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USE OF EXPERT SYSTEMS IN NUCLEAR SAFETY

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ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

One dominant aspect of improvement in safe nuclear power plant operation is the very high speed in the development and introduction of computer technologies. This development commenced recently when advanced control technology was incorporated into the nuclear industry. This led to an increasing implementation of information displays, annunciator windows and other devices inside the control room, eventually overburdening the control room operator with detailed information. The logical next step is therefore to concentrate the collected data in a well structured display with prioritization capabilities. Advancement in software development subsequently helped to transform the computerized operator support system from a relatively simple status-quo presentation into a veritable decision aid with capacities for diagnosis, trend analysis and checking of recovery actions. Expert systems are a further step in this direction being designed to apply large knowledge bases to solve practical problems. These "intelligent" systems have to incorporate enough knowledge to reach expert levels of importance and represent a very advanced man-machine interface.

SUMMARY

The aims of the Technical Committee were addressed by the three Working Groups and summarized in Sections 2, 3 and 4 of this report.

Section 2 summarizes the results and discussions on the current capabilities of expert systems and identifies features for the future development and use of Expert Systems in Nuclear Power Plants.

Section 3 provides an overview of the discussions and investigations into the current status of Expert Systems in NPPs. This Section develops a method for assessing the overall benefit of different applications and recommends a broad strategy for priority developments of Expert Systems in NPP's.

Section 4 assesses the overall use of PSA type studies in Expert Systems in NPP's and identifies specific features to be adopted in the design of these systems in future applications.

The conclusions of the three Working Groups are presented in Section 5.

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1. INTRODUCTION

Expert Systems and Artificial Intelligence technology in general, have become increasingly prominent as a potential solution to a number of previously intractable problems in many phases of human activity over the last few years. The industrial sector has identified expert systems as a means of :

- increasing plant safety
- distributing the knowledge of experts to less qualified workers
- increasing industrial reliability and throughput
- preserving the expertise of retiring workers

The world nuclear industry has also begun to look at expert systems for all of the above reasons, but particularly to increase plant safety. The extensive application of Probabilistic Safety Analysis (PSA) studies to nuclear power plants leads to the question of the best way to use the results of these studies with expert system technology to produce operation aids which will enhance plant safety.

A Technical Committee Meeting was convened by IAEA, October 17-21, 1988 at the Vienna International Centre, to bring together a cross section of the world nuclear power community with several aims in mind.

From a technical point of view, the aims were:

1) to describe expert system initiatives in the various national programs,
2) to identify the current state of the art in the use of expert systems in nuclear power,
3) to identify useful and feasible applications to pursue, and
4) to identify how PSA studies can be incorporated into knowledge based tools

As the list of participants demonstrates, the meeting attracted wide participation from the world nuclear power community. The national presentation sessions produced an even wider technical spectrum of papers. These papers were quite remarkable in their different emphases and directions and yet, many common threads appeared. A purpose of this report is to summarize the contents of the presentations and to identify the common threads.

In addition to purely technical concerns, there was the issue of co-operation in implementing artificial intelligence technology in the nuclear power industry. In order to ease the introduction of this technology into their respective operations, many organizations have formed interest groups to share experience with:

1) techniques and tools,
2) case histories of actual projects, and
3) basic concepts.

In some cases, consortia have been established to fund pilot projects. Usually such consortia represent different interests with related goals. However, even competing organizations have joined such groups since they recognize the value of sharing generic information at this pre-competitive stage of the technology. One of the purposes of this meeting was to identify technical areas in which co-operation would be valuable and to recommend to the IAEA that action to initiate such co-operation be pursued.

In order to accomplish this task, the committee was divided into three working groups following the national presentations. Each group was responsible for preparing a report on a distinct aspect of the meeting. These aspects were:

1. Capabilities of Expert Systems and their Estimated Impact on Safety

2. Areas of Application for Expert Systems in Nuclear Power Plants

3. Utilization of Plant Specific PSA Results in Advisory Systems for operational support.

The detailed choice of these topics was made by the working groups themselves as part of their deliberations. Each group was responsible for producing a chapter for this report relevant to the group's topic. Draft reports were discussed in several plenary sessions of all committee members during the group deliberations in order to obtain feedback from the entire committee, to ensure consistency of the final report, and to discuss possible recommendations arising from the working group reports.

2. CAPABILITIES OF EXPERT SYSTEMS AND THEIR ESTIMATED IMPACT ON SAFETY

Based on the presentations made to the technical committee by members it is clear that the application of expert systems to nuclear problems and in particular nuclear safety topics is of interest to the majority of countries using nuclear power. Although not all national organizations were directly represented at the meeting, there is sufficient information in the literature to support the above statement. At the present time, the application of expert system techniques is appropriate to the enhancement of existing facilities in use which themselves will typically be applied using more conventional approaches. Expert systems offer an additional way of solving some problems and in certain instances they may offer the only practicable solution.

Expert systems can now be regarded as sufficiently mature to be used to create a variety of user support facilities; there is no question of them either replacing fully the human contribution to a system or being used in a superior role to a human. For the forseeable future, the inclusion of expert systems into overall system design is not expected to change in any way the existing, proven approaches to the achievement of nuclear safety in installations. It is probable that they will impact on the way in which such systems are designed and justified.

The decision to employ expert system techniques for a given application must be based on clearly defined criteria. Such criteria are well established and can be found in the literature. For use in a nuclear application, certain additional criteria must be met and these are discussed in the various sections of this Technical Committee Report.

Of particular concern are the questions of system software (language/shell, etc) and application software (knowledge base) quality control. There is no justification for the often observed casual approach to system development. Expert Systems, particularly in nuclear applications, require similar attention to software engineering and to conventional systems. Continued development of tools and methodologies to assist with these activities is required and developers and suppliers of expert systems should be encouraged to improve what is presently available. In this respect, purchasers and users of expert systems must set suitable standards to which developers can respond. The availability of adequate ways of justifying the integrity of expert system software may be a key factor in establishing the technology within the nuclear industry and in integrating new facilities within the framework of licenced systems. A variety of approaches and techniques are in use, depending on the stage of development of both nuclear technology and of computing technology in each member country and this breadth of approach is entirely justifiable. There are at this stage, no perceived benefits from recommending a standardized approach.

Expert systems represent the outward manifestation of a relatively new and rapidly developing science. As a science, it has not yet reached the stage of formal mathematical definition, and in many areas this situation may prevail for many years to come. It is therefore important that existing knowledge about designing, building and operating expert systems is recorded, structured and made available to the nuclear power community as soon as possible. The evolutionary nature of the technology will require that knowledge be regularly reviewed and updated if it is to be a credible source of design information for users. Improved mechanisms for achieving this knowledge assembly and dissemination are urgently required.

As a new technology, it is important that user confidence is rapidly established and a proven method of achieving this is the use of a demonstrator project. To be successful, such demonstrator projects must be realistic in size and scope and must be applied to a realistic problem if they are to be credible. This is particularly important in the case of real-time systems. The use of full-scope simulators for demonstration purposes should be exploited where possible.

In discussing the application of expert systems to nuclear power it is useful to be able to classify systems and it is also necessary to adopt certain definitions of terms, peculiar to the topic. A possible classification scheme for this purpose could use the following basic attributes:

Notional System Size	Small--Medium--Large
Qualitative System Complexity	Low--High
	Off-Line/on-Line

Based on this scheme, Figure 2.1 illustrates the types of nuclear applications being considered or developed using expert system techniques based on the material presented at the meeting and certain other known applications. Certain specialist terms have been defined in appropriate sections of the report.

Included in the figure is a small but interesting set of applications using rule-based systems for closed-loop process plant control. These have been included to present a comprehensive picture but they differ significantly from the other applications shown in that all other sets are used in open loop mode, i.e. advice to the human user. Appendix 1 gives a fuller explanation. The small group termed Safety Analysis relates to systems which can be used to assess the status of plant or components such as primary coolant circuit, neutronics, auxiliary systems, etc. Other groupings should be self-explanatory.

The chain dotted line on the figure indicates the approximate state of development of expert systems in the various application areas. Also shown is a predicted state of development which might be hoped for in say 2-3 years time. It can be seen that the areas of real-time, on-line diagnostics, fault management/accident response, operating procedures and PSA offer significant challenges and give an indication of where future resources might be allocated.

2.1. Methods and Techniques for Expert System Development

2.1.1. Overall Project Management Approach

Project Management with respect to Knowledge Based Systems (KBS) development usually differs from other software project management. This is due partly to the inherent nature of the field, as it addresses highly complex knowledge, and partly to its youth and the resulting state of the art. A typical feature here is that prototyping methods are relatively important, in contrast to conventional software engineering. Other methodologies have been reported at the meeting that will become of interest, but they are still in the research stage.

This experimental nature of KBS development does not imply, however, that elements of the more classical software management approach of separating stages can not or should not be introduced. Elementary principles like the reduction of complexity through problem decomposition remain valid in the context of expert systems.

First of all, the feasibility of the proposed KBS has to be assessed. In a preliminary study, a cost-benefit analysis should be made. The relevant points here are covered in many textbooks in a general fashion. For the present applications specific remarks are made in Sections 3 and 4 of this Report. It suffices to point out here that in the field of expert systems resources are needed that are novel and non-standard for many organizations. These can be externally acquired only at high cost. Therefore, exploiting and strenghtening in-house capabilities is an important issue. Furthermore, relative newcomers in the field are recommended to start with modest and isolatable problems in order to build up the necessary hands-on experience.

Given a perceived benefit of a proposed expert system, two distinct activities have to be carried out:

- a knowledge analysis;
- an external requirements analysis.

The knowledge analysis is concerned with the question of whether the specific expertise is suited for being represented in a knowledge-based system. To this end, often a prototype or demonstration system is developed. This is a focal point in the general literature, and some remarks are made in Sections 2.1.2. and 2.1.3. below. It is desirable that the knowledge, because it is valuable already in its own right, remains discernable in an intermediate representation form and is not only available in a final coded form. Further, more emphasis than usual has to be put on the fact that existing expert system shells and languages are limited in various ways. Accordingly, the selection of such tools before the aspects of knowledge representation and structure have been carefully considered (a quite common practice nowadays) must be cautioned against.

Also the need for carrying out a requirements analysis is accentuated here, since it tends to be underestimated in KBS projects and literature. The intended KBS has to ultimately operate in a certain environment. Early involvement of the operating staff as end users of the system and of the knowledge-base maintenance team is crucial.

Many of the associated requirements can be and should be identified at an early stage, e.g., end-user characteristics, operational interfacing and performance requirements, verification and maintenance aspects. These are addressed in Sections 2.2.to 2.4. In assessing the feasibility of a proposed KBS these factors are equally important compared to a satisfactory problem-solving behaviour of the prototype knowledge base.

With due regard to the experimental nature of KBS development as exemplified in prototyping, it is concluded that several elements of conventional software management can be used with advantage also in KBS project management already in the initial stages. This applies even more strongly to the later stages of implementation, testing, introduction and maintenance. Here, it is to be recognized that there may be an important difference between the effort needed in producing a prototype and that in building and maintaining a fully-fledged expert system. Also, a conscious choice might be made in favour of differing development and operation environments (e.g., prototyping in PROLOG and final implementation in another language like C or PASCAL for speed or interfacing reasons).

Given such a careful overall management approach, it is strongly felt that expert systems will provide a useful addition to existing software capabilities for the improvement of nuclear safety.

2.1.2. Knowledge Acquisition and Representation Techniques

Knowledge acquisition, defined here as the extraction, interpretation and organization of the necessary facts, knowledge, heuristics and other expertise, is generally recognized to be a crucial activity. The generation of the data is done through a variety of techniques such as structured and unstructured interviews, questionnaires, think-aloud protocols, observation of task performance and so forth. Whilst there is some limited guidance from the literature this is not specific to the nuclear field. Equally important, the availability of adequate software tools is not satisfactory. However, experience shows that the generation and interpretation process is clearly facilitated if the knowledge engineer is knowledgeable about the domain. Since knowledge acquisition is very labour-intensive, much can be gained here by employing in-house expertise. Furthermore, it can be remarked that the use of elicitation techniques depends on the context and the stage of the project and, therefore, the ability to employ a variety of techniques probably will give the best results.

The analysis of the data is usually done with the aid of prototyping methods, implying that the knowledge representation is in terms of implementation constructs (e.g., rules, frames) available in expert system shells and tools. Thus, in prototyping the information analysis is interwined with implementation although these are conceptually distinct activities. The advantage of prototyping is that it is a very practical and visible approach, also from the point of view of the expert, that lends itself to incremental improvement. On the other hand, it tends to lead to poor documentation, which for management purposes is undesirable.

Also, it is less helpful in providing an intermediate, higher-level representation of the knowledge. As has been mentioned, often the explicit description and analysis of domain expertise is very valuable in its own right (e.g. for training or for improving procedures), even without being encoded in an expert system. Finally, it has to be recognized that the knowledge representation constructs provided by most existing shells are limited and may not be suited to the task. This has been borne out by the experience of several participants, as reported at the meeting.

The above points deserve careful attention in all KBS projects. It is evident that there is a definite need for better tools for knowledge acquisition and representation. It is therefore of importance that the users, including the nuclear industry, make clear their needs to the suppliers of these tools, e.g., with respect to customizing rules and frames, interfacing capabilities, knowledge editing and documentation aids, etc. In general, further work should be encouraged on improving the methodologies on knowledge acquisition, representation, inferencing and structuring and on the associated activity support tools. Also, knowledge-base handling in various consistency and integrity aspects appears to lag behind the state of the art in data bases and is felt as a problem. It is noteworthy in this context that several participants reported in-house efforts in these directions, including intermediary high-level modelling languages and specialized inference engines and knowledge representations.

2.1.3. Expert System Shells, Tools and Languages

Expert systems technology is developing rapidly with regard to both hardware and software. Therefore, the comments below as well as existing general reviews have to be qualified in that what applied at the time of the Meeting of the Technical Committee could be subsequently outdated.

The relevant languages can be, roughly speaking, grouped into two classes: conventional languages like C, PASCAL and FORTRAN and AI-oriented languages such as LISP, PROLOG, OPS5 and SMALLTALK. Although the latter languages are the most appropriate ones from the knowledge point of view they are not always adequate (e.g., due to garbage collection problems) with respect to speed and efficiency. This limits their real-time capabilities in view of the demands of nuclear safety. Thus, consideration should be given to the possibility, mentioned earlier, of using different languages for development and run-time versions. In comparison to shells and tools, it can be said that languages provide a maximum of flexibility and a minimum of direct support.

Tools and shells can be globally classified into two groups:

- High-end general-purpose tools like KEE, ART, LOOPS, which provide a wide range of knowledge representation and support capabilities and may be utilized across different classes of problems and tasks.
  However, they are expensive and still offer very limited support for knowledge elicitation, editing and high-level modelling, the desirability of which was pointed out previously under the heading of knowledge acquisition.
- Domain-specific tools/shells such as G2 and most of the smaller systems, which are restricted in their support and are focussed on special classes of problems. Thus, although they are much less expensive, they are also much less wide in scope compared to the high-end tools.

Hence, unambiguous guidelines cannot be established. Nevertheless, a useful list of evaluation criteria for tools/shells includes the following items:

- Knowledge elicitation and acquisition support;
- Range and flexibility of knowledge representation constructs (rules, frames, objects), including the possibility of customizing these;
- Knowledge base structuring support (modules, meta-rules);
- Inference techniques (forward-backward chaining, flexible search, reasoning with uncertainty, conflict handling);
- Knowledge-base handling with respect to various integrity and consistency aspects;
- Development and debugging facilities;
- Multi-processing capability (several knowledge bases);
- Interfacing features, including user-interfacing (customized windows, menus, icons, buttons etc.), integrated process display, interfacing to software in other languages, data communication;
- Cooperation with external software of various kinds (databases, real-time data acquisition and analysis, numerical computer codes), a point to be stressed since shells tend to be too closed for the applications considered here;
- Performance aspects such as CPU and memory requirements, speed and efficiency.

Many of these items should already have been addressed in the early requirements analysis, which has been outlined. Again, it is important that the nuclear users express their needs in these respects to the suppliers of tools and shells and, where felt necessary, do some own effort in these directions.

2.2. Knowledge Base Verification

Knowledge base (KB) verification is a key point in the development of an ES based application because of two reasons:

- safety of the knowledge stored in the KB
- inference engine may be sensitive to bad data

Knowledge base verification means to perform the appropriate tests for checking:

- KB consistency (no conflict, absence of loops)
- KB completness
- adequate handling of the connecting relations in case of structured or multiple cooperating knowledge bases
- the fidelity of the knowledge as a model of some part of the real world

For rule-based systems verification the following mathematical methods can be applied:

- consistency and completeness verification using logical function representation of the rules by the application of the methodology developed in the theory of electrical circuits
- search for minimal cut sets and loops using the graph representation of the rule set by graph theoretical procedures

Anyway, the task is critical because the increase of the KB size may determine a "combinatory explosion" of all possible situations that the expert system must be able to manage.

Actually the scientific literature on this matter does not offer a definite solution for this problem, and the following described recommendations for KB verifications are extrapolated from the analysis of the software engineering methods already successfully applied in the large real time system verification and will be in the future integrated with new strategies defined as quality assurance methodology for large KB verification.

The best approach to solve a large problem is to try to decompose the problem into different smaller solvable subproblems. In particluar the following strategy is recommended:

- avoid performing a global verification of the KB at the end of the system development
- perform a decomposition of the knowledge base relative to the global application domain into several smaller knowledge bases relative to smaller subdomains
- perform separate verifications of these KB : these tests can be complete if the knowledge bases are sufficiently small.

Using this methodology the knowledge engineer can be confident in producing knowledge sets "correctly tested" in their own application domains. Then at a different level, the produced knowledge sets must be linked together to form the complete system structure.

Other verification tests will be performed at this new level: their goals will be to verify the correct validity of the various knowledge sets during the inference execution.

It is also necessary to demonstrate that for a correct utilization of this methodology, the following constraints must have been previously applied in the definition phase of the knowledge representation models:

- a restriction in the modelling to achieve safe and consistent system structure
- a preliminary study to establish the most correct approach to decompose the global KB

Finally it is to be noted that for a good KB verification an efficient interaction must be provided between the expert system to be tested and the expert of the application domain.

To facilitate this interaction it is recommended that the following features be incorporated:

- the representation of the knowledge must be "explicit" inside the programming structure and must be "easy" to add/remove knowledge from the KB
- a reasoning "explanation module" must be present in the system visualising all the selected rules and verified hypotheses during the inferential process.

2.3. Performance

There is a growing interest in many of the member countries to develop expert system applications in different areas of nuclear technology. Therefore there is an urgent need to formalize aspects or guidelines on how to evaluate a given expert system application. These aspects should address the question "how good is my finished expert system based application?" in more exact terms.

A commonly accepted performance evaluation guideline would help in reviewing and in comparing the activities of the Member States in the area.

Aspects of performance evaluation of an expert system based application in nuclear industry can be applied from the field of software technology, carefully adapting them to the needs of the nuclear industry, especially for nuclear safety. This adaptation needs further activity both from IAEA and from the member states possibly in the framework of another Technical Committee Meeting. This point formalizes the key concepts in the area of performance evaluation.

2.3.1. Criteria

It is to be emphasized that the importance and even the meaning of different performance criteria vary with different application areas within the nuclear industry. Moreover the desired (expected) value of different performance criteria is different in all major application areas.

Three commonly used performance measures can be identified in the literature and as practiced by Member States:

- size type measure which reflect the capacity of the knowledge base and the inference engine;
- speed type measure which are usually given in terms of response times;
- sensitivity type measure which characterise the performance degradation in extraordinary situations.

Following the classification of the expert system based applications in nuclear industry,illustrated in Figure 2.1., the importance and the trend of the desired values for the above three performance measures can be derived.

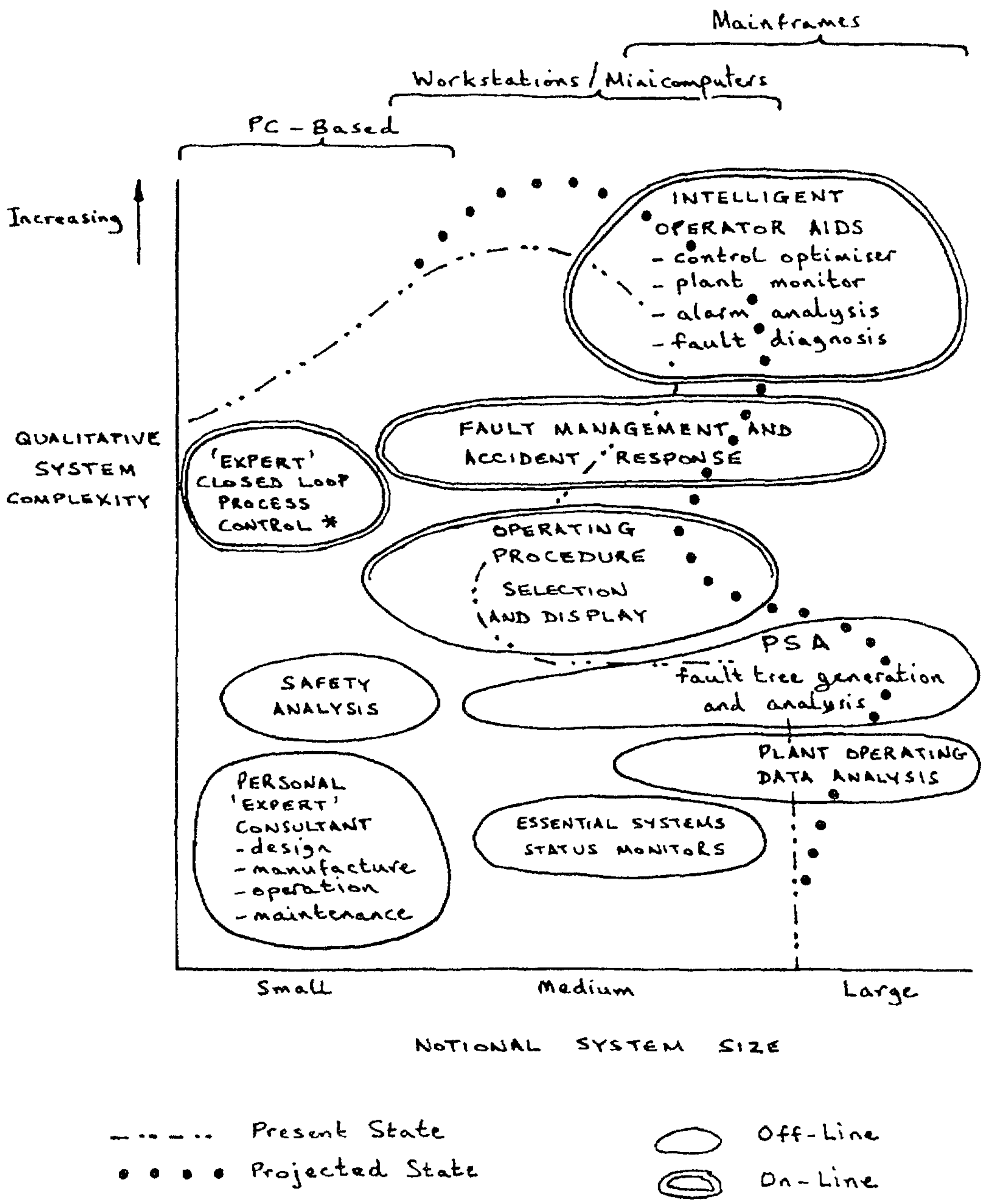


FIG.2.1. Classification of expert systems applications.

2.3.2. Size

The size of a given expert system based application can be understood in different ways:

- the size of the application domain (how many external variables or signals are or can be present);
- the number of knowledge items (rules, frames, objects, etc.) to be handled (note that this number is usually given in terms of "number of rules" even in those cases when more sophisticated knowledge representation forms are available);

- some numbers expressing the measure of complexity in the knowledge base, for example the number of hierarchical levels, the number of available knowledge bases, etc.;
- the size of the software product itself, as:
  - memory requirement for the inference engine,
  - memory requirement for the knowledge base,
  - dynamic memory needed for the continuous operation (see garbage collection problems).

2.3.3. Speed

The speed of an expert system based application is usually measured in some kind of response time, i.e. the time between the occurrence of a task and the response obtained by the operator or engineer-expert. In case of on-line applications the occurrence of the task is usually a signal value change recorded (or signaled) by the control system, in off-line case the operator or expert engineer initiates the task.

Although the response time is normally a main design parameter it is not an easy task to define (and measure) its exact value because the response time varies with the concrete task to be solved. It is a well-known fact from the literature that expert systems may behave very poorly in response time if they are given a task far from their designed average use (but within their scope).

Therefore it is desirable to give not only an average value of the response time, but also some information about its "statistics", for example values for the best and worst case obtained (possibly together with the description of the cases) or some kind of empirical standard deviation on the test cases. To make a more precise and well-grounded suggestion about the response time characterization would need further investigation and collection of the practical experiences in the member countries.

It should be noted that for on-line applications the overall response time contains the following items:

- acquisition time for the data processing system of the plant,

- acquisition time for the expert system (including, for example networking),
- preprocessing time to fill data into the fact part of the knowledge base,
- expert system processing time (reasoning),
- presentation (display) time.

The analysis of the times taken by the above items enables the weak points of an on-line expert system based application to be identified.

2.3.4. Sensitivity

For safety requirements it is particularly important to have systems which are not sensitive to various types of errors and mistakes. In the field of expert system based applications in nuclear power plants the following sensitivities in performance are of importance:

- sensitivity with respect to measurement errors;
- sensitivity with respect to sensor failure;
- sensitivity with respect to knowledge base extension or structure modifications;
- sensitivity with respect to improper constant data (including conflicting rules);
- sensitivity with respect to inadequate or bad operator action ("fool safety").

The measurement of the sensitivity to some changes (cases above) can be chosen as

- qualitative: if the system remains fully functionable or not;
- quantitative: the variation of other measure of performance (e.g. speed) to these changes (not only an average value, but a "statistic" is needed).

It is often not evident what the causes of a high sensitivity can be in a complicated expert system based system. This topic needs further investigation and experience from practical examples.

It is important to note that the sensitivity cases above are more or less common in evaluating, for example commercial software tools or commercial compact controllers. It would be worthwhile to review the experiences in these fields and continue with the specific requirements in nuclear industry to make a revised (enhanced) list of the sensitivity cases together with more detailed definitions and suggestions on how to improve bad sensitivity situations.

2.4 Interfacing

Each expert system has to communicate, as in the case of other computerized applications, with its direct environment. Two types of interface can be distinguished. The first one is the man-machine interface, and the second one the software interface, i.e.software within which it is contained, software which it includes and software with which it has to co-operate.

Man-machine interface: two main groups of people are involved in the use of expert systems:

- the developers
- the users

For developers a more flexible interface is required and for certain applications this may be quite sophisticated. For the final system users, e.g. operators, engineers, designers, the interface must be specifically designed for the tasks involved.

However, in general, the man-machine interface is well defined and achieved by complete studies in this area for nuclear applications particularly since expert systems shells provide good man-machine interface.

Two points require attention:

- the consistency of the expert-system man-machine interface and the other system's man-machine interfaces when they are used together, for example in control rooms.
- the definition of priorities in presenting the messages to alert the users, when real-time supervision expert systems are running in cycle.

Interface with other software systems

In several cases the expert-system has to communicate with other applications. Some of these may be:

- external data bases or knowledge bases
- communication networks
- conventional computerized applications such as mathematic computerized modules, chaining tasks modules, etc.

This variety of possible interfaces needs to be addressed in consistency with the distributed systems (hardware, operating systems), in the transaction organisation (reasoning process interruptions, priorities, reset or restart of the reasoning process), in the mutual protection of each application.

It should be emphasized that there is extensive research and development in the area of real-time expert-systems. Interfacing can be a critical problem in implementing an expert-system, particular attention must therefore be paid to this throughout the implementation process. Further, the problems of co-operating inference engines are still significant.

It is concluded that "open" expert-systems and cooperating inference engines are required with the goal of consistency in the whole operating system and in particular in the data exchange. In the same way, portable expert-systems are recommended to easily facilitate the expert-system implementation in the computers used for power plant control and command if real-time or off-line operation are expected.

2.5 System Quality Assurance

Throughout all phases of any expert system project there is a need to demonstrate adequate quality control of the product being produced. Key items in such a process will include:

- Validity of acquired knowledge
- Validity of knowledge representation models used
- Validity of coded knowledge
- Validity of expert system inference mechanisms and data handling

- System Repeatability
- System performance in relation to targets and to process and user needs

It is not possible to discuss these matters separately since they are implicit in much of the material contained in this section of the report. Each subsection therefore addresses the question of validation and verification in the appropriate context.

3. AREAS OF APPLICATION FOR EXPERT SYSTEMS IN NUCLEAR POWER PLANTS

The potential applications of expert systems in the nuclear industry is wide and, in order to focus the discussions on the subject matter of the Technical Committee Meeting, the following constraints were agreed for the identification of potential application areas:

(i) Only applications directly associated with NPP operation to be identified, i.e. potential applications in areas such as fuel fabrication, fuel reprocessing, test reactors, waste management, etc., not to be considered.

(ii) Only the identified application areas to be discussed in detail (see Section 3.1).

(iii) Only safety-related but not necessarily safety qualified applications to be considered.

3.1 Application Areas

The following general application areas were considered.

- Plant Data Management
- Training
- Condition/Safety Status Monitoring
- Alarm Analysis and Diagnosis (Before Trip)
- Accident Management
- Emergency Planning

3.2 Task Paradigms

The tasks carried out by expert systems fall within certain 'task paradigms'. For the range of expert systems applications identified in the working group the following paradigms were considered:

- diagnostic
- prognostic
- analytical
- monitoring
- configuring
- planning/scheduling

Diagnostic

A Diagnostic system analyzes the knowledge base state (wherein the 'facts' can include stated facts, inferred facts and observed data) and makes recommendations for the remedy for a particular problem.

Prognostic

A prognostic system analyzes the current knowledge base state to predict the likely outcome or consequences if no intervening action is taken,i.e. looks into the future.

Analytical

An analytical system carries out a pre-determined task based on the current knowledge base state, an example would be an expert system to actually carry out (say) automatic fault tree generation.

Monitoring

A monitoring system keeps track of performance over time, e.g. monitoring of process parameters or the condition of plant monitoring.

Configuring

A configuring task is one in which a finite number of components are arranged in a particular combination to suit the problem constraints. An example of such a system is one for the configuration of large commercial computers.

Planning/Scheduling

A scheduling task co-ordinates the various elements of an operating organisation in order to improve efficiency. An example would be a system which schedules when a particular maintenance operation is needed to be carried out, with the recommended resources required. Scheduling tasks normally refer to routine operations wherein the procedural requirements of the operations are known from task to task.

Planning tasks are similar, but are more variable from task to task.

3.3 Real Time Expert System Definitions

3.3.1. Real Time Expert System

The plant data is collected automatically using the plant instrumentation and control, data acquisition systems (DAS), including event sequence monitoring.

The data obtained is processed by the expert system and the results are obtained before the next set of data is considered. The computational capability of the system fits within the time-contraints or delays in the physical process/scenarios under considerations and user needs for information. The expert system meets the information needs of the user in relation to the task he performs.

The on-line system computing machinery should have the capability to accept interrupts, re-schedule tasks, assign priorities etc. to achieve the necessary performance.

3.3.2. Near-Real Time Expert System

This is an expert system which can compute information and furnish data in a time which is close to that of a real time system. Such systems perform sufficiently well to allow their use as on-line tools for plant supervision or to provide operators with decision-making information, where the process dynamics are comparatively slow.

3.4 Known Applications

A number of expert system applications in the NPP industry were discussed at the Technical Committee Meeting. These are summarized in Table 3.1 with the basic features of each application noted. Other known applications were also identified by the working group and these have also been included in the Table (approximately 40 known applications were identified by the Working Group). It is important to note that there are likely to be a number of other NPP applications which have not been identified at the meeting; particularly from the USA and Japan. The applications included in Table 3.1 do not all fall within the constraints agreed in this Section for identifying potential applications for NPPs.

The following features are included in the Table:

- a brief description of the expert system
- whether it is safety-related
- task paradigm(s)
- size of system, i.e.
  - small (e.g.< 500 rules)
  - medium (e.g. 500 - 3000 rules)
  - large (e.g.> 3000 rules)

- status of the system, i.e.
  - under development
  - implemented

- type of system, i.e.
  - demo/prototype
  - implemented

There is a wide range of knowledge representation formalisms, software and hardware being used for the identified systems, although rule-based systems on PCs tend to dominate. This probably indicates the current state of the art in nuclear power plants, i.e. most countries who have made a committment are still in the prototyping stage, with only PC/rule-based systems implemented and more-sophisticated systems (to actually be used!) are still under development.

Text continued on p. 44.

TABLE 3.1. KNOWN APPLICATIONS OF (KB) EXPERT SYSTEMS FOR NUCLEAR POWER PLANTS

NAME			EXPRESS		EDES		ES for Turbine Vibration Dia.
DESCRIPTION			PSA Tool				
SAFETY RELATED		X		X			
TASK			Analysis		Diagnostics		Diagnostics
SIZE	- small			X	500 Rules	X	
	- medium						
	- large	X					
STATUS	- under development			X		X	
	- operation	X					
TYPE	- demo/prototype	X				X	
	- implemented						
KNOWLEDGED REPRESENTATION	- rules	X		X		X	
	- frames						
	- object oriented						
	- O-A-V						
SOURCE OF INPUT	- data base						
	- signal processing			X			
	- dialogue	X				X	
	- simulation						
SOFTWARE	- language	X	Genesia	X	'C' Unix		Genesia
	- hybridtools						
	- shell					X	
	- conv. language						
EXPECTED MANPOWER							6 Engineering years
COUNTRY			France		USSR		Hungary
DEVELOPING ORGANIZATION			EDF		Ministry of Atomic Energy		VEIKI
HARDWARE	- main frame	X	IBM 3090				IBM PC
	- mini			X	PDP-II clone		
	- special w.s.						
	- PC					X	
REAL TIME	- off line	X					
	- on-line/near RT						
	- RT			X			
NOTES			Integration in the LESSEPS Software				
REFERENCES			(Computerizing PSA)				
DEFINITIONS:							

TABLE 3.1. (cont.)

NAME		ARTICON		AIDE		NO NAME YET
DESCRIPTION		Diagnostic Tool for Secondary side availability		Instrument Air System Impairment Diagnosis		Operator Aid for Condenser Seawater Leak Det. & Action
SAFETY RELATED		Availability related	X			Partly
TASK		Diagnostics		Diagnostics		
SIZE - small - medium - large		~ 1000 rules		See Developers	 X	
STATUS - under development - operation	X			May not be done. Another domain may be substituted.	X	
TYPE - demo/prototype - implemented	X		 X		 X	
KNOWLEDGED REPRESENTATION - rules - frames - object oriented - O-A-V	X			Not decided		Rules supplemented by conventional language data structure & procedures
SOURCE OF INPUT - data base - signal processing - dialogue - simulation	X X X X		 X		 X	
SOFTWARE - language - hybridtools - shell - conv. language	 X	Insight 2 (level 5)		Not decided	 X	 Pascal
EXPECTED MANPOWER		5 Engineering years				2 man-months
COUNTRY		Hungary		Canada		Canada
DEVELOPING ORGANIZATION		VEIKI		Hydro/Quebec		AECL-NBEPC
HARDWARE - main frame - mini - special w.s. - PC	 X			Not decided	 X	
REAL TIME - off line - on-line/near RT - RT	X		 X		 X	
NOTES				AECL may participate		See J. Anderson (AECL)* P.K. Patterson A.D. Rosevear (NBEPC)**
REFERENCES		Conference Paper		Roy Olmstead of AECL can identify participants		

DEFINITIONS:
*AECL = Atomic Energy of Canada Limited
**NBECP = New Brunswick Electric Power Commission
***UNB = University of New Brunswick

TABLE 3.1. (cont.)

NAME		DEFECT DETECTIVE		SEE DEVELOPERS		NO NAME YET
DESCRIPTION				Operator Aid		On-Line Diagnosis
SAFETY RELATED		Somewhat-but-mostly economic		No	X	
TASK		Diagnosis & Evaluation		Start up of liquid zone control system for CANDU		Diagnosis
SIZE - small - medium - large	X			See developers	X	50 rules
STATUS - under development - operation	X		X		X	
TYPE - demo/prototype - implemented	X			Not decided - depends on whether can port to cheaper hardware	X	
KNOWLEDGED REPRESENTATION - rules - frames - object oriented - O-A-V	X	Frames	X			Rules supplemented by conventional language variables and procedures
SOURCE OF INPUT - data base - signal processing - dialogue - simulation	X		X		X	
SOFTWARE - language - hybridtools - shell - conv. language	X	Prolog.	X	Art	X	Pascal
EXPECTED MANPOWER		6 man-months		1 man-year		2 man-months
COUNTRY		Canada		Canada		Canada
DEVELOPING ORGANIZATION		AECL		NBEPC/UNB		AECL/NBEPC
HARDWARE - main frame - mini - special w.s. - PC	X		X	Symbolic LISP Machine	X	
REAL TIME - off line - on-line/near RT - RT	X		X		X	
NOTES		Will be modified for better performance & more seamless data access		See B. Nickerson (UNB)*** G. Mallette ***		Contact J. Anderson B.K. Patterson A.D. Rosevear **
REFERENCES		Manzer, Anderson, So 1988 CNS Conference				

DEFINITIONS: *AECL = Atomic Energy of Canada Limited
**NBECP = New Brunswick Electric Power Commission
***UNB = University of New Brunswick

TABLE 3.1. (cont.)

NAME		SEE DEVELOPER		ON-LINE SAFETY SYSTEM IMPAIRMENT MANUAL		FUELLEM
DESCRIPTION		Diagnostic Tool		Guides Operator to find impairment level advises approp. action		Chooses CANDU channel to refuel next
SAFETY RELATED	X		X			No
TASK		Diagnosis		(see description)		Analysis
SIZE - small - medium - large	 X 	 250 rules 	 X 			 See Developers
STATUS - under development - operation	 X		X 		 X	Being revised
TYPE - demo/prototype - implemented	X 		 X		 X	
KNOWLEDGED REPRESENTATION - rules - frames - object oriented - O-A-V	X			Rules supplemented by conventional language data structures and procedures		See Developers
SOURCE OF INPUT - data base - signal processing - dialogue - simulation	 X		 X 		X 	
SOFTWARE - language - hybridtools - shell - conv. language	 X 	Guru 	 X	 Pascal	 X	 FORTRAN
EXPECTED MANPOWER		See Developer		2 man-months		See Developers
COUNTRY		Canada		Canada		Canada
DEVELOPING ORGANIZATION		Ontario Hydro		AECL-NBEPC		AECL
HARDWARE - main frame - mini - special w.s. - PC	 X		 X		X 	Super Micro Apollo DN 3000
REAL TIME - off line - on-line/near RT - RT	X 		 X			
NOTES		ECC System Status & Fault Diagnosis for Bruce NGS		Contact J. Anderson* B.M. Patterson**		See B. Rouben or D. Jenkins
REFERENCES		(see O.B. Chou for details)				

DEFINITIONS: *AECL = Atomic Energy of Canada Limited
**NBECP = New Brunswick Electric Power Commission

TABLE 3.1. (cont.)

NAME		PLANT CONFIGURATION & EQUIPMENT STATUS MONITOR		STAR GENERIS		EXTRA A
DESCRIPTION		Operator Advisor				Electric Power Supply Supervisor
SAFETY RELATED	X		X		X	
TASK				Plan status diagnosis signal validation		Diagnostic/monitoring design/planning schedule
SIZE - small - medium - large	X (medium)				X (large)	5000 rules
STATUS - under development - operation	X (under development)		X (operation)		X (under development)	
TYPE - demo/prototype - implemented	X (implemented)				X (demo/prototype)	
KNOWLEDGED REPRESENTATION - rules - frames - object oriented - O-A-V		To be decided			X (rules)	
SOURCE OF INPUT - data base - signal processing - dialogue - simulation	X (data base)	Database + dialog	X (signal processing)		X (signal processing) X (dialogue) X (simulation)	
SOFTWARE - language - hybridtools - shell - conv. language		Hypercard + an expert US system on a separate but communicating computer			X (language)	Genesia
EXPECTED MANPOWER		1-5 man-year				16
COUNTRY		Canada		FRG		France
DEVELOPING ORGANIZATION		Atomic Energy of Canada		GRS		EDF
HARDWARE - main frame - mini - special w.s. - PC		Multiple networked PCs				SPG 7 Bull
REAL TIME - off line - on-line/near RT - RT	X (on-line/near RT)		X (on-line/near RT)		X (off line) X (RT)	
NOTES		See L. Lupton AECL-Chalk River Nuclear Labs				Implemented in early 1989
REFERENCES						

DEFINITIONS: *AECL = Atomic Energy of Canada Limited
**NBECP = New Brunswick Electric Power Commission

TABLE 3.1. (cont.)

NAME		TEX-1		ESSM		DIAREX
DESCRIPTION		Real-time monitoring & Plant diag.		Living PSA		Diagnosis (BWR) Fault Tree Analysis (FTA)
SAFETY RELATED	X		X		X	
TASK		Diagnosis/Monitoring		Analysis/Monitoring		Diagnostic/Analysis
SIZE - small						
- medium			X			
- large	X				X	
STATUS - under development						
- operation	X		X		X	
TYPE - demo/prototype		Plant is not yet ready				
- implemented			X		X	
KNOWLEDGED REPRESENTATION - rules	X					
- frames	X				X	
- object oriented						
- O-A-V	X		X			
SOURCE OF INPUT - data base			X			
- signal processing	X		X		X	
- dialogue	X		X		X	
- simulation	X					
SOFTWARE - language			X	Fortran 77	X	PRL (process description language)
- hybridtools	X	Babylon				
- shell						
- conv. language						
EXPECTED MANPOWER						
COUNTRY		FRG		U.K.		Japan
DEVELOPING ORGANIZATION		Interatom		CEGB		
HARDWARE - main frame					X	
- mini		Symbolic LISPMACH		PDP 11		
- special w.s.						
- PC						
REAL TIME - off line	X					
- on-line/near RT			X		X	
- RT	X				X	
NOTES						
REFERENCES						Nuclear Tech. 79 (1987)

DEFINITIONS:

TABLE 3.1. (cont.)

NAME				ICS-Expert		
DESCRIPTION		Advance Alarm System		PRA		Diagnosis/Analysis
SAFETY RELATED	X		X		X	
TASK		Diagnostic/Monitoring Plant Management		Diagnostic/Monitoring Analysis		Monitoring/Physical Parameters of PWR
SIZE - small			X	70	X	
- medium						
- large	X					
STATUS - under development						
- operation	X				X	
TYPE - demo/prototype	X		X		X	
- implemented						
KNOWLEDGED REPRESENTATION - rules			X		X	50 with learning period
- frames						
- object oriented						
- O-A-V						
SOURCE OF INPUT - data base					X	
- signal processing	X		X		X	
- dialogue			X			
- simulation						
SOFTWARE - language			X	Smart (LISP)		CLIPS (C)
- hybridtools						
- shell						
- conv. language						
EXPECTED MANPOWER						0.6
COUNTRY		USA		USA		Netherlands
DEVELOPING ORGANIZATION		Westinghouse		EPRI		ECN
HARDWARE - main frame						
- mini					X	
- special w.s.						
- PC			X		X	
REAL TIME - off line						
- on-line/near RT						
- RT					X	
NOTES		Alarm Management/trouble shooting computerized proc/plant information system				Expert system with learning period
REFERENCES		Nucl. Eng. International May 1988		Nucl. Tech. Vol. 82 August 1988		
DEFINITIONS:						

TABLE 3.1. (cont.)

NAME		ERICE		BCD MONITOR		APEX
DESCRIPTION		Expert System		On-line Plant Monitor		Alarms Process. Real-Time Expert System
SAFETY RELATED			X			Yes - in part but conventional plant
TASK		Analysis-Diagnosis		Diagnosis		Diagnosis
SIZE - small - medium - large		200 rules	X		X	
STATUS - under development - operation	X		X	Desk top feasibility study only		Project complete
TYPE - demo/prototype - implemented	X		X	Proposed prototype	X	Demonstrator
KNOWLEDGED REPRESENTATION - rules - frames - object oriented - O-A-V			X		X X	
SOURCE OF INPUT - data base - signal processing - dialogue - simulation	X			On-line Data		On-line Data
SOFTWARE - language - hybridtools - shell - conv. language		Personal Consultants PLUS	X	3 shells considered	X	
EXPECTED MANPOWER				Study 2 man-months Proj. 1 man-year		4 man-years
COUNTRY		Italy		U.K.		U.K.
DEVELOPING ORGANIZATION		ENEA		CEGB Jenkinson		CEGB/SIRA IESC Jenkinson
HARDWARE - main frame - mini - special w.s. - PC	X		X			LMI LISP Mach.
REAL TIME - off line - on-line/near RT - RT	X		X	On-line	X	Real time
NOTES	*			Study indicated that type of data was more suited to conven. approach		Project formed basis of spec. for future nucl. application
REFERENCES						Various papers

DEFINITIONS: * The expert system performs the analysis and the diagnosis of the corrosion rate inside the heating circuits of a nuclear power plant. It is connected to a specialist machine that requires corrosion-rate data on-line from the plant.

TABLE 3.1. (cont.)

NAME		NAMS-EXPERT		RGL		SEPIA
DESCRIPTION		Design procedure Expert		Maintenance Aid		Steam Gen. T. Rupt. Training
SAFETY RELATED	X					
TASK		Control of Design Modifications				Training
SIZE - small - medium - large	X		 X		 X	
STATUS - under development - operation	 X	Implemented	X		 X	
TYPE - demo/prototype - implemented	X	Demo/Prototype	X		X	
KNOWLEDGED REPRESENTATION - rules - frames - object oriented - O-A-V	X				X	
SOURCE OF INPUT - data base - signal processing - dialogue - simulation	 X	User dialogue	 X		 X X	
SOFTWARE - language - hybridtools - shell - conv. language	 X		 X			
EXPECTED MANPOWER		1 man-year				
COUNTRY		U.K.		France		France
DEVELOPING ORGANIZATION		CEGB Jenkinson		EDF Merlin Gerin		EDF FRAMENTEC
HARDWARE - main frame - mini - special w.s. - PC	 X		 X			 Sun
REAL TIME - off line - on-line/near RT - RT	X	Off-line	X		X	
NOTES		User feedback being obtained appraisal carried out				
REFERENCES						

DEFINITIONS:

TABLE 3.1. (cont.)

NAME		DIVA		MIGRE		MAC II
DESCRIPTION		Vibration Analysis		Noise analysis		Scheduling Path for Refuelling PWR Core
SAFETY RELATED			X		X	
TASK		Diagnosis		Diagnosis		Scheduling
SIZE - small						
SIZE - medium			X		X	
SIZE - large						
STATUS - under development	X		X			
STATUS - operation					X	
TYPE - demo/prototype	X		X			
TYPE - implemented					X	
KNOWLEDGED REPRESENTATION - rules					X	
KNOWLEDGED REPRESENTATION - frames						
KNOWLEDGED REPRESENTATION - object oriented			X			
KNOWLEDGED REPRESENTATION - O-A-V						
SOURCE OF INPUT - data base						
SOURCE OF INPUT - signal processing	X		X			
SOURCE OF INPUT - dialogue	X		X		X	
SOURCE OF INPUT - simulation					X	
SOFTWARE - language	X	Prolog	X		X	Genesia
SOFTWARE - hybridtools						
SOFTWARE - shell						
SOFTWARE - conv. language						
EXPECTED MANPOWER						
COUNTRY		France		France		France
DEVELOPING ORGANIZATION		EDF Alsthom		EDF		EDF/Research Studies
HARDWARE - main frame					X	3090 IBM
HARDWARE - mini		Sun				
HARDWARE - special w.s.						
HARDWARE - PC			X	Mackintosh		
REAL TIME - off line	X		X		X	
REAL TIME - on-line/near RT						
REAL TIME - RT						
NOTES						
REFERENCES						

DEFINITIONS:

TABLE 3.1. (cont.)

NAME						
DESCRIPTION		Configuration Core Aid		Chemical State of PWR Secondary Circuite		Intelligent Operator Advisor
SAFETY RELATED	X		X		X	Monitoring/Diagnosis
TASK		Analysis		Diagnosis		
SIZE - small			X			
- medium	X					10,000 rules
- large					X	
STATUS - under development	X		X		X	
- operation						
TYPE - demo/prototype	X		X		X	
- implemented						
KNOWLEDGED REPRESENTATION - rules	X		X		X	
- frames						
- object oriented					X	
- O-A-V						
SOURCE OF INPUT - data base	X				X	
- signal processing					X	
- dialogue	X		X			
- simulation					X	
SOFTWARE - language	X	Genesia	X	Genesia		
- hybridtools					X	
- shell						
- conv. language					X	
EXPECTED MANPOWER						24 man-years
COUNTRY		France		France		Hungary
DEVELOPING ORGANIZATION		EDF/Research & Studies		EDF/Research & Studies		Paks Nuclear Power Plant & Autom. Inst. HAS
HARDWARE - main frame	X	3090 IBM				
- mini					X	VAX
- special w.s.						Intelligent Terminal
- PC			X		X	
REAL TIME - off line	X		X			
- on-line/near RT					X	
- RT						
NOTES				On-line for future		
REFERENCES						

DEFINITIONS:

TABLE 3.1. (cont.)

NAME		SESAME				
DESCRIPTION		Operator/maintenance Engineer Support Tool		AGR Reactor Gas Chemical Analysis		Standing order expert
SAFETY RELATED			X		X	
TASK		Fault Diagnosis				Rule compliance monitor
SIZE - small - medium - large	 X	 50,000	X		 X	 500 rules
STATUS - under development - operation	 X		X		X	
TYPE - demo/prototype - implemented	 X		X		X	
KNOWLEDGED REPRESENTATION - rules - frames - object oriented - O-A-V					X	
SOURCE OF INPUT - data base - signal processing - dialogue - simulation	X X		 X			Hand-held electronic logger
SOFTWARE - language - hybridtools - shell - conv. language		Written in 'C'	 X		 X	
EXPECTED MANPOWER						6 man-years
COUNTRY		U.K. - Scotland		U.K. - Scotland		U.K.
DEVELOPING ORGANIZATION		SSEB		SSEB		CEGB
HARDWARE - main frame - mini - special w.s. - PC	 X	 Micro VAX	 X		 X	 IBM PS/2
REAL TIME - off line - on-line/near RT - RT	 X		 X	 Near RT	X	
NOTES						
REFERENCES				References (SSEB) John Morrison		
DEFINITIONS:						

3.5 Scoring Parameters for Identified Applications

In addition to identifying potential application areas the working group also attempted to estimate the worthiness of each application by applying a series of scoring parameters in group-consensus sessions. Parameters which affect an application's worthiness were agreed under two areas:

(i) Potential Benefits

(ii) Achievability

'Achievability' refers to the prospect of achieving success now, i.e it is recognized that certain applications are probably feasible, but to achieve success further research work is necessary.

A third area 'Need' was originally considered but it was found during the analysis that this area was closely-correlated to 'Potential Benefits'. Indeed the benefits of a project are directly dependent on its need, for instance, a project which is urgently needed will inevitably obtain a high scoring on benefits.

A similar scoring approach was adopted to that used in EPRI EL-4323 ("Artificial Intellingence Technologies for Power System Operation", January 1986). The scoring was carried out on a scale of 0 to 4, with '4' indicating a favourable attribute. The following parameters were considered during the exercise:

Potential Benefits

This range of parameters indicates the benefits which would be obtained if a particular problem is successfully solved.

- Reduction in Risk
  0 - no significant reduction in risk
  4 - significant reduction in risk

- Operational Payback: i.e. cost savings achieved during the actual running of the plant
  0 - no savings
  4 - large savings

- Capital Cost Saving (for new plants and retrofits)
  0 - no savings
  4 - large savings

- General application : i.e. can the methodology be applied in other areas
  0 - no benefit to other problem areas
  4 - research in this problem area will be of great use in many other problem areas

- Political; i.e. would the application have important social and political benefits
  0 - no political effect
  4 - large political effect

<u>Achievability</u>

This range of parameters indicated whether a particular application is <u>achievable</u> at this moment in time.

- Knowledge: does it exist and is it well-formulated?
  0 - knowledge is ill-defined and difficult to formalise
  4 - knowledge is well-defined, accessible and easy to formalise

- Software: is the necessary software to solve the problem readily available?
  0 - not availabe and unlikely to be obtained in the near future
  4 - available now

- Hardware: is the necessary hardware readily available
  0 - not available and unlikely to be obtained in the near future
  4 - available now

- Development costs:
  0 : more than 10 man-years of effort
  1 : 7-10 man-years of effort
  2 : 4-7 man-years of effort
  3 : 2-4 man-years of effort
  4 : less than 1 man-year of effort

- Portability:
  - 0 - virtually none of the program can be reused in moving it from one utility to another
  - 4 - minimal changes/new knowledge will be needed in moving the program from one utility to another

In order to emphasize the agreed dominant importance of 'reduction in risk' a factor of 2 was applied to this parameter for each application considered. It should be noted that 'risk' refers to the combination of the frequency of an event and its consequence, i.e.

Risk = Frequency x Consequence

The analysis was only subjective but it is thought that the results should indicate the potential worthiness of some applications compared to others. Further work would be needed to validate the results of this subjective exercise.

The final summation of the individual scoring parameters is taken as an indication of the worthiness of a particular application, with a higher score indicating greater worthiness.

3.6 Identified Applications

3.6.1 General

The working group considered each of the application areas in Section 3.1. with the specific aim of (bearing in mind the time constraints in the analysis/study):

- identifying those areas which the participating experts considered had the most impact on safety
- restricting the maximum choice in any application area to 4 applications. This was generally found to be more than suitable for the classification exercise (see Table 3.2) and scoring exercise (see Tables 3.3 and 3.4) which was carried out on each application, indeed some of the application areas considered were quite specific and less than 4 potential applications were analysed.

Text continued on p. 54.

TABLE 3.2. CLASSIFICATION OF APPLICATIONS

APPLICATION AREA: **PLANT DATA MANAGEMENT**

SPECIFIC APPLICATION		Plant standing orders database	Plant experience (outside) database	Plant experience (internal) database	Technical information support database
TASK PARADIGMS		Monitoring	Design, Monitoring, Diagnostic, Planning	Monitoring, Diagnostic	Monitoring, Diagnostic, Planning
SIZE	small	-	-	-	-
	medium	x	x	x	-
	large	-	x	x	x
SUITABILITY AS DEMONSTRATOR	low	-	-	-	x
	medium	x	x	-	-
	high	-	-	x	-
KNOWLEDGE REPRESENTATION	simple	x	-	-	-
	medium	-	x	x	x
	complex	-	-	-	-
SOURCE OF INPUTS	database	-	x	x	x
	signals	-	-	-	-
	dialogue	x	x	x	x
	simultation	-	-	-	-
SOFTWARE	A.I. Language	-	x	x	x
	hybridtool	-	x	x	x
	shell	x	-	-	-
	conv. lang.	-	-	-	-
HARDWARE (MINIMUM REQUIREMENTS)	mainframe	-	-	-	-
	special W.S.	-	-	-	-
	mini	-	x	x	x
	PC	x	-	-	-
REAL-TIME	off-line	x	x	x	x
	on-line / near r/t	-	-	-	-
	real time	-	-	-	-
EASE OF VALIDATION	easy	x	-	x	-
	medium	-	x	x	-
	difficult	-	-	-	x
KNOWLEDGE SOURCE	documents	x	x	x	x
	measurements	-	-	-	-
	head of expert	x	x	x	x
	database	-	x	x	x
POTENTIAL USER		Supervisors	Plant management, Designers	Designers, Plant management	Plant management, Supervisors, Technical
NOTES			Improved MMI Text searching Graphics? Pictures?		Important information for PSA

TABLE 3.2. (cont.)

APPLICATION AREA: TRAINING

SPECIFIC APPLICATION		Operator Simulator	Maintenance Simulator/Aid
TASK PARADIGMS		Diagnostic, Prognostic, Analysis	Diagnostic, Planning, Scheduling
SIZE	small	-	x
	medium	x	-
	large	x	-
SUITABILITY AS DEMONSTRATOR	low	x	-
	medium	x	-
	high	-	x
KNOWLEDGE REPRESENTATION	simple	-	x
	medium	-	-
	complex	x	-
SOURCE OF INPUTS	database	x	x
	signals	-	-
	dialogue	x	x
	simultation	x	-
SOFTWARE	A.I. Language	x	-
	hybridtool	x	-
	shell	-	x
	conv. lang.	x	-
HARDWARE (MINIMUM REQUIREMENTS)	mainframe	-	-
	special W.S.	-	-
	mini	x	-
	PC	-	x
REAL-TIME	off-line	x	x
	on-line } near r/t }	-	-
	real time	-	-
EASE OF VALIDATION	easy	-	x
	medium	x	-
	difficult	-	-
KNOWLEDGE SOURCE	documents	x	x
	measurements	-	-
	head of expert	x	x
	database	x	-
POTENTIAL USER		Operator	Maintenance
NOTES			

TABLE 3.2. (cont.)

APPLICATION AREA: CONDITION/SAFETY STATUS MONITORING

SPECIFIC APPLICATION		Equipment Monitoring	Systems Behaviour Monitoring
TASK PARADIGMS		Diagnostics, Analysis, Monitoring, Prognosis	Diagnostics, Analysis, Monitoring, Prognosis
SIZE	small	x	-
	medium	x	x
	large	-	x
SUITABILITY AS DEMONSTRATOR	low	-	-
	medium	-	x
	high	x	-
KNOWLEDGE REPRESENTATION	simple	-	-
	medium	x	x
	complex	-	x
SOURCE OF INPUTS	database	x	x
	signals	x	x
	dialogue	x	x
	simultation	x	x
SOFTWARE	A.I. Language	-	x
	hybridtool	x	x
	shell	x	-
	conv. lang.	x	-
HARDWARE (MINIMUM REQUIREMENTS)	mainframe	-	-
	special W.S.	-	x
	mini	x	x
	PC	x	-
REAL-TIME	off-line	x	-
	on-line } near r/t }	x	-
	real time	-	x
EASE OF VALIDATION	easy	-	-
	medium	x	-
	difficult	-	x
KNOWLEDGE SOURCE	documents	-	-
	measurements	x	x
	head of expert	x	x
	database	x	x
POTENTIAL USER		Maintenance, Supervisors, Management, Operator	Operators, Supervisors
NOTES			

TABLE 3.2. (cont.)

APPLICATION AREA: **ALARM ANALYSIS AND DIAGNOSIS**

SPECIFIC APPLICATION		Alarm Filtering and reduction	Prioritization of Alarms	Alarm Analysis and event diagnosis
TASK PARADIGMS		Diagnostic, Monitoring	Diagnostic, Monitoring, Analysis	Diagnostic, Analysis, Monitoring
SIZE	small	-	x	-
	medium	x	x	-
	large	x	-	x
SUITABILITY AS DEMONSTRATOR	low	-	-	x
	medium	x	x	-
	high	-	-	-
KNOWLEDGE REPRESENTATION	simple	-	x	-
	medium	x	-	-
	complex	-	-	x
SOURCE OF INPUTS	database	-	-	x
	signals	x	x	x
	dialogue	-	-	x
	simultation	x	-	x
SOFTWARE	A.I. Language	-	-	x
	hybridtool	-	x	x
	shell	x	x	-
	conv. lang.	x	-	-
HARDWARE (MINIMUM REQUIREMENTS)	mainframe	-	-	-
	special W.S.	-	-	-
	mini	x	x	x
	PC	-	-	-
REAL-TIME	off-line	-	-	-
	on-line }	-	-	-
	near r/t }			
	real time	x	x	x
EASE OF VALIDATION	easy	x	x	-
	medium	-	-	-
	difficult	-	-	x
KNOWLEDGE SOURCE	documents	-	-	x
	measurements	x	x	x
	head of expert	x	x	x
	database	-	-	x
POTENTIAL USER		Operators	Operators	Operators, Supervisors
NOTES				

TABLE 3.2. (cont.)

APPLICATION AREA:		POST-TRIP ANALYSIS AND DIAGNOSIS	EMERGENCY PLANNING
SPECIFIC APPLICATION		Post-trip analysis and diagnosis	Emergency planning advisor
TASK PARADIGMS		Diagnostic, Prognostic, Analysis, Monitoring	Diagnostic, Analysis, Prognostic
SIZE	small	-	-
	medium	-	x
	large	x	-
SUITABILITY AS DEMONSTRATOR	low	x	-
	medium	-	x
	high	-	-
KNOWLEDGE REPRESENTATION	simple	-	-
	medium	-	-
	complex	x	x
SOURCE OF INPUTS	database	x	x
	signals	x	x
	dialogue	x	x
	simultation	x	x
SOFTWARE	A.I. Language	x	x
	hybridtool	x	x
	shell	-	-
	conv. lang.	x	-
HARDWARE (MINIMUM REQUIREMENTS)	mainframe	-	-
	special W.S.	x	-
	mini	x	x
	PC	-	-
REAL-TIME	off-line	-	x
	on-line / near r/t	-	x
	real time	x	x
EASE OF VALIDATION	easy	-	-
	medium	-	-
	difficult	x	x
KNOWLEDGE SOURCE	documents	x	x
	measurements	x	x
	head of expert	x	x
	database	x	x
POTENTIAL USER		Supervisors, Operators, Technical supporting team	Supervisors, Management, Technical Emergency safety personnel
NOTES:			

TABLE 3.3. SCORING CHART FOR IDENTIFIED APPLICATIONS

	B E N E F I T S						A C H I E V A B I L I T Y							
Application	Risk Reduction	Payback	Capital Cost Saving	Gen- eral	Polit- ical	Sub Total	Knowledge Exists	Soft- ware	Hard- ware	Develop- ment Cost	Port- ability	Tolerate Errors	Sub- Total	Totals
Plant Datab.														
-plant stand. orders	1x2=2	2	0	3	1	8	4	3	4	4	1	2	18	26
-plant exper. (outside)	3x2=6	3	3	1	3	16	4	2	4	2	3	2	17	33
-operation. exper. (internal)	3x2=6	3	2	1	3	15	4	2	4	2	3	2	17	32
-techn. info support	1x2=2	2	0	3	1	8	4	2	4	1	2	3	16	24
Training:														
-operator simulator	4x2=8	1	0	1	1	11	4	2	4	0	1	3	14	25
-maintenance simulator	2x2=4	3	0	3	0	10	4	3	4	3	3	3	20	30
Condition/ Safety Status Mon- itoring														
-Equipment	2x2=4	3	0	2	1	10	3	3	4	3	2	3	18	28
-System Behavior	3x2=6	3	0	1	3	13	2	2	3	2	1	1	11	24
Alarm Analysis & Diagnosis (before trip):														
-alarm filter. &reduction	2x2=4	1	0	3	2	10	4	4	4	4	3	0	19	29
-prioritisation of alarms	2x2=4	1	0	3	2	10	3	4	4	4	2	1	18	28
- alarm analy. & event diagn.	3x2=6	3	0	3	3	15	1	2	3	1	1	0	8	23
Post-Trip Alarm Anal. & Diagnosis	4x2=8	3	0	2	4	17	2	2	3	0	1	0	8	25
Emergency Planning Adviser	4x2=8	2	0	1	4	15	3	1	4	0	1	1	10	25

TABLE 3.4. WORTHINESS OF IDENTIFIED APPLICATIONS

APPLICATION	SUB-TOTAL FOR 'BENEFITS'	SUB-TOTAL FOR 'ACHIEVABILITY	TOTAL WORTHINESS
Plant database: Outside Plant Experience	16	17	33
Plant database: Operational Experience (Internal)	15	17	32
Training: Maintenance Simulator/aid	10	20	30
Alarm filtering + Reduction	10	19	29
Equipment Condition Monitoring	10	18	28
Prioritisation of Alarms	10	18	28
Plant Database: Plant Standing Orders	8	18	26
Post-Trip Alarm Analysis and Diagnosis	17	8	25
Training: Operator Simulator	11	14	25
Emergency Planning Advisor	15	10	25
Condition Monitoring: System Behaviour	13	11	24
Alarm Analysis and Event Diagnosis	15	8	23

It should be noted that other application areas such as 'plant operation' and 'maintenance' were also originally to be analysed. It was subsequently found, however, that most of the important safety - related applications for these areas had already been adequately covered in other application areas, for example, 'maintenance simulator aid' had already been considered under 'Training', although it was recognized that the type of system being discussed had wider ramifications than just safety.

The working group emphasised, however, that the one important area which had not been adequatley covered is that of risk management. Although Section 4 addresses this Area it is considered that this area is worthy of a detailed Conference in its own right. Such a conference could consider the applications of AI technology in three main areas:
- Improving/providing the tools for carrying out PSA more effectively
- using the PSA results more effectively as a risk management tool
- consideration of human factors (i.e. both qualitative (task analysis) and quantitative analysis of potential human error).

The consideration of human factors should include both operators' primary failure and management/organisational effects.

3.6.2 Plant Data Management

The operation of a nuclear power plant needs various data types and huge amounts of data to be handled. Finding, collecting and interpretation of the data consumes a lot of plant engineers' time. The difficulty of the data management problem is even more complex due to diversity of the users, where each engineering task has its own requirements for data management.

To be useful, data management tools need to:

- Access various types of data bases
- Aid users in locating data
- Interpret data
- Be simple enough to interact with users of different backgrounds.

The type of expert system discussed by the Working Group was one which accessed conventional data bases but used AI techniques to present the information to the user both more efficiently and in a more usable form.

Possible specific applications which were identified, were:

1. Operational experience of the plant
2. Technical information of the plant (technical specification, PSA,etc.).
3. Operational plant experience from similar types of plants (i.e. a general data base covering plant experiences on a large number of NPP's).
4. Plant standing orders (i.e. an expert data base which collated, monitored and managed the wide array of plant standing orders which are used on NPPs).

All the above stated applications can be used for plant management. Regarding potential benefits the applications (1) and (3) had the highest scoring. Indeed, applications incorporating operations experience of NPP scored the highest of all the applications considered.

Regarding achievability of such applications all the stated applications had similar scoring. It was considered, however, that only the incorporation of the Plant Standing Orders with an expert system/database were achievable on a PC. The amount of data required for plant operations and technical support information is vast.

3.6.3 Training

The usual objective of a computer-aided instructional tool is to teach concepts and support the thought processes, rather than practice mechanical operations with the system. Computer-aided tools could be developed for operators (both in normal and emergency operations) for:

- auxiliary operations
- maintenance personnel
- specialists

The training aid could be a stand-alone interactive graphics system which contains a detailed model of the plant being taught, and a detailed model of the correct maintenance equipment specifications. The explanation facilities in ES lend themselves to computer aided instruction. Potential problems focus on the need of the developer to set up reasonable diagnostic procedures for multiple fault detection, and this is very plant dependent.

Operator's simulators would be much more difficult to construct than those for maintenance because the required system size is greater and the knowledge is complex.

The scoring analysis indicated that the provision of a maintenance simulator/aid was more achievable in the immediate future, and this particular application was one which prompted much enthusiasm in the working group. Particularly favourable aspects of a maintenance simulator/aid, i.e. a simulator to train maintenance personnel in difficult procedural tasks, were its perceived significance in reducing risk and also its possible significant 'Payback' savings. The Working Group also considered other types of systems which could be used for the maintenance staff to aid/advise them during the actual tasks (there are also important planning/scheduling possibilities for this application). Other favourable aspects of a maintenance simulator resulted from:

- implementation on a PC
- the program knowledge in a shell
- easy validation
- low cost

These aspects made this particular application an extremely favourable demonstrator-type project.

3.6.4 Condition/Safety Status Monitoring (excluding post-accident situations)

This type of applicaton can be divided into two main areas - equipment monitoring and system behaviour monitoring. The first application is concerned with a piece of machinery, however complex, which provides advance warning before partial or complete failure, so that preventive maintenance can be provided. Various types of sensors can be used, such as vibration sensors, microphones for sensing noise, ion-specific electrodes for measuring corrosion product, etc.

The second application deals with system behaviour, and gives information on whether the system is behaving "normally" or "abnormally" before it actually gets into an alarm trigerring situation. Obviously this task is more complex, as the system components and their connectivity should be defined dynamically. However, the benefits of such advance warning, for preventing failure situations would justify this effort. The inputs to such a system behaviour monitor are normally the plant sensors. However, it may be useful to observe also the control system outputs, as these are normally controlled within operation bands and higher/lower than usual output may indicate abnormal load or a fault condition.

The working group concentrated on the following generic range of applications within this area, i.e.:

- equipment condition monitoring
- system behaviour monitoring

The former of these applications scored high in the subjective analysis carried out (see Table 3.3), with reasonably good 'Benefits' and high 'Achievability'.

This particular type of application can be achieved on a micro if the equipment data capture is not too complex, and indeed there are numerous non-nuclear applications wherein this type of condition monitoring has been achieved, e.g. the expert system shell 'XiPlus' is being used in a number of industrial firms for vibration analysis. This type of application is also well suited as a demonstrator project.

3.6.5 Alarm Analysis and Diagnosis

This type of application, which is one of the most widely applied, deals with the treatment of alarm information once the plant equipment or systems have gone outside the safe limits. The application can be divided into three parts: alarm filtering or reduction, alarm prioritisation, and alarm analysis for event diagnosis.

Alarm filtering is needed for reducing the load on the operator for correcting the alarm situation. Redundant or secondary alarms should be filtered out, usually by advance analysis. For instance, if an electrical breaker connected to a pump is opened there is no need to give dependent alarms such as a low pressure at the outlet of this pump.

Alarm prioritisation deals with the order of importance of the alarms existing in a given moment. The operator should get the information on what should be his most urgent task, and not on the alarms in their order of appearance.

Alarm analysis for event diagnosis is important for comprehending the source of a given alarm combination, so as to deal with the primary cause and not only with the symptoms. This is a more complicated task, as many event or fault 'trees' may be involved, but again, the usefulness of this type of analysis for correct and quick identification of a fault situation is important for the safe operation of a nuclear power plant.

The following three specific applications were considered for this application area:

- alarm filtering and reduction
- prioritization of alarms
- alarm analysis and event diagnosis

The analysis indicated that only the latter application was likely to be useful as a demonstrator project, and this would probably be using a simulator or user input to indicate alarm situations.

The second of the above applications is the most difficult to achieve, but also presents the greatest benefits. This is an area in which further research is needed to improve the possibility of successful implementation of such a system.

3.6.6 Accident Management

In emergency situations when multiple alarms are presented, operators of the plant are under pressure to decide on actions to bring the reactor to a safe shut-down condition.

An expert system can be used for accident management to aid operators, but this is the most challenging application of expert system in nuclear safety.

The requirements for such an expert system are as follows:

- signal validation and alarm filtering allowing the operator to concentrate on the most important information
- diagnosis of the plant status (with explanation) which may be overlooked by the operator
- track of the accident progress
- evaluation of system and component availability
- list and explanation of the emergeny procedures

- track of recovery actions
- success path monitoring
- real time work

Potential benefits of expert system application in accident management are high (especially so in risk reduction) but achievability is relatively low.

A good starting point for this stated application would be the extraction of the knowledge about specific plant behaviour under accident conditions.

3.6.7 Emergency Planning

In NPP, the emergency planning is a very important feature of operation and design to meet Design Basis (DB) accidents and beyond, including post accident operations. During an accident condition the Safety Engineer/Operator is left with too many decisions to be made, subject to the condition of the plant prevailing at that time and propagation of the accident scenario. The operator's decision is subject to the context of the accident situation and the resources at his disposal. An emergency planning advisor could advise the operator on the severity of the accident, the plausible consequence and the measures to be taken for mitigating the situation. It could compute the status of the plant and advise progressively on what action to take. The advisor could have the feature of identifying the various built-in accident scenerios and advise accordingly. The system could be real-time or near real-time so that the operator action can follow the scenario. The expert system could be built around design basis parameters and could update the system values as the event propagates.

Such a system would have the potential benefit of a high reduction in the risk of a severe accident and thereby increase the credibility of reliable operation. The system, although fairly complex, should be achievable with existing software and hardware. Development costs, however, could be high.

3.7 Worthiness of Identified Applications

Table 3.3 gives the results of the 'scoring' exercise which was carried out to indicate the perceived worthiness of the above potential applications considered. The results on Table 3.3 are also summarised in order of worthiness, with the sub-totals for 'Benefits' and 'Achievability' also included, in Table 3.4.

From Table 3.4 it can be seen that the two applications "Plant database: outside (or internal) plant experience" were scored with the highest 'worthiness'. These results were discussed by the working group and it was generally agreed that these were applications which provided good benefits and were achievable with current technology. The group also noted, however, that it is difficult to discriminate between the top five or six applications, thus giving an indication of the possible 'error-band' in the subjective analysis.

It is important to note that all of the applications included in the analysis are considered to be worthy of implementation at some stage. The applications, which have been awarded the lowest 'overall' scoring were generally those for which the 'benefits' was high but the 'achievability' low, eg. "Alarm Analysis and Event Diagnosis" is considered to have considerable benefits but not easily achievable at this moment in time. Had this application received the same ranking for 'achievability' as the current top-scored application then this would be of equal worthiness.

It is considered that this analysis has highlighted the following:

(i) The applications (with the exception of the current top-scored application) which provide the greatest potential benefit are the most difficult to achieve with current technology.

(ii) The applications which are easily achievable do not provide the greatest benefits (although the benefits are still worthy enough to pursue)

It would appear that 'benefits' and 'achievability' are currently inversely proportional to each other. This relationship is not dependent, however, and it is therefore feasible to raise the 'worthiness' of the applications lower down in the Table by improving the achievability of success.

These latter events tended to score poorly on:

- software
- development cost
- portability
- tolerability of error (generally)

It is therefore recommended that research projects should be initiated with the aim of improving the success of these parameters. It is recognized that the nature of these large-scale applications probably precludes eliminating a high development cost. The provision of suitable software with improved QA and validability, however, should be achievable in time. The portability of these large scale applications tends to be low since there is a large amount of plant specific data which needs to be incorporated. It is important to note that all applications for which validation is critical were given a low rating; this is in recognition of the current 'state-of-the-art' with validation of AI software, ie. there is a significant chance of the software not performing as required due to a programming error or basic software 'bug'. In order for the achievability of identified applications to be improved it is essential that vigorous and robust QA/validation of the AI software is carried out. Furthermore it is essential that the AI software developers produce tools which enable the developer(s) to also validate the expert system program.

3.8 Regulatory supports and constraints

The Working Group considered that it was very difficult to make constraints in an active and recent research and development area. The regulatory supports and constraints should be focussed to increase the confidence in expert systems for the operation staff, authorities, and also the public.

One of the requirements for the implementation of an ES was an independent verification and validation of the ES itself. The verification could include:

- definition of the extent to which the sytem is safety-related and the expected level of reliability and maintainability.

- definition of the sensitivity of the expert system to changes in the input data

- inclusion of the safety and radiological criteria in the knowledge base. This means checking the consistency of the expert knowledge with respect to safety guidelines and instructions.

The validation must point out the limitations of the expert system and direct the attention to cases when the ES should be used with caution.

Another requirement is for those ES designed to advise in critical areas (i.e alarm diagnosis and monitoring post trip analysis and diagnostic and emergency planning etc.) to work in real time (or 'near-real-time') and to handle the changing scenarios appropriately.

4. THE UTILIZATION OF PLANT SPECIFIC PSA-TYPE KNOWLEDGE IN ADVISORY SYSTEMS FOR OPERATIONAL SUPPORT

The use of PSA-type knowledge for Operational Support was identified in various papers presented by members of the Technical Committee. Some of these papers described systems which use PSA information directly, i.e. used cut-set analyses resulting from PSA. Other papers described systems which used PSA information either in the form of fault tree/event tree topology or in the form of logical statements derived from the fault trees and event trees used in the PSA.

Some of the systems described in the papers were not expert systems, although they demonstrated some characteristics of expert systems. Although the Technical Committee was primarily concerned with expert systems, for completeness all current systems were reviewed which used PSA knowledge to provide advice for operational support at nuclear power stations. This review, the results of which are described in Section 4.1. identified some features which the Committee considered important in the design of future expert systems for providing this operational support.

4.1 Current Status of PSA-type Advisory Systems

4.1.1 Systems using PSA results

Two advisory systems were identified, neither was an expert system.

The first, the Plant Risk Status Information Management System (PRISIM) [Ref. 1] use the cut-set results of a PSA/fault tree analysis. The system provides rapid access to this specific stored information, providing an on-line evaluation of probabilistic safety information based on current plant status. The system is essentially a near-real time advisory system for operational support.

The second system identified, the Essential Systems Status Monitor, ESSM, was the subject of a paper presented to the Technical Committee [Ref. 2]. This system uses PSA type fault trees in its knowledge base and has the capability of modifying these fault trees in very limited ways to follow specific station system reconfigurations.

The ESSM also modifes its knowledge base to follow unavailabilities of a plant which has failed or is being maintained. The ESSM provides two types of advice, firstly a quantified estimate of the risk, as defined by the failure frequency of the safeguards systems assessed over all the initiating events, and secondly advice on the combinations of plant functions currently unavailable which would significantly improve the risk assessed if these functions were made available. The ESSM is also a near-real time advisory system for operational support.

4.1.2 Systems using PSA knowledge

The Working Group recognised that not every advisory expert system had been identified, but only those systems known to the Working Group. The salient features of each case are summarised as follows:

4.1.2.1 SMART

Electrical Power Research Institute, EPRI has developed an expert system supporting operators in the process of diagnosing and controlling ICS-related transients [Ref. 4]. Results of PRA studies were used to create knowledge bases represented by 70 logic rules. The computer code was written in a LISP based software called SMART and is in use on PC.

This expert system begins to function when a plant is running at power and suddenly experiences a transient and becomes unstable. Questions are asked of the operator who responds by answering TRUE or FALSE. Based on his responses, the code will either ask additional questions or when an intermediate conclusion can be reached, it will be displayed, possibly with suggestions on what to do next. In this way, the operator, providing input to the expert system, receives information that helps him in his task of reaching shutdown. It goes without saying that this system is not smarter than the operator is, but in certain abnormal situations can support the operator's selfconfidence.

The knowledge base (rules) can be directly adopted for other commercial shell tools, like Insight 2+ (LEVELS) [Ref. 11] and easily modified if required.

As an example, an ATWS event without ICS was considered. The Line Reasoning Report capability of Insight 2+ can provide an excellent explanation of the scenario.

4.1.2.2 DIARAEX

DIAREX [Ref. 8] is a real-time expert system which has been developed to diagnose plant failure and to offer a corrective operational guide for Boiling Water Reactor (BWR) power plants. The failure diagnosis model used in DIAREX was systematically developed, based mainly on deep knowledge, to cover heuristics. Complex paradigms for knowledge representation were adopted, i.e., the process representation language and the failure propagation tree. The system is composed of a knowledge base, knowledge base editor, preprocessor, diagnosis processor, and display processor. The DIAREX simulation test has been carried out for many transient scenarios, including multiple failures, using a real-time full-scope simulator modeled after the 1100 MWe BWR power plant. Test results showed that DIAREX was capable of diagnosing a plant failure quickly and of providing a corrective operational guide with a response time fast enough to offer valuable information to plant operators.

4.1.2.3 IFTREE

Using Artificial Intelligence (AI) techniques, a tool has been developed [Ref. 6] to support scoping analysis of very large problems. The IFTREE program can be used to perform limited real time assessments of full plant PRA models. Current efforts are directed at modifying IFTREE to address the functional needs of emergency response personnel. The primary emphasis is on development of explanation capabilities to allow direct support of personnel without formal PRA training. This involves not only decoding of the analysis results but also capabilities to answer menu developed questions based on the model knowledge. A second issue is the specialization of the IFTREE cut set generation algorithm to address specialized analyses based on confirmed system statuses.

4.1.2.4 GOAL TREE

Goal tree success tree model [Ref. 7] is a new type of deep knowledge representation model (herein called GTST model) for developing expert systems for plant monitoring, controlling and alarm handling activities. This model can completely and rigorously describe a process plant and its operations. It incorporates a structured approach that shows how a specific objective in a plant is achieved. This is done by defining the objective and partitioning it into a series of related sub-objectives or goals; these goals are then broken down into subgoals.

The partitioning of goals and subgoals continues until their description can not be made without referring to plant hardware. At this point logical model of the hardware should be represented in the form of success trees.

A GTST model may be used to represent an expert system's knowledge base. An expert system for a typical feedwater system operation of a pressurized water reactor is developed on an IBM-AT, by using the micro-PROLOG language. The system showed that the GTST model can be effectively used in developing the expert system knowledge base and improves search strategies.

The expert system is developed such that it can be used to operate in a near-real time environment, in order to provide quick response to operator's needs.

4.1.3. Systems Modifying PSA Knowledge

One existing system is the EXPRESS tool [Ref.5]. The method implemented by the EXPRESS software revolves around two main phases performed by the second order GENESIA I engine.

The first phase is aimed at grouping the components of a system into larger components categories - or macrocomponents - according to the consequence of their failures (blockage or leakage).

The second phase deduces the failure consequences, for each component, in terms of path losses, according to the macrocomponent(s) to which the said component belongs. Moreover, the possible causes of these failures are stated according to the configuration and the mission studied. These two phases lead to the generation of a simple rule base of order 0 (GENESIA I-AUTO) written in GENESIA I language.

If a hand-written rule base (GENESIA I-MANU) concerning the undesirable event and the boundary conditions mainly is added (GENESIA I-AUTO), all the data required to build a fault tree are then available.

The method described above has been successfully used and validated on the Auxiliary Feedwater System (AFS), the Residual Heat Removal System (RHRS), and the Safety Injection System of the PALUEL

Nuclear Power Plant. The basic principles underlying this method remain valid, whatever the reliability model set as a goal (fault tree, state graph).

The future developments will consist in the integration of these expert systems in the LESSEPS software that is developed for the complete computerization of the French PSA.

The aim of the LESSEPS software is to allow the organization of all data managed (reliability data, reliability models as fault trees, state graphs and event trees) and the updating of the PSA when reliability data are changed.

When the above described expert systems are integrated in this software the objective is then to use the computerized PSA as a tool for design conception (by taking into account a modification of the system topology) and for an operator aid (by evaluating the Allowable Operating Times, for example).

4.1.4. Features of current systems

Most of the papers presented during the Technical Committee Meeting used an approach revolving around three main phases:

a) The choice of a plant knowledge representation and the extraction of the needed information for the modelization of the problem to be solved (diagnosis, reliability studies....).

b) The modelization, using the extracted information for creating new concepts specific to the application: an increased but specific knowledge base is then obtained.

c) The processing of the so constructed models for obtaining the required goals (diagnosis, safety assessment, maintenance strategies....)

This approach seems to be valid, either for systems using PSA results and knowledge (as described before), or for systems using a plant knowledge representation but not PSA type knowledge.

In this way, EXTRA is an expert system for industrial process control [Ref 3]. The main objectives are diagnosis and operational aids. From a methodological point of view, EXTRA is based on a deep knowledge of the plant operation and topology and on qualitative physics principles.

A specific application of EXTRA is developed for the BUGEY unit 2 (a 900 MWe pressurized water nuclear unit) concerning the electrical power supplies. This system called "electrical power supplies supervision" gives diagnosis in real time of the electric incidental situation origin. A connection to a database makes the expert system able to supply operators with information concerning the consequences of the electrical power system failures on the safety systems, the equipment measurement sensors and certain automatic devices (availability, unavailability, validity, etc..)

The following diagram (Fig. 4.1) presents these fundamental concepts and interrelationships between current advisory systems, whether they use a PSA knowledge or not.

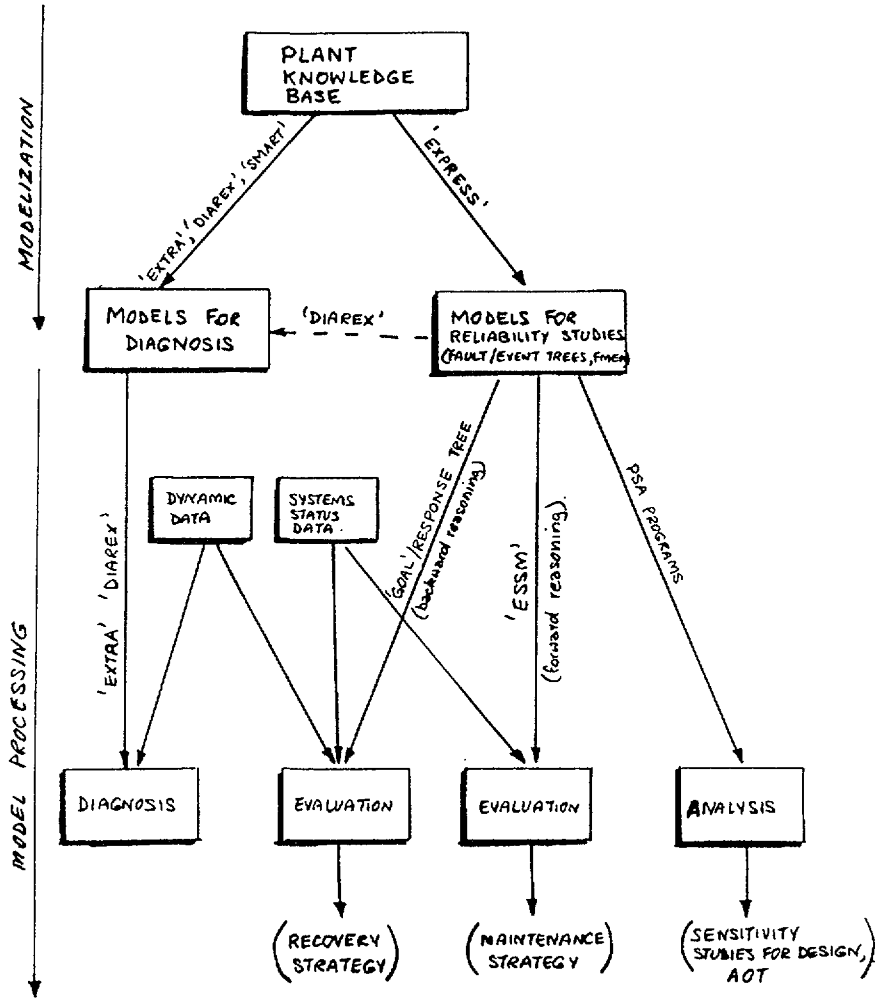


FIG.4.1. Interrelationships between current advisory systems.

What Can Be the Role of Expert Systems in Such a Diagram?

Expert systems (or, generally speaking, artificial intelligence techniques) can be used:

- for the knowledge representation
- for the generation of the models (man-machine interface considered as an aid for constructing these models, or even an automatic generator of these models)
- for the processing of the models, particularly for diagnosis.

The models constructed by the expert systems may then be used:

- for direct evaluation of sensitivity studies for design optimisation, determining allowable outage times, etc., through the use of standard PSA computer programs
- for evaluation of maintenance strategies by using further knowledge of systems status and through forward reasoning processes (e.g. ESSM system)
- for evaluation of recovery strategies by using further knowledge of dynamic data and systems status data and through backward reasoning processes such as response tree (e.g. 'GOAL' system).

4.1.5. Implementation of PSA-type knowledge into a knowledge base

Knowledge of PSA may be particularly useful for emergency response applications because a constructed PSA reflects the best knowledge about all the ways a plant might fail, including various transients an operator will experience. The comprehensiveness of PSA provides the operator with detailed information on events that might occur.

When a PSA of a plant is constructed, a very thorough analysis of each type or class of accident is carried out. This is usually described by a diagramatic/probabilistic process where fault and event trees are drawn and evaluated using performance and failure rate information based on past plant and industry experience. From such analysis, important sequences can be ranked using severity and/or frequency of occurrence as a measure.

However, the examination of an actual incident occurrence at a plant identifies that a PSA does not treat the detail of any actual sequence. Such detail is often unnecessary in the characterization of these sequences for risk comparison and reliability purposes, but to provide a knowledge based expert system supportive to an operator just such specific information is needed. It does not matter that it is a rare occurrence, because the operator has to deal with it at that time. It is therefore concluded that FTA and PSA information is not enough. The quantitative results from a PSA can play only a guiding role in this analysis. [Ref. 4]

As an example in DIAREX (a real-time expert system has been developed to diagnose plant failure and to offer a corrective operational guide for BWR) the knowledge representation realizes this concept for failure diagnosis and operational guidance [Ref. 8].

