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FOREWORD

The maintenance of systems, structures and components in nuclear power plants (NPPs) plays an important role in assuring their safe and reliable operation. Worldwide, NPP maintenance managers are seeking to reduce overall maintenance costs while maintaining or improving the levels of safety and reliability. Thus, the issue of NPP maintenance is one of the most challenging aspects of nuclear power generation.

There is a direct relation between safety and maintenance. While maintenance alone (apart from modifications) will not make a plant safer than its original design, deficient maintenance may result in either an increased number of transients and challenges to safety systems or reduced reliability and availability of safety systems.

The confidence that NPP structures, systems and components will function as designed is ultimately based on programmes which monitor both their reliability and availability to perform their intended safety function. Because of this, approaches to monitor the effectiveness of maintenance are also necessary. An effective maintenance programme ensures that there is a balance between the improvement in component reliability to be achieved and the loss of component function due to maintenance downtime. This implies that the safety level of an NPP should not be adversely affected by maintenance performed during operation.

The nuclear industry widely acknowledges the importance of maintenance in NPP safety and operation and therefore devotes great efforts to develop techniques, methods and tools to aid in maintenance planning, follow-up and optimization, and in assuring the effectiveness of maintenance.

The IAEA officer responsible for this publication was A. Gomez of the Division of Nuclear Installation Safety.

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ANNEX: PAPERS PRESENTD AT THE TECHNICAL COMMITTEE MEETING

1. INTRODUCTION

Many years of operational experience, related mainly to the occurrence of operational events and to the need for availability and reliability of safety important equipment, confirm the link between maintenance and safety.

Even though maintenance alone will not make a plant safer than its original design, maintenance is very important to ensure that the original design basis is maintained or not unacceptably degraded.

Experience has shown that despite efforts made by NPPs in accomplishing maintenance activities (programme content and implementation) and despite regulations, operational events occur.

This has lead to the consideration of a result oriented process for the assessment of the effectiveness of maintenance.

Traditional maintenance practices include detailed processes, requirements and instructions. While they have the advantage of being in most cases clear, detailed and easy to implement and regulate, they sometimes may lead to focus attention on compliance without perhaps adequate consideration of performance and results.

Current trends in safety related maintenance take into account the plant as a whole and its global safety performance rather than individual components and their individual performance. General ideas and concepts that are important in the modern approaches for the development and monitoring of effective maintenance programmes can be summarized as follows:

- The scope (structures, systems and components) of a maintenance programme needs to be adequately defined. The items falling within the scope of the programme need to be ranked according to their safety significance in order to adequately focus resources and efforts.
- The effectiveness of maintenance has to be evaluated against some sort of performance criteria, i.e., reference values for performance need to be defined.
- An effective maintenance programme must not only consider random equipment failures, but also potential failures caused by maintenance practices and activities.
- The safety impact of equipment out of service has to be taken into account.
- An effective maintenance programme needs to include considerations on the balance between availability and reliability.
- An effective maintenance programme needs to take into account worldwide industrial operational experience and needs the support and involvement of NPP departments other than maintenance.
- The effectiveness of the maintenance programme needs to be periodically assessed.
- Personnel training is a key aspect of an effective maintenance programme.

Practices emerging from the development of these concepts and ideas are basically results oriented and include risk considerations, i.e., rather than on processes, they are focused on results which can be monitored by maintenance effectiveness indicators.

In September 1997 the IAEA convened a Technical Committee Meeting in Vienna to compile information on the most advanced techniques, approaches, methods and tools used in connection with NPP safety related maintenance activities and to discuss their advantages and drawbacks. The first part of the TCM was dedicated to the presentation of papers. The papers that were presented are included in the Annex to this document. During the second part of the TCM the participants distributed in five working groups discussed the following topics:

- Maintenance decisions and applicable tools, methods and constraints
- Plant processes related to maintenance
- Use of PSA in maintenance decisions
- Deterministic/engineering considerations and their interfaces with PSA based evaluations
- Implementation considerations and interfaces

The following Sections summarize the discussions and conclusions of the five working groups.

2. MAINTENANCE DECISIONS AND APPLICABLE TOOLS AND APPROACHES

2.1. INTRODUCTION

Safety related maintenance has long been recognized as essential to plant safety. Methods and tools are constantly in development to improve the quality of maintenance and to maintain or even reduce costs.

The purpose of this Section is to provide an overview of the design and execution of the safety related maintenance programme. This Section also includes, for each major process step described, brief discussions on developments in tools and methods available to assist decision-making.

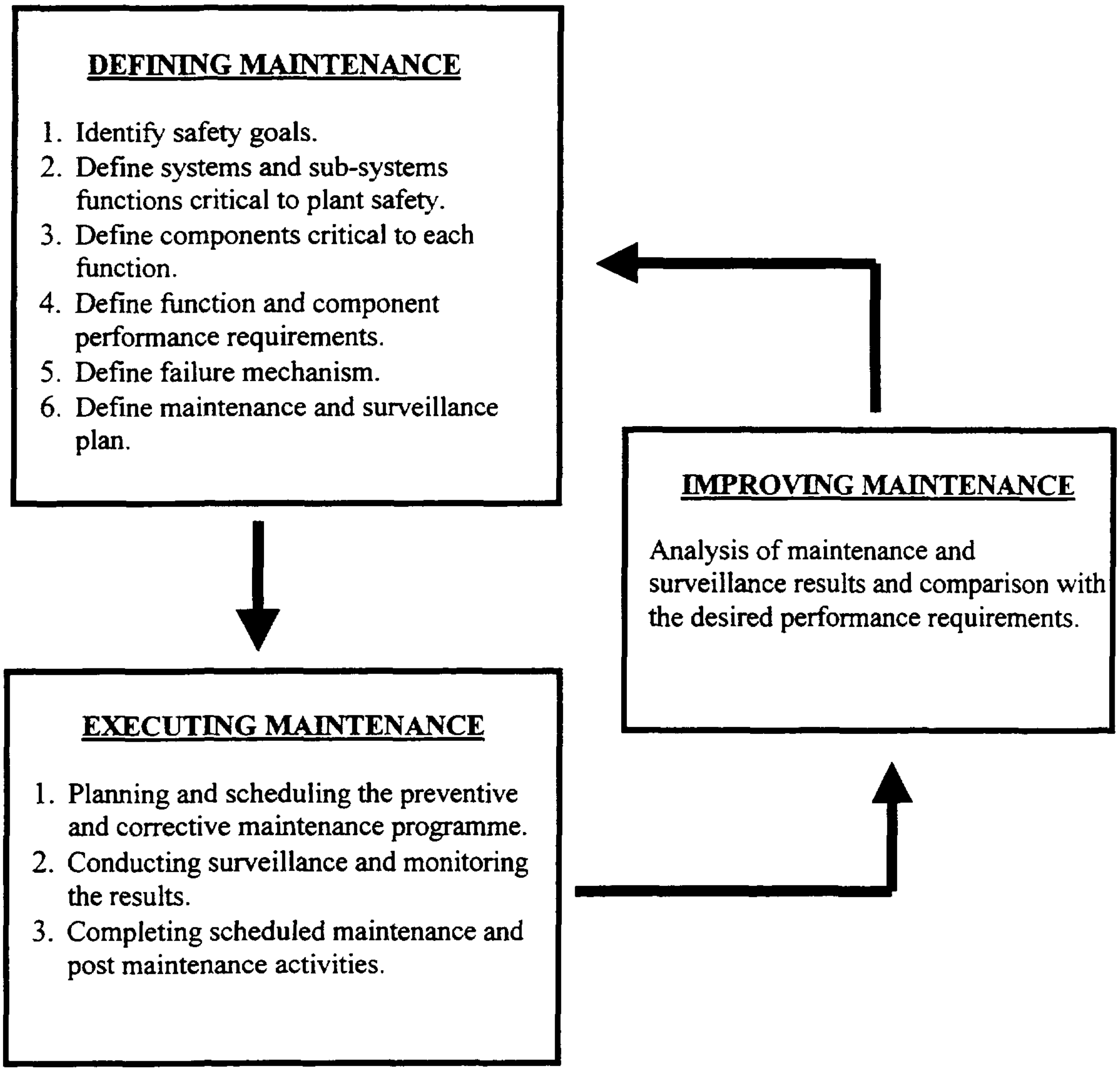
Figure 1 represents the general consensus regarding the essential elements of a living safety related maintenance programme. Within each major process such as defining maintenance, executing maintenance and performance feedback, a subset of key processes requiring important decisions related to safety related maintenance can be found.

2.2. PROCESSES FOR DEFINING MAINTENANCE

2.2.1. Establishment of plant safety goals

The initial step to ensure that all safety related maintenance is adequately handled within the plant maintenance programme requires a clear identification of the overall plant safety goals.

These goals are usually established and implemented into regulatory requirements under the direction of the regulatory body. From a safety related maintenance perspective, these goals are useful in determining the impact of certain plant configuration or maintenance strategies on the public safety objectives and can be used to optimize maintenance scheduling during plant operation and outages.



DEFINING MAINTENANCE

1. Identify safety goals.
2. Define systems and sub-systems functions critical to plant safety.
3. Define components critical to each function.
4. Define function and component performance requirements.
5. Define failure mechanism.
6. Define maintenance and surveillance plan.

EXECUTING MAINTENANCE

1. Planning and scheduling the preventive and corrective maintenance programme.
2. Conducting surveillance and monitoring the results.
3. Completing scheduled maintenance and post maintenance activities.

IMPROVING MAINTENANCE

Analysis of maintenance and surveillance results and comparison with the desired performance requirements.

FIG. 1. Maintainance related processes.

2.2.2. Identification of systems, sub-systems, and functions that are important to safety

Process description

It is necessary to define which systems, sub-systems and components are important to safety so that attention can be focused on the maintenance requirements for satisfying the plant safety goals. In addition, it is important to identify the safety functions that each of the identified systems and sub-systems have to fulfil. This then allows the identification of the components that are effective in supporting these safety functions.

Personnel involved

An expert panel of three categories of personnel may carry out this task:

- The *plant designers* (or system engineers with access to all the design documentation) can identify which ones among all the systems, sub-systems and components are important to safety, in reliance on the detailed plant description. Such a selection will be made on the basis of understanding the safety functions that the systems are expected to perform to fulfil the plant safety objectives.
- The *safety analysts* can help to identify the safety roles of the systems, supported by their knowledge of the Final Safety Analysis Report. In addition, the results and conclusions of the plant PSA will complement this information, since they are

intimately connected with the systems requirements to achieve the plant safety goals.

- The participation of *operations staff*, with their practical knowledge of Emergency Operating Procedures, and of the system functions required for each emergency scenario, is necessary in order to complete the expert panel.

Remarks

If PSA is used for the selection of systems, sub-systems and functions important to safety, it must be plant specific. The validity and applicability of the PSA can only be ensured by periodic updates to incorporate experience accumulated in operating and maintaining the plant. In other words, the PSA should be "living"[1] in order to maximize its contribution to this activity.

There are costs associated with creating and updating such a living PSA model. However, if these costs can be accommodated, the increased quality and validity of the information available to the expert panel will help to ensure that the number of identified systems, sub-systems and components is not unnecessarily large and that only those items which are genuinely important to safety are selected.

Since the scope of PSA does not cover all maintenance needs, expert panels can supplement the information needed to select components. A note of caution: poorly managed expert panels may tend to overfill the list of safety related systems, sub-systems and components.

2.2.3. Definition of components critical to function

Process description

Having defined the safety functions of systems and sub-systems, the next logical step is to define or identify, for each of the previously identified systems and sub-systems, the components that are critical for the accomplishment of the safety function.

There are many components in each system/subsystem; however, not all the possible failures of components necessarily jeopardize the safety functions of the system. Therefore, it is necessary that the system components critical to the safety functions of the system be identified.

Personnel involved

The following personnel may be involved in this task:

- *System designers*, using their theoretical knowledge of the system, equipment and failure modes, can prepare a list of the critical components, their possible failure modes and the effect of these failures on the system.

[1] A Living PSA (LPSA) can be defined as a PSA of the plant which is updated as necessary to reflect the current design and operational features and is documented in such a way that each aspect of the model can be directly related to existing plant information, plant documentation or the analysts' assumptions in the absence of such information. The LPSA would be used by designers, utility and regulatory personnel for a variety of purposes according to their needs, such as design verification, assessment of potential changes to the plant design or operation, design of training programmes and assessment of changes to the plant licensing basis. (IAEA-TECDOC-1106 on "Living Probabilistic Safety Assessment (LPSA)".

- *PSA/safety analysts,* using PSA methods and results, FMEA (failure mode and effect analysis) or other safety analysis techniques, can determine the components which, if they fail, hinder the system from fulfilling its safety function.
- *System engineers* are normally in charge of the evaluations covering system performance. This is accomplished by collecting system operating data, maintenance history and other relevant data. The system engineer may sometimes suggest modifications in order to increase the reliability of the system. The system engineer is a link between a design engineer and an operating and maintenance engineer.

Methods and tools

The 'input' information that may be used to identify the components which are critical to the safety functions can be the following:

- Design information
- FSAR (Final safety analysis report)
- Operational experience feedback
- Maintenance experience feedback
- Plant-specific PSA

2.2.4. Definition of performance criteria: function level and component level

Description of process

Specific criteria need to be established to monitor performance at function level and component level.

Performance criteria are defined to provide a basis for establishing the maintenance, surveillance and testing frequencies as well as for monitoring for satisfactory performance.

Consideration should be given to the achievement of an adequate balance between availability and reliability through the examination of performance criteria.

Personnel involved

Maintenance staff, technical support, operations, PSA analysts and system designers can provide input for establishing performance criteria.

Benchmarking, experience feedback, technical specification and industry experience can be used as a reference.

Methods and approaches to establish targets

The following methods can be used to establish targets for performance criteria:

- consistency with PSA assumptions, for example unavailability of components or systems;
- benchmark results with other nuclear power plants;
- company business plans or other policies.

Establishing performance criteria values at the function/component level might prove difficult due to the limited availability of reference data (different design, operating history, age of the plant, previous plant performance, etc.).

2.2.5. Identification of failure modes and mechanisms

Description of process

One of the most important parts in the practical implementation of methods for maintenance optimization is the identification of component failure modes that need to be considered and the failure mechanisms that lead to those failure modes The results of analytical and practical methods to identify failure mechanisms are considered vital for this purpose

Personnel involved

The personnel who may be involved in the development of these activities are

- *System/component engineers* using an analytical method to determine failure mechanisms through the design review Interaction with risk and reliability analysts to identify functional failures may be an advantage,
- *Maintenance personnel* assisted by system engineers and specialists in reliability analysis for the review of operational experience from the plant and from other plants

Methods

From an analytical standpoint, the most commonly used tools in the identification of failure modes and mechanisms are

- Failure Mode and Effect Analysis
- Remnant component life studies (ageing studies), that pay special attention to existing degradation processes.
- Operational Experience Feedback. It is possible to perform a review of the internal maintenance history to identify failure modes and mechanisms that have really occurred Resorting to external operational experience can also support the final identification of failure modes and mechanisms.

2.2.6. Definition of the maintenance and surveillance plan

Process description

Once the performance requirements of components have been defined and their failure mechanisms identified, the next obvious step is to ensure that the maintenance and surveillance activities performed on those components are adequate so that the components in the system/sub-system critical to safety remain able to perform their designated duty

Surveillance of the components is to be done on a regular basis so that any degradation can be detected and corrected through maintenance before a failure occurs. The kind of surveillance, and the frequency, methods and procedures chosen for surveillance will normally depend on the component, its complexity and its safety significance The type of maintenance needed depends on the type of degradation that has occurred, this is based on the operating condition of the component and the surveillance data. Maintenance can consist in repairing/reconditioning a component or replacing it altogether.

Personnel involved

The following persons will be involved in deciding the type and frequency of the maintenance activities.

- *Maintenance staff* are the personnel with wider experience in deciding which type of surveillance and maintenance is needed in order to keep the system effective.
- *Operating staff* normally perform surveillance by means of periodical testing based on technical specifications and operational routines. Therefore, they can detect when a component needs attention. In addition, they can provide input regarding operational load and resources.
- *Regulators* prescribe or approve certain surveillance tests and their frequencies based on design basis accident reports. These schedules can also be based on risk analyses and other assessment methods.

Tools and methods

The following tools may be available to serve as input in the definition of maintenance and surveillance schedules:

- reliability centered maintenance results,
- condition monitoring of the system/equipment/component and analysis of the results obtained,
- regulatory body guidelines,
- risk assessment analysis.

In addition, the data required are:

- user manuals,
- technical specifications,
- operating data,
- equipment history, etc.

Remarks

In order to correct degradations found during surveillance without shutting down the unit, it is normally required that redundant trains/equipment be available. In these cases, it may be advisable to take out of service the degraded train/equipment to perform corrective maintenance, provided, however, that the overall regulatory requirements are met. In addition, a risk analysis can be used to assess the increased risk level during the proposed configuration.

Improper interpretation of data that is collected on equipment degradation can lead to the performance of unnecessary maintenance and potential remnant human errors due to maintenance of the component (i.e., maintenance-induced errors, post-maintenance misalignments, etc.).

Another consideration is to determine whether the actual testing practices detect the failure modes and mechanisms identified.

2.3 PROCESSES FOR EXECUTION OF SAFETY RELATED MAINTENANCE AND SURVEILLANCE

2.3.1. Planning and scheduling preventive and corrective maintenance

Process description

Good planning and scheduling will ensure that equipment unavailability and reliability are closely monitored and controlled over time.

Adjustments to the maintenance plan are sometimes necessary to ensure that the objective of preventing failures through maintenance is appropriately balanced against the objective of minimizing unavailability due to preventive maintenance and monitoring activities.

Personnel involved

Maintenance planners are responsible for these tasks. The operations department normally reviews the maintenance plan. For maintenance planning, it is necessary to take into account the resources available (staff, documents, materials), constraints due to operational requirements, etc. A key point is, for example, to decide if the job can be done during power operation or shutdown.

Tools and methods

Some tools and methods that can be used for maintenance planning and scheduling are:

- technical specification,
- a risk monitor,
- risk matrix,
- computerized planning tools,
- job control information systems,
- severity index.

Remarks

To improve the effectiveness of the planning and scheduling process, help will be needed by the planners. This may include: the use of a safety monitor to keep the risk during the cycle as low as possible, a PSA analyst and technical system experts to assist in this area, etc.

2.3.2. Executing surveillance of safety related systems and components

Process description

The surveillance plan defined for the critical safety system and components consists of the collection of a set of system and component parameters and their analysis to identify potential deviation from the normal acceptable range. The outcome of this monitoring can result in the necessity to perform additional diagnostic work or corrective maintenance before the component degrades to a point where its safety function is affected.

Personnel involved and methods used

Surveillance is carried out by plant operators during field rounds, inspection of control room panels and monitoring of the plant annunciation system. Operations staff also carry out functional tests on systems to verify that standby equipment is operational.

The tools and methods used for monitoring and surveillance vary considerably. While operator rounds are commonly done at plants, the details regarding the recorded parameters are not the same. Some plants use portable PC equipment to facilitate the recording of the data and the transmission of the data to the plant system engineers and records department through the information network. Some plants with more recent design have taken advantage of lower cost computer technology by remotely monitoring thousands of data points from the field and communicating the results to plant staff on a need-to-know basis. Historical records are also easily maintained. These technology advances have the advantage of providing easily retrievable records and may be cost effective by reducing the workload associated with manual recording of field data. Nevertheless, it should be noted that this technology is no substitute for field rounds, since many aspects of system and equipment performance such as minor leaks, unusual control valve movements, air leaks, pipe vibration, etc., may not be monitored otherwise.

Surveillance is also carried out by maintenance personnel through the use of condition based maintenance. This involves the collection of equipment performance data such as vibration, temperatures, oil analysis, thermographs, valve diagnostics, etc. Again different technologies exist to facilitate the data collection, recording and even presentation of data to the engineer for analysis. The precision of these measurements is such that early deterioration of a component can be detected and corrective maintenance scheduled before a failure can have a safety impact and result in costly repairs. It should be noted that not all degradation mechanisms can be detected through operator and maintenance surveillance.

Similarly, system engineer surveillance involves long term trending of system and component performance as well as system test results for the purposes of anticipating deterioration and initiating maintenance. Again, a number of tools have been developed for data gathering, historical data storage, retrieval and trend analysis which provide the information to the engineer. These techniques may be cost effective, but the analytical skills and experience of the system engineer are still essential for an effective surveillance programme.

2.3.3. Performing maintenance and post-maintenance testing

This Section addresses the performance of maintenance activities, post-maintenance tests and functional tests.

Personnel involved and methods

Before performing the maintenance work, all activities need to be adequately prepared and clearly understood by the maintenance personnel.

The performance of maintenance tasks is by nature the responsibility of the maintenance department. However, these tasks may also be performed by a qualified contractor under the supervision of the NPP maintenance department.

Operations and maintenance staff participate in post-maintenance testing which is effective in monitoring the quality of the work completed and in ensuring that the equipment maintained is not left out of service (i.e., misaligned, disconnected, etc.). The results of the maintenance activities will be reviewed by the system engineers or other suitably qualified equipment personnel.

The operation department is normally responsible for performing the functional tests required by the Technical Specifications once the maintenance tasks have been completed. The purpose of these tests is to ensure that the system is able to accomplish its safety function. It is important that these tests be performed independently of any post-maintenance testing activity carried out as part of the maintenance work.

Remarks

The effectiveness of the maintenance work also depends on the following factors:

- procedures; which need to be complete, unambiguous and must clearly define the work steps
- appropriate job preparation
- skilled personnel; personnel need to be qualified through training for each particular task in advance
- availability of spare parts and tools.

2.4. IMPROVING MAINTENANCE

Maintenance optimization is essentially based on the data analysis of the maintenance and surveillance results; it may be completed with external feedback.

The following paragraphs discuss data collection, the use of external experience and data analysis for maintenance optimization. It should be recognized that this process is only effective in achieving an improved maintenance programme if the original programme was established on a sound basis using a systematic approach and had the benefit from the review and input of an expert panel.

Personnel involved

The personnel dedicated to analyse maintenance results for optimization are plant technicians and system engineers who evaluate internal plant information and data from external experience. The sources of information needed for data analysis are: results of preventive and corrective maintenance consigned in maintenance records filled by the maintenance technicians and engineers, results of risk impact assessment, and previous availability and reliability results of system performance impacting plant safety.

Tools and methods

The inputs most commonly used for maintenance optimization are:

- root cause analyses performed for the most safety significant failures;
- cause determination for functional failures[2];

[2] A determination of the basic cause for the occurrence of the failure. The depth of the investigation will normally depend on the safety significance of such failure and the evidence of the cause.

- plant databases containing maintenance history data and test and surveillance results;
- review of external operational experience to take and compare feedback insights to evaluate the performance of systems and equipment;
- the current maintenance strategy, including maintenance cost evaluations.

Remarks

Limitations can sometimes arise owing to the "poor" quality of the data contained in the maintenance and surveillance records or in the plant databases. Therefore, the specific plant information should be evaluated for completeness and applicability before using it as input to the optimization process. The quality of data depends strongly on the maintenance personnel who collect it. It is therefore important that this personnel be aware of the importance of the required data.

Finally, it is important to bear in mind that the success in the maintenance optimization process depends on the continuous review for trending purposes of the parameters analysed in the optimization. Hence, the results of periodical evaluations should be reviewed to take into consideration the trends resulting from the current maintenance practices.

3. SPECIFIC ISSUES IN PLANT PROCESSES

3.1. INTRODUCTION

A maintenance programme at a nuclear power plant encompasses many different activities and a large number of personnel. An effective maintenance programme at a plant involves defining the plant processes, co-ordination of activities and personnel and performing the activities.

The elements of a "living" safety related maintenance programme have been discussed in detail in the previous Section. This Section focuses more on specific issues related to the plant maintenance processes and plant implementation.

The elements in a maintenance programme can, in simple terms, be defined as follows:

- To determine the maintenance to be done based on the strategies and the techniques available.
- To plan and schedule bearing in mind the resources available.
- To train maintenance personnel.
- To execute the work using skills, procedures, tools and spares.
- To close out the work, collect information and to test.
- To analyse the results.
- To review the maintenance programme based on the experiences and the lessons learned.

Safety culture has to be an umbrella covering all activities in maintenance. The awareness of safety in maintenance is a continuous process highlighted in training, briefings, maintenance documentation, etc.

3.2. DISCUSSION OF ISSUES RELATED TO PLANT PROCESSES

Maintenance objectives and strategies can be revised and improved to reflect the benefits of the risk and reliability based approaches available and to meet the challenges of cost reduction in a way allowing to focus on safety critical equipment.

The following considerations can be useful in the revisions of the maintenance objectives and strategies:

System and equipment ranking

In the past, equipment classification was based on engineering judgement. With the development of risk based classification techniques, ranking of plant equipment now can be done based on these techniques in combination with engineering judgement and analysis of internal and external experience.

Application of these new approaches can be more cost effective if updated design basis documentation is available.

Revision of maintenance programmes

Maintenance programmes can be optimized based on modem approaches such as reliability centered maintenance, RCM, risk focused maintenance, RFM, or other similar techniques, taking cost effectiveness considerations into account.

Ageing considerations can be addressed by the use of a Life Management Programme. In this case, measures need to be taken to ensure that information exchange between different programmes (e.g. RCM, life management, etc.) is considered.

An important consideration in the revision of maintenance programmes is that upgraded and integrated information systems be available.

Long term planning

Long term planning has to be *proactive* and take into account challenges such as ageing, experience, etc., to be effective.

Scheduling, execution and testing

The use of new scheduling tools can be helpful in this respect (e.g. net grid planning).

Likewise, the use of risk monitoring tools can contribute to safer configurations of the plant due to maintenance.

On-line maintenance programmes can contribute to improve quality, safety and diminish time constraints and other similar considerations which affect personnel during outage periods.

Coaching and supervision by managers are important. The use of communication techniques (pre-job meetings and job safety analysis) can be useful in this phase.

Documentation

Systematic, accurate and the timely recording of information is another important criterion of success.

3.3. CONSIDERATIONS REGARDING SUPPORT ACTIVITIES

3.3.1. Training of personnel

All plant staff who are involved in plant maintenance should be provided with at least a basic knowledge of plant processes. Furthermore, all personnel working on equipment, including contractors, must be well informed on the plant rules and procedures in relation to working in the installation.

Maintenance oriented training

Personnel working in the maintenance area need to understand the plant specific maintenance strategy. Furthermore they should be able to use and understand maintenance related documents, understand safety classes, system functions and possible failure consequences.

Specific training in maintenance

Individual, professionally oriented training has to be provided for personnel working in different phases of the maintenance process (planning, execution, QA, QC).

Personnel qualification control

All personnel training and qualification records have to be regularly updated in order to ensure that work is performed by properly qualified individuals who meet work specific requirements (qualification particulars should be integrated in the Plant Information System).

3.3.2. Interface between different working groups in the maintenance area

An excessive number of interfaces can contribute to possible maintenance failures. Working group teams and co-ordination groups can be helpful in this area.

To reduce the number of interfaces, the trend is to have multi-skilled staff and teams in maintenance. The teams and the groups can be organized either in a permanent or temporary manner. Interface control can also be ensured through a set of meetings in order to follow up maintenance activities and to control processes.

3.3.3. Information systems

Maintenance information systems need to be able to communicate with other supporting information systems in the plant (e.g., PSA, personnel qualifications, spare part control system, etc.). These systems are designed to support, step by step, the main maintenance activities in planning, scheduling, execution, control and evaluation of the work.

In order to obtain a timely, accurate and reliable information, it is important to have an unique database managed by responsible staff. The data needs to be handled and updated at its place of origin.

3.3.4. External and internal experience feedback

The sources of experience feedback can be external or internal. Recommended internal feedback sources may be:

- work order reports
- event reports
- maintenance reports
- health physics reports
- internal audits
- failure cause determination/root cause analysis reports.

External feedback may come from:

- official agencies such as IAEA, WANO, INPO, etc.
- technical meetings held by plant manufactures and plant designers
- service information letters (vendor recommendations)
- meetings (several sources).

3.4. MAINTENANCE RELATED INDICATORS

Performance indicators are useful tools to evaluate maintenance effectiveness. Indicators can be used for trending of equipment/systems and plant performance. Indicators to trend organizational performance can also be established.

To improve maintenance performance evaluation, it is useful to separate failures of equipment due to maintenance from other causes. In addition, this kind of cause determination will be useful to define corrective actions to prevent future recurrences.

IAEA-TECDOC-1141 "Operational Safety Performance Indicators For Nuclear Power Plants"[3] proposes a framework for the definition of plant specific indicators to monitor several areas related to the operational safety performance of the plant. This is not specifically focused towards monitoring the effectiveness of maintenance but rather, of all the operational aspects which bear upon the safety performance of the plant. Several examples of indicators that can help to monitor the effectiveness of the maintenance programme are discussed in said TECDOC. However, each plant has to select plant specific meaningful indicators, define them and establish targets and action plans. Also, it is expected that at the level of the maintenance department, a larger number of specific indicators be defined. It may be that for the purpose of closely monitoring individual systems, a number of specific indicators at the system level also be defined.

[3] International Atomic Energy Agency, Operational Safety Performance Indicators For Nuclear Power Plants, IAEA-TECDOC-1141, Vienna (2000).

3.5. EVALUATION OF MAINTENANCE EFFECTIVENESS

Ineffective maintenance may result in an increased number of transients, increased challenges to safety systems and a reduction of reliability and availability. It is necessary to evaluate the maintenance effectiveness not only to ensure the reliability and design requirements of equipment, but also to gain an understanding of the effectiveness of the changes and improvements being made to the maintenance programme.

The evaluation of maintenance effectiveness can be based on:

- systematic and periodical assessment of equipment/system/plant performance (following the logic of the maintenance rule, 10-CFR-50.65)
- results from audits, peer review and similar activities
- benchmarking techniques.

The performance criteria defined for the systems and equipment (as discussed in Section 2.2.4) form a good basis for evaluating the effectiveness of maintenance.

4. USE OF PSA IN MAINTENANCE RELATED DECISION MAKING

4.1. INTRODUCTION

Probabilistic Safety Assessments (PSAs) are increasingly being used to provide input to many aspects related to maintenance related decision-making.

PSAs can be used to address many aspects related to maintenance, such as:

- maintenance planning and scheduling
- selection or gradation of equipment
- decisions related to on-line maintenance
- configuration control during maintenance
- follow-up of the risk impacts of maintenance
- technical specification changes to accommodate maintenance needs
- regulatory inspection of maintenance activities
- the establishment of performance indicators and criteria.

A living PSA (LPSA) is necessary to conduct the applications discussed above. Some plants may have a *risk monitoring system* completed and available to conduct maintenance applications.

4.2. ISSUES ASSOCIATED WITH PSA APPLICATIONS IN MAINTENANCE

The use of PSA to support NPP maintenance involves addressing/resolving a number of issues relating to PSA models and how they are applied.

4.2.1. PSA quality and scope for different types of maintenance applications

Assuring that the PSA is of appropriate quality and uses standards similar to, for example, to IAEA Safety Series No. 50-P-4, and that the scope of the PSA accommodates the needs of maintenance applications is an important starting point in the use of PSA for maintenance applications.

This Section focuses on the PSA scope requirements for different maintenance applications, addresses many of the current limitations, and provides good practices in PSA developments that will effectively address the needs of safety related maintenance.

4.2.1.1. Scope

Currently, many of the PSA based maintenance applications are conducted using Level 1 internal event PSA. These applications provide useful inputs, but the use of PSA can be more powerful if the scope of the available PSA is enhanced. For example, the availability of a shutdown PSA (SPSA) would allow the evaluation of the impact of maintenance carried out during the shutdown period as compared with the corresponding measure during power operation. This would help to decide whether the maintenance activities under consideration should take place during the power or shutdown operational modes.

Another example is the evaluation of maintenance activities related to components in the containment safeguards. These evaluations would require PSA models beyond Level 1, i.e., a Level 2 PSA or, at least, a Level 1 + PSA[4]. Moreover, since the containment may be open during a significant part of the shutdown operational state, even Level 3 evaluations can provide insights.

The modelling scope should include internal fires and floods and significant external hazards. These are often categories of common cause initiators, and thus important to conditional risk, assuming part of the systems are down for maintenance. The plant models need to be reviewed to ensure that the hazard analyses are also adequate to support maintenance related applications.

PSA must include best estimate modelling assumptions and data to ensure that failure events, unavailability events, etc., are correctly ranked. When very conservative data is used for some components, the risk ranking can be unrealistic and the components can be unduly ranked as more safety significant than they actually are.

Techniques have been developed by ASME to perform risk ranking for the passive pipes in the plant. These techniques determine the most risk significant pipes considering both direct effects, loss of pipe function, and indirect effects, effects on other systems. Leaks, disabling leaks, and pipe breaks are considered. Indirect effects are as a result of spray, flooding, and pipe whip.

4.2.1.2. Human reliability

Typical human errors derived from maintenance activities modelled in the PSA are miscalibration of I&C equipment and misalignment of components. Usually, administrative checks and functional tests follow each maintenance activity in order to check the functionality of the maintained component. However, it may be that these controls are not sufficient to detect all potential human errors associated with each maintenance activity.

[4] Level 1+ PSA is a PSA for which accident sequences are developed to Plant Damage States (instead of core damage states, which are the end states for the level 1 PSA sequences) taking into consideration the status of containment safeguard systems and other features which affect the progression of the severe accident.

The identification and analysis of the maintenance related human errors requires a comprehensive review and understanding of the maintenance tasks and procedures.

In modifying pre-existing maintenance procedures and practices both the PSA analyst and the maintenance planner must recognize the potential for introducing new human errors or modifying the probability of the ones already identified. Modified maintenance procedures and strategies need to be reviewed to identify these potential human errors. Then, the PSA has to be modified accordingly.

4.2.1.3. Level of detail

The level of detail of the models in the plant specific PSA may not correspond with the needs for a maintenance programme. For example, electrical and instrumentation components are sometimes grouped in macro-components for modelling purposes. One can decide to increase the level of detail to facilitate the application (i.e. to establish a direct link between PSA events and plant components as used in the maintenance programme). However, it has to be borne in mind that this will slow down the quantification process significantly and it will lead to a very large number of cut sets, which are difficult to review and to draw conclusions from.

The same result can be achieved by performing post-processing of the PSA results. This can be done by hand, and sometimes automatically. Many plants have databases available where electrical and instrumentation components are connected to the dedicated front line component. Such a database could be used to perform post-processing.

4.2.2. Specific modelling considerations for maintenance related PSA applications

In order to use PSA to support maintenance, changes to the available PSA models may be necessary. These changes may relate to specific needs in maintenance applications or may involve transforming the PSA to facilitate its use.

Examples of these are:

- modification of the fault tree model to reach the required level of detail
- modifications necessary to link the PSA model with Risk Monitor codes or other software developed for the purpose of specific maintenance issues
- adjusting the model to perform system reliability calculations.

The model needs to include all significant maintenance related pre-accident human errors. Special attention needs to be paid to the contributions from maintenance activities on components not included in the models. For example, components within the PSA scope may be aligned in order to perform maintenance on components outside of the PSA scope. The probability that these PSA components are left in the wrong position after such maintenance activities needs to be analysed. Modifications to these contributions can happen if the maintenance strategies are changed.

Other features which may affect the applicability of the PSA to support maintenance are the following:

- modelling maintenance events and test strategies inside fault tree structure by using “not” gates and “house events”
- definition of test and maintenance input to CCF groups.

4.2.3. PSA limitations for maintenance related applications

It needs to be stated that PSA is only one tool among several that can provide input to support NPP testing and maintenance.

It is necessary to acknowledge that there can be issues in the PSA related to the scope and quality of the models which might limit its applicability to support maintenance. This does not imply a limitation of PSA as a technique in itself, but rather, stems from the lack of adequacy in the models, data, documentation and QA of some PSAs. These problems could lead to results that are not sound. Hence, these PSAs are inadequate to support decision making.

Depending on the quality of the PSA (e.g. completeness and PSA scope, level of HRA, CCF modelling, consequential and recovery events modelled) the resulting contributors to the risk can vary significantly, thus resulting in a risk based ranking of components, which is required as an input to some of the maintenance related applications, that could be far from realistic. This issue can have an important impact on the results of the application. For example, it can lead to focusing maintenance efforts where they are least required.

An important limitation of many PSAs to support testing and maintenance is asymmetric modelling[5]. This modelling choice can lead to an unrealistic ranking of components according to their risk significance.

PSA based maintenance optimization deals, in general, with the balance between component unavailability due to maintenance — leading to a risk increase, and improved component reliability due to increased preventive maintenance — leading to a decrease in risk. However, the decrease in risk due to increased or better preventive maintenance cannot be explicitly expressed in the PSA, i.e., the estimation of the potential decrease in failure rate with enhanced maintenance strategies is not an easy or straightforward task.

Another approach is to focus maintenance on the most risk significant equipment, and, in the meantime, to prevent unacceptable configurations due to maintenance in combination with other plant conditions. For this, the PSA has to provide not only the correct component ranking but also credible results. In addition, PSAs based on expected average plant operational status and developed under standard PSA software are often not flexible enough to reflect and analyse, in a reasonable time frame, changes of expected states or series of scheduled conditions at the plant. This is clearly a limitation for a PSA application such as configuration control. Risk monitors are much more suitable and flexible for this type of application.

PSA can provide input to the Reliability Centered Maintenance (RCM) programme. One limitation is that, usually, PSAs only model a limited scope of plant components with given scope of failure modes, and all are risk/safety related. Also, the selected limit of definition of some modelled components, which may be perfectly adequate for the purposes of the PSA, may not be detailed enough for the purpose of RCM. On the other hand, the scope of RCM also includes plant components and failure modes that are considered important for reasons other than safety (i.e., availability, production, cost). Therefore PSA has to be combined with other techniques such as FMEA (failure mode and effect analysis) that help to reveal other

[5] Asymmetric modelling: for rotating systems, only one line-up is chosen to be modelled in most PSAs. In addition, initiators such as LOCA and SGTR are supposed to take place in one loop. This is done for reasons of simplicity. In this way the correct numerical result are obtained, but from the point of view of the logic (cut sets/qualitative results) the results are not correct: one pump might appear more important than a similar pump in the same system.

components, sub-components and failure modes that need to be included in the scope of the RCM.

Finally it must be mentioned that, in general, PSA does look at risk coming from core damage events. Other undesirable situations are either not covered in the PSA or are assigned to a success end state. PSA must then be supplemented by other approaches to include these insights in the list of critical components.

4.2.4. Need for uncertainty analysis

Risk importance indicators are usually used in order to show in which components maintenance activities should be focused. Higher efforts are put on components which have high risk significance. If the uncertainty associated with the resulting ranking is large, attention has to be paid to this uncertainty.

That is, if the low risk significant component (ranking according to the point estimate probability) has a large uncertainty associated, this implies that this component may actually be risk significant. In this case, the uncertainty has to be considered in order to prevent the component from being left out of the scope of the maintenance programme.

Sensitivity analysis also offers similar benefits. Sensitivity analyses can consider modelling uncertainties, changes of assumptions etc. The results of these analyses may lead to a ranking of components different from that obtained in the *base case.* This will help to include in the scope of the maintenance programme all risk significant components and components suspected to be risk significant.

However, it may be hard for decision makers to understand how uncertainty information associated with component ranking or other results of the maintenance related PSA application should be used. Therefore, the PSA team has to provide clear information in a form that is easy to interpret by non-PSA experts.

PSA is used to define/optimize AOTs (allowed outage times) and STIs (surveillance test intervals); the uncertainty associated with the input data, modelling assumptions, etc., also affects the results. Therefore, the risk based AOTs and STIs may be given as a single point estimate value with an uncertainty range. A decision-maker can then decide whether to consider the range of recommended AOTs and STIs. Other factors such as cost, operational considerations, etc., can be of help in reaching the final decision.

5. DETERMINISTIC/ENGINEERING CONSIDERATION AND THEIR INTERFACES WITH PROBABILISTIC EVALUATIONS

This Section focuses on the way in which PSA based evaluations and performance monitoring can be used to complement deterministic considerations in relation to NPP testing and maintenance.

5.1. DETERMINISTIC/ENGINEERING CONSIDERATIONS

Maintenance actions include such activities as testing, inspections, preventive maintenance, and corrective maintenance. Traditionally maintenance decisions have been based on engineering analyses and deterministic considerations. These include:

- defence in depth
- single failure criteria
- functional performance based on accident analysis
- perceived need for high reliability
- manufacturers' recommendations
- experience
- standards and codes.

5.2. ELEMENTS OF A MAINTENANCE PROGRAMME POTENTIALLY AFFECTED BY PSA

The current maintenance programmes have been developed during the operation lives of the NPPs. They were originally based on the recommendations of the plant designer or manufacturers and have developed as a result of operating experience and plant condition monitoring.

The advent of modern PSA techniques has opened up many possibilities to improve maintenance strategies which can have demonstrable cost and safety benefits. As already discussed in the previous Section, PSA may help to identify and rank the key systems and components on the one hand and the failure modes on the other hand.