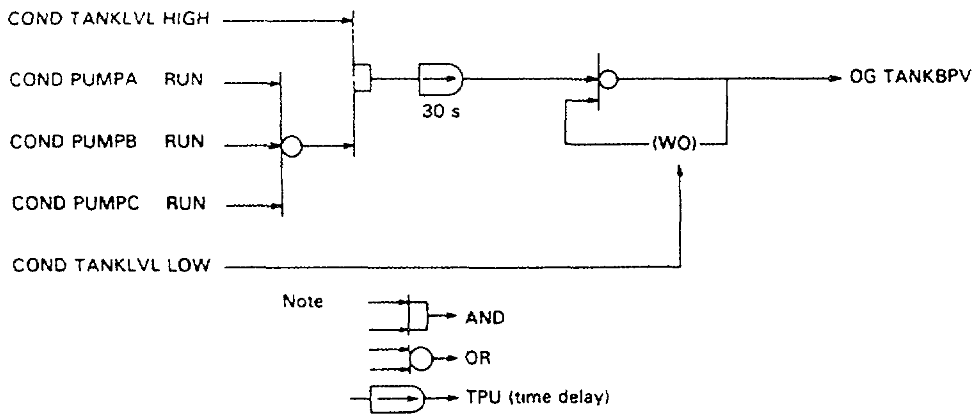
To construct the failure diagnosis model exhaustively and systematically, FTA (fault tree analysis) and FMEA (failure mode and effects analysis) were utilized.

As a top-down approach to event analysis, a FTA is performed by developing the top event, that is the objective event, into the causal events. On the other hand, FMEA is a bottom-up approach to event analysis and is performed by evaluating consequences of a causal event.

The knowledge base consists of failure identification rules (FIR), failure propagation knowledge and operational instructions.

FIR contains bits of knowledge used in failure identification and failure diagnosis. It is described in a so called process representation language (PRL) which is similar to production rules (AND, OR). Time dependent logical operators are also introduced to represent the dynamic behavior of the plant process, (HOLD TPU, time delay pickup).

As an example see Fig.4.2 [Ref. 8]. This illustrates the rule representation too.



OG TANKBPV = (COND TANKLVL LOW HOLD (COND TANKLVL HIGH .AND (COND PUMPA RUN OR COND PUMPB.RUN OR COND PUMPC RUN) TPU. 30)

FIG.4.2 An example of a rule described by PRL (see Fig 7 of Ref. [8]).

Failure propagation knowledge represents the dynamic behavior of process failure propagation in a tree style and each mode in the tree has several attributes (AND, OR or INHIBIT). Because the failure propagation is described through the physical relations of the plant process, including control system and plant interlocks, it is preferable to represent these relations in tree style, from the inference efficiency viewpoint (FPT representation, failure propagation tree). See Fig.4.3 [Ref. 8]*

In DIAREX the FPT structure was developed into many "cut sets" as accomplished in FTA to improve inference efficiency.

From Fig.4.4 [Ref. 8], the rule representation of the top part of this cut set is:

IF FWH break OR FWH drain trouble THEN FWH temperature decreases
IF MCV closure OR MSV closure THEN Dome pressure increase
IF PLR flow increase OR FWH temperature decrease OR
    Dome pressure increase THEN Neutron Flux
IF Dome pressure increase THEN Dome pressure high
IF Neutron flux OR Dome pressure high OR Reactor water
    level low OR MSIV closure THEN Reactor scram

---

* It is easy to give rule representation, i.e.,

IF NODE-1 OR NODE-2 THEN NODE-5
IF NODE-6 AND NODE-5 THEN NODE-7

NODE is used instead of FAILURE or PROCESS state because it is shorter.

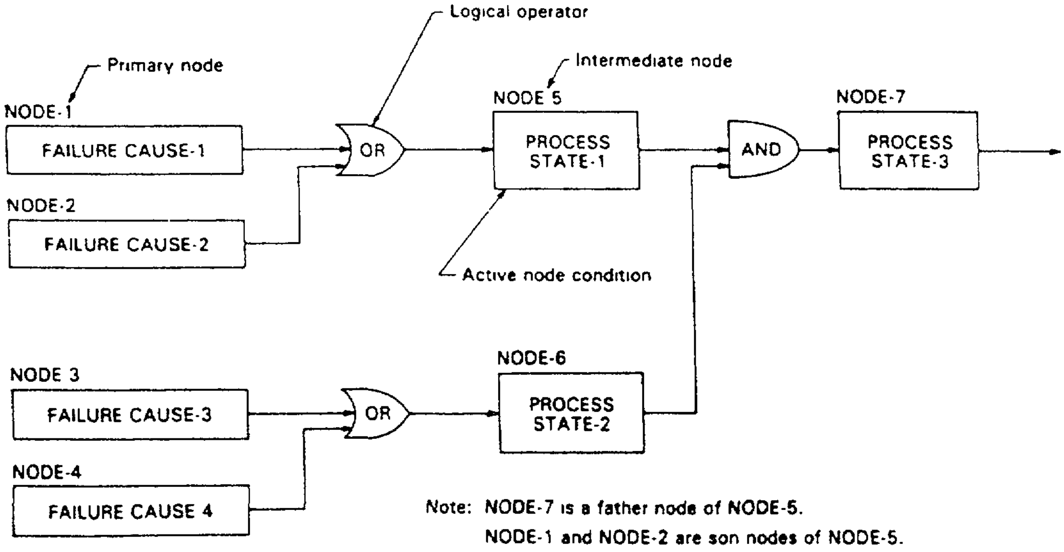


FIG.4.3. An FPT example (see Fig.8 of Ref. [8]).

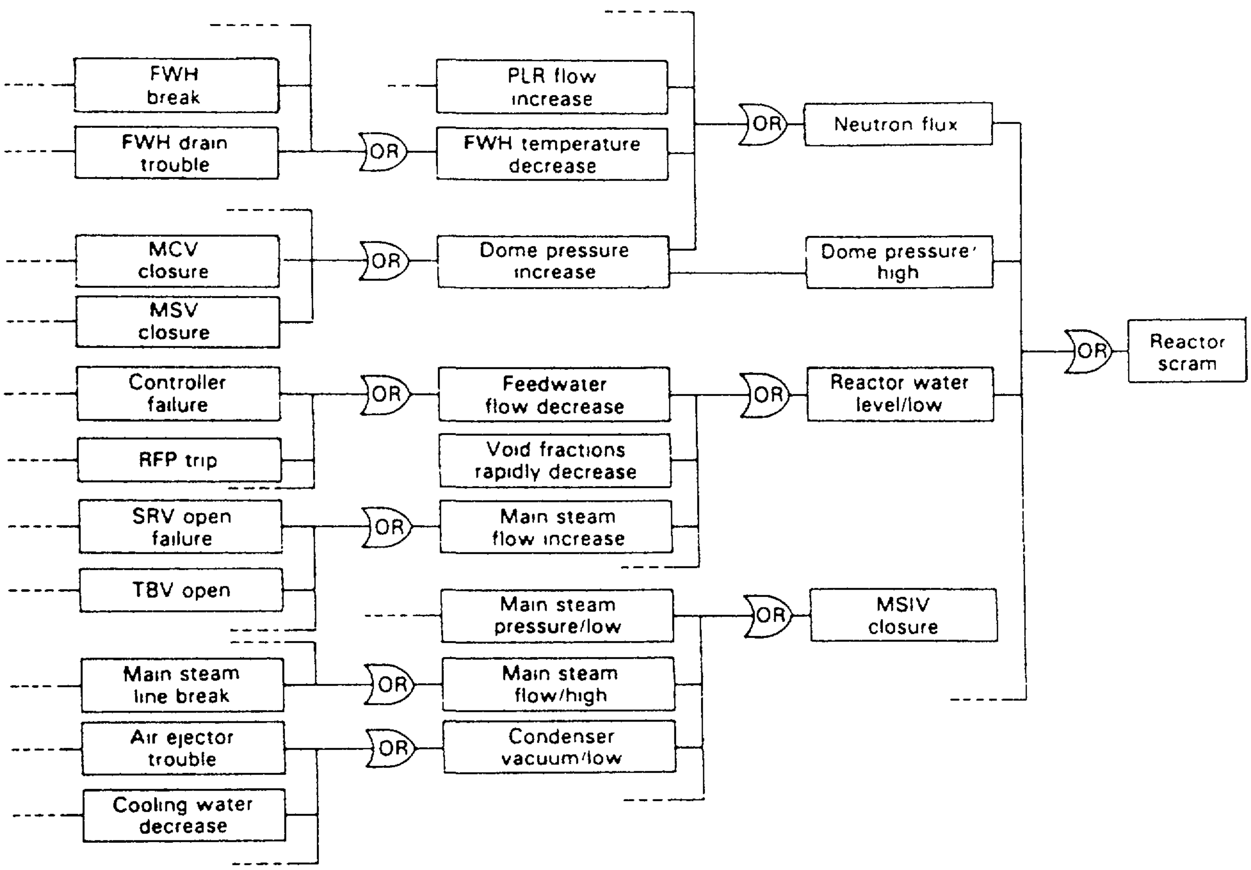


FIG.4.4. A part of the FPT. In this figure, FWH = feedwater heater, MCV = main steam control valve, MSIV = main steam isolation valve, MSV = main steam stop valve, SRV = safety relief valve, and TBV = turbine bypass valve (see Fig.9 of Ref. [8]).

This knowledge base representation can be readily put into a shell like Insight 2+ [Ref. 11] and possible sequences are checked using the line Reasoning Report of this shell.

FTA can also be utilized in automated fault diagnostics based on fuzzy mathematics [Ref. 12].

4.2 Design Features for Future Developments

The review of current PSA-type advisory systems described in Section 4.1 identified some characteristics in the design of these systems which were considered important for future designs of more comprehensive advisory systems, particularly where these systems are expected to be expert systems. These characteristics were related to possible application scenarios, to whether the systems are to be near-real time or time independent, to the reasoning techniques required, and to the information requirement.

4.2.1 Application Scenarios

The current PSA-type advisory systems operate in conditions within the design basis of the plant and generally for the normal power operation of the reactors. The information base used from the PSA-type analysis is derived from the full PSA (level 1), i.e. includes contributions from all initiating events. The information is based on plant fault tree models and event tree models. These systems are currently not expert systems. Advice given for these operational conditions within the design basis of the plant may be:

a) at the design stage, system optimization
b) operationally, risk management advice in the form of advice for planning plant maintenance, advice on acceptable times to operate at reduced plant availabilities, and advice on plant replacement strategies.

The next major area of application for an advisory system will be for major plant failure conditions, where the objective is to establish the reactor in a 'safe' shut down state. For this application the advice required will include diagnosis, accident management to minimize risk and advice on maintaining the 'safe' shut down state when plant failures occur. The advisory system will not only need a knowledge base derived from PSA-type information, but also, structured information on Emergency Operating Procedures. The PSA-type information will now not require information on all possible mitigating events, but only on the possible event(s) diagnosed. This information is likely to require modification as the advice requirements change. The modification will include changes to the failure characteristics data of the plant components, and changes to the logical structure of the fault and event trees information. It is expected that for accident management the PSA-type information will need to be expressed in success form.

The final area of application is envisaged as the beyond design basis conditions. This is expected to be the most demanding on the advisory system in that not only will the knowledge base be considerably larger and subject to greater changes, but also the basis for the advice may change. The advice required may now be in relation to:

a) critical safety functions, e.g. reactivity control, clad and fuel integrity control, primary heat removal, secondary heat removal, containment integrity, steam generator tube rupture prevention
b) economic plant damage
c) containment integrity
d) consequence limiting
e) evacuation planning

The characteristics of the scenarios are summarized in Table 4.1.

4.2.2 Time Independent Systems

Utilization of PSA knowledge may be for example:

Design Evaluation

For the conception of particular safety systems, very detailed reliability studies may be useful for the detection of weak points (failure combinations that could not be imagined using deterministic reasoning).

TABLE 4.1. POSSIBLE SCENARIOS FOR EXPERT SYSTEMS TO PROVIDE OPERATIONAL SUPPORT ADVICE

Plant Condition	Expert System Advice	Information Required	Criteria
Design	System Optimisation	Full PSA	DB
Normal power operation	Risk management (Planning maintenance Defining times)	Full FTA	DB Operation
'Minor' plant failure	Risk management (Plant replacement)	Full FTA	DB Operation
'Major' plant failure	1. diagnosis 2. accident mgmt. 3. shutdown maintenance	EOP'S and PSA with limited events (data changes) (structure changes)	DB Shutdown
Beyond DB incident	- Accident mgmt. - Development of original procedures	Accident Management and EOP's and PSA with limited events (data changes) (structure changes)	- critical safety functions - economic plant damage - containment integrity - consequen. limiting - evacuation plans

For performing <u>sensitivity studies</u> when reliability data or topological data are changed (in this case, the whole knowledge included in the PSA must be used); for example, the core melt probability will be recalculated when the topology of a critical safety system is changed.

In these two cases, expert systems may be used for updating safety and reliability studies when the knowledge base is modified,and they mainly contribute to the implementation of a "living" PSA.

Evaluation of important operational parameters

PSA results may be used for evaluating the increased risk when operation is allowed, despite the partial or total unavailability of a safety system (Allowable Operating Times).

PSA results may be also used as decision criteria allowing or not the starting of a plant, despite the partial or total unavailability of a safety system. In this case also, expert systems will be useful for updating all of these operational parameters when the knowledge base is modified.

Another example of a time independent system for accident/incident management and operator training is described in Appendix 2.

4.2.3. Near-real time systems

Considerable benefit operationally may be derived from the use of advisory systems in "near-real" time. In the context of this paper, the description near-real time has been used for systems which provide operational advice in the short-term.

In normal operational conditions it is envisaged that such systems will provide advice similar to that provided by the ESSM [Ref. 2]. Advice based on probabilistic information is used within defined limitations for maintenance strategies, plant replacement advice,etc. The defined limitations may be non-probabilistic information such as deterministic rules defining acceptable plant availabilities.

In accident management it is recognised that the use of an advisory system in near-real time which is based on an expert system can be particularly beneficial. The system is able to respond to the developing accident situations and advise the operator of the most appropriate recovery actions. Thus, the operator is able to concentrate on the fewer systems needed to manage the accident, and he is able to check the reasoning of the system.

4.2.4. Backward reasoning techniques

In case of a severe accident the emergency management or the safety engineer has to decide the short and long-term recovery actions in order to transfer the NPP to safe conditions and has to control the efficiency of the chosen recovery procedure.

How to get from the current emergency status to safe conditions has to be defined by backward reasoning from the goal, the safe plant status, to available counteractions.

A set of counteractions and available components by which a safe plant status can be achieved, is called a success path. Several success paths are possible because of redundancies in a NPP and their interconnections. The success paths are represented by the so-called response tree. A description of response trees is given in Appendix 3.

The response tree method has been tested at the Low Pressure Injection System at LOFT [Ref. 9]. All success paths are not equally probable and desirable; that means not equally reliable.

Before the counteractions are undertaken, those responsible require information about the most reliable recovery procedures at the current plant and the expected (precalculated) event sequences.

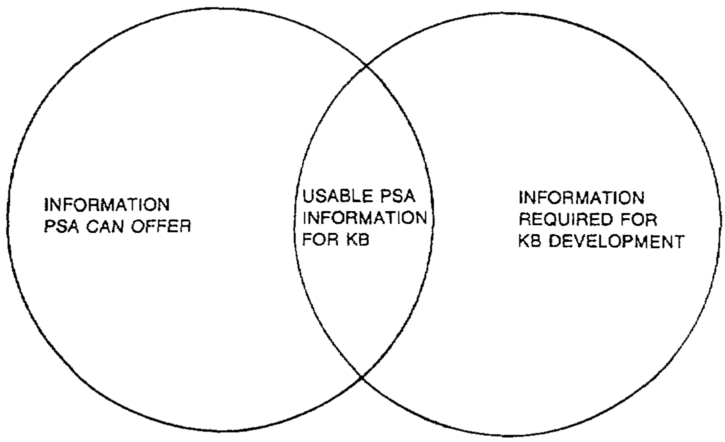
A PSA, which is restricted on the needed (safety-, operational-,...) systems and components in the course of recovery procedures, has to take into account the most severe loads and failures in the course of the accident due to, for example running time of a pump, cycles of valves, exhausting of water storage.

If current and estimated plant conditions exceed design values the probability for system failure will rise. Therefore the most reliable procedures will be those ones without any violation of design basis values.

The results of the PSA about the recovery procedures will be used to qualify the procedures depending on the probability of success. The response tree method is expected to be used in the German Accident Management Expert System [Ref. 10].

4.2.5 Information requirements

Concerning PSA as a basic source of knowledge base (KB) development there are two aspects. On one hand KB development requires some information which PSA can offer, on the other hand not all information which PSA can provide is usable for the K.B.



It is evident that PSA knowledge is not enough to build up a comprehensive expert system knowledge base. There are many other resources which can contribute to generating the KB, such as:

- results of simulation model
- plant knowledge
- safety, maintenance and testing documents
- reports on performance history
- interviews with experienced personnel staff
- technological knowledge
- available software and hardware environment
- experimental data
- international and national experiences

All this information has to be filtered, classified, structured and organised in order to get a useable, realistic knowledge base.

In this process one may utilize some well-known technique used in PSA, for instance handling, clustering and structuring events.

The KB development must be an iterative process involving real-life field testing. In this sense, the experience gained during employing such tools is also a natural information source for creating an acceptable working tool.

The methodology for representing a KB has to be also investigated and developed with consideration to logical processors in the field of AI technology.

Data which consist of quantitative values of possibilities can be employed in only a very loosely and restricted way, especially for real-time AI applications in safety situations. Unfortunate examples, such as the shutdown incident of the airbus aircraft, should caution the guarded use of the data.

5. CONCLUSIONS

5.1. Conclusions on the Development and use of Expert Systems in NPPs

1. Expert Systems, or more precisely Knowledge-Based Systems (KBS), are now a significant element in the research and development programmes of many Member States. They are not yet, however, an established feature of nuclear installations. The state of the technology now justifies increased effort, in particular areas which have been identified, to develop techniques and methods for specific nuclear applications.

2. The Technical Committee considers that a basic premise is that the nuclear industry cannot afford to be left behind compared to other technologies in using modern, sophisticated computing techniques to enhance all facets of its activities from design to operation, to maintenance, to waste disposal, all with the paramount imperative of safety.

3. It is very important therefore that expert systems modules should become established elements in the modernisation of control rooms of nuclear power plants and other nuclear installations, and in the design of the advanced next generation of these projects.

4. Expert systems in the nuclear industry for the areas of management and decision-making, maintenance, planning and design, operation, training, PRA/PSA, nuclear fuel handling, waste disposal, uranium process engineering and other areas, are and should be developed if the nuclear industry does not want to lag behind other industries in its methods of operation.

5. One of the many conclusions of our deliberation is the importance of involving the user in the process of developing an expert system application. Involve him again as the expert systems evolve through the stages of prototype, experimental system, demonstration system, operating system, growing system.

5.2. Conclusions Specific to the Applications of Experts Systems in NPPs

1. A subjective analysis has been carried out on the relative benefits of implementing Expert Systems in the different areas of safety in NPPs. This analysis identified some applications which should be practically achievable in the short term, such as a plant experience database, maintenance simulator etc.

2. A demonstrator-type project should be considered for short-term implementation on a PC for application as either a maintenance simulator, equipment condition monitor or a plant experience database.

3. There is an urgent requirement for research into:

   a) the provision of specific ES for complex NPP applications.
   b) the QA and validation of AI software

4. NPP applications in specialist areas such as real-time and condition monitoring should co-ordinate efforts with similar specialist areas in the non-nuclear sector.

5. NPP applications of Expert Systems which offer the greatest benefit to safety should be systematically identified and the research and development required for implementation pursued in a co-ordinated manner in member states.

5.3. Conclusions specific to the design of Expert Systems in NPPs

1. For the immediate future expert systems using PSA-type knowledge should be designed as systems providing operational advice, as distinct from systems providing direct plant control functions.

2. Advisory systems, using expert systems, should be developed for all phases of plant design and licensing systems providing long term advice and be investigated for all phases of operational and accident management support (system providing real-time advice).

3. The use of PSA type information should be investigated for use in all the advisory systems.

4. For each nuclear installation, a common knowledge base should be used for all advisory systems, and a co-ordinated approach should be adopted in the design of the different advisory systems e.g. similar to that suggested by the RAPID (EPRI) system.

5. For PSA type information to be of best use in advisory expert systems for different nuclear installations a common approach, common objectives, for performing PSA and common criteria should be adopted.

6. The development of the concept of 'living' PSA should be supported.

7. The use of response trees should be regarded as an appropriate information processing strategy for accident management advisory systems.

8. The methodical development of knowledge bases should be supported to take account of the type of knowledge to be used and the design of the expert systems used.

9. Due account should be taken of developments in the US and Japan in the development of future advisory systems for operation support.

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Appendix 1

EXPERT CLOSED LOOP PROCESS CONTROLLERS

(Rule-based controllers)

As described in the rapidly growing literature on the theory of "expert control", prototype closed loop process controllers are available, which contain a small rule-based knowledge base integrated in the software system of these controllers. With the help of the rules different setpoint changes, start-up, shutdown, etc. procedures can be carried out. The process controller works in such a way that under defined conditions it consults its rules on how to proceed.

These expert closed loop process controllers may have great potential in the computer control of nuclear plants in case of distributed, hierarchically organized control systems. In this case the low level process controllers can be equipped with inbedded intelligence expressed in rules. The increased intelligence of the low level process controllers may lead to a decrease of load on a higher (supervisory) level.

Additional information can be found in the literature of control theory and applications especially in the material of recent Symposia and Conferences of IFAC (International Federation of Automatic Control).

Based on the above, the area of expert closed loop controller seems to be worthy of further investigation for nuclear applications. Through the Agency it would be possible to call the member countries to state their interest and activity in this field.

Appendix 2

DEVELOPMENT OF COMPUTER SOFTWARE FOR USING PSA IN NUCLEAR POWER PLANT OPERATIONAL SAFETY MANAGEMENT

Plant operators and the regulator will occasionally receive reports on the occurrence of incidents which appear unimportant e.g. valves which were found opened when they should be closed and vice versa, human errors, maintenance errors, hardware failures, electrical equipment failures during the operation of the plant. The issue to be addressed is:

Do we need to do something about this?
Should the plant be closed down?
Can the plant continue operating?

and therefore is this incident important or can we forget about it.

If an incident is evaluated to be significant the decision could be made to model it on a real time full scope simulator so that operators are well prepared to ameliorate the effect of the incident happening in the plant. Such an incident could have happened in another plant somewhere else in the world.

Evaluation of Abnormal Event

The approach to utilizing PSA for the evaluation of abnormal events is to assess both qualitatively and quantitatively the contribution to core damage probability. Such an assessment is made possible by considering three measures of importance:

- Whether the abnormal event is part of an accident sequence which leads to core damage.
- If the sequence is one leading to core damage, whether the probability P_{mi} (the core melt probability of sequence i) is above a certain threshold, for example 10^{-5} per reactor year?
- If P_{mi} is greater than or equal to 10^{-5} per reactor year, the contribution P_{mij} of the abnormal event j to P_{mi} needs to be assessed.

The exact threshold values for deciding if an abnormal event is significant and therefore warrants action, may be influenced by the analysts and utility or by national policy. The basis of assessing the importance of an abnormal event may be considered as an evaluation of three factors:

a) Importance of the accident sequence to the overall plant.

b) Importance of the system to the accident sequence probability.

c) Importance of the event to the unavailability of the system.

Two risk important measures to evaluate a feature's importance in further reducing the risk and its importance in maintaining the safety level, may be identified, i.e. the risk reduction ratio and the risk achievement ratio. Figure A2.2 shows results for the safety systems of one plant. The risk reduction ratio indicates the factor by which core melt frequency could be reduced at the plant by improving system reliability. This factor may be used in assigning priorities to future improvements.

The risk-achievement ratio indicates the factor by which core-melt frequency would increase if the system had a failure probability of unity, i.e. if the system was not operable. This ratio may be used for assigning priorities to features that are most important in reliability assurance and risk maintenance.

Figure A2.1 shows considerations which should be given to all reported events. It is advantageous if a Level-1 PSA has been performed for the specific plant where the abnormal event has occurred since a search of the event tree bank will reveal which accident sequence and system, or systems, the incident influences. If the event belongs to an accident sequence leading to core damage it should be incorporated into the simulator usually used in that plant's operator training programme. If the incident or its precursors could initiate or belong to a previously unrecognized accident sequence this information could be used to generate a new accident sequence to update the plant PSA and also be programmed into the simulator. Concurrent with this activity an investigation can be made, by experienced PSA analyses and plant operators working together, with the objective of identifying and tabulating the human errors of commission and omission and system and component faults which are part of or related to the same accident sequence.

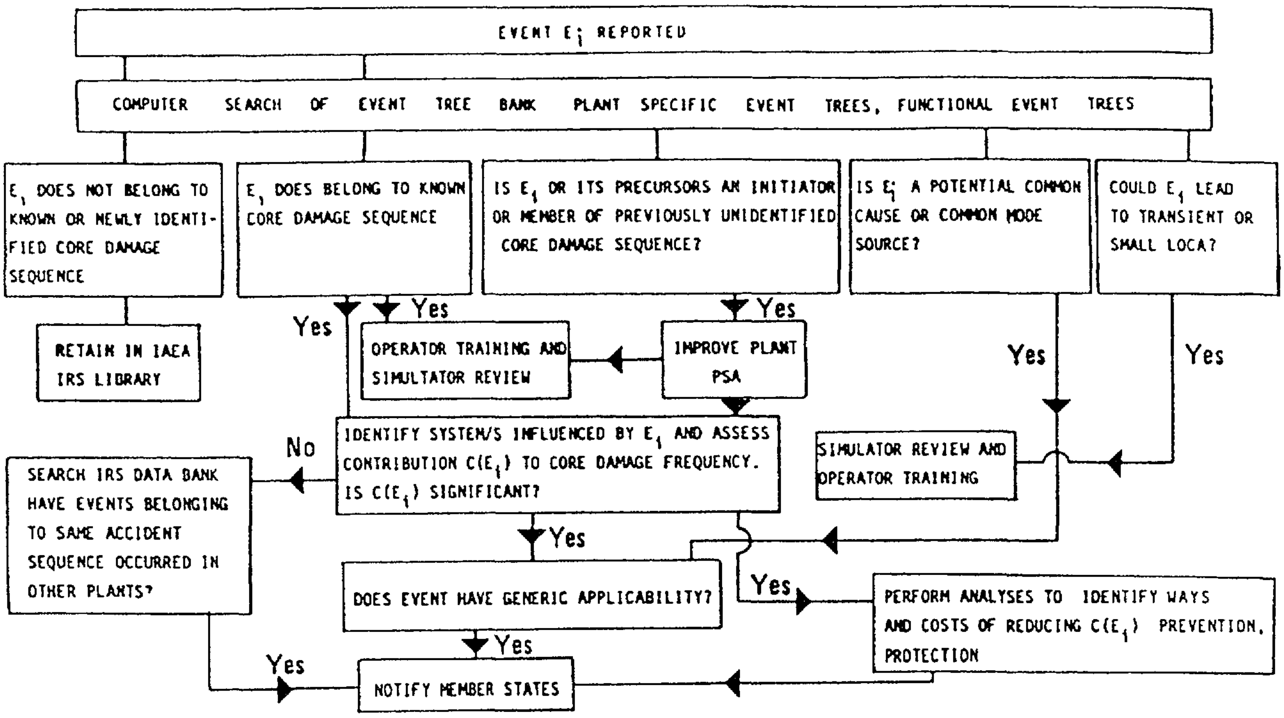

FIG A2 1 Overview of considerations of abnormal events

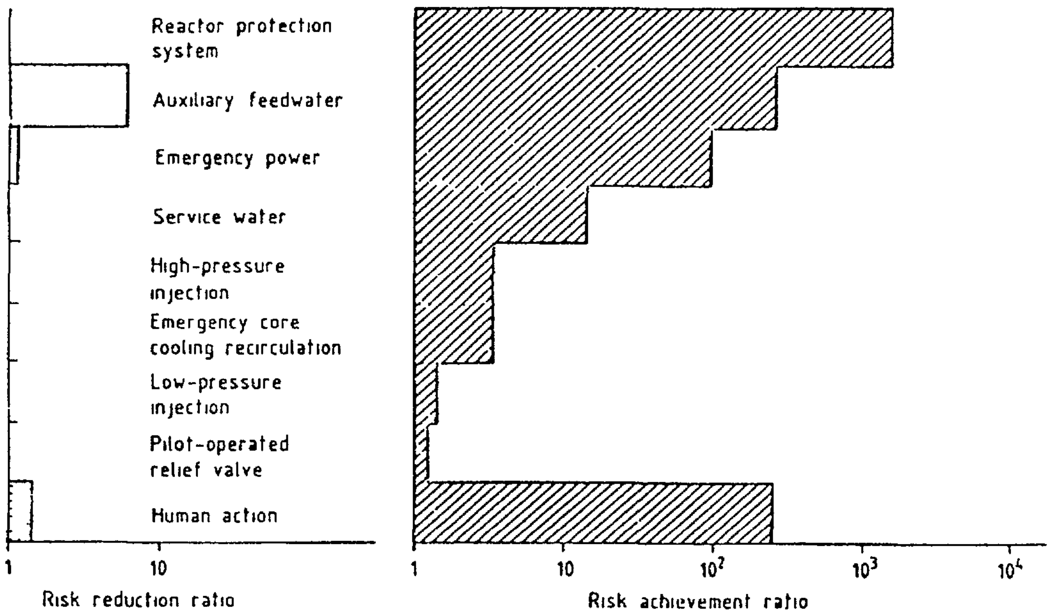

FIG A2 2 Example of the use of risk reduction ratios (left-hand scale) and risk achievement ratios (right-hand scale) to indicate contributions to core melt frequency in a nuclear power plant

When these data are tabulated they can be compared with real failure data such as the IAEA-IRS data collection. If several failures contributing to the same accident sequence have occurred either in the same plant or distributed among several plants this is a criterion suggesting prompt communication of this information to all active members of the data reporting system. If functional event trees which may be applicable to many plants of the same type, but having some differences in design, are used in the identification of an important event the resultant information is likely to be of international use. If the incident could lead to: a transient, a small loca (e.g. valve in primary coolant system failed open) or a common cause/common mode (particularly human error) event it may be labelled important before detailed quantitative assessment of the contribution to core melt frequency has been completed.

Evaluation of a System Fault Tree

Having determined that the event belongs to a particular system which is part of a core damage sequence, the next task is to evaluate the fault tree for that system. The main features of evaluation of a fault tree are outlined below.

The type of results obtained from a fault tree evaluation are:

Qualitative Results

a) Minimal cut sets:	Combinations of component failures causing system failure.
b) Qualitative importances:	Qualitative rankings of contributions to system failure.
c) Common cause potentials:	Minimal cut sets potentially susceptible to a single failure cause.

Quantitative Results

a) Numerical probabilities:	Probabilities of system and cut set failures.
b) Quantitative importances:	Quantitative rankings of contributions to system failure.
c) Sensitivity evaluations:	Effects of changes in models and data and error determinations.

The Minimal Cut Sets are used both in the qualitative and quantitative evaluations.

Appendix 3

RESPONSE TREES*

A response tree is a graphical representation of the success paths that can be used to provide a safety function. A safety function is ".. a group of actions that prevent melting of the reactor core or minimize radiation releases to the general public". A success path is a specific set of components and actions that can be used to provide a safety function. Because of the redundancies built into nuclear plant systems, there are usually many success paths that can be used to provide a safety function. However, not all success paths are equally desirable for use, and different success paths may require different plant states for implementation. In addition, the failure of equipment during an accident can disable certain success paths. Thus, the assessment of safety function status, monitoring of success path performance, and selection of a success path to implement is a task that can require a significant problem solving effort. Response trees were developed to assist this task.

The response trees used for the response tree evaluation project are based on the Low Pressure Injection (LPIS) at LOFT. The LPIS is used to provide the core cooling safety function when other systems are not available or appropriate for use. Figure A3.1 is a simplified schematic of the LOFT LPIS.

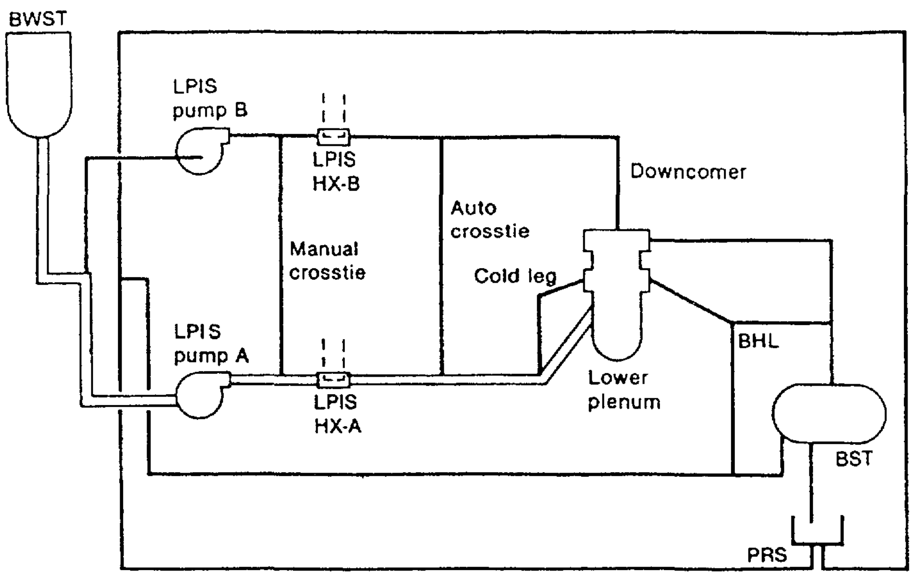

FIG.A3.1. LOFT low pressure injection system (LPIS).

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* Extracted from - NUREG 4272: Response Tree Evaluation, W.R. Nelson, H.S. Blackman, Sept. 1985.

To provide core cooling using the LPIS (or any typical cooling system), five basic types of elements are necessary. These are the water source, pump, route, injection point, and heat sink. The water source is where the water for cooling is stored. The pump is used to transfer the water to the reactor vessel. The route is the piping that directs the water to an injection point. The injection point is the piping where the water is actually injected into the primary coolant system. The heat sink is the mechanism for removing heat from the cooling water. (An active heat sink may not be required for a short-term LOCA situation, since cold water from one of the water sources is supplied to the primary system. When the cold water source is exhausted and recirculation from a sump is required, an active heat exchanger may become necessary.) Any combination of the five types of elements that can be used to provide core cooling can potentially serve as a success path.

The LPIS has four water sources: the borated water storage tank (BWST), the blowdown suppression tank (BST), the pressure reduction sump (PRS), and the blowdown hot leg (BHL). The system has two pumps (Pump-A and Pump-B) and three injection points for providing water to the reactor vessel. The three injection points are the downcomer, lower plenum, and cold leg. To provide coolant to the injection points, water can flow directly from each pump to the injection points on the same side of the system. These piping routes are known as the normal routes. Alternatively, flow can be directed from one side of the system to the other through two crosstie lines: the automatic crosstie and the manual crosstie. The LPIS also has two heat sinks: HX-A (heat exchanger A) and HX-B.

Because of the redundancy built into the LOFT LPIS, the system can be aligned to form numerous different success paths. These paths can then be displayed graphically using a response tree. Figure A3.2 is the response tree for the LOFT LPIS. Each path from the bottom of the response tree to the top represents a different success path. The various levels of the tree show which injection point, route, pump, water source, and heat sink is used by each success path. Because the success paths shown on the tree represent the major groupings of elements but not the details of individual components, each path on the tree actually represents three or four different success paths. The complete LPIS response tree used for the experiment contains 144 success paths.

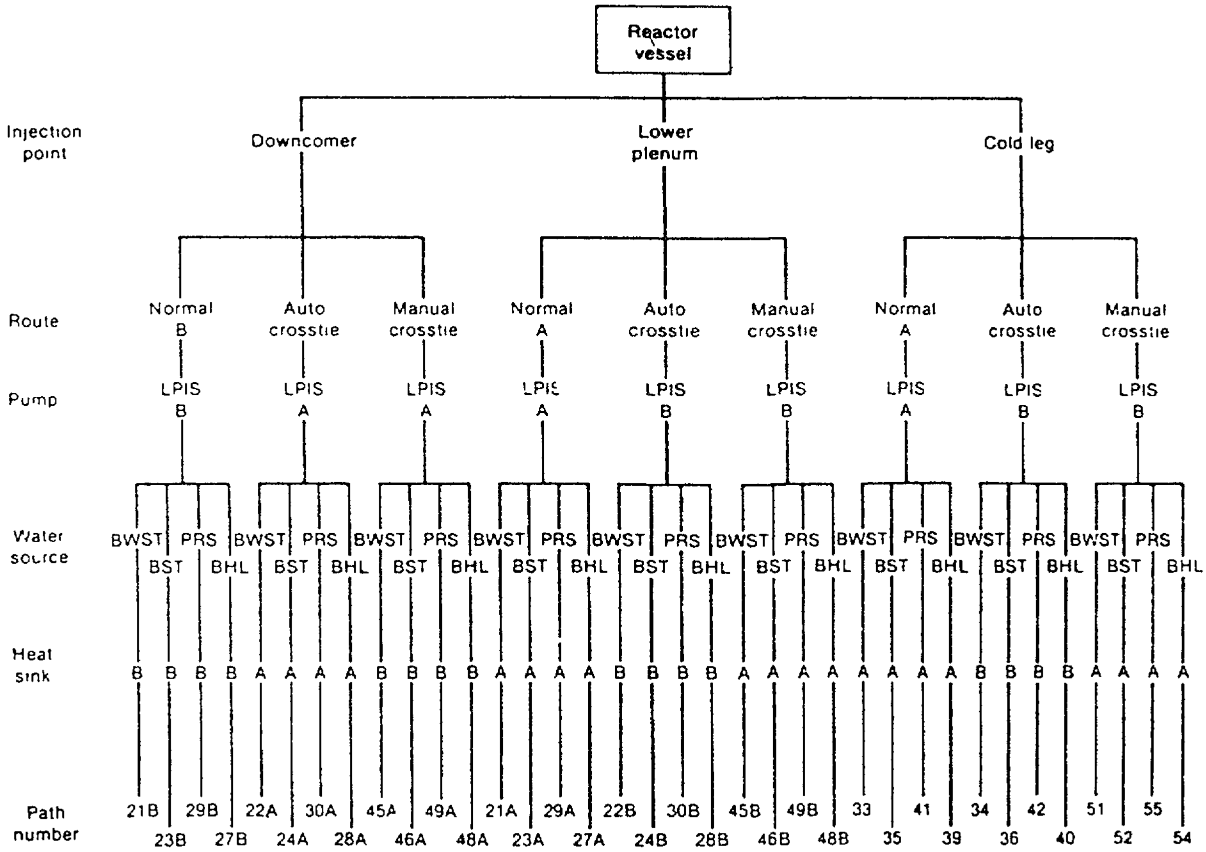

FIG.A3.2. LPIS response tree.

The success paths for the LOFT LPIS are not equally desirable. For example, the injection points are not equally effective for providing cooling water to the core. The water sources do not all have the same cleanliness controls. To position valves in the manual crosstie, someone must leave the control room and enter the containment basement, which requires a substantial period of time. Using considerations such as these, rules governing success path prioritization were formulated during discussions with LOFT operators. The rules are arranged in decreasing order of importance.

The path numbers shown at the bottom of the response tree reflect the application of the operating rules to the success paths of the LPIS. The success paths with the lowest numbers are the most desirable. Thus, when core cooling using the LPIS is needed, the available success path with the lowest path number should be implemented.

When failures disable equipment in the LPIS, the response tree can be used to determine which success path to implement. For failures in the BWST, Pump-A, and the downcomer injection point, no success path that utilizes one of these failed components is available for use. In this case, Path 24B has been selected, as shown by the darkened lines.

It is also possible to generate a customized procedure for implementing the success path. This is done by taking a general outline of a procedure for implementing a success path and adding appropriate component-level details for the specific path.

Appendix 4

SURVEY OF INIS DATABASE CONCERNING EXPERT SYSTEMS IN NUCLEAR POWER PLANTS

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As part of Working Group 2 effort to survey the known applications of expert systems (ES) in nuclear power plants, an interactive literature search was made using the International Nuclear Information System (INIS) data-base at the IAEA Headquarters in Vienna. The resulting print-out of 276 references, with abstracts, is given in the Bibliography Chapter to help newcomers to this field to learn about previous efforts by others. It is not claimed that the search is complete, as only the records which include the combination of (Reactors or Power Plants) and (Artificial Intelligence or Expert Systems or Knowledge Based Systems) in their names, abstracts or chosen key-words, were printed out. There may be important references which can be found in the INIS data-base with more careful search with related key-words. However, several interesing trends can be seen in the summary table.

In the following table, the lower percentage number indicates the distribution within the organizational classificaiton, while the upper percentage number is for the geographical classification.

Institutional / Geographical	Universities and National Research Institutes	Power Plant Vendors, Operators other Companies	Regulatory and Safety Organisations	Total
North America	62% 68%	55% 28%	55% 4%	142 59%
Western Europe	22% 63%	22% 30%	45% 7%	54 23%
Japan	3 % 23%	23% 77%		22 9%
Eastern Europe	13% 100%			21 9%
Total	157 (66%)	73 (30%)	9 (4%)	239 100%

North American references, predominated by the USA, represent 59% of the total number, and Western Europe 23%. Japan and the Eastern European Countries contribute 9% each. There is a significant difference in the organizational classification within each geographical block. While universities and national research institutes have roughly the same dominance (2/3 of the total), North America and Western Europe, and 100% in Eastern Europe, Japanese research is carried out mostly by the power plant vendors, utility users, and other private companies (77%), versus 23% in universities and national research institutes. The small number of regulatory and safety organization references points out one of the main problems in the eventual application of ES in nuclear power plants - the verification and validation of the ES by the regulatory authorities.

The organizations with a large number of references were in the USA: Oak-Ridge National Laboratory/Tennessee University (23), Idaho National Laboratory (18), Ohio State University (13) and Westinghouse Electrical Co. (10). The French CEA (11) stands out in Western Europe. Czechoslovakia (9) in Eastern Europe and Toshiba Co. (7) in Japan. The earliest reference found dates from 1976, but most of the publications date from 1982 onward. It seems that the ratio of 2:1 between research and industrial type references indicates the present status of the application of expert systems in nuclear power plants, which is still in the R&D stage.

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Annex

PAPERS PRESENTED AT THE MEETING

DEMANDS ON EXPERT SYSTEMS FROM THE REGULATORY SIDE: APPLICABILITY CONSTRAINTS

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Abstract

The need of licensing expert systems helping in the management of accidental scenarios is emphasized as an essential requirement for its practical application. This, in turn, will increase the confidence of the operating staff and the regulatory authority in these tools. The interacting role of models, heuristics and decision making under public radiation protection constraints is discussed in the context of pre and post core damage. Consideration is given to the methods for knowledge representation and built-in models with regard to their validity.

INTRODUCTION

The application of expert systems (ESs) is a fast growing area of research and development in many areas of technology and the nuclear field is no exception.

The use of ESs to assist the operator during an abnormal transient, post-trip analysis and emergency planing is justified because the human expertise is needed in a hostile enviroment and the sought solution has a high payoff. But it is not so clear if will be achievable, at least in the next few years, because each of those tasks require models and knowledge in many different fields. The scope of the knowledge involved seems to be too broad to be manageable by only one ES, and one of the rules of ES development ("the scope of the ES should be limited"[1]) may be broken.

The peculiarities of the activities related to radiological safety pose some interesting problems for the application of ESs, at least for the people from the regulatory side engaged in the auditing of such systems. These problems may be related to the degree of coincidence of the criteria implied in the body of the regulatory standards and the representation of this criteria in the knowledge base (KB) and its associated evaluative models, which may be essential for decision making.

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The points of view of the regulatory authorities (RA) and that of the utilities, which must be coincident with regard to public protection, may differ in practice because of the different weights that experts may assign to the various stages of the time evolution of an accidental sequence of events. This undesirable possibility defines the need for the independent assessment of ESs by the RA and its subsequent licensing.

The plant system status should be directly input into the ES via machine-computer interfaces. However, there are many reasons to keep the operator in the loop, among them his creativity and his capacity to handle unexpected events or sequences of an accident by using his own knowledge and common sense. Programs have had little success in doing this. It would be desirable that the operator would keep the control of the plant and should not be bypassed by the ES. This fact would be unacceptable for the operator and for the public. The only exception to this may be the case of safety systems that must be fired inmmediately after the initiating event.

In what follows we discuss the aspects of ESs implementation for radiological safety assistance in the context of regulatory demands.

CONSTRAINTS ON ESs FROM THE RA POINT OF VIEW

The ultimate goal of an ES aimed at providing help in the evaluation of an emergency must be the radiological protection of the public. All the stages of the analysis must imply this basic requirement. In the management of the accident sequence before core damage occurs, the goal is attaining reactor shutdown under safe conditions. This, in turn, means assuring the integrity of the containment system, keeping the primary damage as low as possible and limiting the release of effluents to a minimum. Some ESs [2-5] have been developed up to date which are ussually based on the procedures for handling abnormal transients (PHAT), on the trees of events originated in the probabilistic risk analysis of the installation, on the operating experience and on the integrated knowledge of the plant behaviour under incident (or accident) conditions. These sources of knowledge should provide enough information as to get a closed system for the inference of recommendations.

The introduction of a new tool in the operating room of a nuclear power plant brings along potential help to the operating staff and considerable "noise" in the usual management of an accidental sequence of events. This situation may arise because of the possibility of different interpretation of the system state by the operator and the ES. Two licensing constraint emerges here, namely:

[C01] The ES must reflect the PHAT and recommend the same actions.

[C02] All private knowledge embedded in the ES must be validated and licensed.

The first constraint wich may appear obvious, is not an easy task to acomplish and put under test the boundness and coherence of the procedures as well as their appropriate representation. This brings along an interesting (and ussually additional) requirement as is the complete compatibility of the data as stated in the design manuals and that of the plant.

It must be mentioned here that a complete plant implemented with redundant safety systems such as the Central Nuclear de Embalse (CNE) in Argentina has a great number of alarms of initiating events. Many of the indicating panels have common warning signals which are properly recognized by additional messages at the operator's console. The recommended actions imply verification of these signals and a procedure leading to a (hopefully) fast and safe reactor shutdown based on the symptoms (alarms) instead of the initiating event. The philosophy here seems to be: stop the reactor first and ask for the causes later. If the cause of the problem is well understood (or if it agrees with the initiating event as described in the PHAT) things will behave smoothly. However, a long transient may be initiated and different causes must be analyzed at the same time. Now the magnitude of the problem grows because parameter monitoring changes the scenario. A new constraint appears here:

[C03] The methodology of knowledge representation must adequate to changing scenarios.

C[04] The ES must be operating in real time.

A set of recommended actions for safe reactor shutdown implies the knowledge of plant behaviour under many different states (failed or not). While symbolic modelling can be used for much of the simulation, some aspects require mathematical modeling (e.g. the evolution of pressure and temperature)[6]. Then, procedural systems dominate the scene and the ES may loss its leading role as and advising tool. It is well known that the numerical modelling of such situations is subject to uncertainties originating in various causes, namely: physical models and correlations, nodalization sensitivity and loss of detailed multi-dimensional effects in plant components, among many others. A basic constraint that will put a warning in almost all recommendations inferred by a ES is presented here:

[C05] Only those systems free from or with determined uncertainties in their modelling may be considered as free of uncertainty in an inferred conclusion. It must be mandatory to consider the degree of confidence associated to the inference.

After core damage occurs, the conditions dramatically change, and the decisions concerning containment integrity, nuclide releases and emergency management loss any possibility of being unrelated. This is the case in which the RA must be more careful in the auditing of the ESs responses to a given scenario. The different time scales involved in the post-release stage also imply different strategies and tasks in the implementation of an ES.

There are many strong differences between a smooth shutdown (i.e. with an intact primary circuit) and an accident progressing toward core damage. This later imply the release of nuclides in the containment and its potential release into the enviroment under a variety of conditions.

The authors are not aware of ESs dealing with these situations. However, some considerations regarding licensing constrains may be established. We postulate that reliable warnings on core degradation come from activity monitoring. If the system status tends to one of predicted core disruption and if an ES is to be used as a helping system, then some basic constrains must be satisfied, namely:

[C06] The ES must predict core degradation.

[C07] The ES must be connected as to detect activity releases.

[C08] The ES must incorporate licensed safety criteria related to sudden energy releases, containment integrity and nuclide release into the enviroment.

It is our experience (based on simulacres) that differences in the confidence assiqned to the values implied in the implementation of the safety standards between the requlatory body and the utilities may appear here. Then a proper balance must be assured to deal with the risk-benefit of venting part of the inventory and containment inteqrity.

The most important problem in this case was not qiven consideration up to now. The source term is a difficult task for a predictive tool and with the present state of the art it would be preferable to rely on measurements. Otherwise a mathematical model, once aqain, will take the leading role, increasinq the uncertainty of the response as a whole.

Then we consider that two qeneral constraints must be introduced:

[C09] Decisions on facts with a larqe deqree of uncertainty must be dependent on system parameter monitorinq takinq into account, the uncertainties due to meassurement errors and instrumentation failures

[C10] After a decision has been adopted, relevant parameters must be feed back to the ES.

Durinq a nuclear accident it is very difficult to predict the actual amounts of radioactive material to be released to the enviroment and its pathway. The estimation of the radioactive material would be based on the existinq radiation field and, if it is possible, on the concentration of the different nuclides, on the evaluation of the mitiqation actions, on the estimated release of nuclides to the different compartments of the plant and finally on the inteqrity of the plant. All these evaluations must be done in real time and recommend the operator whether or not it would be necessary to beqin the countermeasures in the outside of the plant. So it is necessary to do a dosimetric evaluation with a dispersion model and a meteoroloqical forecastinq. The countermeasures in the early staqe avoid the non-stochastic effects and reduce the risk of stochastic effects. Countermeasures should achieve a positive net benefit to the individuals involved. The countermeasures to be analized are shelterinq, stable iodine administration, evacuation, relocation due to nuclide deposition and, later, the reentry. The order of the above countermeasures is related to the order in which they would be considered in an emerqency. In a lapse in the order of a day it will be necessary to beqin the analysis of the countermeasures on foodstuffs and its possible cost and extention. This must be done using transference models taking into account a deposition model or field measurements. It will be necessary to analyse all the possible ways to intervene the food chain using an economical model and a model for radiological consequences.

It is difficult to establish new constraints here, except those concerning the recommended practices. The radiological safety cost-benefit analysis of this type of countermeasures is nowadays an active field of research and we feel that the general constraints given above cover most of the problems. Obviously economical decisions are closely correlated with the radiological safety of the public and new criteria is now emerging.

Finally It would be appropriate to comment some characteristics of the knowledge representation techniques and ESs implementation languages and their relative merits in this type of applications.

The most popular techniques to represent knowledge are rules, semantic nets and frames. The knowledge representation in the rule based methods is centred on the use of clauses of the form:

IF <condition> THEN <action>

Rules provide a natural way for describing processes driven by a complex and rapidly changing enviroment like a nuclear power plant under an abnormal transient. The rules specify how the program ought to react to the changing data without a very specific flow of control.

Different types of languages may be used to construct an ES:

- Problem-oriented languages such as FORTRAN or PASCAL
- Symbol manipulation languages as LISP or PROLOG
- Knowledge Engineering languages.

Problem oriented languages are mainly oriented to scientific computation and symbolic manipulation is very hard. Symbolic manipulation languages are designed for artificial inteligence. They have embedded the manipulation of list structures, recurrence, backtraking, sophisticated editing and debugging, but they fail when a model is inserted in the ES to predict facts. In this case the inference engine can be designed to fit the specific heuristics of the solution. A knowledge engineering language is a tool for developing ESs which has the inference engine and the support facilities. They offer little flexibility because the knowledge engineer must use the control scheme provided by the inference engine. On the other hand they provide knowledge representation guidelines and a method to access the KB. In the authors' experience, different strategies are convenient for different stages of the ESs development and implementation, namely: to begin with a knowledge engineering tool for a purely interencing prototype because the major efforts are oriented to the knowledge adquisition from the experts and formalization into a knowledge base. Later, when the problem's heuristics is well understood and the KB is large, it is preferable to use a low level and efficient language to construct a problem-oriented inference engine and KB. Finally in the case in which procedural methods become essential, problem oriented languages may be considered and recent literature[7] show the tendency to this approach as reasonable.

It is difficult to impose a constraint on this facts, but system maintenance must be accessible, at least for data analysis and checking. The RA should put a constraint on this aspect and we define:

[C11] Plant data built-in the ES must be accessible to the RA and the sensitivity of the ES responses to its change must be validated.

CONCLUSIONS

The basic characteristics of ESs designed for helping in accident diagnosis in nuclear installations have been discussed in the context of a RA requirements. A set of constraints, open to additions, to be satisfied by such systems has been proposed as to comply with the auditing process. The need for licensed ESs has been defined as mandatory as well as the availability of plant data built-in in the ES.

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A 'LIVING' PSA BASED ON USE OF EXPERT SYSTEMS

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Abstract

INTRODUCTION

The French Probabilistic Safety Analysis (PSA) is performed on unit 3 of the PALUEL Nuclear Power Plant which belongs to the 1300 MW-P4 series. The concept of a "living" PSA, i.e. a study that can be updated to allow for changes in data and knowledge, led EDF to make an important effort on codes. In the framework of the automation and computerization of reliability studies, expert systems are used.

PRESENTATION OF THE METHOD DEVISED FOR STATIC THERMOHYDRAULIC SYSTEMS

The method implemented by the EXPRESS software revolves around two mains phases performed by the second order GENESIA I engine.

The first phase is aimed at grouping the components of a system into larger components categories - or macrocomponents - according to the consequence of their failures (blockage or leakage).
The second phase deduces the failure consequences, for each component, in terms of path losses, according to the macrocomponent (s) to which the said component belongs. Moreover, the possible causes of these failures are stated according to the configuration and the mission studied.
These two phases lead to the generation of a simple rule base of order 0 (GENESIA I-AUTO) written in GENESIA I language.

If a hand-written rule base (GENESIA I-MANU) concerning the undesirable event and the boundary conditions mainly is added to (GENESIA I-AUTO), all the data required to build a fault tree are then available.

The method described above has been successfully used and validated on the Auxiliary Feedwater System (AFS), the Residual Heat Removal System (RHRS) system, and the Safety Injection System of the PALUEL Nuclear Power Plant.

OBJECTIVES AND PROSPECTS

The future developments will consist in the integration of these expert systems in the LESSEPS software that is developed for the complete computerization of the French PSA.

The aim of the LESSEPS software is to allow the organization of all data managed (reliability data, reliability models as fault trees, state graphs and event trees) and the updating of the PSA when reliability data are changed.

When the above described expert systems will be integrated in this software, the objective is then to use the computerized PSA as a tool for design conception (by taking into account a modification of the system topology) and for operator aid (by evaluating the Allowable Operating Times, for example).

Reflections are also carried out at EDF for the unification of knowledge representations used in different types of applications (reliability assessment, diagnosis...).

INTRODUCTION

The French PSA is performed on unit 3 of the Paluel nuclear power plant which belongs to the 1300 MW-P4 series. The concept of a "living" PSA, i.e. a study that can be updated to allow for changes in data and knowledge, led EDF to make an important effort on codes. In the framework of the automation and computerization of reliability studies, expert systems are used.

The aim of this presentation is to describe one of them, EXPRESS, which helps the analyst to build fault trees, and to show how the concepts used can be extended to the generation of a state graph for sequential systems.

1. KNOWLEDGE REPRESENTATION

The EXPRESS software uses two types of representation : one in connection with the GENESIA II second order inference engine and the other with the knowledge language, i.e. GENESIA I . These two tools have been developed by the Direction des Etudes et Recherches at EDF.

1.1 GENESIA II

GENESIA II is an inference engine in forward chaining with a knowledge representation based on production rules <if....then....> in second order logic and facts written as triplets <object, relation, object>.

1.2 GENESIA I

GENESIA I is a language used to represent knowledge in - order 0 - propositional logic. In the EXPRESS software, only the associated demonstrator is used to build a fault tree by means of backward chaining, starting from an undesirable event.

2. PRESENTATION OF THE METHOD DEVISED FOR STATIC THERMOHYDRAULIC SYSTEMS

2.1 General Principles

The method implemented by the EXPRESS software is issued from a basic observation :

> according to their consequences on the system, failures relating to the components of thermohydraulic system can always be grouped under a few

large categories. The main categories are :

- fluid flow interruption called "blockage"
- loss of fluid outwards also called "external leak"

Based on the above observations, the EXPRESS software revolves around two main phases (figure 1) performed by the second order GENESIA II engine.

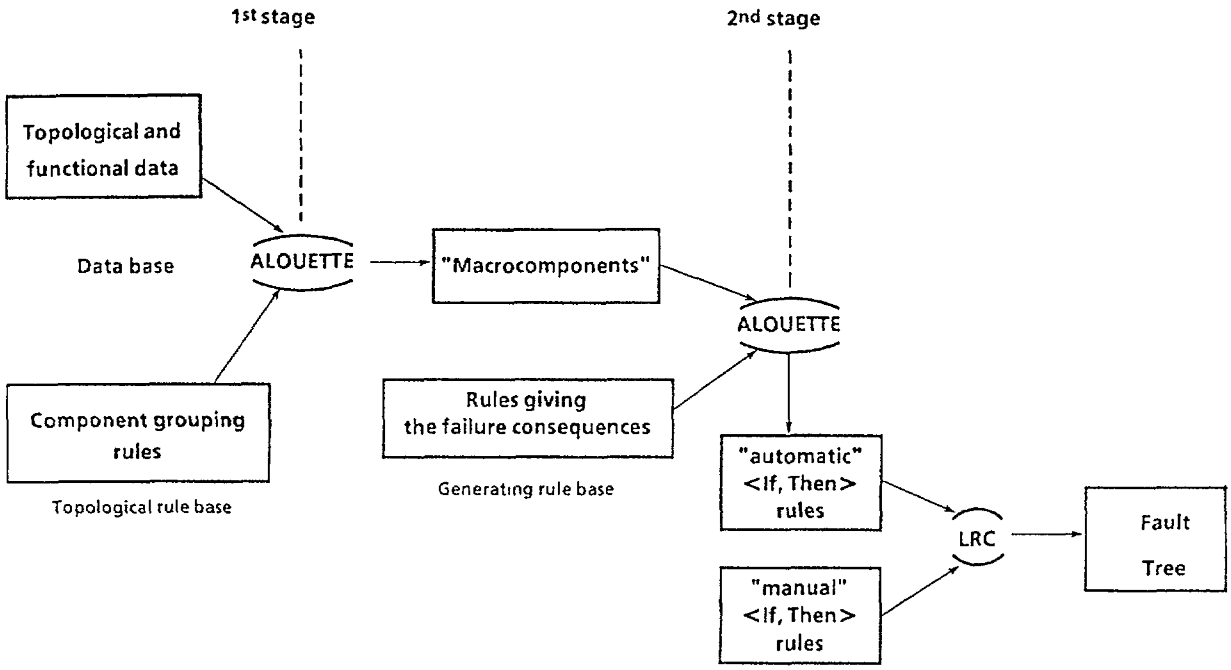

FIGURE 1.

The first phase is aimed at grouping the components of a system into larger components categories - or macrocomponents - according to the consequences of their failures (blockage or leakage).

The second phase deduces the failure consequences, for each component,in terms of path losses, according to the macrocomponent(s) to which the said component belongs. Moreover, the possible causes of these failures are stated according to the configuration and the mission studied.

These two phases lead to the generation of a simple rule base of order 0 (GENESIA I-AUTO) written in language.

If a hand-written rule base (GENESIA I-MANU) concerning the undesirable event and the boundary conditions mainly is added to (GENESIA I-AUTO), all the data required to build a fault-tree are then available.

The fault tree is then obtained by applying a demonstrator to the entire rule base. This demonstrator presents the results in a form which can be handled by a conventional code computing minimal cut-sets.

2.2 The Initial Facts Base

All the data required for the first phase are contained in the initial facts base.
The base includes two types of facts :

. topological facts which describe the system topology, i.e. :

. The sources,
. The goals,
. The sequential relations between the components.

COMMENT

Topological facts are obtained simply by reading a system diagram. They do not, therefore, depend on the mission or the configuration studied.

. functional facts which :

- on the one hand, specify the failure consequences (blockage, leakage) and especially indicate :

  . possible paths between a source and a goal,
  . the components having an isolation function
    * real (normally-closed valve during the system mission)
    * potential (valve that can be closed during the system operation)
    * upon a pressure drop (presence of membranes...),

- on the other hand, state the failure causes (blockage, leakage) and indicate, for instance :

  . the component nature
  . their initial position
  . their position during operation
  . whether their position is displayed in the control room..
  ...

COMMENT

Most functional facts depend on the studied mission or configuration. Moreover, the way they are formulated often depends on the functional assumptions adopted (Can a given component be considered as an isolating device ?Is there enough time to send someone on the spot ?...)

2.3 Topological Rule Base

The topological rule base builds all the concepts needed to express an undesirable event in terms of path losses, i.e. :

- The arcs connecting a source to a node, a node to another...
- The paths running from a source to a goal,
- The "leakage" blocks which are built, starting from a node, by incoporating the components, one after the other, up to the first

isolating devices met whatever actually the nature of the isolation -whether one-sided in the case of check valves,or two-sided in that of valves,

- The "blockage" blocks incorporating all the components whose blockage has the same effect on the system.

By applying the topological rule-base to the initial facts base, all the above-described entities can be constructed and added to this facts base. The knowledge on the system is then contained in a base called "intermediate facts base" which will be used in the second phase to generate order 0 rules (GENESIA I-AUTO)

2.4 Generating Rule Base

It includes two types of rules :

topological rules which use the results obtained during the first phase and express the component failure consequences in terms of "paths losses", according to the "leakage" and or "blockage" block(s) to which they belong.

functional rules which :

. on the one hand, state the causes of the blockage or leakage ; thus, for an initially shut down pump, the failure modes - maintenance outage, primary failure - are automatically generated.

. on the other hand, contribute to the automatic processing of certain specific problems :

- either to facilitate the work : when a given problem must be repeated for several components, it is sometimes convenient to have an automated system for taking the problem into account. Then as a generatic rule is available, manual drafting of numerous rules can be avoided.

- or because those who study the systems think that they are likely to meet the same specific problems when studying other systems.

An Example of Generating Rule

. topological rule

```
Blockage rule :
   IF          nature (BB) = block-blockage
               nature (P)  = path
               item   (P)  = (BB)
               nature (X)  = component
               item   (BB) = (X)
  KNOWING THAT name (C)    = PATH1
  THEN           write       (in  Syntax)
              "IF            X - BLOCKAGE = YES
              "THEN       PATH1 - LOSS     = YES
```

. <u>functional rule</u>

```
Pump maintenance rule :
   IF        nature (PV) = pump
   THEN      write
             "IF          PV - MAINTENANCE = YES
             "THEN        PV - BLOCKAGE    = YES
```

2.5 <u>The GENESIA I-AUTO Base</u>

By applying the generating rules to the intermediate facts base developed in the first phase, a set of very simple rules written in GENESIA I language is obtained. These rules give the lost paths.

An example of GENESIA I rule is given below :

```
RULE NUMBER 11
IF    TANK - LEAKAGE - EXTERNAL = YES
THEN  PATH 1 - LOSS = YES
```

2.6 <u>The GENESIA I-MANU Base</u>

Like the initial facts base, the GENESIA I-MANU base is entered manually by the user and is concatenated to the GENESIA I-AUTO base.

It essentially contains :
- the undesirable event,
- the boundary conditions (electric power supplies, cooling systems, controls..)
- certain specific problems.

3. <u>THE MERITS OF THIS METHOD</u>

The method described above has two obvious advantages :

- short execution time for most of the reliability study. The reliability specialist can then devote much more time to tricky problems in his system analysis.

- greater exhaustivity as most of the failures and failure combinations are automatically generated ; indeed, the method combines :

  . the exhaustivity of an inductive method (FMEA or Failure Combination Method for instance) for the generation of the consequences.
  . convenience and evocative power of a deductive approach (such as fault trees) for the processing of automatically generated rules.

The following two principles epitomize the whole method foundation :

1. to compel the specialist who devised the method to clearly state the concepts he wants to automate and to produce the controls which the user needs to check the validity of his modelling constantly.

2. to compel the method user to formalize and structure his knowledge.

Beyond the good performance of the inference engine used, the implemented method will be effective if these two principles, which impart all its merits to the method as well as all its complexity, are complied with.

But, then the proposed method :

- becomes a true communication tool for system designers and operators ;
- confers the tool an inherent understandability ;
- ensures the consistency of the reliability studies as regards both the quality and presentation as well as the adopted "detail" level ;
- facilitates the application of these studies to other problems (identification of common cause failures affecting several systems...).

4. THIS METHOD WITHIN THE GENERAL RELIABILITY ANALYSIS APPROACH

Up to now, we have merely described the method leading to the fault tree construction.

Note, however, that whatever the reliability model set as a goal (fault tree, state graph...), the basic principles underlying the method presented here remain perfectly valid. Differences can only be found at the implementation level (concepts to be automated, algorithmic processing of the generated reliability model...).

Whether the final model is of the fault tree type or of the state graph type, the main steps of the technique are the following (figure 2) :

- **knowledge representation and reliability concept generation** ; as a matter of fact :

  . some of the generated concepts are common to both methods : the "outleakage" block and "blockage" block concepts can be used for both static and sequential systems.

  On the contrary, some concepts required to construct failure sequences (using the GSI tool, also developed by EDF) are entirely different from those that have to be used to determine minimal cut sets ; hence, to each approach corresponds a specific generating ruleset, producing, on the one hand, the set of GENESIA I rules required for the construction of the fault tree and, on the other, the GSI model used to identify the failure sequences.

  . the knowledge that must be formalized is, for the most part, common to both models ; in fact, the GSI model only requires additional information (initial component state, reliability data..).

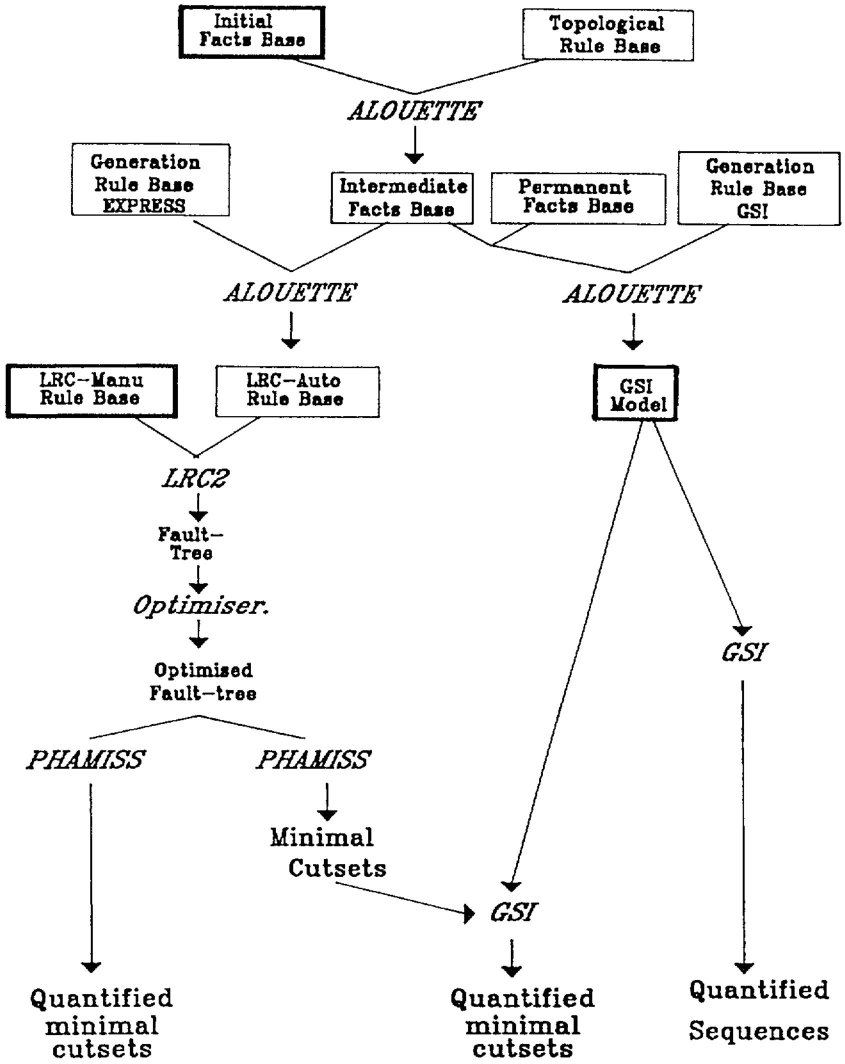

FIGURE 2.

- processing of the reliability model resulting in :

  . on the one hand, the quantification of minimal cut sets following the boolean reduction of the fault tree ;

  . on the other hand, the quantification of the sequences leading to the system failure.

5. PROSPECTS

The following objectives are :

- a methodological one, consisting in the clarification and homogenization of the concepts used for generating the different types of reliability models,

- an operational one, consisting in the integration of these expert systems in the LESSEPS software that is developed for the complete computerization of the French PSA (figure 3).

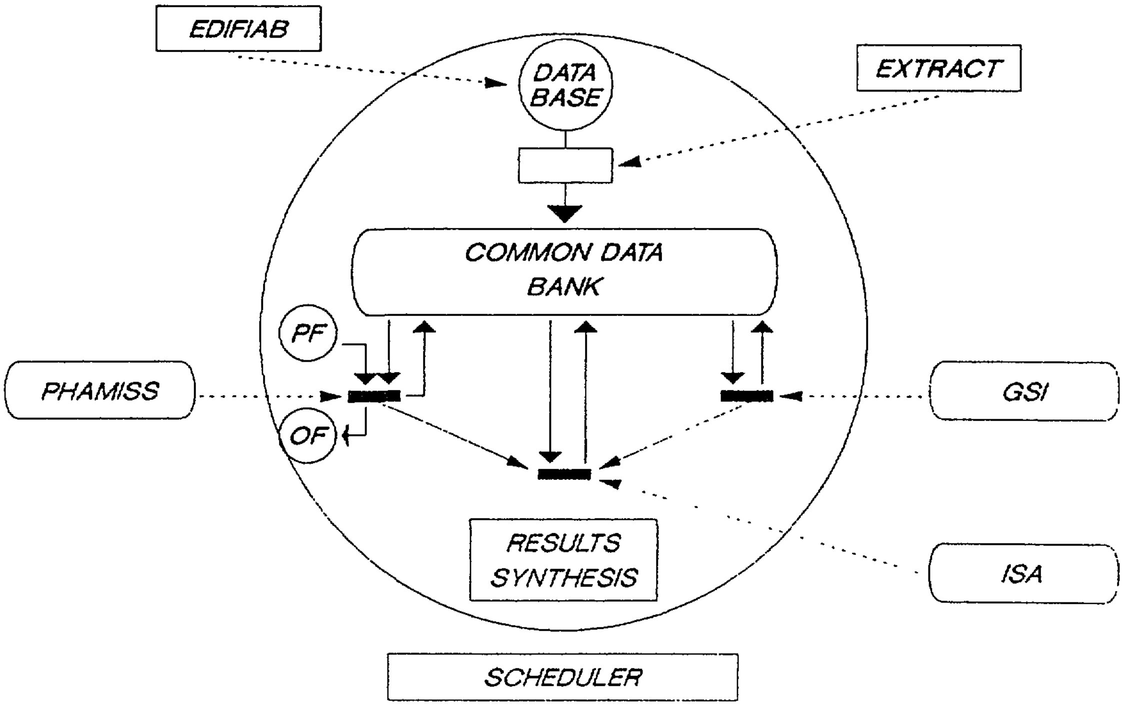

FIGURE 3.

The aim of the LESSEPS software is to allow the organization of all data managed (reliability data, reliability models as fault trees, state graphs and event trees) and the updating of the PSA when reliability data are changed.

When the above described expert systems will be integrated in this software, the objective is then to use the computerized PSA as a tool for design conception (by taking into account a modification of the system topology) and for operator aid (by evaluating the Allowable Operating Times, for example).

Reflections are also carried out at EDF for the unification of knowledge representations used in different types of applications (reliability assessment, diagnosis...).

EXTRA: A REAL TIME KNOWLEDGE-BASED MONITORING SYSTEM FOR A NUCLEAR POWER PLANT

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Abstract

EXTRA is an expert system for industrial process control. The main objectives are diagnosis and operation aids. From a methodological point of view, EXTRA is based on a deep knowledge of the plant operation and topology and on qualitative physics principles. This system represents a considerable step forward in the field of expert-systems because of the size of the knowledge - base and the real-time requirements.

A specific application of EXTRA is developped for the BUGEY unit 2 (a 900 MWe pressurized water nuclear unit) concerning the electrical power supplies. This system called "electrical power supplies supervision" gives diagnosis in real time of the electric incidental situation origin. A connection to a data base make the expert system able to supply operators with informations concerning the consequences of the electrical power system failures on the safety systems, the equipment measurement sensors and certain automatic devices (availability, unavailibity, validity, etc...).

A simulation part of the system, out of real time, can help the operators or the maintenance team to prepare the withdrawal from service of electric equipments by giving informations on the consequences of it, in particular, informations concerning the technical specifications regard. The system will be independently used and managed by the operating crews and maintenance team, but a priority is given to the diagnosis real time supervision.

This expert-system will be installed for the beginning of 1989.

---

1. INTRODUCTION

EXTRA is a real-time expert system for industrial process control. It is being developed jointly by the Study and Research Division and the Fossil and Nuclear Generation Division of Electricité de France.

In 1986, the merits of expert systems in this field were demonstrated with a large prototype coupled to a full scale operator training simulator [1].
It was therefore decided to continue the development effort in an industrial environment and, to this end, to install a system in one of the PWR units of BUGEY nuclear power plant. The system developped for the BUGEY unit 2 is specially oriented on electrical power supplies failures and their consequences on the equipments.

2. AIMS

This monitoring system is a computerized aid designed to help the operator to process all the electrically-induced failures occurring in the facility.

It is used for the following purposes :

- permanent and real-time processing of events caused by electric power system failures,

- preparation for the withdrawal from service of electric equipment,

- operation training in independent electric power supplies and shared power systems.

3. METHODOLOGY

This application uses artificial intelligence techniques to a large extent and, in particular, expert systems [2].

The system is based on a two-level structure reflecting the separation between the design aspect and the on-line application one.

3.1. Design

At the center of the system there is a single knowledge-base, that is a set of data providing a detailed description of the equipment and operation of the plant electric power supplies.

To use expert system terminology, this set of data consists of :

- a fact base including :

  . topological data describing the component nature and their connections,

  . data describing normal equipment configurations according to the state of other components or of the overall plant conditions,

  . functional data describing the behavior of components in case of proper operation or malfunctions.

- the following four rulesets respectively including :

  . data validation and consistency principles combining syntactic control with higher level control and based on the rules governing the design of electric power systems,

  . the principles to identify the plant state on the basis of the information provided by the unit computer system,

  . the simulation principles,

  . the diagnosis principles.

It should be noted that the facts base describing a given facility, as defined above, is not restricted to a particular application but covers a whole range of applications of the operation-aid type (such as those described in this paper) and of the computer-aided reliability study type... Similary, the various rulesets are applicable to all the facilities described with facts bases having the same structure.

The expert system generator chosen at this level is the GENESIA 2 [3]. It consists of an inference engine based on predicate calculus and using forward chaining.

The various processing modules actually performing the monitoring system functions are automatically generated from this knowledge base. Thanks to this approach, the homogeneity and consistency of each module in the system is guaranteed.

Moreover, with the architecture, the system is readily adaptable when the plant is modified or when functions are extended.

3.2. On-line application

Each of the monitoring systems modules is a specialized expert system performing one of the required functions.

- real-time identification of the state of the plant electric power supplies,

- real-time diagnosis and choice of the procedure with, upon demand, the reasoning which led to the diagnosis,

- simulation of the electric power supplies behavior based on an actual plant state (prediction) or a reference state (training).

In fact, these modules are rulesets pertaining to a given application and plant.

The facts used by these rules are on/off or analog data automatically acquired on the unit computer and data processing system.

The expert system tool chosen at this level is tailormade from the GENESIA 1 inference engine [4]. For performance's sake, this engine is written in C language. It is based on propositional calculus and uses forward chaining.

This two -level structure combines the short execution time of the GENESIA 1 for on-line processing and the GENESIA 2 capability for the design of large knowledge bases.

This methodology is summarized in the picture 1.

4. SCOPE

The monitoring system monitors all the electric power supplies of a 900 MW PWR nuclear power unit excluding the thermal-hydraulic phenomena. To be more precise, the field of the application encompasses :

- all the components, together with their controls, supplying electric power to the unit and shared systems,

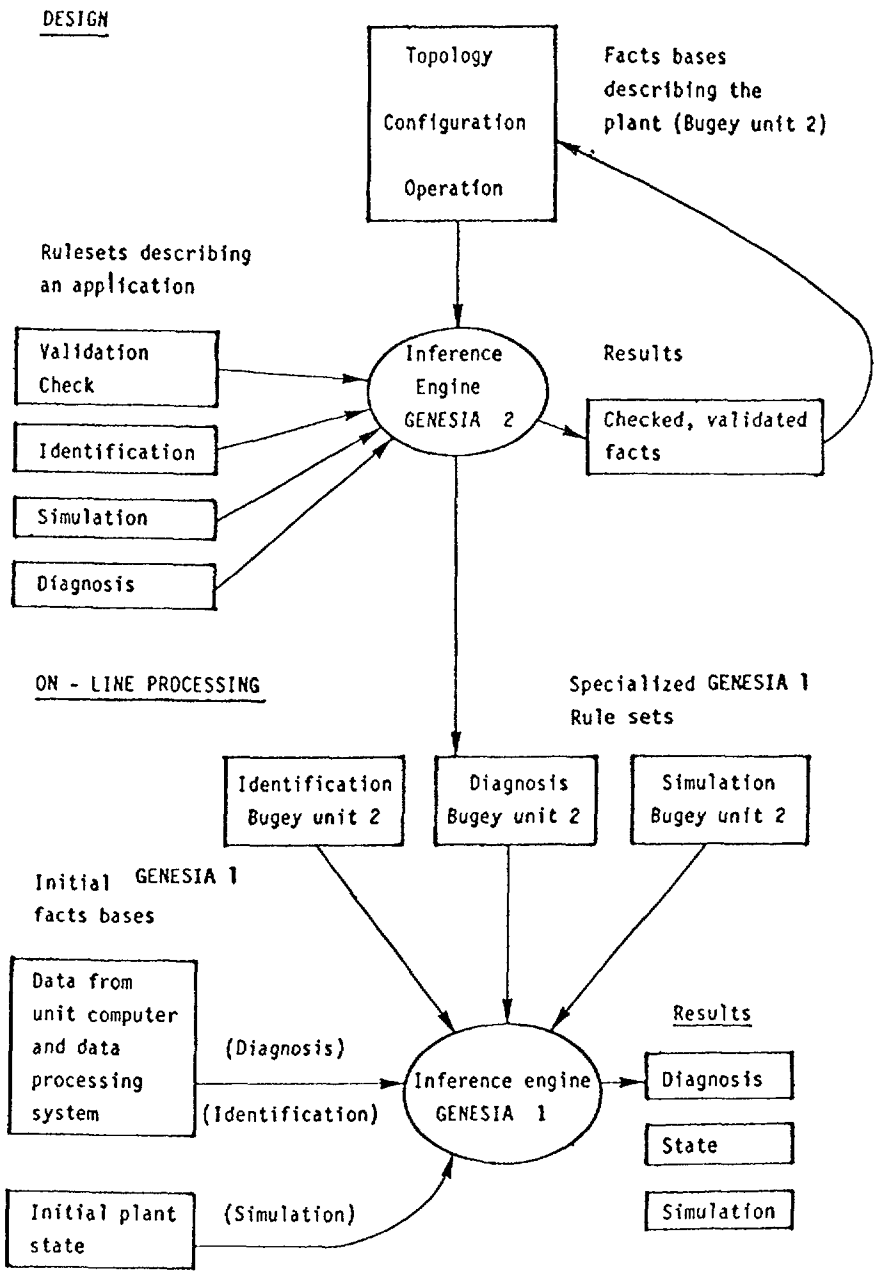
- the actuators of the pumps and valves controlled from the control-room, together with their controls (250),

- the process instrumentation channels,

- the alarms (2000 indicating lights) in the control rooms as well as the unit computer system variables (5024 on/off values and 1550 analog values),

- the reactor protection channels,

- the control room recorders and indicators (250).



Picture 1

5. FUNCTIONS DESCRIPTION

The application centers around the following four main functions :

- plant status identification,
- diagnosis,
- simulation,
- out of real time consultation.

5.1. Real-Time Identification of the Plant State

This function is aimed at producing an as detailed as possible reconstruction of the state of the plant electric power supplies only using the data provided by the unit computer. This same reconstruction process is used for the diagnosis and therefore takes place on a permanent basis. The operator merely consults the results.

Through this process, the state and availability of each component involved in the application is identified.

This function is performed by a specialized expert system. In a first phase, this system translates the values of the data acquired by the unit computer into component states. In a second phase, using its knowledge of the plant topology and operation, the system simultaneously completes step by step and validates the partial plant state obtained during the first phase.

This approach has several advantages. Among others, it has the merit of :

- separating the description of the information sent by the instrumentation from the description of the topology and operation of the facility itself, this faciliting the system updatings,

- spotting inconsistencies in the data on the unit computer, generally revealing instrumentation failures [5] and thus providing a valuable aid for processing this type of failure.

5.2. Diagnosis

The diagnosis is provided to inform the operator of any abnormal event occuring in the facility. The diagnosis is made by a special expert system which interprets the result of the unit state identification function. This expert system is based on cause-consequence relationships between components and on the concept of normal conditions.

In practice, the processing can be divided in two phases :

- from the recorded initiating events, the system draws up a list of potential failures,

- based on this list and on the identified state of the facility, the system selects the only potential failures which are relevant considering the unit state and which do not result (even indirectly) from another identified failure.

The diagnosis consists of :

- synthetic information describing the overall state of the plant : the name of a procedure, for instance,

- relevant equipment failures pointing out to the operator the areas on which to concentrate this attention to restore more normal operating conditions.

  This processing is continuous so that the maximum time between the occurence of the first alarm and the diagnosis by the system does not exceed 15 seconds.

  Note that in the chosen approach, there is no restriction on the number of failures which can occur simultaneously.

5.3. Simulation

The behavior of the plant power supplies can be simulated starting from :

- a reference state,

- a state directly stored from the informa-provided by the unit computer system,

- a state previously derived by simulation.

The operator modifies this initial state as he deems necessary and then starts the simulation. As a result, a so-called final state is derived which corresponds to the equilibrium state reached by the plant at the end of the transient.

From a practical point of view, this simulation is run by an expert system based on the knowledge of the facility topology and operation. The overall simulation is achieved by combining the component local behaviors.

5.4. Ex-post consultations

The user can have access to three types of consultations :

- consultation of the state of equipment :

Using this function the user secures knowledge of the state of equipment memorized either during real time diagnosis or in the course of a simulation. The information listed below can be obtained :

. on the one hand, information about state;
. position of actuators (open, closed, etc...),
. state of valves, pumps, etc...(open, closed, in service, decommissioned, etc...),
. voltage (on, off),
. validity of danger warnings, protections, indications, etc.

- and on the other, information relating to the electrical availability or outage of equipment, especially for redundant equipment or apparatuses covered by rules defined by the Technical Operating Specifications (safety equipment).

- Justification of the diagnosis :

The user can ask for proof of the diagnosis of an incident. Going back in time, the expert system, stage by stage, then produces the string of rules that have led from the information collected on the unit to the diagnosis presented.

- Consultation of a database relating to equipment :

The user can secure technical and functional information on all the components included in the application through a database originating from that used for the development of the system.

6. OPERATOR INTERFACE

The monitoring system will be moderatly used under normal conditions by very different users with no particular training in the computer technique.

This fact as well as the system location in the control room, among numerous other data and operating means, has led us to choose a straight forward man-machine interface largely resorting to the concept of window and to mice.

As for as the hardware is concerned, this interface consists of two totally independent and interchangeable workstations. One is located in the control room and the other in a separate room used by the power plant maintenance personnel. A station is made up of :

- a color graphics display unit,
- a mouse whith a click button,
- an alphanumeric keyboard.

To dialog, the main device available is the mouse with its button. The use of the keybord has been on purpose reduced to a minimum.

The image seen by the operator on the screen at a given instant is a combination of a number of basic graphic objects. There are four basic graphic objects :

- the symbol, which is a simple and compact means of access and progression in the dialog,

- the dynamic process diagram, which is a graphical animated representation of a part of the electric power supplies,

- the list, which contains a set of components and data on the plant state that can be consulted, printed or pointed at,

- the data acquisition grid, which is a special list used to feed alphanumeric data into the computer via the keyboard.

7. HARDWARE CONFIGURATION

As regards the hardware, two BULL SPS7 computers are used for the application :

- a so-called application computer connected to the unit computer and data processing system through an ARLIC network and which performs all the monitoring system functions.

- a so-called design computer supporting the knowledge-base describing the plant.

  It is used in particular :

  . to update the knowledge-base whenever the plant is modified,

  . from this knowledge-base, to automatically generate specialized expert systems pertaining to the various monitoring system functions.

This architecture has the merit of minimizing the monotoring system unavailability during plant modifications as well as the costs if the system is to be adopted in the 4 Bugey PWR units. Indeed, in this case, there will be 4 application computers and only one design computer which will be devoted to the management of the knowledge-bases of all 4 units.

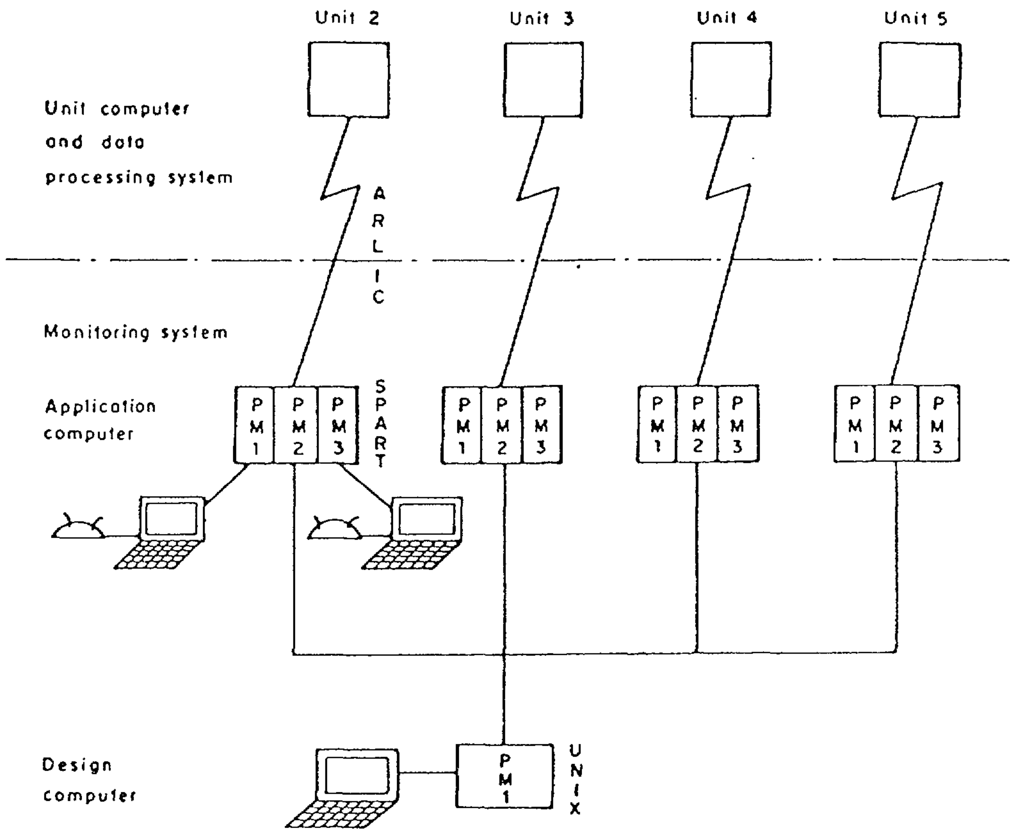
The application computer incorporates, among others :

- three processing modules using 32-bit microprocessors running in parallel under the SPART Operating System,

- two color graphics workstations with a mouse each.

The design computer is made of a single 32 bit microprocessor module and runs under UNIX

The system architecture is therefore as shown on picture 2.



Picture 2

8. CONCLUSION

Now that the design phase is over, the actual implementation has just started, requering some hundred engineers month. The target sought is to have an operational system in one of the Bugey power plant units by the beginning of 1989. Note that this system represents a considerable step forward in the field of artificial intelligence because of :

- the size of the knowledge-base,

- the application is operational. It will be independently used and managed (when changes will be introduced in the plant) by the operating crews,

- the real-time requirement.

Finally, this project is not restricted to an application. It should also make it possible to demonstrate the approach technical maturity and also to train a team for subsequent larger scale applications.

9. PROSPECTS

The monitoring system of electric sources presented above should prove itself in its planned industrial environment in 1989. Should it be satisfactory, it will be extended as such to the other three units of the BUGEY power station.

In the light of the teachings drawn from the implementation of the principles developed in EXTRA for other applications, a certain number of studies are presently under way to broaden the field of application of the system, in particular relating to an extension towards more thorough supervision of safety systems from both standpoints of the consequences of electrical faults and identifiable mechanical or hydraulic faults. In the longer run, an extension can be envisaged for operators assistance by automatic output of control procedures for incident and accident scenarios.