The estimated ranking of the future impact, high (H) or low (L), of PSA to existing deterministic and engineering based maintenance programmes, as shown in Table I, is based on the papers included in the Annex to this document and their importance in terms of cost effectiveness and safety improvements. Currently, there is already extensive experience on the use of PSA to modify AOTs and STIs. Further, in some plants, scheduling decisions are already being influenced by risk profiles produced by risk monitor tools. This is expected to increase significantly in the future, since many risk monitors are currently under development/implementation. Also for the scope and frequency of tests and inspections a certain shift due to risk based priorities is to be expected.

TABLE I. ESTIMATED PSA IMPACT ON EXISTING MAINTENANCE PROGRAMMES

Test/Inspection	**Preventive Maintenance (Servicing)**	**Corrective Maintenance (Repair/Replacement)**
• Test type (L) • Scope (H) • Frequency (H) • Scheduling (H) • AOT (H)	• Task • Scope (L) • Frequency (L) • Scheduling (H) • AOT (H)	• Design specification • Scheduling (H) • AOT (H but only exemptions)

Some maintenance decisions may remain predominantly based on expert judgement. These include:

- maintenance based on failure diagnosis or condition monitoring.
- relating the type or form of maintenance to the equipment's performance and potential failure mechanisms.
- determination of maintenance task duration.
- review of defect history to categorize events into real failures and trivial occurrences.

- resolving arguments on the balance of safety between workers and the public (e.g. we may get more confidence in a low CDF (core damage frequency) as a result of increased weld inspections, but this may mean higher doses to workers in actuality, not just potentially).

There are also other elements which would be expected to feature alongside PSA based information. Apart form those listed above, important engineering inputs are detailed knowledge and understanding of how the equipment actually works.

There are, however, some dangers associated with undue reliance on PSA. One such danger relates to showing that overall risk targets/goals are met during periods of maintenance outage. Using PSA it may be possible to show that removal of all protection against a certain fault for a short period of time is numerically acceptable. A deterministic safeguard should be considered, and this could be expressed as:

> *"For all maintenance operations, there should be protection provided for all faults at all times."*

In cases of redundant equipment or systems, the requirement expressed as — do not maintain all items at the same time, or render them unavailable by mismatch of maintenance of front line and support systems — is clear. In some other cases it is less clear, but for fire doors for example, they may need to be held open, and alternative contingency plans can be established by posting a firewatch.

Another possible danger with PSA is its lack of comprehensiveness in that it may be of limited scope and the possibility of unanticipated faults or failure modes is always present. The validity of some of the input data to PSAs is also a concern. There are ways of interpreting PSAs which enable to derive information about areas which are not explicitly included in the PSA, but these are relatively new developments (i.e., risk importance of pipes or other passive components not modelled in the PSA which are within the scope of the in-service inspection programmes).

5.3. ROLE OF PSA IN STRENGTHENING THE DETERMINISTIC BASIS

Relying solely on deterministic maintenance considerations is unlikely to provide the optimum means of ensuring the desired level of safety. Complementary use of PSA and performance monitoring can help by addressing some of the deterministic deficiencies.

Examples of drawbacks and deficiencies associated with the deterministic approach are:

- the original basis of deterministic maintenance requirements are often not clear — maintenance activities can be performed without an understanding of why they are being done and of their impact on safety.
- deterministic considerations generally lead to only two categories — for example safety related or not safety related.
- risk and reliability are not considered in a consistent and systematic way
- plant based dependencies and common cause failures are not adequately addressed

The way in which PSA and performance monitoring can help is shown in the Table II.

TABLE II. THE WAY IN WHICH PSA AND PERFORMANCE MONITORING CAN HELP TO OVERCOME THE DRAWBACKS OF TRADITIONAL APPROACHES

DEFICIENCY	HOW PSA AND PERFORMANCE MONITORING HELP
Unclear basis	*PSA* can help to establish the most important components and systems, and will show how changes in reliability influence public risk *Performance monitoring* indicates the most likely failure modes, the reliability under different conditions and the impact of different maintenance, surveillance and testing programmes
Two categories only	*PSA* can provide a much better indication of relative safety between components hence aids in defining priorities *Performance monitoring* will enable changes in relative importance to be seen, and if necessary acted upon (e.g. test frequencies)
Risk & reliability	Systematic and consistent approach provided by *PSA* *Performance monitoring* will enable reality to be reflected
Dependency & CCF	*PSA* explicitly models dependencies of front line plant on support systems and reveals any potential problems. *PSA* also enables the impact of CCF on similar components to be identified *Performance monitoring* is important, but for CCF and other rare events this may need to be done as part of a national or international effort

The following is a summary of features of performance monitoring that can be advantageous to support NPP testing and maintenance:

- identification of most likely failure modes and causes
- influence of environment, service condition
- effectiveness of maintenance activities
- relative reliability of different components.

The following is a summary of PSA features that can be advantageous to support NPP testing and maintenance:

- importance of functions, components and failure modes whether or not they are classified as safety related

- importance of reliability
- integrated numerical measure for risk
- groups of components susceptible to CCF and their importance.

Clearly any changes to safety related maintenance must necessarily take account of the prevalent regulatory regime.

5.4. EXAMPLES WHICH SHOW THAT PSA SERVES AS A GOOD COMPLEMENT TO OTHER APPROACHES

The examples discussed below clearly indicate how PSA methodology complements current deterministic insights.

5.4.1. Allowable outage time (AOT)

The deterministic basis for several key components of a Pressurized Water Reactors (e.g., Refuelling Water Storage Tank (RWST), Control Room Ventilation System) were based on their role in Design Basis Accident mitigation. For example, since RWST is the single source of water for LOCA mitigation, the AOT assigned to that component in many plants can be as short as 1 hour. Similarly, if a single failure such as a degraded seal would render the control room inoperable, the operators will be required to fix it within a very short time (e.g., within one hour) or start shutting down the reactor. PSA insights will provide recommendations that can lead to reconsideration of the regulatory requirements dictated by the Technical Specifications. For example, based on the state-of-the-art knowledge on transition risk (risk associated with starting up and shutting down a nuclear reactor) and in consideration of the frequency of LOCAs and the performance of mitigating systems, public safety would be better served by extending these AOTs to provide reasonable time for repairs while the plant is in operation. PSA based approaches to address these issues are already under way in some countries.

5.4.2. Test method

During an accident, the emergency diesel generator is expected to start and come to full power within a relatively short time, which may be as short as 10 seconds. Therefore, the deterministic basis of the diesel test method required the emergency diesel generator to start and load within this short time while it was tested. Operating experience and PSA provided several insights that required a change in this test scheme. Plant operating experience showed that the cold start of diesels has the potential to damage the diesels. The PSA insight highlighted the fact that most of the accident sequences which rely on the diesels (e.g., Loss of Off-site Power) do not require the diesels to start within 10 seconds. In consideration of the above, the testing method of the emergency diesels have been changed.

5.4.3. Safety management

The deterministic basis of technical specifications overlooks the safety significance associated with interaction between initiating event potential and the mitigating systems. For example, in maintenance scheduling, the deterministic basis gives latitude to the operator to schedule a high risk surveillance (e.g. MSIV part stroke test) while a critical mitigation system train (e.g., AFW) is out of service. The technical specifications also overlooked the elevated risk associated with synergistic effects of taking multiple system trains out of service. For example, the deterministic basis and the technical specification derived from that deterministic basis allow taking out train A of emergency diesel and train A of the High and Low Pressure Injection systems concurrently. However, PSA insights would strongly discourage such practices. In consideration of the above, some nuclear plants use risk matrices, risk monitors, safety monitors, or risk management guidance documents to complement technical specifications.

6. OTHER IMPORTANT TOPICS IN SAFETY RELATED MAINTENANCE

6.1. SPECIAL TECHNIQUES AND APPLICATIONS

Although most of the generic aspects in connection with NPP maintenance have been covered in the previous Sections, important topics remain to be discussed. Some maintenance projects and activities are very closely linked to safety. Examples of these are:

- Internal vessel inspections: intergranular stress corrosion cracking (IGSCC) susceptibility. Evaluation of each one of the components, classification according to their implicit susceptibility to IGSCC. Inspection plan. New tools for inspections and developments to mitigate IGSCC.
- Erosion-corrosion: new methods developed (inspections without removal of the insulation).
- Inspections of SG tubes.
- Robotics.
- Condition monitoring of components.

The present report does not discuss the above topics. It is believed that other task groups are working on these issues and that extensive experience is available both from the industry and research organizations. However, these issues cannot be ignored when discussing "safety related maintenance" and this is why they are briefly mentioned here.

6.2. QUALITY ASSURANCE

Quality assurance (QA) is an integral part of maintenance programmes in a nuclear power plant. The basic intent of such a programme is to assure the quality of the maintenance conducted, i.e., that it is conducted by appropriately trained personnel following proper procedures, using adequate tools and the proper quality of materials and spare parts. It also assures that appropriate records are maintained to detect, improve and audit the maintenance at the plant. The quality assurance programme for maintenance usually consists of the following elements:

- maintenance organization,
- training of maintenance personnel,
- process for developing/changing/upgrading maintenance procedures and their implementation,
- non-conformance control and corrective actions,
- document control and records,
- audit of the maintenance programmes at the plant,
- audit of the subcontractors/suppliers.

Any QA programme needs a clear mission or task the organization is committed to perform. These tasks must be identified in plain language and their results capable of being measured. Reviewing the performance and readjusting the mission or task statement is an essential element of a QA programme. (See IAEA Code and Guide "Quality Assurance for Safety in Nuclear Power Plants and other Nuclear Installations", Safety Series No. 50-C/SG-Q, for more specific guidance).

Safety culture is not restricted to all or part of a QA programme. Safety Culture reflects an attitude and must be present at all levels of activities, from decision-making to individual tasks of operation, maintenance or control.

6.3. ALARA PRINCIPLE IN MAINTENANCE

In many cases, maintenance work causes radiation doses to the workers. Tasks should be organized according to the following principles:

- A radiation dose can only be allowed if it produces a positive effect.
- The dose should be as low as reasonably achievable (ALARA).
- The dose must be lower than the levels imposed by the regulatory authority.

The first principle states that only persons performing the tasks are allowed in the radiation areas.

The second principle requires a careful planning of tasks. The use of training on mock-ups can be useful. Finishing a task without an independent check can also reduce doses, but this requires technicians trained to do self-checking.

In addition, errors in maintenance procedures and in conducting a maintenance job may cause incidents or accidents resulting in doses for the personnel involved. Care should be taken to assure that such events are eliminated, to the extent possible.

6.4. INTERFACE WITH REGULATORY BODY

As the effect of maintenance in the safe operation of the plant is accepted, owners and Regulatory Authorities are obliged to reach agreements in solving this interface.

In many countries rules exist for maintenance which directly address the requirements for the selection of systems, component and structures for the maintenance programme, the scheduling of maintenance, the monitoring of compliance with the maintenance programme, the assessment and evaluation of the maintenance programme, the assessment of working procedures, the witnessing of some maintenance activities and the assessment of maintenance administration.

However, the scoping of structures, systems and components to be covered by regulatory requirements, the definition of maintenance indicators and acceptance criteria, procedures and tools to identify maintenance failures or deficiencies and their causes, the methodologies to quantify on-line maintenance risk are aspects to be established by both industry and regulatory authorities.

The regulatory involvement in new fields such as RCM and risk monitoring and PSA to support the maintenance programme are subjects to be discussed by industry and authorities.

Guidance may have to be developed to support the authorities in their judgement and decisions in these matters.

This will require an important effort from both plant operators and regulatory bodies to maintain continuous communication. To this end, regulatory bodies should have experienced maintenance personnel available and able to discuss and find the reasonable balance between requirements and technically proven solutions.

6.5. INTERFACE BETWEEN MAINTENANCE PERSONNEL AND OTHER NPP STAFF

In addition to the maintenance department, other NPP departments such as Safety, Operation, Technical Support, Training and Procurement play a role in maintenance processes and issues. It is necessary that plant staff in these departments co-operate and co-ordinate efforts bearing in mind the common objectives. Taking this synergy one step further, the creation by individual NPPs of a special task force for solving in the shortest time possible significant system or component failures would be of great advantage.

The interrelations among the different NPP departments need to be clearly established and transparent. Only then can a maintenance programme which is adequately balanced both from the safety and the economic standpoints be achieved.

An improved maintenance interface requires:

- The scheduling and co-ordination of maintenance activities at a high level of decision, the delegation of competencies and direct responsibilities to maintenance staff without loosing overall control and an open-minded way of thinking.
- External experience feedback and sharing and keeping abreast of technical and managerial developments in other countries.

Specific maintenance staff training is needed for developing at least basic system oriented knowledge, an equipment oriented knowledge and radiation protection knowledge

Maintenance staff have to get used to be involved in more detail in the assigned maintenance task having a targeted end-date (within their responsibilities) and to introduce the required corrections to everyday activity, according to the experience feedback system prevailing in the NPP.

Following maintenance activities, modification control and updating of operation/maintenance routines, internal procedures, station instructions and applicable databases should be performed as soon as possible.

Maintenance staff should be trained in the methodologies required to comply with the new regulatory requirements (failure cause determination, loss of function, performances etc.) and maintenance indicators of new maintenance programmes in order to provide the interface data required by engineering, operation and safety personnel.

6.6. LIFE MANAGEMENT

6.6.1. General considerations

Demonstrating that ageing and performance are being effectively managed is a key element of a successful remnant lifetime programme. The safety and the benefit-to-cost opportunities are maximized when there is an early recognition of areas requiring enhancement to achieve effective ageing and/or performance management. The aim of maintenance evaluation for life management is to accomplish these objectives. The characteristics of some system, structure and equipment items in NPPs and the severity of some conditions have produced specific forms of degradation which are not always covered by current maintenance practices.

Maintenance evaluation should determine the effectiveness of current plant programmes to address age related degradation and to monitor ongoing performance for each system, structure or component. The evaluation supported by a systematic methodology should ensure that systems, structures and components are subject to testing, inspections, sampling or controls at intervals commensurate with their remnant life. Additionally, there are obvious benefits in integrating these programmes with the new regulatory requirements on maintenance and with complementary programmes such as maintenance optimization.

The nuclear industry is collecting experience on these degradation phenomena and their evolution over time. New mitigation methods and monitoring tools are being developed and proven methodologies to support these maintenance evaluation programmes are available finally. Efforts must be undertaken to seek how best to share this information and to facilitate the interchange of experience between plants performing such programmes.

Consideration should be paid to life management programmes and PSA correlations. PSA can provide inputs for the identification of priorities in life management programmes; these programmes will facilitate precise information on the remaining design margins in critical components and structures and their evolution over time which will become a valuable input for PSA.

6.6.2. Structure and contents of maintenance evaluation for life management

Remnant life management programmes should be supported by the basic activities that are described below:

- Selection of important systems, structures, components and populations, according to safety and economic indicators.
- Identification of significant degradations and their evolution over time.
- Evaluation of the effectiveness of the maintenance practices and their adequacy to the basic objective of life management which is conservation, mitigation and/or monitoring of ageing phenomena which affect plant safety and profitability.
- Analysis, selection and implementation of the life management measures in the following areas:
  - Repair, replacement and/or modification in components effectively affected.
  - Modifications to operating procedures to reduce adverse impact where appropriate (e.g.: chemistry of fluids, transients, etc.).
  - Modifications to maintenance practices to make them more effective for mitigating or monitoring the effect of ageing and evolution over time.

In particular, maintenance evaluation and improvement should be supported in proven methodologies using relevant evaluation guides to determine systematically and in detail the weaknesses of each maintenance practice in some of the following areas:

- Appropriate scope and depth to detect and mitigate the degradation
- Suitability of the frequencies and acceptance criteria
- Sufficiency of information generated for evaluation purposes
- Formal and updated procedures.

6.6.3. Regulatory aspects

The involvement of the regulatory body in maintenance and life management activities differs from country to country according to the existing national legal system.

Generally, the regulatory body's interest is focused on maintenance of safety related systems, components and structures which can primarily affect the safety of an NPP.

Operational experience and aspects and results of maintenance and in-service inspection have lead regulatory bodies to ask for life monitoring of selected safety related systems, components and structures; this has lead to an early recognition of degradation and to the adoption of proper corrective actions to remove and/or to mitigate the ageing phenomena.

In addition, regulatory bodies assess a quality assurance programme which covers all aspects associated with maintenance activities in utilities.

It should be in the interest of a regulatory body that any changes in maintenance programme of safety related systems, components and structures be supported by appropriate analyses (e.g. PSA) to confirm that the design license basis is not affected.

6.6.4. Integration of programmes

There are substantial advantages in integrating the activities of these maintenance evaluation programmes for life management with other programmes imposed by regulatory requirements, the aim of which is to ensure efficient maintenance in key plant systems — a guarantee of safety. Moreover, maintenance improvement programmes for life management go hand in hand with the maintenance optimization programmes (e.g., cost reduction and reliability centered maintenance programmes) as they are complementary in their objectives, scopes, methodologies for identification of critical areas and in the parameters they use as indicators.

For these reasons, all the maintenance related programmes discussed above far from being mutually exclusive should be implemented in parallel and fully integrated in order to solve overlaps and gain the benefits resulting from their synergies, such as:

- long term performance improvement
- common monitoring tasks and tools
- operation and maintenance records, statistics, trends, etc. to be shared
- co-ordinated definition and design of the information systems (databases, information supports and sources, etc.).

Even though the goals and required personnel expertise for these programmes are different, it is strongly recommended to co-ordinate and integrate tasks and logistics so that the maximum benefit can be obtained.

7. FINAL REMARKS

Current maintenance programmes have been developed during the operating lives of NPPs using engineering and deterministic considerations such as *defence in depth, functional performance based on accident analyses,* and *manufacturers recommendations*. The advent of modern PSA techniques which use the *risk significance* concept and industry operating experience has opened up many possibilities to improve maintenance strategies. PSA insights have highlighted changes which can be applied to maintenance practices to improve public health and safety by reducing the risks associated with nuclear plant operations.

Increasing competitiveness and liberalization of electricity generation are putting emphasis on operating plants with reduced costs. And clearly, maintenance has a role to play in reducing costs. The objective is to optimize cost effectiveness up to the point where other overriding considerations come into play. If there are certain points beyond which maintenance costs cannot be reduced without jeopardizing safety, decision-makers should always bear in mind that safety should not be compromised in order to achieve cost reduction targets.

Annex
PAPERS PRESENTED AT THE TECHNICAL COMMITTEE MEETING

REGULATORY CONTROL OF MAINTENANCE ACTIVITIES IN ARGENTINE NUCLEAR POWER PLANTS

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Abstract

The main maintenance objective is to assure that the safety features of structures, components and systems of nuclear power plants are kept as designed. Therefore, there is a direct relationship between safety and maintenance.

Owing to the above mentioned, maintenance activities are considered a relevant regulatory issue for the Argentine Nuclear Regulatory Authority (ARN).

This paper describes the regulatory control to maintenance activities of Argentine nuclear power plants. It also addresses essential elements for maintenance control, routine inspections, special inspections during planned outages, audits and license conditions and requirements.

1. INTRODUCTION

The Law N° 24,804 "Nuclear Activity National Law" defines the regulatory activity's scope and gives to Nuclear Regulatory Authority the responsibility for the nuclear activities control and regulation referring to radiological and nuclear safety issues. Maintenance activities are considered important to plant safety so it is one of the relevant regulatory issues.

The Regulatory Standard AR 3.7.1 - "Documentation to be submitted to the Regulatory Authority prior to the commercial operation of a nuclear power plant", requires the presentation of the installation Maintenance Program within one month prior to the request of an operating license for full-power operation.

The operating license of the argentine NPP's is granted by Regulatory Authority and, it establishes that degradation of components, equipment and systems shall be prevented by means of adequate preventive and predictive maintenance. Besides, such license requires the implementation of both in-service inspection and surveillance programs.

2. NPP'S REGULATORY CONTROL

The regulatory control in each NPP is performed by two on-site inspectors in charge of inspection regarding radiological and nuclear safety to ensure that plants are operated in accordance to regulatory requirements and license conditions. In case of inspection activities requires specialized expertise, it is foreseen that other specialists of the Regulatory Authority supporting and supplementing on-site inspectors activities.

The regulatory activities related with NPP's maintenance are mainly focused on the safety-related systems, and includes: selecting the safety related maintenance activities, assessing the applicable procedures and work instructions and, witnessing such activities by regulatory inspectors.

The above mentioned regulatory activities are divided into regulatory inspections, audits and evaluations. Regulatory inspections are subdivided into routine inspections and special inspections (non-routine inspections).