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THE TEX-I REAL-TIME EXPERT SYSTEM APPLIED TO SITUATION ASSESSMENT FOR THE SNR-300 REACTOR*

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Abstract

Within the German TEX-I Project, which was sponsored by the Federal Ministry for Research and Technology, several companies developed industrial applications of technical expert systems for data interpretation, diagnosis and process control. The purpose of the diagnosis expert system reported here is to support the operators of the LMFBR SNR-300 in assessing plant status in real-time, based on readings from a large number of sensors. By online connection to the process control computer, it can monitor all incoming signal values, check the consistency of data, continuously diagnose the current plant status, detect unusual trends prior accidents, localize faulty components and recommend operators response in abnormal conditions. The systems architecture consists of two basic subsystems, an inference engine and an intelligent process interface, implemented in Lisp on a Symbolics- Workstation. The inference engine has been derived from BABYLON, a hybride shell developed by the German computer research institute GMD. This shell includes rules, prolog, constraints and an object oriented frame processor. The extended version has a component description language and a top-down diganosis scheme including a mechanism of attention focussing. Inference run as endependent, quasi-parallel processes. These so-called inference tasks can interrupt or abort each other, if higher priority events must be processed. The complete system is modelled in an object oriented matter and is divided into several subsystems and each subsystem into its physical components. At present the expert system is connected to a real-time simulation of the reactor. The simulation is based on a thermohydraulic code for simulation of the transient behaviour of temperatures and flow rates in the reactor core, plena, pipes, pumps, valves, intermediate heat exchangers and cooling components. Additionally, the systems response to an asynchronous operator interaction can be simulated. Several types of anomalies have been modelled up to now and can be diagnosed by the expert system.

1. Introduction

Interatom, a subsidiary company of Siemens, develops expert systems for the technical domain. These systems operate in various industrial applications such as flexible manufacturing or plant configuration, based on a domain specific expert system shell, designed by Interatom. Additional projects focus on real-time diagnostics, e.g. for nuclear power plants.

* Supported by the Federal Ministry for Research and Technology of the Federal Republic of Germany under contract ITW 8503.

In this paper we report on the architecture of a real-time expert system tool, which has been developed within the German national venture TEX-I, and its application to diagnostics and process monitoring for the fast breeder reactor SNR-300, built by Interatom GmbH at Kalkar.

2. The TEX-I project

TEX-I is one of the eight national German projects for research of knowledge based systems, which are sponsored in part by the German Federal Ministry for Research and Technology (BMFT). All these projects are based on close collaboration between industrial companies and research institutes and cover a broad range of artificial intelligence applications. One of the technical oriented projects is TEX-I, an acronym for "Technical EXpert systems for data Interpretation, diagnosis and process control".

The aim of the TEX-I project is to design an expert system tool, which can be used for error diagnosis and process control in a wide range of technical applications. The project started in 1985 and will be finished by the end of 1988. Participants are the two institutes Gesellschaft für Mathematik und Datenverarbeitung in Bonn (GMD), the Fraunhofer Institute in Karlsruhe (IITB) and six companies.

The industrial partners and their applications are:

- Bayer (Supervisory control of a sewage clarification plant)
- ESG (Fault-diagnosis of electronic components)
- Interatom (Situation assessment in a fast breeder reactor)
- Krupp Atlas Elektronik (Diagnosis of electric networks)
- Siemens Erlangen (Configuration of computer networks)
- Siemens Karlsruhe (Alarm management in process control systems).

At the beginning of the project it was decided to implement the TEX-I system in COMMON-LISP on a Symbolics Lisp Machine and to use the hybrid shell BABYLON of GMD as the basis for the design of its knowledge processing part. BABYLON, which was available as source code, had to be extended for use in a technical, real-time environment. The resulting expert system kernel, as decribed in chapter 4.4, is one of the two basic subsystems of the TEX-I system.

The second subsystem, implemented with traditional (not knowledge based) programming techniques, is the so called Intelligent Process Interface (IPI). The structure of the IPI, including modules for computer network access, signal processing and user dialog, is described in chapter 4.3.

3. Objectives

3.1 Current status

In the control room of the SNR-300, in addition to the automatic control, a conventional system is already installed to help the operators in assessing and managing failure events.

The alarm system primarily consists of three components: A board with two group alarm signals (blinking lights for failure and for abnormal operation), the recorder instruments for important signals and - as the most important part - a monitor displaying the alarm detector signal, the location of the detector and a hint for the dialog scheme in the instruction manual (fig. 1).

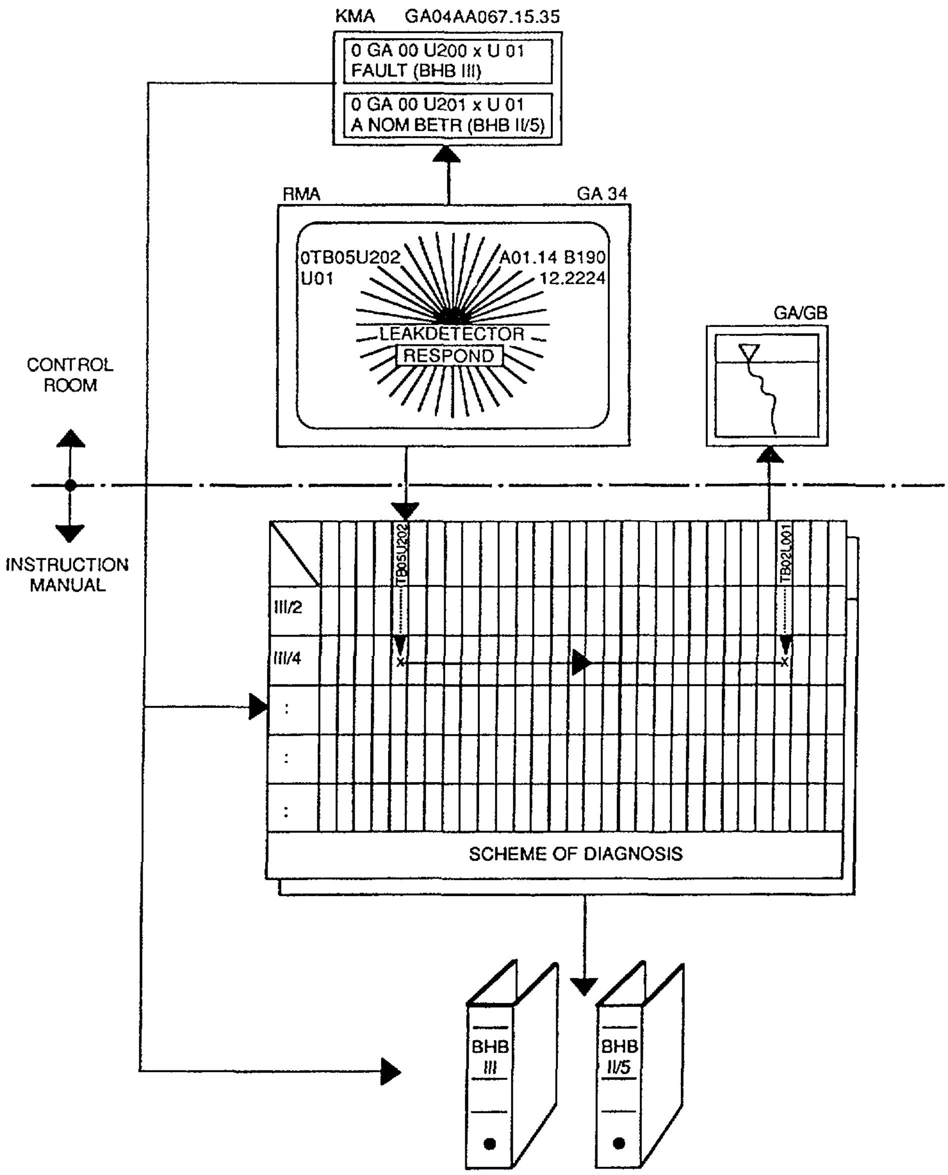

FIG.1. Fault tracing instruction at the SNR-300.

The instruction manual contains a diagnosis matrix, which for each detector signal in correlation to other signals gives hints to the corresponding chapter in the instruction manual. The diagnosis system contains about 60 possible detector signals, half a dozen main failures, and a dozen abnormal operating conditions.

3.2 Aims of the expert system approach

The main purpose of our expert system is the assessment of the reactor status in real-time by monitoring the readings from hundreds of sensors. Via an online connection to the process control computer, it has to support the reactor operators by detecting unusual trends prior to accidents, localizing faulty components and recommending operator responses in abnormal conditions.

We are aware of the fact, that the expert system cannot react fast enough in all situations, because rapid changes in the operation of the nuclear power plant from normal into abnormal states will always lead to automatic nuclear shutdown by the redundant and diversitary protection systems within fractions of a second. Instead, we want to concentrate on slow changes in the operation of the core, coolant circuits, pumps, valves, heat exchangers and control systems, where the operator has a fair chance to react and to bring the reactor in a safe state, before the automatic reactor safety assembly will trigger scram.

3.3 Testing the expert system

Testing the expert system's reaction to different normal and abnormal transients during its design and implementation phase is usually a problem. Obviously, it is not convenient to induce severe disturbances for test purposes, when the reactor is operating in normal mode. Instead, a large scale simulation program has to be developed to generate all relevant signals with the same timing conditions as in the control room.

4. Implementation

4.1 Basic knowledge processing

An expert system, based on BABYLON, can be configured by defining individual interpreters for each knowledge base. The configuration can include interpreters for one or many of the following different knowledge representation formalisms:

- Frame-Interpreter, for an object-oriented system, including frames, behaviors (methods), instances, inheritance, possible value restrictions, active values

- Rule-Interpreter, for forward and backward chaining, different logical junctors and action types, tracing and explanation

- Prolog-Interpreter, a complete Prolog language implemented in Lisp with interfaces to Lisp and the object-oriented frame system

- Constraint-Interpreter, supporting constraints and constraint-nets, used to test consistency between values, remove inconsistent values and to compute unknown values from given boundary conditions

- Lisp-Interpreter, used to evaluate behaviors and instructions for a knowledge base or any lisp function in the condition or action part of rules

- Free-text-Interpreter, taking any expression, not handled by any of the previous mentioned interpreters as true facts, which are stored into a dynamic knowledge base.

By combining different formalisms it is possible to set or get slot values, compare these values by some relation, use a prolog clause, or a constraint relation, or free text, or a complicated lisp function, all within one rule.

A knowledge base itself is divided into several parts. One part contains the instructions to a Meta-Processor, defining the global control of an inference, the knowledge interpreters, used by the knowledge base, the user interface and optional extensions. In other sections all frames, instances and their behaviors are defined, other sections contain different rule-sets, prolog clauses, or constraint definitions.

4.2 Modelling and diagnosing plant components

A complex technical system, such as a reactor, is composed of several interacting subsystems. In our prototype expert system for situation assessment of the SNR-300 reactor we selected only a few important subsystems, including reactor core, primary and secondary coolant loops, and some smaller subsystems as for instance sodium level control or delayed neutron detection.

Each subsystem is composed of many connected components, e.g. pumps, vessels, or valves. A great number of sensors is attached to each component. As one important precondition for a diagnosis, the expert system must have knowledge about this static plant structure and its dynamic behavior, as monitored by the sensor signals.

The modelling of dynamic signals is decribed below in chapter 4.3. The static structure of the reactor system was modelled in a hierarchical manner. Therefore, an universal component description language was developed, which allows the declaration of the physical structure (component hierarchy, defined by parts/part-of relations) and the functional structure of a plant (kind of relations between the components).

Methods are available for finding all direct and indirect components of an object or for finding all compound objects that contain a specific component. Similar methods are used to investigate the relations between components. Different relations indicate connections of different types, like pipes or wires.

In order to identify faulty components, a diagnosis formalism, using a top-down "establish and refine" strategy, was implemented. The plant model is extended by mixing diagnostic information into each component description. This includes a list of all possible fault hypotheses for that component, rules to establish its own degree of certainty by assigning situation dependent focus values to each hypothesis and refinement rules to test more detailed sub-hypotheses.

Starting from a rough localisation of an error, an end diagnosis is reached by repeatedly invoking the sub-diagnoses with the highest focus value. Functions for assigning focus values can be defined by the user. In this way we follow a diagnosis tree with the same structure as the plant model. If this strong hierarchical focus of attention mechanism is not adequate, an agenda-like control structure can also be used.

4.3 Intelligent process interface (IPI)

Our first prototypes, using this diagnosis scheme, were standalone off-line operating systems where process data could only be asked from the operator. In contrast to this conventional, dialog oriented system a real-time expert system has to be directly coupled to the plant, it has to monitor. It has to operate continuously, in order to analyse the incoming measurements and alarms and has to store them into the knowledge base.

In chapter 4.4 an overview is given about the complete architecture of the TEX-I system, developed for this purpose. Here we concentrate on the IPI, that manages the connection to the process control system (PCS) with real-time data acquisition and preprocessing.

The IPI is divided into 3 modules, a communication module, a signal module and a monitor. The communication module is the lowest layer of the IPI. It manages the data transfer from the process control computer to the Lisp Machine and vice versa, and it has to be loaded on both computers. The principal setup is shown in fig. 2.

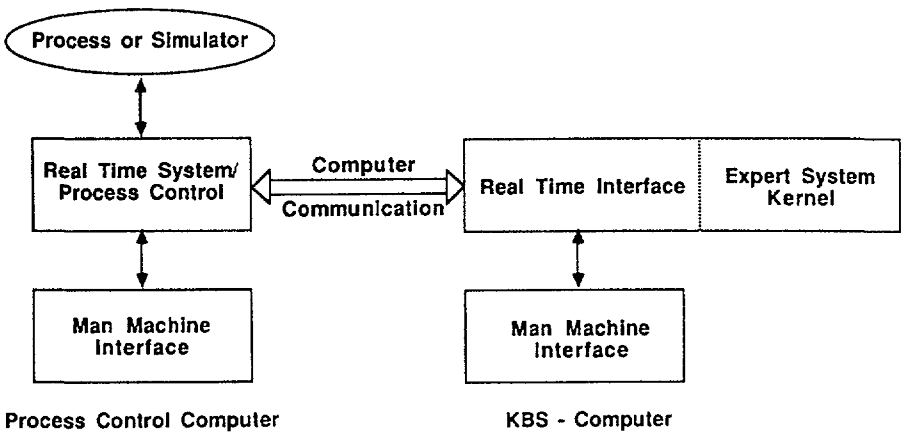

FIG.2. Interface KBS — dynamic process.

Process data can be active or passive signals. Signals are called active, if their value has to be requested by the IPI from the PCS. It is possible to request active signals once or periodically with a given cycle time. Passive signals, typically alarm messages like "EM-Pump temperature too high" or device status messages like "EM-Pump operating" are sent asynchronously from the PCS to the expert system. Both, active or passive signals can be handled by the communication module, and, depending on the actual reasoning process, additional signals can be switched on or off.

The main part of the IPI is the signal module, which consists of two processes, a signal manager and a situation manager. For each sensor in the PCS, there exists one corresponding signal object in the expert system, implemented as an instance of the Lisp object "primary signal". All incoming signal values, together with the time of measurement, are dynamically written into a cyclic history list, which is stored in memory within a signal slot. Other slots define the history length, cycle time and give information about the type of this signal and its measuring place.

The most important purpose of the signal module is to provide a data reduction by abstraction. Data reduction is essential in real-time expert systems for runtime purposes, because it is not useful to fire diagnostic rules with every new signal entry, when typically hundreds of signals per second change their values by small amounts. This example shows, that some preprocessing functions are needed to decouple the rapidly changing sensor data from the diagnosis system. Only signals have to be processed in real-time.

The IPI provides methods for filtering out those events that have to be diagnosed by the expert system. First, a hierarchy of signals is introduced, with the primary process signals at lowest level. Only the top-level signals, called situations, can directly trigger an inference. Computed signals are another type of signals, existing only within the IPI. Their value is calculated from the values of imported primary or other computed signals. A typical example is the computation of signal trends.

At higher levels of abstraction we have situations, which can also be hierarchically ordered. A situation characterises a specific process condition and can be viewed as a partially frozen process picture. Situations can be modelled to import all signals belonging to one physical component, or to import signals or even other situations that are necessary to verify a special fault condition. Please note the close analogy of the second approach to the conventional diagnosis scheme for the SNR-300, as shown in fig. 1.

Supervisory functions, which have to be defined by the user for each signal, can be programmed to convert real number signal values into discrete symbolic values. As an example, temperature surveillance in our reactor expert systems is performed by functions, that know upper and lower warning and alarm thresholds and produce abstract signal values like "high alarm", "normal" or "low warning". Only if these values change, e.g. from "high warning" to "normal", a higher level situation is informed.

The activated situtation then has to decide by its own user defined supervisory function, if an inference has to be started immediatly or if further information has to be collected. Before actually starting an inference, a disturbed situation has to reevaluate its status after a given delay time to be sure, that the error reasons are still there and are not caused by fluctuating signal values near the alarm limits.

4.4 Architecture of the TEX-I System

Fig. 3 shows, how the two TEX-I subsystems, intelligent process interface and the BABYLON expert system kernel, cooperate. Starting an inference means nothing else than starting a session like in a conventional dialog-oriented expert system. A so called inference process is created on the Lisp Machine, including all knowledge processing, tracing and dialog functions of a complete BABYLON. The original BABYLON therefore had to be extended by process properties.

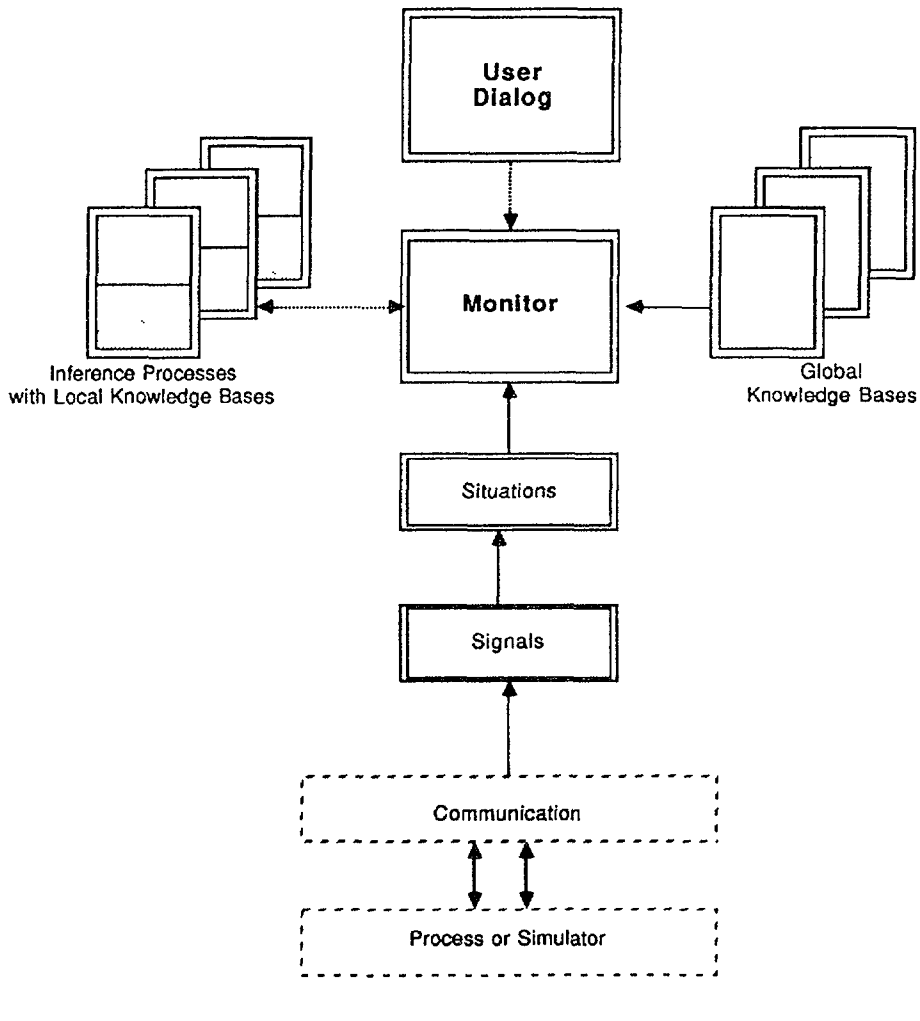

FIG.3. Architecture of the TEX-I system.

If a long running inference process is active while other situations get disturbed and need expert system support, new inference processes can be created to run quasiparallel with others.

Inference processes are priority controlled and can interrupt or abort each other. Creation and scheduling of inference processes is done by a monitor, as shown in Fig. 3. The monitor also has to manage the user dialog.

An inference process has its own knowledge base, but can also get access to global knowledge bases. In our reactor expert systems, one global knowledge base contains the static plant model. Dynamic signals are not included in the static model, but via direct links between components and signals, an inference process can access each signal at any time, available in the history lists.

5. Prototypes for the SNR-300

The framework of the TEX-I system was designed by all partners together and implemented by the institutes. The industrial partners tested the system with their application and provided the necessary feedback and suggestions for further development in the next project phase. The final TEX-I system is available now since the mid of 1988.

5.1 Off-line versions

Based on the "establish and refine" diagnosis strategy, we developed an expert system for the primary system of the SNR-300 reactor with prototypes for the subsystems sodium level control, measurement subsystem for delayed neutrons, primary cooling loops, and the supply and control system for gas pressure in the primary vessel.

The hierarchy of subsystems and components is modelled with the component description language. For each component the possible fault situations are predefined. They are analysed by comparing their focus values. Focus values are evaluated during a session after classification of the signal values of all connected sensors. We take into account the fact, that in a technical system the measured value of a physical quantity has an error, because of inexact measurement or instrumentation limits. The classification procedure therefore uses fuzzy logic methods to assign possibility values for statements like "the pressure is high" or "normal" or "low". Finally, these values are accumulated into one focus value.

We found, that these systems are excellent fault finders, but they are not suitable for efficient on-line use. In many cases, the alarm messages are unambiguous and no further reasoning is needed to propose actions to the operator. Methods to filter out such situations and to perform the necessary consistence checks to confirm that situation, are provided by the IPI.

5.2 Prototype for the sodium level control system

The prototype for the sodium level control system (SLC) was extended to make use of the signal processing functions of the IPI. Signal values are provided by simulation. The SLC was selected, because it is a very important but small subsystem of the sodium-cooled breeder reactor SNR-300. The complete set of sensors, given in the technical documentation of the fully automated SLC, could be modelled in the dynamic knowledge base.

The main purpose of the SLC is to keep the level of the sodium coolant in the reactor vessel constant. It can also indicate small leakages in the reactor vessel, the SLC itself, and in the primary cooling loops. It consists primarily of a level control tank, where the overflow from the reactor vessel is buffered, and 2 EM-pumps, which operate in "1 out of 2" reduncancy mode in order to transport the sodium back to the reactor vessel.

Some useful features of the expert system approach can be demonstrated with the SLC prototype. As a simple example, the early warning function, as shown in fig. 4, will briefly be discussed:

If the temperature of one EM-Pump becomes too high, it has to be switched off automatically. To prevent this, the IPI is continuously monitoring the two thermocouples, connected to the pump tube. By looking into the temperature history, the trend, "decreasing", "normal" or "increasing" is calculated, whenever a new temperature value is stored. If at least one of the temperatures is "rather high" and "increasing", the expert system reacts, like an operator should do. It tries to detect a possible error by starting a complete diagnosis of the SLC itself and the connected subsystems, like the nitrogen gas cooling system for the pumps by a procedure, similiar to the one described in chapter 5.1.

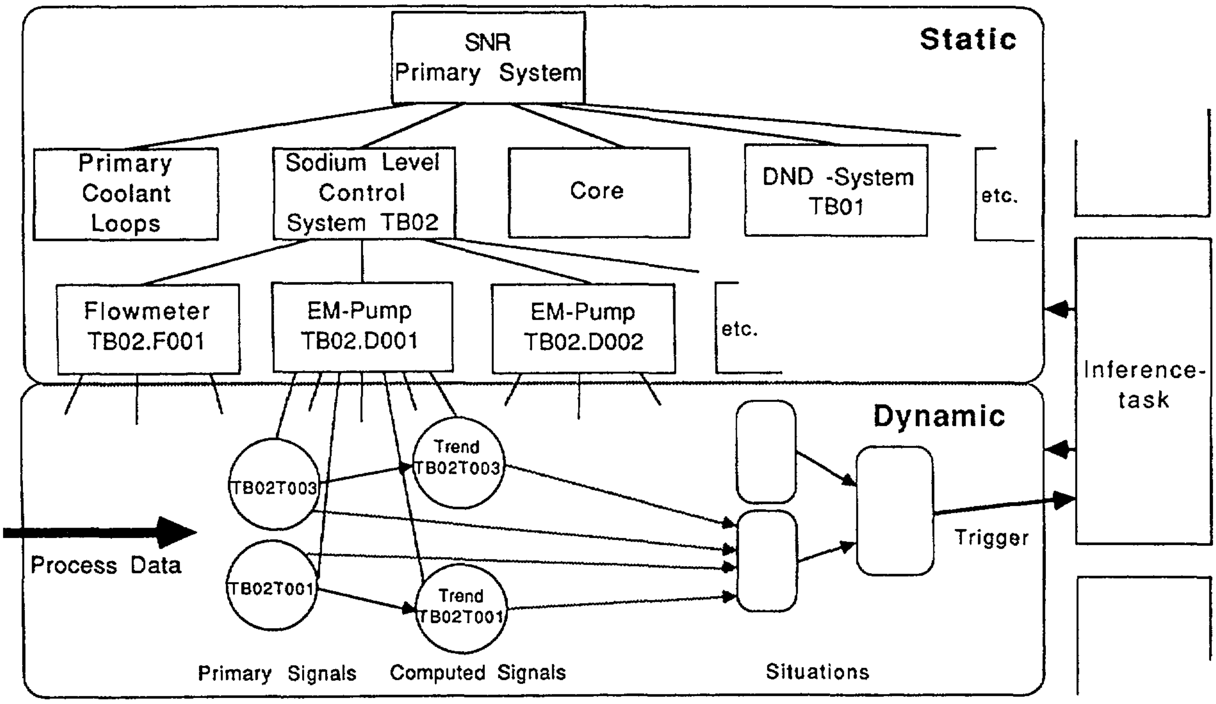

FIG.4. Plant model.

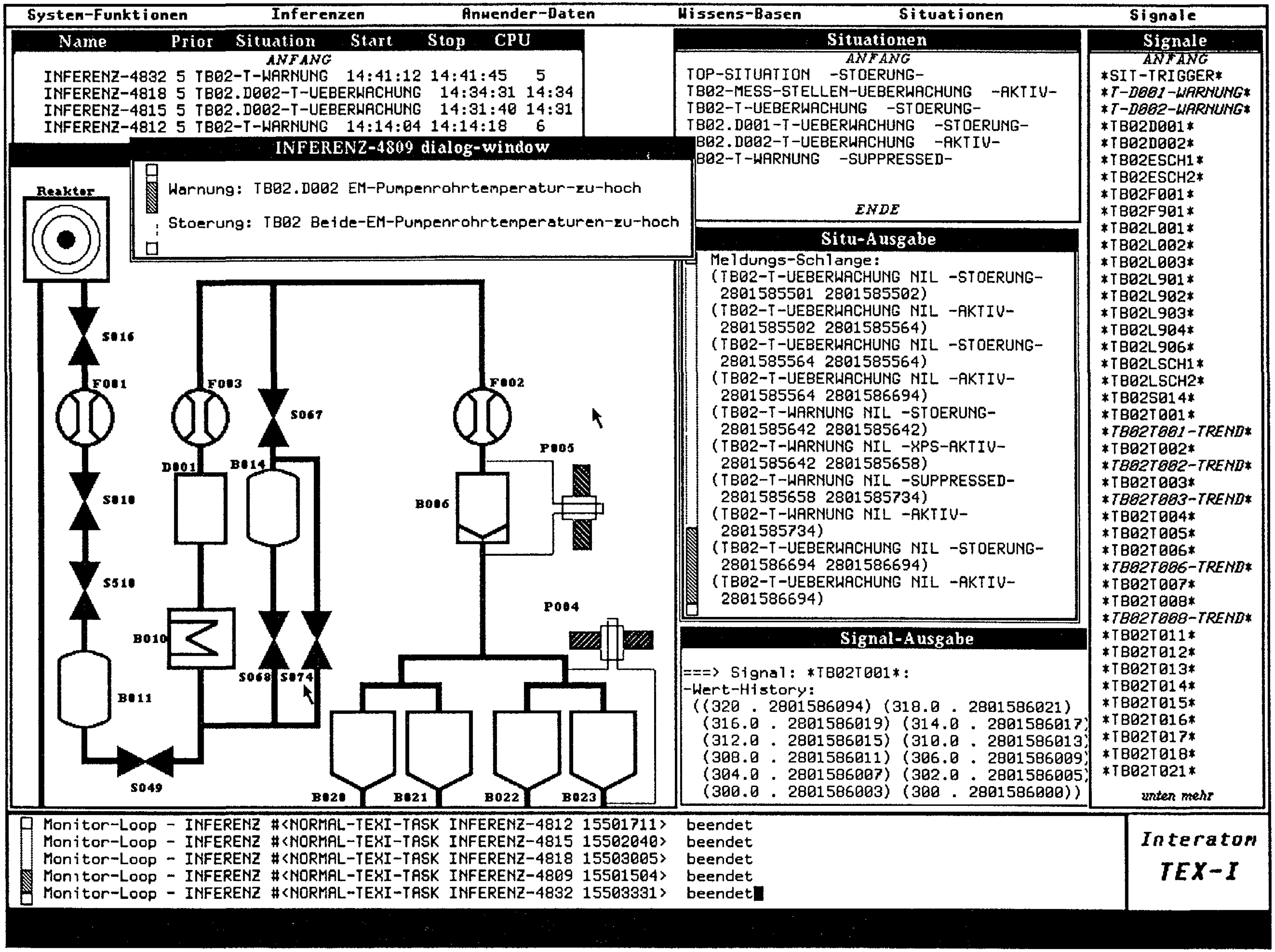
System-Funktionen
Inferenzen
Anwender-Daten
Wissens-Basen
Situationen
Signale
Name Prior Situation Start Stop CPU
ANFANG
INFERENZ-4832 5 TB02-T-WARNUNG 14:41:12 14:41:45 5
INFERENZ-4818 5 TB02.D002-T-UEBERWACHUNG 14:34:31 14:34
INFERENZ-4815 5 TB02.D002-T-UEBERWACHUNG 14:31:40 14:31
INFERENZ-4812 5 TB02-T-WARNUNG 14:14:04 14:14:18 6
INFERENZ-4809 dialog-window
Warnung: TB02.D002 EM-Pumpenrohrtemperatur-zu-hoch
Stoerung: TB02 Beide-EM-Pumpenrohrtemperaturen-zu-hoch
Reaktor
S016
F001
F003
S067
F002
D001
B014
P005
S010
B006
S510
P004
B010
S068
S074
B011
S049
B020
B021
B022
B023
Situationen
ANFANG
TOP-SITUATION -STOERUNG-
TB02-MESS-STELLEN-UEBERWACHUNG -AKTIV-
TB02-T-UEBERWACHUNG -STOERUNG-
TB02.D001-T-UEBERWACHUNG -STOERUNG-
B02.D002-T-UEBERWACHUNG -AKTIV-
B02-T-WARNUNG -SUPPRESSED-
ENDE
Situ-Ausgabe
Meldungs-Schlange:
(TB02-T-UEBERWACHUNG NIL -STOERUNG-
2801585501 2801585502)
(TB02-T-UEBERWACHUNG NIL -AKTIV-
2801585502 2801585564)
(TB02-T-UEBERWACHUNG NIL -STOERUNG-
2801585564 2801585564)
(TB02-T-UEBERWACHUNG NIL -AKTIV-
2801585564 2801586694)
(TB02-T-WARNUNG NIL -STOERUNG-
2801585642 2801585642)
(TB02-T-WARNUNG NIL -XPS-AKTIV-
2801585642 2801585658)
(TB02-T-WARNUNG NIL -SUPPRESSED-
2801585658 2801585734)
(TB02-T-WARNUNG NIL -AKTIV-
2801585734)
(TB02-T-UEBERWACHUNG NIL -STOERUNG-
2801586694 2801586694)
(TB02-T-UEBERWACHUNG NIL -AKTIV-
2801586694)
Signal-Ausgabe
===> Signal: *TB02T001*:
-Wert-History:
((320 . 2801586094) (318.0 . 2801586021)
(316.0 . 2801586019) (314.0 . 2801586017)
(312.0 . 2801586015) (310.0 . 2801586013)
(308.0 . 2801586011) (306.0 . 2801586009)
(304.0 . 2801586007) (302.0 . 2801586005)
(300.0 . 2801586003) (300 . 2801586000))
Signale
ANFANG
SIT-TRIGGER
T-D001-WARNUNG
T-D002-WARNUNG
TB02D001
TB02D002
TB02ESCH1
TB02ESCH2
TB02F001
TB02F901
TB02L001
TB02L002
TB02L003
TB02L901
TB02L902
TB02L903
TB02L904
TB02L906
TB02LSCH1
TB02LSCH2
TB02S014
TB02T001
TB02T001-TREND
TB02T002
TB02T002-TREND
TB02T003
TB02T003-TREND
TB02T004
TB02T005
TB02T006
TB02T006-TREND
TB02T007
TB02T008
TB02T008-TREND
TB02T011
TB02T012
TB02T013
TB02T014
TB02T015
TB02T016
TB02T017
TB02T018
TB02T021
unten mehr
Monitor-Loop - INFERENZ #<NORMAL-TEXI-TASK INFERENZ-4812 15501711> beendet
Monitor-Loop - INFERENZ #<NORMAL-TEXI-TASK INFERENZ-4815 15502040> beendet
Monitor-Loop - INFERENZ #<NORMAL-TEXI-TASK INFERENZ-4818 15503005> beendet
Monitor-Loop - INFERENZ #<NORMAL-TEXI-TASK INFERENZ-4809 15501504> beendet
Monitor-Loop - INFERENZ #<NORMAL-TEXI-TASK INFERENZ-4832 15503331> beendet
Interaton
TEX-I

FIG.5.

If the temperature is too high, the expert system has to supervise the correct automatic switch-over to the reserve pump, which could fail. In case of failure, the operator is alarmed by the expert system that there is a fault in the automatic switching system. This message is not provided by the process control system. Supervising all steps of automatic switch-over means, that many time restrictions, signal delays and interlocking regulations have to be taken into account, which even depend on the actual operating conditions as stored in the static knowledge base. The timing functionality is provided by the IPI. Note that conventional expert systems do not provide support for this class of time dependent problems.

Fig. 5 shows a typical screen layout. Without going into detail, one can see a partial list of signals, whose values and histories can be shown on mouse-click; a list of active and disturbed situations, whose corresponding alarm queues can also be shown with mouse-click; a process graphic; a list of active and completed inference processes; and the messages produced by the last inference.

5.3 Application to reactor operation

5.3.1 Simulator

In order to test the expert system in a real-time environment, a large scale thermohydraulic simulation program for the SNR-300 power plant was used. The simulator, running on a Lisp Machine, is on-line coupled to the expert system computer by an ethernet connection.

The numerically simulated plant model covers all subsystems, shown in the schematic reactor layout of fig. 6. It consists of the heat producing core, three primary sodium coolant loops,

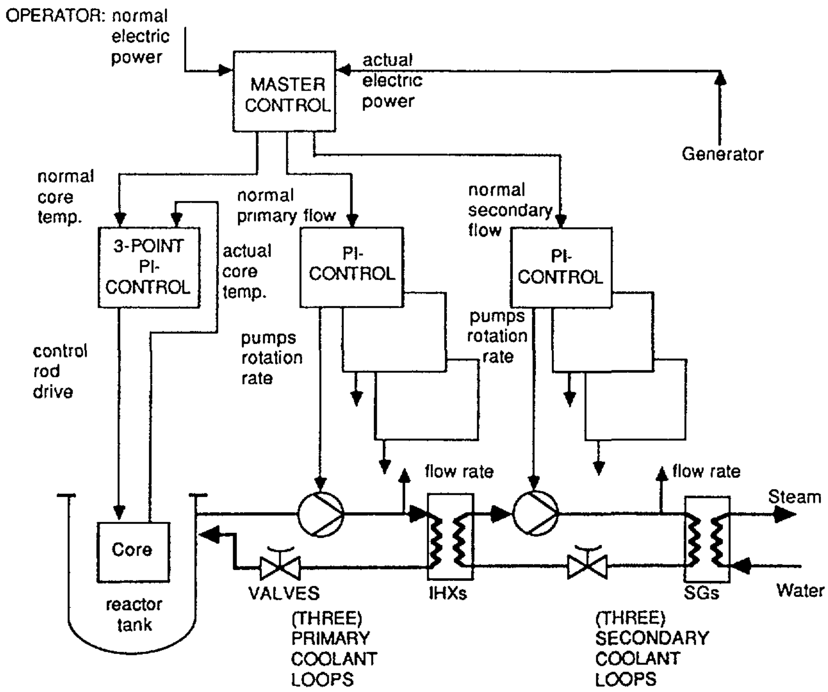

FIG.6. Fast breeder reactor layout.

three intermediate heat exchangers, three secondary coolant loops, three steam generators and three water/steam loops leading each to an electricity producing generator.

The core and coolant loops are regulated by a master controller with inputs: normal and electric power and outputs: normal values of core temperature and primary and secondary flow rates. These are themselves input to the core 3-point PI-controller and the primary and secondary PI-controllers, which regulate the core temperature and the coolant flow rates via the control rod drive and the pumps rotation rates, respectively. Thus, the controlling system will automatically adjust the reactor to a new operating state, if the operator chooses to change the power, e.g. from 100% to 90%.

The simulation code, written in FORTRAN, was optimized to run on a Lisp Machine in real-time or even faster. By solving the differential equations of the neutronics of the primary and secondary coolant loops including the complete controlling system, each second a set of variable values is produced by the code. The values correspond to the measureable neutron-flux, pressures, temperatures, flow rates, pump rotation rates, slide and control rod positions of the reactor core, plena, pipes, pumps, valves, intermediate heat exchangers and cooling components.

These signal values can be plotted against time using a graphic interface, as shown in fig. 7. Additionally, the whole set of about 100 signal values is transferred each second to the on-line coupled expert system, residing on a second Lisp Machine. The architecture of the whole system is shown in fig. 8.

The user has the choice to run one out of ten prefabricated accident cases, which have a combination of faults, like
- reactivity disturbance
- control rod getting stuck
- primary pump revolution rate remaining constant
- control of secondary pump not available

and normal operating procedures, like change of electric power.

Alternatively, the user can run the normal operating case and has the options to interactivly manipulate parameters, which characterise deviations from the normal behavior of controllers, pumps, valves and control rod velocity.

It is worth noting that in contrast to the expert system interface (fig. 5), which is designed for the operator and knowledge engineer, the simulator graphic interface is designed also to support the reactor expert, whose knowledge (also heuristic knowledge) has to be aquired. By analysing the time dependencies and correlations of sensor data, which can simply be reproduced, it is often easier to formulate or reformulate the experts declarations in terms of the available signals.

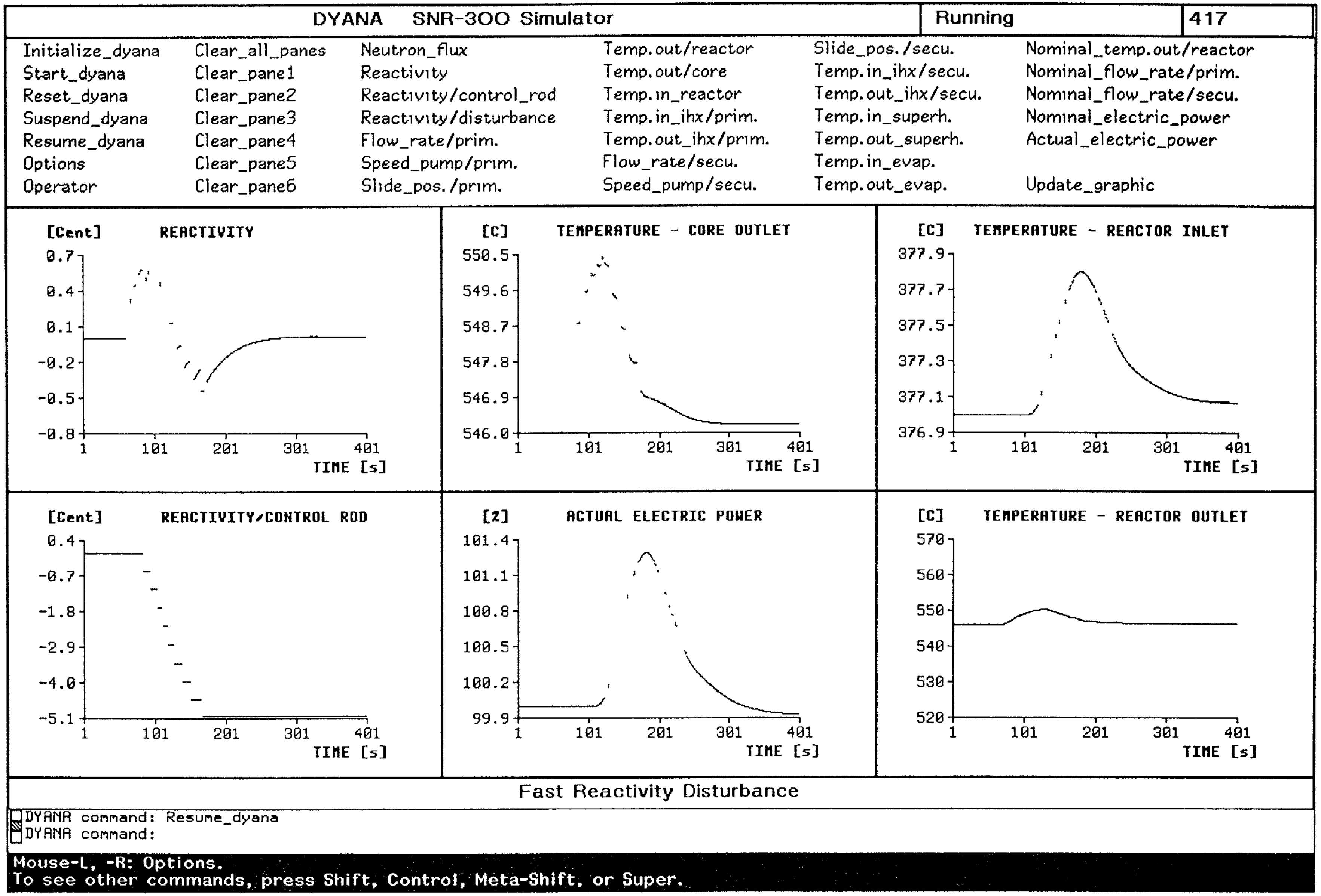
DYANA SNR-300 Simulator
Running
417
Initialize_dyana
Start_dyana
Reset_dyana
Suspend_dyana
Resume_dyana
Options
Operator
Clear_all_panes
Clear_pane1
Clear_pane2
Clear_pane3
Clear_pane4
Clear_pane5
Clear_pane6
Neutron_flux
Reactivity
Reactivity/control_rod
Reactivity/disturbance
Flow_rate/prim.
Speed_pump/prim.
Slide_pos./prim.
Temp.out/reactor
Temp.out/core
Temp.in_reactor
Temp.in_ihx/prim.
Temp.out_ihx/prim.
Flow_rate/secu.
Speed_pump/secu.
Slide_pos./secu.
Temp.in_ihx/secu.
Temp.out_ihx/secu.
Temp.in_superh.
Temp.out_superh.
Temp.in_evap.
Temp.out_evap.
Nominal_temp.out/reactor
Nominal_flow_rate/prim.
Nominal_flow_rate/secu.
Nominal_electric_power
Actual_electric_power
Update_graphic
[Cent] REACTIVITY
0.7
0.4
0.1
-0.2
-0.5
-0.8
1 101 201 301 401
TIME [s]
[C] TEMPERATURE - CORE OUTLET
550.5
549.6
548.7
547.8
546.9
546.0
1 101 201 301 401
TIME [s]
[C] TEMPERATURE - REACTOR INLET
377.9
377.7
377.5
377.3
377.1
376.9
1 101 201 301 401
TIME [s]
[Cent] REACTIVITY/CONTROL ROD
0.4
-0.7
-1.8
-2.9
-4.0
-5.1
1 101 201 301 401
TIME [s]
[%] ACTUAL ELECTRIC POWER
101.4
101.1
100.8
100.5
100.2
99.9
1 101 201 301 401
TIME [s]
[C] TEMPERATURE - REACTOR OUTLET
570
560
550
540
530
520
1 101 201 301 401
TIME [s]
Fast Reactivity Disturbance
DYANA command: Resume_dyana
DYANA command:
Mouse-L, -R: Options.
To see other commands, press Shift, Control, Meta-Shift, or Super.

FIG.7.

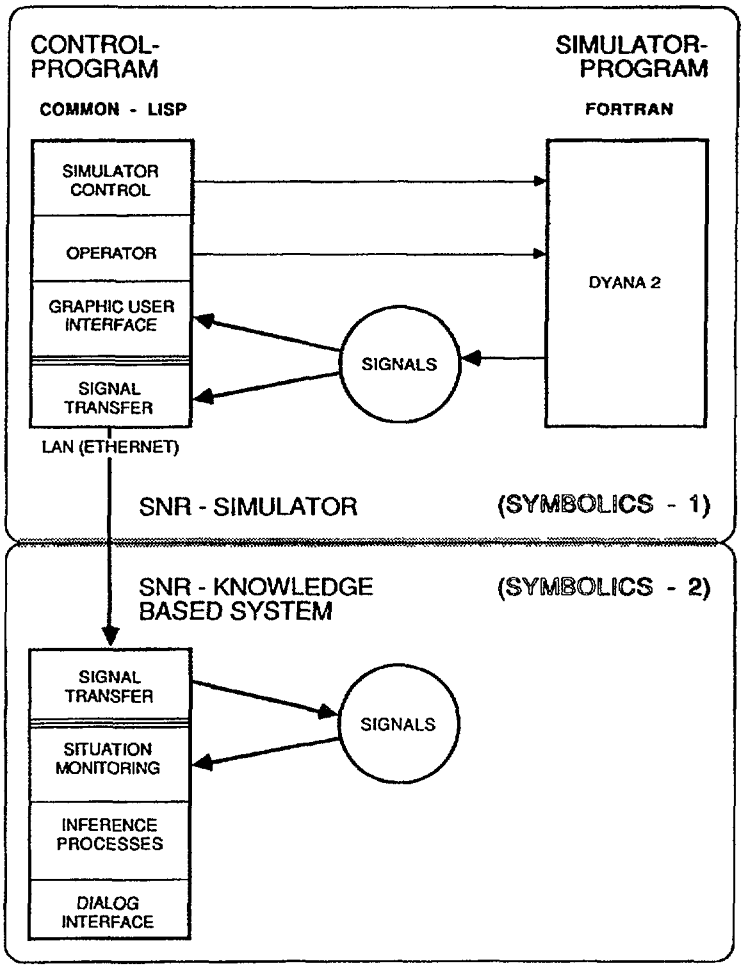

FIG.8. Simulator connection to KBS.

5.3.2 Supervision and fault diagnosis

In the on-line coupled expert system, the information carried by the primary signals and relevant to fault diagnosis is condensed into a smaller set of computed signals, such as

- difference between normal and actual values of
  - o core outlet temperature
  - o primary or secondary flow rates or
  - o electric power

- trends of
  - o actual electric power
  - o pump rotation rates or
  - o coolant flow rates

- differences between the pressure steps at the pumps and the pressure steps derived from the pump characteristic.

These signals are updated as soon as one of their primary signal values changes and are written into their cyclic history list. Each of the computed signals is kept under surveillance by the "situation manager", as decribed in chapter 4.3, and each change from normal to abnormal values and vice versa will be written into a message queue of a particular situation for further processing.

The situations, which can be in states such as

- active
- normal
- disturbed
- XPS-active (an inference process is running, triggered by this situation)
- XPS-finished or
- suppressed,

correspond to the normal or abnormal behavior of the interesting components control rod, pumps, valves and controllers of the reactor.

Thus, at the level of normal or disturbed situations, the desired fault diagnosis has already been realized. The following types of operating conditions can be diagnosed up to now

- disturbances of components of the reactor core, primary and secondary cooling system, such as control rod position, valve position, driving torques of pumps
- normal and abnormal operation of the controllers for core outlet temperature, primary and secondary coolant flow rates and the master control
- occurence of scram conditions
- realibility of n-flux measurements
- value of nominal electric power.

The setup of the complete knowledge base is still in progress.

6. Conclusions

It has been demonstrated, how real-time fault diagnosis of small or slowly developing normal or abnormal transients in the fast breeder reactor SNR-300 can be performed within the TEX-I system. All monitoring, diagnosing and controlling tasks we considered so far, could be implemented with this expert system.

The TEX-I system provides many design features of real-time knowledge based systems like

- static and dynamic knowledge bases
- interface for online coupling to a process control computer
- real-time data processing of asynchronous events
- multitasking of inference processes
- interrupt and priority driven scheduling.

There are, however, some limitations: Continuous monitoring cannot be guaranteed because the system has to stop periodically for a few seconds for "garbage collection", if many data are transferred to the Lisp Machine. Other performance limitations have to be expected in really big systems with hundreds of signals per second and tens of situations, because of the very time consuming signal processing.

But these problems arise mainly from hardware and software limitations of present Lisp Machines. The concept of the TEX-I systen is prepared for new parallel hardware architectures and has been proven to be successful in even more real-time applications than just situation assessment in a reactor.

APPLICATION OF INSIGHT 2+ TO SAFETY ANALYSIS

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Hungary

Abstract

The paper examine possible application of an expert system shell Insight 2+ for improvement in plant safety.
The paper briefly reviews the advantage as well as the limitation of this tool and gives two real life examples including ICS related transients and condenser fault diagnosis to illustrate the range of possible benefits and problems.

Scope

There are two main software directions, one can take expert system shell or a programming language. A shell is a high-level language and some of its benefits are basic support for rule entry built in inference engines, explanation facilities, screen output and data storage utilities.
One of the advantages of these tools is to allow the domain engineer to concentrate on building the expert system rather than having to expend resources developing the necessary tools. The main emphasis is on creating rule based knowledge base with commercial low cost shell for safety related problems.

Conclusions and Significances

Although the interest in employing AI technology in power plant operation is considerably high, the real life realization of each type of projects relatively slow. The success of practical application of AI technology at power plant level depends on how economical, effective and last but not least how clear and

understandable the performance of these tools for the plant staff. Economical consideration implies employing microcomputers having attraction because of their cheapness and availability. It is also important to use low-cost software product which can be easily changed or modified without considerable losses if it would be necessary after the experience with small prototypes.
The effectivity strongly depends on how realistic is the knowledge base consists of facts and rules, which has to emulate the operator's thought process. Therefore close collaboration between operators, power plant engineers, supervisors all having not necessarily special programming or AI knowledge, and domain expert is inevitable. Consequently, the applied tool must be simple, and not too sophisticated. The plant personnel need clear explanation why the expert system recommended certain solution.
All of these requirements can be reasonably satisfied by using a shell.
Insight 2+ has additional capabilities for direct use of programming language like Pascal and access to Dbase III. as well as graphic supports like Halo and Paintbrush.
Despite of its low price and commercial availability, it has considerable capacity about 4000 facts and 2000 rules, which is far beyond the requirement of a small prototype system. Therefore its application as first approach is highly recommended.

Why Insight 2+ ?

Insight 2+ is an integrated expert system shell consisting of four parts:

Inference Engine -- The inference engine controls how Insight 2+ pursues goals, applies rules, performs queries, and reaches conclusions.

Knowledge-Base Compiler -- Knowledge bases are executed in an internal compiled form, which minimizes search time and optimizes execution speed.

Knowledge-Base Editor

-- The multiwindowing rule editor is used to create Insight 2+'s knowledge bases. It is fully integrated into Insight 2+ and accessed through the main menu.

Database-Access Modules

-- Database calls are embedded within rule grammar and require no outside routines. Through the appropriate interface, Insight 2+ knowledge bases can access popular commercial databases to extract relevant data on the specific area of analysis.

Special Features

Inference Engine supports a wide variety of control strategies, including:

* Backward chaining logic-starts with a goal and works backward to determine conditions required to reach that goal.
* Forward chaining logic-starts with known facts and works forward through rules.
* Goal outlining-allows goals to be pursued in a logical outline, not just sequentially.

English Syntax is used to develop the rule grammar, making the knowledge base virtually self-documenting. The time spent maintaining and modifying the knowledge base is dramatically reduced.

Report capability supports customized output reports to document user's interaction with knowledge base for audit purposes and output creation.

Mathematical Capability supports scientific and engineering applicationswith logarithmic and trigonometric functions, floating-point arithmetic to six significant digits and parentetical nesting to 100 levels.

Confidence Weighting designates the confidence factors of each fact and the strength of correlation between a rule 's premise and conclusion. Integer values (0 to 100) express trueness of a particular fact.

Rebugging Tools (such as Line of Reasoning Reporting, Knowledge Tree Reports and What-If Scenario Player) help users build expert system.

Comparing with other shells Insight 2+, respectively its enhanced version LEVEL 5, have a good position concerning overall tool rating and its cost weighted modification, see Fig.1.

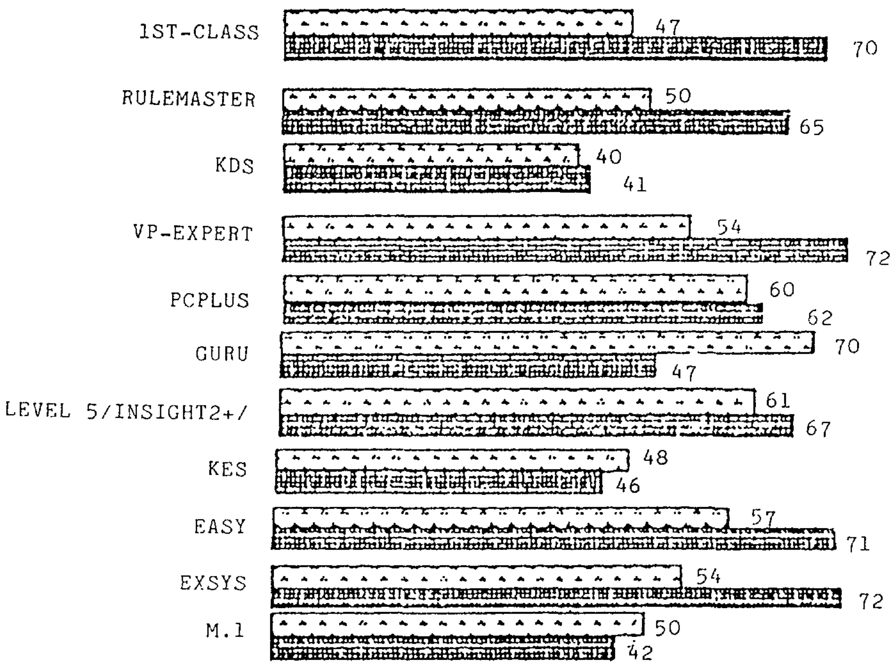

FIGURE 1.

This comparison has been carried out according to 10 important features of the expert system namely: knowledge representation, search strategies, multivalued information, certainty, interfaces, explanation facilities, productivity tools, integration, security and cost factors.

Overall Tool Rating in Expertise Associates' The PC Expert Systems Shoot-Out

Tool Rating with cost weighted at 10 and all other set of features weighted at 1.

Examples for Application of Insight 2+

Example 1.: Diagnosting and controlling ICS-related transients. Electric Power Research Institute (EPRI) has developed an aexpert system supporting operator in the process of diagnosing accident and bringing that accident under control. This system consists of 70 logic rules that were programmed for IBM-PC computer using a software called SMART written in LISP. The knowledge base of this system was adopted for Insight 2+. The adoptation took several hours even for a layman becuase of the very transparent code of Insight 2+.

The dialog between operator and computer in case of an anticipated transient without scram (ATWS) event without ICS is presented in Table 1. The scenario of this transient can be seen in Fig. 2. Table 2. shows the Line of Reasoning Report of Insight 2+ explaining the conclusion has been reached.

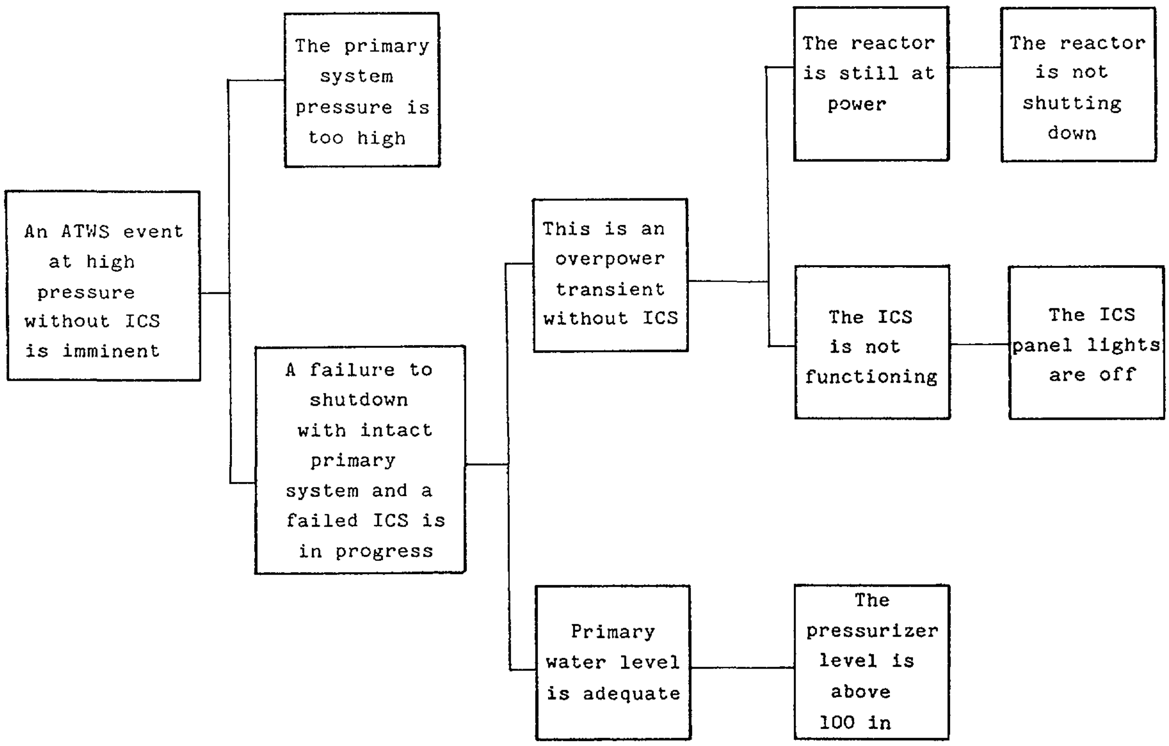

FIGURE 2.

TABLE 1.

Is it true that:

There are alarms sounding in the control room

TRUE

MESSAGE: Verify that there is off-site power

Is it true that:

The primary system pressure is above 2200 psi

TRUE

Is it true that:

Intermediate range detector indicates smaller than 10% power

FALSE

MESSAGE: Manually insert control rods

Is it true that:

the ICS panel lights are off?

TRUE

MESSAGE: The ICS has failed.

MESSAGE: Manually insert control rods. Feedwater pump flow control at midpoint. Use hand control to maintain water level on secondary side of OTSG. Use remote shutdown panel for turbine bypass valve(TBV) and and automatic dump valve (ADV) control. Verify that the turbine has tripped.

MESSAGE: Check ICS cabinet circuit breakers, reset then and check for loose wires.

MESSAGE: The reactor has not shut down. The ICS has failed.

Is it true that:

The pressurizer level is above 100 inches.

TRUE

MESSAGE: This is an ATWS event with the ICS failed. Begin emergency boration. Open PRVs. Follow ATWS guidelines without using ICS-powered controls or instruments.

MESSAGE: An ATWS at high pressure is in progress. The ICS has failed. Attempt to depressurize and scram the reactor again. Follow ATWS procedures.

TABLE 2.

```
Line of Reasoning Report      18/10/1988   13:39:47

Knowledge Base: Expert

The following goal was pursued:
    There is a possible loss of reactor power
The following simple fact was obtained:
    There are alarms sounding in the control room = True
As a result the following conclusion was reached:
    The reactor is in an unacceptable state = True
As a result the following conclusion was reached:
    There is a possible loss of reactor power = True
The following goal was pursued:
    An ATWS event at high pressure without ICS is imminent
The following simple fact was obtained:
    The primary system pressure is above 2200 psi = True
As a result the following conclusion was reached:
    The primary system is in an overpressure state = True
The following simple fact was obtained:
    Intermediate range detector indicates smaller than 10%
    power                                         = False
```

```
Line of Reasoning Report       18/18/1988  13:48:39

Knowledge Base: Expert

As a result the following conclusion was reached:
   Reactor is at power = True
As a result the following conclusion was reached:
   Reactor is still at power = True
The following simple fact was obtained:
   The ICS panel lights are off = True
As a result the following conclusion was reached:
   The ICS is not functioning = True
As a result the following conclusion was reached:
   Normal plant control is compromised = True
As a result the following conclusion was reached:
   ICS repair is worth considering = True
As a result the following conclusion was reached:
   This is an overpower transient without ICS = True
The following simple fact was obtained:
   The pressurizer level is above 100 inches = True
```

```
Line of Reasoning Report      10/10/1988   13:41:32

Knowledge Base: Expert

As a result the following conclusion was reached:
    The primary water level is adequate = True
As a result the following conclusion was reached:
    A failure to shutdown with IPS with a failed ICS = True
As a result the following conclusion was reached:
    An ATWS event at high pressure without ICS is imminent = True
```

The expert system begins to function when the plant is running at power and suddenly experiences a transient and becomes unsteady. Questions are asked of the operator who responds by answering TRUE or FALSE. Based on his responses, the code will either ask additional questions or when an intermediate conclusion can be reached, It will be displayed, possibly with suggestions on what to do next.

Example 2.: Condenser fault diagnostic

This real life problem which can undirectly also lead to safety transient is a good example for the transparent code of insight 2+. Let us consider the following scenario, see Fig. 3.:

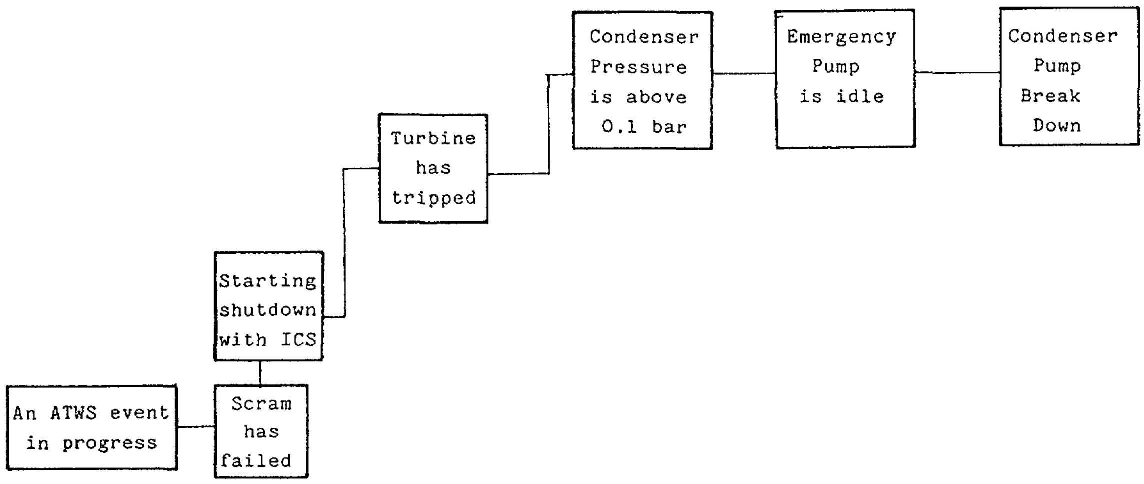

FIGURE 3.

From detectable symptoms the program concludes the possible faults. These faults and symptoms are in Table 3.

TABLE 3.

FAULTS	SYMPTOMS
1. Steam Ejector Pump Failure	1. Condensate Temperature Increase
2. Fouling of Heat Transfer Surface	2. Steam Temperature Increase
3. Condenser Pipe Break	3. Steam Temperature is Higher then Saturation Temperature
4. Emergency Pump is Idle	4. Condenser Pressure slow Increase
5. Level Control Valve Failure	5. Condenser Pressure Rapid Increase
6. Reduction Valve Leakage	6. Condensate Level Decrease
7. Condensate Pump Break Down	7. Condensate Level Increase
8. Leakage in Condenser System	8. Water Quality getting worse
9. Condenser Pump Cavitation	
10. Waste Water Feeded into Condenser	

```
RULE 1.
IF Steam Temperature Increase
AND Steam Temperature Is Higher Than Saturation Temperature
AND Condenser Pressure Rapid Increase
OR Condenser Pressure Slow Increase
AND Condensate Temperature Increase
AND Water Quality Setting Worse
THEN Steam Ejector Pump Failure
AND DISPLAY Message 1.
AND Coal
!
RULE 2.
IF Steam Temperature Increase
AND Condenser Pressure Slow Increase
AND Condensate Temperature Increase
THEN Fouling of Heat Transfer Surface
AND DISPLAY Message 2.
AND Goal
!
RULE 3.
IF Condenser Pressure Rapid Increase
AND Condensate Level Increase
AND Water Quality Setting Worse
THEN Condenser Pipe Break
AND DISPLAY Message 3.
AND Goal
!
```

```
RULE 4.
IF Condensate Temperature Increase
AND Steam Temperature Increase
AND Condenser Pressure Rapid Increase
AND Condensate Level Increase
THEN Emergency Pump is Idle
AND DISPLAY Message 4.
AND Goal
!
RULE 5.
IF Condenser Pressure Rapid Increase
AND Condensate Level Increase
OR Condensate Level Decrease
THEN Level Control Valve Failure
AND DISPLAY Message 5.
AND Goal
!
RULE 6.
IF Steam Temperature Increase
AND Condenser Pressure Slow Increase
THEN Reduction Valve Leakage
AND DISPLAY Message 6.
AND Goal
!
RULE 7.
IF Condensate Temperature Increase
AND Steam Temperature Increase
AND Condenser Pressure Rapid Increase
THEN Condensate Pump Break Down
AND Display Message 7.
AND Goal
!
RULE 8.
IF Condensate Level Decrease
THEN Leakage in Condenser System
AND DISPLAY Message 8.
AND Coal
!
```

```
RULE 9.
IF Condenser Pressure rapid Increase
THEN Condenser Pump Cavitation
AND DISPLAY Message 9.
AND Goal
!
RULE 10.
IF Condensate Temperature Increase
THEN Waste Water Feeded into Condenser
AND DISPLAY Message 10.
AND Goal
!
RULE 11.
IF NOT Coal
THEN Stand
AND Display Message 11.
!
DISPLAY Message 1.
        MESSAGE : Steam Ejector Pump has broken down !

!
DISPLAY Message 2.
        MESSAGE : Heat Exchanger Surface has been fouled!

!
DISPLAY Message 3.
        MESSAGE : Condenser Pipe has broken !
!
DISPLAY Message 4.
        MESSAGE : Condensate Pump Break Down and Emergency
                  Pipe Idle !

!
DISPLAY Message 5.
        MESSAGE : Level Control Valve Failure !

!
DISPLAY Message 6.
        MESSAGE : Reduction Valve Leakage !

!
DISPLAY Message 7.
        MESSAGE : Condensate Pump Break Down !

!
```

```
DISPLAY Message 8.
        MESSAGE : Leakage in Condenser System !
!
DISPLAY Message 9.
        MESSAGE : Condenser Pump Cavitation !
!
DISPLAY Message 10.
        MESSAGE : Waste water has been feeded into Condenser !
!

END
```

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ENVISAGED ROLE OF EXPERT SYSTEMS IN CONTROL AND INSTRUMENTATION OF INDIAN PHWRs

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Abstract

The control and instrumentation hardware in Indian PHWRs is on the verge of undergoing a transition from conventional hard wired systems to distributed computer networks for monitoring and control. The new systems will have more sophisticated data acquisition systems for gathering larger amount of information and presenting in flexible formats suitable for operator comprehension. Implementation of knowledge based systems to process this vast amount of information and provide timely and relevant messages to control room operators are known to be of great help to them especially during plant upset or accident conditions. Attempts are therefore being made to initiate development of such knowledge based systems even while the hardware and software for computer based control systems are being developed. This has been the result of the realisation that achievement of plant safety and good availability hinges as much on operator reliability and good man-machine interface as on the plant design.

1. Introduction

The control and instrumentation(C&I) in nuclear power plants have grown considerably in scope and complexity, mainly because of increased concern for safety and the desire to achieve higher plant availability. This has resulted in significant increase in the volume and types of information presented to the operators, thus making the job of comprehension of plant operating situation more difficult in the present control room concept. Use of computer based C & I systems makes it possible to incorporate advanced control room features and better man-machine interface functions to present information in readily comprehendable manner and thus increase operator reliability.

The extraction of maximum returns from the large investments in nuclear power plants (NPPs) can be realised only through plant operation close to designed power with maximum availability. This can be achieved by minimizing spurious outages during minor upsets through improvements in accuracy and effectiveness of process monitoring and control. Use of computers for data acquisition, processing, control and display enable convenient incorporation of these improvements. The plant operability, availability and safety can be enhanced further by judicious implementation of expert systems to aid the operator in plant monitoring and fault diagnosis.

Availability of cheap computing power and innovations in knowledge based systems as operator aids have made development of computer control and application of expert systems for nuclear reactors quite attractive. This paper describes an approach which is envisaged for implementing these concepts in Indian PHWRs.

2. Present C&I Philosophy in Indian PHWRs

In the present nuclear power plants in India, a major portion of control and instrumentation is implemented using conventional hardwired systems. Computer based systems have been developed for some selected applications like data acquisition systems (DAS), control of on-power refuelling machines, channel temperature monitoring etc.

It is now well recognised that use of computer based real time systems in control and instrumentation enable incorporation of advanced control algorithms, better man-machine interface and self checking diagnostic features which result in greater flexibility and enhanced reliability leading to better plant availability and safety. To realise these advantages in future PHWRs, development work on distributed computer control schemes has been initiated.

This improved control system makes available a much larger amount of plant information in DAS computer and there exists a good potential to make use of expert system techniques to aid the operator in efficient perusal of this database. However, the realisation of the same requires considerable work in identification of potential areas, development of suitable knowledge bases, artificial intelligence tools and appropriate operator interface features. Hence a judicious step by step approach as explained in the ensuing paragraphs is envisaged.

3. Need for 'expert' systems to aid the operator

While increased automation of NPP operation and more extensive monitoring of plant processes are resorted to with the aim of a better safety assurance and enhanced plant availability, the diagnosis of faults developing in such complex automated systems becomes a more taxing job for the operator.

Monitoring automated systems and taking proper corrective actions at the instances of failures of automatic controls require the operator to possess even better understanding of the plant processes and their controls than while operating the plant manually. The alarms and other messages coming from the plant DAS during an abnormal operating condition might run into several hundreds per minute. If this large volume of information is presented to the operator without some cuing, the real message may get lost and it defeats the very purpose of increased sophistication in instrumentation. This problem can be largely overcome by means of substantial information reduction that is possible through suppression of low-priority, irrelevant or consequential alarms during plant upset conditions. Identification of plant state by suitable integration of measured values and presentation of this status information in summary form on a plant mimic through CRT would also help the operator fast identify the essence and source of current abnormalities.

However, on-line computer programs incorporating fairly large amount of plant knowledge have the potential of providing far superior assistance to operators especially during plant upset conditions. Computerisation of the knowledge about the correspondence between different sequences of messages in control room and the root causes initiating the plant disturbances has long been recognised to yield a diagnostic aid to the operator at a qualitatively higher level [1,2,3]. Such a knowledge based system, though requiring enormous amount of plant modelling effort, would provide a faster and more objective diagnosis of plant abnormalities. This is expected to reduce the operator's stress in information handling and enable faster corrective steps to restore normalcy. But the limitation of such automatic diagnostic systems, stemming from the finite number of equipment failure modes modelled, requires a cautious approach. While dealing with novel failure modes, the above system, though unable to identify the root cause of the disturbance, should help the operator through structured presentation of plant status and conclusions drawn from the combination of observed symptoms. In any case, the aim of such a diagnostic aid should be to enhance the operator's knowledge about developing faults rather than to substitute for him. The co-operative effort of the human operator and the computerised aid is expected to find a better solution to the diagnosis problem than either of them alone. While the aid can assimilate large volume of messages during abnormalities for which interpretation exists in the knowledge base provided through the programs, there will be substantial enhancement of the operator's ability to concentrate with his own knowledge on the grey areas left.

4. Desirable scope and features of on-line diagnostic aids

The structural connections between various components in a nuclear power plant as well as the various modes of fault propagation amongst them are well understood and a lot of effort has already gone into representing this type of knowledge in the form of fault trees, event trees and signed digraphs [2,4,5]. This favours a 'deep model' approach in building a diagnostic system rather than rule based expert systems dependant on shallow reasoning. Functional analysis of each plant sub-system and its links with adjacent ones would be necessary to arrive at the cause-consequence relations in various forms viz. truth tables, trees or analytical models. These relationships can be coded using a suitable computer language to constitute the knowledge base of the aid. The initiating events considered in such functional analysis are generally selected from those used for plant safety studies and also based on the experience in operating plants. Once the identification of the initiating event by the knowledge based system is complete, the computerised aid can also display the mitigative steps to be taken by operator from the stored files on emergency operating procedures (EOPs). These plant specific EOPs for each of the events are already developed in detail. Though a vast majority of abnormal conditions may be caused by one of these initiating events, the diagnosis of the event from observed symptoms proceeds in stages. Even during this period, the operator actions based on symptoms observed thus far may help reduce the abnormal parameter deviation or accident severity considerably. Also many novel failure modes (or combination of failures) may not be identifiable with any of the modelled events. To enable prompt operator action in such cases, symptom oriented procedures may have to be developed by both design analysis of various scenarios as well as from the knowledge of experienced operators. Suitable interfacing of such shallow knowledge with the deep models discussed above is an area requiring careful study and experimentation.

Expert systems can also be of use in performance monitoring of plant equipments, detection of any degradation and indication of the type of maintenance required. Signature analysis techniques used in vibration monitoring and noise analysis can be extended to more areas where periodically measured performance signatures can be compared with the corresponding normal signatures stored. By trending the performance, a knowledge based system can predict when and what type of maintenance is required to avoid failure.

Development of such systems requires parallel efforts in identification of critical equipment parameters requiring such monitoring, suitable sensor development and a methodology to reliably interpret the deviations in signature patterns.

5. Implementation Plan

Even before the knowledge based systems are implemented as on-line operator aids, there is a substantial improvement in operator convenience achievable by refinements in existing information presentation practice. Computerised DAS permits moving away from the conventional 'one sensor , one signal' philosophy and present the process data in a form giving more directly the information of operator's current interest. Data reduction by automatic suppression of superfluous information is also achievable through suitable logics. Experience elsewhere [6,7] has shown that prioritization and hierarchical presentation of alarms provides a satisfactory solution to the problems of handling 'alarm bursts'. These steps of information pre-processing are expected to be complete before attempting more advanced diagnostic aids.

The methodology for knowledge acquisition and reasoning in the context of on-line diagnostic aids to process plants has already attracted enough attention of the industry as well as the academic community [2,4,5]. The attempts to have the fault propagation models represented by cause-consequence trees in the STAR project of F.R.G. and the Disturbance Analysis System of U.S. EPRI have set a trend among NPP diagnostic aids [2,8].

Like in any knowledge based system construction, the primary steps to be taken are thorough enumeration of facts, rules and diagnostic principles through which the root causes for abnormalities are to be inferred. A major part of the knowledge base to be provided would be the plant topology i.e. the various components, their interconnections, normal configurations for each plant operating mode etc. The functional knowledge or the behaviour of components and systems following malfunctions is another important constituent. This part of the knowledge has to be made available through construction of cause-consequence diagrams as well as compact analytical models (where quantitative information is of interest.) While all this would constitute the model data base, the current values of all the process parameters made available through DAS computer would constitute the dynamic data base. Periodic scanning of this dynamic data base to check for any parameter crossing the set thresholds would enable record all symptoms of disturbances and initiate diagnostic procedures. The diagnostic rules operating on the models are to be formulated as much exhaustively as possible. In this respect, the lessons learnt by various attempts to make similar diagnostic aids along with their varied and distinctive features in implementation are expected to help us make correct choice in the methodological details as we go ahead with the job.

The demonstration of the feasibility of knowledge based systems for automatic fault diagnosis and the development of the methodology in all its details would be possible through construction of knowledge base pertaining to a single sub-system of the plant (e.g. primary heat transport, boiler or turbine-generator system). To start with, the plant knowledge can be represented in any one of the standard knowledge representation languages (e.g.Prolog) on a modest hardware that can be linked to DAS computer. The system design should provide for convenient addition of further modules of knowledge base so that it can smoothly evolve into a comprehensive plant diagnostic aid. At every stage of evolution of such a system, the correctness of plant behaviour representation has to be well tested with a part task or full scope training simulator which provides for all the malfunctions considered. This phase of testing would not only verify the correctness of logical, causal and temporal connections between causes and consequences represented but also help assess the operator reaction to the automatic aids. The operator feedback on the type of information needed to complement his own knowledge and diagnostic ability would be one of the most vital inputs for decisions on presentation of information.

Even after this aid is proved satisfactory during all phases of simulator testing, it has to be on probation for a period of time while connected to plant DAS and facing all disturbed plant conditions. The recorded output of the aid during all the abnormalities need to be analysed to prove that it doesn't give any wrong or misleading message even during extraordinary process circumstances (that may not have been modelled in simulator). The display of the diagnostic aid may be kept in the shift supervisor's room during this period. Only after ensuring that probability of erroneous diagnosis by such an aid is several orders of magnitude smaller compared to that of average human operators it can be released for use in plant control room.

The efforts in mathematical modelling for control system design, evaluation of plant operational transients, modelling for training simulator needs, accident analysis alongwith the construction and analysis of fault trees for the purpose of safety assurance have already generated enough experienced professionals in these areas. Refinements of these techniques to construct fault propagation models for a vast majority of equipment failures of operational interest, though a voluminous job, can hence be attempted with confidence. The critical job in this attempt is to communicate with process engineers involved in design or operation, acquire the process data and provide a well structured representation of this data for efficient use by the computer. This would require a band of dedicated system analysts and a good coordination with design as well as plant operating staff.

The operators in existing plants can opine from the past experience as well as from their knowledge on emergency operation, the stressful areas and situations where automatic aids could be of help to him. The format of the results of automatic diagnosis has to be designed to suit his convenience. Also, the type and amount of information to be made available on operator's request on a conversational mode have to be planned taking his opinion into account.

6. Conclusions

It is well recognised that development of knowledge based systems involves knowledge acquisition, computer representation and validation for different plant systems covering several tens of man-years of effort by the knowledge engineers. A significant part of this job can start off with the availability of a full scope simulator to test the expert system behaviour.

Deployment of expert systems is envisaged as a natural extension of computer control concepts and it is felt that work in the two areas should proceed concurrently. The basic capability to implement expert systems i.e. acquisition and pre-processing of required plant parameters shall be built-in into the distributed computer control system and expert systems shall be connected in the network whenever sufficiently validated expert systems are developed.

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EXPERT SYSTEMS IN NUCLEAR POWER PLANTS — HOW SOON CAN THEY BE IMPLEMENTED?

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Abstract

Real-time Expert Systems (RTES) techniques will probably be implemented in an eventual nuclear power station in Israel.
In preparation for this goal, expertise in this field can be developed locally and maintained by applying these techniques in similar environment, such as the chemical processing industry. An example of demonstration RTES is described, in which spread-sheet software was used as a shell for a fault analyzing operator-aid in low-level radioactive wastewater evaporation facility. The status of RTES implementation in the chemical industry is discussed, and it seems that the needed development effort is higher than expected. The only reported major industry/university project had to invest 12 man-years in actual plant implementation of RTES for fault analyzing of relatively simple chemical reactor operation. A major part of the effort was concerned with the simulation, checking and verification of the RTES software prior to it's installation in the control room. The implication for nuclear power plant use, with it's much higher complexity and stringent safety requirements and regulations, is that full scale RTES development and implementation will be both expensive and slow. The perceived increase in the safety and plant availability seems, however, attractive enough to warrant continued efforts in RTES development, even only for sub-systems, with speciál attention to the problem of verification and validation needed for the regulatory authorities certification.

Introduction

There are no nuclear power reactors in Israel, but the Israeli Atomic Energy Commission is convinced that there will be a need for such power plants in addition to the oil and coal burning power stations. At that time, there would be a for Israeli team of engineers and scientists well aware of current and advanced technologies, so as to be able to evaluate vendor proposals. One of the areas that is thought to be important, is the use of advanced concepts in the power plant control system, from the operational, safety and public acceptance aspects.

The TMI and other well publicized plant accidents resulting partly from human operators judgment errors, have pointed out the need for better operator training and readily available real-time guidance and help. The newly emergent field of real-time expert systems (RTES) is seen as one way of providing an answer for these needs, and it was in this context that the activity described in this presentation was started in 1984.

How to become an expert in "expert systems"?

As in any other new technology, one should be acquainted with previous and current activities of other organizations, and try to learn from their experience so as to concentrate on the more promising directions. But that in itself is not enough, and one should get "hands-on" expertise by applying the technology to an existing problem, preferably on a small scale. In our case, there was also the need to maintain the expertise for indefinite time until the need for power reactor application will appear, so there must be a non-nuclear area in which this expertise can be utilized with apparent benefits.

If we consider the perceived risk to the public from industrial operations, the chemical and petrochemical industries are high in the list, and the Bhopal disaster is an obvious example. With these considerations in mind we began our involvement in the real-time expert systems field, trying to learn about the application of such systems in the industrial process control area, and to develop a small scale application of our own.

Development of a demonstration real-time expert system as an operator aid

The evaporator used for volume reduction of the low-level radioactive wastewater at the Nuclear Research Center - Negev (NRCN) was chosen for testing the problems involved in applying real-time expert systems in actual operating plant.(1) As no special RTES hardware and software were available to us at that time (1984), it was decided to develop the system on a PC, using integrated spread-sheet software. The basic approach chosen was to define a normal operating range for each of the analog inputs from various sensors used for the evaporation control and monitoring system, and to learn from the senior operating staff, who had many years of experience with that system, what combination of symptoms (low, normal or high readings) at a definite status would indicate a specific fault. About 400 "rules" were identified, based on the 24 analog instrument readings and 14 on/off states of valves and pumps.

As sometimes not all the symptoms were present after a particular fault had occurred, or a counter-indicating symptom was present, "low" probability was assigned for each rule in these cases.

These rules were implemented on the PC, using the spread-sheet structure, and once the operator indicated the existing symptoms on a schematic display of the system elements and instruments (generated by the spreadsheet software and pointed at by use of the cursor and by top line menu), the program displayed a list of faults in a descending order of probabilities, in about 10 seconds. This was done using the query features of the software as an inference engine .An explanation facility, reciting the rules used in generating this list was also provided at the operator choice.

The operator involvement in pointing out the symptoms can be eliminated once a real-time data collection software transfers automatically the information directly from the multiplexing hardware into the spread-sheet structure.

A specific example of a fault identification procedure is given in the appendix.

Survey of current experience in applying expert systems in the chemical industry process control

From a recent survey of expert systems application in the chemical industry, (2) it can be seen that the most prevalent application of ES is in diagnosis, with prescription running second. More equipment than process applications were reported (Fig. 1). The computer systems used are mostly PC-based systems (Fig. 2). Although many RTES are in development or even in plant use , only one large scale experimental industrial implementation was reported extensively in the literature, and this is the FALCON (Fault Analyzer CONsultant) project (3). The demonstration project, carried out jointly by University of Delaware and the Dupont and Foxboro companies, was

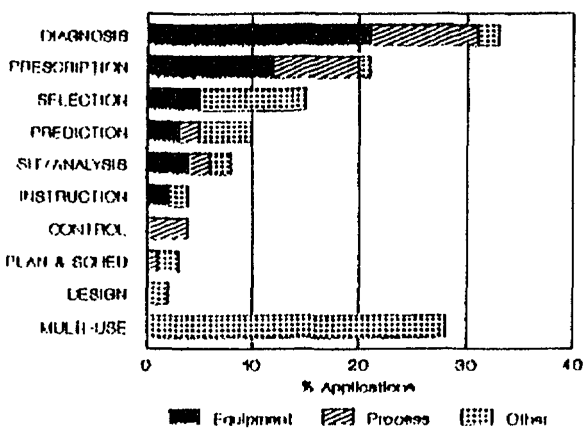


FIG.1. Expert system roles (from Ref. [2]).

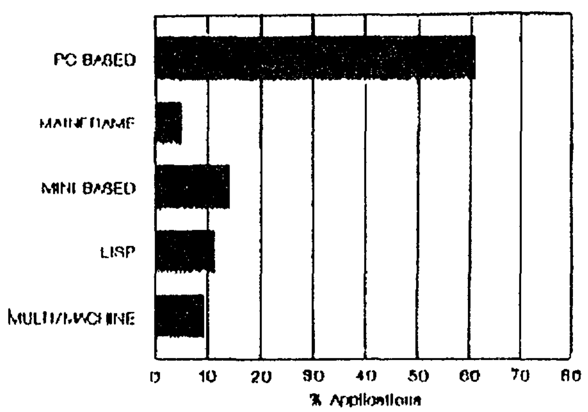


FIG.2. Expert system hardware (from Ref. [2]).

expected to develop an expert system to aid the operators in detecting and identifying process faults of adipic acid production reactor.

The project began in 1983 and the RTES had been implemented in the plant by the end of 1987. Although the reactor system is relatively simple, having 30 process variables, it took about 12 man-years to develop the RTES. The main items needed were the expert system software, using about 650 rules and mass and energy equations, dynamic simulation program of the plant behavior for the RTES checking and verification, and on-line effective man-machine interface using touch-screen. Figures 3-5 indicate of the basic approach and solutions.

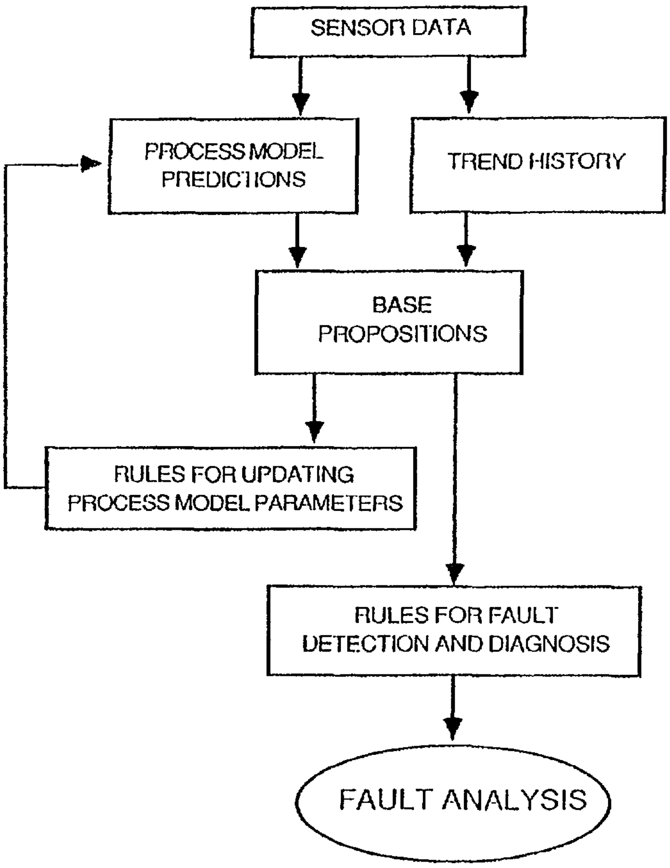


FIG.3. FALCON knowledge base inference procedure (from Ref. [4]).

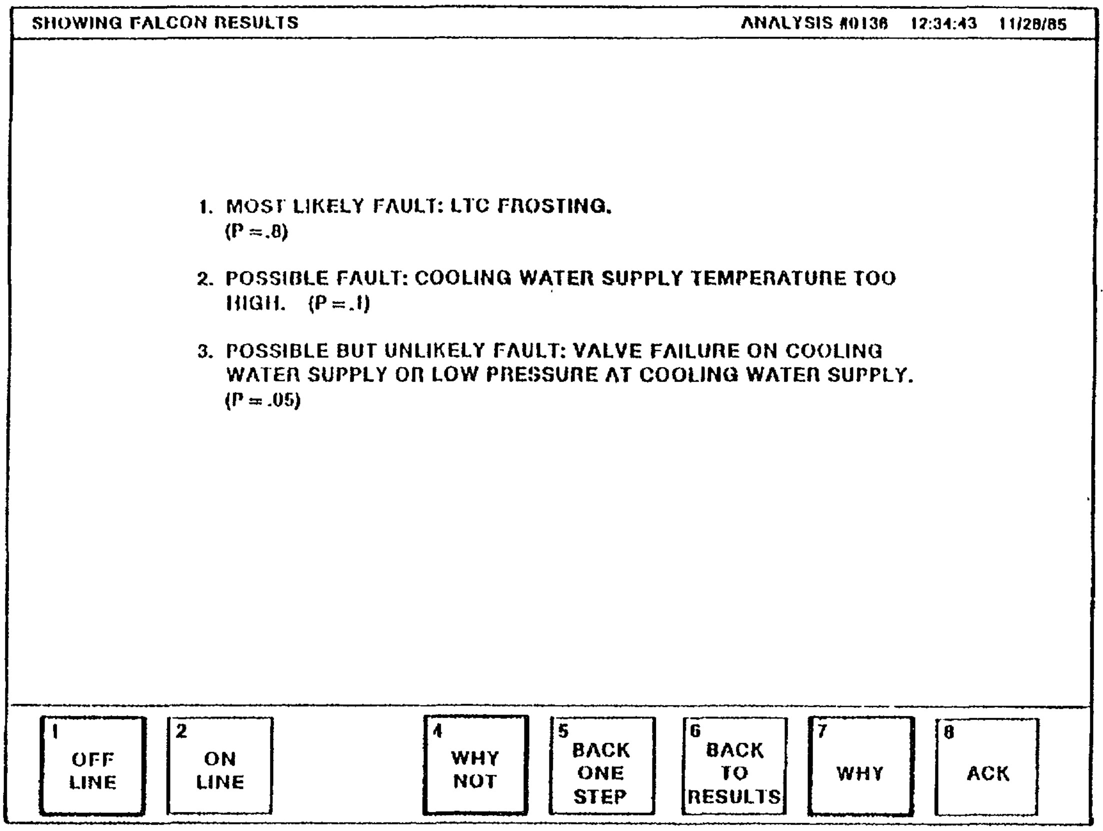


FIG.4. CRT display layout (button 3 is unassigned) (from Ref. [3]).

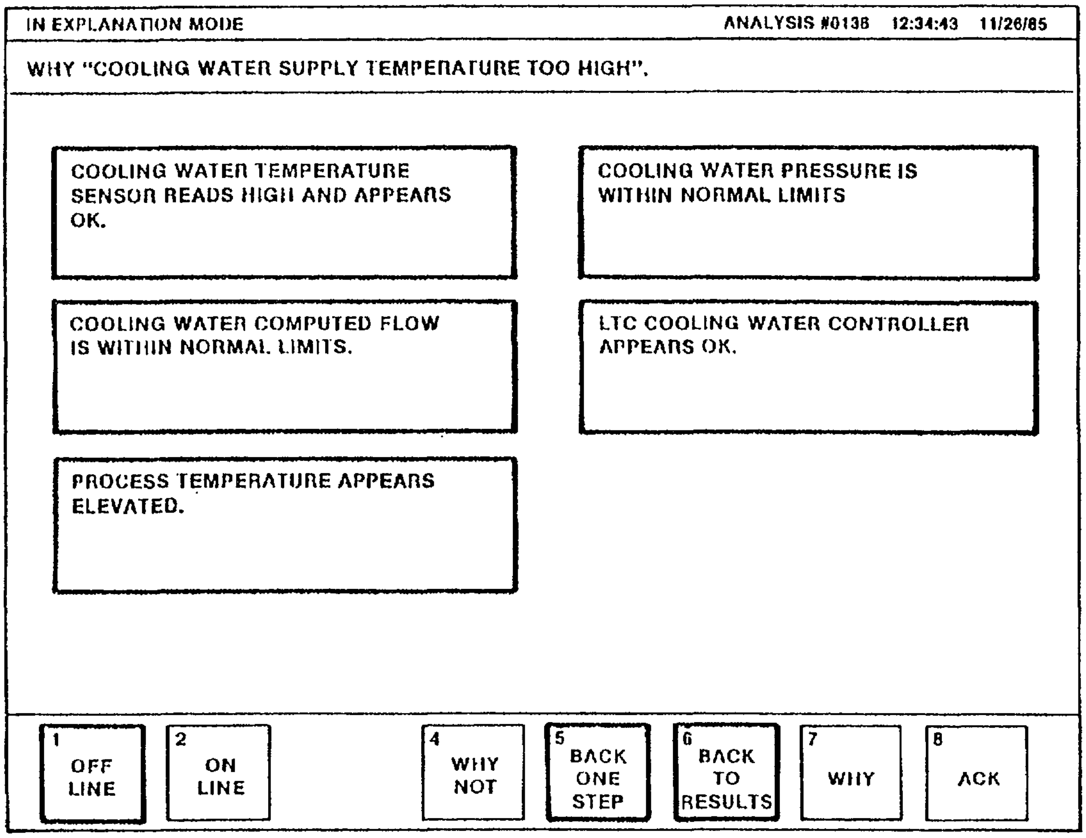


FIG.5. Explanation display (from Ref. [3]).

The application of ES in nuclear power reactors

The potential uses of expert systems as an operator aids in power reactor plants was appreciated long time ago. An example of early intelligent display is "Wolff diagram" in which various analog inputs are normalized to unit circle and displayed as spokes of a wheel. Abnormal situations are diagnosed using fault specific circle distortion pattern, recognized by an experienced operator. An nuclear power reactor example is shown in Fig 6.

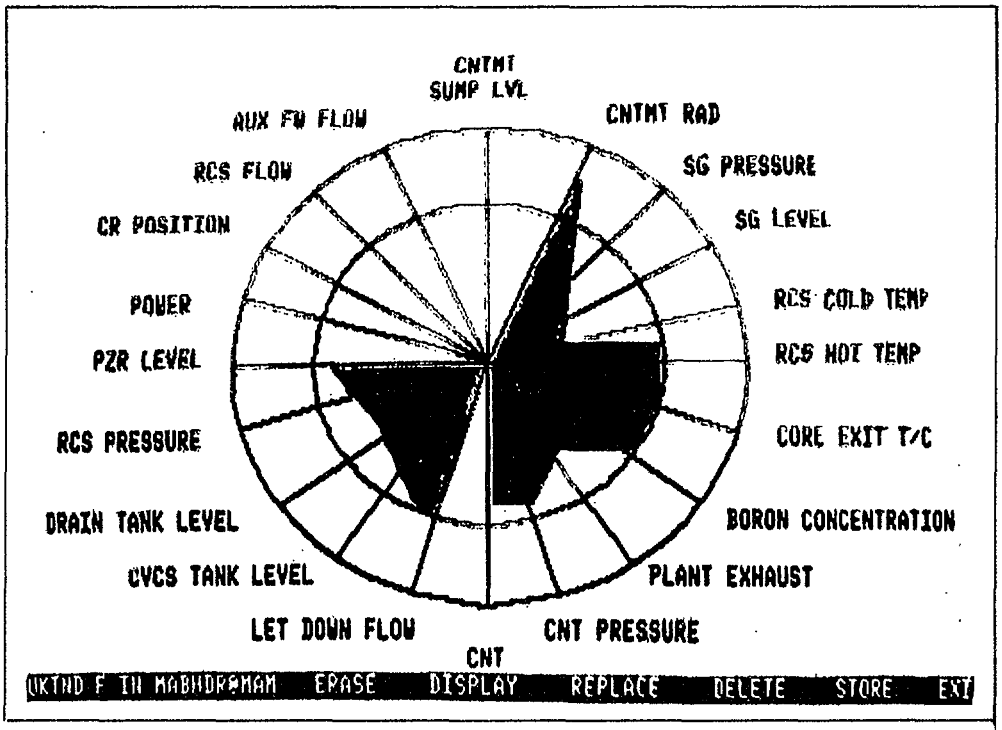


FIG.6.

Much effort was made in this field, and it was summarized in two recent conferences - Artificial Intelligence and other Innovative Computer Applications in the Nuclear Industry, organized by the American Nuclear Society in Snowbird, Utah in August 1987, and the IAEA international conference on Man-Machine Interface in the Nuclear Industry, Tokyo, February 1988. However, there is a general feeling of concern about using RTES in safety related functions in nuclear power reactor, as there are unresolved program validation issues, and because of regulatory and certification problems.

Problems in applying RTES to nuclear power reactors

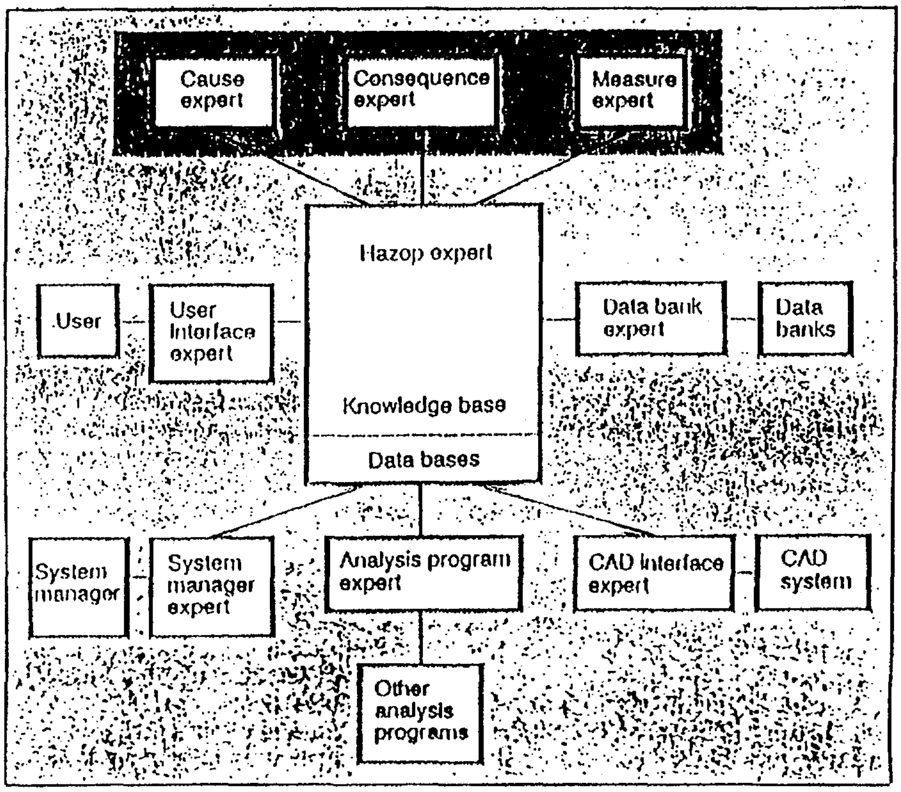
There is an agreement that although ES are not a solution for all problems, it should be applied to problems that satisfy certain conditions. These conditions are shown in Fig 7. Nuclear power reactors satisfy these criteria, but there is a question wether the size and complexity of the systems in nuclear power reactors do not prohibit the actual development of RTES for this purpose. One consideration is the size of the team needed for such effort. A recent news-item about the development of ES for chemical plant safety in Finland identifies the need for 9 experts in different fields (Fig 8) The FALCON project had to apply 12 man-years of highly trained experts and researchers to analyze a chemical reactor with 30 process variables and a minimum set of 40 faults. Even assuming that a similar project will demand less effort as the initial development was done, it will take several man-years to implement. So the question of how soon RTES can be applied in the much more complex systems of nuclear power reactor is very important, even without considering the special safety considerations of verification and validation of such ES to meet regulatory requirements.

Expert systems are most apt to be appropriate when the following criteria are met:

- Symbolic logic is required (such as IF-THEN rules).
- The results are time-critical; they must be obtained in less time than it takes to contact an expert for a solution.
- The problem is moderately complex: it would take an expert more than two minutes and less than two days to solve it.
- Formal mathematical specifications are not feasible, but experts can solve the problem.
- The solution of the problem has a high value.
- The expertise to solve the problem exists, but is scarce.

From: R.S. Shirley: The practicalities of expert systems (For Process Control Applications). Private communication.

FIG.7.



from: ref. 7

FIG.8.

One way of overcoming such obstacles is to try to develop RTES for sub-systems, and to integrate these "modules" into a system once each is tested and checked in actual use. The operators will have to continue to relay on the current operating manuals and procedures, with the RTES providing some help in some subsystems, especially for fault and probable cause identification.

Appendix: Real-time fault analysis example

In Fig A1 a schematic diagram of a wastewater evaporation facility is shown. In Fig A2 an example of the logic diagram used in developing the expert system rules is shown. A conclusion, "feed system fault" is reached by the combined occurrence of the "high" value of the (calculated) standard deviation of the feed pump electrical current

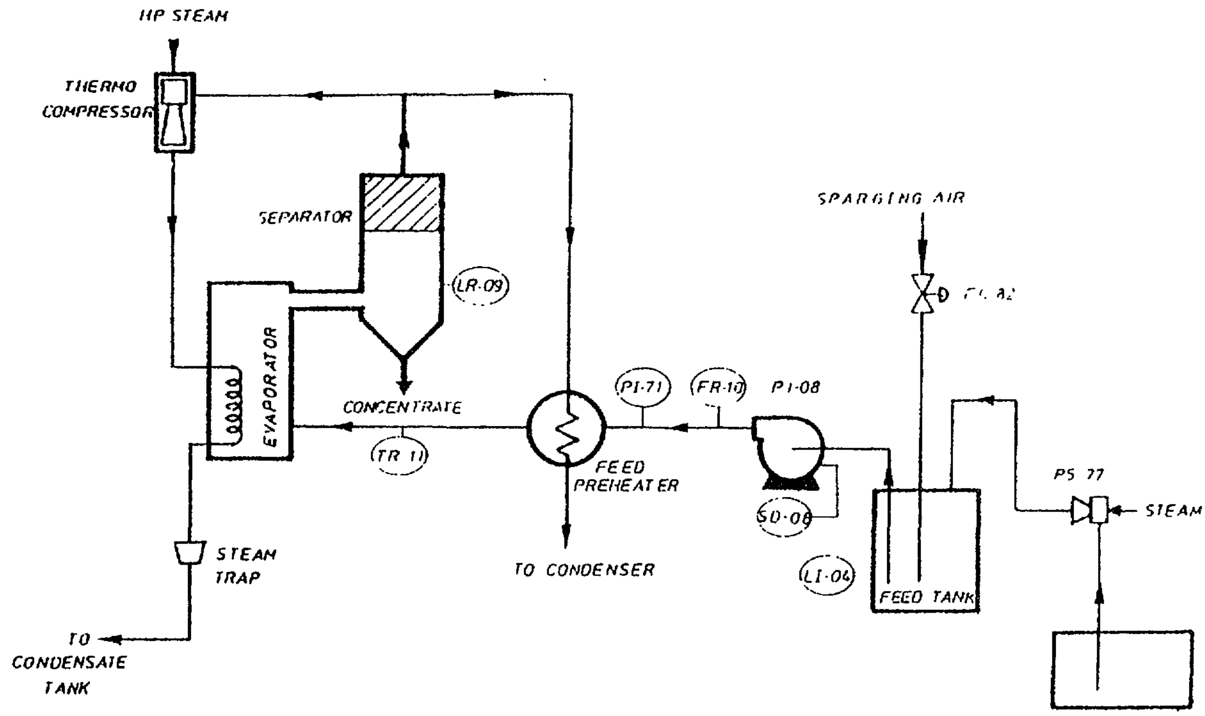

HP STEAM
THERMO COMPRESSOR
SEPARATOR
LR-09
EVAPORATOR
CONCENTRATE
TR 11
FEED PREHEATER
TO CONDENSER
PI-71
FR-10
P1-08
SD-08
LI-04
FEED TANK
SPARGING AIR
PS 77
STEAM
STEAM TRAP
TO CONDENSATE TANK

FIG.A1.

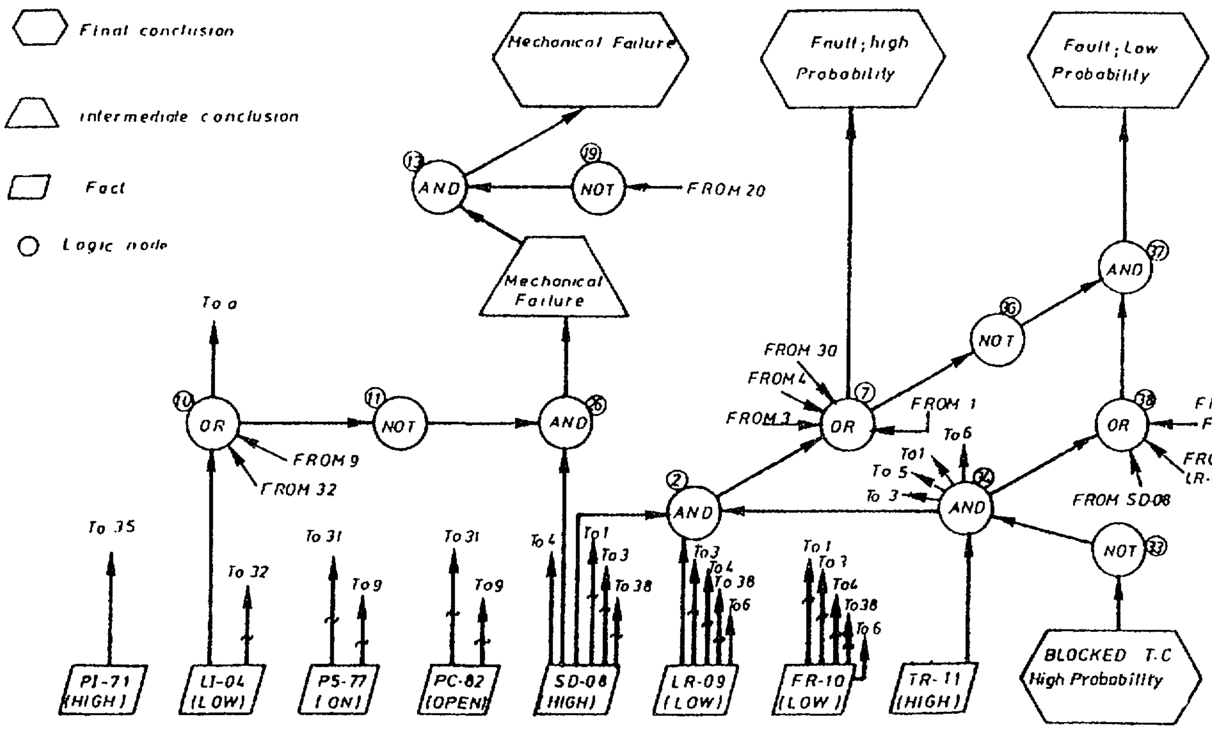

Final conclusion
intermediate conclusion
Fact
Logic node
Mechanical Failure
Fault; high Probability
Fault; Low Probability
AND
NOT
FROM 20
Mechanical Failure
To a
OR
NOT
AND
FROM 9
FROM 32
FROM 30
FROM 4
FROM 3
FROM 1
FROM FR-10
FROM LR-09
FROM SD-08
PI-71 (HIGH)
LI-04 (LOW)
PS-77 (ON)
PC-82 (OPEN)
SD-08 (HIGH)
LR-09 (LOW)
FR-10 (LOW)
TR-11 (HIGH)
BLOCKED T.C High Probability

FIG.A2.

consumption, "low" value of the liquid level in the separator, and "high" value of feed temperature. The "high probability" assignment is caused by the fact that the conclusion, "blocked thermo-compression high probability" conclusion was not reached. The diagnosis of the "pump mechanical failure" probable cause was reached by the reasoning that the other causes for the pump erratic behavior is not air drawn into it as result low liquid level in the feed tank, or open sparging air valve. Thus the operator is informed that there is a "high probability" that a fault exists in the evaporation feed sub-system, and the probable cause of the fault is some mechanical failure in the pump which behaves erratically.

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NEURAL NETWORKS — POTENTIAL APPLICATION IN THE NUCLEAR INDUSTRY: PRELIMINARY COMMENTS AND REFLECTIONS

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Abstract

Neural Networks (or Parallel Distributed Processing, Connectionist Models, Layered Self-Adaptive Systems, Self-Organizing Systems) are an emerging technology which is perceived to have potential for solving complex computation problems which cannot be solved by standard computational methods. A neural network is a system composed of many simple processing elements operating in parallel whose function is determined by network structure, connection strengths and the processing performed at computing elements. Neural networks' architectures are inspired by the architecture of biological nervous systems which use a large number of simple processing elements, the neurons, operating in parallel to obtain high computation rates. Neural networks can learn and can be taught. They learn from examples, adjusting internal parameters to match the examples. Neural networks can be considered as a different approach to computation. In the future, neural networks' modules may possibly be embedded in expert systems and conventional applications. Thus, the three information processing technologies, procedural languages, expert system and neural networks, running on the same hardware, can be used either separately, to solve specific types of problems, or in various combinations where each brings its unique capabilities to the overall application.

In the 1980's, there has been an extraordinary growth of interest in neural models and their computational capabilities. This occurred because of the convergence of several factors, recent advances in neurobiology, widespread availability of powerful computers, growing interest in parallel computation, new concepts in the mathematics and simulation of neural models. In 1986, only one neural network modelling tool was available; now there are about twenty offered commercially by various companies in North America. Large programs on neural networks have been recently launched in the United States, Europe and Japan, and some applications are already coming out of the labs into commercial use.

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1. Introduction

Neural Networks are an emerging technology which is perceived to have potential for solving complex computation problems which cannot be solved by standard computational methods. One such example is the inverse kinematics problem which is considered to be the most difficult problem in robotics.

A large number of research and development evaluations and programs on neural networks are being conducted by major institutions and companies.

A study on the applicability of neural networks has been recently completed by DARPA, the Defense Advanced Research Projects Administration in the United States. The report will probably be released before the end of 1988.

As reported at the IAEA meeting on the use of expert systems in nuclear safety, Vienna, October 17-21, 1988, AECL (Atomic Energy of Canada Limited) is monitoring developments in neural networks and will review the possibility of using them for parts of the Operator Companion expert system, particularly in dealing with improved alarm annunciation.

In 1986, only one neural network modelling tool was available; now there are about twenty offered commercially by various companies in North America. Some of these products are designed only for education purposes; others may be used to create applications and some are user products. The cost of these software packages, most of them for PC-AT's and PS-2's, varies between about $500 and $60,000. Some of these softwares run under the Microsoft Windows environment such as <u>ANSim</u> (Artificial Neural Systems Simulation) of SAIC, the Science Applications International Company of San Diego, Ca., and <u>AI-NET</u> of the AIWare Company of Cleveland, Ohio. A Canadian company is offering a neural network learning tool for $250.

2. <u>What is a Neural Network?</u>

The fundamental element of any neural network is <u>the processing element</u>. Basically, the neural network uses processing elements and weighted connections, both very loosely analogous to neurons and synapses in the brain. <u>Synapses</u> are the regions of specialized interconnection and communication between neurons. It is believed that the ease of these interconnections and communications in the brain is connected with learning.

A neural network consists of "<u>layers</u>", linear arrays of processing elements grouped together. Typically data is applied to an <u>input layer</u>. Connections transfer information from the input layer to one or several "<u>hidden" layers</u> from which connections lead to the <u>output layer</u>. This "<u>connectionism</u>", a completely different approach to computation, is characterized mathematically by <u>large matrices</u>, the connection matrices, that relate input, hidden and output layers. Instead of representing information in rules, as in expert systems, the information is stored, as it were, in the distribution of weights in the connection matrix. <u>Neural networks can learn</u>, and can be taught. During the learning phase, exponential functions of the connection matrices must be computed many times. Pairs of inputs and outputs are applied to the network. These pairs of data are used to teach the network. The network learns by adjusting or adapting the strengths of the connections between processing elements. The method used for doing the adaptation is called the "learning rule". Several "learning rules", or paradigms, have been proposed and tried in neural network development. The architecture of the network consists of the number of processing elements, the way connections are made and the neurodynamics (sum, transfer and learning rules). When the network is given some input, the output obtained depends on the values of the connections. During the learning phase, the connections are adjusted so that the difference between the network output and the desired output is minimized.

Neural networks learn from examples, adjusting internal parameters to match the examples. Neural networks may be viewed as a statistically-based mapping technology. Neural networks can produce continuous mappings from an n-dimensional space to an m-dimensional space.

Neural networks can be considered as a different approach to computation. While classical procedural languages require a detailed statement of the solution to a problem, and expert systems require a symbolic statement of the solution, a neural network requires a statistically valid representation of the problem mapping to produce a solution.

A neural network uses a large number of simple cheap processing elements to perform operations rapidly in parallel. A conventional computer simulation uses a small number of complex expensive processing elements to perform operations slowly in serial.

3. Engineering Applications and Potential Applications in the Nuclear Industry

Neural networks are typically used to develop a relationship between several inputs and several outputs. Neural networks can be used to compute the inverse of a mathematical function.

Several Examples of Applications:

- Complex pattern recognition
- Noise analysis
- Modelling including financial and economic modelling
- Forecasting

Examples of Computation of the Inverse of a Mathematical Function:

- The inverse kinematics problem in real time (robotics)
- CAD/CAM modelling of three dimensional surfaces
- Structural design
- Sensor processing

Neural networks have been used to analyse electrocardiograms, which may be considered some type of noise analysis.

Neural Networks and Expert Systems

Expert systems have been developed for the diagnosis of various engines: aircraft engines, automobile engines, diesel engines. In these diagnostic expert systems there is a need to interpret certain signals such as engine noise. Different signals of the engine may mean either a good-running engine or a faulty one. Now expert humans may be part of the loop to interpret these engine signals. Neural networks modules could be trained to differentiate between these signals and help remove the human from the loop and thus automate the system.

Other Applications

We may mention the famous NetTalk experiments at John Hopkins University where a neural network was trained to take text and convert it directly to speech. This may be considered as mapping from text space to speech space.

Another area of possible military applications is the identification of a particular type of object in a picture.

Finally, let us say a few words about the application mentioned above, the inverse kinematics problem in real time (robotics).

In conventional manipulator control in robotics there is the problem of how should a robotic arm be oriented when there are redundant degrees of freedom. The Cartesian end point position is used to convert that position, say x, y, into joint coordinate angles, say θ_1, θ_2, and then use a control algorithm to move all the point angles to their desired positions. The inverse transformations, from the endpoint to the angles of the robotic arm, are made up of many transcendental functions that have to be derived and programmed, requiring extensive computation. A neural network can avoid the derivation programming and computation of the transformation equations, since it can learn from examples only.

It is clear from the listing of the above potential applications, and the few comments about some of them, that there is a wide scope of potential applications for neural network in the nuclear industry. Comprehensive surveys and evaluations should be conducted to identify and specify the applications that would be feasible and useful.

4. Conclusion

In view of the above, one of the recommendations of the Technical Committee Meeting on the Use of Expert Systems in Nuclear Safety, Vienna, 17-21 October 1988, is the following:

"To monitor and follow-up developments on the potential uses of Neural Networks in the nuclear industry and take necessary steps to disseminate relevant information when justified by developments".

It is clear that IAEA member states cannot just monitor and follow-up developments. They should take, and some of them are taking, active steps to explore the potential of using neural nets in the nuclear industry and translate promising applications into actual projects that could then be monitored and folowed-up by the IAEA.

APPENDIX

(a) A Short List of Books on Neural Network

(1) Anderson, J.A., and Rosenfeld, E., Neurocomputing: Foundations of Research, MIT Press, 1988.

(2) McClleland, J.L., and Rumelhart, D.E., Explorations in Parallel Distributed Processing - A Handbook of Models, Programs and Exercises, MIT Press, Cambridge, Mass., 1988.

(3) McGregor, R.J., Neural and Brain Modeling, San Diego, CA., Academic Press, 1987.

(4) Arbib, M.A., Brains, Machines and Mathematics, Second Edition, New York, N.Y., Springer-Verlag, 1987.

(5) Rumelhart, D.E., and McClleland, J.L., Parallel Distributed Processing, 2 volumes, MIT Press, 1986.

(b) A Short List of Neural Network Software Available Commercially

(1) Neural Works, from Neural Ware Company, Sewickley, Pa. $495 on PC and PS2. Version 2.0, about to be released: $995 on PC and $2995 on Sun workstations. A stripped-down version of the original Neural Works called Neural Works Explorer, will be available for $199 on the PC and $795 on the Sun.

(2) ANSim from SAIC, San Diego, Calif. As a stand-alone, the network modelling tool is $495 and requires a PC-AT,EGA, Windows and a mouse.

(3) AI-NET from AI Ware Inc., Cleveland, Ohio $1500, windows-based.

(4) NDS 1000.SP from the Nestor Corp., Providence, R.I., $7500, requires an AT, EGA and a mouse, N500: $500.

(5) Awareness from Neural Systems, Vancouver, Canada, $250, PC:PS2, educational tool.

USEFULNESS OF MODEL-DRIVEN APPROACHES TO KNOWLEDGE ACQUISITION IN TECHNICAL DOMAINS

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Abstract

Knowledge acquisition is often considered as a major bottleneck in furthering the use of expert systems technology. Opinions on causes and solutions regarding this bottleneck diverge, however. In addition, current methods of knowledge acquisition and expert system development strongly deviate from good standard software engineering practices as well as from the ways knowledge is analysed and modelled in most other technical domains. An approach is advocated in which expert system development is seen as an activity focussing on the formal modelling of specific types of knowledge. In this context, a short survey of discussions relating to the knowledge acquisition bottleneck is given and some research suggestions are brought forward that seem promising, also in reducing the gap with methods that have proved valuable in the technical and physical sciences.

[Keywords: Expert system bases; methods/techniques for expert system development]

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1. Introduction

It is generally recognized that knowledge acquisition -- the elicitation and interpretation of (often verbal) data from experts -- is an essential activity in constructing expert systems [e.g.,1,2]. Moreover, it is often seen as the major bottleneck in the further development and industry-wide application of expert systems technology [3]. Views diverge, however, on which aspect this bottleneck is to be attributed to and on how it could be solved [1-6]. In this paper we will approach this question from the angle of experiences gathered in the technical and physical sciences (Sec.2).

In this perspective, we will discuss a recent methodology [5,6] (Sec. 3) and put forward some suggestions (Sec.4), of a programmatic and research nature, for addressing some of the problems associated with the so-called knowledge acquisition bottleneck.

2. Knowledge acquisition: mining or modelling?

Knowledge acquisition is commonly thought to be composed of two different activities. First, there is the elicitation phase in which the necessary data are collected through interviewing the expert, consulting written sources, making observations during task performance, obtaining think-aloud protocols, etcetera. Second, we have the stage of analysis, in which the collected data are interpreted, organized and transformed into a framework that can serve as the design basis for a knowledge-based system (KBS) or already into a prototype expert system itself. Note that these two stages need not be clearly separated in time and are often intertwined, because of their iterative nature.

One possible view is to attribute the difficulties associated with knowledge acquisition to the problems of elicitation. Upon acquiring expertise in a domain, much of the knowledge and skill will be "compiled out" and become implicit. Accordingly, it becomes difficult for both expert and knowledge engineer to get mental access to this expertise and to articulate what is going on. This leads to a "mining" picture [3,4] of knowledge acquisition, its essence lying in "extracting" [4] or "mining those jewels out" [3] of the expert's head. In this view, progress is to be made by expanding the cognitive techniques for knowledge elicitation (e.g., repertory grid analysis) and making them available in automated tools. Without denying the importance of this, it is obvious that this approach is generally very remote from what is customary in technical domains like nuclear engineering. Not the exploration of the inner depths of the expert or psychological plausibility, but the simulation of reliable knowledge and controlled reasoning is what counts in supporting the safe operation of complex devices such as nuclear facilities by means of expert systems. Thus, a mining approach might inhibit the diffusion of expert systems technology as a standard application software tool (like numerical simulation is) in these fields.

An alternative view is to associate the knowledge acquisition bottleneck with the complexities of interpretation and analysis. Even a few short interviews of an expert generate a wealth of data, and a knowledge engineer often has a hard time in capturing their meaning, distinguishing what is relevant and what is not, delineating what is "solid" knowledge and what is heuristic, and generally in bringing organization into the data. Here, knowledge acquisition is basically seen as the problem of the modelling of expertise, the latter being considered as adequate problem-solving behaviour [1,5,6]. The notion of a model is used here in a sense quite close to the concept one encounters in the natural sciences: an abstract and idealized description of some phenomenon on the basis of which one is able to make empirically verifiable predictions. This is an attractive feature of the modelling approach from the viewpoint taken in this paper. Progress in the modelling view of knowledge acquisition consists in finding and validating theories, languages and tools able to describe the functional aspects of classes of problem solving behaviour. Nevertheless, concerning the methods to be used in the modelling of expertise there are still many open questions. At present, they usually show not much resemblance with the modelling methods that are standard (or even sacrosanct) in technical-scientific fields.

This latter point may be elucidated by considering in some detail the method that still dominates practical expert systems building: rapid prototyping. Here, the elicited data are effectively analysed by mapping them into a prototype implementation of the knowledge base (in a moderated form this method is also advocated in the well-known handbook [2]). Insofar as one may speak of modelling, the maxim here is: the system is the model. Various observations may be made regarding this situation. First, a direct mapping is constructed from data to code. This is in remarkable contrast with basic principles of software engineering (which on their turn may be seen as a specialization of good project management) that recommend a development process in clearly distinct stages of analysis, design and, ultimately, implementation. Also in other engineering disciplines one would hardly think of building a computer simulation on top of empirical data without several intermediate steps of formal theoretical analysis. Second, prototyping is necessarily implementation-driven, implying that the modelling of knowledge must be tailored towards the implementation constructs available in the selected coding language (e.g., expert system shell). This is a questionable approach from the cognitive point of view. At the same time it is also far from modelling in the exact sciences, where decisions about model elements (e.g., differential equations) are not made dependent on implementation decisions. Furthermore, the level of abstraction employed in prototyping is usually low, whereas in the exact sciences it is generally high and even has the historical tendency to increase due to ongoing mathematization. These points -- lack of phased and structured development, implementation-drivenness, low abstraction level -- are, given the experience of long-standing technical-scientific disciplines, in our opinion sufficient to argue that the current prototyping methods will not be central in future, more matured knowledge engineering.

The above argument yields what we see as important requirements or desiderata respecting knowledge acquisition methodologies, as can be derived from experiences of older branches of engineering, namely:

(i) a distance between data and implementation mediated by theoretical analysis;

(ii) ability to deal with abstract concepts that may not be apparent from the empirically observed data;

(iii) generation of a model simulating the factual and inferential knowledge regarding the system or task at hand, in a controlled and verifiable manner.

To this we may add another desideratum that is difficult to meet in young sciences (we recall that physics and astronomy needed some centuries for this):

(iv) concepts, models and theories should be expressible in a precise and appropriate mathematical apparatus [7,8].

In the next section we will give a short account of one methodology of interest in that it partially meets the requirements formulated above. Subsequently, we will tentatively consider the feasibility of mathematization of such a methodology.

3. The structured model-driven methodology KADS

KADS [5,6,9] is an acronym for a knowledge-engineering methodology developed in the framework of the European ESPRIT programme (project P1098). Its main characteristics that distinguish KADS from implementation-driven prototyping are:

- it advocates a structured and phased KBS lifecycle approach, similar to conventional software engineering;
- each phase is considered as a modelling activity at a relatively high level of abstraction.

This is depicted in Fig. 1.

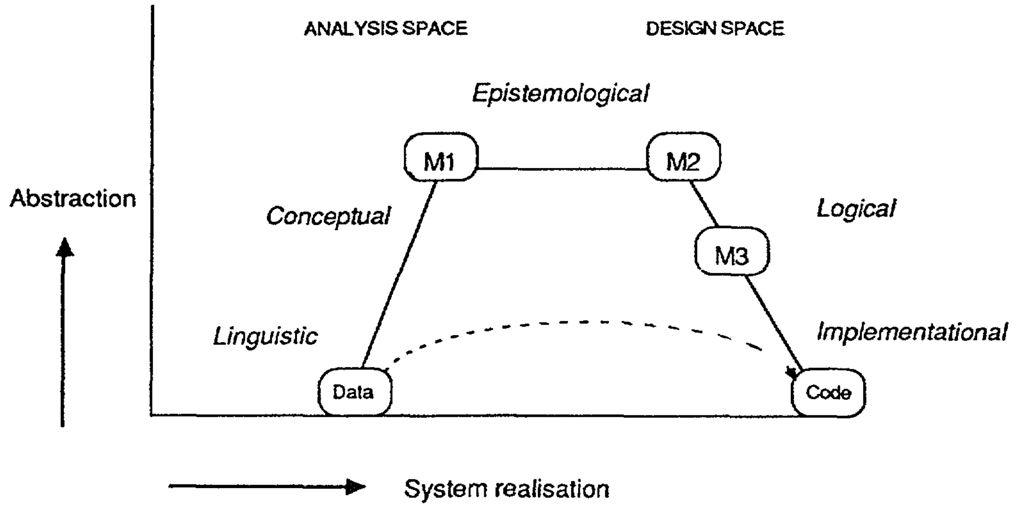

Fig. 1: Models and the development of expert systems (from [6]).

The dotted line in Fig. 1 denotes the development trajectory of current prototyping. The solid line indicates the KADS trajectory, from the underlying data passing through M1 (conceptual model) via M2 (global design model) and M3 (detailed design) to the final artefact (the KBS code). M1, M2 and M3 are also directly coupled to separate stages in KBS software project management. Below, we will focus on the conceptual modelling (M1), since this is closest to modelling as understood in the technical and physical disciplines.

A conceptual model is an abstract and organized description of the problem-solving expertise (explicating, so to speak, the laws of inferential motion). It is the outcome of the interpretation and knowledge analysis of the verbal data obtained from the expert. No commitment is made in the conceptual model with regard to any kind of implementation decision. The role of a conceptual model is a high-level specification of the required knowledge, the inference competence and the use and flexible control of knowledge, that serves as a basis for later KBS design and implementation stages. In order to build a conceptual model, KADS supplies a four-layered framework for describing expertise. Each layer is considered as having its own, relatively independent, functionality. The four-layer structure may be viewed as a more detailed explication of the distinction in cognitive science between the domain or object knowledge and the strategic meta-knowledge that controls the use of the object knowledge. A summary of these four layers is given in Table I.

Table I: Layers of the conceptual model (after [9]).

Layer	Elements	Relation to lower layer
Strategy	Plans, meta-rules	
		controls
Task	Goals, task routines	
		applies
Inference	Knowledge sources, metaclasses	
		describes
Domain	Concepts, relations, laws	

The lowest layer contains the static knowledge of the domain: domain concepts (e.g. represented as a kind of frames) and their relations and more complex structures, such as models of processes or devices (e.g., consist-of hierarchies). This represents the strict domain theory and is used as the support knowledge of the KBS. The second layer is the inference layer. It describes the inference-making competence that is available for solving a problem based on the domain knowledge. It only indicates what inferences can be made (so, it is a kind of route map), not how and when they are made. The how is a function of the underlying domain knowledge, the when is specified in the higher layers. Typical concepts used in the inference layer are knowledge sources and metaclasses. Knowledge sources are inference-making functions that map domain concepts onto other domain concepts. The domain concepts in their specific role of input and output arguments of inferential functions are called metaclasses (e.g., norm, parameter, symptom, hypothesis, constraint). The triplet of knowledge sources with the input and output metaclasses can be seen as an inference element; a composition of these is called an inference structure (the complete road map). A simple typology of knowledge sources that has appeared useful in practice, is given in Table II. A good example of an inference structure is heuristic classification, as discussed in Clancey [10].

The third layer is the task layer. It contains fixed task structures, i.e., standard problem solving routines. These are specified in terms of goal and control statements. Goal statements define the invocation of a knowledge source, or of a structure of knowledge sources. It is at the task layer where control notions like iteration and selection enter for the first time. Thus, the task layer represents reasoning and problem

Table II: Typology of knowledge sources (after [9]).

Group	Operation (Knowledge Source)	Arguments
Generate concepts	Instantiate	Description → instance
	Classify - Identify	Instance → description
	Generalize	Set → description
	Abstract	Description → description
	Specify	Description → description
Differentiate between concepts	Compare	Values
	Match	Structures
Structure manipulation	Assemble	Set → structure
	Sort	Set → sets
	Decompose	Structure → set
	Transform	Structure → structure
Change concept	Assign_value - Compute	Variable → value

solving strategies, but only to the extent that these are standard and fixed. The flexible utilization of knowledge and of meta-knowledge is modelled in the top level, the strategy layer. This type of strategic reasoning is often of a meta-nature (knowledge about problem solving itself) and commonly also impasse- or failure-driven. However, in current knowledge-based systems this top level is often empty. The totality of the descriptions of the separate layers is called a conceptual model.

4. Possibility of mathematical formalization of knowledge analysis

From the above review it appears that the requirements (i) and (ii) as formulated at the end of Sec. 2 are satisfied to a large extent in a structured modelling methodology like KADS. This is already more than can be said of the prototyping and mining approaches. We point out, however, that KADS mainly supports the conceptual modelling stage (which also may be called the information analysis) and, to a lesser extent, the global system design. At present, there is no support for the later stages of the KBS software lifecycle (but here existing shells and languages are helpful) and a very limited support for the first stage of preliminary or feasibility study. On the other hand, this is quite in line with the situation concerning modelling as carried out in the natural sciences. There is an important difference, however. Although the modelling concepts employed in KADS have shown their pragmatic and intuitive appeal in applications, they still are of a qualitative and informal nature. Hence, our desideratum (iv) of a mathematized language is not met at all and, as a consequence, this also limits the "objective", verifiable nature of the ensuing model (so that also our requirement (iii) can only be partially fulfilled). Not meeting requirement (iv) is common to all knowledge acquisition methodologies, except in those cases that the problem at hand is so simple that it can be immediately translated into a PROLOG-type of logical program (but this violates all other requirements).

The question then arises in which direction a mathematization of knowledge-engineering methods might be sought. As indicated in the previous section (see also [1],[5-10]), "structures" and "models" abound in approaches dealing with KBS conceptual modelling and design. At the same time we observe that in mathematical logic precise and formal definitions exist of these notions (cf. [7,8] and refs. therein) which are furthermore closely related to the concept of (many-sorted) algebras encountered in abstract data types [11] and also in mathematical physics (e.g., groups, matrix algebra). Moreover, well-known knowledge representation constructs like frames and semantic nets can be reformulated in such terms [8]. Therefore, we propose to take the notion of a many-sorted structure in the sense of mathematical logic as a point of departure of mathematization. Below we will give a global and preliminary outline of how to proceed with this idea (compare, in a different context, also Weyhrauch [12]).

A structure S is an ordered sequence

$$S = \langle U, c_i, R_j, f_k \rangle ,$$

where U is a non-empty set called the universe (of discourse) or the domain or the carrier of the structure, the c_i are constants denoting the individual objects that are the elements of the universe U, R_j is a set of relations (=subsets) on this universe, and f_k is a set of functions on U; i,j,k are members of index sets. It is implied that together with the relations and functions, their arities are indicated. If the universe is divided into various sorts or types (e.g., reals, integers, character strings), the structure is said to be many-sorted [7]. A structure not containing relations (except for the equality relation) is usually called an algebra. Thus, a structure may be understood as a generalization of the mathematical concept of an algebra, which for KBS purposes will be more powerful.

Next, it is convenient to introduce the (strictly syntactical) notion of a signature [7,13], a basic concept in the algebraic specification theory of abstract data types. A signature Σ is a set, containing a declaration of names of: the universe and its sorts; individual objects in this universe (constants for each sort); relations and functions, including the sorts of their arguments and their arities. Now, a structure of signature Σ can be redefined as an assignment of "real" relations, functions etc. (i.e. referring to the part of reality the structure intends to describe) to the names in the signature set. In other words, a structure of signature Σ or Σ-structure is a pair

$$S_{\Sigma} = \langle U, I \rangle ,$$

where I is an interpretation function, mapping the names of the signature Σ onto constants, functions and relations over the universe U of the structure.

Evidently, the signature is the basis of the syntax, i.e., of a predicate language L(S) of the structure, which is generated by the symbols contained in the signature plus the usual variable symbols, logical connectives and quantifiers, and possibly additional auxiliary symbols. Quite clearly, on this basis one can express propositions about that part of the real world the structure is supposed to represent, in the language of (usually first-order) predicate logic. Thus, one may practically write down a set of predicate-logical assertions Γ that reflects the true state of affairs in the world we want to describe via the structure S. Then the structure S is formally said to be a model of Γ. It is easy to see that Γ may be interpreted as a knowledge base in a sense that is a generalization over the widely used if-then-rules. If we supplement the language L(S) with some deduction system (for example, resolution à la PROLOG, or backward-forward chaining as found in expert system shells), it is also clear how we can automatically prove new facts and rules from Γ, which in this context acts as a set of axioms. This discussion indicates how we might make mathematically precise the idea of data structuring in knowledge acquisition and how one might develop a formally clear specification $\langle \Sigma, \Gamma \rangle$ of a KBS conceptual model.

These remarks still belong to the realm of pure theory. If the research program suggested above can be successfully carried out, it will be important to make available to the knowledge analyst corresponding automated tools as part of a knowledge-engineering workbench. Here, we feel that an object-oriented language should be used to implement the afore-mentioned theoretical ideas. First, we need to represent various levels of abstraction which are seen (Sec. 3) as rather independent; this idea fits quite well with the object-oriented class mechanism. Next, our notion of model and structure may be interpreted as complex abstract data types/data structures, these having a natural representation in classes/objects [13]. Finally, in any kind of knowledge representation inheritance mechanisms are very helpful. Both theoretical and implementational aspects are presently being investigated in the context of certain classification problems.

Finally, as a simple and yet relevant application and adstruction we point out how the physical concept of time can be modelled as a structure in the sense of mathematical logic. In many physical laws of motion (for instance, the Boltzmann and the Schrödinger equations) time enters as a continuous parameter with certain properties. Here, time can be interpreted as a structure:

$$T_1 = \langle R, <, \text{differentiation}, +, \ldots \rangle,$$

where the universe R is the set of the real numbers, < is a linear ordering ("earlier"), etcetera. This is an elementary structural description of the common continuous time. A completely different concept of time is encountered in the theory of random walks and Markov chains:

$$T_2 = \langle N, <, \ldots. \rangle,$$

where N denotes the natural numbers and < is the ordering of events ("steps of the walker"). This is a discrete concept of time. Both these concepts of time are actually (but implicitly) being used in the physical modelling of the same problems, for example in theories applicable to the evaluation of nuclear data for fission and fusion reactors. A detailed exposition is given in [14]. This suggests that mathematical logic may also be useful in clarifying physical and technical model concepts.

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THE ESSENTIAL SYSTEMS STATUS MONITOR FOR HEYSHAM 2 NUCLEAR POWER STATION

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Abstract

This Paper summarises the principal features in the development of an operator aid, the Essential Systems Status Monitor (ESSM), now installed at CEGB's Heysham 2 nuclear power station. The facility enables the operator to carry out, interactively, probabilistic evaluations of the risk arising from the unavailability of the essential safeguards plant. The Paper describes the significant benefits in the operational use of the ESSM in terms of simplification of Operating Instructions and increased flexibility in operation.

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1. INTRODUCTION

The operation of a nuclear power station requires that safety objectives are adequately satisfied in order to assure the safety of the operators and the general public. For these objectives to be satisfied, the essential systems of a nuclear power station are installed and operated with considerable redundancy in plant and equipment. Different systems contain different levels of redundancy of plant, depending on the reliability requirements for the essential functions provided. It is a necessary requirement that, in the presence of faults or outages of plant due to maintenance, this redundancy is always maintained at an acceptable level.

Traditionally, in the United Kingdom, the minimum levels of plant redundancy in each essential and supporting system have been stipulated to operational staff in concise and unambiguous instructions. However, because of the requirement for conciseness, and the inherent complexity and inter-relation of the essential systems when considered as a whole, these instructions have interpreted the requirements for minimum levels of plant redundancy in ways which are pessimistic. The instructions have been based on deterministic fault tolerance criteria applied on a system by system basis.

The extensive experience within CEGB of PSA techniques, in particular in fault tree analysis techniques, suggested the possibility of applying some probabilistic methods to the derivation of these operating instructions. An extensive programme of software development was therefore initiated to provide a fault tree model of the essential systems plant which could be assessed interactively from the Central Control Room as an operational aid, and from the planning office as a maintenance planning aid. The fault tree model had to be capable of being continuously updated to accommodate plant outages and plant changes as they occurred.

The development of this software has been completed (Ref. 1) and the facility, known as the Essential Systems Status Monitor (ESSM) has been installed on a dedicated mini computer at Heysham B Nuclear Power Station.

This Paper briefly reviews the development of the ESSM and then discusses its operational use and the operational benefits that result from the use of the facility.

2. DEVELOPMENT OF THE ESSM FACILITY

2.1 Objectives

The latest CEGB Nuclear Power Stations are designed to meet probabilistic targets. One fundamental target is that the total frequency of exceeding the design basis for reactor trip, shutdown and post trip cooling should not be greater than 10^{-6}/annum. Part of the safety case therefore includes demonstrating the acceptability of the design in relation to this probabilistic target. In this safety demonstration certain assumptions are made, in advance of operation, concerning the effect that plant outages will have on this frequency of exceeding the design basis.

An overall objective in the development of the ESSM was to provide a facility which could derive the effect, on the frequency of failure, of the actual plant outages during operation. This would enable plant outages to be controlled to ensure that the probabilistic claims made in the safety case would not be invalidated.

For the ESSM to be suitable as an operator aid the facility was required to:

i) assess the probability of failure of the essential systems for all the relevant initiating events;

ii) assess the effects of all possible combinations of plant outages and plant configurations;

iii) carry out the assessments quickly, i.e. to enable it to be used "interactively";

iv) provide the operator with advice on what plant should be returned to service from a maintenance state in order to optimise the improvement in the overall integrity, i.e. to provide a maintenance priority function;

v) provide a separate facility to enable plant outages to be planned;

vi) be capable of running on a dedicated mini-computer at the Station.

2.2 Design Development

The design of the ESSM facility is shown in Figure 1. The software has been developed on a Honeywell DPS6/92 machine, and an IBM mainframe machine, with the software written in FORTRAN 77. The size of the machine required to run the ESSM is basically a function of the size of the input fault trees.

There are 2 distinct software modules, the 'pre-processor' and the 'interactive' module.

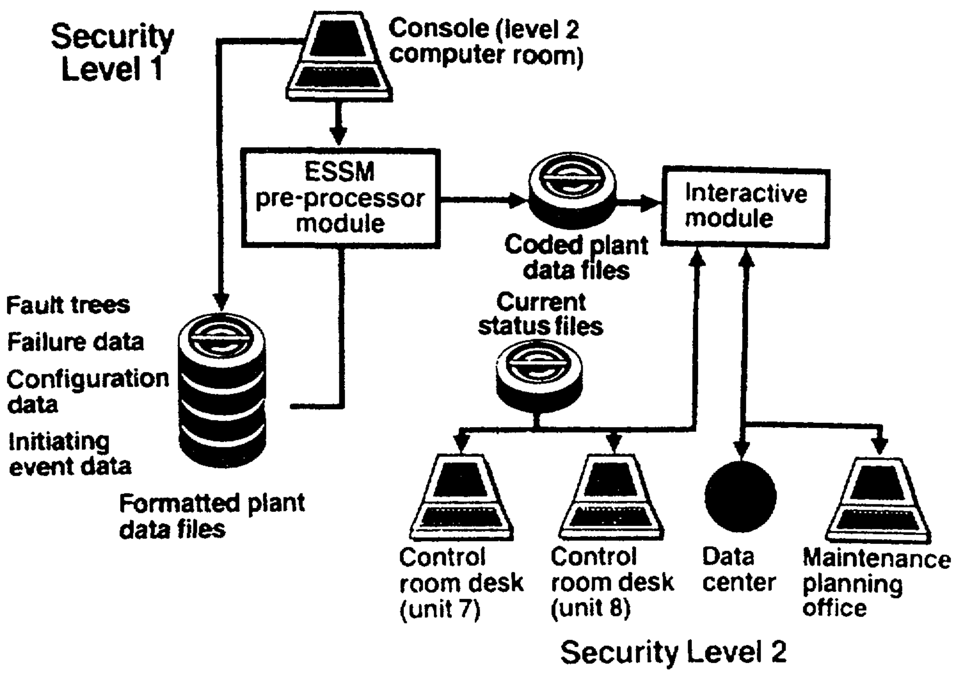

FIG.1. ESSM computer system.

The 'pre-processor' is responsible for checking the input data files which define the fault trees, item probabilistic failure data, initiating events and plant configuration details. The 'pre-processor' module can only be accessed by personnel with the necessary security clearance and is used infrequently (e.g. to make modifications to the original fault tree structure). In addition, the 'pre-processor' is responsible for coding the input data into a form ready for use by the 'interactive' module.

The 'interactive' module is responsible for providing the main functions of the ESSM, namely the integrity assessment and maintenance priority functions. The 'interactive' module is accessed by the operators and is in continual day-to-day use. Facilities are provided within the 'interactive' module of the ESSM to allow the operator to enter the current status of the essential systems.

The operator informs the ESSM that various plant items are unavailable due to faults or preventative maintenance. In addition, the operator may input some system configuration changes, such as changes in the electrical system, to the ESSM. The ESSM reconfigures and re-analyses the effects of such outages, and configuration changes, on the original basic fault trees.

Special highly efficient and fast analytical methods were developed for the ESSM to enable it to be interactive. The minimal cut-sets for the fault tree representing all the essential systems are evaluated taking into account the logical effect of plant outages and plant configurations. The ESSM does not work by modifying previously calculated lists of minimal cut-sets, but re-calculates the minimal cut-sets for each assessment. This is essential to the accuracy of the ESSM.

TABLE 1. ESSM — ESSENTIAL SYSTEMS MODELLED

Post trip sequencing equipment
Pressure support system
Start/standby boiler feed system
Emergency boiler feed system
Essential electrical system
Decay heat boiler feed system
Reactor sea water system
Inlet guide vane system
Gas circulators
Circulators auxiliaries cooling system
Circulators auxiliaries diverse cooling system

Fault trees modelling the eleven essential and supporting systems listed in Table 1 and some forty two initiating events are contained in the plant data files of the ESSM. Data for the frequencies of the initiating events and failure data for components are also contained in the files. These are all used for the probabilistic assessments carried out by the ESSM.

Computing times taken by the ESSM in evaluating the complete fault trees are very significantly reduced over the times that would be taken by traditional fault tree programs. The ESSM takes around three minutes to complete these evaluations whereas the traditional programs would take many hours. These short computing times enable the ESSM to be interactive, i.e. provide on-line computing facilities for the control room operator.

Also contained in the data files are logic modules of the "back stop rules". These rules describe overriding deterministic criteria for the plant availability conditions for which immediate remedial action is required, such as a controlled reactor shutdown. These rules are also contained in hard copy format in the main control room. The plant unavailability conditions requiring some immediate action therefore are not identified solely from an assessment performed by the ESSM, thus avoiding undue reliance on the ESSM by the operator. In practice the ESSM automatically identifies to the operator the particular back stop rules satisfied in any operating condition and this provides a quick means of reference for the operator to the hard copy rules.

2.3 Control Room Facilities

The facilities provided in the Control Room of the power station for operating the ESSM consist of two computer terminals with VDUs, one for each reactor. One hard copy printer is located near to the Control Room.

The ESSM is a menu driven facility in which the operator selects options from successive menus for the precise function he requires. The interactive functions of the ESSM provide:

1. information on the current status in terms of the acceptability, or otherwise, of continuing operation following plant failures, or on a future status if plant is taken out;

2. the capability to re-configure the system fault tree models either to model existing conditions or to model future planned plant outages;

3. information on the most effective options for plant replacement which will improve the status of the systems.

The information on current status is displayed to the operator in terms of one of three conditions; 'normal maintenance', 'urgent maintenance', or 'immediate remedial action', as shown in Figure 2. Precise probabilistic information is not displayed. Although the condition of 'immediate remedial action' may be displayed by the ESSM the operator would not normally use this as a basis for initiating a reactor shutdown. He is always required independently to confirm that the system status is consistent with the limits of the overriding simple deterministic rules as contained in hard copy form in the Control Room. The ESSM does also incorporate these limiting deterministic rules within its modelling so that before any probabilistic assessment is carried out the ESSM first checks that the system status satisfies these rules.

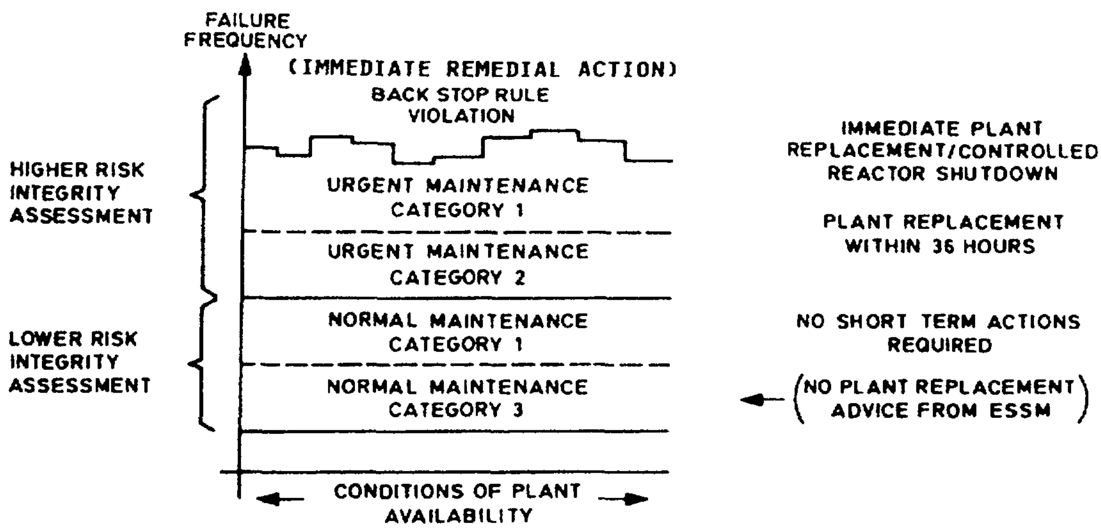

FIG.2. ESSM integrity assessment categories.

Having completed an assessment of the status of the essential systems, the ESSM may then be used to provide information on the optimum combinations of plant for restoration from unavailability conditions which could improve the overall integrity of the systems. This information is provided in the form of advice to the operator so that other factors not available to the ESSM, such as availability of maintenance personnel, may be taken into account in adopting the most appropriate action.

The hard copy printer located near to the Central Control Room is provided so that every time the plant availability conditions of the essential systems are changed then a record of this is made in hard copy format. This is expected to be of particular use at the beginning of an operator shift, and for providing a record of the system changes.

2.4 Maintenance Planning Facility

The facility for using the ESSM for advice on maintenance planning is provided on a separate terminal located in the planning office.

For this facility the ESSM generates a separate fault tree model to that used by the control room operator so that the planning office does not access the fault trees modelling the current status of the systems. This facility therefore allows future outages of plant to be modelled and assessed. This allows an operator to plan the outages of plant necessary for scheduled maintenance so that coincidences of plant unavailabilities have a minimal effect on overall system integrity.

2.5 Assessment Principles

The probabilistic integrity assessments provided by the ESSM are system assessments summated over all the initiating events in turn factored by their estimated frequencies of occurrence.

An overall frequency of failing to achieve defined minimum success criteria is therefore calculated which is similar to that derived from a conventional PSA for the operation of the essential cooling systems following a reactor trip.

The integrity status of a particular system condition is assessed by comparing this overall risk frequency figure with fixed pre-set frequency levels, typically one decade apart. The system status is thereby defined in one of the three conditions shown in Figure 2.

3. OPERATIONAL BENEFITS OF THE USE OF PROBABILISTIC EVALUATIONS

3.1 Simplification of Technical Specifications/Operating Instructions

The most significant implication of the use of probabilistic evaluations rather than the traditional deterministic considerations, is the potential for the simplification of the Technical Specifications/Operating Instructions which define the minimum acceptable levels of plant redundancy. Probabilistic analysis of the deterministic instructions shows that the assessed failure frequencies for the different minimum plant conditions can vary considerably. One plant condition may be defined in deterministic instructions as acceptable only for a restricted time period, whereas probabilistically it may be assessed to be acceptable on a continuous basis. Traditional deterministic instructions tend to incorporate pessimisms to varying degrees and to consist of a significant number of separate instructions. Probabilistic assessment techniques allow pessimisms to be incorporated in a controlled manner and require very few instructions to be defined. A single probabilistic criterion may define a whole boundary of acceptable plant availability conditions which previously had been defined by a large number of instructions to cover the various plant combinations. In the UK three conditions of essential plant availability are defined, a "normal maintenance" condition which is continuously acceptable, an "urgent maintenance" condition which is acceptable on a limited time basis only (e.g. 36 hours), and a third condition which requires some immediate action for improvement to restore plant or a controlled reactor shutdown. These conditions are illustrated in Figure 2. In principle the boundaries between all these conditions could be defined probabilistically.

3.2 Increased flexibility in planning plant outages for maintenance

The use of the ESSM significantly improves the flexibility for the control of planned outages of essential plant. This is because:-

i) It significantly increases the number of permissible combinations of plant outages.

With the traditional deterministic instructions, outages of plant for maintenance are restricted to be planned according to the limited instructions which have been defined to the operator, and these are normally in terms of each system considered individually. By using probabilistic assessments the outages of plant may be planned, according to any combinations of unavailable plant and equipment, provided only that the results of the probabilistic assessment are within acceptable limits.

ii) It enables plant outages to be planned in such a way as to minimise the increase in risk.

3.3 Increased flexibility in restoring plant to service

In the same way as the use of probabilistic assessments signficiantly improves the flexibility in planning plant outages, so this also increases the flexibility of advice to the operator for restoring plant to improve an operating condition, i.e. "remedial advice".

This might be particularly useful, for example, when a plant failure occurs at a time of a planned combination of plant outages. With the use of deterministic instructions the advice to the operator on which equipment should be restored is limited by the particular deterministic instructions defined to the operator. Using probabilistic methods extends this advice, over a large number of combinations of plant, and this allows the operator greater freedom to select plant which may more easily be resotred to availability. Unplanned restoration of plant like this typically requires some rescheduling of priorities for completing planned maintenance outages. Any increase in the choice of plant which may be considered for this is an operational advantage.

Probabilistic methods also enable plant reconfigurations to be considered by the operator for improving an operational condition. These methods also allow priorities for plant restoration to be identified, i.e. some relative significances to be attached to the restoration of different plant. The significances may be either a function of a probabilistic "importance" function or defined in relation to increasing the order of dominant cut-sets in the overall assessment.

4. CONCLUSIONS

Significant advantages in the use of probabilistic evaluations for controlling the unavailability of essential safeguards plant at a commercial nuclear power station have been identified. CEGB has successfully developed a facility which enables these evaluations to be used in an operating power station, and has developed an approach for introducing them at Heysham II AGR nuclear power station.

ACKNOWLEDGEMENT

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EXPERT SYSTEMS APPLICATIONS IN PROBABILISTIC RISK ASSESSMENT

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Abstract

Over the last few years SRD have been researching into the application of expert system technology to risk assessment. A number of prototyping projects have been carried out, using a variety of expert system 'shells' and also using the AI (Artificial Intelligence) programming language PROLOG. This paper summarises the projects which have been undertaken and also those which are currently in progress.

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1. Introduction

The emergence of expert system technology has generated considerable interest in the field of probabilistic risk assessment. Some of the features of expert systems which have attracted this interest are:

- the handling of uncertainty
- the incorporation of heuristics* within a knowledge base
- the provision of good user interfaces
- the use of logic programming
- high level knowledge representation formalisms such as 'frames' and 'objects'

The risk assessment process consists of five main stages:

i Hazard identification

ii Frequency analysis

iii Consequence analysis

iv Calculation of risk levels

v Consideration of the acceptability of the risk results (with sensitivity analysis as required).

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*A heuristic is a 'rule of thumb' or simplification to a problem.

SRD have been researching into applications concerned with the first four stages of the PRA. Research work is also being undertaken into real-time-applications on nuclear power plant, with particular emphasis on the provision of information to operators for:

- fault diagnosis
- safety status of the plant

SRD carry out both non-nuclear (ie conventional chemical plant such as chlorine storage and LPG, and also explosives storage) and nuclear probabilistic risk assessments. The projects described in this paper cover all these domains; it is important to note, that the AI principles involved in the non-nuclear projects are directly applicable to the nuclear field.

2. Expert System Philosophy

The approach adopted at SRD in the expert system field has been to:

i Carry out an appraisal of various expert system products, ie expert system 'shells' and AI programming languages such as PROLOG.

ii Carry out a number of prototyping projects in current specialist domains. A key aim in these prototyping exercises has been to study different knowledge representation formalisms.

iii To aim towards the selection of a number of specific development tools for expert systems. Standardisation of software is an important consideration for any large organisation.

iv To provide effective interfacing with current computer codes in use at SRD, eg large Fortran-based dense-gas dispersion codes and fault tree analysis codes.

v To develop in-house knowledge-based tools to enhance current risk assessment practices.

vi To develop an AI capability such that an important commercial service can be offered to both the non-nuclear and nuclear industry in:

- advice on the choice of AI tools for particular applications
- knowledge representation formalisms
- hardware advice
- prototyping assistance

3. Summary of Expert System Developments

The following projects have been, or are currently being undertaken, at SRD:

- SAPIEXT: SRD all-purpose integrated expert tool
- TREEBASE: fault tree analysis
- BLEVE: fault tree analysis on LPG project (using the expert system shell CRYSTAL)

- EXHUME: human error expert system (using the expert shell CRYSTAL)
- HAZCHECK: hazard identification using 'expert checklists'
- RFME: PWR real-time
- EX-STORE: expert screening tool for risk assessment of explosives storage sites (written in PROLOG-2)
- CDES: consequence code decision aid (prototyping for SAPIEXT: using Xi Plus)
- FRACAS: the integration of fire modelling codes
- LOCA: PWR event trees application (prototyping using the UKAEA Harwell (Prolog) shell SPICES)

Details of a number of these projects are given in this paper (details of other projects can be obtained upon direct request to SRD).

4. Standardisation of Expert System Software

The standardisation of software is important in any large organisation. If this is not achieved then problems are encountered with:

- incompatibility between different developments
- lack of consistency in problem solving on similar domains
- inefficient use of resources

The potential applications for expert systems, however, covers a wide domain. As a consequence, it is unlikely that a single AI tool will prove to be applicable across all domains. It is considered that most organisations will require a standard tool for each class of application, ie;

- domains that require use of induction (ie generating rules from examples).
- real-time applications
- domains which require complex knowledge formalisms such as frames or object-orientated approaches
- rule-based domains

A cautionary note is included now on the subject of INDUCTION.

Induction can be an invaluable tool on certain applications, particularly those in which the generation of rules is proving to be a difficult task. It is important, however, that any rules which are generated using induction are fully validated. The problem with induction is that it is possible to generate incorrect (or incomplete) rules. On applications concerned with safety therefore, it is vital that all rules which are generated are fully validated. This leads us onto an important point for real-time safety applications:

ANY RULES WHICH ARE GENERATED (IN REAL-TIME) MUST BE FULLY VALIDATED.

Clearly this is a goal which would not be achievable. Therefore, it is considered that induction should only be used during the development of the knowledge base and should not be used to generate rules in real-time in applications in the domain of safety. The above precautionary note should not be seen as a vilification of induction, quite the contrary: it is considered that induction can be a useful tool during formalisation of the knowledge base.

5. Description of SAPIEXT

5.1 Aims of the Project

SAPIEXT (SRD All Purpose Integrated Expert Tool) is to provide an expert system environment which integrates the risk assessment tools which are currently in use at SRD. SAPIEXT is currently being developed for non-nuclear PRA but the intention is to incorporate nuclear codes at a later date. The basic aims of the project are to provide:

- an integrated environment for SRD computer codes
- a consistency/standardisation of use of various complex consequence codes (thereby formalising procedures)
- a training aid

5.2 Development Environment

SAPIEXT is being developed on an IBM-PC (386-based) compatible and the expert system development tools are GOLDWORKS and XiPlus.

GOLDWORKS is a LISP-based 'hybrid tool' which supports full frame representation formalisms, and can also be directly integrated with LISP. One of the advantages of GOLDWORKS compared to most other PC-based tools is the facility to run GOLDWORKS in extended memory, thereby leaving most of the 640K RAM still available for running MS-DOS applications 'underneath' the controlling expert system environment. This is an important feature for SAPIEXT, since the larger (PC-based) consequence codes in use at SRD use up most of the available 640K.

XiPlus is being used for prototyping work for SAPIEXT.

5.3 Analysis Areas

SAPIEXT will be used at SRD to encompass all areas of the risk assessment process, ie

- Hazard Identification
- Frequency Analysis
- Consequence Assessment
- Risk Calculations/Evaluations

In the hazard identification area it is intended to integrate computer-based tools for:

- HAZOP
- FMEA
- HAZCHECK ('intelligent' checklists)
- Chemical and hazardous properties databases
- MHIDAS (a Major Hazard Incident 'Database' developed at SRD)

Frequency analysis involves the determination of the frequency of predicted events. The frequency of occurrence of the identified failure events are currently quantified using the (PC-based) fault tree analysis program ORCHARD which hs been developed at SRD. SAPIEXT will provide direct access to ORCHARD and the further development and inclusion of TREEBASE is also planned. TREEBASE is a prototype fault tree analysis program written in Prolog-2 which can compute the top event of a fault tree using either:

- numerical values with Boolean Logic
- 'natural language' values with Fuzzy Logic

TREEBASE carries out Boolean elimination (using a 'heuristic' approach for the determination of the minimum cut sets) for numerical evaluation of a fault tree. Natural language computation is also performed on terms such as 'ALMOST DEFINITE', 'UNLIKELY', 'ALMOST IMPOSSIBLE', etc using Fuzzy Logic, ie

- at an 'AND' gate in the fault tree take the minimum likelihood value of the sub-events as the result
- at an 'OR' gate in the fault tree take the maximum likelihood value of the sub-events as the result

SAPIEXT will provide access, with help screens available as required, to the following range of sophisticated consequence models:

- CRUNCH - for 'continuous' releases of toxic or flammable material
- DENZ - for 'instantaneous' releases of toxic or flammable material
- GASP - evaporation from spreading liquid pools
- ENGULF - fire engulfment of vessels
- TRAUMA - 2-phase source term code
- DRIFT - this code will supercede both CRUNCH and DENZ.

5.4 Work to Date

SAPIEXT is an on-going project which will be continuously expanded as more PRA codes are developed at SRD, and as the nuclear domain is included. Progress on SAPIEXT up to date has included:

- providing standard 'C' (menu-based) interfacing to various consequence codes, to ease integration problems in SAPIEXT

- prototyping in XiPlus, eg CDES is a decision and which advises on how to model particular hazardous release situations and which consequence code to apply

- development work on HAZCHECK and TREEBASE

- assessment of frame-based products with the choice of GOLDWORKS as the development environment.

An interesting point here is that the CDES expert system may be left in XiPlus (it was originally intended to re-code this in GOLDWORKS) and access to this system could be achieved through GOLDWORKS. That is, with the provision of sophisticated AI software which can run in extended memory we have have the ability to integrate/interface different expert system environments.

6. Description of RFME

6.1 General

The Reactor Fault Management Expert (RFME) system is being developed at Sheffield University in the UK Under a PhD research project. The work is being funded by SRD and is concerned with the domain of real-time alarm analysis on a PWR nuclear reactor. The work is being applied on a model of the Loss of Fluid Test (LOFT) reactor. Particular emphasis is being made on the man-machine interface on the project.

6.2 The LOFT facility

The LOFT facility (Ref 1) is located at Idaho National Engineering Laboratory and it is designed to test the response of an operating nuclear reactor and associated safety systems under active emergency conditions. It is a 50 MW (thermal) reactor which models a Pressurised Water Reactor.

It is composed of:

i A reactor vessel with a nuclear core.

ii An intact loop with an active steam generator, pressuriser and two primary coolant pumps in parallel.

iii A broken loop with a simulated pump, simulated steam generator and two quick-opening valve assemblies.

iv A blowdown suppression system consisting of a header, suppression tank and a spray system.

v An emergency core coolant injection system consisting of two low-pressure injection system pumps, two high-pressure injection system pumps and two accumulators.

vi A pressure relief pipeline from the top of the pressuriser to the suppression tank containing the Power Operated Relief Valve (PORV) and Safety Relief Valve (SRV) used for experimental testing in parallel with the pressure relief pipelines containing the plant PORV and SRV's.

6.3 System Design and Architecture

RFME is being developed using the expert system 'shell' EXTRAN-7, from Intelligent Terminals Ltd, UK. RFME is based mainly on the behaviour of an operator in response to a deviation from normal operating conditions. In response to a deviation which requires emergency shut down of the reactor, the operator is called on to perform the following tasks:

i Diagnose the event, generally based on the following characteristics:

- use of casual relationships
- use of data-driven reasoning
- sensor data not always reliable

ii Determine an appropriate response.

iii Evaluate the effectiveness of the response.

iv Choose a decay heat removal (DHR) mode to remove the decay heat from the primary coolant system when the reactor is shut down.

v Evaluate the effectiveness of this DHR mode.

To detect, diagnose and treat an emergency condition, the expert system requires the following information about the plant:

i Heuristic rules and knowledge relating to primary deviations.

ii The normal operating band for measured and calculated process variables.

iii Current process measurements.

iv Cause-and effect models (Ref 2).

v Response trees (Ref 3). A response tree is a pictorial representation of a number of cooling modes of a reactor. A cooling mode is a set of components used to cool the reactor core. It generally consists of five elements:

- a heat sink
- a water source
- a pump
- a route
- an injection point

The main features of RFME are:

1. Detection of events
2. Alarm analysis

3. Fault diagnosis

4. Response tree evaluation

5. Advice and Recommendations

6.4 Software Configuration

RFME in its current version is being developed to operate on an IBM PC compatible. It employs two different tools: EXTRAN-7 and GEM.

EXTRAN (EXpert TRANslator) is a product of Intelligent Terminals (Ref 4). It is written in standard Fortran-77 and is a rule-based product. It can be event-driven as well as goal-driven. EXTRAN is composed of two parts:

i ACL Tran (Analog Concept Learning Translator). This part is the development tool by which the problem and examples are defined, decision rules are induced and FORTRAN code is generated.

and ii The Driver which is a set of object files that should be linked to the developed decision-rules to create the expert system.

EXTRAN provides facilities for flexible decision-rule structuring allowing hierarchical dependency to be established between decision-rules in a simple way.

RFME has been linked to GEM by building an assembler interface. Once a decision concerning a cooling mode is reached, the GEM application sub-routine is called to display graphically the selected path of the cooling mode and the faulty components.

RFME is currently configured to run in an interactive mode to obtain process parameters. Future developments will include data input to RFME from an external file, with the user only to be consulted at the end of the decision; the user at this stage will then be able to query the system on the decision path and obtain information about the system, etc.

6.5 Pressuriser Model

The pressuriser sub-system has been simulated using the interactive simulation package PSI (Ref 5). A detailed automated fault diagnostic expert system (FDES) for the pressuriser has been developed. The following stages were carried out for the development of FDES:

i The pressuriser has been broken into small well-defined units.

ii Based on functional models, an alarm tree has been created for each unit.

iii Finally, using the process flow diagram, the unit alarm trees were connected, thus creating the pressuriser alarm network.

Current simulation runs have shown that FDES performs well, with several simulated faults being quickly detected and diagnosed.

The following interesting points are worth considering in relation to this problem-domain:

1. Care should be taken to use realistic values for alarm limits, which are the normal operating band for a process variable.

2. The steering commands to FDES should be sensor inputs, so there is a need to determine the quality of this sensor data since sensor devices fail or degrade over time.

3. An important concept to consider when evaluating a system for suitability in a particular operation is that of future system expansion. In this application the knowledge about the process system can continuously be expanded and revised because it is embodied in an easily retrievable knowledge base.

The current developments within RFME/FDES are still not on real-time data acquisition. However, even without real-time capability the system developed is a useful interactive tool for operator support and could also be used for the training of operators.

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THE APEX PROJECT: A COLLABORATIVE EXPERT SYSTEM DEMONSTRATOR PROJECT

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Abstract

This paper describes an expert system demonstrator project sponsored by the CEGB as part of its work on the use of large, real-time operator support systems. The project was carried out at a conventional plant as a pre-cursor to a nuclear application. The project was intended to examine various areas of concern and to gain hands-on experience of the application of a working system. The results of the project have formed the basis of further proposals for the application of an expert system to an Advanced Gas-cooled Reactor system.

1. Origins

The adoption of a centralised approach to process control over the last fifteen years and the resulting re-allocation of functions from men to machines has increased the demands on the process operator, particularly under fault or abnormal operating conditions when automatic functions become discontinuous or fail. There is a clear need to support operation decision-making using modern information processing technology and methods. To date, the application of available techniques has provided solutions to many problems in the areas of control and plant performance optimisation but little has been done to cope with the problems of operator overload associated with faults on complex plant.

The Central Electricity Generating Board have had experience of the application of alarm analysis schemes on a number of its power plants. These schemes were conceived as means of reducing the number of alarms presented to the operator, particularly during faults, and also improving the comprehensibility of that information by presenting messages which more clearly reflected the 'prime cause' of the disturbance or fault. These schemes have not been totally successful and over the years, operating staff have chosen to simplify the alarm presentation systems. Computing systems are now available which appear to have advantages in processing complex problem domains using so-called expert system techniques and dedicated processing machines. The CEGB and others were keen to establish the value of these for use in power plants and other large industrial processes.

The CEGB, along with some 60 other industrial organisations, are members of the SIRA Industrial Expert Systems Club organised by the Scientific and Industrial Research Association. The club, which enjoys financial support from UK Government Department of Industry (DoI), provides members with knowledge and hands-on experience of commercially available expert systems. Members share experiences and information and a number of collaborative application projects are being developed. Against this background the Alarm Processing Expert System or APEX, was originally proposed in November 1984.

A sub-group was formed to manage the APEX project. Those who joined were :-

British Gas
British Petroleum International
British Petroleum Research
CEGB
Sira Ltd

2. Project Objectives

The objectives of the project were developed by the club through discussions between interested members.

They are:-

- To apply a commercially available expert system to a realistic industrial problem.
- To demonstrate the potential of the technology to handle large and variable quantities of information by intelligent information processing.
- To improve operator comprehension of complex events, particularly by improving alarm processing and presentation.
- To avoid plant shutdowns by early identification and analysis of disturbances and incipient failures.
- To minimise the downtime incurred following equipment failure by cause analysis and provision of remedial advice to operators.
- To obtain hands-on experience of knowledge elicitation and knowledge engineering.
- To quantify the engineering resources required.
- To evaluate the performance of the resulting system, on-line and under realistic plant operating conditions with a view to its possible use on future plants.

The deliverables of the project, i.e. the tangible benefits to participating members, were seen as:-

- A detailed case-history of the problem in the form of the project design history and an operational log.
- A system training package coupled with training and experience for all involved parties involved.
- A practical demonstration of a working system under realistic industrial conditions with quantification of its speed and accuracy.

- A statement of resource inputs including financial, engineering design, knowledge representation and operating involvement.

- A detailed knowledge base reference.

- A general functional specification on which to base future industrial plant information processing expert systems.

3. The Application

In order to identify a suitable application, a steering group produced a questionnaire which was sent to all club members. This document identified the key features of a host process application which were deemed necessary for the proper implementation of the system and for a adequate demonstration of the techniques.

The attributes of a suitable plant were deemed by the group to be as follows:-

- Must be automated, with a currently installed computer system.

- There should be easy access to the computer through a highway.

- Must exhibit disruptions or disturbances.

- Operation of the plant should present a challenge to operators.

- The plant should belong to a member of the participating group.

- The plant should be commercially non-sensitive.

- There should be a record of past incidents and corrective actions.

The plant which was chosen as a suitable site by the club is a coal-fired power station at Thorpe Marsh near Doncaster. The station, which was commissioned in June 1967 comprises two 500 MW units with pulverised fuel fired boilers. Although approaching "middle age" the Station is high in the CEGB's merit order. The design of the furnaces with their twin combustion chambers and eight coal mills in combination with the cross compound turbines introduces certain interesting operational characteristics.

In 1981 the original automatic boiler control scheme which was based on pneumatic equipment was replaced with a distributed direct digital control scheme. The replacement scheme is based on a host computer supporting the CEGB CUTLASS language and a number of target computers each associated with a portion of the plant control. The computer control scheme covers the five major control loops and operates some 56 plant actuators. The plant is complex in design and operation but has the advantage of a long operating record and a well developed operating regime. The plant is operated under a combination of defined rules, codes of practice and heuristics. Much of the engineering data base is well documented.

In order to interface the expert system to the existing computer system whilst providing adequate data security and isolation of the control functions an additional PDP 11/23 microcomputer was installed in the existing scheme to act as a 'bridgehead' machine. This simplified the interfacing software and provided data in a form which could be readily accessed by the expert system.

In order to manage the on-site interfacing activities the CEGB set up its own small management group.

4. Expert System Selection

At the time of selection in mid-85 the only suitable expert system was felt to be the PICON system produced by Lisp Machines Incorporated. This is a real-time process control system designed particularly for alarm information management. The system runs on a purpose built hybrid machine consisting of a Motorola 68010 front end process monitor combined with a dedicated Lisp processor, known as a LAMBDA-PLUS. Several of these machines are in use in the USA, in both industrial and research environments, and favourable reports had been received from satisfied users. The Actual expert system shell, PICON was then still developmental.

The system which was installed comprised:-

- Lambda-Plus computer including 68010 real-time processor combined with a 32-bit LISP processor.
- Lambda system terminal.
- PICON process message display terminal.
- 0.5 in. tape streamer for system loading and backup.
- Hard copy printer.
- Communications card to provide links to process control system.
- PICON software package.

5. Work Programme

The relative merits of equipment purchase and equipment rental were closely examined. The problems of joint ownership of assets with a group of companies and the attendant disposal problems were considered as were the implications of the government support for the project. These factors, together with the overall project timescale led the working group to conclude that a rental arrangement offered the best choice.

The estimated cost of the expert system equipment was some $150,000 at 1985 prices. In addition, the customisation, installation and management and support charges were estimated to bring the total project cost to about $180,000. Allowing for government support and by sharing the costs between a number of club members, the outlay per participant was expected to be a maximum of £16,000.

The sub-group members wished to have direct hands-on involvement in the various aspects of the project but also had clear ideas on when final answers were required. A 60-week programme of work was devised and the project commenced in June 1986.

The project was divided into a number of work packages covering preparation of the interface and communications software, knowledge acquisition, knowledge base construction and testing of the system.

From the outset, members felt that the project would benefit from a formal evaluation in order to provide an objective element in the final assessment. An assessment panel consisting of three persons from the sub-group member organisations was set up to plan and document the assessment.

6. Knowledge Acquisition and Engineering

The project was a collaborative one in which all participants wished to derive maximum benefit from their involvement. However, it was recognised that practical considerations dictated that certain tasks fell to particular organisations. The staff at the application site expressed support for the proposal as did the CEGB research organisations. The often quoted essential criteria of an available and willing domain expert is therefore satisfied. In addition to on-site operations staff the project involved collaboration between management, maintenance, and research staff within the CEGB organisation.

The bulk of the knowledge on plant design and operations was derived from the host's engineering database and from discussions with plant operatives, plant operations ad maintenance engineers and research department staff. Knowledge acquisition was predominantly carried out using unstructured interviews. A number of teams were involved in the process and each team employed a slightly different "style".

Three sets of knowledge were produced:- relating to the boiler combustion process and associated coal-milling plant. The basic operation of the coal mills is to crush coal between a rotating table and a set of three conical rollers. The coal is fed to the mills by drag link feeders supplied from overhead hoppers. The coal falls from the feeder down a central chute onto the rotating table. A particular problem at Thorpe Marsh is that of coal flow failure. If coal stored in a bunker is wet or sticky, it is inclined to form bridges at the bunker nozzle, and thereby interrupt the flow of coal to a mill. If the flow of coal to a mill ceases, the ratio of air to coal rises and, in the classifier duct particularly, may enter a range in which the suspension is explosible. The risk of explosion can be prevented by purging the mill with inert gas once the failure has been detected, but clearly this introduces an undesirable disturbance to the operation of the furnace.

The initial problem chosen was that of detecting coal flow failure. This was chosen because:-

- it is a relatively simple problem and is well bounded
- current instrumentation does not provide a reliable solution
- there is a potential explosion hazard associated with coal flow failures if not detected.

The global objective was to provide consistent high quality information and advice to assist operation.

Another application of interest was the monitoring of the general condition of a coal mill, for example its state of wear, which may be determined from its process parameters. Also the interactions of a number of mills must be regulated for the smooth operation of the furnace. Lastly the combustion process itself benefits from the application of operating expertise.

Knowledge engineering fell therefore into three parts:-

- that associated with a single coal mill
- that associated with eight mills operating together
- that associated with the combustion process.

Knowledge was input to the PICON system and tested using stored operation data in the form of histories and by controlled testing with on-line data. It was possible to introduce plant disturbance and monitor the system performance.

In July 1987 the system was made available to plant operators in the central control room. Operators on the night shift was asked to consider the messages produced by the PICON system and respond with either "Agree", "Disagree" or "Can't Say". These responses were automatically logged by the system for analysis the next day. In this way the knowledge was progressively refined.

7. Assessment

The formal assessment process was based on a set of criteria which were derived from the original project specification and objectives. From these a set of assessment tasks was produced involving all members. Independence was preserved by arranging that no person assessed his own contribution to the project.

The performance of the system hardware and software was examined as were the extent of facilities provided by it. This was done in a number of ways including direct observation, specific probing and controlled tests.

The execution of the project as a whole was audited and the resources used were analysed. In addition, an attempt was made to examine the future potential of expert system technology in relation to more conventional approaches in the context of process monitoring and control.

An important element in gathering data for the assessment was the questionnaire which contained some 90 questions and was sent to 30 participants.

8. Lessons Learned

The detailed findings from the project remain confidential to members, however, the overall results can be summarised in the following way:-

- Collaborative projects such as this are an effective way to gain hands-on experience and knowledge at a reasonable cost and with limited risk to each member.
- A working system can be created using in-house skills in a realistic time.
- As reported by many workers in the field, knowledge acquisition is a significant element in such a project.
- The knowledge gained is useful in its own right as an aid to better operations, even if it is not encapsulated in an expert system.

- Knowledge acquisition can provide interesting feedback on the effectiveness of operator training.

- For a large process such as a power generating unit significant amounts of data must be processed at a fast rate if real-time performance is to be achieved. Current expert system technology cannot yet provide us with fully adequate tools for this.

- To provide an effective system a range of knowledge representation techniques are required including generic rules and the ability to customise frames to the particular process domain.

- Many questions of software quality assurance remain to be solved before expert systems can be used in critical or safety-related roles.

9. Conclusion

It can be predicted that expert system shells will have a place in real-time process monitoring when sufficiently mature products with adequate performance can be made available. Meanwhile, the study of expert systems and associated knowledge-based approaches will provide valuable information on how systems including the human operator perform.

KEY TOPICS ARISING FROM THE APEX PROJECT

. PRODUCT MATURITY
  Language/tool kit/shell

. SOFTWARE MANAGEMENT TOOLS

. KNOWLEDGE ENGINEERING TOOLS
  Knowledge engineering 'overheads' for maintenance

. SOFTWARE VALIDITY
  Inference mechanisms
  Quality assurance

. REAL TIME PERFORMANCE
  Interface to process information system

. KNOWLEDGE REPRESENTATION TECHNIQUES
  Generic rules, customised frames, deep knowledge?

. HARDWARE DEPENDENCE
  'exotic' hardware

. INTEGRATION WITH PROCESS M.M.I.
  Operator interface, developer interface

. CLEAR SAFETY/COST BENEFITS
  Expert systems vs. conventional approaches

USE OF EXPERT KNOWLEDGE IN MONITORING AUTOMATED SYSTEMS FOR NPP POWER UNIT OPERATION

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Abstract

This paper describes two expert system projects developed in VNIIAES Institute, Moscow. One of the systems (PEX) is a ES-shell of classical type able to manipulate fuzzy expert assesments.
The system used as a shell for ES-advisor for MCP failures diagnostics and in some applications of the same sort.
Another system (ESDS) is on-line express-diagnostical ES for NPP unit emergency regimes identification. EDES is acomponent of NPP system of control and operation conditions diagnostics.

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1. Modern NPP can be considered as managment object having the highest category of complexity according to the number of functional components of connections beetwen them and responsibility of tasks for safe and simultaneously effective control. The complexity is determined not only by complexity of processes going in the system but by these processes velocities changing from seconds (emergency reognition) to months (reactor company) and even to years (equipment reliability).

Class of tasks to be solved in the process of control has the same wide scope. In particular a man has to solve tasks connected with signals interpretation, diagnostics and analyses, tasks of tracking and tasks of actions planning.

As experience has shown it's impossible to transfer solving these tasks completely on automated systems. However it's necessary to create man-machine "automated systems to support" deciding persons (DP). For example desire to provide more detail control by increasing the number of parameters to be controlled has led to the fact that now operating personnel is to control indications of approximately 10000 transducers. Such large information volume make it difficult to decide correctly especially when it shold be done in short terms or if transducers indication require taking measures which don't coordinate with each other.