2.1. Maintenance routine inspections

Routine inspections are performed during plant normal operation by on-site inspectors, emphasizing control on safety-related systems. The main routine inspections activities are the following:

- Procedures control review.
- Controlling that all maintenance activities are carried out in such a manner that the radiation exposure of site personnel is kept as low as reasonably achievable (ALARA).
- Witnessing during maintenance works and post maintenance testing.
- Verifying compliance of preventive and predictive maintenance program. Checking that frequencies of maintenance - with the procedures applied - are performed in accordance with such program.
- Periodic tests follow-up: The plant periodic tests are performed in accordance with the surveillance program. Such periodic tests are carried out on safety systems to check that components availability is maintained all the time. The Regulatory Authority has implemented an updated data base of periodic tests that includes the component performance during such tests.
- Regulatory requirements follow-up.
- During plant construction works progress follow-up: stored components condition, mechanical assembly, electrical assemblies and civil works.

2.2. Maintenance special inspections

Special inspections are performed by both on-site inspectors and specialists of Regulatory Authority. Such inspections are basically performed during planned outages and in case of abnormal events occurrence. The main special inspections activities are the following:

- Planned outages:

During planned outages the inspection activities are similar those routine inspections. Additional regulatory control in this case of both fulfillment in service inspection program and design modification are included.

- Abnormal events:

In case of abnormal event occurrence an inspection team is organized to review: corrective actions, root cause analysis and lessons learned.

2.3. Audits

Audits are prepared on a "check-list" based on: applicable documentation analysis and assessment, audit team members expertise and both specialists and on-site inspectors recommendations.

Upon the audit completion, the audit team issued an audit report that includes: findings, strengths, weaknesses, observations and recommendations of the audited activities. Then, as required, follow up audits to verify the finding related corrective actions taken by the utility will be performed.

2.4. Evaluations

Evaluations consists in the analysis and assessments of data resulting from routine and special inspections, audits, operational experience and abnormal event occurrences. Such evaluations involve the use of deterministic and probabilistic methods, computer codes, termohydraulic analysis, reactor kinetics, and reliability calculations, etc. The main evaluation activities related with maintenance are the following:

- Abnormal event assessment occurred at both argentine and foreign NPP's.
- Operating experience assessments.
- Radiological safety assessment to detect weaknesses in practices and to propose measures to reduce personnel doses (ALARA).
- Periodic test: procedures assessments and review of acceptation criteria
- Assessment of preventive, predictive and corrective maintenance activities.
  This activity includes the evaluation of the scope maintenance works and criteria applied. Results trend to evaluate component performance and aging effects are analyzed.
- Definition and implementation of performance indicators.
- Design modifications and backfitting assessment.
- Assembly procedures assessment.
- Commissioning procedures assessment.
- Regulatory requirements.

3. MAINTENANCE ACTIVITIES PERFORMED BY UTILITIES

The NPP's maintenance program is fundamentally based on manufacturer recommendations, operating experience, safety analysis and engineering judgment. The licensee's preventive and predictive maintenance programme is defined establishing the scope, methods to be implemented, the planning activities and the applicable controls in accordance with the following considerations:

- Maintaining and improving reliability and availability of components, equipment and systems

- Reducing failures to minimize outages.

- Reducing maintenance costs.

- Reducing doses by applying adequate techniques and procedures.

- Collecting historical maintenance data from all plant components to evaluate component performance.

- Assessing operational parameters of equipment, components and systems for early detection failures.

4. REVIEW OF MAINTENANCE REGULATORY POLICY

Considering the recognized dependency between maintenance and plant safety, that argentine regulatory philosophy is based on performance-based regulation (non-prescriptive regulation) and regulatory applications of PSA methodology, the Regulatory Authority decided to face a reviewing process of the maintenance regulatory policy.

The overall objective of such reviewing process is to improve the maintenance activities regulatory control. The effort will be focused on monitoring the results of the maintenance activities, assessing the evaluation equipment performance carried out by utility and verifying the safety assessment before programming the maintenance activities schedule.

The above mentioned reviewing process is in progress. However, at present it is possible to comment that the following issues have been highlighted:

- Need to issue a specific maintenance regulatory standard based on monitoring results of maintenance activities.

- Maintenance related indicators: Presently the Regulatory Authority is working in a regulatory project aimed at defining the performance indicators that include preventive, predictive and corrective maintenance and will be used to assess the maintenance programs effectiveness. Some of the issues below are being discussed:

    - Number of deficiency reports.
    - Number of pending deficiency reports.
    - In service inspection programme compliance.
    - Maintenance re-working
    - Spare parts availability applied to safety systems.

- Encourage the use of Probabilistic Safety Assessment (PSA) applications and Reliability Centered Maintenance (RCM) by the licensees. Such tools are useful for maintenance optimization.

5. CONCLUSIONS

The maintenance activities are regulated through regulatory standards. license conditions and limiting condition for operation. To verify the above mentioned compliance, both on-site inspectors and specialists personnel inspect. audit and evaluate the NPP's maintenance activities.

The maintenance regulatory policies review in progress, will produce:

- Strengthen the licensee's maintenance self-monitoring system related with its effectiveness.

- Use of performance safety indicators to assess the maintenance programs effectiveness.

- Encourage utilities to use maintenance optimization tools as probabilistic safety assessment and reliability centered maintenance.

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MAINTENANCE AND TEST STRATEGIES TO OPTIMIZE NPP EQUIPMENT PERFORMANCE

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Abstract

This paper proposes an approach to maintenance optimization of nuclear power plant components, which can help to increase both safety and availability. In order to evaluate the benefits of preventive maintenance on a quantitative basis, a software code has been developed for component performance and reliability simulation of safety related nuclear power plant equipment. A three state Markov model will be introduced, considering a degraded state in addition to an operational state and a failed state.

Introduction

In the field of safety culture, maintenance activities are more and more considered to play a major role. The past history has shown that in many cases nuclear power plant equipment failures could have been avoided with an appropriate maintenance schedule. Avoiding failures means not only an increasing state of safety, but also reducing costs due to forced component outage times and repairs. The following paper shows a possible approach to maintenance optimization of nuclear power plant components, which can help increasing both safety and availability.

In standard reliability and probabilistic safety assessment (PSA) modeling, only two states for each component are considered: success and failure. Yet in many cases there is no immediate transition from a success state to a failed state. Components may show significant degradation, indicating a more serious failure to occur. Component degradation can, however, in many cases be detected and corrected. Thus a total loss of function, which could seriously impact plant reliability and safety, can be avoided. With the inclusion of degraded states, especially scheduled, preventive maintenance activities turn out to have remarkable benefits regarding component performance. Preventive maintenance can be seen as a scheduled periodic activity with the objective to repair any degraded or failed equipment and to assure the proper functioning of the equipment after it has been maintained.

In order to evaluate the benefits of preventive maintenance on a quantitative basis, a software has been developed for component performance and reliability simulation of safety related nuclear power plant equipment. A three state Markov model will be introduced, considering a degraded state in addition to an operational state and a failed state. The degraded state occurs when the component's performance degrades below some threshold value defining normal designed performance.

The three state Markov model allows not only the immediate transition from an operational state to a failed state, but also the transition from an operational state to a failed state through a degraded state. The component may of course remain in a degraded state, depending on the degradation mode. With the inclusion of a degraded state, the advantages of preventive maintenance actions can be explicitly quantified.

The application of this three state Markov model is, however, not restricted to components in standby; simulations for running components may also be performed. Whereas for standby components catastrophic failures only can be detected through demand, test or maintenance, it can be assumed that for running components in most cases catastrophic failures will be detected immediately. But for both standby and running components, a degraded state may remain undetected for a certain amount of time. In such a case, preventive maintenance can correct degradation before the transition to a catastrophic failure.

Once a degraded state is defined, optimal preventive maintenance intervals can be evaluated, depending on the components' reliability parameters. This is one of the main objectives of the reliability simulation. In order to simulate the three state Markov model, the following reliability parameters need to be known or estimated:

- catastrophic failure rate;
- degraded failure rate;
- average repair time;
- (allowed) outage time due to preventive maintenance.

The application of the reliability simulation to safety related pumps show the state probabilities and the availability depending on the reliability parameters and on the preventive maintenance interval

An interesting and also very important study is the dependence of component reliability and availability on failure detection probabilities. Whereas it can be assumed that a catastrophic failure will always be detected by preventive maintenance or by surveillance/test, the detection of degraded failures may not always be possible by surveillance/test, based, however, on the degraded failure modes. The inclusion of failure detection probabilities plays a major role for components with high degraded failure rates. It can be shown that a high detection rate of degraded failures increases both reliability and availability.

Another very important issue is a comparison between scheduled and unscheduled maintenance. It can be shown that planned maintenance activities lead to a significant higher operational state probability than unscheduled maintenance activities, even if the scheduled and the unscheduled maintenance activities have the same cumulative outage time for a certain time period. This emphasizes the importance of an appropriate periodic maintenance schedule, which should be determined depending on the failure history over the past operating period.

As an extension, this model can also be applied to power production systems, where a degraded state can be defined in terms of lower output. In this case the economical consequences of the components' reliability parameters can be shown quantitatively.

The equipment performance and reliability simulation was applied to nuclear power plant safety related standby pumps. The data available from the maintenance records are the total operating period, the number of catastrophic failures, the number of degraded failures and the average repair time. The allowed preventive maintenance outage times were assumed to be in the range of the average repair times. The range of the operating periods is long enough to yield reasonable results (5 – 11 years). However, future investigations may show changes in pump reliability performance, which may also be based on changing maintenance procedures.

Operational State Probabilities

In the following the operational state probability for various standby pumps are compared. For graphical reasons, the complementary probability, 1 - operational state probability, is plotted in Fig. 1 and Fig. 2.

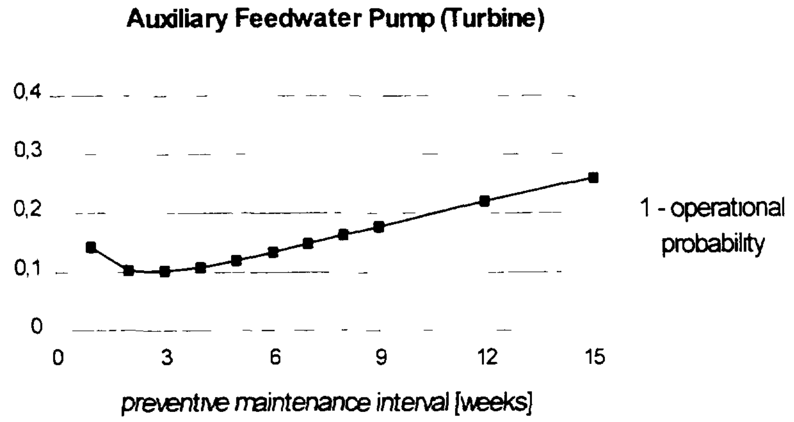

Fig. 1. Operational state probability of the auxiliary feedwater pump (turbine) of Plant I. For graphical reasons, the complementary probability, 1 - operational state probability, is plotted. It can be seen that after the maximum of the state probability at a preventive maintenance interval of 3 weeks, the operational state probability decreases only slightly with increasing preventive maintenance interval.

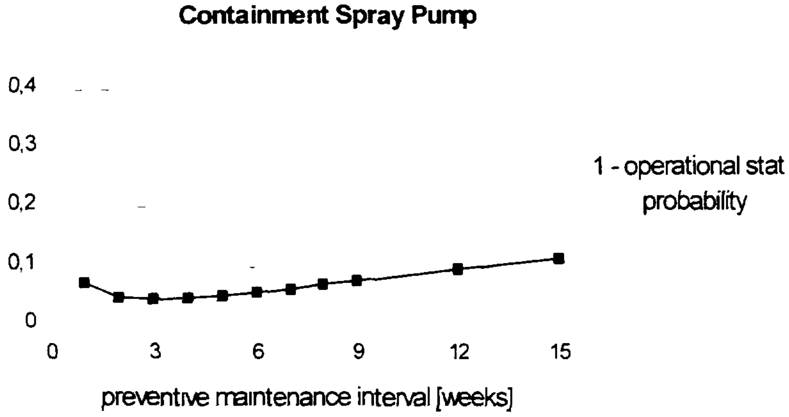

Fig. 2. Operational state probability of the containment spray pump of Plant I. For graphical reasons, the complementary probability, 1 - operational state probability, is plotted. It can be seen that the operational state probability decreases slightly with increasing preventive maintenance interval.

Availability

During one preventive maintenance interval plus the planned outage due to preventive maintenance activities, the maximum availability which is achievable is:

$$\frac{\textit{maintenance interval}}{\textit{maintenance interval} + \textit{preventive maintenance outage time}}$$

In this case it is assumed that no outages due to failures occur within one preventive maintenance interval.

Taking into account pump failures and in the case of failure the unavailability to perform its function upon demand, the pump availability can be written in the form:

$$\frac{\textit{maintenance interval} - \textit{time in failed state}}{\textit{maintenance interval} + \textit{preventive maintenance outage time}}$$

Figures 3 and 4 show the maximum availability and the availability taking into account the outage time due to failures of sample pumps. The gap between the maximum availability and the "real" availability is shown, depending on the preventive maintenance interval. With constant maintenance duration, the maximum availability is always increasing with preventive maintenance. The "real" availability shows a maximum, indicating the optimum balance between planned preventive maintenance outages and outages due to failures. The following decrease of the availability is caused by the increasing influence of the failure outages. As can be seen in the following figures, a high failure rate is reflected by a rapid decrease in the availability.

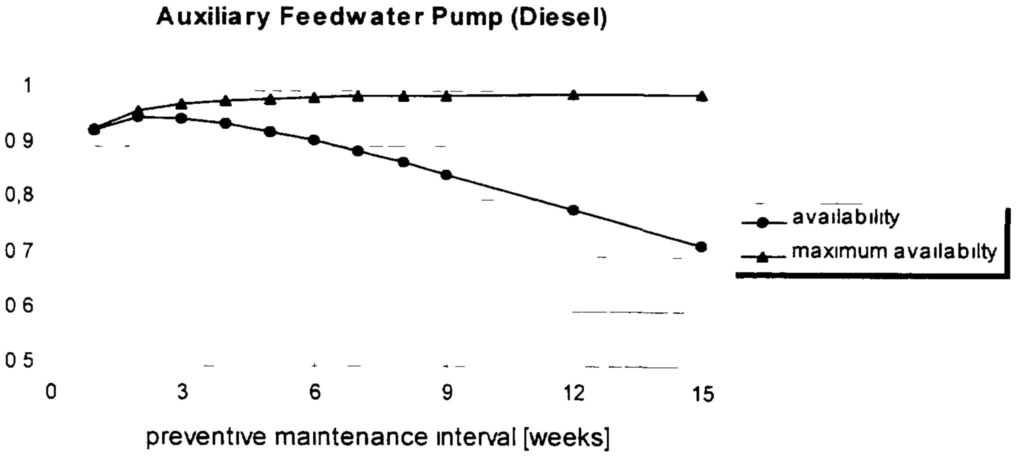

Fig. 3. Availability of the auxiliary feedwater pump (diesel). The maximum availability considers only outages due to preventive maintenance activities. In addition to preventive maintenance outages, the "real" availability also takes into account outages due to failures The rapid decrease of the availability reflects the high pump failure rate.

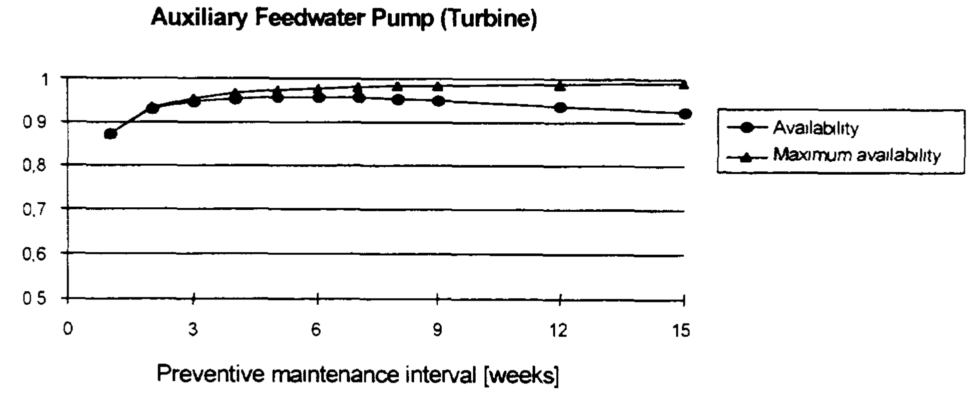

Fig. 4. Availability of the auxiliary feedwater pump (turbine). The maximum availability considers only outages due to preventive maintenance activities. In addition to preventive maintenance outages, the availability also takes into account outages due to failures. The slow decrease of the availability reflects the low pump failure rate.

Tests between Preventive Maintenance Activities

Under certain circumstances it is valuable to perform tests between preventive maintenance activities. In most cases, tests are much easier to perform and less time consuming than preventive maintenance activities and are therefore more cost-effective. The inclusion of regularly performed tests may allow the extension of the preventive maintenance interval, thereby hardly affecting equipment reliability and availability.

Tests are in most cases less costly than maintenance activities. Benefits obtained through an appropriate inclusion of tests within a regular preventive maintenance interval are therefore not only of probabilistic nature, improving the component's reliability performance, but also of financial terms. Allowing the extension of the preventive maintenance interval through the inclusion of tests, financial resources could be saved and allocated for spare parts or improved training courses for maintenance personnel. However, tests performed too frequently may also have a negative impact on plant reliability performance, as a higher number of demands, either due to tests or emergency, may accelerate component aging. In our work, we concentrate on probabilistic safety assessment calculations and do not consider financial aspects. A future extension of this work might be to include financial considerations in addition to probabilistic calculations of component reliability performance.

In our model, tests are assumed to be capable of failure detection, but not of detecting equipment degradation. Fig. 5 shows a possible trajectory for equipment that undergoes regular preventive maintenance with a predetermined maintenance interval and with tests performed within each preventive maintenance interval.

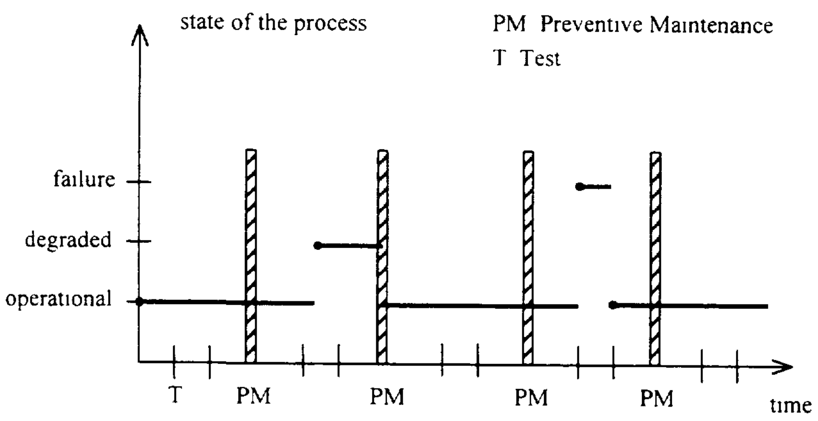

Fig. 5. A possible trajectory of equipment with the states operational, degraded and failed is shown for a Markov process that is interrupted not only by preventive maintenance, but also by tests performed during the preventive maintenance interval. In our model, tests are assumed to be capable of failure detection, but not of detecting any kind of degradation.

We assume that a test is capable of detecting a failed state. This assumption is reasonable because the main objective of performing tests is to identify the functional performance of a component. For the time being, any kind of degradation will in our model not be detected through a test:

$$Test: \begin{cases} operational\ state & \rightarrow\ operational\ state \\ degraded\ state & \rightarrow\ degraded\ state \\ failed\ state & \rightarrow\ operational\ state \end{cases}$$

One of the most remarkable benefits obtained by tests performed between preventive maintenance activities is the reduction of the failed state probability and the increase in the component's availability.

In order to show the effect of regularly performed tests within each preventive maintenance interval, 2 pumps were selected for the numerical simulation, quantifying changes in component reliability performance compared to a maintenance strategy not considering tests:

- *the Auxiliary Feedwater Pump (Diesel) of Plant I,*
- *the Fire Pump of Plant II.*

In Table 1 and Table 2 the pump input parameters for the numerical reliability performance simulation are listed.

Table 1 shows the 2 selected pumps' transition rate from the operational state to the failed state, denoted by the catastrophic failure rate, the transition rate from the operational state to a degraded state, denoted by the degraded failure rate, and the transition rate from a degraded state to the failed state, denoted by the degraded to catastrophic failure rate.

Standby Pump	catastrophic failure rate	degraded failure rate	degraded to catastrophic failure rate
	/ 1,000,000 h	/ 1,000,000 h	/ 1,000,000 h
Plant I: (operating period = 5 years = 43800 h)			
Auxiliary Feedwater Pump (Diesel)	23.0	757.8	803.8
Plant II: (operating period = 6 years = 52560 h)			
Fire Pump	19.3	159.3	197.9

Table 2 shows the 2 selected pumps' average repair duration and average preventive maintenance duration. The average preventive maintenance durations were estimated to be in the range of the respective pump's average repair durations.