Existing automated system have already shown sufficient efficiency and reliability. So according to emergencies analysis made in USA during resent years erroneous personnel actions were emergencies causes in most cases. Thus we have no doubts that it's necessary to develop automated systems supporting NPP personnel operation. The problem is to increase these systems intellect, their flexibility, dynamics and to make man-machine interface with the systems as much convenient for user as possible.

One way to decide the problem is to create new generation of program systems so-called Expert System (ES). Programmes or apparatus-programme tools using experts knowledge in specific area while solving problems are called Expert Systems (ES). ES formally simulate experts work. ES have the following features:

- decisions are the result of symbolic reasoning based on hearistiks;
- decisions are clear, of high quality and require small resources;
- systems are able to analise and explain their actions and knowledges;
- systems are able to assimilate new knowledges and correspondingly change their behaviour;
- systems provide friendly interface with the user;
- problems that are to be solved by systems are rather complicated and significant;
- at last if systems knowledge base has be worked out by experts having high qualification the system permits to increase quality of decisions made by DP using this system.

Now there are already about several hundreds of ES operating in different areas including ES in atomic industry.

Researchers from Oak Ridge National Labs (USA) came to the conclusion that in order to solve tasks connected with NPP operation it's worth while creating Expert System (I). There have been determined the following fields of investigation:

- component state examination;
- automatic control of equipment characteristics and its ageing;
- failures contribution in trees of events;
- interpretation and signals checking;
- dynamic treatment of emergency and warning signals for operator;
- timely recognition of regime deviations;
- methodical help to operator at normal conditions and at emergenciens;
- preliminary assessment of requlating actions of operator on plant and so on.

Analysis of enumerated tasks shows that for their decision it's necessary to develop expert systems of different classes. It's necessary to develop ES-Advisors capable to operate in real time conditions (for example for dynamic treatment of signals), ES having large knowledges bases (for example to control equipment characteristics) and also ES hybrids where expert knowledge are represented not only in the form of rules but in the form of program modules (to asses requlating actions of operator).

2. In different scientific centres of the USSR including VNIIAES there have been working out projects on developing ES for nuclear industry. VNIIAES has developed expert system named PEX, simple expert shell system being ES of classical type intended for creating system-advisors on different issues connected with operation of unit and separate equipment components (2). The second system is express-diagnostical ES - EDES is created for emergency conditions identification and is one components belonging to system of control and NPP operation conditions diagnostics.

The goal of creating PEX was to develop Expert Systems shell meeting the following requirements:

- the system is to posses all base function and possibilities

that majority of ES has including the ability if knowledge acquisition in the form of facts and rules to make logical concusion;
- to hold active dialogue with the user and to explain its actions;
- knowledge representation in the system should be comprehensive to the terminal user;
- there are to be permitted fuzzy expert assessments in knowledge base facts and rules and logical inference system should be able to manipulate with them;
- and at last operation with the system should be easy to unprepared user and in connection with it there should be developed apparatus of menu, prompts and help and protection from user's errors. Besides there were put additional technological requirements:
- system response time should be convenient for user;
- system should have small volume and should be easily transferred on minicomputers, microcomputers and PC's.

According to these requirements there have been made the following decisions:

The system is realized on programming language Pascal and has open interface with standard DBMS. Facts in the system are dupletes of the type OBJECT-VALUE. The idea of OBJECT is interpreted rather arbitrarily as denomination of object, process or phenomenon. Linguistically object is represented as a rule as subject group for example such expressions as "temperature-of-coolant-in-primary-circuit" or "cause-of-failure" are objects of the system. Such phrases as for example "between-500-and-600-degress" or "corrosive-cracking-of-metal" are also VALUES of objects. Rules in the system have usual form IF {antecedent} THEN {consequent} where antecedent and consequent are conjunctions of the facts. This facts and rules are rather clear to the user however it's rather difficalt to manipulate with them.

For solution of this collision there has found simple but as we think practically justifiable decision. There has been introduced the idea of "permissible" value. Knowledge engineer (this role fulfil now development engineers) introduces into knowledge base permissible values list of some objects according to the agreement with expert (there are as a rule terminal objects, i.e. such objects values of wich can't be inferred out of other rules). Besides question connected with the given object is also introduced into knowledge base. The knowledges permits to introduce into knowledge base the facts not only in the form of direct duplet task but also in the regime for answers on the menu formulated by the system. For example if we connect permissible values "nominal", "above-nominal", "below-nominal" and question "What is the meaning of current pressure in steam generator ? " with the object "pressure-in-steam-generator", then system will give the following question from menu (if it's neccessary to know the meaning of the object in the course of logical inference):

What is the value of current pressure in steam generator?

1. Nominal
2. Above-nominal
3. Below-nominal

In answer user indicates answer's number for example 2. At this the duplet "pressure-in-steam-generator" is "above-nominal" is introduced into knowledge base.

Usage of analogue apparatus of questions and permissible values for special objects (variable) permits the system to organized active dialogue with user not only for terminal objects.

System has the possibility to set truth level (FC - from factor of certainty) for any fact and rule (so-called coefficients of attenuating). Repeated calculation of certainties is beeng done on Shortliff scheme: if fact F is derived in the system with two different certainties: FC1(F)=ALPHA and FC2(F)=BETA then resultant certainty FC(F)=ALPHA + (I - ALPHA)Beta.

Attenuation is obtained in the following way: there is a rule IF A THEN C and A=A1 &...&An and C=C1 &...&Cm and

FC(A(i))=ALPHA(i); i={i:n}

FC (C(j))= BETA(j); j={i:m}

At applying the rule into knowledge base there are introduced facts from consequent having

FC: FC(C(j))=BETA(j)*min ALPHA(i), i={1:n}. It should be noted that the given calcutation factor of certainty level corresponds to human ideas of intuition.

At present on the base of PEX there have been developing two ES-advisor system for MCP failures diagnostics (main circulating pumps) and system-advisor for most typical cases of developing event and equipment and NPP systems states. Fragment of simplified dialogue with PEX system is given on Fig 1.

<u>Condition?</u>
What condition from enumerated ones takes place?
    1. Pressure_rise_in_fience_off
    2. Pressure_drop_in_SG*
    ...   .
<u>1</u>
        condition=measurement_chemistry_regime
<u>Actions?</u>
    To_do_control_measurement_chemistry_regime
<u>CR</u>
        Actions=see list of actions-3
List of action-3?
What_is_chemistry_regime?
    1. Corresponds_to_norm
    2. Doesn't_corresponds_to_norm
<u>2</u>
Correction regime?
    1. Yes
    2. No
<u>2</u>
    List_of_actions-3=1. partly_dismount_fence_of
                      2. grind_out_cracs_of_the_collector
                      3. weld_on_places_of_grinding
                      4. introduce_correction_chemistry_regime_and_meter_out_hydrazine_and_chelatometers
                      5. introduce_ACC**
                      6. introduce_100%_condensate_cleaning

Fragment of simplifield dialogue with PEX system. Information set User questions and answers is underlined.

FIGURE 1.

*SG - Steam Generator
**ACC- automatic chemical control

3. The second system - EDES - is a system of quite another type. It is on-line system for fact automatic identification of emergencies on the possible earliest stage and for representing information about the emergencies (3). Task solved by EDES is a classification task or pattern recognition . Recognition of NPP emergency is characterized by the following features:

1. Necessity of great volume information treatment: the number of emergencies possible types is equal to 100 but the number of sensors is about 10000 and even after eliminating of sensors giving no information it is about 100.
2. Identification should be done in short term due to great velocity of emergencies processes development.
3. Great damage from erroneous or illtimed identification of emergency.
4. Significant uncontrollable error used at information identi fication.

   The error has two sources : error of transducers indication and possible error in reference redimes great part of which is obtained by means of calculations on mathematical model.

In VNIIAES there is mathematical model of physical processes and strenth characteristics for VVER-1000 power units.

The model simulates operation of the whole main unit equipment. In particular the model is used as information source at reference regimes forming. Comparison with these regimes perimits to determine emergency operation type. However due to complexity of mathematical algorythms it's impossible to use the given model directly at regime recognition. Direct realization of these algorythms in on-line diagnostics regime requires computers having two-three orders greater power.

With account of identification task features enumerated above from majiority of possible approaches there has been selected expert system combination building recognition algorythm on the base of expert knowledge about regime features and statistical method of the least damage (according to Bias method) or fuzzy reasoning.

Expert knowledge about emergency operations have been developed on the basis of the following scheme. The full list of emergency operations possible types has been worked out with the help of experts-technologists (it should be noted that the lists worked out by experts of the USSR and Finland correspond with each other that indicates on their sufficientvalidity and completeness). Then after careful analysis of all emergency operatuons made by experts there have been selected the most characteristics parameters (in the theory of patern recognition they are often called symptoms) for every operatio. Experts had unlimited number of characteristic parameters but their number proved to be very small and for different regimes it was equal to 4-8 parameters (different operations, generally speaking are characterized by different sets of parameters).

For describing regime symptoms it's not sufficient to name characteristics parameters, it's necessary to describe their characteristics values or characteristic dynamics of their values. It's impossible to set calculation values of parameters having dinamic indication of sensors due to inaccuracy of source data. For example on Fig.2 there are shown two curves different from each other only by small tension of temporary axis. However in points 1,2 values of

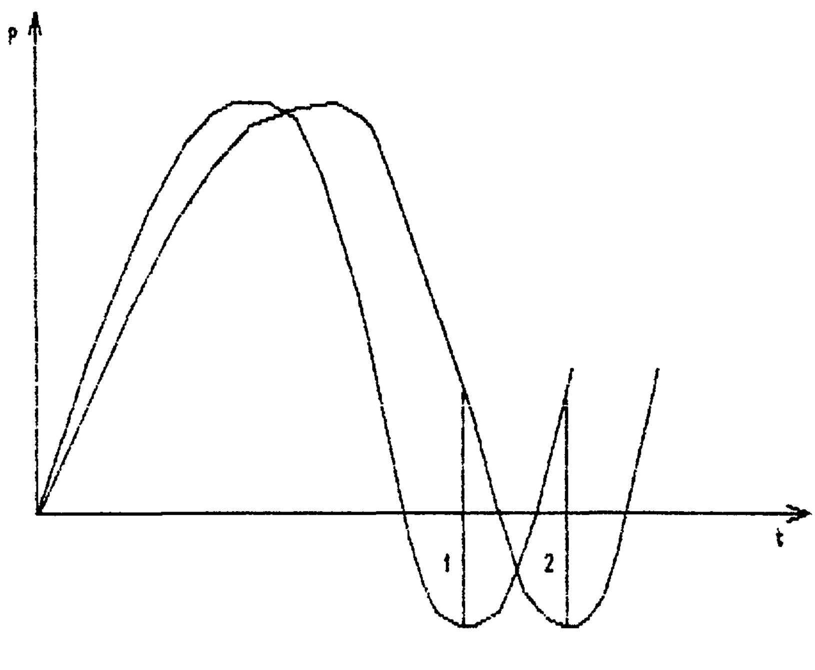

FIGURE 2.

parameters differ even by signs. In order to solve the problem it was necessary to work out adequate language for describing parameters behaviour wich could be insensible to different errors. There has been made analysis of experts reasoning at operations' identification which permitted to fick out six most characteristics types of operation parameters behaviour : rise, drop, local minimum or maximum, sharp rise or sharp drop with stabilization on a new level. Besides the type of parameter behaviour it's characterized by concrete value (velocity, oscillation, value of level). This sufficiently rough language for describing parameters behaviour turns out to be quite adequate and it allows to increase system's performance in terms of error probability and significantly increase its time of response because each type of behaviour is determined on the base of three points not more.

Determining of parameter behaviour belonging to the behaviour type is determined by means of fuzzy sets theory. There has been determined the idea of similarity for parameter behaviour with reperence behaviour. Similarity degree has been determined by the correspondent function of belonging. Typical form of belonging function is shown on Fig. 3. P(i) - calculated - is accurate reference value of parameter. The value of delta(i) is determined with the account of correspondent sensor errors, reference

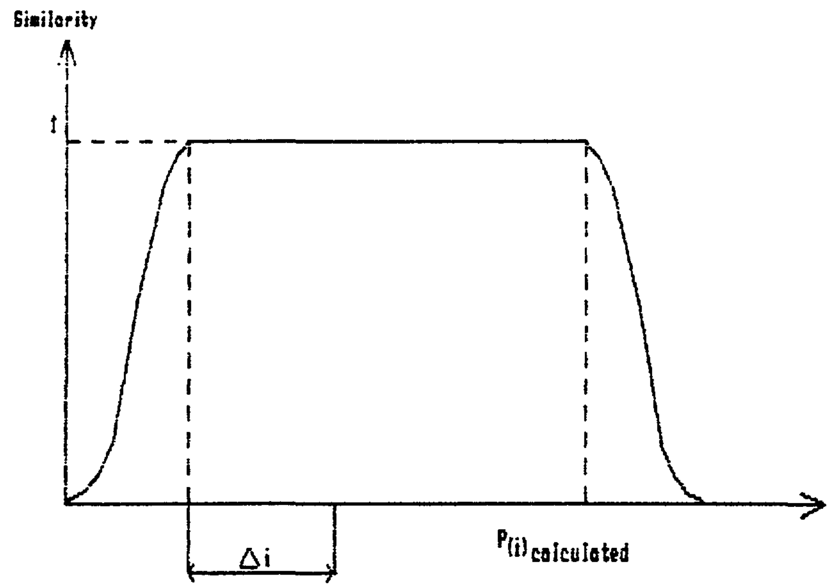

FIGURE 3.

meaning, determining of the time for emergence operation beginning and others. Belonging function form and expert assessment procedure of emergencies were selected in such a way that erroneous identification probability was less than the one given in advance (with the account of signals dispersion and reference operation's error).

Calculation have shown that EDES permits to identification synonymously almost all regimes (list of possible regimes consists out of one-two claimants) right up to 1% error in pressure and 3% error in levels and consumptions. List of claimants containa similar regimes up to error of 2% and 6%. The length of claimants list increases at large error.

To reduce list of claimants on the final stage of identification method of Bies has been used : optimum (in the meaning of minimum mathematical expectation of damage) version of emergency causes is determined on the basis of the diven a priori probabilities of emergency course and forecasted value of damage from incorrect actions.

When completing representation of expert knowledge into EDES it's necessary to indicate one more circumstance that complicating decision of emergency identification task. It's necessary to take into account not only error in sensor's indications but their complete failure. In the system of representing knowledge to EDES it means that for certain emergency one should foresee the possibility of some symptoms nondeveloping. In connection with it we accepted heuristical rule permitting regime identification according to (n-1) symptom where n - general number of symptoms describing emergency. Similar attenuation of identification conditions has led to increasing of the possible information messages number on about 1,5 orders (emergency operation + failure of transducer).

It's clear that direct realization of similar system in real time regime is practically impossible at such volumes of emergency regimes descriptions (they can be interpreted as rules), at such volumes of sensor's indication (they can be interpreted as facts) and at rather complicated algorithms of information analisis and logical inference. It's impossible because time of response on regime change should be equal to approximately 10 seconds. The problem has been escaped by means of dividing EDES into two subsystems : statical and dynamical.

The main task of statical subsystem EDES-S is to build inference tree of claimant diagnostic messages list of the type "emergency operation + failure of transducer" (10000 messages). It's filfilled on the base of the given expert descriptions for emergency regimes(about 100 types). Since the given tree is being built a priori without account of actual indications of transducers, so as delta for belonging function is selected delta(i) = 3 * sigma(i) where sigma(i) - variance of i-th parameter (it's assumed that parameters have normal distribution). Thus while moving along obtained inference tree all alternatives wich don't correspond to definite set of parameters values according to the well-known rule of three sigmas are sifting out. It should be noted that specially developed algorithms for determining dichotomical parameters and their values were used while building the tree. It was necessary in order to reduce to maximum average of steps of inference for the list of claimants-diagnostic messages.

Besides EDES-S represents for user usual for ES set of functions: acgusition and modification of knowledges (in the given case description of emergencies), explanation of actions (in the given case-sequence of steps for determining list of claimants messages).

The task of dynamic subsystem EDES-D is to determine diagnostic message (or several messages but not more than 3) about the most possible type of emergency and about failed sensors. The message is determined from available set of sensors values for n resets. System EDES-D switches on when indication of any asensors exceeds the limits of the setpoint established in advance. Then EDES-D determines list of claimats for relatively small number of steps (the number is about 20-25) by using inference tree which has been built by EDES-S. Since inference tree is building in paradigm "almost likely NO" so naturally list of claimants is sufficiently large. Therefore to reduce the list EDES-D dynamically includes either function calculation of emergency symptoms set belonging (there applied other, more severe belonging functions for fuzzy relation similar where delta 3*sigma) or at once recognition according to Bias.

Offered scheme of EDES according to experiments and preliminary calculations permits to provide sufficiently early and reliable identification of emergencies and thus to increase NPP safety.

In conclusion it should be noted that VNIIAES is going to develop expert systems in two directions : to create systems-advisors on the base of developed shell-systems and to create hybrid logical-calculating systems.

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AN EXPERT SYSTEM FOR DIAGNOSIS OF NUCLEAR POWER PLANT TRANSIENTS AND ACCIDENTS

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Abstract

The paper presents a concept of an expert system to aid operator's of nuclear power plants in diagnosing of safety challenging events when abnormal plant conditions occur.

1. Introduction

The electric utility industry have already made an investment for developing and implementation systems for operator-plant interface. Reactor operators are provided with additional real time information aids, with the aim to increase plant availability and to improve plant safety.

Besides, there are efforts to explore other approche like expert systems application with the same goal to improve plant availability and safety.

Very simply described, an expert system (ES) is an encoding of an expert's knowledge in a particular area of speciality in a computer code. In a different field the expert systems have proved themselves as a powerful tool for solving problems which include data interpretation and diagnosis suggestion.

This paper presents the concept and some idea in developing the expert system for dignosis of nuclear power plant transients which challenge plant safety.

In designing and application of the expert system several steps have to be solved:
1. Expert system conception elaboration
2. Knowledge base design
3. Inference engine design
4. Prototype development and testing
5. Refinement and modification

The first three of the stated problems will be dealt with in the next sections.

2. Conception of the expert system

The main idea is to make use of previously extracted knowledge about plant behaviour during the transients in order to help in diagnosing of some new transients. A lot of calčulation usually were performed for studing the particular plant dinamics in off set condition. These analyses can be used with non conventional computer programs such as expert systems for diagnosing purpose.

Besides knowledge extracted from previously calculated transients this expert system contains the others knowledge about plant such as plant specific, etc. In adition some simple standard computer codes are implemented to help and improve the work of the expert system.

Regarding input data it is assumed that some data (measurements) are available in real time for expert system throu existing electronical acquisation system. User of this expert system has to provide some other input data such as: status of the equipment (for instance on , off) and operator action. However, the intention is to ask the users for a minumum amount of data. Interactively he may be queried for aditional data.

Measurements and status of the equipment are both time depedent data. Once there are obtained there are stored with corespoding time.

The structure of such an expert system is shown on fig. 1.

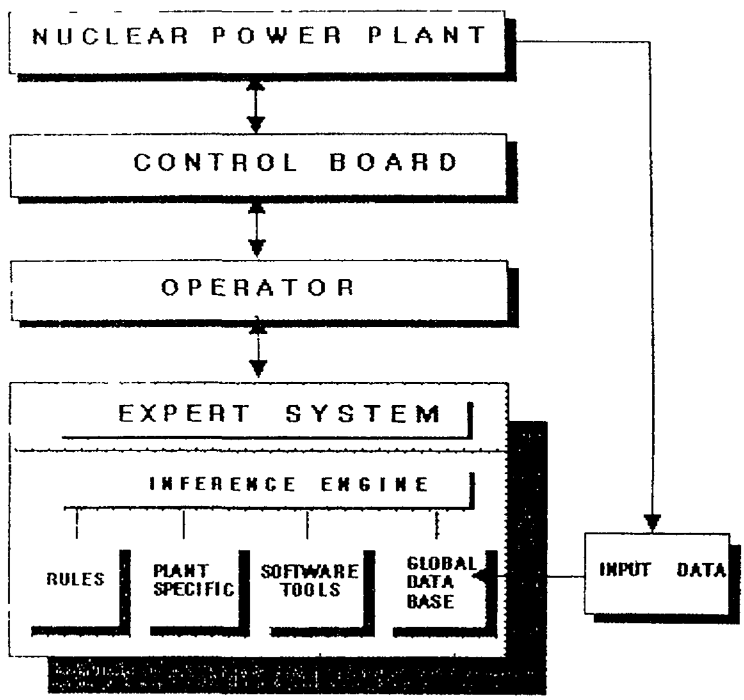

FIG.1. The expert system structure.

3. Knowledge base design

The knowledge base is designed to contain rules, specific knowledge about the plant, simple software programs and global data base.

Rules are long-term information about how to generate new facts and hypotheses from what is presently known. Essentially IF-THEN rules are made from previously analysed transients. Besides that rules are formed after analyses of the system operating states (including failed states) which could affect the transient, and the specific initiating events which could lead to transient.

Specific knowledge about the plant contains nominal values of plant variables under different conditions and plant set points.

Emergency procedures guidelines are part of this knowledge base and they can be displayed by using the same computer system.

Some simple software tools or software programs wich can help in diagnosing proces are contained in knowledge base. For instence, program for calculation of subcooled margin .

Global data base is a temporary store of: short-term information which can change frequently and of all findings derived about specific case. Measurements, equipment status, operator actions are stored here.

4. Inference engine design

Inference engine or control strategy applies the rules in the corect order, deals with data, request additional data or calls procedures.

The design about inference engine design has to be based upon knowledge representation technique applied. In this case, as a method for knowledge presentation first order logic is applied. Therefore, Turbo PROLOG is recognized as very valuable tool for implementation on personal computer.

5. Conclusion

The paper has been written from a conceptual rather then formal standpoint, because it is early stage of the development of the expert system to aid operator's of nuclear power plants in diagnosing of safety challenging events.

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THE USE OF EXPERT SYSTEMS FOR OPERATIONS SUPPORT IN THE CANDU NUCLEAR POWER SUPPLY SYSTEM

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Abstract

The recent emergence of artificial intelligence technology from the laboratory promises to provide new tools for the designers and operators of nuclear power plants to exploit the use of computers. Expert systems, in particular, offer benefits in all phases of plant design and operation.

Atomic Energy of Canada Limited (AECL), the designer of the CANDU* nuclear power supply system, is a pioneer in the application of digital computer control and monitoring in nuclear power plants. It is therfore natural for AECL to examine new techniques for extending the advantages of its current designs. In order to guide the choice of applications and to incorporate operations experience into expert system knowledge bases, AECL is working closely with Canadian CANDU plant owners in this area.

This paper provides an overview of the work which is currently underway and being planned at AECL on the design of expert systems and their integration into CANDU stations. Examples of particular systems which have been developed and are being developed as part of this overall plan will also be discussed.

INTRODUCTION

In the last few years, arificial intelligence (AI) technology has made the move out of the research environment and into commercialization and widespread use in many fields. The vendors and owners of nuclear power plants have realized the potential of this technology to reduce design and operating costs. Benefits are expected to be realized in operations through streamlining plant tasks, increasing plant availablity, and minimizing operator error.

Atomic Energy of Canada Limited (AECL), the designer of the CANDU (CANadian Deuterium Uranium) power plants, is a pioneer in the application of digital control to nuclear power plants.[1-3] The next generation of CANDU plants will extend this through the use of distributed control systems and data highways.[4] In addition, AECL has used computers to implement on-power refuelling, automatic station shutdown and for special maintenance tasks.

* CANadian Deuterium Uranium

AECL intends to use AI technology, particularly expert systems, to assist operations in the next generation of CANDU plants. Operating utilities are already beginning to apply the technology to existing plants. Applications will include the full spectrum of station activities from day to day operations to the management of infrequent events. This will enhance plant and operator reliability, increase plant availability, and decrease the stress on plant staff.[5]

While the gross capacity factors for CANDU power plants are quite high, there is a strong incentive to push them even higher (one percentage point is equivalent to $1.5M Cdn./600 MW/annum). A feasability study has shown that diagnostic expert systems would be an effective method of increasing plant availability.[6]

This paper will describe the the current activities related to expert system development at AECL including a discussion of several applications which have been developed or are under development. In addition, the role of expert systems in future CANDU designs will be discussed.

CURRENT WORK ON EXPERT SYSTEMS AT AECL

In order to gain expertise and experience with AI technology, AECL established a unit called the Knowledge Systems Engineering Group in 1986. This group is designed to be a resource centre to supply expertise for AI projects within AECL and to acquire contracts from external organizations.

AECL has also had consultations with Canadian CANDU owner utilities in order to bring operations experience to bear on the selection and development of applications. A task force on Advanced CANDU Computer Technology has been established with these utilities.

FUELEM

CANDU reactors are designed to be refuelled while operating. Because the fuel is contained in 380 to 400 fuel channels, determining which channel to refuel next is a complex task which requires factoring channel power and fuel burnup distribution, refuelling rates, regional power distribution and recent refuelling history.

AECL has developed an expert system, FUELEM, to pre-select channels which are candidates for refuelling using output from a fuel management computer code that accumulates the detailed operating and refuelling history of a reactor.

Defect Detective

In the rare event of a fuel defect in a CANDU reactor, the use of on-power refuelling and individual fuel channels allows the possibility of removing defective fuel immediately. A prerequisite to removal, however, is an assessment of the seriousness of the defect and an identification of the its location because refuelling a channel will remove some non-defective fuel bundles.

Defect Detective is an expert system developed at AECL to automate and improve the evaluation and location of fuel defects. Since the program can be easily modified to incorporate different evaluation and location criteria, it can be used as a development tool which can quickly assess criteria using historical fuel failure data. The system is in use and is undergoing further development in cooperation with the New Brunswick Electric Power Commission (NBEPC).

Defect Detective is written entirely in Prolog, including inherently procedural parts of the program such as the data input module, the linear regression module, and the plotting module. These modules were implemented very easily in Prolog despite its reputation as a declarative, logic based programming language and they illustrate the power of Prolog as a general purpose programming language.

Plant Configuration and Equipment Status Monitor

Operation of a complex system requires knowledge of the status of the components which comprise the system. A program has been developed which uses a graphical user interface based on actual plant flowsheets to inform operators of current plant status. The system will employ one or more central file servers and several workstations connected by a local area network. The plant database which is maintained by the system contains current component status information, a working history of the components, and plant process parameters. The plant may be viewed on several levels of detail from an overall view to a view of individual components. A demonstration system which illustrates the graphical interface and the network architecture has been developed.

Shell and Tube Heat Exchanger Design Advisor

AECL has developed an expert system to assist in the design of shell and tube heat exchangers. The system considers factors such as working fluid characteristics to provide advice on the selection of components for the heat exchanger.

Pump Seal Replacement Advisor

If a problem is detected with a pump seal in a nuclear power plant, it may be possible to continue normal operation until a scheduled plant outage. The condition of the seal can be determined by conventional data analysis, but determining the expected lifetime of the seal requires judgement and experience. AECL is currently working on an expert system to provide such advice.

Eddy Current Inspection Interpreter

Eddy current inspection of tubing to find incipient defects produces complex two dimensional traces which require experience and skill to interpret. Both quantitative and qualitative aspects of the traces need to be considered in order to determine the type and seriousness of a tube defect. AECL has acquired considerable expertise in this field and is currently developing an expert system to capture this expertise.

PROPOSED EXPERT SYSTEMS FOR CANDU PLANTS

Expert System for Automatic Fault Tree Construction

The production of a Probabilistic Safety Assessment (PSA) for a nuclear plant has been a large undertaking. For example, 85,000 personhours were required for the PSA for the Darlington Generating Station currently under construction in Canada. Some of the labor involved in producing a PSA, such as drawing fault trees, can be automated using conventional software. However, many tasks require expert knowledge. These include the determination of initiating events, the development of event sequences, and the development of fault trees. AECL is considering the use of knowledge based software to handle such tasks. Improvement in efficiency by a factor of between five and seven is predicted.

Operator Companion

Operator aids can be developed for any number of plant functions and subsystems. Such aids form a distributed computing system of satellite expert systems. This distributed system will be coordinated and communicate with the operators through a central program module, or executive. The combination of the executive and the satellites is called Operator Companion.

Each satellite system will contain a component to monitor sensor inputs, an expert system adviser component, a component to communicate with the executive and other satellites and, when appropriate, a component to simulate the function or subsystem for which it is responsible. The plant operators will have access to all of the simulation functions and a complete justification trail for the decisions made by the expert system component.

Since the entire system as described relies heavily on operating experience and encompasses a wide spectrum of plant functions and substems, the development of the system will require cooperation with utilities which operate CANDU plants and the establishment of multi-disciplinary working teams.

Several key functions which Operator Companion will address are the following:

a) Improved Alarm Annunciation Strategy

Annunciation of large numbers of control room alarms is a problem for nuclear plants and also affects petrochemical and process plants. The current CANDU design provides some relief through alarm conditioning, system parameter recorders, classification of major and minor alarms, and sorting alarms by systems. However, expert systems offer more effective ways to reduce alarm flooding and help the operators to diagnose the cause of alarms.

Neural networks[7] are an emerging technology which have great potential in the field of complex pattern recognition. They have two desirable characteristics compared to conventional computer architectures. First, they can recognize patterns from inputs which do not exactly match the patterns they were constructed to recognize. Second, the system can learn by being presented with examples and adjusting its internal parameters to match the examples. This form of machine learning can save a great deal of programmer effort. AECL is monitoring developments in this area and will review the possiblity of using neural networks for parts of the Operator Companion, particularly in dealing with improved alarm annunciation.

b) On-Line Fault Detection and Diagnosis

Presently, fault detection is conveyed to the operator by computerized alarm annunciation. The operator must then often diagnose the root problem using knowledge of the plant and experience. This process can be slow.

Operator Companion will be designed to analyze all available information to identify and diagnose faults. Models will be constructed for each important process system in order to recognize initiating events as opposed to events caused by the initiating events.

c) Plant Configuration and Equipment Status Monitoring

As mentioned above, in the discussion of the demonstration program for implementing this function, the purpose of this function is to provide the operator with on-line information on the physical state of the plant using operational flowsheets.

d) Monitoring Vital Plant Parameters

This function of the system will be designed to monitor vital plant parameters and normal and safety operating limits under all operating conditions. This will reduce the heavy demand placed on the operator during trip or multiple fault conditions when a large volume of data must be considered.

e) Displaying Plant Operating Procedures

Information related to plant operations is distributed among several types of documents, including plant operating manuals, procedure manuals, training manuals, safety analyses, and PSA documents, as well as in the experience of the plant operators. It is important that correct procedures are used during all phases of plant operation.

This function of Operator Companion will consolidate operating procedures in a single knowledge base. It will provide convenient and rapid access to procedures which are relevant to the current plant state and recommend appropriate actions.

Examples of subsystem specific modules which have been proposed for early development are:

a) Fault Diagnosis for Programmable Digital Comparators

Programmable digital comparators are microcomputers which are used in the shutdown systems of some CANDU reactors. Faults in these systems must be diagnosed and repaired in a timely manner since a fault results in an impairment of a shutdown system.

These systems are very reliable so that only a few of the staff in a typical plant are experienced in diagnosing failures. These staff must be called in at unpredictable times to service the PDCs which leads to other work being delayed. An expert system is an ideal way to transfer this expertise to a larger number of personnel. In addition, improved diagnostic capability would reduce unnecessary board swapping during maintenance leading to less wear on components.

This system is unusual for an expert system in that it must co-exist with other programs in a time sliced, Pascal based environment. This is because this environment is used to access the on-line data that the system requires. For this reason, the system is implemented in Pascal using rule-like syntax with boolean valued functions to play the role of logical predicates and a forward chaining inference engine.

This system will be implemented jointly by AECL and NBEPC.

b) On-Line Safety System Impairment Manual

The Safety System Impairment Manual contains the procedures which are to be used by operators to classify impairments to plant safety systems and the procedures which are to be used to respond to the different categories of impairments. In order to follow the procedures, operators must work through a complicated sequence of checks and decisions at the same time that they are trying to remove the cause of the impairment.

An expert system with access to data on the state of the shutdown systems will make the use of these procedures much simpler. This system will also be implemented in the Pascal based time sliced environment.

This system will be implemented jointly by AECL and NBEPC.

c) Condenser Seawater Leak Advisor

Plants which are cooled by seawater require the use of procedures to minimize the effects of corrosion to the condenser, feed train, and steam generators in the event of a seawater leak into the condenser. Depending on the nature of the leak and the operating state of the plant, derating, shutdown, or cooldown may be required. An expert system with access to station chemistry data is capable of working through these procedures much more quickly than the operators in order to recommend the appropriate action. This system will also be implemented in the Pascal based time sliced environment.

This system will be implemented jointly by AECL and NBEPC.

d) Instrument Air System Fault Diagnosis

The instrument air system supplies the pneumatic driving force to operate many plant components. A full or partial loss of instrument air can therefore potentially lead to a requirement for the operators to perform a large number of tasks to determine which equipment is out of service and to compensate for this.

Hydro Quebec intends to build an expert system, AIDE (Air Instrumentation Diagnostic Expert) to diagnose faults in the instrument air system of the Gentilly 2 plant. AECL is taking an active interest in this project and is proposing to take part in the development of this system.

CONCLUSION

Expert Systems are a promising tool for enhancing nuclear plant operating reliability and safety. AECL has begun to use this technology to enhance the performance of existing plants and to integrate the technology into the design of the next generation of CANDU reactors.

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2. J. Pauksens, J.R. Popovic, and S. Groom, "Development of Computerized Man-Machine Interface Systems for CANDU Nuclear Power Plants", Proceedings of the ISA Symposium on New Technologies in Nuclear Power Plants, Instrumentation and Control, Washington, D.C., November 28-30, 1984, pp. 373-384.

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4. G.L. Brooks, "Advances in Commercial Heavy Water Reactor Power Stations", 6th Pacific Basin Conference, Beijing, China, September 7-11, 1987.

5. J.W.D. Anderson, A. Natalizio, and J.E.S. Stevens, "Development in the Application of Knowledge Base Systems in the Canadian Nuclear Industry", ANS Topical Meeting on Artificial Intelligence and Other Innovative Computer Applications in the Nuclear Industry, Snowbird, Utah, August, 1987.

6. J.T. Dunn, J.J.H. Pisett, M.J.F. Notley, and N.J. Spinks, "Future Trends in the Design of CANDU Reactors", AECL-9179, 1986.

7. D.E. Rumelhart et al, "Parallel Distributed Processing", Vols. 1 and 2, The MIT Press, Cambridge, Ma., 1986.

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DOCUMENT NUMBER = INI19:095809

TI: Recent developments in man-machine **systems**. Neue Entwicklungen bei Mensch-Maschine-Systemen.
AU: Johannsen, G. (Kassel Univ. (Gesamthochschule) (Germany, F.R.). Fachbereich 15 - Maschinenbau).
LA: German
JR: Automatisierungstechnik (at). ISSN 0178-2312. CODEN: ATRTE. (1987). v. 35(10) p. 385-395.
AB: The field of man-machine **systems** is introduced with its subareas and a short outline of its history of 45 years. Three current lines of development in university and industrial research are emphasized. Today, the human problem solving activities are experimentally investigated and analytically described more vigorously than the control activities. Further, improved information presentations and decision support are made possible through new technologies of computer graphics and **expert systems**. At last, work on a general design methodology for man-machine **systems** is in progress. The aim is to better support human operators of dynamic technological **systems** as well as designers of graphics for visual display units and of dialogue styles. Thereby, safety and availability of the complete **system** can be increased. (orig.).

DOCUMENT NUMBER = INI19:093385

TI: Serus, an **expert system** for the ultrasonic examination of fuel rods.
AU: Gondard, C.; Papezyk, F.; Wident, P. (and others).
CO: CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. de Technologie.
LA: English
RP: CEA-CONF--9359.
IM: 1987. 7 p. Availability: INIS.
CF: (4. European conference on Nondestructive Testing. London (UK). 13- 18 Sep 1987.)
AB: The use of pattern recognition functions and the modelization of the human **expert** reasoning, allow the automatic identification of defects in welds or structures. The proposed application uses an ultrasonic examination to detect and classify 3 types of defects in end plug welds of PWR fuel rods.

DOCUMENT NUMBER = INI19:093361

TI: An **expert** display **system** and nuclear **power** plant control rooms.
AU: Beltracchi, L. (U.S. Nuclear Regulatory Commission, Washington, DC (USA)).
LA: English
JR: IEEE Trans. Nucl. Sci. ISSN 0018-9499. CODEN: IETNA. (Apr 1988). v. 35(2) p. 991-1000.
AB: An **expert** display **system** controls automatically the display of segments on a cathode ray tube's screen to form an image of plant operations. The image consists of an icon of: 1) the process (heat engine cycle), 2) plant control **systems**, and 3) safety **systems**. A set of data-driven, forward-chaining computer stored rules control the display of segments. As plant operation changes, measured plant data are processed through the rules, and the results control the deletion and addition of segments to the display format. The icon contains information needed by control rooms operators to monitor plant operations. One example of an **expert** display is illustrated for the operator's task of monitoring leakage from a safety valve in a steam line of a boiling water reactor (BWR). In another example, the use of an **expert** display to monitor plant operations during pre-trip, trip, and post-trip operations is discussed as a universal display. The viewpoints and opinions expressed herein are the author's personal ones, and they are not to be interpreted as Nuclear Regulatory Commission criteria, requirements, or guidelines.

* Source: International Nuclear Information System (INIS) of the International Atomic Energy Agency.

DOCUMENT NUMBER = INI19:093338

TI: A proof-of-concept transient diagnostic **expert system** for BWRs / (Boiling Water **Reactors/**).
AU: Yoshida, K.; Naser, J.A.
CO: Electric **Power** Research Inst., Palo Alto, CA (USA).
LA: English
RP: EPRI-NP--5827-SR.
IM: May 1988. 160 p. Availability: Research Reports Center, Box 50490, Palo Alto, Ca 94303.
AB: A proof-of-concept transient diagnostic **expert system** has been developed to identify the cause and the type of an abnormal transient in a boiling water nuclear **power** plant. For this **expert system** development, the calculational results of the simulation code RETRAN were used as the **knowledge** source. The **knowledge** extracted from the RETRAN analyses was transformed into IF-THEN rules in the **knowledge** base for the **expert system.** An important feature of this **expert system** is the introduction of certainty factors to allow diagnosis even in the cases where data may be either missing or marked as invalid. To increase the capability of this diagnostic **system** to distinguish between similiar transients, backward chaining reasoning is used to support the forward chaining reasoning with certainty factors. Through this effort, it has been demonstrated that an **expert system** can be successfully used to create a transient diagnostic **system.** It has also successfully demonstrated that RETRAN can be used as the **knowledge** source for developing the **knowledge** base of the diagnostic **system.**

DOCUMENT NUMBER = INI19:093329

TI: Reactor safety assessment **system.**
AU: Sebo, D.E.; Bray, M.A.; King, M.A.
CO: EG and G Idaho, Inc., Idaho Falls (USA).
LA: English
RP: EGG-M--09487. CONF-870832--11.
IM: 1987. 8 p. Availability: INIS. Available from NTIS, PC A02/MF A01;1 as DE88007732.
CF: (Topical meeting on **artificial intelligence** and other innovative computer applications in the nuclear industry. Snowbird, UT (USA). 31 Aug - 2 Sep 1987.)
AB: The Reactor Safety Assessment **System** (RSAS) is an **expert system** under development for the United States Nuclear Regulatory Commission (USNRC). RSA is designed for use at the USNRC Operations Center in the event of a serious incident at a licensed nuclear **power** plant. RSAS is a situation assessment **expert system** which uses plant parametric data to generate conclusions for use by the NRC Reactor Safety Team. RSAS uses multiple rule bases and plant specific setpoint files to be applicable to all licensed nuclear **power plants** in the United States. RSAS currently covers several generic reactor categories and multiple **plants** within each category.

DOCUMENT NUMBER = INI19:091975

TI: Plant corrosion: prediction of materials performance.
AU: Strutt, J.E.; Nicholls, J.R. (Cranfield Inst. of Tech., Bedford (UK). School of Industrial Science) (eds.).
CO: Institution of Corrosion Science and Technology, Birmingham (UK).
LA: English
IM: Chichester (UK). Ellis Horwood Ltd. 1987. 332 p. ISBN 0-7458- 0200-1. Price Pound 45.00.
AB: Seventeen papers have been compiled forming a book on computer-based approaches to corrosion prediction in a wide range of industrial sectors, including the chemical, petrochemical and **power** generation industries. Two papers have been selected and indexed separately. The first describes a **system** operating within BNFL's Reprocessing Division to predict materials performance in corrosive conditions to aid future plant design. The second describes the truncation of the distribution function of pit depths during high temperature oxidation of a 20Cr austenitic steel in the fuel cladding in AGR **systems.** (U.K.).

DOCUMENT NUMBER = INI19:089946

TI: **Knowledge based systems** in nondestructive evaluation.
AU: Melton, R.B.
CO: Pacific Northwest Lab., Richland, WA (USA).
LA: English
RP: PNL-SA--15670. CONF-8708225--1.
IM: Aug 1987. 7 p. Availability: INIS. Available from NTIS, PC A02/MF A01 as DE88006653.
CF: (NATO advanced research workshop on signal processing and pattern recognition in nondestructive evaluation of materials. Quebec (Canada). 19-22 Aug 1987.)
AB: This paper dicusses the application of **knowledge based systems** technology to problems in nondestructive evaluation (NDE). The Saft Image Interpretation Assistant (SIIA) is used as an illustrative example. SIIA is a prototype **knowledge based system** designed to assist in making the operation of the Synthetic Aperture Focusing Technique (SAFT) Ultrasonic Inspection **System** more reliable and efficient. The two primary motivations for developing SIIA were to insure that the SAFT **system** is used correctly and consistently and to assist in interpreting the results of a SAFT inspection. Our initial goal was to develop a **system** to assist in the interpretation of the images resulting from a SAFT inspection. As we started to identify the structure of the inspection problem, however, we realized **knowledge based system** technology could be more effectively applied to develop a **system** that is in essence an on-line procedure generator that guides a user through a SAFT inspection. When fully developed such a **system** could assist in proper setup of the inspection equipment for each of the steps in a SAFT inspection and in interpreting the inspection results for each step. The first section of the paper describes the structure of the SAFT inspection problem. The next section discusses three forms of **knowledge**: procedural, structural, and inferential as they relate to the SAFT problem. The final section discusses the implications of this type of **system** for other NDE techniques and applications.

DOCUMENT NUMBER = INI19:089476

TI: Vibration monitoring of Kraftwerk Union pressurized water **reactors** - Review, present status, and further development.
AU: Stolben, H.; Wehling, H.J. (Kraftwerk Union AG, Postfach 3220, 8520 Erlangen (Germany, F.R.)).
LA: English
JR: Nucl. Technol. ISSN 0029-5450. CODEN: NUTYB. (Mar 1988). v. 80(3) p. 400-411.
AB: Incipient damage to mechanical structure may be detected early in time by deviations from normal dynamic behavior. For vibration monitoring of coupled **systems**, only a small number of transducers are necessary, in general. On the basis, Kraftwerk Union has been involved in the development and construction of vibration monitoring **systems** for pressurized water **reactors** over the last 20 yr. The current state of the art permits vibration monitoring during normal operation by reactor personnel without **expert** assistance. The new SUS-86 microprocessor-based **system** allows further expansion toward an **expert system**.

DOCUMENT NUMBER = INI19:089363

TI: Processing of alarms by means of an **expert system**.
AU: Legaud, P. (Electricite de France, 75 - Paris. Direction de la Production et du Transport).
LA: English
JR: Reliab. Eng. Syst. Saf. CODEN: RESSE. (1988). v. 22(1-4) p. 401- 409.
NO: Held in conjunction with the 9. international conference on structural mechanics in reactor technology.
CF: (International seminar on accident sequence modeling. Munich (Germany, F.R.). 24-25 Aug 1987.)
AB: After describing the existing alarm **system**, this paper presents a **system** for real time alarm processing using the **expert system** approach. The findings, both at design and performance levels, are described, as well as the immediate perspectives. (author).

DOCUMENT NUMBER = INI19:089331

TI: **Artificial intelligence** and training of nuclear reactor personnel.
AU: Uhrig, R.E.; Buenaflor, M.T. (Univ. of Tennessee, Knoxville (USA)).
LA: English
MS: Proceedings of the CSNI specialist meeting on training of nuclear reactor personnel. Nuclear Energy Agency, 75 - Paris (France). Committee on the Safety of Nuclear Installations.
RP: NUREG/CP--0089.
IM: Dec 1987. p. 225-237. Availability: INIS. NTIS, PC A21/MF A01 - US Govt. Printing Office. as TI88900305.
CF: (CSNI specialists meeting on training of nuclear reactor personnel. Orlando, FL (USA). 21-27 Apr 1987.)
AB: **Expert** computer **systems** offer an excellent and effective means to reduce the potential for operator error, and improve plant safety and reliability. For the training field the benefits are twofold. First, the inclusion of advisory **expert systems** in the control environments (the physical control room and its simulator) offer a continuous source of on-the-job diagnostic training. Second, **expert systems** specifically designed for training are feasible for specialized license/requalification training in higher order analytical skills. This paper consists of two parts. In the first section, the improvements for on-the-job training are examined. In the second section, the benefits for the overall training program are explored in terms of technical and educational rationales.

DOCUMENT NUMBER = INI19:089298

TI: Communications interface for plant monitoring **system**.
AU: Lee, K.L.; Morgan, F.A. (Public Serve Electric and Gas Co., Newark, NJ (USA)).
LA: English
MS: Proceedings: 1986 integrated **power** plant computer communications seminar. Halter, M.; Colley, R.; Divakaruni, S.M. Pacific Gas and Electric Co., San Francisco, CA (USA); Electric **Power** Research Inst., Palo Alto, CA (USA).
RP: EPRI-NP--5641-SR.
IM: Feb 1988. p. 4.7.1-4.7.13. Availability: Research Reports Center, Box 50490, Palo Alto, CA 94303.
CF: (Seminar on integrated **power** plant computer communications. San Francisco, CA (USA). 25-27 Aug 1986.)
AB: This paper presents the communications interface for an intelligent color graphic **system** which PSE and G developed as part of a plant monitoring **system**. The intelligent graphic **system** is designed to off-load traditional host functions such as dynamic graphic updates, keyboard handling and alarm display. The distributed **system's** data and synchronization problems and their solutions are discussed.

DOCUMENT NUMBER = INI19:089255

TI: Intelligent decision support **systems** for nuclear **power plants** in Japan.
AU: Ogino, Takamichi (Mitsubishi Electric Corp., Amagasaki, Hyogo (Japan). Central Research Lab.); Nishizawa, Yasuo (Hitachi Ltd., Ibaraki (Japan). Energy Research Lab.); Morioka, Toshihiko (Toshiba Corp., Isogo, Yokohama (Japan). Isogo Engineering Center); Naito, Norio (Nippon Atomic Industry Group Co. Ltd., Kawasaki, Kanagawa. Nuclear Research Lab.); Tani, Mamoru (Mitsubishi Heavy Industries Ltd., Minato, Tokyo (Japan). Nuclear **Systems** Engineering Dept.); Fujita, Yushi (Mitsubishi Atomic **Power** Industries, Inc., Minato, Tokyo (Japan). Electrical and Control Engineering Dept.).
LA: English
JR: Reliab. Eng. Syst. Saf. CODEN: RESSE. (1988). v. 22(1-4) p. 387- 399.
NO: Held in conjunction with the 9. international conference on structural mechanics in reactor technology.
CF: (International seminar on accident sequence modeling. Munich (Germany, F.R.). 24-25 Aug 1987.)

AB: Projects for decision support **systems** for nuclear **power plants** in Japan are described. In particular the main features of the Man- Machine **System** MMS-NPP are discussed. This is composed of three **systems, Knowledge** Base Management **System,** Operating Method Decision **System** and Man-Machine Communication **system.** (author).

DOCUMENT NUMBER = INI19:089253

TI: Intelligent decision aids for abnormal events in nuclear **power plants.**
AU: Kafka, P.; Polke, H. (Gesellschaft fuer Reaktorsicherheit m.b.H. (GRS), Garching (Germany, F.R.)).
LA: English
JR: Reliab. Eng. Syst. Saf. CODEN: RESSE. (1988). v. 22(1-4) p. 355- 370.
NO: Held in conjunction with the 9. international conference on structural mechanics in reactor technology.
CF: (International seminar on accident sequence modeling. Munich (Germany, F.R.). 24-25 Aug 1987.)
AB: German nuclear **power plants** are characterized by a high degree of automation, not only for normal operation but also for abnormal events. Therefore the role of the operating personnel is mainly a supervisory function. Nevertheless, for a spectrum of unexpected events the operating personnel have to react with manual recovery actions. In order to minimize human error in such recovery actions, different kinds of intelligent decision aid support the operators today. In this paper such aids are discussed and one of them integrated disturbance analysis, IDA, is described in more detail. It is applied in Biblis B reactor to the whole secondary circuit with special account of the condensate from the hotwell to the steam generator including all the support **systems.** (author).

DOCUMENT NUMBER = INI19:089251

TI: A logic flowgraph-based concept for decision support and management of nuclear plant operation.
AU: Guarro, S.B. (Lawrence Livermore National Lab., CA (USA)).
LA: English
JR: Reliab. Eng. Syst. Saf. CODEN: RESSE. (1988). v. 22(1-4) p. 313- 332.
NO: Held in conjunction with the 9. international conference on structural mechanics in reactor technology.
CF: (International seminar on accident sequence modeling. Munich (Germany, F.R.). 24-25 Aug 1987.)
AB: The architecture of an automated decision support **system** for nuclear plant operators is presented and discussed. The **system** is **based** on the use of 'logic flowgraph' process models and is designed in a hierarchical fashion. Its functionality spans from 'function oriented' plant status and alternative success path information displayed to the plant operators at its higher access levels to 'process oriented' diagnostic and recovery information deduced and displayed at its lowest. The design basis for this architecture is the 'defense in depth' plant safety concept. The decision support **system** goal is to provide plant operators, in the presence of an unforeseen transient, with the best and safest alternative between plant stabilization after shutdown and recovery of normal operation **based** on early diagnosis. Examples of the **system** capability to interpret and diagnose abnormal plant conditions and of the information that it can supply to the operators at its three access levels are presented and discussed. (author).

DOCUMENT NUMBER = INI19:089243

TI: Modeling human intention formation for human reliability assessment.
AU: Woods, D.D.; Roth, E.M. (Westinghouse Electric Corp., Pittsburgh, PA (USA). Research and Development Center); Pople, H. Jr. (Pittsburgh Univ., PA (USA); Seer **Systems,** Inc., Pittsburgh, PA (USA)).
LA: English
JR: Reliab. Eng. Syst. Saf. CODEN: RESSE. (1988). v. 22(1-4) p. 169- 200.
NO: Held in conjunction with the 9. international conference on structural mechanics in reactor technology. Contract NRC-04-85-103.

CF: (International seminar on accident sequence modeling. Munich (Germany, F.R.). 24-25 Aug 1987.)

AB: This paper describes a dynamic simulation capability for modeling how people form intentions to act in nuclear **power** plant emergency situations. This modeling tool, Cognitive Environment Simulation or CES, was developed **based** on techniques from **artificial intelligence.** It simulates the cognitive processes that determine situation assessment and intention formation. It can be used to investigate analytically what situations and factors lead to intention failures, what actions follow from intention failures (e.g. errors of omission, errors of commission, common mode errors), the ability to recover from errors or additional machine failures, and the effects of changes in the NPP person machine **system.** One application of the CES modeling environment is to enhance the measurement of the human contribution to risk in probabilistic risk assessment studies. (author).

DOCUMENT NUMBER = INI19:084742

TI: **Expert systems** for space **power** supply: design, analysis, and evaluation.

AU: Cooper, R.S.; Thomson, M.K.; Hoshor, A. (Apogee Research Corp., San Diego, CA (USA)).

LA: English

MS: Space nuclear **power systems,** 1986: Volume 5. El-Genk, M.S.; Hoover, M.D. (eds.). New Mexico Univ., Albuquerque (USA). Inst. for Space Nuclear **Power** Studies; Lovelace Biomedical and Environmental Research Inst., Albuquerque, NM (USA). Inhalation Toxicology Research Inst.

IM: Malabar, FL (USA). ORBIT Book Company, Inc. 1987. p. 259-266.

CF: (3. symposium on space nuclear **power systems.** Albuquerque, NM (USA). 13-16 Jan 1986.)

AB: The authors evaluated the feasibility of applying **expert systems** to the conceptual design, analysis, and evaluation of space **power** supplies in particular, and complex **systems** in general. To do this, they analyzed the space **power** supply design process and in associated **knowledge** base, and characterized them in a form suitable for computer emulation of a human **expert.** The existing **expert system** tools and the results achieved with them were evaluated to assess their applicability to **power system** design. They applied some new concepts for combining program architectures (modular **expert systems** and algorithms) with information about the domain to create a deep **system** for handling the complex design problem. They authors developed, programmed and tested NOVICE, a code to solve a simplified version of a scoping study of a wide variety of **power** supply types for a broad range of missions, as a concrete feasibility demonstration.

DOCUMENT NUMBER = INI19:084483

TI: Optimization of fuel exchange machine operation for boiling water **reactors** using an **artificial intelligence** technique.

AU: Sekimizu, K.; Araki, T.; Tatemichi, S.I. (Nippon Atomic Industry Group Co., Ltd., **System** Analysis Dept., 4-1, Ukishima-cho, Kawasaki- ku, Kawasaki City, Kanagawa Prefecture 210 (Japan)).

LA: English

JR: Nucl. Technol. ISSN 0029-5450. CODEN: NUTYB. (Nov 1987). v. 79(2) p. 196-209.

AB: Optimization of fuel assembly exchange machine movements during periodic refueling outage is discussed. The fuel assembly movements during a fuel shuffling were examined, and it was found that the fuel assembly movements consist of two different movement sequences;one is the "PATH," which begins at a discharged fuel assembly and terminates at a fresh fuel assembly, and the other is the "LOOP," where fuel assemblies circulate in the core. It is also shown that fuel-loading patterns during the fuel shuffling can be expressed by the state of each PATH, which is the number of elements already accomplished in the PATH actions. **Based** on this fact, a scheme to determine a fuel assembly movement sequence within the constraint was formulated using the **artificial intelligence** language PROLOG. An additional merit to the scheme is that it can simultaneously evaluate fuel assembly movement, due to the control rods and local **power** range monitor exchange, in addition to normal fuel shuffling. Fuel assembly movements, for fuel shuffling in a 540-MW(electric) boiling water reactor **power** plant, were calculated by this scheme. It is also shown that the true optimization to minimize the fuel exchange machine movements would be costly to obtain due to the number of alternatives that would need to be evaluated. However, a method to obtain a quasi-optimum solution is suggested.

DOCUMENT NUMBER = INI19:084418

TI: How **artificial intelligence** can help /(man-machine interface/).
AU: Elm, W.C. (Westinghouse Electric Corp., Pittsburgh, PA (USA)).
LA: English
JR: Nucl. Eng. Int. ISSN 0029-5507. CODEN: NEINB. (May 1988). v. 33(406) p. 36-40.
AB: The operator is ultimately responsible for the safe and economical operation of the plant, and must evaluate the accuracy of any **system**- recommended action or other output. Decision support **systems** offer a means to improve the man-machine interface by explicitly supporting operator problem solving, rather than complicating decision-making by the need to request an explanation of the rationale behind an **expert system**'s advice during a high stress situation. (author).

DOCUMENT NUMBER = INI19:084415

TI: Method for automatic control rod operation using rule-based control.
AU: Kinoshita, Mitsuo; Yamada, Naoyuki; Kiguchi, Takashi (Hitachi Ltd., Ibaraki (Japan). Energy Research Lab.).
LA: Japanese
JR: Nippon Genshiryoku Gakkai-Shi. ISSN 0004-7120. CODEN: NGEGA. (Mar 1988). v. 30(3) p. 276-285.
AB: An automatic control rod operation method using rule-based control is proposed. Its features are as follows: (1) a production **system** to recognize plant events, determine control actions and realize fast inference (fast selection of a suitable production rule), (2) use of the fuzzy control technique to determine quantitative control variables. The method's performance was evaluated by simulation tests on automatic control rod operation at a BWR plant start-up. The results were as follows; (1) The performance which is related to stabilization of controlled variables and time required for reactor start-up, was superior to that of other methods such as PID control and program control methods, (2) the process time to select and interpret the suitable production rule, which was the same as required for event recognition or determination of control action, was short (below 1 s) enough for real time control. The results showed that the method is effective for automatic control rod operation. (author).

DOCUMENT NUMBER = INI19:083177

TI: Applying IT to accident management: the Nordic project.
AU: Andersen, V. (Risoe National Lab., Roskilde (Denmark)).
LA: English
JR: Nucl. Eng. Int. ISSN 0029-5507. CODEN: NEINB. (Apr 1988). v. 33(405) p. 65-66.
AB: The NKA/INF project, a Nordic collaborative research effort, is examining ways in which modern information technology, including **expert systems**, can help in the hugely complex task of handling a major emergency. (author).

DOCUMENT NUMBER = INI19:078829

TI: Optimized signal monitoring for boiling experiments in the KNK-2 reactor. Optimierte Signalueberwachung bei Siedeexperimenten am Schnellen Brueter.
AU: Scherer, K.P. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Datenverarbeitung in der Technik).
LA: German
MS: Annual meeting on nuclear technology '88. Proceedings. Session 3: Safety of nuclear facilities. Jahrestagung Kerntechnik '88. Tagungsbericht. Sektion 3: Sicherheit kerntechnischer Anlagen. Deutsches Atomforum e.V., Bonn (Germany, F.R.); Kerntechnische Gesellschaft e.V., Bonn (Germany, F.R.).

IM: Bonn (Germany, F.R.). INFORUM Verl. May 1988. 789 p. p. 215- 218.
CF: (Annual meeting on nuclear technology (JK '88). Luebeck-Travemuende (Germany, F.R.). 17-19 May 1988.)
AB: Published in summary form only.

DOCUMENT NUMBER = INI19:078679

TI: French operators get EXTRA **expert** help /(a new **expert system**/).
AU: Anon.
LA: English
JR: Nucl. Eng. Int. ISSN 0029-5507. CODEN: NEINB. (Mar 1988). v. 33(404) p. 24.
AB: Electricite de France is to put its new **expert system**, EXTRA into real time operation at the Bugey station this year, and is considering extending EXTRA to the alarm **system** in all its nuclear stations. EXTRA helps operators to respond faster and more effectively to alarms. (U.K.).

DOCUMENT NUMBER = INI19:078569

TI: Proceedings of the US Nuclear Regulatory Commission fifteenth water reactor safety information meeting: Volume 5, Industry safety research, International Code Assessment Program.
AU: Weiss, A.J. (comp.).
CO: Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research.
LA: English
RP: NUREG/CP--0091-Vol.5. CONF-8710111--Vol.5.
IM: Feb 1988. 469 p. Availability: INIS. Available from NTIS, PC A20/ MF A01; 1 as TI88006492.
NO: Portions of this document are illegible in microfiche products.
CF: (15. water reactor safety information meeting. Gaithersburg, MD (USA). 26-30 Oct 1987.)
AB: This six-volume report contains 140 papers out of the 164 that were presented at the Fifteenth Water Reactor Safety Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, during the week of October 26-29, 1987. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. This report, volume 5, discusses Industry Safety Research and the International Code Assessment Program. Twenty reports have been cataloged separately.

DOCUMENT NUMBER = INI19:078500

TI: Development of advanced methods for signal processing in the monitoring of sodium-cooled **reactors**. Entwicklung fortschrittlicher Verfahren der Signalverarbeitung bei der Ueberwachung Na-gekuehlter Reaktoren.
AU: Schleisiek, K.; Aberle, J.; Massier, H.; Scherer, K.P.; Vaeth, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)); Leder, H.J.; Schade, H.J. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)).
LA: German
MS: Man and the chip in nuclear engineering - techniques in information processing. Proceedings. Mensch und Chip in der Kerntechnik - Techniken der Informationsverarbeitung. Berichtsband. Deutsches Atomforum e.V., Bonn (Germany, F.R.).
IM: Bonn (Germany, F.R.). INFORUM Verl. Dec 1987. 500 p. p. 462- 482. ISBN 3-926956-01-1.
CF: (Technical meeting and exhibition of Deutsches Atomforum e.V.: Man and the chip in nuclear engineering - techniques in information processing. Bonn (Germany, F.R.). 27-28 Oct 1987.)
AB: Selected examples (acoustic boiling detection, pattern recognition method, identification of fuel element vibrations, diagnosis **system** for KNK II) are used to

demonstrate the benefits of up-to-date information technology in the monitoring of nuclear facilities. The methods used range from intelligent frequency analysis to AI methods like pattern recognition and **expert systems.** (DG).

DOCUMENT NUMBER = INI19:078479

TI: An approach to build a **knowledge** base for reactor accident diagnostic **expert system** disket.
AU: Aoyagi, Toshihiko (Kyushu Railway Co., Fukuoka (Japan). Management Administration Dept.); Yoshida, Kazuo; Hirota, Yasuhiro; Fujiki, Kazuo; Kohsaka, Atsuo.
CO: Japan Atomic Energy Research Inst., Tokyo.
LA: Japanese
RP: JAERI-M--88-044.
IM: Mar 1988. 63 p. Availability: INIS.
AB: In the development of a rule **based expert system**, one of key issues is how to acquire **knowledge** and to build **knowledge** base (KB). On building the KB of DISKET, which is an **expert system** for nuclear reactor accident diagnosis developed in JAERI, several problems have been experienced as follows. To write rules is a time consuming task, and it is difficult to keep the objectivity and consistency of rules as the number of rules increase. Further, certainty factors (CFs) must be often determined according to engineering judgement, i.e. empirically or intuitively. A **systematic** approach was attempted to handle these difficulties and to build an objective KB efficiently. The approach described in this report is **based** on the concept that a prototype KB, colloquially speaking "an initial guess", should first be generated in a **systematic** way and then is to be modified and/or improved by human experts for practical use. Statistical methods, principally Factor Analysis, were used as the **systematic** way to build a prototype KB for the DISKET using a PWR plant simulator data. The source information is a number of data obtained from the simulation of transients, such as the status of components and annunciators etc., and major process parameters like pressures, temperatures and so on. The results of diagnoses shows that the statistical method, Factor Analysis, is powerful for building a prototype of **knowledge** base of an **expert system** for reactor accident diagnosis like DISKET. (author).

DOCUMENT NUMBER = INI19:076763

TI: Evolution of emergency response facility data acquisition and display **systems.**
AU: Vroman, V.; Michelbacher, B. (Energy Inc., Idaho Falls, ID).
LA: English
MS: Proceedings: 1986 seminar on emergency response facilities and implementation of safety parameter display **systems.** Divakaruni, S.M. (ed.). Electric **Power** Research Inst., Palo Alto, CA (USA).
RP: EPRI-NP--5510-SR.
IM: Nov 1987. p. 2A.5.1-2A.5.13. Availability: Research Reports Center, Box 50490, Palo Alto, CA 94303.
CF: (Seminar on emergency response facilities and implementation of safety parameter display **systems.** Boston, MA (USA). 6-8 May 1986.)
AB: Since the requirement for an Emergency Response Facility Data Acquisition and Display **System** (ERFDADS) was imposed on the utilities by the Nuclear Regulatory Commission, significant advances in these **systems** have come about. **Systems** specified in the early 1980s generally contained only the base requirements called out by NUREG-0696. Specifications have evolved to the current standards for integration of plant process computer functions and plant performance maximization programs with these **systems.** These new requirements necessitated changes in **system** hardware and software configurations. Trends have been toward distributed **systems** with powerful hosts and advanced data processing and display presentation capabilities. Future trends are toward advancing the distributed processing concept with the use of microprocessors and minicomputers. With the use of **expert systems,** ERFDADS will likely become a more useful tool to plant operators during both normal operations and transient conditions.

DOCUMENT NUMBER = INI19:076748

TI: Use of probabilistic risk assessment (PRA) in **expert systems** to advise nuclear plant operators and managers.
AU: Uhrig, R.E.
CO: Oak Ridge National Lab., TN (USA); Tennessee Univ., Knoxville (USA). Dept. of Nuclear Engineering.
LA: English
RP: CONF-880464--1.
IM: 1988. 6 p. Availability: INIS. Available from NTIS, PC A02; 3 as DE88005437.
NO: Paper copy only, copy does not permit microfiche production.
CF: (Applications of **artificial intelligence** conference. Orlando, FL (USA). 4-8 Apr 1988.)
AB: The use of **expert systems** in nuclear **power plants** to provide advice to managers, supervisors and/or operators is a concept that is rapidly gaining acceptance. Generally, **expert systems** rely on the expertise of human experts or **knowledge** that has been modified in publications, books, or regulations to provide advice under a wide variety of conditions. In this work, a probabilistic risk assessment (PRA)"3 of a nuclear **power** plant performed previously is used to assess the safety status of nuclear **power plants** and to make recommendations to the plant personnel. 5 refs., 1 fig., 2 tabs.

DOCUMENT NUMBER = INI19:076747

TI: Simulation-based **expert system** for nuclear-power-plant diagnostics.
AU: Hassberger, J.A.
CO: Michigan Univ., Ann Arbor (USA).
LA: English
IM: 1986. 119 p. Availability: University Microfilms Order No. 87- 02,743.
NO: Designation: Thesis (Ph. D.).
AB: Applications of **expert systems** to the diagnostics of nuclear **power** plant accidents is considered. In this work, dynamic simulators, Kalman filtering, pattern recognition, fuzzy diagnostics and **artificial intelligence** are combined in a unique algorithm for diagnosing and analyzing nuclear plant transients targeted for use on-line and in real time. Knowledge-based reasoning is used to monitor plant data and hypothesize about the status of the plant. Fuzzy logic is employed as the inferencing mechanism and an implication scheme **based** on observations is developed and employed to handle scenarios involving competing failures. Hypothesis testing is performed by simulating the behavior of faulted components using numerical models. A simulation filter was developed **based** on the structure of the Kalman filter for **systematically** adjusting key model parameters to force agreement between the simulation and actual plant data. The unique feature of the simulation filter is that it operates only on the discrete time- series of inputs and associated outputs of a dynamic simulation program, thus admitting arbitrary **system** dynamics and being readily applicable to any **system** for which a simulation program for computing **system** states is available. Detailed simulation results of various nuclear **power** plant accident scenarios are presented to demonstrate the performance and robustness properties of the diagnostic algorithm developed.

DOCUMENT NUMBER = INI19:073244

TI: Smart sensor application, to nuclear plant thermocouple channels.
AU: Cannon, N.S.; Martin, J.D.; Patterson, B.A.; Stringer, J.L.; Zimmerman, B.D.; Heinisch, H.L.; Cannon, C.P.
CO: Westinghouse Hanford Co., Richland, WA (USA); Pacific Northwest Lab., Richland, WA (USA); Cannon Technology, Kennewick, WA (USA).
LA: English
RP: WHC-SA--0015. CONF-870832--10.
IM: 1987. 12 p. Availability: INIS. Available from NTIS, PC A03/MF A01 as DE88004470.
CF: (Topical meeting on **artificial intelligence** and other innovative computer applications in the nuclear industry. Snowbird, UT (USA). 31 Aug - 2 Sep 1987.)
AB: The application of smart sensor technology to thermocouple (TC) channels at the Fast Flux Test Facility (FFTF) has been accomplished. The objective of this work was to

provide a first step toward the development of nuclear plant sensor **systems** that not only produce a transducer output signal, but also perform diagnostics on the sensor channel to provide signal validation. When TC impairment was detected, the **system** provided an analysis of the severity of the damage, its location, and probable cause. Two **systems** were constructed: a laboratory **system** for development purposes, and a follow-on **system** for installation at FFTF. Since its installation, the FFTF **system** continues to provide TC trending data for lifetime predictions, and a test facility for new concepts. The laboratory **system** has been adapted to include a new sensor type, an FFTF low-level neutron flux monitor.

DOCUMENT NUMBER = INI19:073111

TI: The structure of an **expert system** to diagnose and supply a corrective procedure for nuclear **power** plant malfunctions.
AU: Hajek, B.K.; Stasenko, J.E.; Hashemi, S.; Bhatnagar, R.; Punch, W.F. III; Yamada, N.
CO: Ohio State Univ., Columbus (USA); Hitachi Ltd., Ibaraki (Japan). Energy Research Lab.
LA: English
RP: CONF-870832--9.
IM: 1987. 8 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE88004919.
NO: Portions of this document are illegible in microfiche products.
CF: (Topical meeting on **artificial intelligence** and other innovative computer applications in the nuclear industry. Snowbird, UT (USA). 31 Aug - 2 Sep 1987.)
AB: During the past two years, two prototype **knowledge based systems** have been developed at the Ohio State University. These **systems** were the result of collaboration between the Nuclear Engineering Program and the Laboratory for **Artificial Intelligence** Research (LAIR). The first **system** uses hierarchical classification to diagnose malfunctions of the coolant **system** in a General Electric Boiling Water Reactor (BWR). The second **system** provides a plan of action, through a process of dynamic procedure management, to stabilize the plant once an abnormal transient has occurred. The objective of this paper is to discuss the structure that has been designed to integrate the two **systems.** The combined **system** will be capable of informing plant personnel about the nature of malfunctions, and of supplying to the operator the most direct corrective procedure available. Two important features of the integrated **system** are faulty sensor detection, **based** on malfunction context and unlike sensor data, and procedure management **based** on the initial state of the plant. Since the two **knowledge based systems** were developed separately, the integration has required a separate component currently under development, the Plant Status Monitoring **System** (PSMS). The task of PSMS is to monitor plant parameters in order to detect an abnormal condition developing within the plant. **Based** on the nature of the event, PSMS is capable of directing control to either the procedure management or diagnosis component. The integrated **system** plays only an advisory role, and any suggested action would be executed by the plant personnel.

DOCUMENT NUMBER = INI19:072272

TI: An **expert system** for sensor data validation and malfunction detection.
AU: Hashemi, S.; Hajek, B.K.; Miller, D.W.
CO: Ohio State Univ., Columbus (USA).
LA: English
RP: CONF-870832--8.
IM: 1987. 7 p. Availability: INIS. Available from NTIS, PC A02; 3 as DE88004920.
NO: Paper copy only, copy does not permit microfiche production.
CF: (Topical meeting on **artificial intelligence** and other innovative computer applications in the nuclear industry. Snowbird, UT (USA). 31 Aug - 2 Sep 1987.)
AB: Nuclear **power** plant operation and monitoring in general is a complex task which requires a large number of sensors, alarms and displays. At any instant in time, the operator is required to make a judgment about the state of the plant and to react accordingly. During abnormal situations, operators are further burdened with time constraints. The possibility of an undetected faulty instrumentation line, adds to the complexity of operators' reasoning tasks. Failure of human operators to cope with the conceptual complexity of abnormal situations often leads to more serious

malfunctions and further damages to plant (TMI-2 as an example). During these abnormalities, operators rely on the information provided by the plant sensors and associated alarms. Their usefulness however, is quickly diminished by their large number and the extremely difficult task of interpreting and comprehending the information provided by them. The need for an aid to assist the operator in interpreting the available data and diagnosis of problems is obvious. Recent work at the Ohio State University Laboratory of **Artificial Intelligence** Research (LAIR) and the nuclear engineering program has concentrated on the problem of diagnostic **expert systems** performance and their applicability to the nuclear **power** plant domain. There has also been concern about the diagnostic **expert systems** performance when using potentially invalid sensor data. Because of this research, an **expert system** has been developed that can perform diagnostic problem solving despite the existence of some conflicting data in the domain. This work has resulted in enhancement of a programming tool, that allows domain experts to create a diagnostic **system** that will be to some degree, tolerant of bad data while performing diagnosis. This **expert system** is described here.

DOCUMENT NUMBER = INI19:069506

TI: Engineering Physics and Mathematics Division progress report for period ending September 30, 1987.
CO: Oak Ridge National Lab., TN (USA).
LA: English
RP: ORNL--6425.
IM: Dec 1987. 269 p. Availability: INIS. Available from NTIS, PC A12/ MF A01; 1 as DE88006141.
NO: Portions of this document are illegible in microfiche products. Original copy available until stock is exhausted.
AB: This report provides an archival record of the activities of the Engineering Physics and Mathematics Division during the period June 30, 1985 through September 30, 1987. Work in Mathematical Sciences continues to include applied mathematics research, statistics research, and computer science. Nuclear-data measurements and evaluations continue for fusion **reactors**, fission **reactors**, and other nuclear **systems**. Also discussed are long-standing studies of fission-reactor shields through experiments and related analysis, of accelerator shielding, and of fusion-reactor neutronics. Work in Machine **Intelligence** continues to feature the development of an autonomous robot. The last descriptive part of this report reflects the work in our Engineering Physics Information Center, which again concentrates primarily upon radiation-shielding methods and related data.

DOCUMENT NUMBER = INI19:068713

TI: The development of an **expert system** for defect identification and its assessment.
AU: Okamoto, A.; Kataoka, S.; Watanabe, M. (Ishikawajima-Harima Heavy Industries Co. Ltd., Yokohama (Japan)); Miyoshi, S.; Kurokawa, A. (Japan **Power** Engineering and Inspection Corp., Tokyo).
LA: English
MS: Transactions of the 9th international conference on structural mechanics in reactor technology. Vol. G. Fracture mechanics and NDE. Wittmann, F.H. (ed.).
IM: Rotterdam (Netherlands). Balkema. 1987. 575 p. p. 247-252. ISBN 90-6191-768-9.
CF: (9. biennial international conference on structural mechanics in reactor technology (SMIRT-9). Lausanne (Switzerland). 17-21 Aug 1987.)
AB: A set of parallel works has been performed in Japan. The project includes the development of advanced NDE devices using computer tomography, supersonic wave holography and electromagnetic supersonic wave and also the development of software for the evaluation. Defect Identification Program (DIP), the **expert system** being described, is one of the products of the coorperative works. The program is designed to infer the cause and the kind of the defect found in ISI, and assess the defect for further operation in accordance with the flaw evaluation criteria. The development of an **expert system** has been performed which is used for the inference of the cause and the kind of the defect found in ISI. The prototype (SSC in a BWR) of the computer program is accomplished. An actual cracking incident was used as a

sample problem to show the effectiveness of the prototype. The result shows that the program can be a useful tool to identify the inspected crack in nuclear **power plants.** (orig./HP).

DOCUMENT NUMBER = INI19:068629

TI: Development of **expert system** for abnormal-event recurrence prevention.
AU: Nishiyama, Takuya; Shinohara, Yasushi (Central Research Inst. of Electric **Power** Industry, Tokyo (Japan)).
LA: Japanese
JR: Nippon Genshiryoku Gakkai-Shi. ISSN 0004-7120. CODEN: NGEGA. (Apr 1987). v. 29(4) p. 268-272.
AB: As for the **expert system** for helping to prevent the recurrence of accidents and troubles in nuclear **power plants,** of which the development has been advanced by the Central Research Institute of Electric **Power** Industry, it is considered to show its features, and the background of the development, the role and the outline of the function are reported. As to the background of the development, it is mentioned that this **system** can effectively utilize the specific character of **knowledge** engineering. As to the role, its features are shown by placing emphasis on the comparison of this **system** with a so-called abnormality diagnosis **system.** As to the outline of the function, the fundamental function (the elucidation of factors, the forecast of influence, the forecast of similar events, the evaluation of importance, the planning of countermeasures for preventing beforehand) and interface function, that this **system** executes, are explained. This Consultation **System** for Prevention of Abnormal Event Recurrence extracts the **knowledge** and know-how which serve to prevent the recurrence of similar abnormal events from the information obtained through information acquisition and transmission **systems,** puts in order and stores them, and offers to users any time by processing them into required forms. (Kako, I.).

DOCUMENT NUMBER = INI19:068624

TI: Employing **expert systems** for process control. Einsatz von Expertensystemen in der Prozessleittechnik.
AU: Ahrens, W. (Bayer A.G., Leverkusen (Germany, F.R.). Ingenieurbereich Prozessleittechnik).
LA: German
JR: Autom.tech. Prax. (atp). ISSN 0178-2320. CODEN: ARTPE. (1987). v. 29(10) p. 475-485.
AB: The characteristic features of **expert systems** are explained in detail, and the **systems'** application in process control engineering. Four points of main interest are there, namely: Applications for diagnostic tasks, for safety analyses, planning, and training and **expert** training. For the modelling of the technical **systems** involved in all four task fields mentioned above, an object-centred approach has shown to be the suitable method, as process control techniques are determined by technical objects that in principle are specified by data sheets, schematic representations, flow charts, and plans. The graphical surface allows these data to be taken into account, so that the object can be displayed in the way best suited to the individual purposes. (orig./GL).

DOCUMENT NUMBER = INI19:068606

TI: Man-machine communication in reactor control using AI methods.
AU: Klebau, J.; Lindner, A.; Fiedler, U. (Akademie der Wissenschaften der DDR, Zentralinstitut fuer Kernforschung Rossendorf, Dresden (German Democratic Republic)).
LA: English
MS: IAEA, NPPCI specialists' meeting on the human factor information feedback in nuclear **power:** implications of operating experience on **system** analysis, design and operation. International Atomic Energy Agency, Vienna (Austria). International Working Group on Nuclear **Power** Plant Control and Instrumentation.
IM: Roskilde (Denmark). Risoe National Laboratory, Department of Computer and Information Science. May 1987. 379 p. p. 343-354. Available on loan from Risoe Library, DK-4000 Roskilde.

CF: (IAEA, NPPCI specialists' meeting on the human factor information feedback in nuclear **power:** implications of operating experience on **system** analysis, design and operation. Roskilde (Denmark). 26-28 May 1987.)

AB: In the last years the interest in process control has expecially focused on problems of man-machine communication. It depends on its great importance to process performance and user acceptance. Advanced computerized operator aids, e.g. in nuclear **power plants,** are as well as their man-machine interface. In the Central Institute for Nuclear Research in Rossendorf a computerized operator support **system** for nuclear **power plants** is designed, which is involved in a decentralized process automation **system.** A similar but simpler **system,** the Hierarchical Informational **System** (HIS) at the Rossendorf Research Reactor, works with a computer controlled man-machine interface, **based** on menu. In the special case of the disturbance analysis program SAAP-2, which is included in the HIS, the limits of menu techniques are obviously. Therefore it seems to be necessary and with extended hard- and software possible to realize an user controlled natural language interface using **Artificial Intelligence** (AI) methods. The draft of such a **system** is described. It should be able to learn during a teaching phase all phrases and their meanings. The **system** will work on the basis of a self-organizing, associative data structure. It is used to recognize a great amount of words which are used in language analysis. Error recognition and, if possible, correction is done by means of a distance function in the word set. Language analysis should be carried out with a simplified word class controlled functional analysis. With this interface it is supposed to get experience in intelligent man-machine communication to enhance operational safety in future. (author).

DOCUMENT NUMBER = INI19:064090

TI: Waste management '87: Waste isolation in the US, technical programs and public education.

AU: Post, R.G.

LA: English

RP: CONF-870306--.

IM: Tucson, AZ (USA). University of Arizona Nuclear Engineering Dept. 1987. 823 p.

SE: Volume 3 - Low-level waste.

CF: (Waste management '87. Tucson, AZ (USA). 1-5 Mar 1987.)

AB: This book contains 12 sections, each consisting of several papers. The section titles are: Utility Low-Level Waste Management: Plans and Practices; Power-Low-Level Waste Disposal Activities and Waste Management at the IMPC; Siting Developments for LLW Disposal Facilities; Utility LLW Management-Waste Characterization; LLW Containers; Packages and Casks Technologies; Low-Level Waste Processing; LLW Filtration and Ion Exchange Media Processes; Utility- Low Level Waste Management: Processing; Mixed, Hazardous and Short- Lived Wastes; Low Level Waste Regulatory Compliance; LL Waste Processing/Form/Transportation and Associated Equipment/Facilities/ Plans; and LL Waste Form Technology.

DOCUMENT NUMBER = INI19:063938

TI: EIDOMATIX: automatic control **system** for welds by numerical X-ray radiography. EIDOMATIX: un **systeme** de controle automatique de soudures par radiographie X numerique.

AU: Darier, P.; Favier, C.; Furlan, J.

CO: CEA Centre d'Etudes Nucleaires de Grenoble, 38 (France). Inst. de Recherche Technologique et de Developpement Industriel (IRDI).

LA: French

RP: CEA-CONF--9181.

IM: 1987. 5 p. Availability: INIS.

CF: (Meeting on **expert systems** in nuclear field: diagnosis, maintenance, decision aid. Paris (France). 14 Oct 1986.)

AB: Short communication.

DOCUMENT NUMBER = INI19:063841

TI: An **artificial intelligence system** for assisting nuclear **power** plant operators in the diagnosis of and response to plant faults and transients.
AU: Hajek, B.K.; Stasenko, J.E.; Bhatnagar, R.; Hashemi, S.
CO: Ohio State Univ., Columbus (USA). Dept. of Mechanical Engineering.
LA: English
RP: DOE/NE/37965--T2.
IM: Dec 1987. 6 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE88004212.
NO: Portions of this document are illegible in microfiche products.
AB: This report discusses the **Artificial Intelligence** (AI) **system** being developed using the Conceptual Structures and Representation Language (CSRL) developed at the Ohio State University Laboratory for **Artificial Intelligence** Research (LAIR). This **system** combines three sub-systems which have been independently developed to perform the tasks of: detecting changes in the state of the plant that may lead to conditions requiring operator response, and then managing the actions taken by the other two subsystems; diagnosing the plant status independent of alarm states by analyzing the status of basic operating parameters such as flow rates, pressures, temperatures, and water levels, and providing a determination of the validity of sensor indications; and providing and/or synthesizing an appropriate procedure for the operator to follow to correct the transient or abnormal state of the plant. These three **systems** are tied into the main plant computers, including both the process computer and the safety parameter and display **system** computer, through the use of a compatible database. The architecture of the **system** is shown in Figure 1. The **system** is being developed using the Perry Nuclear **Power** Plant (a BWR/6) as the reference plant, and the General Electric ERIS and GEPAC Plus **systems** as key data sources. Scenarios are run on the Perry plant referenced simulator for testing of the AI **system**. Future testing plans call for the **system** to be interfaced directly to the Perry simulator.

DOCUMENT NUMBER = INI19:063808

TI: An examination of qualitative plant modelling as a basis for knowledge-based operator aids in nuclear **power** stations.
AU: Herbert, M.; Williams, G. (Central Electricity Research Labs., Leatherhead (UK)).
LA: English
MS: **Expert systems** and optimisation in process control. **Based** on papers delivered at a seminar on 3-5 December 1985, organised by Unicorn Seminars Ltd. Mamdani, A.; Efstathiou, J. (Queen Mary Coll., London (UK)) (eds.). Unicorn Seminars, London (UK).
IM: Aldershot (UK). Gower Technical Press Ltd. 1986. 275 p. p. 184- 205. ISBN 0-291-39710-7. Price Pound 50.00.
CF: (Seminar on **expert systems** and optimisation in process control. London (UK). 3-5 Dec 1985.)
AB: New qualitative techniques for representing the behaviour of physical **systems** have recently been developed. These allow a qualitative representation to be formally derived from a quantitative plant model. One such technique, Incremental Qualitative Analysis, is **based** on manipulating qualitative differential equations, called confluences, using sign algebra. This is described and its potential for reducing the amount of information presented to the reactor operator is discussed. In order to illustrate the technique, a specific example relating to the influence of failures associated with a pressurized water reactor pressuriser is presented. It is shown that, although failures cannot necessarily be diagnosed unambiguously, the number of possible failures inferred is low. Techniques for discriminating between these possible failures are discussed. (author).

DOCUMENT NUMBER = INI19:063805

TI: Reducing the operator's burden with PODIA, A-PODIA and I-PODIA.
AU: Itoh, Matsumi; Tomizawa, Teruaki (Toshiba Corp., Chiyoda, Tokyo (Japan)); Iwamoto, Takanori (Toshiba Corp., Fuchu, Tokyo (Japan). Fuchu Works).

LA: English
JR: Nucl. Eng. Int. ISSN 0029-5507. CODEN: NEINB. (Jan 1988). v. 33(402) p. 41-43.
AB: The PODIA (Plant Operation by Displayed Information and Automation) advanced control panel **system** is now in use at operating **power plants** in Japan. Toshiba is currently working on a more advanced version (A-PODIA) and a version incorporating elements of **artificial intelligence** (I-PODIA). (author).

DOCUMENT NUMBER = INI19:063800

TI: Development of **expert system** on personal computer for diagnosis of nuclear reactor malfunctions.
AU: Kameyama, Takanori; Uekata, Tomomichi; Oka, Yoshiaki; Kondo, Shunsuke; Togo, Yasumasa (Tokyo Inst. of Tech. (Japan). Faculty of Engineering).
LA: Japanese
JR: Nippon Genshiryoku Gakkai-Shi. ISSN 0004-7120. CODEN: NGEGA. (Jan 1988). v. 30(1) p. 42-48.
AB: An **expert system** on a personal computer has been developed for diagnosis of malfunction of the fast experimental reactor 'JOYO'. Prolog-KABA is used as the language. The **system** diagnoses the event which causes scram or set-back of the control rod after an alarm at steady state operation. The **knowledge** base (KB) consists of several sub-KBs and a meta-KB. Using the forward chaining, the meta-KB decides which sub-KB should be accessed. The cause of the malfunction is identified in the sub-KB using the backward chaining. The terms expressing the characteristics of the events are involved in the production rules as attributes in order to use the Prolog function of pattern matching and back-tracking for efficient inference. The total number of the rules in the **system** is about 400. The experiments using the plant simulator of 'JOYO' have shown that malfunctions are successfully identified by the diagnosis **system.** It takes about 10s for each diagnosis using the 16-bits personal computer, PC-9801 VM. (author).

DOCUMENT NUMBER = INI19:063799

TI: **Knowledge based system** for control rod programming of BWRs.
AU: Fukuzaki, Takaharu; Yoshida, Ken-ichi; Kobayashi, Yasuhiro (Hitachi Ltd., Ibaraki (Japan). Energy Research Lab.).
LA: English
JR: J. Nucl. Sci. Technol. (Tokyo). ISSN 0022-3131. CODEN: JNSTA. (Feb 1988). v. 25(2) p. 120-130.
AB: A **knowledge based system** has been developed to support designers in control rod programming of BWRs. The programming searches through optimal control rod patterns to realize safe and effective burning of nuclear fuel. **Knowledge** of experienced designers plays the main role in minimizing the number of calculations by the core performance evaluation code. This code predicts **power** distibution and thermal margins of the nuclear fuel. This **knowledge** is transformed into 'if-then' type rules and subroutines, and is stored in a **knowledge** base of the **knowledge based system.** The **system** consists of working area, an inference engine and the **knowledge** base. The inference engine can detect those data which have to be regenerated, call those subroutine which control the user's interface and numerical computations, and store competitive sets of data in different parts of the working area. Using this **system,** control rod programming of a BWR plant was traced with about 500 rules and 150 subroutines. Both the generation of control rod patterns for the first calculation of the code and the modification of a control rod pattern to reflect the calculation were completed more effectively than in a conventional method. (author).

DOCUMENT NUMBER = INI19:063755

TI: Fault detection and reliability, **knowledge based** and other approaches.
AU: Singh, M.G.; Hindi, K.S. (Manchester Univ. (UK). Inst. of Science and Technology); Schmidt, G. (Technische Univ. Muenchen (Germany, F.R.)); Tzafestas, S.G. (Technical Univ., Athens (Greece)) (eds.).
LA: English

IM: Oxford (UK). Pergamon Press. 1987. 333 p. ISBN 0-08-034922-6. Price Pound 43.00.
SE: International series on **systems** and control. v. 9.
CF: (2. European workshop on fault diagnostics, reliability and related **knowledge based** approaches. Manchester (UK). 6-8 Apr 1987.)
AB: These proceedings are split up into four major parts in order to reflect the most significant aspects of reliability and fault detection as viewed at present. The first part deals with knowledge-based **systems** and comprises eleven contributions from leading experts in the field. The emphasis here is primarily on the use of **artificial intelligence, expert systems** and other **knowledge- based systems** for fault detection and reliability. The second part is devoted to fault detection of technological **systems** and comprises thirteen contributions dealing with applications of fault detection techniques to various technological **systems** such as gas networks, electric **power systems,** nuclear **reactors** and assembly cells. The third part of the proceedings, which consists of seven contributions, treats robust, fault tolerant and intelligent controllers and covers methodological issues as well as several applications ranging from nuclear **power plants** to industrial robots to steel grinding. The fourth part treats fault tolerant digital techniques and comprises five contributions. Two papers, one on reactor noise analysis, the other on reactor control **system** design, are indexed separately. (author).

DOCUMENT NUMBER = INI19:059710

TI: KLM's boric acid reclamation **system** (BARS). An update.
AU: Schuelke, D.; Kniazewycz, B.G.; Markind, J.; Brossart, M.A.; Choi, R.C. (Northern States **Power** Co., Prairie Island Nuclear Generating Plant, 1717 Wakonade Drive, Easte, Welch, MN 55089 (USA)).
LA: English
MS: Waste management '87: Waste isolation in the US, technical programs and public education. Post, R.G.
IM: Tucson, AZ (USA). University of Arizona Nuclear Engineering Dept. 1987. p. 123-126.
CF: (Waste management '87. Tucson, AZ (USA). 1-5 Mar 1987.)
AB: KLM Technologies has implemented its Department of Energy Phase II Small Business Innovative Research (SBIR) demonstration program for a radioactive waste Boric Acid Reclamation **System** (BARS). Preliminary performance indicates enhanced treatment by the BARS technique over state of the art process methods for selective removal of silica and other impurities from borated water matrices. At optimal **system** recovery of 96-97 percent. BARS removes nominal levels of boric acid while achieving significant rejection for soluble silica and selective radioisotopes. This is indicative of superior performance compared to existing data governing standard boric acid process treatment in the presence of silica and other contaminants. Conventional technologies have also proven to be relatively expensive, utilizing costly chemically treated disposable resins for primary waste removal. The overall BARS program indicates substantial savings regarding off-site disposal costs **based** on reduced waste generation. Optimization of the BARS technology could have potential impact on conventional process technologies that are essentially non-selective in removal capacities. Within the scope of the project, a variety of contaminated process stream and mixed radwaste sources have been evaluated at Northern States Power's Prairie Island Nuclear Generating Station. The design of an advanced prototype BARS as an optimized process alternative was the result of KLM's initial Phase 1 SBIR program with the DOE in 1984 and 1985.

DOCUMENT NUMBER = INI19:059412

TI: Sextant: an **expert system** for transient analysis of nuclear **reactors** and integral test facilities.
AU: Barbet, N.; Dumas, M.; Mihelich, G. (and others).
CO: CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. d'Etudes Mecaniques et Thermiques.
LA: English
RP: CEA-CONF--9234.
IM: 1987. 8 p. Availability: INIS.
CF: (Topical meeting on **artificial intelligence** and other innovative computer applications in the nuclear industry. Snowbird, UT (USA). 31 Aug - 2 Sep 1987.)

AB: **Expert systems** provide a new way of dealing with the computer-aided management of nuclear **plants** by combining several **knowledge** bases and reasoning modes together with a set of numerical models for real- time analysis of transients. New development tools are required together with metaknowledge bases handling temporal hypothetical reasoning and planning. They have to be efficient and robust because during a transient, neither measurements nor models, nor scenarios are hold as absolute references. SEXTANT is a general purpose physical analyzer intended to provide a pattern and avoid duplication of general tools and **knowledge** bases for similar applications. It combines several **knowledge** bases concerning measurements, models and qualitative behavior of PWR with a mechanism of conjecture-refutation and a set of simplified models matching the current physical state. A prototype is under assessment by dealing with integral test facility transients. For its development, SEXTANT requires a powerful shell. SPIRAL is such a toolkit, oriented towards online analysis of complex processes and already used in several applications.

DOCUMENT NUMBER = INI19:059389

TI: A real-time **expert system** for nuclear **power** plant failure diagnosis and operational guide.
AU: Naito, N.; Sakuma, A.; Shigeno, K.; Mori, N. (Nippon Atomic Industry Group Co., Ltd., 4-1 Ukishima-cho, Kawasaki-ku, Kawasaki-shi 210 (Japan)).
LA: English
JR: Nucl. Technol. ISSN 0029-5450. CODEN: NUTYB. (Dec 1987). v. 79(3) p. 284-296.
AB: A real-time **expert system** (DIAREX) has been developed to diagnose plant failure and to offer a corrective operational guide for boiling water reactor (BWR) **power plants.** The failure diagnosis model used in DIAREX was **systematically** developed, **based** mainly on deep **knowledge**, to cover heuristics. Complex paradigms for **knowledge** representation were adopted, i.e., the process representation language and the failure propagation tree. The **system** is composed of a **knowledge** base, **knowledge** base editor, preprocessor, diagnosis processor, and display processor. The DIAREX simulation test has been carried out for many transient scenarios, including multiple failures, using a real-time full-scope simulator modeled after the 1100-MW(electric) BWR **power** plant. Test results showed that DIAREX was capable of diagnosing a plant failure quickly and of providing a corrective operational guide with a response time fast enough to offer valuable information to plant operators.

DOCUMENT NUMBER = INI19:059337

TI: Assessment of seismic damages in nuclear **power** plant buildings.
AU: Corsanego, A.; DelGrosso, A.; Ferro, G. (Istituto di Scienza delle Costruzioni, Univ. of Genova (Italy)).
LA: English
MS: Structural mechanics in reactor technology. Wittmann, F.H.
IM: Accord, MA (USA). A.A. Balkema Publishers. 1987. p. 653-660. ISBN 90-6191-772-7.
CF: (9. SMIRT: international conference on structural mechanics in reactor technology. Lausanne (Switzerland). 17-21 Aug 1987.)
AB: Performance of nuclear **power** plant sites, buildings and components is in today's practice continuously evaluated by means of monitoring **systems** composed by a variety of instruments, allowing records of the most significant behavioral parameters to be gathered by electronic data acquisition equipment. A great emphasis has been devoted in recent years to the development of "intelligent" monitoring **systems** able to perform interpretation of the response of structures and components automatically, only requiring human intervention and sophisticated data processing techniques when degradation of the safety margins is likely to have been produced. Such computerized procedures can be formulated through logic or algorithmic processes and normally are consistently **based** upon simplified, heuristic behavioral models and probabilistic reasoning schemes. This paper is devoted to discuss the development of an algorithmic procedure intended for automatic, real-time interpretation of the recorded response of nuclear **power** plant buildings and foundations during seismic events.

DOCUMENT NUMBER = INI19:056168

TI: **Expert systems** for process control in nuclear research and nuclear technology. Expertensysteme in Prozesskontrolle der Kernforschung und Kerntechnik.
AU: Jankowski, L. (Akademie der Wissenschaften der DDR, Leipzig. Zentralinstitut fuer Isotopen- und Strahlenforschung).
LA: German
JR: Informatik. ISSN 0019-9915. CODEN: IIDWA. (Jan 1988). v. 35(1) p. 28-30.
AB: The use of **expert systems** results in a qualitative change in the possibilities of process control and optimization of technical **plants**. In the case of **plants** used for nuclear research and nuclear technology, **expert systems** for process supervision touch on another aspect of enormous importance, i.e. increasing the safety of the plant. Their design, development and constant improvement are, therefore, among the number one tasks of research. These can be solved only in close cooperation of specialists in the fields of computer technology, scientific information, nuclear research and plant engineering. (author).

DOCUMENT NUMBER = INI19:056164

TI: Cognitive environment simulation: An **artificial intelligence system** for human performance assessment: Cognitive reliability analysis technique: /(Technical report, May 1986-June 1987/).
AU: Woods, D.D.; Roth, E.M.
CO: Westinghouse Research and Development Center, Pittsburgh, PA (USA).
LA: English
RP: NUREG/CR--4862-Vol.3.
IM: Nov 1987. 76 p. Availability: INIS. NTIS, PC A05/MF A01 - US Govt. Printing Office. as TI88900193.
AB: This report documents the results of Phase II of a three phase research program to develop and validate improved methods to model the cognitive behavior of nuclear **power plant** (NPP) personnel. In Phase II a dynamic simulation capability for modeling how people form intentions to act in NPP emergency situations was developed **based** on techniques from **artificial intelligence**. This modeling tool, Cognitive Environment Simulation or CES, simulates the cognitive processes that determine situation assessment and intention formation. It can be used to investigate analytically what situations and factors lead to intention failures, what actions follow from intention failures (e.g., errors of omission, errors of commission, common mode errors), the ability to recover from errors or additional machine failures, and the effects of changes in the NPP person-machine **system**. The Cognitive Reliability Assessment Technique (or CREATE) was also developed in Phase II to specify how CES can be used to enhance the measurement of the human contribution to risk in probabilistic risk assessment (PRA) studies. 34 refs., 7 figs., 1 tab.

DOCUMENT NUMBER = INI19:056163

TI: Cognitive environment simulation: An **artificial intelligence system** for human performance assessment: Modeling human intention formation: /(Technical report, May 1986-June 1987/).
AU: Woods, D.D.; Roth, E.M.; Pople, H. Jr.
CO: Westinghouse Research and Development Center, Pittsburgh, PA (USA).
LA: English
RP: NUREG/CR--4862-Vol.2.
IM: Nov 1987. 135 p. Availability: INIS. NTIS, PC A07/MF A01 - US Govt. Printing Office. as TI88900192.
AB: This report documents the results of Phase II of a three phase research program to develop and validate improved methods to model the cognitive behavior of nuclear **power plant** (NPP) personnel. In Phase II a dynamic simulation capability for modeling how people form intentions to act in NPP emergency situations was developed **based** on techniques from **artificial intelligence**. This modeling tool, Cognitive Environment Simulation or CES, simulates the cognitive processes that determine situation assessment and intention formation. It can be used to investigate analytically what situations and factors lead to intention failures, what actions

follow from intention failures (e.g., errors of omission, errors of commission, common mode errors), the ability to recover from errors or additional machine failures, and the effects of changes in the NPP person-machine **system.** The Cognitive Reliability Assessment Technique (or CREATE) was also developed in Phase II to specify how CES can be used to enhance the measurement of the human contribution to risk in probabilistic risk assessment (PRA) studies. 43 refs., 20 figs., 1 tab.

DOCUMENT NUMBER = INI19:055585

TI: Trends in development of method of probability assessment of nuclear **power** plant safety. Vyvojove tendence v metode pravdepodobnostniho oceneni bezpecnosti jadernych elektraren.
AU: Dach, K. (Ustav Jaderneho Vyzkumu CSKAE, Rez (Czechoslovakia)).
LA: Czech
MS: Safety and operating reliability of reactor WWER. External load. Bezpecnost a provozni spolehlivost reaktoru VVER. Vnejsizatizeni. Pecinka, L. (ed.). Ceskoslovenska Komise pro Atomovou Energii, Prague.
RP: INIS-mf--11222.
IM: 1987. 158 p. p. 125-131. Availability: INIS.
CF: (5. seminar on safety and operating reliability of reactor WWER. Marianske Lazne (Czechoslovakia). 5-6 Nov 1986.)
AB: Trends are discussed in the development of methods of the probability safety assessment of nuclear **power plants.** The research effort is focused on the study of uncertainties, the study of the human factor effect, the study of failures with common cause and on the creation of **expert systems.** It is stated that at present the methods of probability assessment of safety are suitable for assessing the safety of nuclear **power plants** allowing the use of detailed **knowledge** of risks for raising the level of nuclear safety. At the present level of development they can serve as an important instrument for improving the quality of safety documentation and the decision-making sphere. (Z.M.). 2 figs., 3 tabs., 4 refs.

DOCUMENT NUMBER = INI19:052244

TI: Implementation of PATREC nuclear reliability program in PROLOG.
AU: Koen, B.V.; Koen, D.B. (Univ. of Texas, Austin).
LA: English
RP: CONF-870601--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1987). v. 54 p. 54-55.
CF: (Annual meeting of the American Nuclear Society. Dallas, TX (USA). 7-11 Jun 1987.)
AB: PROLOG, the de facto computer language for research in **artificial intelligence** in Japan, is a logical choice for research in the pattern recognition strategy for evaluating the reliability of complex **systems** expressed as fault trees. PROLOG's basic data type is the tree, and its basic control construct is pattern matching. It is also **based** on recursive programming and allows dynamic allocation of memory, both of which are essential for an efficient reduction of the input tree. Since the inference engine of PROLOG automatically examines the user-defined data base in a **systematic** order, an additional advantage of this language is that the largest known pattern will always be found first without coding complex tree searches of the pattern library as was required in other computer languages such as PL/1 and LISP.

DOCUMENT NUMBER = INI19:052241

TI: Computerized plant maintenance **expert system:** a cost/benefit study.
AU: Cruz, P.A.; Frizzell, G.
LA: English
RP: CONF-870837--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1987). v. 54 p. 45.
CF: (13. American Nuclear Society international meeting on nuclear **power** plant operation. Chicago, IL (USA). 30 Aug - 3 Sep 1987.)
AB: Probably the most critical aspect of nuclear **power** plant operation is plant maintenance. The ideal situation, from the plant maintenance perspective, would be to have an **expert** in every phase of plant design and operation constantly available.

There are alternatives, however, to the human **expert** for some aspects of plant maintenance. This paper discusses a microcomputer-based **expert system** designed to assist nuclear **power** plant maintenance activities. Impell Corporation's PbSHIELDING **system**, which is currently in use at nuclear **power plants** throughout the country, including three Commonwealth Edison **plants**, has been developed for the on-site evaluation of temporary lead shielding placement on piping **systems**. This paper focuses on the benefits gained from the use of computerized **expert systems** such as PbSHIELDING for plant operations and maintenance activities. Time and cost savings resulting from the use of such an **expert system** as well as benefits seen from reduced radiation exposure due to increased allowable lead shielding quantities are discussed in detail. This cost/benefit study will be **based** on actual evaluation data gathered from current users of the PbSHIELDING **expert system**.

DOCUMENT NUMBER = INI19:052239

TI: **Artificial intelligence** aid to efficient plant operations.
AU: Wildberger, A.M.; Pack, R.W.
LA: English
RP: CONF-870837--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1987). v. 54 p. 37-38.
CF: (13. American Nuclear Society international meeting on nuclear **power** plant operation. Chicago, IL (USA). 30 Aug - 3 Sep 1987.)
AB: As the nuclear **power** industry matures, it is becoming more and more important that **plants** be operated in an efficient, cost-effective manner, without, of course, any decrease in the essential margins of safety. Indeed, most opportunities for improved efficiency have little or no relation to nuclear safety, but are **based** on trade-offs among operator controllable parameters both within and external to the reactor itself. While these trade-offs are describable in terms of basic physical theory, thermodynamics, and the mathematics of control **systems**, their actual application is highly plant specific and influenced even by the day-to-day condition of the various plant components. This paper proposes the use of **artificial intelligence** techniques to construct a computer-based **expert** assistant to the plant operator for the purpose of aiding him in improving the efficiency of plant operation on a routine basis. The proposed **system**, which only advises the human operator, seems more amenable to the current regulatory approach than a truly automated control **system** even if the latter provides for manual override.

DOCUMENT NUMBER = INI19:051893

TI: Space nuclear **power systems**, 1986: Volume 5.
AU: El-Genk, M.S.; Hoover, M.D. (eds.).
CO: New Mexico Univ., Albuquerque (USA). Inst. for Space Nuclear **Power** Studies; Lovelace Biomedical and Environmental Research Inst., Albuquerque, NM (USA). Inhalation Toxicology Research Inst.
LA: English
RP: CONF-860102--.
IM: Malabar, FL (USA). ORBIT Book Company, Inc. 1987. 466 p. ISBN 0- 89464-017-8.
SE: Space Nuclear **Power Systems** series, Volume 5.
CF: (3. symposium on space nuclear **power systems**. Albuquerque, NM (USA). 13-16 Jan 1986.)
AB: This volume contains the peer reviewed and edited versions of many of the papers presented at the 3rd Symposium on Space Nuclear **Power Systems** held in Albuquerque, New Mexico, January 13-16, 1986. It is the fifth volume in the series Space Nuclear **Power Systems**. The objective of the symposia, and hence this volume, is to summarize the state of **knowledge** in the area of space nuclear **power systems** and to provide a forum at which the most recent findings and important new developments can be presented. Space Nuclear **Power Systems** 1986 includes a total of 45 papers prepared by 119 authors from more than 30 organizations. These 45 papers have been cataloged separately.

DOCUMENT NUMBER = INI19:051725

TI: Situation-assessment and decision-aid production-rule analysis **system** for nuclear plant monitoring and emergency preparedness.
AU: Gvillo, D.; Ragheb, M.; Parker, M.; Swartz, S. (Dept. of Nuclear Engineering, Computer Science Dept., and National Center for Supercomputing Applications (NCSA), Univ. of Illinois, Urbana- Champaign, 103 S. Goodwin Ave., Urbana, IL 61801).
LA: English
MS: Applications of **artificial intelligence** V. Gilmore, J.F.
IM: Bellingham, WA (USA). SPIE Society of Photo-Optical Instrumentation Engineers. 1987. p. 347-354. ISBN 0-89252-821-4.
CF: (5. applications of **artificial intelligence.** Orlando, FL (USA). 18- 20 May 1987.)
AB: A Production-Rule Analysis **System** is developed for Nuclear Plant Monitoring. The signals generated by the Zion-1 Plant are considered. A Situation-Assessment and Decision-Aid capability is provided for monitoring the integrity of the Plant Radiation, the Reactor Coolant, the Fuel Clad, and the Containment **Systems.** A total of 41 signals are currently fed as facts to an Inference Engine functioning in the backward-chaining mode and built along the same structure as the E-Mycin **system.** The Goal-Tree constituting the **Knowledge** Base was generated using a representation in the form of Fault Trees deduced from plant procedures information. The **system** is constructed in support of the Data Analysis and Emergency Preparedness tasks at the Illinois Radiological Emergency Assessment Center (REAC).

DOCUMENT NUMBER = INI19:051706

TI: **Expert system** for diagnosis of reactor transients using confidence levels.
AU: Martin, R.P.; Nassersharif, B. (Texas A and M Univ., College Station).
LA: English
RP: CONF-870601--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1987). v. 54 p. 175-176.
CF: (Annual meeting of the American Nuclear Society. Dallas, TX (USA). 7-11 Jun 1987.)
AB: Transient Analysis of MUltiple-failure Simulations (TAMSU) is a prototype program that is the result of a project investigating and implementing the use of an **expert system** in reactor transient analysis. The purpose of TAMUS is to monitor a simulated nuclear reactor facility, detect deviations from normal operation as they happen, respond with diagnosis of what is happening in the **system,** and recommend operator actions for the transient. TAMUS uses confidence levels to infer diagnosis of multiple-failure transients in real time.

DOCUMENT NUMBER = INI19:051607

TI: Radiation effects on microelectronics.
AU: Gover, J.E. (Sandia National Lab., Albuquerque, NM).
LA: English
RP: CONF-870601--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1987). v. 54 p. 14-15.
CF: (Annual meeting of the American Nuclear Society. Dallas, TX (USA). 7-11 Jun 1987.)
AB: Applications of radiation-hardened microelectronics in nuclear **power systems** include (a) light water reactor (LWR) containment building, postaccident instrumentation that can operate through the beta and gamma radiation released in a design basis loss-of-coolant accident; (b) advanced LWR instrumentation and control **systems** employing distributed digital integrated circuit (IC) technology to achieve a high degree of **artificial intelligence** and thereby reduce the probability of operator error under accident conditions; (c) instrumentation, command, control and communication **systems** for space nuclear **power** applications that must operate during the neutron and gamma-ray core leakage environments as well as the background electron, proton, and heavy charged particle environments of space; and (d) robotics **systems** designed for the described functions. Advanced microelectronics offer advantages in cost and reliability over alternative approaches to instrumentation and control. No semiconductor technology is hard to all classes of radiation effects phenomena. As the effects have become better understood, however, significant progress has been made in hardening IC technology. Application of hardened

microelectronics to nuclear **power systems** has lagged military applications because of the limited market potential of hardened instruments and numerous institutional impediments.

DOCUMENT NUMBER = INI19:051571

TI: An **expert system** for the design of a **power** plant electrical auxiliary **system**.
AU: Puttgen, H.B.; Jansen, J.F. (School of Electrical Engineering, Georgia Institute of Technology, Atlanta, GA 30332).
LA: English
MS: The 1987 industry computer application conference (Conference Papers). Anon.
IM: Piscataway, NJ (USA). IEEE Service Center. 1987. p. 188-195.
CF: (15. PICA: **power** industry computer application conference. Montreal (Canada). 18-22 May 1987.)
AB: The ASDEP **Expert System,** which is oriented toward the electric **power** plant auxiliary **system** design problem, is presented. The **Artificial Intelligence** techniques incorporated into ASDEP are reviewed along with the limitations of the **expert system.** An actual design session is stepped through for a nuclear **power** plant auxiliary **system** design. Finally, some possible extensions of the methods used to other **power system** design problems are alluded to.

DOCUMENT NUMBER = INI19:051556

TI: Information interfaces for process plant diagnosis.
AU: Lind, M.
CO: Risoe National Lab., Roskilde (Denmark).
LA: English
RP: RISO-M--2417.
IM: Feb 1984. 34 p. ISBN 87-550-0982-4. Availability: INIS. Also available from Risoe Library, DK-4000 Roskilde.
CF: (Conference on Data communication in distributed **systems.** Lyngby (Denmark). 2-3 Nov 1983.)
AB: The paper describes a **systematic** approach to the design of information interfaces for operator support in diagnosing complex **systems** faults. The need of interpreting primary measured plant variables within the framework of different **system** representations organized into an abstraction hierarchy is identified from an analysis of the problem of diagnosing complex **systems.** A formalized approach to the modelling of production **systems,** called Multilevel Flow Modelling, is described. A MFM model specifies plant control requirements and the associated need for plant information and provide a consistent context for the interpretation of real time plant signals in diagnosis of malfunctions. The use of MFM models as a basis for functional design of the plant instrumentation **system** is outlined, and the use of **knowledge Based (Expert) Systems** for the design of man-machine interfaces is mentioned. Such **systems** would allow an active user participation in diagnosis and thus provide the basis for cooperative problem solving. 14 refs. (author).

DOCUMENT NUMBER = INI19:050535

TI: Developing sensor-driven robots for hazardous environments.
AU: Trivedi, M.M.; Gonzalez, R.C.; Abidi, M.A. (Dept. of Electrical and Computer Engineering, The Univ. of Tennessee, Knoxville, TN 37996- 2100).
LA: English
MS: Applications of **artificial intelligence** V. Gilmore, J.F.
IM: Bellingham, WA (USA). SPIE Society of Photo-Optical Instrumentation Engineers. 1987. p. 185-189. ISBN 0-89252-821-4.
CF: (5. applications of **artificial intelligence.** Orlando, FL (USA). 18- 20 May 1987.)
AB: Advancements in robotic technology are sought to provide enhanced personnel safety and reduced costs of operation associated with nuclear **power** plant manufacture, construction, maintenance, operation, and decommissioning. The authors describe main characteristics of advanced robotic **systems** for such applications and suggest utilization of sensor-driven robots. Research efforts described in the paper are

directed towards developing robotic **systems** for automatic inspection and manipulation of various tasks associated with a test panel mounted with a variety of switches, controls, displays, meters, and valves.

DOCUMENT NUMBER = INI19:047264

TI: Natural language interface for nuclear data bases.
AU: Heger, A.S.; Koen, B.V. (Univ. of Texas, Austin).
LA: English
RP: CONF-870601--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1987). v. 54 p. 282-284.
CF: (Annual meeting of the American Nuclear Society. Dallas, TX (USA). 7-11 Jun 1987.)
AB: A natural language interface has been developed for access to information from a data base, simulating a nuclear plant reliability data **system** (NPRDS), one of the several existing data bases serving the nuclear industry. In the last decade, the importance of information has been demonstrated by the impressive diffusion of data base management **systems**. The present methods that are employed to access data bases fall into two main categories of menu-driven **systems** and use of data base manipulation languages. Both of these methods are currently used by NPRDS. These methods have proven to be tedious, however, and require extensive training by the user for effective utilization of the data base. **Artificial intelligence** techniques have been used in the development of several intelligent front ends for data bases in nonnuclear domains. Lunar is a natural language program for interface to a data base describing moon rock samples brought back by Apollo. Intellect is one of the first data base question-answering **systems** that was commercially available in the financial area. Ladder is an intelligent data base interface that was developed as a management aid to Navy decision makers. A natural language interface for nuclear data bases that can be used by nonprogrammers with little or no training provides a means for achieving this goal for this industry.

DOCUMENT NUMBER = INI19:045976

TI: **Expert systems** for nuclear **plants.**
AU: John, J. (Czech Technical University, Faculty of Electrical Engineering Prague, Czechoslovakia).
LA: English
MS: **Artificial intelligence** and information-control **systems** of robots - 87. Proceedings of the 4th international conference held in Smolenice, Czechoslovakia, 19-23 October, 1987. Plander, I. (ed.) (Slovenska Akademia Vied, Bratislava (Czechoslovakia). Ustav Technickej Kybernetiky).
IM: Amsterdam (Netherlands). North-Holland. 1987. 516 p. p. 265- 268. ISBN 0-444-70303-9.
CF: (4. International conference on **artificial intelligence** and information-control **systems** of robots. Smolenice (Czechoslovakia). 19-23 Oct 1987.)
AB: At the Department of Automatic Control of the Czech Technical University, two **expert systems** for nuclear **plants** safety program are now developed. The **systems** are **based** on the FEL-EXPERT empty **expert system** and intended for natural circulation emergency cooling of the plant and steam generator state monitoring. Some experience with **knowledge** acquisition and **knowledge** base construction is discussed. 3 refs.

DOCUMENT NUMBER = INI19:045776

TI: **Artificial intelligence** techniques in diagnosis of **power plants.**
AU: Hashemi, S.
LA: English
MS: Future of the nuclear industry: proceedings. Abstracts. Ohio State Univ., Columbus (USA). Published in summary form only.
RP: DOE/ER/75172--1.
IM: 1985. p. 63. Availability: INIS. Available from NTIS, PC A06/MF A01; 1 as DE85016928.
CF: (22. annual American Nuclear Society midwest student conference on the future of the nuclear industry. Columbus, OH (USA). 21-23 Mar 1985.)

DOCUMENT NUMBER = INI19:044925

TI: Development of a semiautonomous robot **system** for surveillance/ maintenance/emergency response work at nuclear **power** stations.
AU: Tulenko, J.S.; Crane, C.; Dalton, G.R. (Univ. of Florida, Gainesville).
LA: English
RP: CONF-870601--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1987). v. 54 p. 325.
CF: (Annual meeting of the American Nuclear Society. Dallas, TX (USA). 7-11 Jun 1987.)
AB: The US Dept. of Energy has awarded grants to four universities (Univ. of Florida, Univ. of Michigan, Univ. of Tennessee, and the Univ. of Texas) to pursue research leading to the development and deployment of an advanced semiautonomous robotic **system** capable of (a) performing tasks that would be hazardous or not achievable for humans, (b) reducing greatly the occupational radiation exposure of plant personnel, and (c) increasing also the availability of the plant to be on line producing **power**. The four universities have decided to work as one team and this paper describes the approach and results to date of the Univ. of Florida's efforts to provide **intelligence** to the robotic **system**.

DOCUMENT NUMBER = INI19:042310

TI: An **expert system** for USNRC emergency response.
AU: Sebo, D.E.; Bray, M.A.; King, M.A. (EG and G Idaho, Inc., Idaho Falls, ID).
LA: English
MS: Proceedings of **expert systems** in government symposium. Karna, K.N.; Parsaye, K.; Silverman, B.G.
IM: Piscataway, NJ (USA). IEEE Service Center. 1986. p. 74-79. ISBN 0-81806-0738-6.
CF: (2. **expert systems** in government conference. McLean, VA (USA). 20- 24 Oct 1986.)
AB: The Reactor Safety Assessment **System** (RSAS) is an **expert system** under development for the United States Nuclear Regulatory Commission (USNRC). RSAS is intended for use at the NRO's Operations Center in the event of a serious incident at a licensed nuclear **power** plant. RSAS is a situation assessment **expert system** which uses plant parametric data to generate conclusions for use by the NRC Reactor Safety Team. RSAS uses multiple rule bases and plant specific setpoint files in order to be applicable to all licensed **power plants**. RSAS currently covers several generic reactor types and **power plants** within those classes.

DOCUMENT NUMBER = INI19:041735

TI: Qualification program of an instrumentation **system** for measuring nuclear **power** plant settlements.
AU: Cremonini, M.G.; Varosio, G. (D'Appolonia S.p.A., Genoa (Italy)); Vanoli, G. (Centro Informazioni Studi Esperienze, Segrate (Italy)); Mirone, M.; Saveri, E. (Ente Nazionale per l'Energia Elettrica, Rome (Italy)).
LA: English
MS: Transactions of the 9th international conference on structural mechanics in reactor technology. Vol. D. Experience with structures and components in operating **reactors.** Wittmann, F.H. (ed.).
IM: Rotterdam (Netherlands). Balkema. 1987. 469 p. p. 159-164. ISBN 90-6191-765-4.
CF: (9. biennial international conference on structural mechanics in reactor technology (SMIRT-9). Lausanne (Switzerland). 17-21 Aug 1987.)
AB: A level indicator **system** was installed in 1984 at the Caorso nuclear **power** plant, for automatic and remote monitoring of foundation settlements. Before discontinuing previous optical surveying activities, a qualification program was developed to validate the continuous records provided by the level indicator **system**. Some of the results of this program, presently under way, are presented herein, i.e. the comparison of the automatic and optical settlement data, and the geotechnical evaluation of the maximum settlement recorded by the **system** during the 1985 refueling of the plant. The level indicator **system** is finally shown to provide a basis for the development of an **expert system** that can perform and automatic, continuous and real-time foundation safety control of the plant. (orig.).

DOCUMENT NUMBER = INI19:036701

TI: Express: the reliability of complex **systems** and **artificial intelligence** techniques. Express: la fiabilite des **systemes** complexes et les techniques de l'intelligence artificielle.
AU: Ancelin, C.; Le, P.; Saint-Quentin, S. de (Electricite de France, 75 - Paris. Direction des Etudes et Recherches).
LA: French
JR: Rev. Gen. Nucl. ISSN 0335-5004. CODEN: RGNUD. (Jul-Aug 1987). v. 4 p. 378-381.
AB: The probabilistic safety study for the Paluel nuclear **power** station, commissioned by EDF in 1986, involved development of data processing methods and equipment which was to be given an entirely new impetus by the use of **artificial intelligence** techniques. The authors describe the salient features of the approach which was adopted and the lessons learnt from the way it was applied in practice.

DOCUMENT NUMBER = INI19:036661

TI: Power-plant simulation and reactor safety.
AU: Hetrick, D.L. (Univ. of Arizona, Tucson).
LA: English
JR: Nucl. Saf. ISSN 0029-5604. CODEN: NUSAA. (Oct-Dec 1987). v. 28(4) p. 473-486.
AB: This article is a review of developments in nuclear power-plant simulation that are particularly relevant to nuclear plant safety. This includes complex fast interactive **systems** called plant analyzers, intermediate-size modular **systems,** and small simulators used for preliminary studies and for illustrating concepts. Applications include safety analysis, computer-assisted plant operation, and test beds for design of control **systems,** disturbance analysis **systems,** and **expert systems** for automated plant operation.

DOCUMENT NUMBER = INI19:036626

TI: The use of **artificial intelligence** in nuclear engineering.
AU: Tye, P.; Garland, W.J. (McMaster Univ., Hamilton, Ontario (Canada). Dept. of Engineering Physics).
LA: English
JR: Energy Newsl. (Hamilton). ISSN 0711-3366. CODEN: ENNLD. (Aut 1985). v. 6(3) p. 21-33.
AB: Many real time problems, such as plant control, require the ability to process a huge amount of data very quickly to determine the state of the **system.** To do this CANDU **reactors** use Digital Control Computers to monitor the reactor and algorithm **based** programs to perform tasks such as 'power-up' and 'power-down'. The problem with this **system** lies in the fact that when the operator most needs help, during an upset, the computer cannot offer any real assistance since the task of determining the cause of a problem lacks the mathematically tractable core needed by most computer programs to function. This paper will examine the use of Artifical **Intelligence** techniques, specifically **expert systems,** as an aid to operators in reactor control.

DOCUMENT NUMBER = INI19:035874

TI: Application of **knowledge** engineering to **power** plant heat rate performance monitoring.
AU: Frogner, B.; Dourandish, R.
LA: English
RP: CONF-861211--.
IM: New York, NY (USA). American Society of Mechanical Engineers. 1987. 8 p.
NO: Technical Paper 86-WA/DSC-23.
CF: (American Society of Mechanical Engineers winter meeting. Anaheim, CA (USA). 7-12 Dec 1986.)
AB: An **expert system** approach has been used to demonstrate how one can implement in a computer program the essential expertise needed to operate nuclear **power plants** at a high thermal efficiency. The focus is on the ability of skilled performance analysis

engineers to monitor the plant heat rate performance as a means of early diagnosing incipient failures, recognizing deteriorating performance, and keeping the plant optimally configured. This essential experience is being integrated with **knowledge** available in reports and the basic understanding of the underlying physics of these **plants.** The resulting body of **knowledge** is being implemented into an **expert system,** called the Performance Diagnosis Assistant (PDA), that can be made available to elss experienced plant performance engineers.

DOCUMENT NUMBER = INI19:035866

TI: Transactions of the fifteenth water reactor safety information meeting.
AU: Weiss, A.J. (comp.).
CO: Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research.
LA: English
RP: NUREG/CP--0090.
IM: Oct 1987. 269 p. Availability: INIS. NTIS, PC A12/MF A01 - US Govt. Printing Office. as DE88900097.
AB: This report contains summaries of paper on reactor safety research to be presented at the 15th Water Reactor Safety Information Meeting held at the National Bureau of Standards in Gaithersburg, Maryland, October 26-29, 1987. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers concerning nuclear safety issues are included from the Office of Nuclear Reactor Regulation, USNRC, in addition to summaries of invited papers that cover the highlights of reactor safety research conducted by the Department of Energy (DOE), the electric utilities through the Electric **Power** Research Institute (EPRI), the nuclear industry, and the research of government and industry in Europe and Japan. The summaries have been compiled in one report to provide a basis for meaningful discussion and information exchange during the course of the meeting, and are given in the order of their presentation in each session.

DOCUMENT NUMBER = INI19:033597

TI: Commutative convergence of a **knowledge** structures **system:** L.R.C. software applied to a project of computer assisted control of a nuclear **power** plant. Convergence commutative d'une base de connaissances: les logiciels L.R.C. appliques a une maquette d'aide a la conduite d'une centrale nucleaire.
AU: Hery, J.F.
CO: Ecole Centrale des Arts et Manufactures, 92 - Chatenay-Malabry (France).
LA: French
RP: FRNC-TH--3166.
IM: Jun 1985. 481 p. Availability: INIS.
NO: Designation: These (D. Ing.).
AB: For operating continuous process, the conceptors of control room are confronted with two major problems: what is the optimal distribution of works between operators and machinery and what interface between them. The development of **expert systems** can made easier the solutions of these problems. In this work, we proposed to present the walk to make a model of **expert system based** on control of nuclear reactor. The major problems described here, are those of convergence and commutativity of knowledge's bases. We have formulated some necessary conditions to assure these two major properties for the bases.

DOCUMENT NUMBER = INI19:033578

TI: Regulatory considerations in extending the operating life of nuclear **power plants.**
AU: Cox, D.L. (Southern California Edison Co.).
LA: English
MS: Proceedings of the 1986 joint ASME/ANS nuclear **power** conference. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 293- 294.
CF: (ASME/ANS bi-annual nuclear **power** conference: safe and reliable nuclear **power plants.** Philadelphia, PA (USA). 20-23 Jul 1986.)

AB: In this paper some recommendations and conclusions with respect to the regulatory process involved in nuclear plant license renewal are presented. One item of primary importance is the establishment of a regulatory approach whereby the NRC could effectively evaluate license renewal applications. A number of approaches were considered, yet no single regulatory policy framework has been proposed or determined at this time. It is attempted to design the optimum approach to meet the broad needs of both the NRC and the nuclear utilities. To ensure that the overall license renewal process is adequate, many issues require consideration and must somehow be incorporated into the finalized approach. These encompass a large spectrum of issues ranging from operational performance and technical acceptance criteria to the overall timing and schedular constraints of the license renewal process. The specific "weighted" importance of each issue remains undetermined insofar as how it may be incorporated in the renewal process and what measures need to be taken by utilities to appropriately address all regulatory concerns. It is clear at this time, though, that the same findings required of the NRC for issuance of an initial operating license are relevant as well to renewal of an operating license. Mainly, the NRC must have reasonable assurance as to the continued safe operation of the facility.

DOCUMENT NUMBER = INI19:033207

TI: LARA: **Expert system** for acoustic localization of robot in a LMFBR. Lara: un **systeme expert** pour la localisation acoustique d'un robot dans un reacteur a neutrons rapides.
AU: Lhuillier, C.; Malvache, P.
CO: CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-lez- Durance (France). Dept. des Reacteurs a Neutrons Rapides.
LA: French
RP: CEA-DRNR-P--365.
IM: Dec 1986. 15 p. Availability: INIS.
CF: (2. International conference on **artificial intelligence**. Amsterdam (Netherlands). 8-11 Dec 1986.)
AB: The **expert system** LARA (Acoustic Localization of Autonomic Robot) has been developed to show the interest of introducing **artificial** intelligency for fine automatic positioning of refuelling machine in a LMFBR reactor. LARA which is equipped with an acoustic detector gives rapidly a good positioning on the fuel.

DOCUMENT NUMBER = INI19:033022

TI: Qualimetrical **expert** method - a tool for determining the safety level of nuclear **power plants.**
AU: Altmann, S.; Zahn, P. (Technische Hochschule, Leipzig (German Democratic Republic)); Kuraszkiewicz, P. (Politechnika Poznanska (Poland)).
LA: English
MS: Transactions of the 9th international conference on structural mechanics in reactor technology. Vol. M. Structural reliability - probabilistic safety assessment. Wittmann, F.H. (ed.).
IM: Rotterdam (Netherlands). Balkema. 1987. 497 p. p. 173-176. ISBN 90-6191-774-3.
CF: (9. biennial international conference on structural mechanics in reactor technology (SMIRT-9). Lausanne (Switzerland). 17-21 Aug 1987.)
AB: We will deal with an alternative method of safety level calculation named Qualimetrical **Expert** Method in which the nuclear safety is considered particularly. The mentioned method was proposed by Altmann for electric **power plants** and used with success for several years. One of the variants of the qualimetrical method isbased on **expert** judgement in which, however, the subjective factors which play an important role cannot be eliminated. To exclude the influence of subjective factors form the **expert** judgement method Zahn proposed the psychological mathematics application. This mathematics fits very well to complex **systems** characterised by meagre statistical data in view of the reliability coefficients calculation. (orig./HP).

DOCUMENT NUMBER = INI19:033021

TI: On the derivation of fragility curves from **expert** opinion.
AU: Mosleh, A. (Pickard, Lowe and Garrick, Inc., Newport Beach, CA (USA)); Apostolakis, G. (California Univ., Los Angeles (USA). Dept. of Mechanical, Aerospace and Nuclear Engineering).
LA: English
MS: Transactions of the 9th international conference on structural mechanics in reactor technology. Vol. M. Structural reliability - probabilistic safety assessment. Wittmann, F.H. (ed.).
IM: Rotterdam (Netherlands). Balkema. 1987. 497 p. p. 167-172. ISBN 90-6191-774-3.
CF: (9. biennial international conference on structural mechanics in reactor technology (SMIRT-9). Lausanne (Switzerland). 17-21 Aug 1987.)
AB: In the assessment of the seismic risk of nuclear **power plants**, the capacity of components and structures to withstand seismically induced stress is presented in the form of fragility curves. A fragility curve is the conditional probability of failure as a function of the response parameter; e.g., the peak ground acceleration. Although rather complex mathematically in their most general form, the formal methods of use of **expert** opinion for estimating fragility curves provide us with useful insights into several aspects of the **expert** opinion problem. They help us understand the often hidden assumptions behind the commonly used ad hoc techniques. They also present a common link among such models and provide a vehicle for assessing the impact of the various modeling assumptions on the final results. Once these assumptions and their implications are understood, simpler models may be used more appropriately and with higher confidence. (orig./HP).

DOCUMENT NUMBER = INI19:029329

TI: **Expert systems** and accident management.
AU: Jenkins, J.P.; Nelson, W.R. (Nuclear Regulatory Commission, Washington, DC).
LA: English
MS: Proceedings of **expert systems** in government symposium. Karna, K.N.; Parsaye, K.; Silverman, B.G.
IM: Piscataway, NJ (USA). IEEE Service Center. 1986. p. 88-95. ISBN 0-81806-0738-6.
CF: (2. **expert systems** in government conference. McLean, VA (USA). 20- 24 Oct 1986.)
AB: Accident Management **Expert System** (AMES) is an operational aid to decision makers concerned with the management of low probability, high consequence severe accidents. Such severe accidents in commercial nuclear **power plants** are seen as events which threaten the integrity of the core or the containment protection **systems**. AMES consists of an **expert system** developed from a deep **knowledge** base and characterized by success paths in the form of a response tree. Lessons learned from two experiments provide the basis for selection of this method of **expert system** development.

DOCUMENT NUMBER = INI19:029324

TI: Expert-systems and computer-based industrial **systems**.
AU: Terrien, J.F. (Societe Franco-Americaine de Constructions Atomiques (FRAMATOME), 92 - Courbevoie (France)).
LA: English
JR: Nucl. Eng. ISSN 0368-2595. CODEN: NUEND. (Jul-Aug 1987). v. 28(4) p. 127-131.
AB: Framatome makes wide use of **expert systems**, computer-assisted engineering, production management and personnel training. It has set up separate business units and subsidiaries and also participates in other companies which have the relevant expertise. Five examples of the products and services available in these are discussed. These are in the field of applied **artificial intelligence** and expert-systems, in integrated computer-aid design and engineering, structural analysis, computer-related products and services and document management **systems**. The structure of the companies involved and the work they are doing is discussed. (UK).

DOCUMENT NUMBER = INI19:028786

TI: Design of an **expert system** to assess structural damage in nuclear primary piping. (EXULT.)
AU: Garribba, S.F. (Politecnico di Milano (Italy). Ist. di Ingegneria Nucleare); Lucia, A.A. (Commission of the European Communities, Ispra (Italy). Joint Research Centre).
LA: English
MS: Transactions of the 9th international conference on structural mechanics in reactor technology. Vol. A. Indexes, abbreviations, supplement. Wittmann, F.H. (ed.).
IM: Rotterdam (Netherlands). Balkema. 1987. 471 p. p. 285-294. ISBN 90-6191-761-1.
NO: Contract CEC 2827-85-11 ED.
CF: (9. biennial international conference on structural mechanics in reactor technology (SMIRT-9). Lausanne (Switzerland). 17-21 Aug 1987.)
AB: In the effort towards designing and constructing the **expert system** four main steps are envisaged. First, are the design of the main architecture of the **expert system** and the identification of the functions which must be assigned to the different **expert** modules or blocks. Second, is the definition of the **knowledge** mechanisms and their representation in terms of rules and metarules. Third, is the preparation of prototypes for the **expert** modules and their testing. As the fourth step, are the design and enactment of a capability for controlling the interconnections among the **expert** modules to make information flows compatible and to optimize the operation of the whole **system.** (orig./HP).

DOCUMENT NUMBER = INI19:028727

TI: The computerized **expert system:** An innovative approach to an easily customized plant maintenance tool.
AU: Cruz, P.A.; Ashley, G.R. (Impell Corp., Bannockburn, IL (USA)).
LA: English
MS: Transactions of the 9th international conference on structural mechanics in reactor technology. Vol. B. Computational mechanics and computer-aided engineering. Wittmann, F.H. (ed.).
IM: Rotterdam (Netherlands). Balkema. 1987. 637 p. p. 193-198. ISBN 90-6191-763-8.
CF: (9. biennial international conference on structural mechanics in reactor technology (SMIRT-9). Lausanne (Switzerland). 17-21 Aug 1987.)
AB: One critical aspect of nuclear **power** plant operation is plant maintenance. Its rapid and efficient performance ensures maximum plant availability at the required safety level. While computerized **expert systems** can provide cost effective solutions to plant maintenance needs, the wide range of individual plant design requirements and installation environments emphasizes the need for a rule-based **expert system** whose components can be easily customized to respond to plant specific design and maintenance requirements. This paper discusses the techniques employed in the recent development of a microcomputer-based **expert system** called PbSHIELDING. Designed for the evaluation of the use of temporary lead shielding on nuclear **power** plant piping, PbSHIELDING will be used to illustrate how a versatile **expert system** can be easily adapted to suit individual **power** plant needs. (orig.).

DOCUMENT NUMBER = INI19:028432

TI: Transactions of the 9th international conference on structural mechanics in reactor technology. Vol. B. Computational mechanics and computer-aided engineering.
AU: Wittmann, F.H. (ed.).
LA: English
IM: Rotterdam (Netherlands). Balkema. 1987. 637 p. ISBN 90-6191-763- 8.
CF: (9. biennial international conference on structural mechanics in reactor technology (SMIRT-9). Lausanne (Switzerland). 17-21 Aug 1987.)
AB: The area to be covered by Division B of SMIRT-Conferences can be briefly summarized as follows: Computational methods for static and dynamic analysis of solids and structures in linear and non-linear regimes: Finite elements, boundary elements, finite differences and other numerical techniques. Methods and software for computer-aided engineering including data bases, computer graphics and **expert systems.** Seventy eight papers have been separately indexed for the database (orig./GL).

DOCUMENT NUMBER = INI19:024201

TI: Promises in intelligent plant control **systems.**
AU: Otaduy, P.J.
CO: Oak Ridge National Lab., TN (USA).
LA: English
RP: CONF-870832--5.
IM: 1987. 8 p. Availability: INIS. Available from NTIS, PC A02; 3 as DE87014780.
NO: Paper copy only, copy does not permit microfiche production.
CF: (Topical meeting on **artificial intelligence** and other innovative computer applications in the nuclear industry. Snowbird, UT (USA). 31 Aug - 2 Sep 1987.)
AB: The control **system** is the brain of a **power** plant. The traditional goal of control **systems** has been productivity. However, in nuclear **power plants** the potential for disaster requires safety to be the dominant concern, and the worldwide political climate demands trustworthiness for nuclear **power plants.** To keep nuclear generation as a viable option for **power** in the future, trust is the essential critical goal which encompasses all others. In most of today's nuclear **plants** the control **system** is a hybrid of analog, digital, and human components that focuses on productivity and operates under the protective umbrella of an independent engineered safety **system.** Operation of the plant is complex, and frequent challenges to the safety **system** occur which impact on their trustworthiness. Advances in nuclear reactor design, computer sciences, and control theory, and in related technological areas such as electronics and communications as well as in data storage, retrieval, display, and analysis have opened a promise for control **systems** with more acceptable human brain-like capabilities to pursue the required goals. This paper elaborates on the promise of futuristic nuclear **power plants** with intelligent control **systems** and addresses design requirements and implementation approaches.

DOCUMENT NUMBER = INI19:024200

TI: Incorporating "fuzzy" data and logical relations in the design of **expert systems** for nuclear **reactors.**
AU: Guth, M.A.S.
CO: Oak Ridge National Lab., TN (USA); Oak Ridge Associated Universities, Inc., TN (USA).
LA: English
RP: CONF-870832--3.
IM: 1987. 9 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE87014410.
NO: Portions of this document are illegible in microfiche products.
CF: (Topical meeting on **artificial intelligence** and other innovative computer applications in the nuclear industry. Snowbird, UT (USA). 31 Aug - 2 Sep 1987.)
AB: This paper applies the method of assigning probability in Dempster- Shafer Theory (DST) to the components of rule-based **expert systems** used in the control of nuclear **reactors.** Probabilities are assigned to premises, consequences, and rules themselves. This paper considers how uncertainty can propagate through a **system** of Boolean equations, such as fault trees or **expert systems.** The probability masses assigned to primary initiating events in the **expert system** can be derived from observing a nuclear reactor in operation or **based** on engineering **knowledge** of the reactor parts. Use of DST mass assignments offers greater flexibility to the construction of **expert systems.**

DOCUMENT NUMBER = INI19:023516

TI: European Reliability Data **System:** main developments and use.
AU: Amesz, J.; Capobianchi, S.; Kalfsbeek, H.W.; Mancini, G. (Commission of the European Communities, Ispra, Italy).
LA: English
MS: International topical meeting on probabilistic safety methods and applications: proceedings. Volume 1. Sessions 1-8. Electric **Power** Research Inst., Palo Alto, CA (USA).
RP: EPRI-NP--3912-SR-Vol.1.
IM: Feb 1985. p. 41.1-41.14. Availability: Research Reports Center, P.O. Box 50490, Palo Alto, CA 94303 $125.00.

CF: (International ANS/ENS topical meeting on probabilistic safety methods and applications. San Francisco, CA (USA). 24 Feb - 1 Mar 1985.)
AB: The paper revises the aims and objectives of the European Reliability Data **System** (ERDS), a centralized **system** collecting and organizing information related to the operation of light water **reactors**. The paper first describes the recent developments of the four data banks constituting the **system**: Component Event Data Bank, Abnormal Occurrences Reporting **System**, Operating Unit Status Report and Reliability Parameters Data Bank. Then several issues are discussed referring mostly to the status of classification schemes and their use, to the operation of the banks (data input and transcoding) and to the retrieval and utilization of the information; in this latter case particularly the analysis potential of the data collection scheme of the AORS (Abnormal Occurrences Reporting **System**) is demonstrated. Finally, emphasis is given to the increasing role which **artificial intelligence** techniques such as natural language and **expert systems** and fuzzy logic may play in improving the future capabilities of the **system**.

DOCUMENT NUMBER = INI19:023477

TI: Development of an integrated signal validation **system** in nuclear **power plants**: Annual report for the period September 30, 1986- September 29, 1987.
AU: Upadhyaya, B.R.; Kerlin, T.W.; Gloeckler, O.; Frei, Z.; Morgenstern, V.; Qualls, L.
CO: Tennessee Univ., Knoxville (USA). Dept. of Nuclear Engineering; Combustion Engineering, Inc., Windsor, CT (USA).
LA: English
RP: DOE/NE/37959--12.
IM: Sep 1987. 194 p. Availability: INIS. Available from NTIS, PC A09; 3 as DE88000187.
NO: Paper copy only, copy does not permit microfiche production.
AB: In large **power** generating stations, signals from various instrumentation **systems** are channeled to control **systems**, protection **systems**, and plant monitoring **systems**. Validation of these signals accomplishes the following goals: minimize plant downtime, increase the reliability of operator decisions, and as an aid for scheduling plant maintenance. The signal validation methods developed under this project are designed to perform various levels of signal processing, and to detect, isolate and characterize faulty signals. A central database stores information about sensors, history of signal validation results, plant data, and subsystem models. The signal validation architecture consists of parallel signal processing modules, each of which implements a diverse validation scheme. The software development and testing of all the modules is performed using IBM PC/AT computers. All the signal validation modules are ready for preliminary implementation and are tested using operational data from a pressurized water reactor (PWR) and PWR data from the Combustion Engineering Nuclear Transient Simulation code.

DOCUMENT NUMBER = INI19:023408

TI: LTREE - a lisp-based algorithm for cutset generation using Boolean reduction.
AU: Finnicum, D.J.; Rzasa, P.W. (Combustion Engineering, Inc.).
LA: English
MS: International topical meeting on probabilistic safety methods and applications: proceedings. Volume 3. Sessions 17-23 and indexes. Electric **Power** Research Inst., Palo Alto, CA (USA).
RP: EPRI-NP--3912-SR-Vol.3.
IM: Feb 1985. p. 177.1-177.10. Availability: Research Reports Center, P.O. Box 50490, Palo Alto, CA 94303 $125.00.
CF: (International ANS/ENS topical meeting on probabilistic safety methods and applications. San Francisco, CA (USA). 24 Feb - 1 Mar 1985.)
AB: Fault tree analysis is an important tool for evaluating the safety of nuclear **power plants**. The basic objective of fault tree analysis is to determine the probability that an undesired event or combination of events will occur. Fault tree analysis involves four main steps: (1) specifying the undesired event or events; (2) constructing the fault tree which represents the ways in which the postulated event(s) could occur; (3) qualitative evaluation of the logic model to identify the

minimal cutsets; and (4) quantitative evaluation of the logic model to determine the probability that the postulated event(s) will occur given the probability of occurrence for each individual fault. This paper describes a LISP-based algorithm for the qualitative evaluation of fault trees. Development of this algorithm is the first step in a project to apply **expert systems** technology to the automation of the fault tree analysis process. The first section of this paper provides an overview of LISP and its capabilities, the second section describes the LTREE algorithm and the third section discusses the on-going research areas.

DOCUMENT NUMBER = INI19:023328

TI: Use of **expert systems** in nuclear **power plants.**
AU: Uhrig, R.E.
CO: Oak Ridge National Lab., TN (USA).
LA: English
RP: CONF-870822--5.
IM: Aug 1987. 7 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE87013680.
NO: Portions of this document are illegible in microfiche products. Published in summary form only.
CF: (American Institute of Chemical Engineers summer national meeting. Minneapolis, MN (USA). 16-19 Aug 1987.)

DOCUMENT NUMBER = INI19:023298

TI: Method of safety evaluation in nuclear **power plants.** Methode zur Sicherheitsbewertung in Kernkraftwerken.
AU: Kuraszkiewicz, P. (Politechnika Poznanska (Poland)); Zahn, P. (Technische Hochschule, Leipzig (German Democratic Republic)).
LA: German
JR: Kernenergie. ISSN 0023-0642. CODEN: KERNA. (Jan 1988). v. 31(1) p. 7-9.
AB: A novel quantitative technique for evaluating safety of subsystems of nuclear **power plants based** on **expert** estimations is presented. It includes methods of mathematical psychology recognizing the effect of subjective factors in the **expert** estimates and, consequently, contributes to further objectification of evaluation. It may be applied to complementing probabilistic safety assessment. As a result of such evaluations a characteristic 'safety of nuclear **power plants**' is obtained. (author).

DOCUMENT NUMBER = INI19:023151

TI: **Expert** robots in nuclear **plants.**
AU: Byrd, J.S.; Fisher, J.J.; DeVries, K.R.; Martin, T.P.
CO: Savannah River Lab., Aiken, SC (USA).
LA: English
RP: DP-MS--87-6. CONF-870832--6.
IM: 1987. 11 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE87014839.
NO: Portions of this document are illegible in microfiche products.
CF: (Topical meeting on **artificial intelligence** and other innovative computer applications in the nuclear industry. Snowbird, UT (USA). 31 Aug - 2 Sep 1987.)
AB: **Expert** robots will enhance safety and operations in nuclear **plants.** E. I. du Pont de Nemours and Company, Savannah River Laboratory, is developing **expert** mobile robots for deployment in nuclear applications at the Savannah River Plant. Knowledge-based **expert systems** are being evaluated to simplify operator control, to assist in navigation and manipulation functions, and to analyze sensory information. Development work using two research vehicles is underway to demonstrate semiautonomous, intelligent, **expert** robot **system** operation in process areas. A description of the mechanical equipment, control **systems,** and operating modes is presented, including the integration of onboard sensors. A control hierarchy that uses modest computational methods is being used to allow mobile robots to autonomously navigate and perform tasks in known environments without the need for large computer **systems.**

DOCUMENT NUMBER = INI19:023150

TI: US Department of Energy Nuclear Energy University program in robotics for advanced **reactors:** Program plan, FY 1987-1991.
AU: Mann, R.C.; Gonzalez, R.C.; Tulenko, J.S.; Tesar, D.; Wehe, D.K.
CO: Oak Ridge National Lab., TN (USA); Tennessee Univ., Knoxville (USA); Florida Univ., Gainesville (USA); Texas Univ., Austin (USA); Michigan Univ., Ann Arbor (USA).
LA: English
RP: DOE/OR--884.
IM: Jul 1987. 137 p. Availability: INIS. Available from NTIS, PC A07/ MF A01; 1 as DE87014967.
NO: Portions of this document are illegible in microfiche products. Original copy available until stock is exhausted.
AB: The US Department of Energy has provided support to four universities and the Oak Ridge National Laboratory in order to pursue research leading to the development and deployment of an advanced robotic **system** capable of performing tasks that are hazardous to humans, that generate significant occupational radiation exposure, and/or whose execution times can be reduced if performed by an automated **system**. The goal is to develop a generation of advanced robotic **systems** capable of performing surveillance, maintenance, and repair tasks in nuclear facilities and other hazardous environments. This goal will be achieved through a team effort among the Universities of Florida, Michigan, Tennessee, Texas, and the Oak Ridge National Laboratory, and their industrial partners, Combustion Engineering, Martin Marietta Baltimore Aerospace, Odetics, Remotec, and Telerobotics International. Each of the universities and ORNL have ongoing activities and corresponding facilities in areas of R and D related to robotics. This program is designed to take full advantage of these existing resources at the participating institutions.

DOCUMENT NUMBER = INI19:022295

TI: The structure of an **expert system** to diagnose and supply a corrective procedure for nuclear **power** plant malfunctions.
AU: Hajek, B.K.; Stasenko, J.E.; Hashemi, S.; Bhatnagar, R.; Yamada, N.; Punch, W.F. III.
LA: English
MS: Application of **artificial intelligence** to the operation of nuclear **power plants:** Progress report, September 15, 1986-June 4, 1987. Hazek, B.K.; Miller, D.W. Ohio State Univ. Research Foundation, Columbus (USA).
RP: DOE/NE/37965--T1.
IM: Jun 1987. Paper 4. Availability: INIS. Available from NTIS, PC A03/MF A01 as DE87010749.
AB: Two prototype **knowledge based systems** have been developed at the Ohio State University. These **systems** were the result of collaboration between the Nuclear Engineering Program and the Laboratory for **Artificial Intelligence** Research (LAIR). The first **system** uses hierarchical classification to diagnose malfunctions of the coolant **system** in a General Electric Boiling Water Reactor (BWR). The second **system** provides a plan of action, through a process of dynamic procedure synthesis, to stabilize the plant once an abnormal transient has occurred. The objective of this paper is to discuss a structure that will integrate the two **systems**. The combined **system** will be capable of informing plant personnel about the nature of malfunctions, and of supplying to the operator the most direct corrective procedure available. Two important features of the integrated **system** are faulty sensor detection, **based** on malfunction context and unlike sensor data, and procedure synthesis **based** on the initial state of the plant.

DOCUMENT NUMBER = INI19:022294

TI: An **expert system** for sensor data validation and malfunction detection.
AU: Hashemi, S.; Hajek, B.K.; Miller, D.W.; Chandrasekaran, B.; Punch, W.F. III.
LA: English
MS: Application of **artificial intelligence** to the operation of nuclear **power plants:** Progress report, September 15, 1986-June 4, 1987. Hazek, B.K.; Miller, D.W. Ohio State Univ. Research Foundation, Columbus (USA).

RP: DOE/NE/37965--T1.
IM: Jun 1987. Paper 3. Availability: INIS. Available from NTIS, PC A03/MF A01 as DE87010749.
AB: During recent years, applications of **expert systems** in different fields of engineering have been under study throughout the world. At the Ohio State University, the theories developed by the Laboratory for **Artificial Intelligence** Research (LAIR) have been implemented for nuclear **power plants** and chemical processing **systems.** For nuclear **power plants,** these techniques have been further developed to reach diagnostic conclusions about malfunctions and faulty sensors, as well as to suggest corrective actions about the malfunctions. This paper concentrates on the AI applications to plant diagnosis and faulty sensor identifications. To achieve the above goals without adding extra sensors in a plant, the use of unlike sensor data (such as relationships between pressure and temperature in a Boiling Water Reactor (BWR)) and diagnostic conclusions about malfunctions as backups for suspicious sensors has been made. This extra evidence is readily available throughout the plant and is not generally used to backup suspicious sensor data in any manner.

DOCUMENT NUMBER = INI19:022293

TI: CSRL application to nuclear **power** plant diagnosis and sensor data validation.
AU: Hashemi, S.; Punch, W.F. III; Hajek, B.K.
LA: English
MS: Application of **artificial intelligence** to the operation of nuclear **power plants:** Progress report, September 15, 1986-June 4, 1987. Hazek, B.K.; Miller, D.W. Ohio State Univ. Research Foundation, Columbus (USA).
RP: DOE/NE/37965--T1.
IM: Jun 1987. Paper 2. Availability: INIS. Available from NTIS, PC A03/MF A01 as DE87010749.
AB: During operational abnormalities, plant operators rely on the information provided by the plant sensors and associated alarms. The sensors' usefulness however, is quickly diminished by their large number and the extremely difficult task of interpreting and comprehending the provided information. Malfunction diagnosis can be further complicated by the existence of conflicting data which can lead to an incorrect diagnostic conclusion. Thus, the value of an operator aid to assist plant personnel in interpreting the available data and diagnosing the plant malfunctions is obvious. Recent work at the Ohio State University Laboratory for **Artificial Intelligence** Research (OSU-LAIR) and the Nuclear Engineering Department has concentrated on the problem of performing **expert system** diagnosis using potentially invalid sensor data. **Expert systems** have been developed that can perform diagnostic problem solving despite the existence of some conflicting data in the domain. This work has resulted in enhancement of a programming tool, CSRL, that allows domain experts to create a diagnostic **system** that will be, to some degree, tolerant of bad data. The domain of Boiling Water Nuclear **Power Plants** was chosen as a test domain to show usefulness of the ideas under real world conditions.

DOCUMENT NUMBER = INI19:022292

TI: **Expert system** application to fault diagnosis and procedure synthesis.
AU: Hajek, B.K.; Hashemi, S.; Bhatnagar, R.; Miller, D.W.; Stasenko, J.
LA: English
MS: Application of **artificial intelligence** to the operation of nuclear **power plants:** Progress report, September 15, 1986-June 4, 1987. Hazek, B.K.; Miller, D.W. Ohio State Univ. Research Foundation, Columbus (USA).
RP: DOE/NE/37965--T1.
IM: Jun 1987. Paper 1. Availability: INIS. Available from NTIS, PC A03/MF A01 as DE87010749.
AB: Two **knowledge based systems** have been developed to detect plant faults, to validate sensor data in a nuclear **power** plant, and to synthesize procedures to assure safety goals are met when a transient occurs. These two **systems** are being combined into a single **system** through a Plant Status Monitoring **System** (PSMS) and a common data base accessed by all the components of the integrated **system.** The **system** is designed to

sit on top of an existing Safety Parameter Display **System** (SPDS), and to use the existing data acquisition and data control software of the SPDS. The integrated **system** will communicate with the SPDS software through a single database. This database will receive sensor values and equipment status indications in a form acceptable to the **knowledge based system** and according to an update plan designed specifically for the **system.**

DOCUMENT NUMBER = INI19:019389

TI: Integrated operator-plant process modeling and decision support for allocation of function.
AU: Schryver, J.C.; Knee, H.E.
CO: Oak Ridge National Lab., TN (USA).
LA: English
RP: CONF-8710101--1.
IM: 1987. 6 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE87011942.
NO: Portions of this document are illegible in microfiche products.
CF: (31. annual Human Factors Society meeting. New York, NY (USA). 19- 23 Oct 1987.)
AB: Human operator simulation models can play an important information role in the allocation of functions in person-machine **systems.** A prototype simulation model **system** developed at ORNL is described in which a human operator model (INTEROPS) and a nuclear **power** plant (NPP) process model are dynamically integrated. INTEROPS is a cognitive/performance simulation model which is itself a dynamic integration of a SAINT task network model and a knowledge-based subsystem which reasons with uncertainty. Potential contributions of INTEROPS to NPP advanced control design are evaluated. 8 refs., 3 figs.

DOCUMENT NUMBER = INI19:019384

TI: Use of an advanced **expert system** tool for fault tree analysis in nuclear **power plants.**
AU: Frogner, B.; Dourandish, R. (Expert-EASE **Systems,** Inc., 1301 Shoreway Road, Belmont, CA 94002).
LA: English
MS: Thirty-second IEEE computer society international conference (Digest of Papers). Anon.
IM: Piscataway, NJ (USA). IEEE Service Center. 1987. p. 454-457. ISBN 0-8186-0764-5.
CF: (COMPCON '87: 32nd IEEE Computer Society international conference. San Francisco, CA (USA). 23-26 Feb 1987.)
AB: Fault tree analysis is used extensively in probabilistic risk analysis of nuclear **power plants.** The results summarized in this paper indicate that the use of an **expert system** approach can improve the state-of-the-art of fault tree construction. It simplifies the way fault trees are specified and it allows the incorporation of prior **knowledge** and experience. In particular, the **expert system** technology is highly effective in promoting consistency throughout the tree and giving "**expert**" assistance in resolution of logical loops.

DOCUMENT NUMBER = INI19:019375

TI: Representation of the deduction with tetravalent logic: realisation of fast inference motors. Representation de la deduction par une logique trivalente: realisation de moteurs d'inference rapides.
AU: Terrier, F.
CO: CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. d'Electronique et d'Instrumentation Nucleaire.
LA: French
RP: CEA-N--2530.
IM: May 1987. 365 p. Availability: INIS.
NO: Designation: These (D. es Sci.).
AB: Building automats for surveillance, control or diagnosis, can be done by **artificial intelligence.** They function according to the formalism of O ordre propositional logic. Deduction modes can then easily be represented in a trivalent logic with both

efficiency and concision. One can obtain workable codes with low storage of high speed (env. 5000 rules per second). This leads us to develop and entry langage with extremely simple statement rules which can be learnt by experts in practically no time. Finally in order to generalize this principle and draw rules with multi-choice conclusions, (if... then the motor stops or the pressure goes down), rewriting these processes enable to return to an equivalent **system** without multi-choice conclusions, providing the same amount of informations with the possibility of deciding. This study has been used on 2 different cases: first it served as a tool to analyse reliability, (OLAF), second to help diagnosis breakdown of pressurized water-reactors.

DOCUMENT NUMBER = INI19:018796

TI: An **artificial intelligence** approach towards disturbance analysis in nuclear **power plants.**
AU: Lindner, A.; Klebau, J.; Fielder, U.; Baldeweg, F. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)).
LA: English
MS: Load following control of nuclear **power plants** including availability aspects (Proceedings of the specialists' meeting held at Bombay during 10-12 Dec 1985). International Atomic Energy Agency, Vienna (Austria). International Working Group on Nuclear **Power** Plant Control and Instrumentation.
IM: Bombay (India). Department of Atomic Energy. Mar 1987. 243 p. p. 123-141, 145-149.
NO: Discussion on p. 145-149.
CF: (Specialists' meeting on load following control of nuclear **power plants** including availability aspects. Bombay (India). 10-12 Dec 1985.)
AB: The scale and degree of sophistication of technological **plants,** e.g. nuclear **power plants,** have been essentially increased during the last decades. Conventional disturbance analysis **systems** have proved to work successfully in wellknown situations. But in cases of emergencies, the operator staff needs a more advanced assistance in realizing diagnosis and therapy control. The significance of introducing **artificial intelligence** methods in nuclear **power** technology is emphasized. Main features of the on-line disturbance analysis **system** SAAP-2 are reported about. It is being developed for application in nuclear **power plants.** 9 refs. (author).

DOCUMENT NUMBER = INI19:018779

TI: A nuclear **power** plant status monitor.
AU: Chu, B.B.; Conradi, L.L.; Weinzimmer, F. (Electric **Power** Research Institute).
LA: English
MS: Proceedings of the 1986 joint ASME/ANS nuclear **power** conference. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 229- 234.
CF: (ASME/ANS bi-annual nuclear **power** conference: safe and reliable nuclear **power plants.** Philadelphia, PA (USA). 20-23 Jul 1986.)
AB: **Power** plant operation requires decisions that can affect both the availability of the plant and its compliance with operating guidelines. Taking equipment out of service may affect the ability of the plant to produce **power** at a certain **power** level and may also affect the status of the plant with regard to technical specifications. Keeping the plant at a high as possible production level and remaining in compliance with the limiting conditions for operation (LCOs) can dictate a variety of plant operation and maintenance actions and responses. Required actions and responses depend on the actual operational status of a nuclear plant and its attendant **systems,** trains, and components which is a dynamic situation. This paper discusses an Electric **Power** Research Institute (EPRI) Research Project, RP 2508, the objective of which is to combine the key features of plant information management **systems** with **systems** reliability analysis techniques in order to assist nuclear **power** plant personnel to perform their functions more efficiently and effectively. An overview of the EPRI Research Project is provided along with a detailed discussion of the design and operation of the PSM portion of the project.

DOCUMENT NUMBER = INI19:018711

TI: Functional consistency checking module for signal validation: Topical report.
AU: Zwingelstein, G.; Kerlin, T.W.; Upadhyaya, B.R.
CO: Tennessee Univ., Knoxville (USA). Dept. of Nuclear Engineering; Electricite de France, 78 - Chatou.
LA: English
RP: DOE/NE/37959--10.
IM: Aug 1987. 31 p. Availability: INIS. Available from NTIS, PC A03/ MF A01; 1 as DE87014338.
NO: Portions of this document are illegible in microfiche products.
AB: This report discusses the basic principles of a functional consistency checking module for signal validation in nuclear **plants**. This module will add additional features to the signal validation **system** currently being designed at the University of Tennessee. The goals of the functional consistency module are to provide solutions to the common mode features and single or dual sensor failures. The basic principle lies in the comparison of the actual outputs and outputs derived from reference static or dynamics models. The only requirement needed for this method is that the plant must be decomposed into subsystems with observable inputs and outputs. The reference models can be either empirical or derived from physical laws. The main feature of this module is its capability to separate component failures from sensor failures. The failures which can be isolated are bias, drifts, absormal noise or changes in the dynamic sensor characteristics. The decision-making process relies basically on truth table look-up. The truth table contents are derived from an apriori **knowledge** of the failure causes. More elaborate decision making processes **based** on pattern recognition techniques or **expert systems** are proposed to improve the reliability of the decisions. A modified sequential probability ratio test is suggested to handle adequately the modeling errors. The proposed functional consistency checking is valid for both steady state and transient conditions. In the latter case, the transfer functions of the instrumentation channels must be taken into account for a fast and accurate failure identification. To conclude the report, anticipated benefits of the functional consistency checking are listed.

DOCUMENT NUMBER = INI19:018064

TI: A unified plant information network.
AU: Niederauer, G.F. (Energy Inc., Idaho Falls, ID).
LA: English
MS: Proceedings of the 1986 joint ASME/ANS nuclear **power** conference. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 219- 221.
CF: (ASME/ANS bi-annual nuclear **power** conference: safe and reliable nuclear **power plants**. Philadelphia, PA (USA). 20-23 Jul 1986.)
AB: Technology is bringing **power plants** fully into the age of computerization. Microcomputers, data base managers, networking, and friendly, **expert** software are principal technology factors. Monitoring will improve, and the number and **power** of computers is increasing. The huge information flow will cause computers to be integrated into a communication network. The total plant operating triangle includes process, engineering, and management **systems.** The total network will integrate all of these into a Total Unified Plant Information Network (TUPIN). Software will take the type of information beyond monitored data. Analysis will improve through direct access to logical, physical, and procedural models by end users. Information management will improve through widespread use of hierarchical, relational, and **expert** data base managers. **Expert systems** will aid in diagnostics and interpretation. The goal is to automate plant operations to enhance safety and performance and to reduce cost by making both the **plants** and the personnel more **expert**.

DOCUMENT NUMBER = INI19:013790

TI: WWER diagnostics. Diagnostika reaktoru VVER.
AU: Erben, O.; Horinek, K.; Houska, Z.; Rejchrt, J.; Rygl, J.; Stulik, P.; Vasa, I. (Ustav Jaderneho Vyzkumu CSKAE, Rez (Czechoslovakia)).

LA: Czech
MS: Nuclear **Power** Day '86. Collection of papers. Den jaderne energetiky '86. Sbornik prednasek. Ustav Jaderneho Vyzkumu CSKAE, Rez (Czechoslovakia). Published in two parts.
RP: INIS-mf--11096.
IM: 9 Dec 1986 . 277 p. p. 143-157. Availability: INIS.
NO: English translation available from Nuclear Information Center, 156 16 Prague 5-Zbraslav, Czechoslovakia at US$ 10 per page.
CF: (Nuclear **Power** Day '86. Rez near Prague (Czechoslovakia). 1 Oct 1986.)
AB: Within the state research project "Development of Nuclear **Power** till the Year 2000", the Institute of Nuclear Research is involved in a task whose scope is the diagnostics of the condition of the reactor core, the in-core inspection **system**, the mechanical behavior of the reactor internals and of the primary circuit piping. The procedures and instruments are briefly described used for noise diagnostics as are **expert systems** with special regard to the application of these techniques in nuclear **power plants.** Currently, the studies are concentrated to modelling the effects diagnosed in the reactor and the primary coolant circuit and to creating conditions for the implementation of an **expert system** for the Dukovany nuclear **power** plant. (Z.M.). 15 refs.

DOCUMENT NUMBER = INI19:013680

TI: Ways to prevent errors of operator personnel at NPP. Puti predotvrashcheniya oshibok operativnogo personala na AEhS.
AU: Chachko, S.A. (Kievskij Inst. Avtomatiki (Ukrainian SSR)).
LA: Russian
JR: Ehlektr. Stn. ISSN 0013-5372. CODEN: EKSTA. (Mar 1987). (no.3) p. 6-10.
AB: Problems of possibilities and limitations of NPP operator personnel are discussed. Necessity of the **expert system** development analysing potential sources of errors and improving operator readiness to solutions by means of participation in antiemergency plays is discussed. Reduction of the error number is possible by means of radiation designing of advanced control rooms, especially code standartization, development of an optimal set of display formays, development of language and procedure of effective dialogue man - computer in the **expert system.** To develop an operator support **system,** it is necessary to form the machine data bank on operative errors.

DOCUMENT NUMBER = INI19:013679

TI: A fundamental study on nuclear **power** plant diagnosis **system.**
AU: Yoshimura, Sei-ichi; Fujimoto, Junzo (Central Research Inst. of Electric **Power** Industry, Komae, Tokyo (Japan). Komae Research Lab.).
LA: Japanese
JR: Denryoku Chuo Kenkyusho Hokoku. CODEN: DCKHD. (May 1987). (no.T86060) p. 1-35.
AB: Diagnosis of nuclear **power** plant is a large application field of **knowledge** engineering. But, the study examples are few and the diagnosis method is not established yet. This report describes the diagnosis method using cross correlation coefficients and describes the **knowledge** acquisition method of undefined transients in order to enhance the **system** performance. The usefulness of the **system** was verified by putting some data into the **system.** Main results are as follows. (1) Diagnosis method. Some transients are selected by the first judgement and one of them is identified by the second judgement using the cross correlation. (2) **Knowledge** aquisition method. When putting new data into the knowledge-base, the **system** indicates the inconsistency by arranging the aquired data, and the operators input new transient names and corresponding manipulation methods after analyzing the indicated results. (3) Usefulness of the **system.** Freedwater controller failures(2 transients), 2 recirculation pumps trip and a dummy datum combined 2 transients(one is feedwater controller failure and one is 2 recirculation pumps trip) were put into the **system.** It was proved that the **system** identified the transients correctly and it indicated the first hit and the inconsisency of the transients in the course of **knowledge** acquisition. (author).

DOCUMENT NUMBER = INI19:013678

TI: A survey on the present status and applicability of human factors engineering methods to the operational and observation task in nuclear **power plants.**
AU: Nagasaka, Akihiko; Yoshino, Kenji; Takano, Ken-ichi (Central Research Inst. of Electric **Power** Industry, Komae, Tokyo (Japan). Komae Research Lab.).
LA: Japanese
JR: Denryoku Chuo Kenkyusho Hokoku. CODEN: DCKHD. (May 1987). (no.T86048) p. 1-58.
AB: To enhance the operational safety and prevent human errors in nuclear **power plants,** it is necessary to work out a overall countermeasure from the point of view of human factors. This paper presents the results of literature survey on the present status of human factors engineering methods applicabilities to the operation and observation task in nuclear **power plants.** Results of this survey reveals the lack of research of man-machine interface. Then we propose Total Human Activity and Performance Optimization Research Technique (THAPORT) with the aim of preventing human errors and also discuss the following. (1) Present status and problems of human performance improvement research to the operation and observation task. (2) The necessity or research about man-machine interface with the aim of preventing human errors. (3) A proposal of THAPORT and the examination of influence of the research technique. (author) 59 refs.

DOCUMENT NUMBER = INI19:013643

TI: Decision support **systems** and **expert systems** for risk and safety analysis.
AU: Baybutt, P. (Battelle Columbus Labs., OH (USA)).
LA: English
MS: Hazards in the process industries: Hazards IX. Symposium on hazards in the process industries: Hazards IX A three-day symposium organised by the Institution of Chemical Engineers (North Western Branch) and the IChemE's Safety and loss prevention Group and held at the University of Manchester Institute for Science and Technology (UMIST), 2-4 April 1986. Institution of Chemical Engineers, London (UK). EFCE event--341.
IM: London (UK). Institution of Chemical Engineers. 1986. ISBN 0 85295 198 1. 327 p. p. 261-268.
SE: Institution of Chemical Engineers symposium series. (no.97).
CF: (Symposium on hazards in the process industries: Hazards IX. Manchester (UK). 2-4 Apr 1986.)
AB: During the last 1-2 years, rapid developments have occurred in the development of decision support **systems** and **expert systems** to aid in decision making related to risk and safety of industrial **plants.** These activities are most noteworthy in the nuclear industry where numerous **systems** are under development with implementation often being made on personal computers. An overview of some of these developments is provided, and an example of one recently developed decision support **system** is given. This example deals with CADET, a **system** developed to aid the U.S. Nuclear Regulatory Commission in making decisions related to the topical issue of source terms resulting from degraded core accidents in light water **reactors.** The paper concludes with some comments on the likely directions of future developments in decision support **systems** and **expert systems** to aid in the management of risk and safety in industrial **plants.** (author).

DOCUMENT NUMBER = INI19:013595

TI: An integrated approach for signal validation in nuclear **power plants.**
AU: Upadhyaya, B.R.; Kerlin, T.W.; Gloeckler, O.; Frei, Z.; Qualls, L.; Morgenstern, V.
CO: Tennessee Univ., Knoxville (USA). Dept. of Nuclear Engineering.
LA: English
RP: DOE/NE/37959--9. CONF-870832--1.
IM: Aug 1987. 9 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE87014227.
NO: Portions of this document are illegible in microfiche products.
CF: (Topical meeting on **artificial intelligence** and other innovative computer applications in the nuclear industry. Snowbird, UT (USA). 31 Aug - 2 Sep 1987.)

AB: A signal validation **system, based** on several parallel signal processing modules, is being developed at the University of Tennessee. The major modules perform (1) general consistency checking (GCC) of a set of redundant measurements, (2) multivariate data-driven modeling of dynamic signal components for maloperation detection, (3) process empirical modeling for prediction and redundancy generation, (4) jump, pulse, noise detection, and (5) an **expert system** for qualitative signal validation. A central database stores information related to sensors, diagnostics rules, past **system** performance, subsystem models, etc. We are primarily concerned with signal validation during steady-state operation and slow degradations. In general, the different modules will perform signal validation during all operating conditions. The techniques have been successfully tested using PWR steam generator simulation, and efforts are currently underway in applying the techniques to Millstone-III operational data. These methods could be implemented in advanced **reactors,** including advanced liquid metal **reactors.**

DOCUMENT NUMBER = INI19:012868

TI: Application of **artificial intelligence** to the operation of nuclear **power plants:** Progress report, September 15, 1986-June 4, 1987.
AU: Hazek, B.K.; Miller, D.W.
CO: Ohio State Univ. Research Foundation, Columbus (USA).
LA: English
RP: DOE/NE/37965--T1.
IM: Jun 1987. 37 p. Availability: INIS. Available from NTIS, PC A03/ MF A01; 1 as DE87010749.
NO: *Portions of this document are illegible in microfiche products.*
AB: This report summarizes progress to date on Department of Energy grant No. AC02-86N37965. Work began on this project September 15, 1986, and is scheduled to be completed December 31, 1987. The project has five major goals: combining two previously designed **knowledge based systems;** evaluation and selection of operational scenarios for testing and demonstration, evaluation and selection of a demonstration site; arrangement of an alliance with an industrial collaborator; and evaluation and specification of hardware and software requirements for connection to a full function simulator. Two papers delivered at the EPRI Seminar on **Expert Systems** Applications in **Power Plants** in May 1987, form a part of this report. In addition, two other paper summaries are included providing greater detail and updating development progress. These four individual papers have been cataloged separately. (FI).

DOCUMENT NUMBER = INI19:009499

TI: Instrumentation in the **power** industry: Volume 29.
AU: Anon.
LA: English
IM: Research Triangle Park, NC (USA). Instrument Society of America. 1986. 220 p. ISBN 0-87664-965-7.
AB: This book compiles the papers presented in the conference. Topics of the papers are: total distribution control and data acquisition; nuclear plant testing and requirements of the plant instrument engineer; advanced control concepts; nuclear plant control and monitor **systems** upgrades; **artificial intelligence** in control; and boiler control and data acquisition **systems** upgrades.

DOCUMENT NUMBER = INI19:009186

TI: Application of Goal Tree-Success Tree model as the knowledge-base of operator advisory **systems.**
AU: *Kim, I.S.;* Modarres, M. (Maryland Univ., College Park (USA). Dept. of Chemical and Nuclear Engineering).
LA: English
JR: Nucl. Eng. Des. ISSN 0029-5493. CODEN: NEDEA. (Oct 1987). v. 104(1) p. 67-81.

AB: The most important portion of an **expert system** development is the articulation of **knowledge by the expert** and its satisfactory formulation in a suitable **knowledge** representation scheme for mechanization by a computer. A 'deep **knowledge**' approach called Goal Tree-Success Tree model is devised to represent complex dynamic domain **knowledge.** This approach can hierarchically model the underlying principles of a given process domain (for example nuclear **power** plant operations domain). The Goal Tree-Success Tree can then be used to represent the knowledge-base and provide means of selecting an efficient search routine in the inference engine of an **expert system.** A prototype **expert system** has been developed to demonstrate the method. This **expert system** models the operation of a typical **system** used in the pressurized water **reactors.** The **expert system** is modeled for real-time operations if an interface between plant parameters and the **expert system** is established. The real- time operation provides an ability to quickly remedy minor disturbances that can quickly lead to a **system** malfunction or trip. A description of both the Goal Tree-Success Tree model and the prototype **expert system** is presented. (orig.).

DOCUMENT NUMBER = INI19:009124

TI: **Expert systems** for the analysis of transients on nuclear **reactors**: crisis analysis, sextant, a general purpose physical analyser.
AU: Barbet, N.; Dumas, M.; Mihelich, G.; Souchet, Y.; Thomas, J.B.
CO: CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. d'Etudes Mecaniques et Thermiques.
LA: English
RP: CEA-CONF--9034.
IM: Apr 1987. 8 p. Availability: INIS.
CF: (International topical meeting on advances in reactor physics, mathematics and computations. Paris (France). 27-30 Apr 1987.)
AB: Two developments of **expert systems** intended to work on line to the analysis of nuclear reactor transients are reported. During an hypothetical crisis occurring in a nuclear facility, a staff of the Institute for Protection and Nuclear Safety (IPSN) has to assess the risk to local population. The **expert system** is intended to work as an assistant to the staff. At the present time, it deals with the availability of the safety **systems** of the plant (e.g. ECCS), depending on the functional state of the support **systems.** A next step is to take into account the physical transient of the reactor (mass and energy balance, pressure, flows). In order to reach this goal as in the development of other similar **expert systems,** a physical analyser is required. This is the aim of SEXTANT, which combines several **knowledge** bases concerning measurements, models and qualitative behaviour of the plant with a mechanism of conjecture- refutation and a set of simplified models matching the current physical state. A prototype is under assessment by dealing with integral test facility transients. Both **expert systems** require powerful shells for their development. SPIRAL is such a toolkit for the development of **expert systems** devoted to the computer aided management of complex processes.

DOCUMENT NUMBER = INI19:008494

TI: Applications and perspective of PRA methods in Mexico.
AU: Schwarzblat, M.; Arellano, J. (Instituto de Investigaciones Electricas, Cuernavaca, Morelos).
LA: English
MS: Proceedings of the international conference on nuclear **power** plant aging, availability factor and reliability analysis. Goel, V.S.
IM: Metals Park, OH (USA). American Society for Metals. 1985. p. 647- 651.
CF: (International conference on nuclear **power** plant aging, availability factor and reliability analysis. San Diego, CA (USA). 7-12 Jul 1985.)
AB: Probabilistic risk assessment (PRA) methods are widely used to evaluate safety **systems,** particularly in the nuclear industry. In Mexico, a considerable effort has been devoted to both implementation and development of computerized methods for PRA studies, as well as to the establishment of procedures and guidelines for the application of PRA methods as a tool for **systems** analysis during the design, construction and operation of complex **systems.** Most of the work in this area was

initiated at IIE, the national electric research institute of Mexico. However, a close coordination with other organizations has been established in order to optimize available resources. In this paper, both the computerized methods and the application studies performed at IIE are presented.

DOCUMENT NUMBER = INI19:005827

TI: CLEO: A knowledge-based refueling assistant at FFTF.
AU: Smith, D.E.; Kocher, L.F.; Seeman, S.E.
CO: Hanford Engineering Development Lab., Richland, WA (USA).
LA: English
RP: HEDL-SA--3359. CONF-851115--54.
IM: Jul 1985. 7 p. Availability: INIS. Available from NTIS, PC A02; 3 as DE87010411.
NO: Paper copy only, copy does not permit microfiche production.
CF: (American Nuclear Society winter meeting. San Francisco, CA (USA). 10-15 Nov 1985.)
AB: CLEO is computer software **system** to assist in the planning and performance of the reactor refueling operations at the Fast Flux Test Facility. It is a recently developed application of **artificial intelligence** software with both **expert systems** and automated reasoning aspects. The computer **system** seeks to organize the sequence of core component movements according to the rules and logic used by the **expert**. In this form, CLEO has aspects which tie it to both the **expert systems** and automated reasoning areas within the **artificial intelligence** field.

DOCUMENT NUMBER = INI19:005619

TI: Project of an online information **system** for monitoring and diagnosing neutron and acoustic fluctuations in fast **reactors**. Use of **artificial intelligence** in the design of a detection and diagnostic **system based** on noise analysis. Projet de **systeme** informatique de suivi et de diagnostic en ligne applique aux fluctuations neutronique et acoustique des R.N.R. Application de l'intelligence artificielle au projet de **systeme** de detection et de diagnostic base sur l'analyse des bruits.
AU: Le Guillou, G.; Malvache, P.; Himbaut, S.; Pham, H. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-lez-Durance (France)).
LA: French
MS: 18th informal meeting on reactor noise. Dach, K. (ed.). Ceskoslovenska Komise pro Atomovou Energii, Prague.
RP: INIS-mf--11077.
IM: 1986. 288 p. p. 268-284. Availability: INIS.
CF: (18. informal meeting on reactor noise. Prague (Czechoslovakia). 10-12 Apr 1985.)
AB: The objective of the project is the development of an information **system** for on-line surveillance and primary diagnostics of a fast reactor nuclear **power** plant, this on the bsis of neutron and acoustic noise. The **system** will be modular and will use real time signal processing methods and pattern recognition methods. The basic steps are described of both stages and implementation criteria are listed. The entire project is scheduled for 3 years and should be finished in 1986. (A.K.) 3 figs.

DOCUMENT NUMBER = INI19:005618

TI: The on-line surveillance **system** by Merlin Gerin.
AU: Indjirdjian, J.; Vullierme, C. (Etablissements Merlin et Gerin, 38 - Grenoble (France)).
LA: English
MS: 18th informal meeting on reactor noise. Dach, K. (ed.). Ceskoslovenska Komise pro Atomovou Energii, Prague.
RP: INIS-mf--11077.
IM: 1986. 288 p. p. 251-267. Availability: INIS.
CF: (18. informal meeting on reactor noise. Prague (Czechoslovakia). 10-12 Apr 1985.)
AB: Automated diagnostic **system** SES was developed for French 1300 MW pressurized water **reactors**. The **system** primarily fulfils the functions of monitoring primary circuit component leaks, loose parts and vibrations inside the reactor core. The core of the **system** is a PDP 11/23 computer to which up to 15 stellite units can be connected. Further development of the SES **system** is focused on the extension of diagnostic

possibilities by the detection of fuel cladding damage and by the monitoring of primary coolant flow and of the vibrations of main circulation pumps. **Artificial intelligence** elements will be incorporated in the existing **system** and it will also be possible to adapt the whole **system** to BWR's and fast **reactors.** (Z.M.) 10 figs.

DOCUMENT NUMBER = INI19:005165

TI: **Expert system** technology for control integration in nuclear **reactors.**
AU: Stabler, E.P. Jr.; Zimmerman, J.J.; Stratton, R.C.
CO: Quintus Computer **Systems,** Palo Alto, CA (USA); Hanford Engineering Development Lab., Richland, WA (USA).
LA: English
RP: HEDL-SA--3511-FP. CONF-8604297--1.
IM: Mar 1986. 13 p. Availability: INIS. Available from NTIS MF A01 as DE87004438.
NO: Microfiche only, copy does not permit paper copy reproduction.
CF: (6. international workshop on **expert systems** and their applications. Avignon (France). 28-30 Apr 1986.)
AB: This report describes the role of **expert system** technology in nuclear **power** plant operation. The use of computers to assist operator decisions would greatly enhance the safety and efficiency of operation. A description of the necessary operator interfaces, data acquisition and validation, plant status and parameter diagnosis, and **system** reliability is presented. (FL).