Standby Pump	average repair duration [h]	average preventive maintenance duration [h]
Plant I: (operating period = 5 years = 43800 h)		
Auxiliary Feedwater Pump (Diesel)	6.6	12.0
Plant II: (operating period = 6 years = 52560 h)		
Fire Pump	6.7	12.0

For the 2 pumps it is assumed that the test duration is 3 hours.

Auxiliary Feedwater Pump (Diesel) of Plant I

Test Interval = 3 Weeks

As already stated, the reduction of the failed state probability and therefore a higher operational state probability and an increase in the pump availability are among the tangible benefits regarding the inclusion of tests within the preventive maintenance interval. In Fig. 6, the two component reliability performance approaches, case (a), considering only preventive maintenance activities and case (b), including tests performed every 3 weeks within the preventive maintenance interval, are compared with respect to the pump's failed state probability. As can be seen, the reduction of the failed state probability through the inclusion of tests is a remarkable positive impact obtained by the regular performance of tests. With tests performed every 3 weeks, the increase in the failed state probability with the extension of the preventive maintenance interval becomes almost negligible. However, it should be noted that tests performed too frequently may effect component performance, an aspect which is not taken into consideration in our calculations.

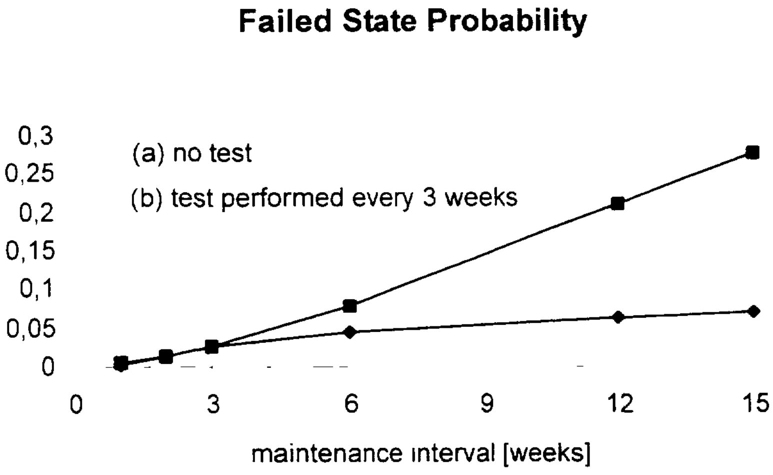

Fig. 6. Failed state probability for 2 two component reliability performance strategies: In case (a) only preventive maintenance activities are considered, whereas in case (b) tests performed every 3 weeks within the preventive maintenance interval are included. It can be seen that the inclusion of tests significantly reduces the failed state probability, thus allowing the extension of the preventive maintenance interval without a major increase in the failed state probability.

The reduction of the failed state probability is equal to an increase in the probability that the pump will be found at the operational state upon demand. This can be seen in Fig. 7, where the operational state probability is compared for the 2 cases (a) and (b). With the preventing maintenance interval exceeding 6 weeks, the gap between the operational state probabilities becomes significantly large, indicating that for the auxiliary feedwater pump the inclusion of tests would contribute to a better component reliability performance.

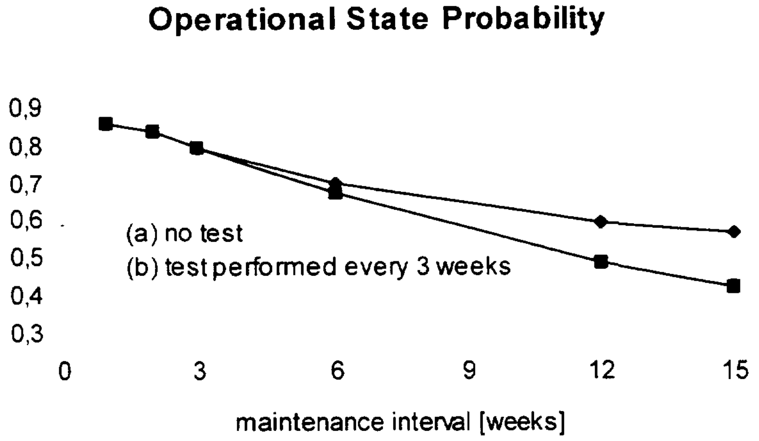

Fig. 7. Operational state probability for 2 two component reliability performance strategies: In case (a) only preventive maintenance activities are considered, whereas in case (b) tests performed every 3 weeks within the preventive maintenance interval are included. It can be seen that the gap between the state probabilities becomes significantly large with the preventive maintenance interval exceeding 6 weeks.

An interesting issue is the availability of the auxiliary feedwater pump depending on the component performance strategies. Comparing the availability for case (a) and case (b), it can be seen that through the inclusion of tests, the availability stays almost constant with increasing preventive maintenance interval, whereas a dramatic decrease in the availability occurs for the performance strategy not considering the regular performance of tests. As already mentioned, the increase in the pump's availability through the inclusion of functional tests is consistent with the decrease in the failed state probability. The auxiliary feedwater pump availability, compared for case (a) and case (b), can be seen in Fig. 8.

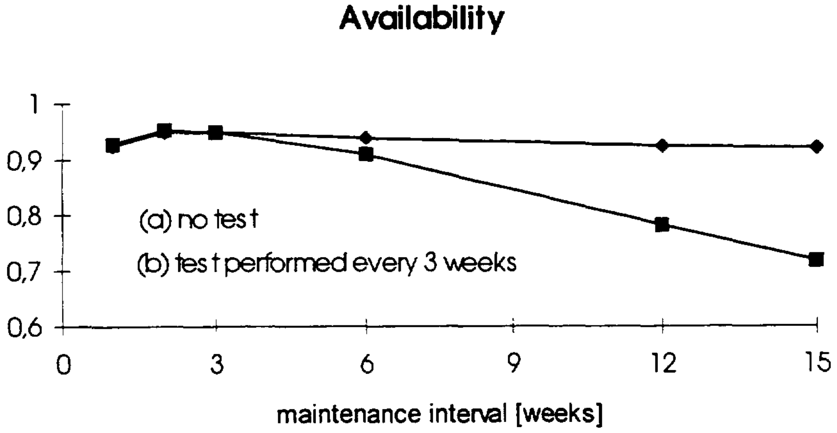

Fig. 8. Comparison of the availability of the auxiliary feedwater pump of plant I for case (a) and case (b). In case (a) only preventive maintenance activities are considered, whereas in case (b) tests performed every 3 weeks within the preventive maintenance interval are included. The difference is significant, emphasizing the benefits obtained by the inclusion of tests within the preventive maintenance interval.

Comparison of Different Test Strategies

We have now compared 2 different test strategies to improve reliability performance of the auxiliary feedwater pump of plant I. We have shown that the inclusion of tests within the preventive maintenance interval contributes significantly to a better pump reliability performance, increasing the pump's operational state probability and availability.

In addition to the 2 test strategies we have already evaluated, a test interval of 6 weeks is now considered. In the following we directly compare 3 different test strategies for the auxiliary feedwater pump of plant I. For this purpose the preventive maintenance is kept constant, being 12 weeks. The test interval of the 3 different test strategies ranges from 3 to 6 weeks:

(1) Test Interval = 3 Weeks

(2) Test Interval = 4 Weeks

(3) Test Interval = 6 Weeks

Comparing the failed state probability in Fig. 9, it can be seen that with the test interval extending from 3 to 6 weeks, the failed state probability becomes twice as high, increasing from 0,06 to 0,12.

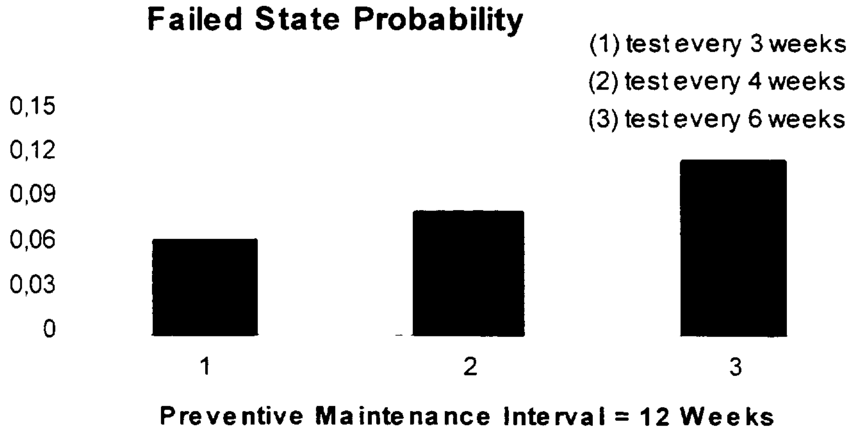

Fig. 9. Compares the failed state probability of the auxiliary feedwater pump of Plant I for 3 different test intervals. It can be seen that with the test interval extending from 3 to 6 weeks, the failed state probability becomes twice as high, indicating that the variation of the test interval remarkably influences reliability performance of the auxiliary feedwater pump.

In Fig. 10 the operational state is compared for the 3 different test intervals at a preventive maintenance interval of 12 weeks. It can be seen that the operational state probability decreases with the extension of the test interval.

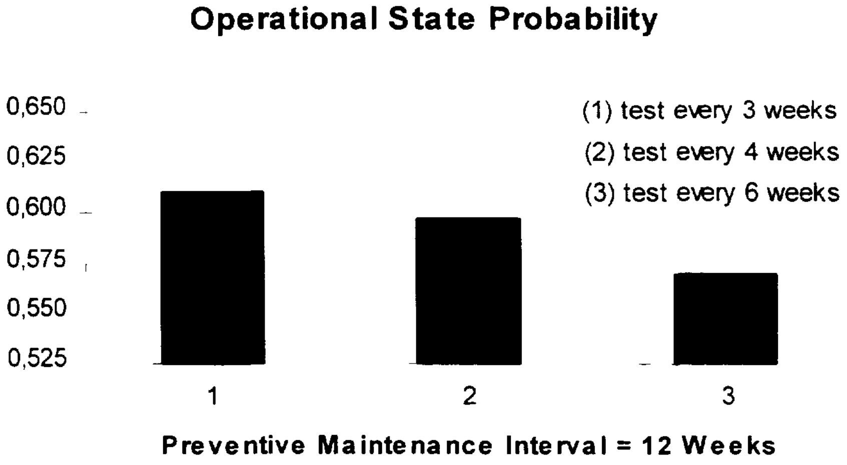

Fig. 10. Compares the operational state probability of the auxiliary feedwater pump of Plant I for 3 different test intervals It can be seen that with the extension of the test interval, the operational state probability increases, caused by the increase of the failed state probability.

The increase in the failed state probability directly affects pump availability, which can be seen in Fig. 11. Comparing the availability for the 3 different test intervals at a preventive maintenance interval of 12 weeks, one observes that the availability decreases with the extension of the test interval. Summarizing the effects of the extension of the test interval on component reliability performance, it can be stated that the higher the failed state probability, the more beneficial is the inclusion of tests into a maintenance optimization strategy.

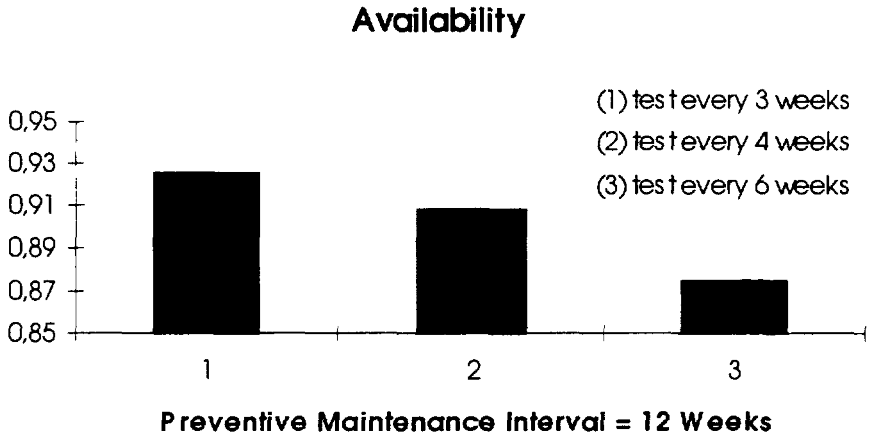

Fig. 11. Compares the availability of the auxiliary feedwater pump of Plant I for 3 different test intervals It can be seen that with the extension of the test interval, the availability decreases, caused by the increase in the failed state probability shown in Fig. 9.

Tests with the Capability of Detecting Component Degradation

In the previous chapter we have included the performance of tests within the preventive maintenance interval. We assumed that tests are only capable of detecting component failures. However, for some kinds of degradation this assumption may not be true. Based on the degradation mode, a component degradation may or may not be detected.

In order to include the possibility of detecting component degradation through tests, we will introduce degradation detection probabilities through tests. Thus it is possible to correct a degraded state through tests, bringing the component back to the operational state.

For the auxiliary feedwater pump (diesel) of plant I, we compared 2 different cases:

- *25% degradation detection probability through test*
- **50%** degradation detection probability through test

<u>25% Degradation Detection Probability Through Tests</u>

In Fig. 12 the effect of including a degradation detection probability through tests is shown on the degraded state probability. Tests being capable of detecting pump degradation with a probability of 25%, denoted by case (a), are compared with tests not being capable of detecting a degraded state, denoted by case (b). The test interval is kept 4 weeks in our numerical simulations. The difference between the degraded state probabilities is obvious, indicating a higher efficient component reliability performance through the inclusion of degradation detection probabilities through tests.

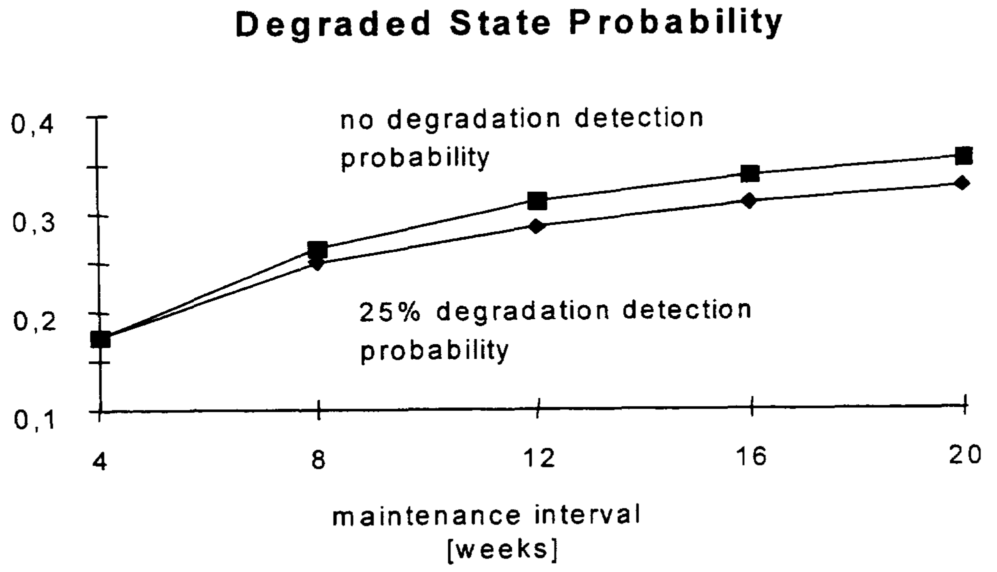

Fig. 12. Comparison of the degraded state probability of the auxiliary feedwater pump of plant I: In case (a), tests are capable of detecting a degraded state with a probability of 25%, whereas in case (b) tests are not capable of detecting any mode of degradation. The test interval is 4 weeks. The difference between the degraded state probabilities is significant, being 5% at a preventive maintenance interval of 20 weeks.

In Fig. 13 the operational state probability is compared for case (a) and case (b).

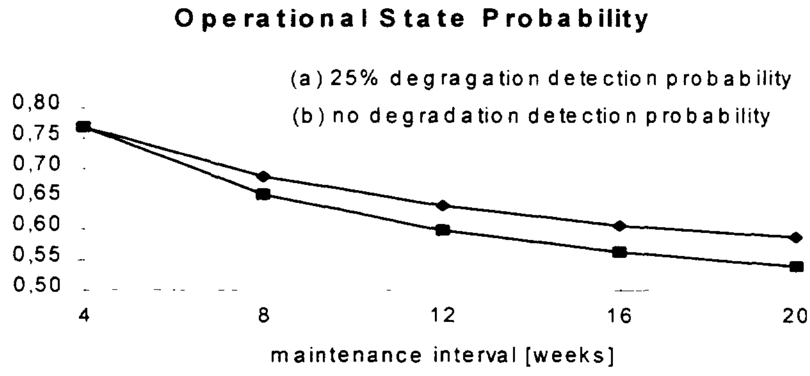

Fig. 13. Comparison of the operational state probability of the auxiliary feedwater pump of plant I: In case (a), tests are capable of detecting a degraded state with a probability of 25%, whereas in case (b) tests are not capable of detecting any mode of degradation. The test interval is 4 weeks. The increase in the operational state probability reflects the decrease in the degraded state probability shown in Fig. 12.

In Fig. 14 the failed state probability is compared for the 2 cases (a) and (b). The difference between the failed state probabilities is not so obvious compared to the degraded and the operational state probabilities shown in Fig. 12 and Fig. 13. The rather small influence on the failed state probability is an indirect benefit obtained by the possibility of detecting degradation through tests, thus decreasing the probability of the pump transiting from a degraded state to the failed state.

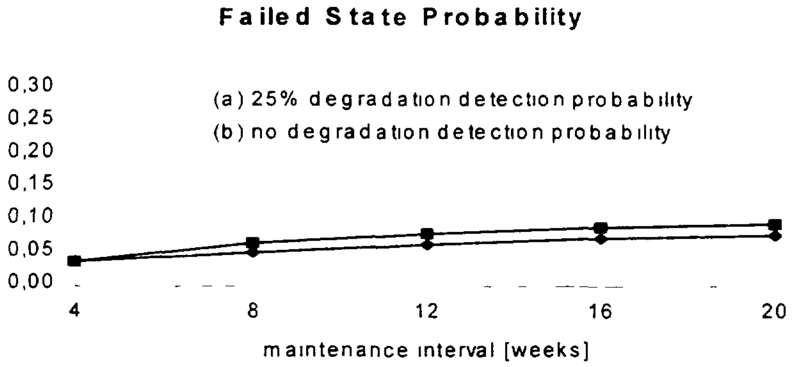

Fig. 14. Comparison of the failed state probability of the auxiliary feedwater pump of plant I: In case (a), tests are capable of detecting a degraded state with a probability of 25%, whereas in case (b) tests are not capable of detecting any mode of degradation. The test interval is 4 weeks. The difference between the failed state probabilities is an indirect benefit of the inclusion of degradation detection probabilities through tests, thus decreasing the probability of the pump transiting from a degraded state to the failed state. However, the effect is not very significant.

The effect on the pump availability is almost negligible, as can be seen in Fig. 15. In this connection it should be mentioned that being in a degraded state, the pump is assumed to be still available. As the inclusion of degradation detection probabilities through tests mainly affects the degraded state probability, the availability does not show major changes.

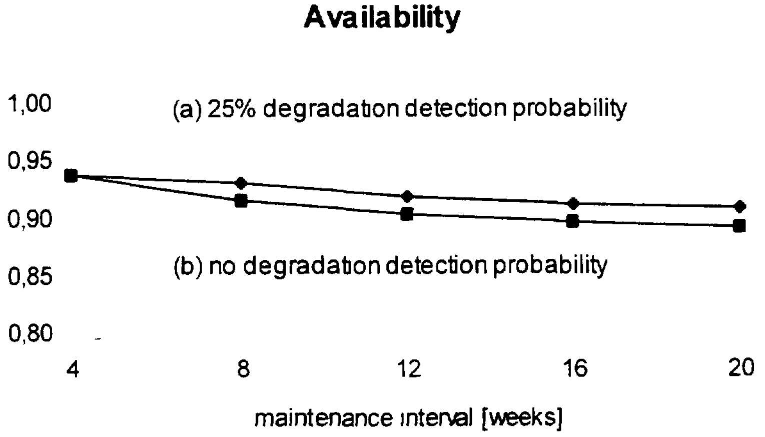

Fig. 15. Comparison of the availability of the auxiliary feedwater pump of plant I: In case (a), tests are capable of detecting a degraded state with a probability of 25%, whereas in case (b) tests are not capable of detecting any mode of degradation. The test interval is 4 weeks. The difference between the availabilities is rather small, as the inclusion of degradation detection probabilities through tests mainly affect the pump's degraded state probability.

50% Degradation Detection Probability Through Tests

Let us now increase the degradation detection probability to 50%, being equal of detecting every second pump degradation. Again we will evaluate the effect of the inclusion of a degradation detection probability through tests on component reliability performance. We of course expect now a greater influence on the state probabilities, especially on the degraded and the operational state probability. In Fig. 16 the degraded state probabilities are shown for tests being capable of detecting pump degradation with a probability of 50%, now denoted by case (a), and for tests not being capable of detecting a degraded state, denoted by case (b). The test interval is again 4 weeks.

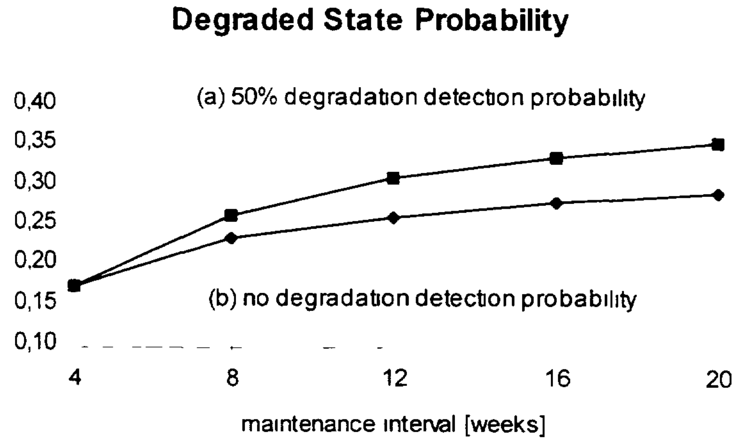

Fig. 16. Comparison of the degraded state probability of the auxiliary feedwater pump of plant I: In case (a), tests are capable of detecting a degraded state with a probability of 50%, whereas in case (b) tests are not capable of detecting any mode of degradation. The test interval is 4 weeks. The difference between the degraded state probabilities is even more significant than in the case of tests being capable of detecting pump degradation with a probability of 25%, shown in Fig. 12. At a preventive maintenance interval of 20 weeks, the difference between the degraded state probabilities is already 10%.

In Fig. 17 the operational state probability is compared for case (a) and case (b). As can be seen, with the inclusion of tests being capable of detecting degraded states with a probability of 50%, the operational state probability becomes significantly higher. The increase in the operational state probability reflects the decrease of the degraded state probability shown in Fig. 16.

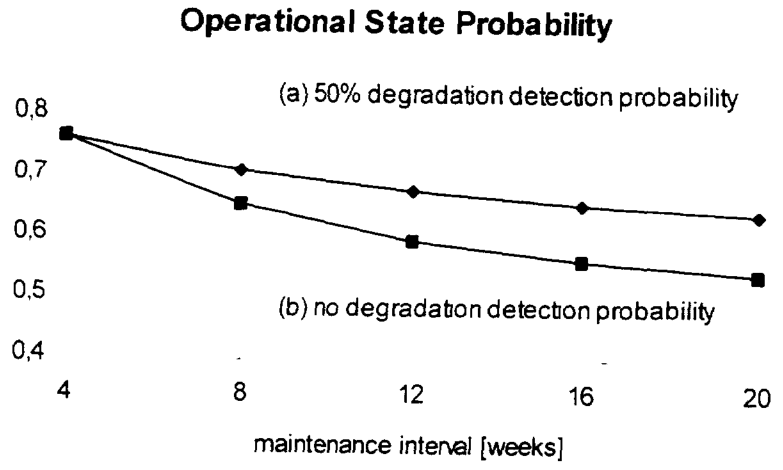

Fig. 17. Comparison of the operational state probability of the auxiliary feedwater pump of plant I: In case (a), tests are capable of detecting a degraded state with a probability of 50%, whereas in case (b) tests are not capable of detecting any mode of degradation. The test interval is 4 weeks. The increase in the operational state probability, now even more remarkable than for tests being capable of detecting degradation with a probability of 25%, reflects the decrease of the degraded state probability shown in Fig. 16.

Again, the failed state probability is compared for the 2 cases (a) and (b), as can be seen in Fig. 18. The difference between the failed state probabilities is not so obvious compared to the differences between the degraded and the operational state probabilities shown in Fig. 16 and Fig. 17. However, the difference between the failed state probabilities has increased due to the higher probability of degradation detection through tests, thereby decreasing the probability of the transiting from a degraded state to the failed state.

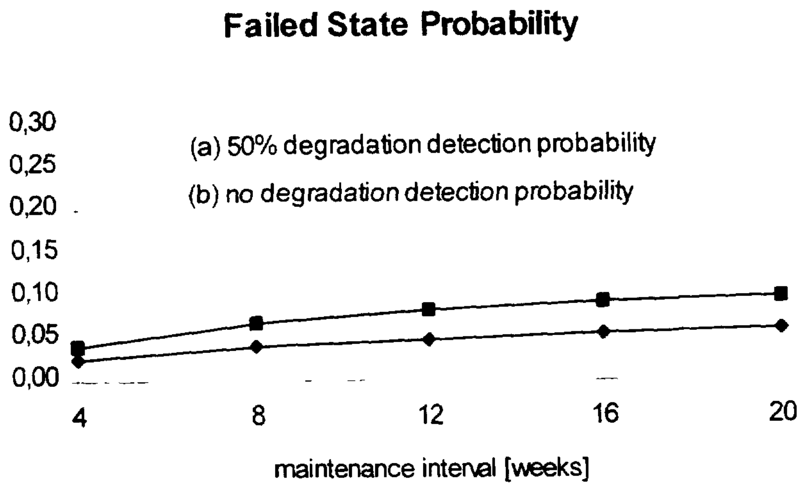

Fig. 18.Comparison of the failed state probability of the auxiliary feedwater pump of plant I: In case (a), tests are capable of detecting a degraded state with a probability of 50%, whereas in case (b) tests are not capable of detecting any mode of degradation. The test interval is 4 weeks. The difference between the failed state probabilities, now more significant than in the case of a degradation detection probability of 25% through tests, is an indirect benefit of the inclusion of degradation detection probabilities through tests, thus decreasing the probability of the pump transiting from a degraded state to the failed state.

Comparing the pump availabilities for the 2 cases (a) and (b), it can be seen in Fig. 19 that the difference between the availabilities is now more significant, which reflects the changes of the failed state probability shown in Fig. 18.

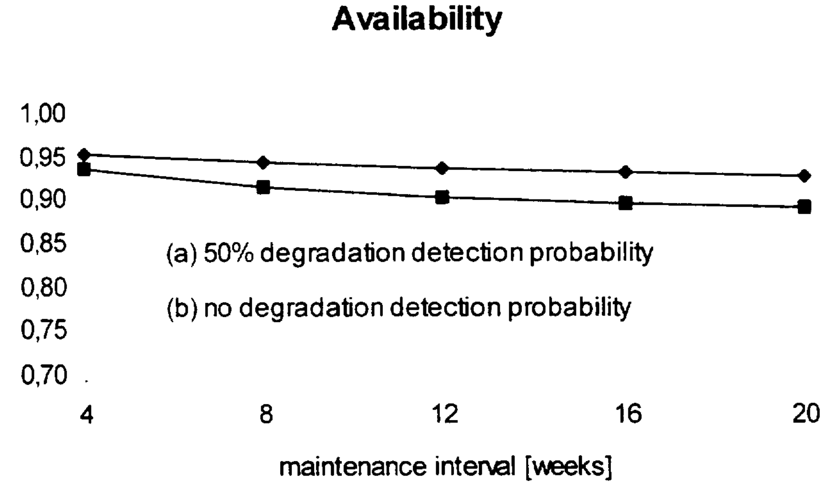

Fig. 19. Comparison of the availability of the auxiliary feedwater pump of plant I: In case (a), tests are capable of detecting a degraded state with a probability of 50%, whereas in case (b) tests are not capable of detecting any mode of degradation. The test interval is 4 weeks. The difference between the availabilities is still small, but more significant than in Fig. 15, where the degradation detection probability through tests is only 25%.

Let us now summarize the effects of the inclusion of degradation detection probabilities through tests: Keeping the preventive maintenance interval constant to be 20 weeks, Fig. 20 shows the degraded state probability for (1) tests not capable of detecting any degraded states, (2) for tests with a degradation detection probability of 25% and 50%. The differences are significant and indicate the benefits that could be obtained with increasing degradation detection probabilities through tests.

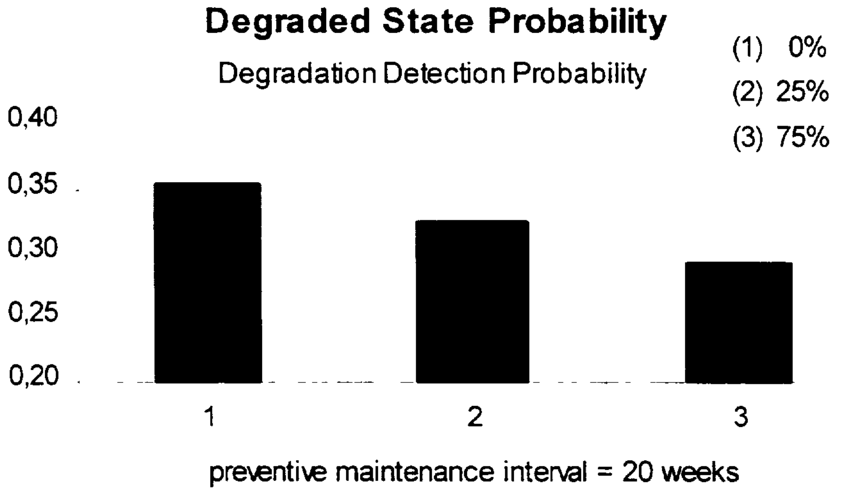

Fig. 20. Degraded state probability of the auxiliary feedwater pump (diesel) of plant I at a preventive maintenance interval of 20 weeks for (1) tests not capable of detecting any degraded state, (2) tests with a degradation detection probability of 25% and (3), tests with a degradation detection probability of 50%. The test interval is 4 weeks. The differences are significant, thus indicating the benefits that could be obtained with increasing degradation detection probabilities.

Comparing the pump availability for tests not capable of detecting any degraded state and for tests with a degradation detection probability of 25% and 50%. It can be seen in Fig. 21 that the differences between the availabilities are less significant than the differences between the degraded state probabilities in Fig. 20. As already mentioned, the increase in the pump availability is an indirect benefit obtained by the inclusion of degradation probabilities through tests, thus decreasing the probability of pump failure after being in a degraded state.

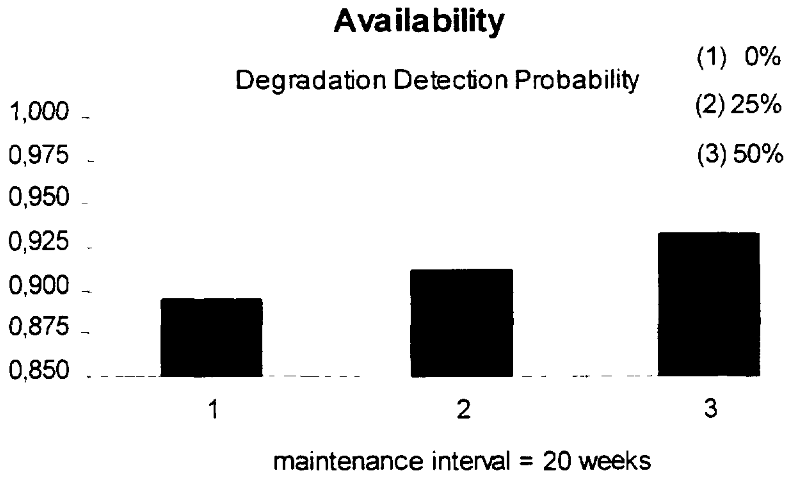

Fig. 21. Comparison of the availability of the auxiliary feedwater pump (diesel) of plant I at a preventive maintenance interval of 20 weeks for (1) tests not capable of detecting any degraded state, (2) tests with a degradation detection probability of 25% and (3), tests with a degradation detection probability of 50%. The test interval is 4 weeks. Compared to the differences between the degraded state probabilities in Fig. 20, the difference between the availabilities is smaller, caused by the assumption that the pump is still available in a degraded state.

Unscheduled Maintenance Activities

In this chapter we want to compare component reliability and availability performance for scheduled preventive maintenance activities and for unscheduled maintenance activities. The objective of this comparison is to show that a scheduled maintenance strategy yields to higher component reliability and availability than a randomly distributed performance of maintenance, comparing both maintenance approaches for the same average maintenance interval and the same maintenance duration.

For our unscheduled maintenance activities, the maintenance intervals are exponentially distributed. With p_m being the probability that a component will be maintained, and λ_m being the inverse of the average maintenance interval, we can write

$$p_m = e^{-\lambda_m t} .$$

Each maintenance interval is determined through a random number generator, with the average maintenance interval being the inverse of λ_m

$$\textit{average maintenance interval} = \frac{1}{\lambda_m}.$$

The occurrence of maintenance is according to the theory of Markov processes. However, we assumed the maintenance duration to be constant and not being exponentially distributed.

In the following figures the 2 maintenance performance strategies are compared for the containment spray pump of plant I. In case (a) preventive maintenance is performed regularly, keeping the interval between preventive maintenance activities constant. In case (b) the maintenance intervals are exponentially distributed. It should be emphasized that for unscheduled maintenance activities both the average maintenance interval and the maintenance duration are the same as for a scheduled preventive maintenance performance.

To explicitly show the effect of unscheduled maintenance activities compared to scheduled preventive maintenance, a maintenance interval of 15 weeks is chosen in Fig. 22 to compare the operational state probability. For a scheduled maintenance strategy, the operational state probability is 8% higher than for unscheduled maintenance activities.

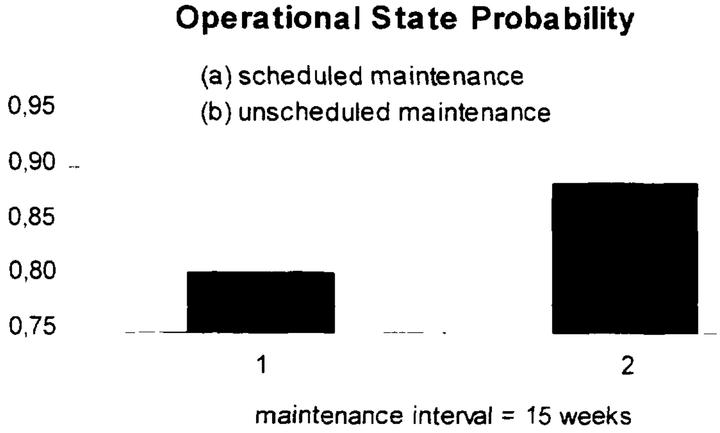

Fig. 22. Comparison of the operational state probability of the containment spray pump of plant I of scheduled and unscheduled maintenance performances at an maintenance interval of 15 weeks. In case (a) preventive maintenance is performed regularly, keeping the interval between preventive maintenance activities constant. In case (b) the maintenance intervals are exponentially distributed. In the case of regular preventive maintenance, the operational state probability is 8% higher, thus indicating the advantage of regular preventive maintenance over unscheduled maintenance activities.

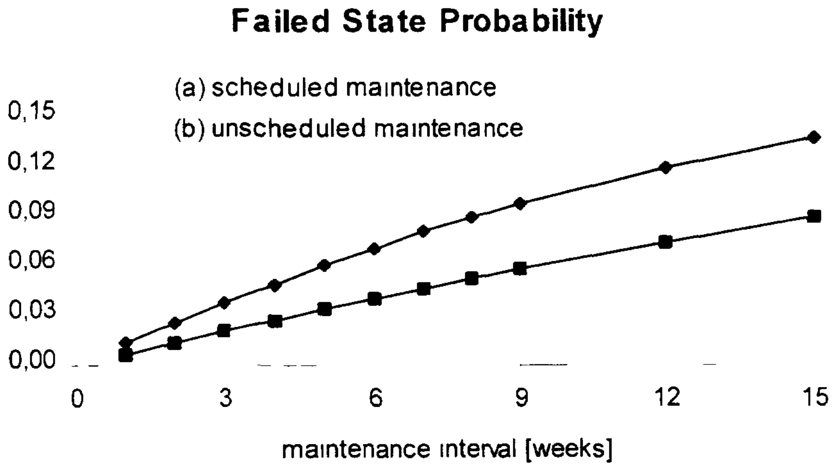

Fig. 23. Comparison of the failed state. The difference between the state probabilities for scheduled and unscheduled maintenance performance is significant.

Comparing the degraded state probability for scheduled and unscheduled maintenance performances, one can see in Fig. 24 the increasing difference between the state probabilities with increasing preventive maintenance interval. The degraded state becomes significantly lower in the case of scheduled maintenance activities and once again emphasizes the advantage of performing regular preventive maintenance.

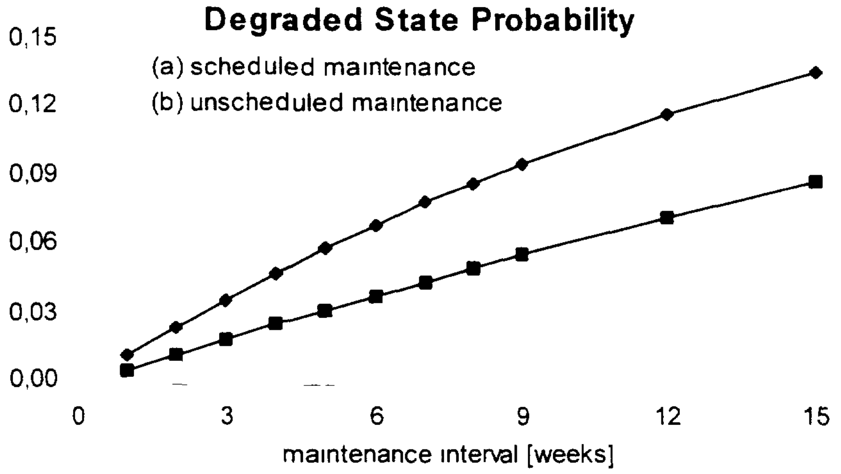

Fig. 24. Comparison of the degraded state probability of the containment spray pump of plant I for scheduled and unscheduled maintenance performance strategies. In case (a) preventive maintenance is performed regularly, keeping the interval between preventive maintenance activities constant. In case (b) the maintenance intervals are exponentially distributed. Comparing the degraded state probability, one can see that the difference between the state probabilities becomes significantly large with increasing maintenance interval.

CONCLUSIONS AND RECOMMENDATIONS

The results show that particularly in large-scale and complex technical facilities appropriate maintenance schedules contribute substantially to equipment and plant reliability, availability and safety. The studies focused primarily on the role of preventive maintenance activities in the optimization of both equipment reliability and availability performance, considering different preventive maintenance frequencies and durations.

An essential role in the optimization process of preventive maintenance is the information available from the databases. Data records should include, *inter alia*, failure frequency and mode, repair time, average repair time, maintenance frequency and duration, test interval and duration. Comprehensive data records are essential for establishing appropriate policies to improve plant performance.

Needless to mention that also economic considerations have to be included in any processes of maintenance optimization. In any maintenance policies, the cost of maintenance personnel, spare parts, repairs as well as maintainability on component and system level play a vital role. However, the recent past has shown that appropriate preventive maintenance has become a substantial contribution to plant reliability, availability and safety.

RCM VS. TRADITIONAL MAINTENANCE PRACTICE — SOME PROBLEMS WHEN RCM IS INTRODUCED FOR THE FIRST TIME

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Abstract

The presentation is focused on some problems which arise when RCM is introduced for first time. These problems were identified from EQE-Bulgaria experience when RCM meters were discussed for first time at number of plants and organizations.

Mainly the following problem areas are covered:

- The first group of problems arise from the nature of RCM.

  The RCM is referred to as a '^{3}new concept", "new philosophy", new maintenance system", etc. On the other hand, RCM has nothing new - it is a structure of established maintenance techniques. It leads to some difficulties to explain what exactly is the "new" to a managerial body and to a staff of a plant which has its own established maintenance system running for years.

  This group of problems covers in particular the following points:

  - RCM definition vs. "Traditional" Maintenance definition;
  - The understanding about the relationship between inherent reliability and the desired performance of an asset;
  - Difficultes to prove the advantages of RCM and the benefits of its implementation in case RCM is not a country-wide practice.

- The second problem area is the terminology. The RCM concept is developed and described mainly in reference documents written in English. The direct translation into a certain language introduces some misunderstanding especially if there is no established terminology in the corresponding national language. The first major task is to get clear understanding about relationship between RCM terminology and the terms used in the current maintenance practice defined in the national standards.

- The next problem area is the availability or unavailability of relevant reliability plant specific data. We found that people tend to trust more the "numerical" (quantitative) results rather than the qualitative ones. So efforts should be spent to explain and even to prove the advantages and disadvantages of both qualitative and quantitative approaches as well as to show clearly the relation between qualitative and quantitative results.