DOCUMENT NUMBER = INI19:002815

TI: A control **system** verifier using automated reasoning software.
AU: Smith, D.E.; Seeman, S.E.
CO: Hanford Engineering Development Lab., Richland, WA (USA).
LA: English
RP: HEDL-SA--3300. CONF-850903--31.
IM: Aug 1985. 7 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE87010415.
NO: Portions of this document are illegible in microfiche products.
CF: (International topical meeting on computer applications for nuclear **power** plant operation and control. Pasco, WA (USA). 8-12 Sep 1985.)
AB: An on-line, automated reasoning software **system** for verifying the actions of other software or human control **systems** has been developed. It was demonstrated by verifying the actions of an automated procedure generation **system.** The verifier uses an interactive theorem prover as its inference engine with the rules included as logical axioms. Operation of the verifier is generally transparent except when the verifier disagrees with the actions of the monitored software. Testing with an automated procedure generation **system** demonstrates the successful application of automated reasoning software for verification of logical actions in a diverse, redundant manner. A higher degree of confidence may be placed in the verified actions of the combined **system.**

DOCUMENT NUMBER = INI19:002787

TI: Signal validation in nuclear **power plants:** Annual progress report for the period September 30, 1986-September 29, 1987.
AU: Kerlin, T.W.; Upadhyaya, B.R.; Gloeckler, O.; Frei, Z.; Qualls, L.; Gaudio, P.J.
CO: Tennessee Univ., Knoxville (USA). Dept. of Nuclear Engineering; Combustion Engineering, Inc., Windsor, CT (USA).
LA: English
RP: DOE/NE/37959--1.
IM: May 1987. 51 p. Availability: INIS. Available from NTIS, PC A04/ MF A01; 1 as DE87010234.
NO: Portions of this document are illegible in microfiche products.
AB: In large sysstems, such as nuclear **power plants,** chemical **plants** and others, signals from various instrumentation **systems** are channeled to control **systems,** protection **systems,** and plant monitoring **systems.** Validation of these signals is necessary to

minimize plant downtime, to increase the reliability of operator decisions and to aid in scheduling plant maintenance. The research program at the University of Tennessee and its subcontractor, Combustion Engineering, Inc.,, is directed towards developing a comprehensive signal validation **system** for implementation in current and future **power plants** (advanced **reactors**). The signal validation architecture consists of parallel signal processing modules, each of which implements a diverse validation scheme. The various modules are being evaluated using PWR plant simulation data from combustion Engineering, Inc., and operational data from Northeast Utilities Millstone III plant. Currently, all the modules are being developed and tested using the IBM-AT computer.

DOCUMENT NUMBER = INI19:002029

TI: Procedure generation and verification.
AU: Sheely, W.F.
CO: Hanford Engineering Development Lab., Richland, WA (USA).
LA: English
RP: HEDL-SA--3573-VT.
IM: 15 Jul 1986. 7 p. Availability: INIS. Available from NTIS, PC A02/ MF A01; 1 as DE87010221.
NO: Portions of this document are illegible in microfiche products. Text of videotape stored at Battelle Northwest, Audio-Visual, Richland, WA.
AB: The Department of Energy has used **Artificial Intelligence** of "AI" concepts to develop two powerful new computer-based techniques to enhance safety in nuclear applications. The Procedure Generation **System,** and the Procedure Verification **System,** can be adapted to other commercial applications, such as a manufacturing plant. The Procedure Generation **System** can create a procedure to deal with the off-normal condition. The operator can then take correct actions on the **system** in minimal time. The Verification **System** evaluates the logic of the Procedure Generator's conclusions. This evaluation uses logic techniques totally independent of the Procedure Generator. The rapid, accurate generation and verification of corrective procedures can greatly reduce the human error, possible in a complex (stressful/high stress) situation.

DOCUMENT NUMBER = INI18:101042

TI: Collection, processing and use of data.
AU: Mancini, G. (JRC-Ispra (Italy)).
LA: English
MS: Safety and reliability in Europe. Proceedings of the pre-launching meeting of the European Safety and Reliability Association (ESRA). Colombo, A.G. (ed.). Commission of the European Communities, Ispra (Italy). Joint Research Centre; Commission of the European Communities, Luxembourg.
RP: EUR--10006.
IM: 1985. 576 p. p. 247-285. ISBN 92-825-5841-X. Availability: INIS.
CF: (Pre-Launching Meeting of the European Safety and Reliability Association (ESRA). Ispra (Italy). 9-12 Oct 1984.)
AB: Data are now bring collected in many types of activities, such as nuclear **reactors** and annexed fuel cycle facilities, transport of hazardous materials. The proliferation in the past few years of data banks related to safety and reliability has somewhat alleviated the problem of lack of information when carrying out safety analysis; but it has succeeded meanwhile to identify several crucial aspects in the overall procedures of collection, processing and use of data. These aspects are reviewed in the paper with special emphasis on the interaction of the information and the various interpretative models of component and **system** behaviour. The need for the analyst of not only validating stated models but also of attempting new interpretations of the facts is particularly stressed. Eventually the role of new **Artificial Intelligence** techniques in supporting man in his search for model structures - that is the "history" underlying the facts - is mentioned. 12 refs.

DOCUMENT NUMBER = INI18:100636

TI: **Expert system** for space **power** supplies.
AU: Cooper, R.S.; Hoshor, A.; Thomson, M.K. (Creative Enterprises, San Diego, CA).
LA: English
MS: Transactions of the symposia on space nuclear **power systems**. Summaries 1986. El-Genk, M.S.; Hoover, M.D. (eds.). New Mexico Univ., Albuquerque (USA). Inst. for Space Nuclear **Power** Studies.
RP: CONF-860102--Summs.
IM: 1986. p. SI-1.1-SI-1.3. Availability: INIS. Available from NTIS, PC A13/MF A01; 1 as DE86005726.
CF: (3. symposium on space nuclear **power systems**. Albuquerque, NM (USA). 13-16 Jan 1986.)
AB: Design and evaluation of space **power** supplies involves many qualitative, uncertain and heuristic factors that cannot be handled by conventional algorithmic computer programs. The authors are applying **Artificial** Intelligence/Expert **Systems** techniques to provide tools for designers and managers for the synthesis and analysis of space **power** supplies. The authors have evaluated the feasibility of an **Expert System** for the identification and selection of supplies for a wide range of NASA missions of various **power** levels (P) and durations.

DOCUMENT NUMBER = INI18:100611

TI: FBR plant operation support **system**.
AU: Sato, Masuo; Fukawa, Naohiro (Toshiba Corp., Kawasaki, Kanagawa (Japan)); Tamaoki, Tetsuo; Aoki, Hiroshi.
LA: Japanese
JR: Toshiba Rebyu. ISSN 0372-0462. CODEN: TORBA. (May 1987). v. 42(5) p. 383-386.
AB: The development of the most suitable knowledge-based operation support **system** is demanded for the purpose of operating FBR **plants** with higher stability and greater safety. This **system** must accumulate the thinking processes of experts as **knowledge**, and in emergency, it must offer necessary information to operators and to support them for understanding plant situation and making necessary corrective action. This paper describes the whole operation **system** for FBR and the JOYO consulting-and-analyzing tool (JOYCAT), which is a knowledge-based alarm-handling **system** using "TOSFile". (author).

DOCUMENT NUMBER = INI18:100428

TI: Mobile robot response to actions associated with the release of hazardous materials.
AU: Meieran, H.B. (HB Meieran Assoc., Pittsburgh, PA).
LA: English
MS: ANS (American Nuclear Society) topical meeting on radiological accidents: Perspectives and emergency planning: Proceedings. Oak Ridge National Lab., TN (USA).
RP: CONF-860932--.
IM: Mar 1987. p. 95-100. Availability: INIS. Available from NTIS, PC A16/MF A01; 1 as DE87007690.
CF: (Radiological accidents, perspectives and emergency planning preparedness. Bethesda, MD (USA). 15-17 Sep 1986.)
AB: This paper presents a rational and composite summary of tasks and missions that could be assigned to mobile robots and other teleoperated devices in response to accidental releases of radioactive and other hazardous/toxic materials to the environment. This paper will also discuss specific missions that have been, or could be, assigned to mobile robots operating at the TMI-2 and Chernobyl-4 nuclear **power** plant sites. Other items and issues that will also be considered are: availability of applicable mobile robot units, expendability/durability/decontamination, portability/ maneuverability/mobility, communication techniques, **power** supply, and **artificial** intelligence/autonomous navigation interactions.

DOCUMENT NUMBER = INI18:100417

TI: Development of maintenance support **system** for nuclear **power plants.**
AU: Ujita, Hiroshi; Kiguchi, Takashi (Hitachi Ltd., Ibaraki (Japan). Energy Research Lab.); Onodera, Katsushige; Komata, Masaoki.
LA: Japanese
JR: Nippon Genshiryoku Gakkai-Shi. ISSN 0004-7120. CODEN: NGEGA. (Jun 1987). v. 29(6) p. 538-547.
AB: A maintenance support **system,** which utilizes maintenance **expert** know- how and failure experience, is developed by applying the method of **knowledge** engineering. The **system** searches failure causes from failure symptoms by repeating forward and backward chaining. It then estimates the certainty factor of the failure causes by using the certainty factors of the symptoms and of the failure cause- consequence relation, and the occurrence probabilities of candidate causes. The **system** also provides a plant maintenance scheme **based** on maintenance priority. Application of the **system** to the diagnosis and maintenance scheme evaluation on BWR plant control rod drive **systems** demonstrates the **system's** usefulness. (author).

DOCUMENT NUMBER = INI18:100328

TI: Intelligent operation **system** for nuclear **power plants.**
AU: Morioka, Toshihiko; Fukumoto, Akira; Suto, Osamu (Toshiba Corp., Kawasaki, Kanagawa (Japan)); Naito, Norio.
LA: Japanese
JR: Toshiba Rebyu. ISSN 0372-0462. CODEN: TORBA. (May 1987). v. 42(5) p. 371-374.
AB: Nuclear **power plants** consist of many **systems** and are operated by skillful operators with plenty of **knowledge** and experience of nuclear **plants.** Recently, plant automation or computerized operator support **systems** have come to be utilized, but the synthetic judgment of plant operation and management remains as human roles. Toshiba is of the opinion that the activities (planning, operation and maintenance) should be integrated, and man-machine interface should be human-friendly. We have begun to develop the intelligent operation **system** aiming at reducing the operator's role within the fundamental judgment through the use of **artificial intelligence.** (author).

DOCUMENT NUMBER = INI18:100316

TI: Operator support and AI: Kansai applies experience through an **expert system.**
AU: Sonoda, Naoki. (Kansai Electric **Power** Co. Inc., 3-22 Nakanoshima 3- chome, Kita-ku, Osaka 530 (Japan)).
LA: English
JR: Nucl. Eng. Int. ISSN 0029-5507. CODEN: NEINB. (Jul 1987). v. 32(396) p. 32.
AB: Kansai Electric in Japan is developing **expert systems** to prevent the loss of valuable operating experience when skilled operators and engineers retire. Several prototype **expert systems** are at present being constructed for use in the operation of nuclear **power plants.** The two **power** plant operations selected as design objectives for the prototype are automatic actuation of safety injection **systems** and minor coolant leakage. (UK).

DOCUMENT NUMBER = INI18:100315

TI: MAPI's knowledge-based **system** will help to deal with abnormal conditions (in nuclear **plants).**
AU: Anon.
LA: English
JR: Nucl. Eng. Int. ISSN 0029-5507. CODEN: NEINB. (Jul 1987). v. 32(396) p. 28, 30-31.
AB: Mitsubishi Atomic **Power** Industries (MAPI) in Japan is incorporating **artificial intelligence** characteristics into a **system** for guiding nuclear **power** plant operators during fault conditions. The central feature of the **system** is the incorporation of the operator's cognitive processes as the framework for its display of information and its mode of reasoning. A small scale prototype **system** has been developed and work is underway to produce and test a full-scale prototype **system** by 1991. (UK).

DOCUMENT NUMBER = INI18:100273

TI: **Expert system** for diagnosis of nuclear **power** plant feedwater **system.**
AU: Yokota, Yutaka; Shibata, Koji; Yokota, Mitsuhisa (Toshiba Corp., Kawasaki, Kanagawa (Japan)).
LA: Japanese
JR: Toshiba Rebyu. ISSN 0372-0462. CODEN: TORBA. (May 1987). v. 42(5) p. 379-382.
AB: **Expert system** for diagnosis of nuclear **power** plant feedwater **systems** has been developed to assist maintenance engineers. This **system,** adopting the superminicomputer "TOSBAC" G 8050, and employing the developing tool TDES 2 (tool for developing **expert system** 2), has a large-scale **knowledge** base consisting of the **expert knowledge** and experience of engineers in many fields. The man-machine **system,** developed exclusively for diagnosis, improves the man-machine interface and realizes the graphic display of diagnostic process and path, stores diagnostic results and searches past reference. (author).

DOCUMENT NUMBER = INI18:100272

TI: Maintenance support **expert system** for nuclear **power** plant.
AU: Sasaki, Yukio; Tai, Ichiro (Toshiba Corp., Kawasaki, Kanagawa (Japan)); Andoh, Yasumasa.
LA: Japanese
JR: Toshiba Rebyu. ISSN 0372-0462. CODEN: TORBA. (May 1987). v. 42(5) p. 375-378.
AB: For the enhancement of reliability and safety in nuclear **power plants,** maintenance support **expert system** has been developed. When anomaly is found in a plant equipment, maintenance guidance or countermeasures is given on the **knowledge** data base of experts and their experiences and simulation, if required. This **system** can explain the reason of the inference and the experience of past occurrences, as well as the maintenance guidance. This **system** is highly useful for technical improvement of maintenance work. (author).

DOCUMENT NUMBER = INI18:095454

TI: Trouble shooting with **expert system**: a solution for sophisticated electronic equipments. **Systeme expert** d'aide au depannage: une solution pour les equipements electroniques complexes.
AU: Colling, J.M.; Franco, A. (Merlin Gerin, 38-Grenoble (France)).
LA: French
JR: Rev. Gen. Electr. ISSN 0035-3116. CODEN: RGELA. (Apr 1987). (no.4) p. 35-40.
AB: The electronic equipments maintenance is thought straight from the conception, and realized through various means such as: self testing, fault display, periodic test campaign, etc. However maintenance problems still remain that only the exploitation **knowledge** can overcome. Merlin Gerin offers hereafter an **expert system** which allows trouble shooting while the equipment is in use. The example of application chosen is one of the most complex Merlin Gerin's equipment, so the most complicated for trouble-shooting: the Digital Integrated Protection **System** (SPIN: **Systeme** de Protection Integre Numerique) of 1300 MW PWRs.

DOCUMENT NUMBER = INI18:095453

TI: **Expert systems,** automation tools: EDF experience. Les **systemes** experts, outils de l'automatisation: l'experience d'EDF.
AU: Gondran, M.; Laleuf, J.C. (Electricite de France, 75-Paris).
LA: French
JR: Rev. Gen. Electr. ISSN 0035-3116. CODEN: RGELA. (Apr 1987). (no.4) p. 18-22.
AB: The article first presents two major applications now being validated at Electricite de France: assistance in alarm management and nuclear **power** plant operation and computer-assisted reliability. The experience from these developments and implementations performed around the world permits one to predict a considerable impact of **expert system** methodology in the automation of the **systems** as regards software design, installation operation, monitoring and maintenance.

DOCUMENT NUMBER = INI18:095452

TI: Providing optimum operating information for PWRs /(in control rooms/).
AU: Aleite, W. (Kraftwerk Union A.G., Erlangen (Germany, F.R.)).
LA: English
JR: Nucl. Eng. Int. ISSN 0029-5507. CODEN: NEINB. (May 1987). v. 32(394) p. 41-42.
AB: The control rooms in Kraftwerk Union PWRs take account of human factors and make use of mimic diagrams. The 3 Convoy stations and the Brokdorf plant also have computer **based** VDU display facilities, providing a Process Information **System** (Prins) for normal operation and fault and accident conditions. (author).

DOCUMENT NUMBER = INI18:095243

TI: Remote technology: a series of articles.
AU: Meieran, H.B. (and others).
LA: English
JR: Nucl. Eng. Int. ISSN 0029-5507. CODEN: NEINB. (Apr 1987). v. 32(393) p. 34-52.
AB: A series of articles on the use of remote technology in the nuclear industry. A worldwide survey of mobile robots suitable for actual or proposed use in nuclear facilities is presented. Details are given of the first Robot Users Group, recently formed in the U.S.A. Robots with **artificial intelligence** are under development at Tokyo Electric **Power** Company. Kansai Electric **Power** and Toshiba are two companies conducting RandD to further the application of robots. Westinghouse have used the Rosa robotic arm in zero-entry steam generator tube sleeving projects, and are now looking at further developments. The 'Warrior' manipulator, by Taylot Hitec, has conducted the first continuous path MIG weld inside a magnox reactor. The articulated boom for the JET fusion device can lift 1t. at full extension. The Savannah River Laboratory is studying an advanced intelligent machine which could lead to the introduction of legged mobile and multi-tasking teleoperated work stations. Plans are being made to equip the Surveyor mobile surveillance **system** at Nine Mile Point with a number of tools. Frastar, developed by Framatome, is a vehicle which can operate inside the containment of a reactor in operation or in hazardous areas. The mobile surveillance robot, Surbot, developed by Remotec, has successfully completed five months of testing at Browns Ferry BWR. (author).

DOCUMENT NUMBER = INI18:091289

TI: Instrumentation in the **power** industry, volume 29.
AU: Anon.
LA: English
RP: CONF-8605245--.
IM: Research Triangle Park, NC (USA). Instrument Society of America. 1986. 121 p. ISBN 0-87664-965-7.
CF: (29. **power** instrumentation symposium. Cleveland, OH (USA). 19-21 May 1986.).
AB: This book presents the papers given at a conference on computerized control **systems** for **power plants.** Topics considered at the conference included human factors considerations in **power** plant control room design, advanced control concepts, nuclear plant testing and requirements of the plant instrument engineer, nuclear plant control and monitor **system** upgrades, the use of **artificial intelligence** in **power** plant control **systems**, and boiler control and data acquisition **systems** upgrades.

DOCUMENT NUMBER = INI18:090958

TI: A safety decision analysis for Saudi Arabian nuclear research facility.
AU: Abulfaraj, W.H.; Abdul-Fattah, A.F. (Dept. of Nuclear Engineering, King Abdulaziz Univ., P.O. Box 9027, Jeddah-21413).
LA: English
MS: Alternative energy sources VII. Veziroglu, T.N.
IM: New York, NY (USA). Hemisphere Publishing. 1985. p. 447-460. ISBN 0-89116-661-0.
CF: (7. Miami international conference on alternative energy sources. Miami Beach, FL (USA). 9-11 Dec 1985.)

AB: The first step in planning for introducing the nuclear energy to Saudi Arabia is to establish a nuclear research facility. Selection of a research reactor type for such a case is not an easy task. The fuzzy set decision theory is selected among different decision theories to be applied for this analysis. Four research **reactors** are selected for this study. These are: the University of Michigan Ford Nuclear Reactor (FNR), Massachusetts Institute of Technology Reactor (MITR), Georgia Institute of Technology Research Reactor (GTRR), and University of Wisconsin Nuclear Reactor (UWNR). The IFDA computer code, which **based** on the fuzzy set theory is applied here. The results show that the FNR reactor is the best alternative for the case of Saudi Arabian nuclear research facility, and MITR is the second best.

DOCUMENT NUMBER = INI18:090647

TI: Operator support **system** for nuclear **power plants.**
AU: Mori, Nobuyuki; Tai, Ichiro; Sudo, Osamu (Toshiba Corp., Kawasaki, Kanagawa (Japan)); Naito, Norio.
LA: Japanese
JR: Karyoku Genshiryoku Hatsuden. ISSN 0387-1029. CODEN: KGHTA. (Feb 1987). v. 38(2) p. 167-177.
AB: The nuclear **power** generation in Japan maintains the high capacity factor, and its proportion taken in the total generated electric **power** exceeded 1/4, thus it has become the indispensable energy source. Recently moreover, the nuclear **power plants** which are harmonious with operators and easy to operate are demanded. For realizing this, the technical development such as the heightening of operation watching performance, the adoption of automation, and the improvement of various man-machine **systems** for reducing the burden of operators has been advanced by utilizing electronic techniques. In this paper, the trend of the man-machine **systems** in nuclear **power plants,** the positioning of operation support **system,** the support in the aspects of information, action and **knowledge,** the example of a new central control board, the operation support **system** using a computer, an operation support **expert system** and the problems hereafter are described. As the development of the man-machine **system** in nuclear **power plants,** the upgrading from a present new central control board **system** PODIA through A-PODIA, in which the operational function to deal with various phenomena arising in **plants** and safety control function are added, to 1-PODIA, in which **knowledge** engineering technology is adopted, is expected. (Kako, I.).

DOCUMENT NUMBER = INI18:086235

TI: Pattern recognition and prognostic methods in parametric signal models for early failure detection. Mustererkennung und Prognoseverfahren in parametrischen Signalmodellen zu Fehlerfruehdiagnosen.
AU: Scherer, K.P.
CO: Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Datenverarbeitung in der Technik; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter; Karlsruhe Univ. (T.H.) (Germany, F.R.). Fakultaet fuer Elektrotechnik.
LA: German
RP: KFK--4197.
IM: Jun 1987. 112 p. Availability: INIS.
NO: Designation: Diss.
AB: For early failure detection in critical devices it is important to use intelligent signal processing methods in learning **systems,** so that a process evaluation can be optimized already on the signal processing level. On the one hand, pattern recognition methods are useful. On the other hand, about surveillance structures with parametric signal models there are a lot of questions not solved. In this report, a boiling detection **system** of real acoustic signals is demonstrated, how to survey such a physical process by methods of pattern recognition with enclosed parametric signal models. By introduction of prognostic models, a decision in form of a classification process can be made in advance. A surveillance is performed by the developed methods and the results are presented. (orig.).

DOCUMENT NUMBER = INI18:086130

TI: Inductive analysis of failure modes of thermohydraulic **systems** by numerical simulation.
AU: Limnios, N.; Guyonnet, J.F. (Universite de Technologie de Compiegne, 60 (France)).
LA: English
JR: Reliab. Eng. ISSN 0143-8174. CODEN: RLEND. (1987). v. 18(2) p. 141-154.
AB: The study of reliability and safety of **systems** includes three complementary approaches: physics of failure, algebraic representation and probabilistic modeling. This paper presents a numerical simulation model of failure modes in water thermohydraulic **systems** and its associated package. In the physics of failure approach, they constitute an efficient tool for the study of failure mode effects in such **systems.** It is possible to integrate this model into more general ones of risk control or **expert systems.** (author).

DOCUMENT NUMBER = INI18:083308

TI: **Artificial intelligence** applications to design validation and sneak function analysis.
AU: Stratton, R.C. (Argonne National Lab., Idaho Falls).
LA: English
MS: Space nuclear **power systems** 1984: proceedings. Volume 2. El-Genk, M.S.; Hoover, M.D. (eds.). New Mexico Univ., Albuquerque (USA). Inst. for Space Nuclear **Power** Studies; Lovelace Biomedical and Environmental Research Inst., Albuquerque, NM (USA). Inhalation Toxicology Research Inst.
IM: Malabar, FL (USA). Orbit Book Company, Inc. 1985. p. 567-576.
CF: (Symposium on space nuclear **power systems.** Albuquerque, NM (USA). 10-13 Jan 1984.)
AB: An objective of the US space reactor program is to design **systems** with high reliability and safety of control over long operating lifetimes. Argonne National Laboratory (ANL) is a participant in the National Man-Machine Integration (MMI) program for Liquid Metal Fast Breeder **Reactors** (LMFBR). A purpose of this program is to promote the development of concepts and technologies that enhance the operational safety and reliability of fast-breeder **reactors.** Much of the work is directly applicable to the space reactor program. This paper reports on one of the MMI projects being developed by ANL. The project reported pertains to an automated **system** that demonstrates the use of **artificial intelligence** (AI) for design validation (DA) and sneak function analysis (SFA). The AI **system** models the design specification and the physical design of the cooling process assigned to the Argon Cooling **System** (ACS) at Experimental Breeder Reactor II (EBR-II). The models are developed using heuristic **knowledge** and natural laws. 13 refs.

DOCUMENT NUMBER = INI18:080849

TI: **Expert systems** in the nuclear sector.
AU: Kretzschmar, J.G. (SCK/CEN).
LA: English
MS: Proceedings of the second international **expert systems** conference. Anon.
IM: Medford, NJ (USA). Learned Infromation Inc. 1986. p. 183-192. ISBN 0-904933-56-3.
CF: (2. international **expert systems** conference. London (UK). 30 Sep - 2 Oct 1986.)
AB: A general description of the nuclear fuel cycle reveals an enormous potential for **expert systems.** Over the past years R and D work has nevertheless almost exclusively been restricted to intelligent aids for the operators in the complex control rooms of nuclear **reactors.** A **systematic** review reveals a somewhat confusing picture with different approaches and many problems still to be solved. A number of tests on prototypes proved that **knowledge based systems** are at least very useful aids for operator training.

DOCUMENT NUMBER = INI18:080565

TI: Application of probabilistic risk assessment models to the safety assurance of the Oconee emergency feedwater **systems.**

AU: Weber, B.J.; Abraham, P.M.; Reed, L.A. (Duke **Power** Co., P.O. Box 33189, Charlotte, NC 28242).
LA: English
MS: Proceedings of the international ANS/ENS topical meeting on thermal reactor safety. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. X6.1- X6.6. ISBN 0-89448-121-5.
CF: (International ANS/ENS topical meeting on thermal reactor safety. San Diego, CA (USA). 2-6 Feb 1986.)
AB: The component failure data base for the Oconee PRA is updated using recent failure information from the Oconee emergency feedwater **systems**. The failure information is incorporated as new evidence into the Oconee PRA data base in the same manner as the original study. An updated **system** reliability is calculated and compared to the previous value. The results of the reanalysis show no decline in emergency feedwater **system** performance since the completion of the PRA.

DOCUMENT NUMBER = INI18:080479

TI: LWR containment loading, performance, and failure modes status versus uncertainties.
AU: Haskin, F.E.; Von Riesemann, W.A.; Williams, D.C.; Clauss, D.B. (Sandia National Labs.).
LA: English
MS: Proceedings of the international ANS/ENS topical meeting on thermal reactor safety. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. XXIII.1-XXIII.15. ISBN 0-89448-121-5.
CF: (International ANS/ENS topical meeting on thermal reactor safety. San Diego, CA (USA). 2-6 Feb 1986.)
AB: Insights regarding containment loads and containment performance have emerged from recent severe accident research. These insights permit the identification of risk-significant threats to containments for a variety of containment designs. Uncertainties in these relative likelihoods exist because of incomplete **knowledge** regarding physical processes which affect containment loading and containment performance. This paper attempts to reflect the current state of **knowledge** regarding containment loading, performance and failure modes by focusing on uncertainties which remain. Key uncertainties are discussed and qualitative perspective regarding their relative importance is provided.

DOCUMENT NUMBER = INI18:080297

TI: Application dependent nuclear data processing wims code.
AU: Kulikowska, T. (Institute of Atomic Energy).
LA: English
MS: Applications in nuclear data and reactor physics. Cullen, D.E.; Muranaka, R.; Schmidt, J.
IM: Philadelphia, PA (USA). World Scientific Pub. Co. 1986. p. 42- 88. ISBN 9971-50-132-5.
CF: (Workshop on applications in nuclear data and reactor physics. Miramare, Trieste (Italy). 17 Feb - 21 Mar 1986.)
AB: In previous sets of lectures the first steps of the nuclear data processing are explained. The resulting neutronic material properties, i.e. multigroup cross sections and associated parameters, are almost independent of the reactor type for analysis of which they are designed. The next step of nuclear data processing is considered. It consists in transformation of the multigroup library data into fewgroup constants for overall reactor calculations. The group constants obtained at this level are the coefficients of equations which have to be solved to deliver the neutron flux and **power** density distributions together with the reactivity characteristics of the reactor **system**. At this level the actual energy spectra are to be taken into account and therefore the actual lattice properties have to enter into the analysis.

DOCUMENT NUMBER = INI18:079722

TI: Modern control technology for improved nuclear reactor performance.
AU: Oakes, L.C. (Oak Ridge National Lab.).
LA: English
JR: IEEE Trans. Nucl. Sci. ISSN 0018-9499. CODEN: IETNA. (Dec 1986). v. NS-33(6) p. 1721-1722.
AB: One of the main complaints leveled at reactor control **systems** by utility spokesmen is complexity. One only has to look inside a **power** reactor control room to appreciate this viewpoint. The high reliability and versatility of modern microprocessors makes possible distributed control **systems** with only performance data and abnormal conditions being relayed to the control room. In a sense, this emulates the human-body control **system** where routine repetitive actions are handled in an involuntary manner. The significance of **expert systems** to the nuclear reactor control and safety **systems** is their ability to capture human and other expertise and make it available, upon demand, and under almost all circumstances. Thus, human problem-solving skills acquired by the learning process over a long period of time can be captured and employed with the reliability inherent in computers. This is especially important in nuclear **plants** when human operators are burdened by stress and emotional factors that have a dramatic effect on performance level.

DOCUMENT NUMBER = INI18:077242

TI: Problems in software development for nuclear robotics.
AU: Shinohara, Yoshikuni (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment).
LA: Japanese
MS: Proceedings of the third seminar on software development in nuclear energy research. Takano, Hideki; Fujii, Minoru (eds.) (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment). Japan Atomic Energy Research Inst., Tokyo.
RP: JAERI-M--86-178.
IM: Dec 1986. 253 p. p. 192-200. Availability: INIS.
CF: (3. seminar on software development in nuclear energy research. Tokai, Ibaraki (Japan). 10-11 Sep 1986.)
AB: Major technical problems in developing softwares for intelligent robots for future nuclear applications are explained briefly. In order that a robot can perform various kinds of complex works, it must be equipped with a high level of **artificial intelligence** which includes sensing functions such as visiual, auditory, tactile, proximity sensing, cognitive functions such as recognition of objects and understanding of working environment, decision-making functions such as work planning and control functions such as manipulator and locomotion controls. A large amount of various kinds of signals and informations must be processed with a high speed for an integrated control of these functions. It will be desirable that the computer program for controlling a robot which must run in a real-time will have a functionally hierarchical and distributed structure from the view point of software development. Parallel processing will be required from the view point of computation time. (author).

DOCUMENT NUMBER = INI18:077224

TI: The CLASS **system** and the scheduling problem. **System** CLASS a problem rozvrhovania.
AU: Zarnovican, V. (Vyskumny Ustav Jadrovych Elektrarni, Jaslovske Bohunice (Czechoslovakia)).
LA: Slovak
MS: Automated **systems** of nuclear **power** plant control. Conference proceedings. Automatizovane **systemy** riadenia jadrovych elektrarni. Zbornik referatov z konferencie. Vyskumny Ustav Jadrovych Elektrarni, Jaslovske Bohunice (Czechoslovakia).
RP: INIS-mf--10958.
IM: 1986. 270 p. p. 111-117. Availability: INIS.
CF: (Conference on automated **systems** of nuclear **power** plant control. Tale (Czechoslovakia). 15-17 Apr 1986.)

AB: The need to process large quantities of data on the machine equipment of nuclear **power plants** and the high level of occurrence of recurrent technological procedures applied in maintenance will create the preconditions for the introduction of automated processing of this data and the subsequent automated control of units of centralized maintenance in nuclear **power plants.** For the said purposes the Research Institute for Nuclear **Power Plants** at Jaslovske Bohunice has selected activities suitable for automated processing and has experimentally applied to these activities the CLASS decision-making **system** operating with the CPM method. It is stated that the automated **system** for the operative control of repairs of nuclear **power plants** requires detailed information on conditions from which ensue the inter-relations between repairs. The choice of the most suitable sequence of operations should be left to the program which uses elements of **artificial intelligence.** Some aspects are discussed of the mathematical model of such a decision-making **system** in CLASS-AI operative control. (Z.M.).

DOCUMENT NUMBER = INI18:076922

TI: Design of **expert system** for diagnostics of operating states of WWER- 440 reactor. Koncepce expertniho **systemu** pro diagnostiku provoznich stavu reaktoru VVER 440.
AU: Dach, K.; Houska, Z.; Rygl, J.; Vasa, I. (Ustav Jaderneho Vyzkumu CSKAE, Rez (Czechoslovakia)).
LA: Czech
MS: Automated **systems** of nuclear **power** plant control. Conference proceedings. Automatizovane **systemy** riadenia jadrovych elektrarni. Zbornik referatov z konferencie. Vyskumny Ustav Jadrovych Elektrarni, Jaslovske Bohunice (Czechoslovakia).
RP: INIS-mf--10958.
IM: 1986. 270 p. p. 281-285. Availability: INIS.
CF: (Conference on automated **systems** of nuclear **power** plant control. Tale (Czechoslovakia). 15-17 Apr 1986.)
AB: A brief description is presented of the development of an **expert system of** diagnostics of operating states of the WWER-440 reactor. The **system** is designed as a hybrid **system** which will allow to evaluate the states of reactor internals both from values measured using "conventional" software and by means of logical programs using **expert knowledge** bases. The **system** should have a modular structure which will allow to continuously widen **knowledge** of the **expert system** and thereby also possibilities of operator consulting. In the first stage software was developed which evaluates the correctness of the function of sensors of neutron flux and the respective measurement chain. The software was tested during the start-up of the first units of the Dukovany nuclear **power** plant and the V-2 nuclear **power** plant at Jaslovske Bohunice. Also constructed was the so-called **knowledge** base for the main closing valve of the primary circuit of the WWER-440 reactor. **Knowledge** bases are being prepared for the evaluation of operating states of the reactor core, the diagnostics of steam generators, and for the evaluation of the safety aspects of nuclear **power plants.** A brief description is presented of the designed hardware for the **expert system** for the Dukovany nuclear **power** plant which is **based** on the prepared IGS 4710 graphic station. (Z.M.).

DOCUMENT NUMBER = INI18:076874

TI: Development of several data bases related to reactor safety research including probabilistic safety assessment and incident analysis at JAERI.
AU: Kobayashi, Kensuke; Oikawa, Tetsukuni; Watanabe, Norio; Izumi, Fumio; Higuchi, Suminori (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment).
LA: Japanese
MS: Proceedings of the third seminar on software development in nuclear energy research. Takano, Hideki; Fujii, Minoru (eds.) (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment). Japan Atomic Energy Research Inst., Tokyo.
RP: JAERI-M--86-178.
IM: Dec 1986. 253 p. p. 176-191. Availability: INIS.
CF: (3. seminar on software development in nuclear energy research. Tokai, Ibaraki (Japan). 10-11 Sep 1986.)

AB: Presented are several databases developed at JAERI for reactor safety research including probabilistic safety assessment and incident analysis. First described are the recent developments of the databases such as 1) the component failure rate database, 2) the OECD/NEA/IRS information retrieval **system**, 3) the nuclear **power** plant database and so on. Then several issues are discussed referring mostly to the operation of the database (data input and transcoding) and to the retrieval and utilization of the information. Finally, emphasis is given to the increasing role which **artifitial intelligence** techniques such as natural language treatment and **expert systems** may play in improving the future capabilities of the databases. (author).

DOCUMENT NUMBER = INI18:076852

TI: Diagnosis and maintenance support **expert system** for reactor instrumentation.
AU: Tai, Ichiro; Fujii, Makoto; Itoh, Toshiaki (Toshiba Corp., Kawasaki, Kanagawa (Japan)).
LA: Japanese
JR: Toshiba Rebyu. ISSN 0372-0462. CODEN: TORBA. (Sep 1986). v. 41(9) p. 803-806.
AB: Diagnosis and Maintenance Support **Expert System** for reactor instrumentation has been developed to assist plant engineers in nuclear **power plants.** This **system** comprises two subsystems : the diagnosis subsystem and the prognosis subsystem. The diagnosis subsystem is a knowledge-based **expert system** which supports the plant engineers to find the cause of abnormality and to take countermeasures. Hierarchical rules are adopted for **knowledge** representation, and special support functions are prepared, such as flexible inference function and measured data processing function. The prognosis subsystem performs statistical processing of various data and information to estimate the life time of neutron detectors. This **system** can highly mitigate diagnosis and maintenance work for reactor instrumentation. (author).

DOCUMENT NUMBER = INI18:076784

TI: **Expert systems** for nuclear **power** plant diagnostics. Expertni **systemy** pro diagnostiku jadernych elektraren.
AU: Dach, K. (Ustav Jaderneho Vyzkumu CSKAE, Rez (Czechoslovakia)).
LA: Czech
MS: Automated **systems** of nuclear **power** plant control. Conference proceedings. Automatizovane **systemy** riadenia jadrovych elektrarni. Zbornik referatov z konferencie. Vyskumny Ustav Jadrovych Elektrarni, Jaslovske Bohunice (Czechoslovakia).
RP: INIS-mf--10958.
IM: 1986. 270 p. p. 271-280. Availability: INIS.
CF: (Conference on automated **systems** of nuclear **power** plant control. Tale (Czechoslovakia). 15-17 Apr 1986.)
AB: Some aspects are discussed of the development and deployment of **expert** diagnostic **systems** in nuclear **power** in different parts of the world. A brief description is presented of the concept of the **expert** diagnostic **system** prepared for the French reactor Phenix. The example of the diagnostics of vibrations of reactor internals is used to document work in the field of the development of **expert systems** for nuclear **power** in Czechoslovakia. (Z.M.).

DOCUMENT NUMBER = INI18:074319

TI: Developing a **knowledge** base for the management of severe accidents.
AU: Nelson, W.R.; Jenkins, J.P. (EG and G Idaho, Inc., P.O. Box 1625, Idaho Falls, ID 83415).
LA: English
MS: Proceedings of the international topical meeting on advances in human factors in nuclear **power systems.** Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 514- 519. ISBN 0-89448-125-8.
CF: (International topical meeting on advances in human factors in nuclear **power systems.** Knoxville, TN (USA). 21-24 Apr 1986.)

AB: Prior to the accident at Three Mile Island, little attention was given to the development of procedures for the management of severe accidents, that is, accidents in which the reactor core is damaged. Since TMI, however, significant effort has been devoted to developing strategies for severe accident management. At the same time, the potential application of **artificial intelligence** techniques, particularly **expert systems,** to complex decision-making tasks such as accident diagnosis and response has received considerable attention. The need to develop strategies for accident management suggests that a computerized **knowledge** base such as used by an **expert system** could be developed to collect and organize **knowledge** for severe accident management. This paper suggests a general method which could be used to develop such a **knowledge** base, and how it could be used to enhance accident management capabilities.

DOCUMENT NUMBER = INI18:074318

TI: Application of **artificial intelligence** to improve plant availability.
AU: Frank, M.V.; Epstein, S.A. (Management Analysis Co., 12671 High Bluff Drive, San Diego, CA 92130).
LA: English
MS: Intelligent simulation environments. Luker, P.A.; Adelsberger, H.H.
IM: San Diego, CA (USA). Society for Computer Simulation. 1986. p. 92- 100.
CF: (Society for Computer Simulation (SCS) multiconference. San Diego, CA (USA). 23-25 Jan 1986.)
AB: Using **artificial intelligence** software techniques, Management Analysis Company (MAC) has developed two complementary software packages that together provide a workstation environment for maintenance and operations personnel to dramatically reduce inadvertent reactor trips. They are called the Reactor Trip Simulation Environment (RTSE) and the Key Component Generation Environment (KCGE). They are not proposed or being designed. They are in use today. The **plants** component and process interdependencies are modeled within RTSE - using modeling practices and notations familiar to engineers which accelerated the acceptance of the software. KCGE provides the groups of key components that would cause a reactor trip.

DOCUMENT NUMBER = INI18:074288

TI: Modeling cognitive behavior in nuclear **power plants:** An overview of contributing theoretical traditions.
AU: Woods, D.D.; Roth, E.M. (Westinghouse Research and Development Center, 1310 Beulah Road, Pittsburgh, PA 15235).
LA: English
MS: Proceedings of the international topical meeting on advances in human factors in nuclear **power systems.** Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 12- 20. ISBN 0-89448-125-8.
CF: (International topical meeting on advances in human factors in nuclear **power systems.** Knoxville, TN (USA). 21-24 Apr 1986.)
AB: This paper reviews the major theoretical literatures that are relevant to modeling human cognitive activities important to nuclear **power** plant safety. The traditions considered include control theory, communication theory, statistical decision theory, information processing models and symbolic processing models. The review reveals a gradual convergence towards models that incorporate elements from multiple traditions. Models from the control theory tradition have gradually evolved to include rich **knowledge** representations borrowed from the symbolic processing work. At the same time theorists in the symbolic processing tradition are beginning to grapple with some of the critical issues involved in modeling complex real world domain.

DOCUMENT NUMBER = INI18:074286

TI: A human factors data bank for French nuclear **power plants.**
AU: Villemeur, A.; Mosneron-Dupin, F.; Bouissou, M.; Meslin, T. (Direction des Etudes et Recherches, 1, av. General de Gaulle BP 408, 92141 Clamart Cedex).

LA: English
MS: Proceedings of the international topical meeting on advances in human factors in nuclear **power systems**. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 368- 373. ISBN 0-89448-125-8.
CF: (International topical meeting on advances in human factors in nuclear **power systems**. Knoxville, TN (USA). 21-24 Apr 1986.)
AB: CONFUCIUS is a computerized data bank developed by Electricite de France to study human factors in nuclear **power plants**. A detailed and homogeneous grouping of described operation and maintenance errors as well as of performance times is possible with CONFUCIUS. It also incorporates a selection of statistical treatment softwares. Readily usable and modifiable, the **system** can easily evolve. It allows a wide range of applications (safety analysis, event analysis, training, human factors engineering, probabilistic analysis). Data derived from the analysis of significant events reported in **power plants** and from the analysis of simulator tests are used as inputs into this data bank.

DOCUMENT NUMBER = INI18:073986

TI: A prototype **expert system** for fault analysis, alarm handling and operator advising in a PWR nuclear **power** plant.
AU: Gondran, M.; Ancelin, J.; Hery, J.F.; Laleuf, J.C.; Legaud, P. (Electricite de France, 1, avenue du General de Gaulle - 92140 Clamart).
LA: English
MS: Proceedings of the international topical meeting on advances in human factors in nuclear **power systems**. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 60- 64. ISBN 0-89448-125-8.
CF: (International topical meeting on advances in human factors in nuclear **power systems**. Knoxville, TN (USA). 21-24 Apr 1986.)
AB: This paper successively describes the methodological advantages of the expert-systems approach and then two major applications of it that are under development in Electricite de France: assistance in alarms and operational management, and computer-assisted reliability. Topics considered include man-machine **systems**, human factors engineering, reactor operation, **artificial intelligence**, and decision making.

DOCUMENT NUMBER = INI18:073983

TI: Application of **artificial intelligence** techniques to TRR operation. (TRR - Thai Research Reactor, Bangkok, Thailand.)
AU: Ho, L.; Tseng, C.; Chang, S. (Institute of Nuclear Energy Research, P.O. Box 3-3, Lungtan 32500).
LA: English
MS: Proceedings of the international topical meeting on advances in human factors in nuclear **power systems**. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 520- 525. ISBN 0-89448-125-8.
CF: (International topical meeting on advances in human factors in nuclear **power systems**. Knoxville, TN (USA). 21-24 Apr 1986.)
AB: It has been over ten years since TRR had its initial critical. To collect the experiences of shift operators and technique staffs and transfer these experts' **knowledge** to a computer and build an **expert system** is a typical application of **artificial intelligence** techniques to nuclear business. The **system** can provide the correct information of TRR operation for shift personnel, new staffs and other technical people.

DOCUMENT NUMBER = INI18:073981

TI: Knowledge-based operator guidance **system** for Japanese PWRs.
AU: Fujita, Y.; Ito, K.; Kawanago, S.; Tani, M.; Murata, R. (Mitsubishi Atomic **Power** Industries, Inc., 4-1, Shibakoen 2-chome, Minato-ku, Tokyo 105).

LA: English
MS: Proceedings of the international topical meeting on advances in human factors in nuclear **power systems**. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 505- 513. ISBN 0-89448-125-8.
CF: (International topical meeting on advances in human factors in nuclear **power systems**. Knoxville, TN (USA). 21-24 Apr 1986.)
AB: A knowledge-based operator support **system** for nuclear **power** plant operation is under development. The main theme of the study is the incorporation of operator's cognitive structure as the framework of the **knowledge** representation and inference control mechanisms. **Based** upon information collected from interviews, and experiments using a real-time simulator, an operator's model related to diagnostic tasks was developed. A knowledge-based **system** incorporating the proposed model demonstrated highly efficient problem solving capabilities and the dynamic fitness to operator's perceptual feeling, thereby suggesting the potential importance and practical benefit of such a study.

DOCUMENT NUMBER = INI18:073958

TI: Experimental evaluation of an operator decision aid for BWR **plants**.
AU: Fukutomi, S.; Yoshimura, S.; Takizawa, Y.; Hattori, Y.; Mori, N. (Nippon Atomic Industry Group Co. Ltd., 4-1 Ukishima-cho, Kawasaki- ku, Kawasaki, 210).
LA: English
MS: Proceedings of the international topical meeting on advances in human factors in nuclear **power systems**. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 358- 367. ISBN 0-89448-125-8.
CF: (International topical meeting on advances in human factors in nuclear **power systems**. Knoxville, TN (USA). 21-24 Apr 1986.)
AB: An experiment was carried out to evaluate the effectiveness of a computerized operator decision aid (ODA) for BWR (boiling water reactor) **power plants**. The ODA was developed to enhance operational safety, reliability and availability by supporting the operators in coping with adverse plant situations. The disturbance analysis and the post trip operational guidance of the ODA are the main support functions under plant transient conditions. To assess the effects of the ODA on operator performance, a realistic experimental control room (equipped with color cathode-ray tubes, control switches, annunicators and paging devices) and a full scope BWR plant simulator were developed.

DOCUMENT NUMBER = INI18:073955

TI: An application of **Expert System** technology to nuclear **power** plant operations.
AU: Reierson, J.D.; Lay, R.K. (The MITRE Corp., McLean, VA 22102).
LA: English
MS: Proceedings of Compcon Fall "84". Anon.
IM: Piscataway, NJ (USA). IEEE Service Center. 1984. p. 11-15. ISBN 0-8186-0546-4.
CF: (COMPCON fall '84. Arlington, VA (USA). 16-20 Sep 1984.)
AB: An **Expert System** is a computer-based **system** that emulates a human **expert** in solving problems in a specific subject area. The **knowledge** of the human **expert** is contained in non-numeric If-Then decision rules. This paper describes an internally funded MITRE project which assessed the current state of **Expert System** technology by constructing a small **Expert System** program to prescribe sequences of control rod withdrawals during start-up and **power** shaping maneuvers of a boiling water reactor in a nuclear **power** plant. The **knowledge** of the station nuclear engineer is contained in seventeen If-Then rules which compare the desired setting of control rod groups with the settings of neighboring rods. The **Expert System** can prescribe a sequence of moves to the target pattern or can check a sequence input by the engineer. The paper concludes with a list of problems and benefits associated with using current **Expert System** technology.

DOCUMENT NUMBER = INI18:073937

TI: Proceedings of the international topical meeting on advances in human factors in nuclear **power systems**.
AU: Anon.
LA: English
RP: CONF-860415--.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. 525 p. ISBN 0-89448-125-8.
CF: (International topical meeting on advances in human factors in nuclear **power systems**. Knoxville, TN (USA). 21-24 Apr 1986.)
AB: This book presents the papers given at a conference on the human factors engineering of nuclear **power plants**. Topics considered at the conference included human modeling, **artificial intelligence, expert systems,** robotics and teleoperations, organizational issues, innovative applications, testing and evaluation, training **systems** technology, a modeling framework for crew decisions during reactor accident sequences, intelligent operator support **systems,** control algorithms for robot navigation, and personnel management.

DOCUMENT NUMBER = INI18:073288

TI: **Artificial intelligence** enhancements to safety parameter display **systems.**
AU: Hajek, B.K.; Hashemi, S.; Sharma, D.; Chandrasekaran, B.; Miller, D.W. (The Ohio State Univ., Nuclear Engineering Program, Robinson Lab., 206 West Eighteenth Avenue, Columbus, OH 43201).
LA: English
MS: Proceedings of the international topical meeting on advances in human factors in nuclear **power systems**. Anon.
IM: La Grange Park, IL (USA). American Nuclear Society. 1986. p. 497- 504. ISBN 0-89448-125-8.
CF: (International topical meeting on advances in human factors in nuclear **power systems.** Knoxville, TN (USA). 21-24 Apr 1986.)
AB: Two prototype **knowledge based systems** have been developed at The Ohio State University to be the basis of an operator aid that can be attached to an existing nuclear **power** plant Safety Parameter Display **System.** The first **system** uses improved sensor validation techniques to provide input to a fault diagnosis process. The second **system** would use the diagnostic **system** output to synthesize corrective procedures to aid the control room licensed operator in plant recovery.

DOCUMENT NUMBER = INI18:070087

TI: Hardware and software for the diagnostics of Czechoslovak nuclear units with WWER-440 and WWER-1000 **reactors.** Technicke prostredky a programove vybaveni pro diagnostiku cs. jadernych bloku VVER 440 a VVER 1000.
AU: Drab, F.; Mlady, Z. (Skoda, Plzen (Czechoslovakia). Zavod Vystavba Jadernych Elektraren).
LA: Czech
MS: Software for diagnostic **systems** of nuclear **power plants.** Programove vybaveni diagnostickych **systemu** jadernych elektraren. Skoda, Plzen (Czechoslovakia). Zavodni Pobocka Ceske Vedeckotechnicke Spolecnosti.
RP: INIS-mf--10947.
IM: 1986. 106 p. p. 31-48. Availability: INIS.
CF: (Seminar on software for diagnostic **systems** of nuclear **power plants.** Plzen (Czechoslovakia). 3-4 Jun 1986.)
AB: The hardware of in-service diagnostics of Czechoslovak nuclear units with WWER-440 **reactors** at present consist of a configuration of diagnostic **systems** for vibration control, control of loose parts and leakage control manufactured by the West German firm KWU, completed with a Czechoslovak computer type evaluation **system.** This **system** does not, however, have continuity with major Soviet made measurement **systems** and cannot therefore solve the whole range of diagnostic and **expert** problems. It will therefore be necessary to equip the WWER-440 units with subsystems for control **system** self- diagnostics, the reactor protection **system,** the diagnostics of the reactor core condition, etc. On the other hand, the concept of in- service

diagnostics of the primary circuit of WWER-1000 units for the Temelin nuclear **power** plant according to the Soviet design is an integral component of the whole automated control **system.** The demands are defined placed on completing the structure of the **automated system** of secondary circuit control with technological equipment of Czechoslovak provenance. Briefly described are areas on which attention should be centred in case components of the primary circuit of WWER-1000 **reactors** are manufactured in Czechoslovakia. (Z.M.).

DOCUMENT NUMBER = INI18:069998

TI: State of the art and prospects in application of robots at NPPs. Sostoyanie i perspektivy primeneiya robototekhnicheskikh ustrojstv na AEhS.
AU: Gavrilov, S.D.
LA: Russian
JR: At. Tekh. Rubezhom. ISSN 0320-9326. CODEN: ATRUA. (Aug 1986). (no.8) p. 3-11.
AB: Main aspects of development and application of robotized technologies and robots at NPPs are considered: robotization programs, modern state of art of maintenance automatization of main equipment, perspectives of robot application and their introduction at NPPs. Feasibility of robots with **artificial intelligence** and examples of local accident elimination by dint of robots nuclear **power** units are analyzed.

DOCUMENT NUMBER = INI18:069973

TI: **Knowledge** base for **power** plant operation and its application to operation guide.
AU: Doi, Atsushi; Sakaguchi, Toshiaki (Mitsubishi Electric Corp., Amagasaki, Hyogo (Japan). Central Research Lab.).
LA: Japanese
JR: Denki Gakkai Ronbunshi, B. ISSN 0385-4213. CODEN: DGRBB. (Jan 1986). v. 106(1) p. 25-30.
AB: The present study is aimed at constructing a **knowledge** base for supervisory control operation in **power plants** and developing an operation guidance by using it. Examination is made to provide dignosis procedures on the basis of an existing alarm **system.** An operation guidance procedure for diagnosis is proposed which is to be followed when several alarms are sounded simultaneously in a **power** plant, and application of the procedure to an existing plant is examined. The operation manual for the plant includes 75 description items for six alarms. It is shown that the number of items related to these alarms can be reduced by 70 % by rearranging them according to the procedure. Another investigation is conducted to provide an operation manual for diagnosis to be used when one alarm is sounded in a plant. The quality of the manual developed is on nearly the same level with that for the existing plant examined. When a **knowledge** base is to be constructed from an existing operation manual, the processing operation generally requires a certain level of linguistic comprehension ability, such as for judgment of synonyms. It is demonstrated that the procedure proposed here is able to develop a high-quality **knowledge** base with standardized terminology. The procedure can also serve to construct operation manuals for **plants** in other industrial fields. (Nogami, K.).

DOCUMENT NUMBER = INI18:069972

TI: Industrial disasters - the **expert systems** solution.
AU: Sachs, P.
LA: English
JR: Control Syst. ISSN 0266-2493. CODEN: CSYEM. (Dec 1986 - Jan 1987). v. 4(1) p. 42-45.
AB: Six mistakes by the operators led to the accident at the Cherobyl nuclear reactor. These have been studied. It is suggested that an **expert systems** approach could prevent similar accidents. The **expert system** is a new approach to software programming where programs are required to perform intelligent analyses of complex situations. It separates the **knowledge** of a problem from the procedural code that performs the decision. An **expert system** will evaluate data and indicate a priority on alarms in real time. Now software **systems** can detect the cause of a problem in a

process plant and present their findings to the operators in the control room. This should enable operators to make the correct decisions as they will know which underlying process faults are causing the alarms to operate. The Chernobyl post-mortem meeting made 13 proposals for improving safety. Two in particular are noted as relevant to **expert** advice **systems;** international collaboration on man-reactor relationships and a conference to explore the balance of automation and human action to minimise operating errors. (U.K.).

DOCUMENT NUMBER = INI18:069952

TI: Knowledge-based framework for procedure synthesis and its application to the emergency response in a nuclear **power** plant.
AU: Sharma, D.D.
CO: Ohio State Univ., Columbus (USA).
LA: English
IM: 1986. 374 p. Availability: University Microfilms Order No. 86- 25,286.
NO: Designation: Thesis (Ph. D.).
AB: In this dissertation a nuclear **power** plant operator is viewed as a **knowledge based** problem solver. It is shown that, in responding to an abnormal situation, an operator typically solves several problems, for example, plant status monitoring, diagnosis, sensor data validation, consequence prediction, and procedure synthesis. It is proposed that, in order to respond to unexpected situations and handle procedure failures the capability to synthesize and modify procedures dynamically or in runtime is required. A **knowledge based** framework for dynamically synthesizing procedures (DPS), a **knowledge** representation language to apply DPS framework for real problems (DPSRL), and a framework for emergency procedure synthesis (FEPS) for nuclear **power plants based** on DPS are developed. The DPS framework views the task of synthesis as a process of selecting predefined procedures to match the needs of the dynamically changing plant conditions. The DPSRL language provides **knowledge** organization and representation primitives required to support the DPS framework. Specifically, it provides mechanisms to build various plant libraries and procedures to access information from them. The capabilities and the use of DPS, DPSRL, and FEPS are demonstrated by developing an experimental **expert system** for a typical boiling water reactor and analyzing its performance for various selected abnormal incidents.

DOCUMENT NUMBER = INI18:069907

TI: Development of a microcomputer **based** nuclear **power** plant maintenance **expert system.**
AU: Ashley, G.R.; Cruz, P.A. (Impell Corp., Bannockburn, IL).
LA: English
MS: Engineering applications of microcomputers. Jones, R.F.; Ahluwalia, K.S.; Rosenberg, R.S.
IM: New York, NY (USA). American Society of Mechanical Engineers. 1986. p. 51-54.
CF: (ASME pressure vessel and piping conference and exhibit. Chicago, IL (USA). 20-24 Jul 1986.)
AB: The advanced capabilities of today's microcomputers have paved the way for the development of extremely portable **expert systems.** These micro-based **expert systems** are proving particularly beneficial to nuclear **power** plant operation and maintenance. This paper details the development of Impell Corporation's PbSHIELDING **system,** a micro-based **expert system** designed for evaluating the use of temporary lead shielding in nuclear **power plants** during maintenance activities. The paper describes the implementation of the major components of an **expert system** as well as the primary advantages and disadvantages of this type of **system.**

DOCUMENT NUMBER = INI18:069906

TI: Interpretation of pipe networks by magnetic sensing.
AU: Whittaker, W.L.; Motazed, B. (Dept. of Civil Engineering, Carnegie- Mellon Univ., Pittsburgh, PA 15213).
LA: English

MS: Applications of **artificial intelligence** in engineering problems. Sriram, D.; Adey, R.
IM: New York, NY (USA). Springer-Verlag New York Inc. 1986. p. 131- 140. ISBN 0387-16349-2.
CF: (1. international conference on applications of AI to engineering problems. Southhampton (UK). 15-18 Apr 1986.)
AB: The imaging and sizing of buried pipes in soil or steel rods in concrete has wide utility. The accurate detection of embedded reinforcement is useful to enforce the correct placement of steel in new concrete, and is even more important in assessing as-built structures. Rebar mapping is essential in post-construction operations such as coring and anchor emplacement in nuclear **power plants.** Pipe mapping is invaluable to the excavation and repair of utility distribution **systems.** This paper identifies some limitations of conventional manual methods of detecting and sizing buried ferrous cylinders. The body of the paper presents an alternate, automated methodology that rationalizes and enhances the mapping process. The methodology integrates automated scanning, image processing and image understanding techniques to characterize buried steel from sensor data.

DOCUMENT NUMBER = INI18:069857

TI: Use of **expert systems** for technical diagnostics in nuclear **power** engineering. Expertni **systemy** a jejich vyuziti pro technickou diagnostiku v jaderne energetice.
AU: Marik, V.; Zdrahal, Z. (Ceske Vysoke Uceni Technicke, Prague (Czechoslovakia). Fakulta Elektrotechnicka).
LA: Czech
MS: Software for diagnostic **systems** of nuclear **power plants.** Programove vybaveni diagnostickych **systemu** jadernych elektraren. Skoda, Plzen (Czechoslovakia). Zavodni Pobocka Ceske Vedeckotechnicke Spolecnosti.
RP: INIS-mf--10947.
IM: 1986. 106 p. p. 112. Availability: INIS.
NO: Published in summary form only.
CF: (Seminar on software for diagnostic **systems** of nuclear **power plants.** Plzen (Czechoslovakia). 3-4 Jun 1986.)

DOCUMENT NUMBER = INI18:062275

TI: **Artificial intelligence** executive summary.
AU: Wamsley, S.J.; Purvis, E.E. III (Westinghouse Advanced Energy **Systems** Division).
LA: English
MS: Nuclear **power** high technology colloquium: proceedings. Argonne National Lab., IL (USA).
RP: ANL/IEP--85-10.
IM: 10 Dec 1984. p. 211-220. Availability: INIS. Available from NTIS, PC A13/MF A01; 1 as DE86012079.
CF: (Nuclear **power** high technology colloquium. Williamsburg, VA (USA). 10-12 Dec 1984.)
AB: **Artificial intelligence** (AI) is a high technology field that can be used to provide problem solving diagnosis, guidance and for support resolution of problems. It is not a stand alone discipline, but can also be applied to develop data bases for retention of the expertise that is required for its own **knowledge** base. This provides a way to retain **knowledge** that otherwise may be lost. **Artificial intelligence** Methodology can provide an automated construction management decision support **system,** thereby restoring the manager's emphasis to project management.

DOCUMENT NUMBER = INI18:061941

TI: Nuclear powerplant functions: control and monitoring, testing and equipment qualification, manufacturing.
AU: Nolan, J. (Westinghouse Hanford Co., Richland, WA).
LA: English
MS: Nuclear **power** high technology colloquium: proceedings. Argonne National Lab., IL (USA).
RP: ANL/IEP--85-10.

IM: 10 Dec 1984. p. 51-80. Availability: INIS. Available from NTIS, PC A13/MF A01; 1 as DE86012079.
CF: (Nuclear **power** high technology colloquium. Williamsburg, VA (USA). 10-12 Dec 1984.)
AB: The author presents an overview of 1) control and monitoring, 2) testing and equipment qualification, and 3) manufacturing by discussing a number of individual topics. The liquid metal reactor FFTF is used as an example in the various topics: sensors, radiation monitors, control rooms, engineered safety **systems**, hydraulic **systems**, reactor shutdown **systems**, failed element detection, man- machine interfaces, **artificial intelligence**, data acquisition **systems**, disciplined testing programs, testing of fuel **systems**, manufacturing of breeder fuel, automated fabrication lines, robotics for materials handling, and nuclear material accountability.

DOCUMENT NUMBER = INI18:061617

TI: RiTSE/KCGE: Application of **artificial intelligence** to improve plant availability.
AU: Frank, M.V.; Epstein, S.A. (Management Analysis Co., San Diego, CA (USA)).
LA: English
MS: Proceedings of second international topical meeting on nuclear **power** plant thermal hydraulics and operations. Wakabayashi, Jiro (Kyoto Univ., Uji (Japan). Inst. of Atomic Energy); Nariai, Hideki (eds.).
IM: Tokyo (Japan). Atomic Energy Soc. of Japan. 1986. 1191 p. p. 9/ 38-9/44.
CF: (2. international topical meeting on nuclear **power** plant thermal hydraulics and operations. Tokyo (Japan). 15-17 Apr 1986.)
AB: Maintenance and other personnel related errors are a major cause of reactor trips in some U.S. **plants.** Software called RiTSE and KCGE, that can operate on a mini-computer work station, has been developed to dramatically reduce trips caused by clearance coordination problems. Such problems arise because of the enormous number of components that interrelate in ways that cannot always be anticipated during plant operation. Concurrent maintenance, test, surveillance and other operations activities contribute to the unanticipated interactions. RiTSE allows the appropriate plant personnel (i.e., those responsible for authorizing work during start- up and operation) to predict if their next authorizations would cause a reactor trip. With KCGE, RiTSE can warn these personnel of a condition of significantly reduced margin to trip. RiTSE also aids post-trip cause analysis by pointing out those components that could have contributed to the trip. It is believable that use of RiTSE is a quicker and more reliable way to authorize work during operation than reliance on memory and drawings. The software provides maximum benefit when used in conjunction with a configuration management program that assures accurate plant presentation and accurate updating of component status during operation. Application of RiTSE and KCGE to a client's PWR demonstrated part of their potential benefits. (Nogami, K.).

DOCUMENT NUMBER = INI18:061616

TI: The Westinghouse Emergency Response Guidelines considered as an **expert system.**
AU: Stella, M.E. (Delian Corp., PA (USA)); Julian, H.V.
LA: English
MS: Proceedings of second international topical meeting on nuclear **power** plant thermal hydraulics and operations. Wakabayashi, Jiro (Kyoto Univ., Uji (Japan). Inst. of Atomic Energy); Nariai, Hideki (eds.).
IM: Tokyo (Japan). Atomic Energy Soc. of Japan. 1986. 1191 p. p. 13/ 30-13/37.
CF: (2. international topical meeting on nuclear **power** plant thermal hydraulics and operations. Tokyo (Japan). 15-17 Apr 1986.)
AB: The Westinghouse Emergency Response Guidelines combined and integrated an extensive body of **knowledge** on nuclear reactor emergency operations. The guidelines resulted from a **systematic** development process supported by all major elements of the commercial nuclear industry. The results of the process and the features of the process itself were analogous to the characteristics that are used to define an **Expert System.** The similarities of the Emergency Response Guidelines to **Expert Systems** can be used to draw conclusions about the possible future development of both reactor emergency operations guidance and on the general application of **Expert Systems** to similar large-scale commercial or industrial problems. (author).

DOCUMENT NUMBER = INI18:061611

TI: **Knowledge** engineering and its applications to nuclear technology.
AU: Motoda, Hiroshi (Hitachi Ltd., Tokyo (Japan)); Ogino, Takamichi; Sekimizu, Koichi; Shinohara, Yoshikuni; Kitamura, Masaharu.
LA: Japanese
JR: Nippon Genshiryoku Gakkai-Shi. ISSN 0004-7120. CODEN: NGEGA. (Sep 1986). v. 28(9) p. 794-805.
AB: The application of **knowledge** engineering to nuclear technology is rapidly advancing over wide ranges. For the problems which have been solved so far by empirical laws and know-hows, the **knowledge** engineering offers new approach, and its potential is very large. In this report, this technology is explained about the various actual examples of application, and its future perspective is given. The reports of the research on the application of **knowledge** engineering presented to the Atomic Energy Society of Japan have increased very much. The fields of the object of application were initially abnormality diagnosis and operation guidance, but recently, alarm dealing, the planning of fuel exchange, the layout of machinery and equipment, the routing of pipings, the forecast of machinery and equipment life, maintenance planning and so on have been included. As the concrete examples of application, the description and analysis of the networks related to phenomena, the diagnosis of nuclear **power plants**, qualitative inference and the acquirement of the **knowledge** for diagnosis, the planning of fuel movement and others are described. The expression of **knowledge** and the mechanism of inference, and the problems of **knowledge** engineering hereafter are discussed. (Kako, I.).

DOCUMENT NUMBER = INI18:061473

TI: Why use microcomputers for nuclear engineering applications?.
AU: Graves, H.W. Jr.
LA: English
RP: CONF-861102--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1986). v. 53 p. 267-268.
CF: (American Nuclear Society and Atomic Industrial Forum joint meeting. Washington, DC (USA). 16-21 Nov 1986.)
AB: The rapid advancement of microprocessor technology over the last five years, coupled with the commercialization of microcomputers, has made these devices an attractive alternative to the use of mainframe computers for many types of engineering analysis. These machines offer advantages in three areas: computational cost, engineering efficiency, and utilization of recent advances in information technology.

DOCUMENT NUMBER = INI18:060571

TI: Enhanced inspection of NPPs using **expert systems** as a new approach.
AU: Hoernes, P.E.; Salame-Alfie, A.; Yeater, M.L.
LA: English
RP: CONF-861102--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1986). v. 53 p. 275-277.
CF: (American Nuclear Society and Atomic Industrial Forum joint meeting. Washington, DC (USA). 16-21 Nov 1986.)
AB: The planning, execution, and post-analysis of a reactor inspection require the inspector to analyze a tremendous amount of data. The review of data before and after an inspection uses up most of the inspector's time. The need to decrease that time motivated the development of a prototype **expert system** for inspection planning. An exploratory **expert system** consisting of 53 rules was built, and it was used to assess the performance of a nuclear **power** plant (NPP) by looking at aspects such as operating level, transient frequency, engineered safety features (ESFs) actuations, and review procedures. The results obtained appear consistent with the experience of NPP inspectors and operators, thereby suggesting the utility of **expert systems** methodology to the nuclear industry.

DOCUMENT NUMBER = INI18:058257

TI: Improving plant availability by predicting reactor trips.
AU: Frank, M.V.; Epstein, S.A. (Management Analysis Co., 12671 High Bluff Drive, San Diego, CA 92130).
LA: English
MS: Proceedings of the 13th inter-ram conference. Anon.
IM: New York, NY (USA). American Society of Mechanical Engineers. 1986. p. 42-49.
CF: (13. Inter-RAM: international reliability, availability and maintainability conference for the electric **power** industry. Syracuse, NY (USA). 2-5 Jun 1986.)
AB: Management Ahnalysis Company (MAC) has developed and applied two complementary software packages called RiTSE and RAMSES. Together they provide an mini-computer workstation for maintenance and operations personnel to dramatically reduce inadvertent reactor trips. They are intended to be used by those responsible at the plant for authorizing work during operation (such as a clearance coordinator or shift foreman in U.S. **plants**). They discover and represent all components, processes, and their interactions that could case a trip. They predict if future activities at the plant would cause a reactor trip, provide a reactor trip warning **system** and aid in post-trip cause analysis. RAMSES is a general reliability engineering software package that uses concepts of **artificial intelligence** to provide unique capabilities on personal and mini-computers.

DOCUMENT NUMBER = INI18:056598

TI: Review of trends in computerized **systems** for operator support.
AU: Cain, D.G. (Electric **Power** Research Institute, Palo Alto, CA).
LA: English
JR: Nucl. Saf. ISSN 0029-5604. CODEN: NUSAA. (Oct-Dec 1986). v. 27(4) p. 488-498.
AB: This article examines past, present, and future trends in the development of computerized operator support **systems**. Previous research in disturbance analysis **systems** and the safety parameter display **systems** requirement were instrumental in the development of computerized operator support **systems**. Work today shows the trend toward increased hardware capability, attention to signal quality, event independence, display-procedures integration, improved design methodologies, on-line analysis capability, standardization, and office automation. Although the future is difficult to predict, one trend will be toward more intelligent software.

DOCUMENT NUMBER = INI18:052456

TI: Some new directions in **system** transient simulation.
AU: Ransom, V.H.
CO: EG and G Idaho, Inc., Idaho Falls (USA).
LA: English
RP: EGG-M--21286. CONF-861035--1.
IM: 1986. 23 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE87000442.
NO: Portions of this document are illegible in microfiche products.
CF: (2. international meeting on simulation methods in nuclear engineering. Montreal (Canada). 14-16 Oct 1986.)
AB: The current research in **system** transient simulation at the Idaho National Engineering Laboratory (INEL) is summarized in this paper and three new directions that are emerging from this work are discussed. The new directions are: development of an Advanced Thermal Hydraulic Energy Network Analyzer (ATHENA) having new modeling capability, use of **expert systems** for enhancing simulation methods, and the trend to individual workstations for simulation.

DOCUMENT NUMBER = INI18:051260

TI: **Expert system** for USNRC emergency response.
AU: Sebo, D.E.; Bray, M.A.; King, M.A.
CO: EG and G Idaho, Inc., Idaho Falls (USA).
LA: English
RP: EGG-M--11886. CONF-8610145--1.

IM: 1986. 7 p. Availability: INIS. Available from NTIS, PC A02/MF A01 - GPO as TI87000469.
CF: (2. **expert systems** in government conference. McLean, VA (USA). 20- 24 Oct 1986.)
AB: The Reactor Safety Assessment **System** (RSAS) is an **expert system** under development for the United States Nuclear Regulatory Commission (USNRC). RSAS is intended for use at the NRC's Operations Center in the event of a serious incident at a licensed nuclear **power** plant. RSAS is a situation assessment **expert system** which uses plant parametric data to generate conclusions for use by the NRC Reactor Safety Team. RSAS uses multiple rule bases and plant specific setpoint files in order to be applicable to all licensed **power plants**. RSAS currently covers several generic reactor types and **power plants** within those classes.

DOCUMENT NUMBER = INI18:051257

TI: Functional relationship **based** alarm processing **system** for nuclear **power**.
AU: Corsberg, D.; Sebo, D.
CO: EG and G Idaho, Inc., Idaho Falls (USA).
LA: English
RP: EGG-M--04986. CONF-860908--17.
IM: 12 Sep 1986. 8 p. Availability: INIS. Available from NTIS, PC A02/ MF A01 - GPO as TI87000466.
CF: (Operability of nuclear **power systems** in normal and adverse environments meeting. Albuquerque, NM (USA). 29 Sep - 3 Oct 1986.)
AB: At the Idaho National Engineering Laboratory (INEL), two **knowledge- based systems**, Alarm Filtering **System** (AFS) and Reactor Safety Assessment **System** (RSAS), are being developed to address the problem of information overload faced by operators during major plant transients. This paper discusses those **systems**, their strengths and weaknesses, and how their basic methodologies can be combined into a **system** that can better support the operator in identifying the plant state.

DOCUMENT NUMBER = INI18:048137

TI: Benefits from joint research activities by universities and national institutions for nuclear programmes in a developing country.
AU: Tzafestas, S.G. (National Technical Univ., Athens (Greece). Computer Science Div.); Zikides, C. (National Research Centre for the Physical Sciences Democritos, Athens (Greece)).
LA: English
MS: Significance and impact of nuclear research in developing countries. Proceedings of an international symposium held in Athens, Greece from 8-12 September 1986. International Atomic Energy Agency, Vienna (Austria).
RP: IAEA-SM--291/26.
IM: Vienna (Austria). IAEA. 1987. ISBN 92-0-070087-X. 371 p. p. 217-225.
SE: Proceedings series.
NO: 34 refs.
CF: (International symposium on the significance and impact of nuclear research in developing countries. Athens (Greece). 8-12 Sep 1986.)
AB: In a developing country such as Greece where a nuclear **power** programme is under consideration and faces special problems, collaboration between universities and national institutions in joint research projects, besides helping to obtain valuable results, is the best way to benefit both parties in their individual programmes. Ten years ago a joint research programme between the Control Group of the University of Patras and subsequently the National Technical University of Athens (NTUA) on the one side, and the Group of Research Reactor Modelling, Reliability and Failure Detection of the Nuclear Technology Department of the NRCPS 'Democritos' on the other, was established. The co-operation proved to be fruitful as regards (a) exchanging information and know-how relating to academically oriented activities and nuclear reactor and laboratory experiments and research; (b) working in joint research programmes on **systems** control theory and applications to nuclear **reactors**; and (c) training of post-graduate students in theory and applications as well as in methods and techniques used in a nuclear laboratory. Particular areas of joint activity and research are nuclear reactor modelling, simulation and identification; control theory methodology as applied to nuclear **reactors**; treatment of nuclear

power systems using large scale **systems** techniques; microprocessor-based **systems** for nuclear reactor measurements, data acquisition and nuclear signal processing and control; analysis and optimization of **system** reliability and maintenance with application to nuclear reactor **systems**; dynamic failure detection of nuclear **systems** using state observers and filters; and application of knowledge-based **systems** and **artificial intelligence** techniques to nuclear reactor **systems** (evaluation, operation, maintenance and safety procedures). Naturally, this co-operation has contributed to the preparation of the necessary manpower for carrying out research and practical work in the **systems** and control area of the nuclear field, and for following up the new technologies. (author).

DOCUMENT NUMBER = INI18:047139

TI: Implicit Kalman filter algorithm for nuclear reactor analysis.
AU: Hassberger, J.A.; Lee, J.C. (Univ. of Michigan, Ann Arbor).
LA: English
RP: CONF-861102--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1986). v. 53 p. 249-250.
CF: (American Nuclear Society and Atomic Industrial Forum joint meeting. Washington, DC (USA). 16-21 Nov 1986.)
AB: **Artificial intelligence** (AI) is currently the hot topic in nuclear **power** plant diagnostics and control. Recently, researchers have considered the use of simulation as **knowledge** in which faster than real-time best-estimate simulations **based** on first principles are tightly coupled with AI **systems** for analyzing **power** plant transients on-line. On-line simulations can be improved through a Kalman filter, a mathematical technique for obtaining the optimal estimate of a **system** state given the information contained in the equations of **system** dynamics and measurements made on the **system.** Filtering can be used to **systemically** adjust parameters of a low-order simulation model to obtain reasonable agreement between the model and actual plant dynamics. The authors present here a general Kalman filtering algorithm that derives its information of **system** dynamics implicitly and naturally from the discrete time step-series of state estimates available from a simulation program. Previous research has demonstrated that models adjusted on past data can be coupled with an intelligent controller to predict the future time- course of plant transients.

DOCUMENT NUMBER = INI18:046284

TI: Quantile arithmetic methodology for uncertainty propagation in fault trees.
AU: Abdelhai, M.; Ragheb, M. (Univ. of Illinois, Urbana).
LA: English
RP: CONF-861102--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1986). v. 53 p. 391-393.
CF: (American Nuclear Society and Atomic Industrial Forum joint meeting. Washington, DC (USA). 16-21 Nov 1986.)
AB: A methodology **based** on quantile arithmetic, the probabilistic analog to interval analysis, is proposed for the computation of uncertainties propagation in fault tree analysis. The basic events' continuous probability density functions (pdf's) are represented by equivalent discrete distributions by dividing them into a number of quantiles N. Quantile arithmetic is then used to performthe binary arithmetical operations corresponding to the logical gates in the Boolean expression of the top event expression of a given fault tree. The computational advantage of the present methodology as compared with the widely used Monte Carlo method was demonstrated for the cases of summation of M normal variables through the efficiency ratio defined as the product of the labor and error ratios. The efficiency ratio values obtained by the suggested methodology for M = 2 were 2279 for N = 5, 445 for N = 25, and 66 for N = 45 when compared with the results for 19,200 Monte Carlo samples at the 40th percentile point. Another advantage of the approach is that the exact analytical value of the median is always obtained for the top event.

DOCUMENT NUMBER = INI18:042990

TI: Development of reactor accident diagnostic **system** DISKET using **knowledge** engineering technique.
AU: Yokobayashi, Masao; Yoshida, Kazuo; Kohsaka, Atsuo (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment); Yamamoto, Minoru.
LA: English
JR: J. Nucl. Sci. Technol. (Tokyo). ISSN 0022-3131. CODEN: JNSTA. (Apr 1986). v. 23(4) p. 300-314.
AB: An accident diagnostic **system** DISKET has been developed to identify the cause and the type of an abnormal transient of a nuclear **power** plant. The **system** is **based** on the **knowledge** engineering (KE) and consists of an inference engine IERIAS and a **knowledge** base. The main features of DISKET are the following : (1) Time-varying characteristics of transients can be treated. (2) **Knowledge** base can be divided into several **knowledge** units to handle a lot of rules effectively. (3) Programming language UTILISP, which is a dialect of LISP, is used to manipulate symbolic data effectively. For the verification of DISKET, performance tests have been conducted for several types of accidents. The **knowledge** base used in the tests was generated from the data of various types of transients produced by a PWR plant simulator. The results of verification studies showed a good applicability of DISKET to reactor accident diagnosis. (author).

DOCUMENT NUMBER = INI18:041137

TI: BWR shutdown analyzer using **artificial intelligence** (AI) techniques.
AU: Cain, D.G.
LA: English
MS: 1985 nuclear **power** plant safety control technology seminar: proceedings. Divakaruni, S.M.; Sun, W.K.H. (eds.). Electric **Power** Research Inst., Palo Alto, CA (USA).
RP: EPRI-NP--4750-SR.
IM: Sep 1986. p. 6.55-6.86. Availability: Research Reports Center, Box 50490, Palo Alto, CA 94303.
CF: (EPRI seminar on nuclear **power** plant safety control technology. Palo Alto, CA (USA). 4-6 Feb 1985.)
AB: A prototype alarm **system** for detecting abnormal reactor shutdowns **based** on **artificial intelligence** technology is described. The **system** incorporates **knowledge** about Boiling Water Reactor (BWR) plant design and component behavior, as well as **knowledge** required to distinguish normal, abnormal, and ATWS accident conditions. The **system** was developed using a software tool environment for creating knowledge-based applications on a LISP machine. To facilitate prototype implementation and evaluation, a casual simulation of BWR shutdown sequences was developed and interfaced with the alarm **system**. An intelligent graphics interface for execution and control is described. **System** performance considerations and general observations relating to **artificial intelligence** application to nuclear **power** plant problems are provided.

DOCUMENT NUMBER = INI18:041136

TI: Signature analysis techniques for plant monitoring.
AU: Fry, D.N.; Kim, J.H.
LA: English
MS: 1985 nuclear **power** plant safety control technology seminar: proceedings. Divakaruni, S.M.; Sun, W.K.H. (eds.). Electric **Power** Research Inst., Palo Alto, CA (USA).
RP: EPRI-NP--4750-SR.
IM: Sep 1986. p. 6.37-6.51. Availability: Research Reports Center, Box 50490, Palo Alto, CA 94303.
CF: (EPRI seminar on nuclear **power** plant safety control technology. Palo Alto, CA (USA). 4-6 Feb 1985.)
AB: Noise signature analysis can provide diagnostic information in nuclear plant operators to aid them in identification of abnormal trends, thus preventing some unscheduled shutdowns. The application of computer-based automated diagnostic methods **based** on pattern recognition, **artificial intelligence**, and **expert systems** will enable noise analysis to be used in the plant environment without the need for on-site trained noise analysts.

DOCUMENT NUMBER = INI18:041133

TI: Software quality - how is it achieved?.
AU: Straker, E.A.
LA: English
MS: 1985 nuclear **power** plant safety control technology seminar: proceedings. Divakaruni, S.M.; Sun, W.K.H. (eds.). Electric **Power** Research Inst., Palo Alto, CA (USA).
RP: EPRI-NP--4750-SR.
IM: Sep 1986. p. 6.3-6.13. Availability: Research Reports Center, Box 50490, Palo Alto, CA 94303.
CF: (EPRI seminar on nuclear **power** plant safety control technology. Palo Alto, CA (USA). 4-6 Feb 1985.)
AB: Although software quality can't be quantified, the tools and techniques to achieve high quality are available. As management stresses the need for definable software quality programs from vendors and subcontractors and provides the incentives for these programs, the quality of software will improve. EPRI could provide the leadership in establishing guidelines for a balanced software quality program and through workshops provide training to utility staff and management on the methods for evaluating the characteristics of quality software. With the more complex **systems** discussed at this workshop and particularly with the trend toward the use of **artificial intelligence**, the importance of quality software will grow dramatically.

DOCUMENT NUMBER = INI18:041109

TI: Role of high technology in the nuclear industry.
AU: Cain, D.G.
LA: English
MS: 1985 nuclear **power** plant safety control technology seminar: proceedings. Divakaruni, S.M.; Sun, W.K.H. (eds.). Electric **Power** Research Inst., Palo Alto, CA (USA).
RP: EPRI-NP--4750-SR.
IM: Sep 1986. p. 1.45-1.60. Availability: Research Reports Center, Box 50490, Palo Alto, CA 94303.
CF: (EPRI seminar on nuclear **power** plant safety control technology. Palo Alto, CA (USA). 4-6 Feb 1985.)
AB: A discussion of high technology identifies the characteristics which distinguish it from conventional technologies, and the impact high technology will have in the nuclear **power** industry in the near future. The basic theme is that high technology is an ensemble of competing technological developments that shifts with time and technological innovation. The attributes which current distinguish high technology are compactness, plasticity, convergence, and **intelligence.** These high technology attributes are presented as a prelude to some examples of high technology developments which are just beginning to penetrate the nuclear industry. Concluding remarks address some of the challenges which must be faced in order to assure that high technology is successfully adapted and used.

DOCUMENT NUMBER = INI18:038090

TI: Computer aided analysis of disturbances. Rechnergestuetzte Analyse von Stoerungen.
AU: Baldeweg, F.; Lindner, A.
LA: German
IM: Berlin (German Democratic Republic). Akademie-Verlag. 1986. 152 p.
SE: Beitraege zur Forschungstechnologie. v. 14.
AB: Computer aided analysis of disturbances and the prevention of failures (diagnosis and therapy control) in technological **plants** belong to the most important tasks of process control. Research in this field is very intensive due to increasing requirements to security and economy of process control and due to a remarkable increase of the efficiency of digital electronics. This publication concerns with analysis of disturbances in complex technological **plants,** especially in so called high risk processes. The presentation emphasizes theoretical concept of diagnosis and therapy control, modelling of the disturbance behaviour of the technological process and the man-machine-communication integrating **artificial intelligence** methods, e.g., **expert system** approach. Application is given for nuclear **power plants.** (author).

DOCUMENT NUMBER = INI18:037469

TI: Applications of industrial machine vision **systems** in the nuclear energy.
AU: Vandergheynst, A.; Vanderborck, Y. (Belgonucleaire, Rue du Champ de Mars, 25, B - 1050 Brussels).
LA: English
MS: Robotics and remote handling in hostile environments. Anon.
IM: LaGrange Park, IL (USA). American Nuclear Society. 1984. p. 97- 104.
CF: (Topical meeting on robotics and remote handling in hostile environments. Gatlinburg, TN (USA). 23-26 Apr 1984.)
AB: In the paper, two multi-functional machine **systems** basically developed for the industrial robotics and representing the state of the art are presented. Their potential applications in the nuclear industry (nuclear **power plants** and fuel cycle facilities) are reviewed.

DOCUMENT NUMBER = INI18:029234

TI: Reactor Safety Assessment **System--A** situation assessment aid for USNRC emergency response.
AU: Bray, M.A.; Sebo, D.E.; Dixon, B.W. (Idaho National Engineering Lab., EGandG Idaho, Inc., P.O. Box 1625, Idaho Falls, ID 83415).
LA: English
MS: **Expert systems** in government symposium. Karna, K.N.
IM: Piscataway, NJ (USA). IEEE Service Center. 1985. p. 423-430. ISBN 0-8186-0686-X.
CF: (**Expert systems** in government symposium. McLean, VA (USA). 23-25 Oct 1985.)
AB: The Reactor Safety Assessment **System** (RSAS) is an **expert system** under development for the United States Nuclear Regulatory Commission (USNRC). RSAS is intended for use at the NRC's Operations Center in the event of a serious incident at a licensed nuclear **power** plant. The **system** uses plant parameter data and status information from the **power** plant. It has a rule base that uses the parametric values, the known operator actions, and the time sequence information in the data to generate situation assessment conclusions for use by the NRC Reactor Safety Team. RSAS rules currently cover one specific reactor type and use setpoints specific to one **power** plant.

DOCUMENT NUMBER = INI18:024485

TI: Paradigm for **expert** display **systems** in nuclear plant and elsewhere.
AU: Gabriel, J.R. (Mathematics and Computer Science Div., Argonne National Lab., Argonne, IL 60439-4844).
LA: English
JR: IEEE Trans. Nucl. Sci. ISSN 0018-9499. CODEN: IETNA. (Oct 1986). v. NS-33(5) p. 1147-1149.
AB: Work on a Rankine cycle display of operation for a nuclear **power** station has recently been published. Discovery of the display was lengthy and difficult. This paper discusses similar CRT displays of plant operation from the point of view of **knowledge** representation and presents an abstraction describing a wide class of such displays. It is suggested that this abstraction is a useful tool for discovery of good safety displays. An example for and idealized, very simple plant is given to explain the concept.

DOCUMENT NUMBER = INI18:023862

TI: Simulation-based **expert system** for nuclear reactor control and diagnostics. Progress report.
AU: Lee, J.C.; Martin, W.R.
CO: Michigan Univ., Ann Arbor (USA). Dept. of Nuclear Engineering.
LA: English
RP: DOE/NE/37945--1.
IM: 31 Jul 1986. 18 p. Availability: INIS. Available from NTIS, PC A02/MF A01 as DE86014668.

AB: This research concerns the development of **artificial intelligence** (AI) techniques suitable for application to the diagnostics and control of nuclear **power** plant **systems.** The overall objective of the current effort is to build a prototype simulation-based **expert system** for diagnosing accidents in nuclear **reactors.** The **system** is being designed to analyze plant data heuristically using fuzzy logic to form a set of hypotheses about a particular transient. Hypothesis testing, fault magnitude estimation and transient analysis is performed using simulation programs to model plant behavior. An adaptive learning technique has been developed for achieving accurate simulations of plant dynamics using low-order physical models of plant components. The results of the diagnostics and simulation analysis of the plant transient are to be analyzed by an **expert system** for final diagnoses and control guidance. To date, significant progress has been made toward achieving the primary goals of this project. **Based** on a critical safety functions approach, an overall design for the nuclear plant **expert system** has been developed. The methodology for performing diagnostic reasoning on plant signals has been developed and the algorithms implemented and tested. A methodology for utilizing the information contained in the physical models of plant components has also been developed. This work included the derivation of a unique Kalman filtering algorithm for using **power** plant data to **systematically** improve on-line simulations through the judicious adjustment of key model parameters. A few simulation models of key plant components have been developed and implemented to demonstrate the method on a realistic accident scenario. The chosen transient is a loss of feed flow exasperated by a stuck open relief valve, similar to the initiating event of the Three Mile Island Unit 2 accident in 1979.

DOCUMENT NUMBER = INI18:023856

TI: Models of cognitive behavior in nuclear **power** plant personnel. A feasibility study: main report. Volume 2.
AU: Woods, D.D.; Roth, E.M.; Hanes, L.F.
CO: Westinghouse Electric Corp., Pittsburgh, PA (USA). Research and Development Center.
LA: English
RP: NUREG/CR--4532-Vol.2.
IM: Jul 1986. 178 p. Availability: INIS. Available from NTIS, PC A09/ MF A01 - GPO as TI86901611.
AB: This report contains the results of a feasibility study to determine if the current state of models human cognitive activities can serve as the basis for improved techniques for predicting human error in nuclear **power plants** emergency operations. **Based** on the answer to this questions, two subsequent phases of research are planned. Phase II is to develop a model of cognitive activities, and Phase III is to test the model. The feasibility study included an analysis of the cognitive activities that occur in emergency operations and an assessment of the modeling concepts/tools available to capture these cognitive activities. The results indicated that a symbolic processing (or **artificial intelligence**) model of cognitive activities in nuclear **power plants** is both desirable and feasible. This cognitive model can be built upon the computational framework provided by an existing **artificial intelligence system** for medical problem solving called Caduceus. The resulting cognitive model will increase the capability to capture the human contribution to risk in probabilistic risk assessments studies. Volume I summarizes the major findings and conclusions of the study. Volume II provides a complete description of the methods and results, including a synthesis of the cognitive activities that occur during emergency operations, and a literature review on cognitive modeling relevant to nuclear **power plants.** 112 refs., 10 figs.

DOCUMENT NUMBER = INI18:022282

TI: A knowledge-based **system** for estimating physical inventories in MBA's involving complex chemical processes.
AU: Argentesi, F.; Costantini, L.; **Power,** R. (Commission of the European Communities, Joint Research Centre - Ispra Establishment, A.I. Lab., 21020 Ispra (Va)).
LA: English
RP: CONF-860654--.

JR: Nucl. Mater. Manage. ISSN 0362-0034. CODEN: NUMMB. (1986). v. 15 p. 105-108.
CF: (27. annual meeting of the Institute of Nuclear Materials Management. New Orleans, LA (USA). 22-25 Jun 1986.)
AB: There is an increasing demand for effective methods of nuclear materials accountancy. In order to keep accounts for any material balance area, it is necessary to measure, or estimate, the inputs and outputs during the assessment period, and the initial and final inventories. Inputs, outputs, and some components of the inventory usually pose no problem since the material is held in containers which can be weighed and sampled. The difficulty lies in the in-process inventory, e.g. material contained in **reactors** and kilns. Although in-process inventories cannot be measured directly, a plant manager usually has a fairly good idea of how much material the plant contains. A possible solution to the problem is, therefore, to create an **expert system** which incorporates the **knowledge** on which such estimates are **based.** An experimental **system** of this kind has been developed. This paper presents a detailed description of the program's **knowledge** base and some preliminary results on its performance.

DOCUMENT NUMBER = INI18:019681

TI: Identify electric-output losses at PWR nuclear **plants.**
AU: Chiu, C. (Southern California Edison, Co.).
LA: English
JR: **Power.** ISSN 0032-5929. CODEN: POWEA. (Sep 1986). v. 130(9) p. 35- 42.
AB: Licensing difficulties and reduced electrical-growth demands of recent years have contributed strongly to the static state of the nuclear industry. At the same time, a substantial electricity growth for the next five years-up to 3.4%/year-is forseen. One method to cope with this demand/supply mismatch is to improve the heat rate of existing **plants.** A 1% heat rate improvement across-the-board for all nuclear **plants,** for example, will compensate for more than 5% of the yearly electricity-demand growth. Techniques used to monitor the heat rate and diagnose the causes of degradation vary greatly from utility to utility. The simplest program, which is still being used by utilities with a small engineering staff, is to periodically monitor the heat rate or electrical output. If the heat rate goes beyond a present value, an engineer is sent to investigate. Unfortunately, the findings are often more qualitative than quantitative. For a utility having a dedicated performance engineer or group, the practice for heat-rate monitoring and diagnosis is more elaborate. It often comprises two subprograms-data trending and heat-rate analysis. Data trending means that a selected group of measured data related to plant heat rate is taken periodically and their trends are closely monitored. A heat-rate analysis, usually in accordance with one of the heat-rate codes, is performed to identify the causes of the degradation and to optimize the fix. The analysis requires a trial-and-error process to pinpoint the causes.

DOCUMENT NUMBER = INI18:015175

TI: **Expert systems** application to plant diagnosis and sensor data validation.
AU: Hashemi, S.; Hajek, B.K.; Miller, D.W.; Chandrasekaran, B.; Josephson, J.R. (Nuclear Engineering Program, Dept. of Mechanical Engineering, Columbus, OH).
LA: English
MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 65.01- 65.13.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: In a nuclear **power** plant, over 2000 alarms and displays are available to the operator. For any given set of alarms and displays, the operator must be able to diagnose and correct the problem (s) quickly and accurately. At the same time, the operator is expected to distinguish the plant **system** faults from instrumentation channel failures and drifts. Needs for plant operator aids have been considered since the accident at TMI. Many of these aids are of the form of the Safety Parameter Display **Systems** and offer improved methods of displaying otherwise available data to the operator in a more concise and summarized format. diagnosis,

however, remains a desirable objective of an operator aid. At The Ohio State University, faculty and students in nuclear engineering and computer science have evaluated this problem. The results of these studies have shown that plant diagnosis and sensor data validation must be considered as one integral problem and cannot be isolated from one another. Otherwise, an incorrect diagnosis **based** on faulty instrument information might be provided to the operator. In this study, the Knowlege **Based System** (KBS) technology is being incorporated to accomplish a final goal of an intelligent operator aid **system.**

DOCUMENT NUMBER = INI18:015174

TI: A new trend in diagnostics: **Expert systems.**
AU: Tanasescu, C. (Institute of Nuclear **Power Reactors**, Pitesti).
LA: English
MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 64.01- 64.10.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: Diagnostic is an intelligent act which means recognition of offnormal and identification of cause of plant. In the diagnosis process the operator has an image of the **system** in his mind by education, training, experience and makes use of his reasoning capacity. The lack of a centralized **system** to take into consideration the overall available data may lead to unnecessary shutdowns because of false alarms. It will be difficult all automatization of diagnosis instead of man, especially automatization of operator's diagnosis process. The solution would be a computerized operator support **system** to assist the operator in decision making. The paper brings into attention the use of an **expert system** for diagnosis in nuclear **power plants** and draws a global overview of such a **system.** As it is known an **expert system** consists in a resolution **system** (application independent) a **knowledge** base (representation of **knowledge** in declarative form) and a communication **system.** The **system** input consists in measured data from the process. This data are compared with rule conditions and associated actions are executed. The **system** provides a classification of possible diagnostics in the decreased order of certainty factors.

DOCUMENT NUMBER = INI18:015173

TI: Intelligent process control operator aid -- An **artificial intelligence** approach.
AU: Sharma, D.D.; Miller, D.D.; Hajek, B.; Chandrasekaran, B. (AI/ **Knowledge Based Systems** Group, Battelle Memorial Institute, 505 King Avenue, Columbus, OH).
LA: English
MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 61.01- 61.20.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: This paper describes an approach for designing intelligent process and **power** plant control operator aids. It is argued that one of the key aspects of an intelligent operator aid is the capability for dynamic procedure synthesis with incomplete definition of initial state, unknown goal states, and the dynamic world situation. The dynamic world state is used to determine the goal, select appropriate plan steps from prespecified procedures to achieve the goal, control the execution of the synthesized plan, and provide for dynamic recovery from failure often using a goal hierarchy. The dynamic synthesis of a plan requires integration of various problems solving capabilities such as plan generation, plan synthesis, plan modification, and failure recovery from a plan. The programming language for implementing the DPS framework provides a convenient tool for developing applications. An application of the DPS approach to a Nuclear **Power** Plant emergency procedure synthesis is also described. Initial test results indicate that the approach is successful in dynamically synthesizing the procedures. The authors realize that the DPS framework is not a solution for all control tasks. However, many existing process and plant control problems satisfy the requirements discussed in the paper and should be able to benefit from the framework described.

DOCUMENT NUMBER = INI18:015172

TI: **Artificial Intelligence** application to surveillance and diagnosis of nuclear **power plants.**
AU: Brunet, E.; Monnier, B.; Zwingelstein, G. (Universite de Compiegne - Genie Informatique - Compiegne).
LA: English
MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 60.01- 60.28.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: Acquisition and representation of **knowledge** are fundamental problems in Aritificial Intelligence and especially in **expert system** domain. In this article, the authors propose a conceptual model allowing to describe the universe the **expert** is working on when trying to make a diagnosis. The **expert** determines the descriptors and predicates with which he describes laws, states, objects involved in his reasoning. With this model, they are presenting design and development of **Knowledge** Acquisition Module allowing a dialogue in a specialized natural language used by the **expert system** to assist in the detection of loose parts in nuclear **power plants.** Using a classical lexical and syntactic analysis, they propose to associate grammatical semantic properties to the specialized language words. The method used allows the design of a sub **expert system** for understanding of the specialized natural language.

DOCUMENT NUMBER = INI18:015171

TI: State and data techniques for control of discontinuous **systems.**
AU: Kisner, R.A. (Oak Ridge National Lab., Oak Ridge, TN).
LA: English
MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 45.01- 45.20.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: The need for automated control **systems** becomes clear as the complexity of nuclear **power plants** increases and economic incentives demand higher plant availability. A control **system** with **intelligence** distributed throughout its controllers allows reduction in operator workload, perhaps reduction in crew size, and potentially a reduction in on-line human error. In automated **systems** of this kind, each controller should be capable of making decisions and carrying out a plan of action. This paper describes a technique for structured analysis and design of automated control **systems.** The technique integrates control of continuous and discontinuous nuclear **power** plant subsystems and components. A hierarchical control **system** with distributed **intelligence** follows from applying the technique. Further, it can be applied to all phases of control **system** design. For simplicity, the example used in the paper is limited to phase I design (basic automatic control action), in which no maintenance, testing, or contingency capability is attempted.

DOCUMENT NUMBER = INI18:014723

TI: A summary of the **artificial intelligence** applications at the HFIR.
AU: Wehe, D.K.; Clapp, N.E.; Clark, F.H.; Mullens, J.A.; Otaduy, P.J. (Oak Ridge National Lab.).
LA: English
MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 62.01- 62.04.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: The AI group within the Instrumental and Controls Division at the Oak Ridge National Laboratory is developing expertise in AI techniques, and applying it to various projects. One such project involves the High Flux Isotope Reactor (HFIR). This paper

summarizes the progress which has been made in the first year of this three-year project. While the HFIR is as a research reactor, it shares many of the characteristics of a full-scale, commercial PWR. It has a pressurized primary **system** (including a component similar to a pressurizer), with multiple primary and secondary coolant legs. In essence, it possesses many of the complexities found in commercial **plants.** The principle differences are its small, loosely coupled, annular core which produces 100 MWt, the concentric, cylindrical control elements which are located external to the core, and the beryllium reflectors which are external to the control elements. Much like a commercial plant, operational emphasis is placed on maximizing fuel utilization and plant, availability, while minimizing safety risks, radiation exposure, and production of low-level wastes. Thus, the HFIR is a realistic platform for developing and testing real-time **expert systems** for the nuclear industry.

DOCUMENT NUMBER = INI18:014659

TI: The construction and use of a **knowledge** base in the real-time control of research reactor **power.**
AU: Bernard, J.A. (Nuclear Reactor Lab., Massachusetts Institute of Technology, 138 Albany Street, Cambridge, MA).
LA: English
MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 57.01- 57.25.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: The construction of a **knowledge** base and its use in developing a rule-based methodology for the transient, closed-loop, digital control of the **power** on the 5 MWt MIT Research Reactor (MITR-II) is described. Material is first presented on the nature of the control task, the procedure used for **knowledge** acquisition, the contents of the **knowledge** base, the verification of the information contained therein, and the use of 'fuzzy' logic to represent that **knowledge.** The rationale, structure, and implementation of a possible set of conditional rules for the transient control of the MITR-II's **power** is then described together with the results of a successful experimental trial of this rule-based controller. The possible role of the rule-based approach in process control is discussed. The proposal is made that rule-based and analytic technologies should be merged in order to obtain a robust controller capable of displaying both the sequence and 'reasoning' of its automatic actions to human supervisory personnel.

DOCUMENT NUMBER = INI18:014657

TI: Condition monitoring **system** for rotating equipment in a CANDU **power** plant.
AU: Elbestawi, M.A.; Lau, D.F.; Tait, H.J. (Mechanical Research Dept., Ontario Hydro Research Div., Toronto, Ontario).
LA: English
MS: Analysis of energy **systems.** Design and operation. Gaggioli, R.A.
IM: New York, NY (USA). American Society of Mechanical Engineers. 1985. p. 49-58.
CF: (American Society of Mechanical Engineers winter annual meeting. Miami, FL (USA). 17-21 Nov 1985.)
AB: The rate of deterioration of **power** plant machinery varies significantly, and it is generally not known when the equipment will require maintenance of what maintenance should be performed. One common approach is to perform regularly scheduled preventive maintenance. The main problems associated with this approach are equipment breakdown prior to scheduled maintenance, performing unnecessary maintenance on machinery in "good" condition, and introducing defects in "healthy" equipment during maintenance. A second approact, which presents a number of advantages over the previous one, is to perform maintenance **based** on the results of condition monitoring. This paper deals with the design and development of an on-line, computer managed condition monitoring **system** for major pumpsets in a CANDU nuclear generating station using virbration analysis. In total 124 pumps are monitored with 508 sensors. The monitoring **system** automatically performs baseline comparisons, generated trend plots, and performs vibration signature analysis to

provide diagnostic information. The diagnostic function is implemented through a knowledge-based **system**. The diagnostic logic incorporated in this knowledge-based **system** automates the decision-making process that a vibration analyst would follow in evaluating a set of vibration signatures.

DOCUMENT NUMBER = INI18:014607

TI: Automated generation of nuclear **power** plant safety information (qualitative simulation and derivation of failure symptom **knowledge**).
AU: Washio, T.; Kitamura, M.; Kotajima, K.; Sugiyama, K. (Dept. of Nuclear Engineering, Tohoku Univ., Sendai).
LA: English
MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 39.01- 39.17.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: The aim of this study is to develop a general purpose information generating (GPIG) **system** for nuclear **power** plant (NPP) safety by using the **artificial intelligence** technique. This **system** makes it possible to derive safety-related **knowledge** necessary for the transient analysis, the risk analysis and the failure diagnosis in an unified manner, utilizing the design information database (DID) of an NPP as the common information resource. The goal of this study is to establish a basic technique for automated generation of safety information for an NPP. This technique can enhance the credibility of the derived safety information by minimizing the risk of introducing human error during the tasks such as construction of simulation code, FT generation and CCT generation.

DOCUMENT NUMBER = INI18:014453

TI: An object-oriented alarm-filtering **system**.
AU: Corsberg, D.R.; Wilkie, D. (Idaho National Engineering Lab., P.O. Box 1625, Idaho Falls, ID).
LA: English
MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 59.01- 59.16.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: This paper discusses an alarm-filtering **system** (AFS) being developed by EG and G Idaho, Inc. for the Advanced Test Reactor (ATR) at the Idaho National Engineering Laboratory. The ultimate goal of this project is to place AFS into ATR's reactor control room to act as an aid during major plant transients. In addition, methods of alarm analysis are investigated **based** on functional relationships rather than on a historical approach utilizing cause-consequence trees. **Artificial intelligence** techniques, including object-oriented programming, are also demonstrated as useful in analyzing alarms and alarm sequences. After a brief description of the problem AFS addresses, this paper discusses the design constraints and human factors that influenced the development of the **system**. The reader is then presented with operational and architectural descriptions of the **system** as well as what directions the future development of AFS may take. The fact that AFS is being considered as a partial solution to the problems discussed in the next section demonstrates the viability of its underlying technology and approach.

DOCUMENT NUMBER = INI18:014395

TI: Heuristic learning parameter identification for surveillance and diagnostics of nuclear **power plants**.
AU: Machado, E.L.; Perez, F.B.; King, T.L. (Dept. of Nuclear Engineering, Univ. of Tennessee, Knoxville, TN).
LA: English

MS: Proceedings of **power** plant dynamics, control and testing symposium. Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
IM: Knoxville, TN (USA). University of Tennessee. 1986. p. 63.01- 63.21.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: An Artifical **Intelligence** Heuristic Learning Algorithm for **System** parameter identification has been developed. This algorithm has been applied to the on-line surveillance of a pressure loop, simulating the dynamics of pressure disturbances in a PWR plant.

DOCUMENT NUMBER = INI18:014379

TI: Proceedings of **power** plant dynamics, control and testing symposium.
AU: Upadhyaya, B.R.; Kerlin, T.W.; Katz, E.M.
LA: English
RP: CONF-860414--.
IM: Knoxville, TN (USA). University of Tennessee. 1986. vp.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: This book presents the papers given at a symposium on reactor control and monitoring **systems.** Topics considered at the symposium included **power** plant modeling and simulation, reactor simulators, **power** reactor surveillance by noise analysis, in core instruments, computer codes, advanced data analysis techniques, loss of coolant simulation, nuclear **power** plant testing, reactor safety, **expert systems,** applications of **artificial intelligence** methodology, human factors, man-machine **systems,** and **power** plant diagnostics.

DOCUMENT NUMBER = INI18:014362

TI: Models of cognitive behavior in nuclear **power** plant personnel. A feasibility study: summary of results. Volume 1.
AU: Woods, D.D.; Roth, E.M.; Hanes, L.F.
CO: Westinghouse Electric Corp., Pittsburgh, PA (USA). Research and Development Center.
LA: English
RP: NUREG/CR--4532-Vol.1.
IM: Jul 1986. 27 p. Availability: INIS. Available from NTIS, PC A03/ MF A01 - GPO as TI86901610.
AB: This report summarizes the results of a feasibility study to determine if the current state of models of human cognitive activities can serve as the basis for improved techniques for predicting human error in nuclear **power plants** emergency operations. **Based** on the answer to this question, two subsequent phases of research are planned. Phase II is to develop a model of cognitive activities, and Phase III is to test the model. The feasibility study included an analysis of the cognitive activities that occur in emergency operations and an assessment of the modeling concepts/tools available to capture these cognitive activities. The results indicated that a symbolic processing (or **artificial intelligence)** model of cognitive activities in nuclear **power plants** is both desirable and feasible. This cognitive model can be built upon the computational framework provided by an existing **artificial intelligence system** for medical problem solving, called Caduceus. The resulting cognitive model will increase the capability to capture the human contribution to risk in probabilistic risk assessment studies. Volume 1 summarizes the major findings and conclusions of the study. Volume 2 provides a complete description of the methods and results, including a synthesis of the cognitive activities that occur during emergency operations, and a literature review on cognitive modeling relevant to nuclear **power plants.** 19 refs.

DOCUMENT NUMBER = INI18:011512

TI: Automation of neutral beam source conditioning with **artificial intelligence** techniques.
AU: Johnson, R.R.; Canales, T.; Lager, D. (Lawrence Livermore National Labs., P.O. Box 808, Livermore, CA 94550).

LA: English
MS: Proceedings of the 11th symposium on fusion engineering. Anon.
IM: Piscataway, NJ (USA). IEEE Service Center. 1986. p. 289-292.
CF: (11. symposium on engineering problems in fusion research. Austin, TX (USA). 18-22 Nov 1985.)
AB: This paper describes a **system** that automates neutral beam source conditioning. The **system** achieves this with **artificial intelligence** techniques. The architecture of the **system** is presented followed by a description of its performance.

DOCUMENT NUMBER = INI18:009924

TI: Instrumentation in the **power** industry. Volume 28.
AU: Anon.
LA: English
RP: CONF-850527--.
IM: Research Triangle Park, NC (USA). Instrument Society of America. 1985. 220 p.
CF: (ISA **power** industry symposium. New Orleans, LA (USA). 20-22 May 1985.)
AB: The thirteen papers here were given at a conference on the monitoring and control of various **power plants.** Types of **plants** covered included nuclear **plants,** cogeneration **plants,** and fossil fuel **plants.** Topics discussed included, retrofitting, economics of control **systems,** reliability, and photovoltaic **power** supplies for control instruments. The emphasis was on the use of computers, including **artificial intelligence** and **expert systems,** in control **systems.** One paper discusses fiber optics in a natural gas pipeline and one discusses combustion efficiency in a coal-fuel stokes.

DOCUMENT NUMBER = INI18:009862

TI: Role of **expert systems** in the operation and control of nuclear **power plants.**
AU: Rozenholc, M.
CO: Societe Franco-Americaine de Constructions Atomiques (FRAMATOME), 92 - Courbevoie (France).
LA: English
RP: FRADOC--6-11.
IM: Sep 1985. 6 p. Availability: INIS.
CF: (International topical meeting on computer applications for nuclear **power** plant operation and control. Pasco, WA (USA). 8-12 Sep 1985.)
AB: This paper summarizes the past improvements and remaining limitations of plant control **systems,** defines **knowledge** engineering and proposes some prospects for future uses of **expert systems** in plant operation, discusses the present involvement in **expert systems** and the status of the first applications and, finally, draws some simple conclusions.

DOCUMENT NUMBER = INI18:009861

TI: **Artificial intelligence** and nuclear **power.** Report by the Technology Transfer **Artificial Intelligence** Task Team.
CO: USDOE Assistant Secretary for Nuclear Energy, Washington, DC.
LA: English
RP: DOE/NE--0064.
IM: Jun 1985. 89 p. Availability: INIS. Available from NTIS, PC A05/ MF A01; 1 as DE86011387.
NO: Portions of this document are illegible in microfiche products. Original copy available until stock is exhausted.
AB: The **Artificial Intelligence** Task Team was organized to review the status of **Artificial Intelligence** (AI) technology, identify guidelines for AI work, and to identify work required to allow the nuclear industry to realize maximum benefit from this technology. The state of the nuclear industry was analyzed to determine where the application of AI technology could be of greatest benefit. Guidelines and criteria were established to focus on those particular problem areas where AI could provide the highest possible payoff to the industry. Information was collected from

government, academic, and private organizations. Very little AI work is now being done to specifically support the nuclear industry. The AI Task Team determined that the establishment of a Strategic Automation Initiative (SAI) and the expansion of the DOE Technology Transfer program would ensure that AI technology could be used to develop software for the nuclear industry that would have substantial financial payoff to the industry. The SAI includes both long and short term phases. The short-term phase includes projects which would demonstrate that AI can be applied to the nuclear industry safely, and with substantial financial benefit. The long term phase includes projects which would develop AI technologies with specific applicability to the nuclear industry that would not be developed by people working in any other industry.

DOCUMENT NUMBER = INI18:006066

TI: **Expert system** for helping diagnostic of electronic equipments. (SPIN: Digital Integrated Protection **System** of PWR 1300 MW.) **Systeme expert** d'aide au diagnostic d'equipements electroniques.
AU: Colling, J.M. (Merlin-Gerin, 38 - Grenoble (France)).
LA: French
MS: Proceedings of the JIE: 86 meeting exhibition. Compte rendu du colloque et de l'exposition JIE - 86. CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Inst. de Recherche Technologique et de Developpement Industriel (IRDI).
IM: Gif-sur-Yvette (France). Commissariat a l'Energie Atomique. 1986. 291 p. p. 223-243.
CF: (Information meeting on electronics at CEN Saclay (JIE - 86). Gif-sur-Yvette (France). 21-23 Jan 1986.)
AB: Merlin Gerin has been constructing for more than 20 years Control-Monitoring equipments for nuclear **reactors**, more particularly for the CEA and EDF. MG wants to allow the company to improve the availability of its equipments giving it a repair computerized tool easy to use. The example of application chosen by MG is one of its most complex equipments, so most difficult to repair: the SPIN (Digital Integrated Protection **System** of the **reactors** PWR 1300 MW). The present study deals more particularly with the computerized part of the SPIN.

DOCUMENT NUMBER = INI17:090358

TI: Knowledge-based **expert systems** for **power** engineering.
AU: Brodsky, S.J.; Tyle, N. (Dept. of Electrical Engineering, Carnegie- Mellon Univ., Pittsburgh, PA 15213).
LA: English
MS: Modelling and simulation. Casetti, E.; Vogt, W.G.; Mickle, M.H.
IM: Research Triangle Park, NC (USA). Instrument Society of America. 1984. p. 1535-1540. ISBN 0-87664-830-8.
CF: (Modeling and simulation conference. Pittsburgh, PA (USA). 19-20 Apr 1984.)
AB: This paper presents a brief review of the development and application of **expert systems** in areas related to electric **power** engineering. The specific examples discussed here include nuclear **power** plant monitoring, **power system** restoration and hydro-electric plant design. In addition, several problems are examined as candidates for future **expert systems**.

DOCUMENT NUMBER = INI17:083411

TI: Demonstration of **expert systems** in automated monitoring.
AU: Otaduy, P.J. (Oak Ridge National Lab., TN).
LA: English
RP: CONF-851115--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (Nov 1985). v. 50 p. 298-299.
CF: (American Nuclear Society winter meeting. San Francisco, CA (USA). 10-15 Nov 1985.)
AB: The Reactor **Systems** Section of Oak Ridge National Laboratory's Instrumentation and Controls Division has been developing expertise in the application of **artificial intelligence** (AI) tools and techniques to control complex **systems**. One of the applications developed demonstrates the capabilities of a rule-based **expert system**

to monitor a nuclear reactor. **Based** on the experience acquired with the demonstration described in this paper, a 2-yr program was initiated during fiscal year 1985 for the development and implementation of an intelligent monitoring adviser to the operators of the HFIR facility. The intelligent monitoring **system** will act as an alert and cooperative **expert** to relieve the operators of routine tasks, request their attention when abnormalities are detected, and provide them with interactive diagnostic aid and project action/effects information as needed or on demand.

DOCUMENT NUMBER = INI17:083407

TI: CLEO: a knowledge-based refueling assistant at FFTF.
AU: Smith, D.E.; Kocher, L.F.; Seeman, S.E. (Westinghouse Hanford Co., Richland, WA).
LA: English
RP: CONF-851115--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (Nov 1985). v. 50 p. 292-293.
CF: (American Nuclear Society winter meeting. San Francisco, CA (USA). 10-15 Nov 1985.)
AB: A computer software **system**, CLEO, is used to assist in the planning and performance of the reactor refueling operations at the Fast Flux Test Facility (FFTF). It is a recently developed application of **artificial intelligence** software with both **expert systems** and automated reasoning aspects. CLEO, an acronym for Cloned LEO, is a logic-based computer program written in Pascal. It imitates the processes that the refueling **expert** for FFTF performs in organizing the refueling of FFTF. The computer assistant seeks to organize the sequence of core component movements according to the rules and logic used by the **expert**. In this form, CLEO has aspects that tie it to both the **expert systems** and automated reasoning areas within the **artificial intelligence** field.

DOCUMENT NUMBER = INI17:083199

TI: Design of an **artificial intelligence system** for safety function maintenance.
AU: Sharma, D.D.; Miller, D.W.; Chandrasekaran, B. (Ohio State Univ., Columbus).
LA: English
RP: CONF-851115--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (Nov 1985). v. 50 p. 294-297.
CF: (American Nuclear Society winter meeting. San Francisco, CA (USA). 10-15 Nov 1985.)
AB: The safety function (SF) maintenance concept provides a **systematic** approach to mitigate the consequences of an unforeseen event. Safety functions are a set of actions for mitigating or limiting consequences of a safety threatening event. The current approach to SF maintenance of selecting a success path (SP) from a library of predefined SPs is inadequate because it includes only anticipated modes of challenging an SF. To cover all possible modes of challenging an SF, the library of success paths would be extremely large and difficult to implement on any existing computer. In this paper the authors describe a method **based** on **artificial intelligence** (AI) theory of planning to synthesize an SP using available resources to satisfy a hierarchy of safety goals. The method has been applied to SF maintenance of a boiling water reactor (BWR) using data from the Perry nuclear **power** plant.

DOCUMENT NUMBER = INI17:083144

TI: Rule-based emergency action level monitor prototype.
AU: Touchton, R.A.; Gunter, A.D.; Cain, D. (Technology Applications, Inc., Falls Church, VA).
LA: English
RP: CONF-851115--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (Nov 1985). v. 50 p. 297-298.
CF: (American Nuclear Society winter meeting. San Francisco, CA (USA). 10-15 Nov 1985.)
AB: In late 1983, the Electric **Power** Research Institute (EPRI) began a program to encourage and stimulate the development of **artificial intelligence** (AI) applications

for the nuclear industry. Development of a rule-based emergency action level classification **system** prototype is discussed. The paper describes both the full prototype currently under development and the completed, simplified prototype.

DOCUMENT NUMBER = INI17:083128

TI: Logic programming for operational analysis of the Savannah River **reactors**.
AU: Suich, J.E. (California Institute of Technology, Pasadena).
LA: English
RP: CONF-851115--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (Nov 1985). v. 50 p. 293-294.
CF: (American Nuclear Society winter meeting. San Francisco, CA (USA). 10-15 Nov 1985.)
AB: The Savannah River Plant (SRP) has installed an on-line reactor monitoring diagnostic computer **system** to direct operators to appropriate procedures for multiple, concurrent alarms. Off-line programs analyze the monitoring logic for consistency. Both applications can be described as logic programs. Emerging **artificial intelligence** (AI) computer technology promises operationally viable **expert systems** for diagnosis and control of reactor **systems**. The core of this technology is the use of computers for logical inference, in contrast to the more traditional number crunching and data processing functions. At present, the diagnostic logic represents surface **knowledge** of reactor operating procedures, while the analytical logic incorporates a small amount of real-world **knowledge** concerning superset/subset relationships among alarms. The challenge for the future is to create a comprehensive operations advisory **system**, with logic as engineering rules of thumb for cause/effect models of malfunction and corrective action, data reflecting the real-time status of all critical reactor components, and control working both bottom up from alarms to malfunctions, and top down from malfunction to corrective procedure.

DOCUMENT NUMBER = INI17:083078

TI: Reviewing the development of an **artificial intelligence based** risk program.
AU: Dixon, B.W.; Hinton, M.F. (EG and G Idaho, Idaho Falls).
LA: English
RP: CONF-851115--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (Nov 1985). v. 50 p. 291-292.
CF: (American Nuclear Society winter meeting. San Francisco, CA (USA). 10-15 Nov 1985.)
AB: A successful application of nonconventional programming methods has been achieved in computer-assisted probabilistic risk assessment (PRA). The event tree sequence importance calculator, SQUIMP, provides for prompted data entry, generic expansion, on-line pruning, boolean reductions, and importance factor selection. SQUIMP employs constructs typically found in **artificial intelligence** (AI) programs. The development history of SQUIMP is outlined and its internal structure described as background for a discussion on the applicability of symbolic programming methods in PRA.

DOCUMENT NUMBER = INI17:078531

TI: Report on nuclear **power** plant control and instrumentation activities in the Federal Republic of Germany.
AU: Bastl, W. (Gesellschaft fuer Reaktorsicherheit m.b.H. (GRS), Garching (Germany, F.R.)).
LA: English
MS: Tenth meeting of the International Working Group on Nuclear **Power** Plant Control and Instrumentation, Vienna, 3-5 March 1986. Summary report. International Atomic Energy Agency, Vienna (Austria). International Working Group on Nuclear **Power** Plant Control and Instrumentation.
RP: IWG-NPPCI--86/1.
IM: Jul 1986. 74 p. p. 57-64. Availability: INIS.
NO: 2 figs, 1 tab.
CF: (10. meeting of the International Working Group on Nuclear **Power** Plant Control and Instrumentation. Vienna (Austria). 3-5 Mar 1986.)

AB: The overall situation in I and C of nuclear **power plants** in the Federal Republic of Germany can be characterized by three aspects. a) The improvement of man-machine communication by introducing integral information concepts for the control room by means of VDUs. b) Along with a) new data acquisition **systems based** upon process computers which facilitate the integration of operator aids like alarm analyses, disturbance analyses, post-mortem analyses, etc. c) The penetration of programmable processors into limitation **systems** in order to provide soft setback measures. d) The transition to I and C **systems** making use of the new generation of electronic components. The most important step towards advanced control rooms was the development of the Process Information **System** (PRINS) by KWU, which will be used with the German convoi-plants. The main emphasis regarding further R and D work in the field of operator aids is placed upon **expert systems.** Work will begin with a two years project aiming at the development of a basic module for a laboratory prototype.

DOCUMENT NUMBER = INI17:078527

TI: NPPCI activities in Finland.
AU: Wahlstroem, B. (Valtion Teknillinen Tutkimuskeskus, Espoo (Finland)).
LA: English
MS: Tenth meeting of the International Working Group on Nuclear **Power** Plant Control and Instrumentation, Vienna, 3-5 March 1986. Summary report. International Atomic Energy Agency, Vienna (Austria). International Working Group on Nuclear **Power** Plant Control and Instrumentation.
RP: IWG-NPPCI--86/1.
IM: Jul 1986. 74 p. p. 44-45. Availability: INIS.
NO: 2 refs, 1 tab.
CF: (10. meeting of the International Working Group on Nuclear **Power** Plant Control and Instrumentation. Vienna (Austria). 3-5 Mar 1986.)
AB: Research activities at the Technical Research Centre of Finland in the control and instrumentation field have been directed both towards immediate needs and towards long term goals. The immediate needs have been connected to different ongoing investigations and to development projects in connection with process industry. Of the more long term projects the following can be explicitly mentioned: the Nordic project on human reliability; work on software reliability; design guides for digital CI **systems**; simulation of technical processes; **artificial intelligence** in nuclear **power.**

DOCUMENT NUMBER = INI17:078514

TI: Paradigm for **expert** display **systems** in nuclear plant and elsewhere.
AU: Gabriel, J.R.
CO: Argonne National Lab., IL (USA).
LA: English
RP: ANL/MCS-TM--64.
IM: Feb 1986. 5 p. Availability: INIS. Available from NTIS, PC A02/ MF A01 as DE86008475.
AB: Display of relevant data concerning plant operation has been a concern of the nuclear industry from its beginnings. Since the incident at Three Mile Island, this matter has had much careful scrutiny. L. Beltracchi, in particular, has originated a sequence of important steps to improve the operator's ability to recognize plant states and their changes. In the early 1980's, Beltracchi (1983, 1984) proposed a display **based** on the Rankine cycle for light water **reactors.** More recently, in an unpublished work (1986b), he described an extension that includes a small, rule-based **system** in the display program, drawing inferences about plant operation from sensor readings, and displaying those inferences on the Rankine display. Our paper examines Beltracchi's rule-based display from the perspective of **knowledge** bases. Earlier (Gabriel, 1983) we noted that analytical models of **system** behavior are just as much a **knowledge** base as are the rules of a conventional **expert system.** The problem of finding useful displays for a complex plant is discussed from this perspective. We then present a paradigm for developing designs with properties similar to those in Beltracchi's Rankine cycle display. Finally, to clarify the issue, we give a small example from an imaginary plant.

DOCUMENT NUMBER = INI17:073808

TI: **Artificial intelligence** applications to the surveillance and diagnostics of nuclear **power plants.**
AU: Zwingelstein, G.; Monnier, B. (Electricite de France).
LA: English
RP: CONF-851115--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (Nov 1985). v. 50 p. 515-516.
CF: (American Nuclear Society winter meeting. San Francisco, CA (USA). 10-15 Nov 1985.)
AB: **Artificial intelligence** has come of age; it can be applied with benefits in the nuclear industry for various applications to improve both plant safety and availability. This paper presents four rule-based **expert systems** currently under development at Electricite de France for the surveillance and the diagnostics of pressurized water **reactors** (PWRs). The objectives of these experiences with **expert system** projects are to demonstrate their feasibility, to provide an evaluation of their potential benefits through on-site implementation, to identify the problems due to the **knowledge** extraction and representation, and finally to specify the classes of problems where **expert systems** are needed and useful. The specific goals of these **expert systems** are to provide an aid for the detection and location of failures occurring in nuclear **plants.**

DOCUMENT NUMBER = INI17:073547

TI: Application of **Artificial Intelligence** to operator assistance.
AU: Wildberger, A.M. (General Physics Corporation, 10650 Hickory Ridge Road, Columbia, MD).
LA: English
MS: New technology in nuclear **power** plant instrumentation and control. Anon.
IM: Research Triangle Park, NC (USA). Instrument Society of America. 1984. p. 295-301. ISBN 0-87664-868-5.
CF: (Symposium on new technologies in nuclear **power** plant instrumentation and control. Washington, DC (USA). 28-30 Nov 1984.)
AB: This paper describes an application of **Artificial Intelligence** to nuclear **power** plant control. An **expert system** is proposed in which the experience of NRC certified instructors, as represented in a **knowledge** base by a series of production rules, is used to recommend control sequences to the operator **based** on the state of the plant at the time.

DOCUMENT NUMBER = INI17:073537

TI: **Knowledge based** development of graphic display **systems.**
AU: Woods, D.D. (Westinghouse Research and Development Center, Pittsburgh, PA 15235).
LA: English
MS: Conference record for 1985 IEEE third conference on human factors and nuclear safety. Hagen, E.W.
IM: Piscataway, NJ (USA). IEEE Service Center. 1985. p. 184-186.
CF: (3. IEEE conference on human factors and **power plants.** Monterey, CA (USA). 23-27 Jun 1985.)
AB: Most human factors guidelines on computer display design attempt to ensure that human sensory limits are not strained or that users can potentially access data. While this is a necessary and too often an overlooked step in interface design, it does not address issues about how to use computer displays to aid human performance at domain tasks. This challenge to human factors guidance is particularly acute in complex environments such as nuclear **power plants** where the goal is more than a useable interface **system;** the interface must support effective human performance at tasks like situation assessment, fault management, problem solving and planning. Studies of human performance in complex domains reveal that human-machine performance failures can often be linked to problems in information handling - data overload, getting lost, keyhole effects, tunnel vision to name but a few. All of these information handling problems represent manifestations of an inability to find, integrate or interpret the "right" data at the "right" time, i.e., failures where critical information is not detected among the ambient data load,

where critical information is not assembled from data distributed over time or over space; and where critical information is not looked for because of misunderstandings or erroneous assumptions (cf., Woods, 1985). Problems of this kind illustrate that the potential to see, read, or access data does not guarantee successful user information extraction. The result is a need for research and guidance on design for enhanced information extraction in addition to design for data availability.

DOCUMENT NUMBER = INI17:073536

TI: Quantitative methods for judging display design quality.
AU: Danchak, M.M. (The Hartford Graduate Center, 275 Windsor Street, Hartford, CT 06120).
LA: English
MS: Conference record for 1985 IEEE third conference on human factors and nuclear safety. Hagen, E.W.
IM: Piscataway, NJ (USA). IEEE Service Center. 1985. p. 178-183.
CF: (3. IEEE conference on human factors and **power plants**. Monterey, CA (USA). 23-27 Jun 1985.)
AB: Since the Three Mile Island incident, a considerable amount of effort was expended in developing display evaluation methods. Techniques that provided quantitative results are expensive and complex. Simpler techniques, such-as checklists, only give qualitative guidance. Using methods recently developed by Tullis, one can now successfully quantify display characteristics such as overall density, local density, grouping and layout complexity. These are the first tools available in what may someday be an **expert system** to evaluate display quality.

DOCUMENT NUMBER = INI17:070318

TI: Annual report 1985 - IFE. (Institute for Energy Technology, Kjeller, Norway.) Aarsberetning 1985 - IFE.
CO: Institute for Energy Technology, Kjeller (Norway).
LA: Norwegian
RP: INIS-mf--10502.
IM: nd . 29 p. Availability: INIS.
NO: Summary in English.
AB: At present Norwegian nuclear energy research is centered around the international OECD Halden Reactor Project which has participants from 40 organisations in 10 countries, including the Nordic nations and most major nuclear **power** countries within OECD. The research programme is concentrated on nuclear fuel and safety technology and on computer-based methods for operation and supervision of **power reactors.** Fuel research is now being concentrated more and more on characterizing long-term effects with regard to the efficiency and reliability of the fuel. Special instruments developed in the Halden project are essential for implementing the fuel testing programme. These instruments provide detailed information concerning external and internal conditions in the fuel rods. In turn such data provide a basis for developing calculation models for use in design, for licensing and when determining operating strategies for the different types of fuel. Work to develop a large computer-based **system** for assisting operators under irregular operating conditions was commenced in 1985. This **system** will consist of several modules that help trace and diagnose disturbances, and it contains procedures for returning the **system** to an acceptable state. A knowledge-based **system** making use of modern techniques for storing alarm sequences and a series of important parameters for the various operating faults, is being developed to help in diagnosing and recognizing fault situations. The nuclear supervision **system** SCORPIO has been supplied for all three pressurized water **reactors** in the Swedish nuclear **power** plant at Ringhals. Operation of the three sets of plant can now be planned from one central workstation. Cooperation with Ringhals is continuing with a view to connecting SCORPIO direct to the plant, so that the operating margins can be monitored in true time at a detailed level. Other signatory nations are displaying an interest in this monitoring **system**, especially West Germany and USA.

DOCUMENT NUMBER = INI17:069695

TI: Object-oriented alarm-filtering **system**.
AU: Corsberg, D.R.; Wilkie, D.
CO: Idaho National Engineering Lab., Idaho Falls (USA).
LA: English
RP: EGG-M--20485. CONF-860414--9.
IM: 1986. 15 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE86009078.
NO: Portions of this document are illegible in microfiche products.
CF: (6. **power** plant dynamics, control and testing symposium. Knoxville, TN (USA). 14-16 Apr 1986.)
AB: This paper discusses an alarm-filtering **system** (AFS) being developed by EG and G Idaho, Inc. for the Advanced Test Reactor (ATR) at the Idaho National Engineering Laboratory. The ultimate goal of this project is to place AFS into ATR's reactor control room to act as an aid during major plant transients. In addition, methods of alarm analysis are investigated **based** on functional relationships rather than on a historical approach utilizing cause-consequence trees. **Artificial intelligence** techniques, including object-oriented programming, are also demonstrated as useful in analyzing alarms and alarm sequences. After a brief description of the problem AFS addresses, this paper discusses the design constraints and human factors that influenced the development of the **system**. The reader is then presented with operational and architectural descriptions of the **system** as well as what directions the future development of AFS may take. The fact that AFS is being considered as a partial solution to the problems discussed in the next section demonstrates the viability of its underlying technology and approach. 10 refs.

DOCUMENT NUMBER = INI17:069176

TI: Microelectronics in LMFBR: Activities sponsored by the Commission 1979-1983.
AU: Nordwall, H.J. de (Commission of the European Communities, Brussels (Belgium)).
LA: English
MS: Use of digital computing devices in **systems** important to safety. Specialists' meeting held at Centre d'Etudes Nucleaires de Saclay - Centrale Nucleaire de PALUEL 28-30 November 1984. International Atomic Energy Agency, Vienna (Austria). International Working Group on Nuclear **Power** Plant Control and Instrumentation; CEA, 75 - Paris (France); Electricite de France, 75 - Paris.
RP: INIS-mf--10499.
IM: Jan 1986. 481 p. p. 457-471. Availability: INIS.
NO: 16 refs, 1 fig.
CF: (Specialists' meeting on use of digital computing devices in **systems** important to safety. Saclay (France). 28-30 Nov 1984.)
AB: These studies were designed to draw attention to signal handling techniques which have become applicable to the automatic control of **reactors** as a result of advances in microelectronics. The underlying concepts being explored are the extent to which more information may be obtained from existing sensors, whether changes in the state of a **system** may be acted upon automatically and the acceptability of stochastic methods of state estimation. If a given state may arise in a number of different ways, automatic action necessarily involves diagnosis of causes if any but the simplest actions, e.g. scram, is required. Up to now this diagnosis has been made by an operator, though a number of schemes intended to identify the primary cause of an abnormal state or to guide an operator to the most easily accessible safe state are in advanced states of development, e.g. at Halden, Westinghouse and GRS-Garching. The activity decribed is continuing, current emphasis being upon diagnostics using approaches **based** upon **artificial intelligence** concepts, **system** optimisation and quantification of the benefits of networking.

DOCUMENT NUMBER = INI17:069139

TI: Westinghouse use of **artificial intelligence** in signal interpretation.
AU: Mark, R.H. (Westinghouse Electric Corp., Pittsburgh, PA).
LA: English
MS: Technology and the world around us. Tuba, S.

IM: Pittsburgh, PA (USA). International Technology Institute. 1984. p. 70-71.
CF: (International congress on technology and technology exchange. Pittsburgh, PA (USA). 8-10 Oct 1984.)
AB: This paper discusses Westinghouse's use of **artificial intelligence** to assist inspectors who routinely monitor the thousands of tubes in nuclear steam generators. Using the AI technology has made the inspection process easier to learn and to apply. The **system** uses pattern recognition to identify off-normal conditions. As part of the in-service inspection program for nuclear **power reactors**, utilities make a practice of inspecting the condition of the large heat exchangers that produce the steam that turns the electric turbine generator. The same data are presented for inspection using form, motion, and color to call attention to off-normal signal patterns.

DOCUMENT NUMBER = INI17:065445

TI: Collection, processing and use of data.
AU: Mancini, G. (Commission of the European Communities, Ispra (Italy). Joint Research Centre).
LA: English
JR: Nucl. Eng. Des. ISSN 0029-5493. CODEN: NEDEA. (May 1986). v. 93(2/3) p. 181-186.
CF: (1. international seminar on the role of data and judgement in probabilistic risk analysis in conjunction with the 8. international conference on structural mechanics in reactor technology (SMIRT-8). Brussels (Belgium). 26-27 Aug 1985.)
AB: The proliferation in the past few years of data banks related to safety and reliability of nuclear **reactors** has somewhat alleviated the problem of lack of information when carrying out safety analysis; but it has succeeded meanwhile to identify several crucial aspects in the overall procedures of collection, processing and use of data. These aspects are reviewed in the paper with special emphasis on the interaction of the information and the various interpretative models of component and **system** behaviour. The need for the analyst of not only validating stated models but also of attempting new interpretations of the facts is particularly stressed. Eventually the role of new **Artificial Intelligence** techniques in supporting man in his search for model structures - that is the "history" underlying the facts, is mentioned. (orig.).

DOCUMENT NUMBER = INI17:065433

TI: Developing a **knowledge** base for the management of severe accidents.
AU: Nelson, W.R.; Jenkins, J.P.
CO: EG and G Idaho, Inc., Idaho Falls (USA); Nuclear Regulatory Commission, Washington, DC (USA).
LA: English
RP: EGG-M--32485. CONF-860415--7.
IM: 1986. 6 p. Availability: INIS. Available from NTIS, PC A02/MF A01 - GPO as TI86007634.
CF: (International topical meeting on advances in human factors in nuclear **power systems.** Knoxville, TN (USA). 21-24 Apr 1986.)
AB: Prior to the accident at Three Mile Island, little attention was given to the development of procedures for the management of severe accidents, that is, accidents in which the reactor core is damaged. Since TMI, however, significant effort has been devoted to developing strategies for severe accident management. At the same time, the potential application of **artificial intelligence** techniques, particularly **expert systems,** to complex decision-making tasks such as accident diagnosis and response has received considerable attention. The need to develop strategies for accident management suggests that a computerized **knowledge** base such as used by an **expert system** could be developed to collect and organize **knowledge** for severe accident management. This paper suggests a general method which could be used to develop such a **knowledge** base, and how it could be used to enhance accident management capabilities.

DOCUMENT NUMBER = INI17:062686

TI: An **artificial intelligence** approach towards disturbance analysis.
AU: Fiedler, U.; Lindner, A.; Baldeweg, F.; Klebau, J. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)).
LA: English
JR: Kernenergie. ISSN 0023-0642. CODEN: KERNA. (Aug 1986). v. 29(8) p. 302-307.
AB: Scale and degree of sophistication of technological **plants**, e.g. nuclear **power plants**, have been essentially increased during the last decades. Conventional disturbance analysis **systems** have proved to work successfully in well-known situations. But in cases of emergencies, the operator needs more advanced assistance in realizing diagnosis and therapy control. The significance of introducing **artificial intelligence** (AI) methods in nuclear **power** technology is emphasized. Main features of the on-line disturbance analysis **system** SAAP-2 are reported about. It is being developed for application to nuclear **power plants**. Problems related to man-machine communication will be gone into more detail, because their solution will influence end-user acceptance considerably. (author).

DOCUMENT NUMBER = INI17:055334

TI: Development of a tutorial **system** for plant preventive maintenance **based** on the **expert systems**.
AU: Terano, Takao (Central Research Inst. of Electric **Power** Industry, Tokyo (Japan)).
LA: Japanese
JR: Joho Shori Kenkyu. ISSN 0388-5038. CODEN: JSKED. (Mar 1985). (no.13) p. 19-36.
AB: A pilot **system** of preventive maintenance of nuclear **power plants** is being developed **based** on the **expert systems**. This report describes basic ideas underlying **the system** development together with the present status of the **expert systems** technology. Fundamental functions of an ideal preventive maintenance **systems** are explained together with some basic requirements. This **system** will be used for supporting the check and review of the initial design and the following design changes. And it is assumed to be used off-line and to have the function of consultation. Logical structure of the **system** and its component subsystems are explained in some detail. These include; (1) **knowledge** basis, (2) deduction mechanism, (3) input support subsystem for constructing **knowledge** basis, (4) **knowledge** basis compatibility check subsystem, and (5) interfaces between users. (Aoki, K.).

DOCUMENT NUMBER = INI17:055299

TI: The future of **artificial intelligence** in nuclear plant maintenance.
AU: Norgate, G. (Spar Aerospace, Ltd., Remote Manipulator **Systems** Div., 1700 Ormont Drive, Weston, Ontario).
LA: English
MS: International conference on robotics and remote handling in the nuclear industry. Anon.
IM: LaGrange Park, IL (USA). American Nuclear Society. 1984. p. 13- 20.
CF: (Robotics and remote handling in the nuclear industry conference. Toronto, Ontario (Canada). 23-27 Sep 1984.)
AB: Robots with vision and force sensing capability, performing tasks under computer control, will offer new opportunities to reduce human exposure to radiation. Such machines do not yet exist and even simple maintenance tasks challenge current robot technology. Recently increased priority for research on **artificial intelligence** and fifth generation computer technology is likely to bring useful maintenance robots closer to reality.

DOCUMENT NUMBER = INI17:045607

TI: Future development and limits of computer aided man-machine- communication. Zu zukuenftigen Entwicklungen und Grenzen der computergestuetzten Mensch-Prozess-Kommunikation.
AU: Grimm, R. (Fraunhofer-Institut fuer Informations- und Datenverarbeitung (IITB), Karlsruhe, Germany).

LA: German
MS: Man-Machine communication in nuclear **power plants** and other complex technical **systems.** SVA Further Education Course. Mensch-Maschinen- Kommunikation in Kernkraftwerken und anderen komplexen technischen **Systemen.** SVA-Vertiefungskurs. Schweizerische Vereinigung fuer Atomenergie, Bern.
IM: Bern (Switzerland). SVA. 1985. 319 p. p. R13-1-R13-23.
CF: (Man-Machine communication in nuclear **power plants** and other complex technical **systems.** Brugg-Windisch (Switzerland). 11-13 Nov 1985.)
AB: An overview of development trends in man-machine-communication is given. Themes such as real-time databases, vocal in- and outputs, integrated information **systems** and **artificial intelligence** are treated.

DOCUMENT NUMBER = INI17:045006

TI: Neutron noise characterization by a pattern recognition method. (PWR regulating rod motion detection.) Caracterisation du bruit neutronique par une methode de reconnaissance de formes.
AU: Invernizzi, M.; Kaoutar, M.
CO: CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Inst. de Recherche Technologique et de Developpement Industriel (IRDI).
LA: French
RP: CEA-CONF--8185.
IM: Nov 1985. 15 p. Availability: INIS.
CF: (Congress on **artificial intelligence** and pattern recognition. Grenoble (France). 25-29 Nov 1985.)
AB: In this paper, a piecewise approximation algorithm is proposed. This method is **based** on a dynamic clustering approach and has been compared with two other ones. One presents an application to control rods movements detection and characterization in a pressurized water reactor.

DOCUMENT NUMBER = INI17:040334

TI: Application of noise analysis methods in nuclear reactor diagnostics. Vyuziti metod sumove analyzy pro diagnostiku jadernych reaktoru.
AU: Dach, K.
LA: Czech
JR: Jad. Energ. ISSN 0448-116X. CODEN: JADEA. (Aug-Sep 1985). v. 31(8-9) p. 330-333.
AB: By statistical evaluation of the fluctuation component of signals from selected detectors, noise diagnostics detects conditions of equipment which might later result in failure. The objective of early diagnostics is to detect the failed integrity of primary circuit components, failed detectors or anomalies of the thermohydraulic process. The commonest method of experimental data analysis is spectral analysis in the frequency range 0 to 50 Hz. Recently, **expert** diagnostic **systems** have been built **based** on **artificial intelligence systems.** Czechoslovakia participates in the experimental research of noise diagnostics in the context of the development of diagnostic assemblies for WWER-440 **reactors.** (M.D.).

DOCUMENT NUMBER = INI17:032924

TI: **Artificial intelligence** program in a computer application supporting reactor operations.
AU: Stratton, R.C.; Town, G.G.
CO: Argonne National Lab., Idaho Falls, ID (USA).
LA: English
RP: CONF-850903--18.
IM: 1985. 14 p. Availability: INIS. Available from NTIS, PC A02; 3 as DE85018372.
CF: (International topical meeting on computer applications for nuclear **power** plant operation and control. Pasco, WA (USA). 8-12 Sep 1985.)
AB: Improving nuclear reactor **power** plant operability is an ever-present concern for the nuclear industry. The definition of plant operability involves a complex interaction of the ideas of reliability, safety, and efficiency. This paper presents

observations concerning the issues involved and the benefits derived from the implementation of a computer application which combines traditional computer applications with **artificial intelligence** (AI) methodologies. A **system**, the Component Configuration Control **System** (CCCS), is being installed to support nuclear reactor operations at the Experimental Breeder Reactor II.

DOCUMENT NUMBER = INI17:032554

TI: Plant experience with an **expert system** for alarm diagnosis.
AU: Gimmy, K.L.
CO: Du Pont de Nemours (E.I.) and Co., Aiken, SC (USA). Savannah River Lab.
LA: English
RP: DP-MS--85-63. CONF-860119--3.
IM: 1986. 4 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE86001966.
CF: (Society for Computer Simulation (SCS) multiconference. San Diego, CA (USA). 23-25 Jan 1986.)
AB: An **expert system** called Diagnosis of Multiple Alarms (DMA) is in routine use at four nuclear **reactors** operated by the DuPont Company. The **system** is wired to plant alarm annunciators and does event-tree analysis to see if a pattern exists. Any diagnosis is displayed to the plant operator and the corrective procedure to be followed is also identified. The display is automatically superseded if a higher priority diagnosis is made. The **system** is integrated with operator training and procedures. Operating results have been positive. DMA has diagnosed several hard-to-locate small leaks. There have been some false diagnosis, and realistic plant environments must be considered in such **expert systems.** 2 refs., 5 figs.

DOCUMENT NUMBER = INI17:025630

TI: Component Configuration Control **System:** an application of logic programming.
AU: Stratton, R.C.; Town, G.G.
CO: Argonne National Lab., IL (USA); Central Washington Univ., Ellensburg (USA). Dept. of Computer Science.
LA: English
RP: CONF-8510161--1.
IM: 1985. 9 p. Availability: INIS. Available from NTIS, PC A02/MF A01 as DE85018455.
CF: (**Artificial intelligence** in engineering conference. Washington, DC (USA). 21-23 Oct 1985.)
AB: A computer application **system** is described which provides nuclear reactor **power** plant operators with an improved decision support **system.** This **system** combines traditional computer applications such as graphics display with **artificial intelligence** methodologies such as reasoning and diagnosis so as to improve plant operability. This paper discusses the issues, and a solution, involved with the **system** integration of applications developed using traditional and **artificial intelligence** languages.

DOCUMENT NUMBER = INI17:025375

TI: **Artificial intelligence** and concurrent computation for robotic applications.
AU: Weisbin, C.R.; de Saussure, G.; Hamel, W.R.; Jorgensen, C.; Lucius, J.L.; Oblow, E.M.; Swift, T. (Oak Ridge National Lab., Oak Ridge, TN).
LA: English
RP: CONF-850610--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1985). v. 49 p. 310-311.
NO: Published in summary form only.
CF: (Annual meeting of the American Nuclear Society. Boston, MA (USA). 9-14 Jun 1985.)

DOCUMENT NUMBER = INI17:023226

TI: Advanced maintenance research programs.
AU: Marston, T.U.; Gelhaus, F.; Burke, R. (Electric **Power** Research Institute, Palo Alto, CA).

LA: English
RP: CONF-850894--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. CODEN: TANSA. (1985). v. 49 Suppl. 2 p. 46-47.
CF: (12. **biennial** conference on reactor operating experience "maintenance and modifications for availability and efficiency". Williamsburg, VA (USA). 4-7 Aug 1985.)
AB: The purpose of this paper is to provide the reader with an idea of the advanced maintenance research program at the Electric **Power** Research Institute (EPRI). A brief description of the maintenance-related activities is provided as a foundation for the advanced maintenance research projects. The projects can be divided into maintenance planning, preventive maintenance program development and implementation, predictive (or conditional) maintenance, and innovative maintenance techniques. The projects include hardware and software development, human factors considerations, and technology promotion and implementation. The advanced concepts include: the incorporation of **artificial intelligence** into outage planning; turbine and pump maintenance; rotating equipment monitoring and diagnostics with the aid of **expert systems**; and the development of mobile robots for nuclear **power** plant maintenance.

DOCUMENT NUMBER = INI17:022924

TI: A **knowledge based system** for plant diagnosis.
AU: Motoda, H.; Yamada, N.; Yoshida, K. (Energy Research Lab., Hitachi Ltd., Ibaraki 316).
LA: English
MS: Fifth Generation Computer **Systems** - 1984. Anon.
IM: New York, NY (USA). Elsevier. 1984. p. 582-588. ISBN-0-444-87673- 1.
CF: (International conference on fifth generation computer **systems**. Tokyo (Japan). 6-9 Nov 1984.)
AB: A **knowledge based system** for plant diagnosis is proposed in which both event-oriented and function-oriented **knowledge** are used. For the proposed **system** to be of practical use, these two types of **knowledge** are represented by mutually nested four frames, i.e. the component, causality, criteriality, and simulator frames, and production rules. The **system** provides fast inference capability for use as both a production **system** and a formal reasoning **system**, with uncertainty of **knowledge** taken into account in the former. Event-oriented **knowledge** is used in both diagnosis and guidance and function-oriented **knowledge**, in diagnosis only. The inference capability required is forward chaining in the former and resolution in the latter. The causality frame guides in the use of event-oriented **knowledge**, whereas the criteriality frame does so for function-oriented **knowledge**. Feedback nature of the plant requires the best first search algorithm that uses histories in the resolution process. The inference program is written in Lisp and the plant simulator and the process I/O control programs in Fortran. Fast data transfer between these two languages is realized by enhancing the memory management capability of Lisp to control the numerical data in the global memory. Simulation applications to a BWR plant demonstrated its diagnostic capability.

DOCUMENT NUMBER = INI17:019625

TI: STAR-GENERIS - a software package for information processing. Concept and application. STAR GENERIS Softwarepaket zur Informationsaufbereitung. Konzept und Anwendungen.
AU: Felkel, L.
CO: Gesellschaft fuer Reaktorsicherheit m.b.H. (GRS), Koeln (Germany, F.R.).
LA: German
RP: INIS-mf--9870.
IM: 1985. 31 p. Availability: INIS.
CF: (9. GRS technical meeting on trends in nuclear **power** plant control technology. Munich (Germany, F.R.). 7-8 Nov 1985.)
AB: Man-machine-communication in electrical **power plants** is increasingly **based** on the capabilities of minicomputers. Rather than just displaying raw process data more complex processing is done to aid operators by improving information quality. Advanced operator aids for nuclear **power plants** are, e.g. alarm reduction, disturbance analysis and **expert systems**. Operator aids use complex combinations and

computations of plant signals, which have to be described in a formal and homogeneous way. The design of such computer-based information **systems** requires extensive software and engineering efforts. The STAR software concept reduces the software effort to a minimum by proving an advanced program package which facilitates specification and implementation of engineering know-how necessary for sophisticated operator aids. (orig./HP).

DOCUMENT NUMBER = INI17:017452

TI: Application of **expert systems** to heat exchanger control at the 100- megawatt high-flux isotope reactor.
AU: Clapp, N.E. Jr.; Clark, F.H.; Mullens, J.A.; Otaduy, P.J.; Wehe, D.K.
CO: Oak Ridge National Lab., TN (USA).
LA: English
RP: CONF-8509131--1.
IM: 1985. 5 p. Availability: INIS. Available from NTIS, PC A02/MF A01; 1 as DE85017084.
CF: (International industrial controls conference and exposition. Long Beach, CA (USA). 16-18 Sep 1985.)
AB: The High-Flux Isotope Reactor (HFIR) is a 100-MW pressurized water reactor at the Oak Ridge National Laboratory. It is used to produce isotopes and as a source of high neutron flux for research. Three heat exchangers are used to remove heat from the reactor to the cooling towers. A fourth heat exchanger is available as a spare in case one of the operating heat exchangers malfunctions. It is desirable to maintain the reactor at full **power** while replacing the failed heat exchanger with the spare. The existing procedures used by the operators form the initial **knowledge** base for design of an **expert system** to perform the switchover. To verify performance of the **expert system**, a dynamic simulation of the **system** was developed in the MACLISP programming language. 2 refs., 3 figs.

DOCUMENT NUMBER = INI17:017221

TI: Combination of **artificial intelligence** and procedural language programs in a computer application **system** supporting nuclear reactor operations.
AU: Town, G.G.; Stratton, R.C.
CO: Argonne National Lab., Idaho Falls, ID (USA); Central Washington Univ., Ellensburg (USA). Dept. of Computer Science.
LA: English
RP: CONF-850903--2.
IM: 1985. 7 p. Availability: INIS. Available from NTIS, PC A02; 3 as DE85007937.
CF: (International topical meeting on computer applications for nuclear **power** plant operation and control. Pasco, WA (USA). 8-12 Sep 1985.)
AB: A computer application **system** is described which provides nuclear reactor **power** plant operators with an improved decision support **system**. This **system** combines traditional computer applications such as graphics display with **artificial intelligence** methodologies such as reasoning and diagnosis so as to improve plant operability. This paper discusses the issues, and a solution, involved with the **system** integration of applications developed using traditional and **artificial intelligence** languages.

DOCUMENT NUMBER = INI17:017192

TI: **Artificial intelligence**: the future in nuclear plant maintenance.
AU: Norgate, G. (Spar Aerospace Ltd., Weston, Ontario).
LA: English
RP: CONF-840916--.
JR: Nucl. News (La Grange Park, Ill.). ISSN 0029-5574. CODEN: NUNWA. (Dec 1984). v. 27(15) p. 57-61.
CF: (Robotics and remote handling in the nuclear industry conference. Toronto, Ontario (Canada). 23-27 Sep 1984.)
AB: The role of robotics and remote handling equipment in future nuclear **power** plant maintenance activities is discussed in the context of **artificial intelligence** applications. Special requirements manipulators, control **systems**, and man-machine

interfaces for nuclear applications are noted. Tasks might include inspection with cameras, eddy current probes, and leak detectors; the collection of material samples; radiation monitoring; and the disassembly, repair and reassembly of a variety of **system** components. A robot with vision and force sensing and an intelligent control **system** that can access a **knowledge** base is schematically described. Recent advances in image interpretation **systems** are also discussed.

DOCUMENT NUMBER = INI17:013244

TI: Reactor Safety Assessment **System**: a situation assessment aid for USNRC emergency response.
AU: Bray, M.A.; Sebo, D.E.; Dixon, B.W.
CO: EG and G Idaho, Inc., Idaho Falls (USA).
LA: English
RP: EGG-M--11385. CONF-851072--1.
IM: Apr 1985. 21 p. Availability: INIS. Available from NTIS, PC A02/ MF A01 as DE85011958.
CF: (**Expert systems** in government symposium. McLean, VA (USA). 23-25 Oct 1985.)
AB: The Reactor Safety Assessment **System** is an **expert system** under development for the United States Nuclear Regulatory Commission (NRC). RSAS is intended for use at the NRC's Operations Center in the event of a serious incident at a licensed nuclear **power** plant. The **system** uses plant parameter data and status information from the **power** plant. It has a rule base which uses the parametric values, the known operator actions and the time sequence information in the data to generate situation assessment conclusions for use by the NRC Reactor Safety Team. RSAS rules currently cover one specific reactor type and use setpoints specific to one **power** plant. 5 figs.

DOCUMENT NUMBER = INI17:013160

TI: Robots for nuclear **power plants**. In the USA, robotics technology used at the TMI-2 cleanup and at other nuclear **plants** has prompted interest and shaped research on how robots might best be used.
AU: Moore, T. (Electric **Power** Research Inst., Palo Alto, CA (USA)).
LA: English
JR: Int. At. Energy Agency Bull. ISSN 0020-6067. CODEN: IAEBA. (Aut 1985). v. 27(3) p. 31-38.
AB: In many industrial applications of robots, the objective is to replace human workers with machines that are more productive, efficient, and accurate. But for nuclear applications, the objective is not so much to replace workers as it is to extend their presence - for example, to project their reach into areas of a nuclear plant where the thermal or radiation environment prohibits or limits a human presence. The economic motivation to use robots for nuclear plant inspection and maintenance is centered on their potential for improving plant availability; a by-product is the potential for reducing the occupational radiation exposure of plant personnel. Robotic equipment in nuclear applications may be divided into two broad categories: single-purpose devices with limited ability to perform different operations, and reprogrammable, multi-purpose robots with some degree of computer-based **artificial intelligence**. Preliminary assessments of the potential for applying robotics in nuclear **power plants** - mainly at surveillance and inspection tasks - have been carried out. Future developments are considered.

DOCUMENT NUMBER = INI17:010079

TI: Experimental applications of an **expert system** to operator problem solving in process control.
AU: Nelson, W.R.; Jenkins, J.P.
CO: EG and G Idaho, Inc., Idaho Falls (USA); Nuclear Regulatory Commission, Washington, DC (USA).
LA: English
RP: EGG-M--13985. CONF-850836--1.

IM: 1985. 13 p. Availability: INIS. Available from NTIS, PC A02/MF A01 as TI85014596.
CF: (American Institute of Chemical Engineers national meeting. Seattle, WA (USA). 25-28 Aug 1985.)
AB: The United States Nuclear Regulatory Commission (USNRC) has sponsored a program to assess the effectiveness of **expert systems** for nuclear reactor operators. The project has included two human factors experimental evaluations of the Response Tree **expert system**, a prototype **expert system** for helping nuclear reactor operators respond to emergency conditions. This paper discusses the Response Tree **expert system**, the experiments which have been performed to test its effectiveness, and the results of the experiments. Reference is made to the accident at TMI. 12 refs.

DOCUMENT NUMBER = INI17:006941

TI: LWR simulation code research at the INEL.
AU: Ransom, V.H.; Wagner, R.J.; Laats, E.T.; Chow, H.; Bray, M.A.; Trapp, J.A.
CO: Idaho National Engineering Lab., Idaho Falls (USA); Colorado Univ., Denver (USA).
LA: English
RP: EGG-M--13785. CONF-850646--5.
IM: 1985. 15 p. Availability: INIS. Available from NTIS, PC A02/MF A01 - GPO as TI85014585.
CF: (2. International specialists meeting on small break LOCA analysis in LWRs. Pisa (Italy). 24-26 Jun 1985.)
AB: This paper summarizes the light water reactor simulation research projects being conducted at the Idaho National Engineering Laboratory. Related research in advanced two-phase models, multi-grid numerical methods, and **knowledge** engineering is also described. Future directions for LWR simulation research efforts are suggested. In particular the potential that **expert systems** hold for improved utility and simulation capability is recognized. Software developments for full utilization of the next generation of supercomputers are seen as pacing LWR simulation progress. The methods of **artificial intelligence** may help speed these software developments.

DOCUMENT NUMBER = INI16:081172

TI: Development of methods for nuclear **power** plant personnel qualifications and training.
AU: Jorgensen, C.C.; Carter, R.J. (Oak Ridge National Lab., TN).
LA: English
MS: Twelfth water reactor safety research information meeting: proceedings. Volume 6. Szawlewicz, S.A. (comp.). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research.
RP: NUREG/CP--0058-Vol.6.
IM: Jan 1985. p. 28-35. Availability: INIS. Available from NTIS, PC A22/MF A01 - GPO $9.50 as TI85900640.
CF: (12. water reactor safety research information meeting. Gaithersburg, MD (USA). 23-26 Oct 1984.)
AB: The Nuclear Regulatory Commission (NRC) has proposed that additions and revisions should be made to Title 10 of the "Code of Federal Regulations," Parts 50 and 55, and to Regulatory Guides 1.8 and 1.149. Oak Ridge National Laboratory (ORNL) is developing methods and some aspects of the technical basis for the implementation and assessment of training programs, personnel qualifications, and simulation facilities to be designed in accordance with the proposed rule changes. The paper describes the three methodologies which were developed during the FY-1984 research. The three methodologies are: (1) a task sort procedure (TSORT); (2) a simulation facility evaluation methodology; and (3) a task analysis profiling **system** (TAPS). TAPS is covered in detail in this paper. The task analysis profiling **system** has been designed to support training research. It draws on **artificial intelligence** concepts of pattern matching to provide an automated task analysis of normal English descriptions of job behaviors. TAPS development consisted of creating a precise method for the definition of skills, **knowledge**, abilities, and attitudes (SKAA), and generating SKAA taxonomic elements. It **systematically** outputs skills, **knowledge**, attitudes, and abilities, and information associated with them.

DOCUMENT NUMBER = INI16:078081

TI: Combination of **artificial intelligence** and procedural language programs in a computer application **system** supporting nuclear reactor operations.
AU: Stratton, R.C.; Town, G.G.
CO: Argonne National Lab., Idaho Falls, ID (USA); Central Washington Univ., Ellensburg (USA). Dept. of Computer Science.
LA: English
RP: CONF-850841--1.
IM: 1985. 7 p. Availability: INIS. Available from NTIS, PC A02/MF A01 as DE85007944.
CF: (Workshop on coupling symbolic and numeric computation in **expert systems**. Bellevue, WA (USA). 27-29 Aug 1985.)
AB: A computer application **system** is described which provides nuclear reactor **power** plant operators with an improved decision support **system**. This **system** combines traditional computer applications such as graphics display with artifical **intelligence** methodologies such as reasoning and diagnosis so as to improve plant operability. This paper discusses the issues, and a solution, involved with the **system** integration of applications developed using traditional and **artificial intelligence** languages.

DOCUMENT NUMBER = INI16:075391

TI: Failure diagnostic **expert systems** for fast **reactors**. **SYSTEME EXPERT** en diagnostic de panne pour un reacteur nucleaire a neutrons rapides.
AU: Himbaut, S.; Le Guillou, G.; Quinton, J.C.
CO: CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les- Durance (France). Dept. des Reacteurs a Neutrons Rapides.
LA: French
RP: CEA-CONF--7634. DRNR-P--292.
IM: May 1984. 18 p. Availability: INIS.
CF: (Colloquium on **expert systems** and applications. Avignon (France). 2-4 May 1984.)
AB: The aim of **SYSTEME EXPERT** is to permit diagnostics of one (or more) casual failure of the "ultimate resort cooling" (RUR) (including monitoring or control sensor failures) from data given (or not) by the various sensors or by the classical alarm **systems**. It comprises the following modules: an interface (in French) for information acquisition with possible suppression of the rules, a demonstrator of SL-resolution type, a data incoherence test an inverse interface for listing the collections.

DOCUMENT NUMBER = INI16:075217

TI: Review of trends in computerized **systems** for operator support.
AU: Cain, D.G. (Electric **Power** Research Inst., Palo Alto, CA (USA)).
LA: English
MS: Diagnosis of and response to abnormal occurrences at nuclear **power plants**. Proceedings of a seminar organized by the IAEA and held in Dresden, 12-15 June 1984. International Atomic Energy Agency, Vienna (Austria).
RP: IAEA-TECDOC--334.
IM: May 1985. 535 p. p. 97-108. Availability: INIS.
CF: (Seminar on the diagnosis of and response to abnormal occurrences at nuclear **power plants**. Dresden (German Democratic Republic). 12-15 Jun 1984.)
AB: The major trends shaping the development of computerized operator support **systems** in nuclear **power plants** are reviewed. These trends are the result of prior research in disturbance analysis **systems** that provided the technology base, and the SPDS requirement, which has been the impetus for change. The process is expected to result in hybrid control rooms with computer-driven supervisory workstations that complement conventional control board lay-outs. In the next three to five year period substantial upgrading of computer hardware will allow new and more sophisticated applications routines to be developed for operator support. Greater attention is being given to on-line validation of input signals for computer applications. A general movement towards operating strategies that are not **based** upon pre-analyzed event sequences is expected to influence the development of

operator aids. The integration of displays with operating procedures will enable the computer **system** to a better coupling between problem detection and its resolution. Improved design methodology will assure that computer applications are accepted and used by operations personnel. Greater on-line analysis capability is stimulating the trend towards more on-site analysis and decision-making at nuclear **power plants.** Software standardization reflects the high cost of software development and the desire by utilities to gain greater independence from suppliers. There is growing realization that control rooms are beset by many of the demands and limitations of other office settings and that some of these may be addressed by the burgeoning office automation technology. Trends beyond the next five years are difficult to predict; however, there will be a trend towards more intelligent software. **Artificial intelligence** technology may play a pivotal role in future applications. Taking these trends into perspective, the author concludes that a promising future exists for computerized operator support in nuclear **power** plant control rooms. (author).

DOCUMENT NUMBER = INI16:072072

TI: Diagnosing the vibrating behaviour of the PWR internal structures using **expert systems** techniques and algorithmic treatment. Diagnostic du comportement vibratoire des internes d'un REP associant la technique des **systemes** experts a un traitement algorithnique.
AU: Carre, J.C.; Gibert, R.J.; Kavenoky, A.
CO: CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France).
LA: French
RP: CEA-CONF--7530.
IM: Oct 1984. 7 p. Availability: INIS.
CF: (4. SMORN symposium on fluctuations in nuclear **reactors.** Dijon (France). 15-19 Oct 1984.)
AB: In order to automate the diagnostic of the vibrating behaviour of PWR internal structures, a program **based** on the **expert system** technology is added to the present time numerical software, the calculation comprises three stages: the signal is first processed with a Fourier transformation, then with a quadrature and an average thus giving the Noise **Power** Spectral Density. From this function a few characteristic resonances (between 5 and 10) are extracted and transmitted to the **expert** software.

DOCUMENT NUMBER = INI16:064518

TI: Application of variational-heuristic learning methods to reactor physics: A new concept in **artificial intelligence.**
AU: King, T.L.; Cacuci, D.G.; Machado, E.L.; Perez, R.B.; Wood, P.T. (University of Tennessee, Knoxville, TN 37996).
LA: English
RP: CONF-841105--.
JR: Trans. Am. Nucl. Soc. ISSN 0003-018X. (1984). v. 47 p. 439-440.
NO: Published in summary form only.
CF: (Joint meeting of the American Nuclear Society and the Atomic Industrial Forum. Washington, DC (USA). 11-16 Nov 1984.)

DOCUMENT NUMBER = INI16:064396

TI: Minimal cut-set methodology for **artificial intelligence** applications.
AU: Weisbin, C.R.; de Saussure, G.; Barhen, J.; Oblow, E.M.; White, J.C.
CO: Oak Ridge National Lab., TN (USA).
LA: English
RP: CONF-841254--1.
IM: 1984. 5 p. Availability: INIS. Available from NTIS, PC A02/MF A01 as DE85004779.
CF: (1. conference on **artificial intelligence** applications. Denver, CO (USA). 5-7 Dec 1984.)
AB: This paper reviews minimal cut-set theory and illustrates its application with an example. The minimal cut-set approach uses disjunctive normal form in Boolean algebra and various Boolean operators to simplify very complicated tree structures

composed of AND/OR gates. The simplification process is automated and performed off-line using existing computer codes to implement the Boolean reduction on the finite, but large tree structure. With this approach, on-line **expert** diagnostic **systems** whose response time is critical, could determine directly whether a goal is achievable by comparing the actual **system** state to a concisely stored set of preprocessed critical state elements.

DOCUMENT NUMBER = INI16:046873

TI: An overview of a **knowledge based system** for preventive maintenance support of nuclear **power plants.**
AU: Terano, Takao; Nishiyama, Takuya; Yokoo, Takeshi (Central Research Inst. of Electric **Power** Industry, Tokyo (Japan)).
LA: Japanese
JR: Denryoku Chuo Kenkyusho Hokoku. (May 1984). (no.583015) p. 1-5, 1- 31.
AB: In recent years, much interest has been paid to **knowledge** engineering techniques for new vehicles of advanced information processing. As a practical application in electric **power** industry, this report discusses a **knowledge based system** for supporting preventive maintenance of nuclear **power plants.** To support preventive maintenance tasks, the **system** must have facilities to reason failures and accidents of the **plants,** to evaluate their significance, to predict any possible troubles, and to indentify appropriate preventive countermeasures for them. This report describes the overview of the prototype **system** from a viewpoint of **knowledge** engineering. The results of the study are as follows: (1) The **knowledge** base of the prototype **system** consists of a data-base on **plants** and a rule-base derived from experts' **knowledge.** Using the information in the **knowledge** base, the **system** diagnoses the plant without real-time interaction with operational **plants.** (2) Expert's **knowledge** in the rule-base is represented in the non-procedural declarative forms. These rules are organized in some hierarchical structure so as to be used efficiently and used in conjunction with the corresponding set of plant information in the data-base. (3) The prototype **system** is incrementally developed with the rapid prototyping techniques, that is, the processes of design, implementation and evaluation are repeated several times. (author).

DOCUMENT NUMBER = INI16:027417

TI: **Expert system** for fast reactor diagnostic. Un **systeme expert** en diagnostic sur reacteurs a neutrons rapides.
AU: Parcy, J.P.
CO: Aix-Marseille-2 Univ., 13 - Marseille (France). Centre Universitaire de Luminy.
LA: French
RP: FRNC-TH--1907.
IM: Sep 1982. 124 p. Availability: INIS.
NO: 68 refs.; Prolog. Designation: These (3e Cycle).
AB: A general description of **expert systems** is given. The operation of a fast reactor is reviewed. The **expert system** to the diagnosis of breakdowns limited to the reactor core. The structure of the **system** is described: specification of the diagnostics; structure of the data bank and evaluation of the rules; specification of the prediagnostics and evaluation; explanation of the diagnostics; time evolution of the **system**; comparison with other **expert systems.** Applications to some cases of faults are finally presented.

DOCUMENT NUMBER = INI16:018350

TI: The STAR concept, **systems** to assist the operator during abnormal events.
AU: Felkel, L. (Gesellschaft fuer Reaktorsicherheit m.b.H. (GRS), Garching (Germany, F.R.)).
LA: English
JR: Atomkernenerg. Kerntech. ISSN 0171-5747. (Dec 1984). v. 45(4) p. 252-260. CODEN: ATKEA.

AB: Man-machine-communication in electrical **power plants** is increasingly **based** on the capabilities of minicomputers. Rather than just displaying raw process data more complex processing is done to aid operators by improving information quality. Advanced operator aids for nuclear **power plants** are e.g. alarm reduction, disturbance analysis and **expert systems.** Operator aids use complex combinations and computations of plant signals, which have to be described in a formal and homogeneous way. The design of such computer-based information **systems** requires extensive software and engineering efforts. The STAR software concept described in this paper, however, reduces the software effort to a minimum by providing an advanced program package which facilitates specification and implementation of engineering know-how necessary for sophisticated operator aids. (orig.).

DOCUMENT NUMBER = INI16:002864

TI: Fault diagnosis using **artificial intelligence.**
AU: Bray, M.A.
LA: English
MS: Proceedings of the eleventh water reactor safety research information meeting. Volume 6. Szawlewicz, S.A. (comp.). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research.
RP: NUREG/CP--0048-Vol.6.
IM: Jan 1984. p. 42-50. Availability: INIS. Available from NTIS, PC A10/MF A01; 1 - GPO* as TI84900650.
CF: (11. NRC water reactor safety research information meeting. Gaithersburg, MD (USA). 14-24 Oct 1983.)
AB: An experiment was performed during 1983 that measured performance of nuclear plant operators with and without a computer-based operator aid. This paper discusses the results of that experiment and implications for design and regulation of advanced computer aids in nuclear control rooms.

DOCUMENT NUMBER = INI15:067989

TI: Response trees and **expert systems** for nuclear reactor operations.
AU: Nelson, W.R.
CO: EG and G Idaho, Inc., Idaho Falls (USA).
LA: English
RP: NUREG/CR--3631. EGG--2293.
IM: Feb 1984. 27 p. Availability: INIS. Available from NTIS, PC A03/ MF A01; 1 - GPO as DE84010223.
AB: The United States Nuclear Regulatory Commission is sponsoring a project performed by EG and G Idaho, Inc., at the Idaho National Engineering Laboratory (INEL) to evaluate different display concepts for use in nuclear reactor control rooms. Included in this project is the evaluation of the response tree computer **based** decision aid and its associated displays. This report serves as an overview of the response tree methodology and how it has been implemented as a computer **based** decision aid utilizing color graphic displays. A qualitative assessment of the applicability of the response tree aid in the reactor control room is also made. Experience gained in evaluating the response tree aid is generalized to address a larger category of computer aids, those known as **knowledge based expert systems.** General characteristics of **expert systems** are discussed, as well as examples of their application in other domains. A survey of ongoing work on **expert systems** in the nuclear industry is presented, and an assessment of their potential applicability is made. Finally, recommendations for the design and evaluation of computer **based** decision aids are presented.

DOCUMENT NUMBER = INI15:053256

TI: Response tree evaluation - implications for the use of **artificial intelligence** in process control rooms.
AU: Bray, M.A.; Nelson, W.R.; Blackman, H.S.; Fowler, R.D.
CO: EG and G Idaho, Inc., Idaho Falls (USA).

LA: English
RP: NUREG/CR--3461. EGG--2272.
IM: Jan 1984. 32 p. Availability: INIS. Available from NTIS, PC A03/ MF A01; 1 - GPO as DE84005826.
AB: An experiment was performed during 1983 that measured performance of nuclear plant operators with and without a computer-based operator aid. This report discusses the results of that experiment and their implications for design and regulation of advanced computer aids in nuclear control rooms. The aid tested is called a response tree and is intended to help operators properly align a piping **system** despite component or support **system** failures. In the experiment, 28 reactor operator subjects were required to align a **system** to inject coolant and stop a temperature excursion in a simulated reactor. The experiment did not show an improvement in operator performance when the response tree aid was used. More important, the experiment did produce several conclusions related to the design and evaluation of computer aids in nuclear control rooms.

DOCUMENT NUMBER = INI15:022127

TI: Fault diagnosis using **artificial intelligence**. (PWR; BWR.)
AU: Bray, M.A.
CO: EG and G Idaho, Inc., Idaho Falls (USA).
LA: English
RP: EGG-M--22083. CONF-8310143--46.
IM: 1983. 10 p. Availability: INIS. Available from NTIS, PC A02/MF A01 as DE84002632.
CF: (11. NRC water reactor safety research information meeting. Gaithersburg, MD (USA). 14-28 Oct 1983.)
AB: An experiment was performed during 1983 that measured performance of nuclear plant operators with and without a computer-based operator aid. This paper discusses the results of that experiment and implications for design and regulation of advanced computer aids in nuclear control rooms.

DOCUMENT NUMBER = INI14:799964

TI: Application of **artificial intelligence** to the detection and diagnosis of core faults in fast **reactors**. Application de l'Intelligence artificielle a la detection et au diagnostic des defauts du coeur dans les reacteurs a neutrons rapides.
AU: Le Guillou, G.; Berlin, C. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)); Parcy, J.P. (Aix- Marseille-2 Univ., 13 - Marseille (France)).
LA: French
MS: Nuclear **power** plant control and instrumentation 1982. Proceedings of an international symposium on nuclear **power** plant control and Instrumentation organized by the IAEA and held in Munich, 11-15 October 1982. International Atomic Energy Agency, Vienna (Austria).
RP: IAEA-SM--265/54.
IM: Vienna. IAEA. 1983. ISBN 92-0-050683-6. 726 p. p. 173-186.
SE: Proceedings series.
CF: (International symposium on nuclear **power** plant control and instrumentation. Munich (Germany, F.R.). 11-15 Oct 1982.)
AB: An automatic diagnostic **system** designed specially for fuel assembly accidents in fast **reactors** of the French type is described. It has an "**expert system**" structure and, at the present stage of the project, is operating on a laboratory scale in an experimental unit (Solar 16/65). The software aspects of the development of such **systems** (structure and formulation of rules, language, etc.) are presented in detail. Various simulation tests are outlined. "Real-time" application of the **system** will begin in 1983 on the Phenix reactor. (author).

DOCUMENT NUMBER = INI14:788353

TI: On the control of HTR numerical models using **artificial intelligence** methods.
AU: Bubak, M.; Gudowski, W.; Kitowski, J.; Moscinski, J. (Akademia Gorniczo-Hutnicza, Krakow (Poland)).
LA: English

MS: Gas-cooled **reactors** today (Vol.3). Performance and safety technology. Status of gas-cooled **reactors.** Proceedings of the conference held in Bristol on 20-24 September 1982. British Nuclear Energy Society, London. 4 vol. set - price Pound110.00.
IM: London. British Nuclear Energy Society. 1982. p. 117-123. Available from B.N.E.S., 1-7 Great George St., London SW1P 3AA.
CF: (Conference on gas-cooled **reactors** today. Bristol (UK). 20-24 Sep 1982.)
AB: A description is given of the application of **artificial intelligence** methods (heuristic approach, learning **systems,** fuzzy sets theory and game theory) to the short and long time control problems of HTR and HTR-plant numerical models. (author).

DOCUMENT NUMBER = INI14:768850

TI: Rule-based approach to cognitive modeling of real-time decision making.
AU: Thorndyke, P.W. (Perceptronics, Inc., Menlo Park, CA).
LA: English
MS: Workshop on cognitive modeling of nuclear-plant control-room operators. Abbott, L.S. (ed.). Oak Ridge National Lab., TN (USA). Portions are illegible in microfiche products.
RP: NUREG/CR--3114.
IM: 1982. p. 147-155. Available from NTIS, PC A10/MF A01; 1 - GPO as DE83006002. Availability: INIS.
CF: (Workshop on cognitive modeling of nuclear plant control room operators. Dedham, MA (USA). 15 - 18 Aug 1982.)
AB: Recent developments in the fields of cognitive science and **artificial intelligence** have made possible the creation of a new class of models of complex human behavior. These models, referred to as either **expert** or knowledge-based **systems,** describe the high-level cognitive processing undertaken by a skilled human to perform a complex, largely mental, task. **Expert systems** have been developed to provide simulations of skilled performance of a variety of tasks. These include problems of data interpretation, **system** monitoring and fault isolation, prediction, planning,. diagnosis, and design. In general, such **systems** strive to produce prescriptive (error-free) behavior, rather than model descriptively the typical human's errorful behavior. However, some research has sought to develop descriptive models of human behavior using the same theoretical frameworks adopted by **expert systems** builders. This paper presents an overview of this theoretical framework and modeling approach, and indicates the applicability of such models to the development of a model of control room operators in a nuclear **power** plant. Such a model could serve several beneficial functions in plant design, licensing, and operation.

DOCUMENT NUMBER = INI14:737365

TI: REACTOR: an **expert system** for diagnosis and treatment of nuclear reactor accidents. (PWR; BWR.)
AU: Nelson, W.R.
CO: EG and G Idaho, Inc., Idaho Falls (USA).
LA: English
RP: EGG-M--09782. CONF-820874--1.
IM: 1982. 7 p. Availability: INIS. Available from NTIS., PC A02/MF A01 as DE83000757.
CF: (National conference on **artificial intelligence.** Pittsburgh, PA (USA). 18 - 20 Aug 1982.)
AB: REACTOR is an **expert system** under development at EG and G Idaho, Inc., that will assist operators in the diagnosis and treatment of nuclear reactor accidents. This paper covers the background of the nuclear industry and why **expert system** technology may prove valuable in the reactor control room. Some of the basic features of the REACTOR **system** are discussed, and future plans for validation and evaluation of REACTOR are presented. The concept of using both event-oriented and function-oriented strategies for accident diagnosis is discussed. The response tree concept for representing **expert knowledge** is also introduced.

DOCUMENT NUMBER = INI08:290595

TI: Computer simulation of nuclear reactor control by means of heuristic learning controller. Symulacja komputerowa sterowania reaktorem jadrowym przy pomocy uczacego sie ukladu heurystycznego.

AU: Bubak, M.; Moscinski, J. (Institute of Physics and Nuclear Techniques, Krakow (Poland)).

LA: Polish

JR: Zesz. Nauk. Akad. Gorn.-Hutn. Stanisl. Staszica, Autom. (1976). (no. 13) p. 53-65.

AB: A trial of application of two techniques of **Artificial Intelligence:** heuristic Programming and Learning Machines Theory for nuclear reactor control is presented. Considering complexity of the mathematical models describing satisfactorily the nuclear **reactors,** value changes of these models parameters in course of operation, **knowledge** of some parameters value with too small exactness, there appear diffucluties in the classical approach application for these objects control **systems** design. The classical approach consists in definition of the permissible control actions set on the base of the set performance index and the object mathematical model. The **Artificial Intelligence** methods enable construction of the control **system,** which gets during work an information being a priori inaccessible and uses it for its action change for the control to be the optimum one. Applying these methods we have elaborated the reactor **power** control **system.** As the performance index there has been taken the integral of the error square. For the control **system** there are only accessible: the set **power** trajectory, the reactor **power** and the control rod position. The set **power** trajectory has been divided into time intervals called heuristic intervals. At the beginning of every heuristic interval, on the base of the obtained experience, the control **system** chooses from the control (heuristic) set the optimum control. The heuristic set it is the set of relations between the control rod rate and the state variables, the set and the obtained **power,** similar to simplifications applied by nuclear **reactors** operators. The results obtained for the different control rod rates and different reactor (simulated on the digital computer) show the proper work of the **system.** (author).

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