Some other problems which are common when RCM is introduced and carried out for first time are outlined briefly.

Paper

The paper is focused on some problems which arise when RCM is introduced for first time These problems were identified from EQE-Bulgaria experience when RCM meters were discussed for first time at number of plants and organizations At the very beginning it should be noted that all discussions we have had up to now were led at plants with long term operational experience and therefore with its own established maintenance systems running for years It should be mentioned also that RCM is a new field of maintenance activities in our country

The difficulties one meet when RCM is discussed for first time could be subdivided into the following problem areas

(1) The first group of problems arise from the nature of RCM.

The RCM is referred to as a "new concept", "new philosophy", "new maintenance system", etc On the other hand, RCM has nothing new - it is a structure of established maintenance techniques It leads to some difficulties to explain what exactly is the "new" to a managerial body and to a staff of a plant which has its own established maintenance system running for years

These difficulties arise from the fact that the transition from the "Traditional" Maintenance to RCM requires that the maintenance people are having to adopt completely new ways of thinking and acting, as engineers and as managers In this aspect the problems cover in particular the following points

(a) RCM definition vs. "Traditional" Maintenance definition

To start with, every physical asset is put into service to fulfill a specific function or functions It follows that when we maintain an asset, the state which we wish to preserve must be one in which it continues to fulfill its intended functions So we come to the definition of the maintenance in "Traditional" Maintenance sense

> ***Maintenance: Ensuring that physical assets continue to fulfill their intended functions***

From the RCM point of view the maintenance is defined as follows

> ***Reliability-Centered Maintenance: a process used to determine the maintenance requirements of any physical asset in its operating context***

As it becomes clear from both definitions, the key words are "*the operational context of the assets*"

On one hand as it is well known, the traditional maintenance practice is based on the use of scheduled maintenance programs The scheduled maintenance programs on their side are based mainly on the concept that every item on a piece of complex equipment has a "right age" at which complete overhaul is necessary to ensure safety and operating reliability

It directly leads us to the question of the relationship between equipment functional failures, failure modes, failure effects and failure consequences Here we faced the first problem - to come to an agreement with the plant staff (especially the managerial body) that the scheduled overhauls at many cases might not have significant effect on the overall reliability of a complex item (unless the item has a dominant failure mode) and that there are many items for which there is no effective form of scheduled maintenance Initially it was a bit surprising that such problem could arise at all because there is a lot of evidence from the operational experience that it is exactly the case

(b) What actually RCM means?

First of all it should be noted that this question is connected to some extend to the problem of definitions and translation which is discussed in the next section On the other hand, RCM as a term which has two aspects (from the point of view if our experience)

(i) It was found that usually when people say "RCM" they mean "A maintenance directed towards improving reliability of equipment, system and a plant as a whole

(ii) The experience shows that in the discussions an special emphasis should be done on the other aspect of the RCM definition, i e that RCM is a maintenance policy which is based on the current reliability state of the equipment

(c) Inherent Reliability vs. Desired Performance

The two performance standards associated with every function are the desired performance and the inherent reliability If the performance which we want any asset to deliver is within its inherent capabilities, then maintenance can help to achieve the desired performance On the other hand, if the desired performance exceeds the built-in capability, no amount of maintenance can deliver the desired performance This is shown in Figures 1a and 1b

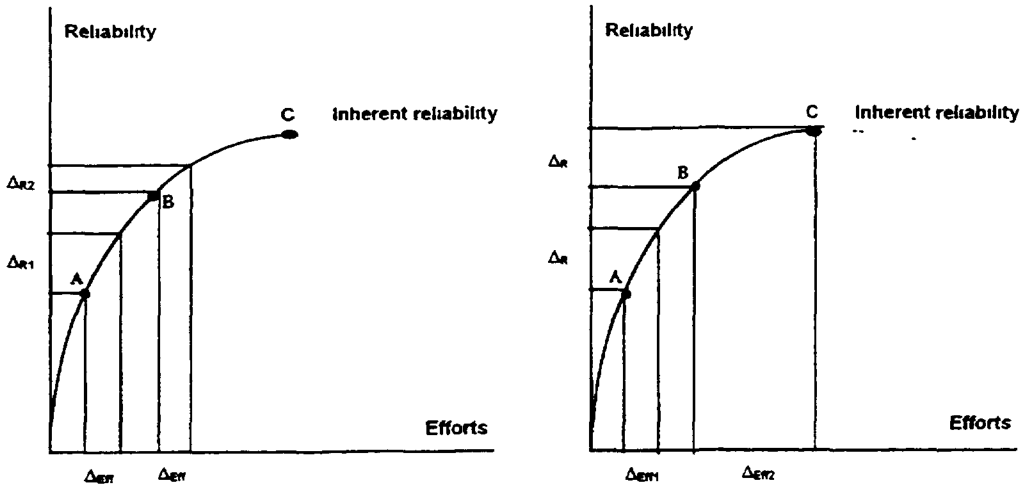

Figure 1a

Figure 1b

The distinction between what is wanted an item to do and what it can do in fact lies in the in the base of many disputes with the plant people This happens because operational staff tends to think in terms what they want out of each asset while in terms of maintenance the question is what that very asset can to do

It should be mentioned that this contradiction between maintenance and the production people always exists, but in case one discuses RCM matters for first time at the some plants it might become quite a problem

(2) The second problem area is the terminology.

The RCM concept is developed and described mainly in reference documents written in English The direct translation into a certain language introduces some misunderstanding especially if there is no established terminology in the corresponding national language (for example, such terms like "operate to failure", "total productive maintenance", etc do not have direct correspondence in Bulgarian standards) The problems which arise so me time are due to the fact that the translator very often seeks to find the corresponding terms in his native language not paying enough attention to the meaning

So, the first major task is to get clear understanding about relationship between RCM terminology and the terms used in the current maintenance practice defined in the national standards

(3) The next problem area is the availability or unavailability of relevant reliability plant specific data.

In the course of the discussions it was found that people tend to trust more the "numerical" (quantitative) results rather than the qualitative ones On the other hand usually there are not relevant (from PSA and RCM point of view) reliability plant specific data which could be used directly and often the efforts should be directed to the qualitative part of the mentioned analyses

So efforts should be spent to explain and even to prove the advantages and disadvantages of both qualitative and quantitative approaches as well as to show clearly the relation between qualitative and quantitative results

(4) Some other problems

Some other problems which are common when RCM is introduced and carried out for first time could be outlined briefly as follows

- In case RCM is not a country-wide practice difficulties could arise to prove the advantages of RCM and especially the benefits of its implementation The benefits of applying the RCM concept can only be measured in a long term perspective The concept generally have to sold based on experience from others, and the mere belief in the concept philosophy,
- Some problems could take place in the course of initial talks with a plant staff if at the very beginning of the discussions it is not stated clearly what is the difference between RCM analyses and RCM Program Misconception that Decision Diagram is RCM also could be mentioned,
- Difficulties arise to assess significance of failure consequences

etc

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CANDU PLANT MAINTENANCE - RECENT DEVELOPMENTS

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Abstract

CANDU units have long been recognized for their exceptional safety and reliability. Continuing development in the maintenance area has played a key role in achieving this performance level. For over two decades, safety system availability has been monitored closely and system maintenance programs adjusted accordingly to maintain high levels of performance. But as the plants approach mid life in a more competitive environment and component aging becomes a concern, new methods and techniques are necessary. As a result. recent developments are moving the maintenance program largely from a corrective and preventive approach to predictive and condition based maintenance. The application of these techniques is also being extended to safety related systems. These recent developments include use of reliability centred methods to define system maintenance requirements and strategies. This approach has been implemented on a number of systems at Canadian CANDU plants with positive results. The pilot projects demonstrated that the overall maintenance effort remained relatively constant while the system performance improved. It was also possible to schedule some of the redundant component maintenance during plant operation without adverse impact on system availability. The probabilistic safety assessment was found to be useful in determining the safety implications of component outages. These new maintenance strategies are now making use of predictive and condition based maintenance techniques to anticipate equipment breakdown and schedule preventive maintenance as the need arises rather than time based. Some of these techniques include valve diagnostics. vibration monitoring. oil analysis. thermography. Of course, these tools and techniques must form part of an overall maintenance management system to ensure that maintenance becomes a living program. To facilitate this process and contain costs, new information technology tools are being introduced to provide system engineers with current system performance trends as well as historical records. This paper discusses the experience gained in CANDU plants with the application of these maintenance tools and the results achieved to date. New technologies being developed by Atomic Energy of Canada Limited for CANDU plants will also be discussed.

1. INTRODUCTION

With the increasing pressure to reduce costs and improve capacity factors, development of a systematic approach to defining maintenance and inspection requirements is a strategy frequently adopted by utilities. The experience in Canada is no different. Originally the maintenance programs established at Canadian CANDU plants were largely based on manufacturer's recommendations and experience at similar plants. The program consisted primarily of time based preventive maintenance developed from manufacturers recommendations as well as experience feedback from other plants. Updates and optimization of the program was largely based on operating experience and lessons learned from plant events or problems. Often. increasing the frequency of overhauls was the solution to equipment problems. This strategy led to a growing volume of preventive maintenance work with little optimization. In the early 1990's increasing unit incapability due to equipment problems and concerns related to safety related system availability led to a reassessment of the maintenance programs and the development of a maintenance model which promoted a systematic process to identify maintenance, surveillance and inspection requirements.

2. MAINTENANCE DECISIONS AND APPLICABLE TOOLS/METHODS AND CONSTRAINTS

In 1993, Ontario Hydro nuclear plants initiated a program to optimize the maintenance program using a reliability centered maintenance methodology. The same basic approach was used by each plant although the degree of detail varied mainly in the analysis phase. Each plant conducted pilot projects to assess the benefits of the approach and then extended the approach to other systems. The process currently in use at the Bruce site consists of the following steps:

1) Starting from a complete list of plant systems, a subset of 20 systems were identified for the first phase of the project. These systems were critical to plant safety as well production. Examples of systems included in the original list are the reactor shut down system, the heat transport system, reactor moderator system, emergency boiler cooling. The importance of the system was established based on an assessment of the system impact on the plant key effectiveness of safety, reliability and cost based on the following considerations:

    - Importance to plant safety and reliability
    - System reliability/availability
    - Maintenance resource requirement
    - Maintenance complexity

2) For each system on the list, the system boundary and sub systems were established. This boundary is important because it delineates the components which must be considered in the analysis and ensures that all support systems such as air, power supplies and service water are taken into consideration. The system design, operating and maintenance historical information was collected for the analysis.

3) The system functions were then identified for each sub system. A functional failure analysis identified the functional failures which could impact on the system fulfilling it's primary role. The components associated with these failures were then identified along with their dominant failure mechanisms. The probabilistic safety analysis was used where the system was modeled to ensure all critical functions and components important to safety were included.

4) Based on the potential and actual failure mechanisms an effective preventive task was established. External experience from other plants, vendor information and regulatory requirements were taken into consideration in arriving at an improved program. For each component, the strategy consisted of one or a combination of the following strategies:

    - Condition Monitoring & Diagnostics

- Time Based maintenance
- Periodic Overhaul
- Component replacement
- Surveillance and testing
- Visual inspections
- Design change
- Run to Failure

5) The recommended program was then compared with the program in existence at the time and the program adjusted accordingly. The revised system surveillance and maintenance requirements were then incorporated into the station documentation for execution. Changes to plant maintenance call ups were revised or updated, the operator routines were redefined and the maintenance procedures updated. Each analysis was documented in a maintenance basis document.

Optimization of the maintenance program using this process requires expertise in the following three main areas:

- Experience with the process of systematically analyzing systems, system functions, failure mechanisms and developing effective strategies.

- A good overall understanding of the system design intent, safety design requirements, system operation

- Experience with maintenance and surveillance of plant components including the technology available for plant maintenance.

The level of effort required varied from system to system dependent on complexity and the quality of information. The detail analysis portion of the process on average required about 6-8 person-weeks of effort and the implementation of the program required about the same level of effort. Using a streamlined analysis process and an expert panel review , the level of effort was down to a few days.

Implementation of this overall program led to the following changes in component and system surveillance and maintenance:

- In many cases, the system surveillance carried out by operators during routine plant rounds were modified. Overall a 20% increase in operator surveillance workload was required to implement the program on 18 major systems at Bruce A.

- The preventive maintenance program changes resulted in no significant change in overall maintenance effort but in some cases up to 40% of the tasks were deleted and new ones added as a result of the analysis.

- In some cases new technology was adopted to eliminate the need for periodic overhauls and make use of condition based monitoring techniques by the maintenance staff. In addition, condition based maintenance surveillance tasks such as vibration monitoring, thermography, oil analysis, system parameters were also added.

The benefits arising from optimization of the maintenance program were measured using a number of plant indicators namely:

- System availability for safety systems
- Unit incapability
- Ratio of preventive maintenance to total maintenance
- Compliance with the preventive program
- Backlogs of corrective maintenance
- Results from audits and plant evaluations
- Maintenance Costs

Implementation of the maintenance optimization process on a number of pilot systems resulted in the following:

- The number of maintenance preventable forced outages dropped from 5 per year in 1992 to 2 in 1996 at Bruce A.
- The ratio of preventive maintenance to total maintenance increased from about 40% to over 55% from 1992-1996 at Bruce A.
- The backlog of corrective maintenance dropped from about 350 work orders per unit to about 150 from 1992 to 1996.
- The result of the review generated 55,673 work hours of new maintenance tasks but 54,067 were deleted resulting in negligible change in resource requirements
- The incapability due to fuel handling systems at Pickering dropped from 4.5% to less than 0.5%.
- While the process considered the need for plant modifications, the number of modifications identified through the maintenance optimization process were small.
- Implementation of a condition based maintenance program on Pickering instrument air compressors resulted in cancellation of 40K$ worth of unnecessary annual overhaul and identified a specific defect which was causing major compressor damage requiring complete rebuild.

With these early successes, a standard approach to maintenance optimization is being developed. This approach will be systematically applied on a priority basis to all plant systems.

3. USE OF PSA IN MAINTENANCE DECISIONS

The Probabilistic Safety Assessment was originally developed to assess the probability of major core damage incidents and to ensure that all potential contributing accident sequences were considered. These tools are now being used in support of plant operation and maintenance as risk management tools. The actual applications vary from plant to plant as a function of the type of PSA and the level of detail modeled. Some of the specific applications are discussed below:

3.1 Changes to Safety Related Component Testing Frequency

Under certain circumstances, safety system tests cannot be carried out as scheduled due to maintenance being underway on redundant equipment. These tests are part of an overall surveillance program to demonstrate safety system availability. The PSA model can be used to assess the public safety implication of delaying a safety system test for a given period of time. If the impact on risk is insignificant, then approval from the appropriate authority for the test deferral is obtained.

Under other circumstances, the testing frequency can be optimized using the PSA models. If a component can be tested at a lower frequency thereby reducing wear and tear without impact on safety, then this leads to improved component reliability and reduced maintenance. This approach was used successfully to change the test frequency of steam reject valves and standby generators for example.

3.2 Removal of redundant standby safety related equipment from service

The PSA models are used to assess the public safety impact of removing certain components from service for regular maintenance with the unit on line. This analysis is essential for the development of a set of requirements documented in the plant operating procedures with respect to system and unit configuration. The planning and scheduling of maintenance activities are then governed by these requirements. Under unusual unit configuration, the PSA is also used to analyze a specific unit configuration before other equipment is taken out of service for maintenance reasons.

3.3 Identifying Critical Structures, Systems and Components

The PSA can be used as a source of information for optimizing maintenance through reliability centred maintenance process. Components critical to plant and public safety can be easily identified including the performance requirements such as reliability and availability and the failure mechanisms considered in the analysis.

3.4 Outage Planning Support

The PSA models are used extensively during plant outages to ensure risk is maintained within prescribe limits during maintenance activities. This is particularly necessary in CANDU where the reactor core remains fueled during the outage. At any time during an outage, two different method of heat removal must remain available as well as ability to maintain the reactor shutdown and the containment boundary intact. With the large volume of work scheduled and the number of parallel activities being carried out, the potential exist for unknowingly reducing the defense in depth. Analysis of the outage sequence prior to the outage using the PSA tool and monitoring progress during the outage provides an added verification that omissions have not been overlooked.

While these applications are extremely beneficial, the PSA still has limitations in support of plant operation and maintenance. In particular, the models only address plant and public safety and therefore only include certain safety related systems. For example, systems intended for production such as the reactor control systems are not modeled in detail. For these systems, separate engineering analysis is required. The potential expansion of existing PSA's to probabilistic reliability analysis to include energy production is under consideration.

4. IMPLEMENTATION CONSIDERATION AND TECHNOLOGY DEVELOPMENT

Optimization of plant maintenance program depends on development of technology and use of innovative information systems to facilitate data collection, analysis and storage for easy retrieval. These developments are essential to maintaining a competitive edge. Atomic Energy of Canada Limited has been developing new technologies in a cooperation with the CANDU utilities. Some of the more important ones are summarized for information:

4.1 Condition Based Monitoring Systems

Advanced plant monitoring and display systems gather 1000's of data points continuously from plant systems. Historically, the data collected was strictly monitored and only parameters outside of a specified range were recorded and provided to the unit operator. With advancement in condition based maintenance, the data being collected by the plant monitoring systems can be trended, recorded and provided to the plant maintenance personnel as well as system engineers on a continuous basis through the plant information network. This allows monitoring of plant critical components as well as system performance including chemistry parameters. These systems can be easily backfitted to existing plants to transmit the parameters already monitored by the plant computer display systems. If additional parameters need to be trended, then modifications are required or alternatively, the data can be collected from field measurements by maintenance and operations staff and transmitted to the engineering staff via hand-held computers.

For new plants, a systematic analysis of the critical plant systems and components similar to the one outlined in section 2 above will determine the critical parameters which should be monitored and provisions can be made in the design.

4.2 Historical Data Storage System

To manage the large amounts of data collected during plant operation and maintenance, a historical data storage system is being developed which interfaces with computer display systems as well as the plant work management systems. This tool allows the systems and equipment engineers to easily retrieve records, display trends, and even compare results with other units in order to analyze and resolve problems.

4.3 System Health Monitoring

To facilitate engineering analysis of plant processes, system health monitors are being developed to easily display systems or multiple systems where their performance is interdependent. For example, steam generator chemistry condition depends on the performance of the condensers, feedwater and condensate systems. Another application would to monitor unit thermal efficiency. The analysis of adverse trends requires monitoring of many interrelated parameters. Presenting the information in schematic or flow diagram form facilitates this analysis.

4.4 On Line Instrument Monitoring

Routine calibration and testing of the numerous instruments and transmitters is a significant workload for the maintenance department and could lead to potential errors and unit disturbance. A transmitter accuracy monitoring system has been developed which can alert the operator of instrument drift before it becomes a potential impairment. For a given process parameter, trending the individual transmitter signals and comparing the results between themselves can alert operator of a developing trend. Maintenance can then be scheduled before a failure occurs or a potential impairment due to drift outside of tolerances. This technology can be used to replace the periodic calibration of instruments.

5. CONCLUSIONS AND NEXT STEPS

Based on a pilot projects at a number of CANDU plants, it has been demonstrated that optimization of the maintenance program is well worth the investment and essential for long term competitiveness. The benefits of implementing a systematic process for the analysis of critical systems structures and components and then developing a surveillance and maintenance program range from improved system performance, plant safety, reliability and cost minimization. This optimization must also be carried out in concert with the implementation of supporting new technology to improve plant surveillance. Furthermore, analytical tools such as probabilistic reliability analysis can assist in the development and planning of maintenance activities.

REFERENCES

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THRMS — A PILOT RISK MANAGEMENT SYSTEM

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Abstract

Daya Bay NPP is the first commercial nuclear power plant in China. This plant is interested in and pro-active towards the development and application of PSA and PSA tools. This, together with the support from the IAEA promoted the project "THRMS: the pilot study of risk management system for NPP". The objectives in development and implementation of THRMS includes: survey and discuss on the development of the approaches used in constructing plant risk models; design and study on the realization of a real-time risk management system. An overview of the project is presented in this paper.

1. Background

For the last years a large number of PSAs have been finished world- widely. It is no doubt that PSA is nowadays the appropriate technology and tool that can evaluate and qualify the base-line risk level for the installation. But the standard/general PSA models are still not suitable enough to be used as assistant tools during the daily plant operation. As we know that many changes on components and systems can occur in the plant configurations during the operation phase. These changes can be originated by planned activities like tests, maintenance and repair or by unplanned actions, mainly random events (failures) on components and systems. This results in a fluctuation of the risk level over operating time and is denominated as the "risk profile" (RP) of the installation. Therefore, PSA models must be regularly updated to reflect the changes of plant availability and configuration, and be made compatible with the actual plant status. To resolve this problem, the development of Living PSA technology and the adequate management tools – Risk/Safety monitoring Systems, which are based on Living PSA models and techniques to study and assess the risk and optimize the operation of the installation with respect to a minimal risk level over the operating time, is nowadays of a growing interest.

There exists internationally a large number of Living PSA and risk monitoring systems that are under developed or already in use in different countries. Some are of great interest like ESSM from United Kingdom, SAFETY MONITOR from NUS, SAS and R&R workstation and EOOS from SAIC in the United States, etc. Till now, some risk or safety monitors have been installed for plant daily use, and some of the applications have gained rather wide acceptance from the plant staff as a means to check safety assumptions and provide an objective basis for expediting or rescheduling work activities based on risk significance. Actually the nuclear industry and the regulators have shown a growing tendency for a major usage of PSA in general, and LPSAs and RMS for safety management and plant operation decisions in particular.

The first commercial nuclear power plant in china – Daya Bay NPP starts operating in 1994. Nearly 10 reactor years of operation experiences makes it possible for the plant to turn from going all out in the operation assurance to being able to do something in the operation improvement and optimization in the meantime. In fact, some advancement activities using PSA techniques have been attempted within certain scope recently, e.g. system reliability analyses on DC power systems and plant trip reduction PSA project. The benefit of using PSA technology in daily operation and maintenance management is gradually being recognized by the installation. Moreover, the Daya Bay NPP is scheduling to extend its fuel cycle from 12 months to 18 months, and at the same time to shorten the refueling outage time as far as possible. Hence on-line maintenance activities must be rapidly increased, the corresponding maintenance, test and repair rules must be rescheduled, and the spare parts storage must be rearranged too. Therefore Daya Bay NPP is comparatively active and cooperative in moving forward towards applying and implementing LPSA and even risk-based technology inside the installation.

The above mentioned as well as the support from IAEA promotes the project "THRMS: the pilot study of risk management system for NPP". The objectives in development and implementation of THRMS includes: survey and discuss on the development of the approaches used in constructing plant risk models; design and study on the realization of a real-time risk management system. Overview of the project is presented in this paper.

2. Study on the approaches

A plant specific PSA model is an integral part of a risk management system, that is so called plant risk models. Two fundamental approaches are usually used when establishing risk models suiting for risk management systems based on fault trees and event trees from standard /general PSA and dealing with the models to provide risk information reflecting the changes of plant availability and configuration in a real-time. One is using pre-solved cutsets equation and the other is through the complete solution. The merits and shortcoming of the two approaches are discussed in detail.

- **Cutsets equation**

Using cutsets equation is to re-quantify the pre-solved Boolean cutsets equation of core damage by changing the numerical values according to new configuration. This approach was widely used in the first developed systems due to its fast calculation capability. But the possibility of losing some potential failure modes and cutsets due to the inevitable truncation when generating the Boolean equation influences the accuracy of evaluated results.

Aiming at the weakness, some methods are studied to recover the potential lost and improve the accuracy. Some of them start with multiple Boolean equations: series of Boolean equations respectively under several groups of possible configurations are pre-generated. When plant configuration changes, the equation under the most closest status is selected to be re-quantified. Whereas some others are trying to find out the lost cutsets. For example, in accordance with a certain criterion, some truncated cutsets involving at least one of the failure components are seek for and then supplemented into the Boolean equation to reduce the loss etc.

Since none of these amending methods has resolved the potential loss completely and basically, and with the rapid development of computer technology, using fast fault tree solution algorithms to allow so called complete dynamic solution of the whole risk model is becoming more realizable, this approach is gradually not adopted in the risk monitoring systems developed recently.

- **Complete solution**

Complete solution of the risk models is to modify the fault trees and event trees according to the current configuration change of the plant and then re-quantify the fault trees and event trees. It can furthest ensure the high precision of the evaluation, but if it is not well associated with fast fault tree solution algorithms or logic model optimization techniques, the whole resolving process will be terribly time consuming. Hence, how to bring it into being in software development and in the meantime improve the solution speed as much as possible to ensure the whole evaluations can be performed within minutes becomes the biggest obstacle on the way of developing the re-quantification approach.

Through the discuss of these two approaches, we decide to adopt the re-quantification approach in our pilot study system THRMS. As described above, the two important issues in resolving the problem of solution speed are developing fast fault tree solution algorithms and logic model optimization techniques. Several fast computing codes for Fault Tree analysis are available in the market. Hence, how to simplify and optimize the risk model purposively in case of ensuring the unaltered logic relationships in order to reduce the model size significantly and make it possible to be evaluated in real time has been chiefly studied.

The common simplification methods using modularization or the largest independent sub-trees are usually used after the fault tree models have been constructed. It is not required that the modules or the sub-trees obtained should have physical meanings, and failure of the components inside does not always lead to the failure of the module or sub-tree. Thus the simplification process is separated from the system prototype, and many information contained in the modules or sub-trees cannot be utilized conveniently. With reference to these methods and through the analysis on the configurations of typical safety systems of nuclear power plant as well as the structure of their fault trees, a optimization method of constructing super-component based on "segment" concept is put forward.

So called "segment" is a series of tandem components in the system, which are in charge of one segment function together. The structural feature determines that any of the components inside the segment fails will fail this segment and the influences on the system respectively from the segment or from the components are same. For example, a pump and its upstream and downstream isolation valves can constitute a segment. Either the pump fails or the valves fail closed will fail the segment function of delivering water. Thus, not only the objective of using modules or independent sub-trees to simplify the fault trees can be achieved by constructing super-component based on segment, but also can the corresponding relation between segment failure and component failures be made consistent. Besides the convenience in fault tree model processing – failure of the components can be transferred to the failure of the segment directly, the certain physical meanings of the super-component can be utilized too, especially in case of considering a number of components are unavailable due to the isolation procedure when one of them is in maintenance.

Of course, the independence of super-component based on segment must be ensured. Events that may appear in other places, e.g. common cause failure events and support systems, should be extracted out of the segment and be treated with individually.

Since tens of basic events can be represented by one basic event, size of the models can be significantly reduced and the solution time can be greatly saved. An example of auxiliary feed-water system shows the comparison of computing time.

	Number of BE	Computing time (second) (Number of MCS)		
Truncation value		0	1E-10	1E-9
Before simplification	142	Overflow	114.45 (4446)	44.71 (2096)
After simplification	41	33.8 (585)	3.52 (172)	1.98 (105)

3. Introduction to THRMS

Risk Management system of THRMS is being developed by Institute of Nuclear Energy Technology of Tsinghua Univ. Aim of the software is to work as an assistant tool in risk-based decision making. Its designed tasks include: monitoring and indicating the plant risk level, giving out risk-based advises on operation or maintenance activities, assessing the influences on plant risk level caused by maintenance schedules, recording plant (systems and components) operation history for reference and PSA updating.

THRMS is developed to operate in a Microsoft Windows environment and provide an estimation of the plant risk , through the complete solution of the risk models, rather than pro-solved cutsets.

The complete solution results are required by specification within minutes on a personal computer. The whole software is based on object-oriented designing and programming techniques, and realized with Microsoft Access for relational database, Microsoft Basic for graphical interface and Microsoft C++ for numerical calculations and algorithms used for numerical analyses.

THRMS is designed to support three kinds of users: plant operators, maintenance schedulers and PSA analysts. In order to meet different user needs, THRMS consists of three main sub-systems: Risk Indicator for plant operators, Maintenance Scheduler for maintenance staff and System Configer for PSA analysts and software system administrator.

© Risk Indicator

As mentioned above, Risk Indicator is designed to support plant operators to control plant risk level and to ensure plant safety. Functions that Risk Indicator supports include:

- ♦ Input of detailed information about the changes of plant status. For example, when and for what reason which component is out-of-service or restored etc.
- ♦ Calculation of the instant risk and the respective AOT. Plant risk model will be modified automatically to reflect changes of plant availability and configuration according to the above change information, and then be re-quantified to give out the current risk level and the AOT based on risk significance.
- ♦ Calculation and ranking of the benefit due to component restoration.
- ♦ Information query and output. Three levels of information can be referenced: plant level (baseline and actual risks, current dominant Minimal Cutsets, plant equipment importance, risk profiles etc.), safety systems level (system unavailability, current important components, system importance, system fault tree graph etc.), and components level (location, type, current status (out-of-service or in service), current risk benefits for components out-of-service etc.).

Risk Indicator demonstrates the most important information in an integrated and user-friendly environment. A small "risk indicator panel" indicates current and baseline risk level as well as risk-based AOT both in text and graphic modes. The color-coded status panel displays the operability of the various plant systems based on the equipment currently out of service. Graphical interface binding with internal database makes it easier to surf through all the queries.

© Maintenance Scheduler

For plant maintenance staff, THRMS helps them to make maintenance schedules and evaluate the candidate schedules in the viewpoint of risk. Consequently two modules are developed.

- ♦ Maintenance schedule editor

  Schedule editor allows users to edit or create a maintenance schedule in text or in graphic mode.
- ♦ Maintenance schedule assessor

  Schedule assessor reads the candidate schedule and determines the status of systems or administrative requirements and creates the respective risk profile. The bar-shaped risk profile helps avoiding any unexpected risks (e.g., two trains of a system cannot be out of service at the same time). It allows schedulers to shift planned work to another time period, prioritize equipment to return to service and remain in service, and recalculate the system status and risk profiles.

© System Configer

System Configer provides administrators and PSA analysts some convenient tools to fulfil the software system administration and risk models updating. Some main modules are:

- ♦ Fault tree converter

  Different PSA analysts often use different software to build their fault trees, like IRRAS, NUPRA, CAFTA and RISK SPECTRUM. Various storage formats of fault trees are big troubles to fault tree resources sharing and modifying. It is a hard work to convert fault trees between two different formats. Fault tree converter is designed to import raw fault trees into THRMS database, and in the meantime give them interfaces to be converted between different formats.

- Risk model creator
  Function of risk model creator is to create master logic fault trees based on event trees and fault trees. That is: event trees and the corresponding fault trees are merged into a master fault tree taking core damage event as the top. The logic relationship of the event tree must be correctly represented in the new fault tree.
- Reliability data modification tool
  Component reliability data must be frequently updated according to the accumulated operation experience. This tool allows PSA analysts to modify the reliability data within the database, including failure modes, failure data and so on.
- System configuration
  THRMS supports several kinds of solution algorithms. PSA analysts and administrators can choose which they prefer in this module. Moreover, some software configurations are performed here, for example, user management, plant model chosen, printer setup etc..

4. Conclusion

Through the development of the project "THRMS: a pilot study on risk management system for NPP", we can say that LPSAs and RMS are promising tools to support risk-based plant configuration control and strategy making, helping operators to optimize the operation with respect to a minimal risk level and improve the economic benefits without reducing plant safety level. Nevertheless, more efforts must be made to complete the technology of LPSA and RMS and gain acceptance from more and more plant staff in advance.

TEMELIN SAFETY MONITOR

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Abstract

Temelin NPP is a WWER-1000/320 two unit plant under construction, originally designed according to the standards of the former Soviet Union. After a series of reviews in the 80s, a decision was taken to upgrade the design of Temelin, including the supply of fuel and instrumentation and instrumentation and control system (I&C). Details on the current design and other related safety matters were presented to the nuclear community in a meeting organized by the IAEA in November 1994.

Based upon recommendations of IAEA OSART missions, post TMI requirements and Temelin Risk Audit recommendations it was decided to perform a Probabilistic Safety Assessment within the Temelin PSA Project. The general purpose of this project was to perform systematic examination of the Temelin Unit 1 NPP for severe accident vulnerabilities by performance of a Level 1 and 2 PSA study.

In addition to the completion of Temelin documented living PSA model, the decision was to develop and implement a PSA based software tool able to analyze real and scheduled plant conditions for determining the risk impact of plant configurations and on-line maintenance activities. This paper provides an overview of the key features of the Temelin Safety Monitor, describes its development activities and its current status and intended use at Temelin NPP for PSA applications.

Introduction

The Temelin NPP is a WWER-1000/320 two unit plant under construction, originally designed according to the standards of the former Soviet Union. After series of reviews in the 1980s, a decision was taken to upgrade the design of Temelin, including the supply of fuel and instrumentation and instrumentation and control system (I&C). Details on the current design and other related safety matters were presented to the nuclear community in a meeting organized by the IAEA in November 1994.

At the present time, a significant number of safety improvements are being or have been incorporated into the Temelin design already. Among most significant measures are: replacement of old core and fuel by new WEC VVANTAGE 6 core, new WEC core monitoring system "BEACON", replacement of I&C by new WEC I&C (PRPS, DPS, RLCS, ESFAS), development of new symptom based emergency operating procedures using COMPRO (computerized procedures system), improved MCR and ECR design and TSC development, two additional non safety grade diesel generators supplying AFW, normal charging system and ADV implemented into design, SGs design modified in terms of the primary header cracking and primary to secondary leak flow rate improvement, enhanced batteries life, replacement of rectifiers and inverters, flame-retardant cables replaced by flame-resistant, containment sump screens and common ECCS suction modified, ECCS/RHR heat exchanger material improved, equipment/structures seismic requalification for 0.1g SSE, PORV fitting into the design, etc.

Based upon recommendations of IAEA OSART missions, post TMI requirements and Temelin Risk Audit recommendations it was decided to perform Probabilistic Safety Assessment within the Temelin PSA Project. The general purpose of this project was to perform systematic examination of the Temelin Unit 1 NPP for severe accident vulnerabilities by performance of a Level 1 and 2 PSA study. The work on the Temelin PSA began in 1993 and it was completed by June 1996. The Project was accomplished by a team consisting of NUS Corporation, NPP Temelin PSA staff and other subcontractor project personnel (EQE International, UJV Rez, EGP Prague, etc.) under the overall direction and responsibility of the NUS.

In addition to the completion of Temelin documented living PSA model, the decision has been made at the plant to develop and implement a PSA based software tool analyzing real and scheduled plant conditions for determining the impact of plant configurations and on-line maintenance on actual operational risk level - Scientech Safety Monitor^{TM} 2.0.

Temelin NPP PSA Project Key Features

Scope:	Level 1 - internal initiating events, external initiating events (fire, floods, seismic, others), Level 2, Living PSA (Temelin Safety Monitor)
Operating modes:	Full power, shutdown (outages, refueling)
Supplier:	NUS Corporation, direct involvement of NPP Temelin
Subcontractors:	EGP Praha, UJV Rez (NRI), RELKO, EQE, others
Client:	NPP Temelin
Financing:	CEZ, a.s. - NPP Temelin
Current status:	All models completed, Safety Monitor - ongoing task

Methodology:

The approach used for the PSA project is given by Temelin PSA Project Plan. Temelin PSA model has been developed using standard small event tree/large fault tree linking methodology using the NUPRA code. The event trees are "Plant Damage State" event trees which have been developed with the Level 2 in mind, to give a smooth interface between Level 1 and Level 2.

Quality Assurance:

As it was intended from the project beginning that the results of the study should be incorporated in a living PSA one of the most important issues was quality assurance. The safety guidelines issued by the IAEA and Decree No 436/90 issued by the State Office for Nuclear Safety both indicate that a quality assurance program should be implemented for activities associated with the design and operation of Nuclear Power Plants. Therefore, a Quality Assurance Program and Plan for the performance of the Temelin PSA Project was developed incorporating the elements of an acceptable NUS QA Program designed to meet 10 CFR 50, Appendix B requirements to the extent possible. The key features of the program involves: design, documentation and software control, verification, review of interim and final work products, and software control.

Independent review:

The key area of independent review was performed at three basic levels. NPP Temelin engineers performed a review of all models and documentation to ensure that the details of the model and assumptions conform with what is known about the plant design. At the second level all work products underwent several stages of verification and review. All system models were independently checked by another analyst within the Project team and designated by the Project manager before their incorporation in the final overall model. All calculations, documents, and computer code inputs and outputs were checked for accuracy by another member of the Project team. The verification was done in accordance with NUS Quality Assurance Program. At the third level an independent review by the IAEA was envisaged. Such independent review of the Temelin PSA Level 1 (internal events) model conducted by IPERS team from the IAEA in the frame of IAEA 1995-1996 TC Biennial Program took place in April/May 1995 at Temelin and the second IAEA IPER mission reviewing external events and Level 2 models proceeded in January 1996. The results and recommendations are summarized in the IAEA reports from these IPER missions.

Features and Development Activities of Safety Monitor^{TM} for Temelin NPP

Temelin PSA staff intends to support actively plant staff in analysis of day-to-day issues related to the areas like:

- Assessment of modifications (design, operation, testing, procedures, etc.)
- Tech specs issues (AOTs, STIs)
- Operating and maintenance strategies based on risk minimization
- Outage Risk Management
- Precursor Analysis

To achieve such day-to-day support of the plant staff requires a dynamic and flexible use of plant specific current PRA models, which is not realistic because of following reasons:

- Extensive scope of PRA models (thousands of BEs, and MCSs)
- Special knowledge of PRA techniques, software and for all the plant specific PRA model(s) is required
- Number of PRA models could potentially exist - Level 1/2, Shutdown, External Events
- Reflection of a current plant status/configuration in the PRA model is time consuming, often requiring large number of PRA model modification steps (model extension, house event settings, CCF and HEPs modification, etc.)
- Quantification process is running for a certain time period itself, depending on the PRA hardware, software used and scope of the plant specific model
- The interpreting of the results obtained requires knowledge of PRA techniques again

Therefore, plant PRA staff decided to extend the PSA project and to implement a real-time risk calculation tool at Temelin analyzing both real and scheduled plant conditions for determining the impact of plant configurations and on-line maintenance on actual operational risk level - Safety Monitor^{TM} 2.0.

Temelin Safety Monitor Key Functional Requirements

1. Must operate in a multi-user PC environment under Microsoft Windows with security access features enabling access of multiple users at the same time
2. Software must be usable by plant personnel without knowledge of PRA techniques
3. Must resolve the complete PRA model(s) within several minutes for each plant configuration/maintenance/testing activities to reflect current (or proposed) plant conditions
4. Must be designed to provide virtually identical results to the original PRA models
5. Re-quantification of cutset libraries is not used, thereby eliminating the risk of truncation errors in the results
6. Must support risk calculations also for other than Level 1 models (external events, shutdown, Level 2 and 3)
7. Must provide the following information:

   - Actual plant risk (displayed in a "gauge" display) as a function of given actual plant configuration and conditions
   - Recommended Allowed Configuration Time
   - Risk profile over the operating cycle
   - Cumulative risk over the cycle
   - Important equipment in current plant configuration
   - Optimal restoration advice for inoperable components to reduce risk
   - Hypothetical risk profile from scheduled maintenance activities

   Some of the Safety Monitor screens are shown in the Appendix A.

Temelin Safety Monitor Development

The Scientech Safety Monitor™ 2.0 has been modified to meet Temelin specific needs:

- Temelin specific Safety Monitor model development
- Data for Safety Monitor development using Temelin component naming conventions
- Development of Czech language displays
- Development of Czech language documentation
- Running software under Temelin LAN Environment
- Testing of completed SM (software and model) at Temelin to ensure proper operation

Status of Temelin Safety Monitor Development Activities

Safety Monitor Model Development

Temelin PSA Level 1 model conversion to a SM master fault tree logic for total core damage risk from all sequences was performed. System fault trees were expanded to consider all possible operating alignments. Safety Monitor models were optimized for fast solution retaining full PSA model fidelity. Optimized SM model results have been carefully validated against the original plant PSA model results.

Safety Monitor Data Development

Plant specific data were developed for Temelin SM including over 25 main database tables, e.g. master system/train/component lists, component to PRA logic mapping list, mutually exclusive events, system alignment list, equipment tag out boundaries, maintenance activities, in addition testing and some other external conditions (e.g. severe weather) that could alter the likelihood of an initiating event, using Temelin specific component naming conventions. Currently, only PSA and some other equipment is included in Temelin Safety Monitor databases.

Development of Czech Language Displays and Documentation

All Safety Monitor 2.0 resource files were translated into Czech language enabling full understanding of Safety Monitor screens and functions by the plant personnel. These files are currently under compilation process. Czech version of Safety Monitor user and administrator documentation will be developed from English version following software final V&V.

Testing of completed SM

Currently, as the software is still under development, it is running for testing at Temelin PSA Dept. LAN computers.

Intended Use of Safety Monitor at Temelin

- Provide an easy-to-use tool for operator/maintainer plant staff to obtain insights from the PSA without detailed knowledge of PRA techniques and terminology

- Provide a PSA oriented tool for active influence on risk level of plant operation

- Serve as a means to optimize safety within Technical Specifications constraints

    - Identify requirements that are too restrictive given their risk significance
    - Identify Tech Specs required testing that may be adverse to plant safety

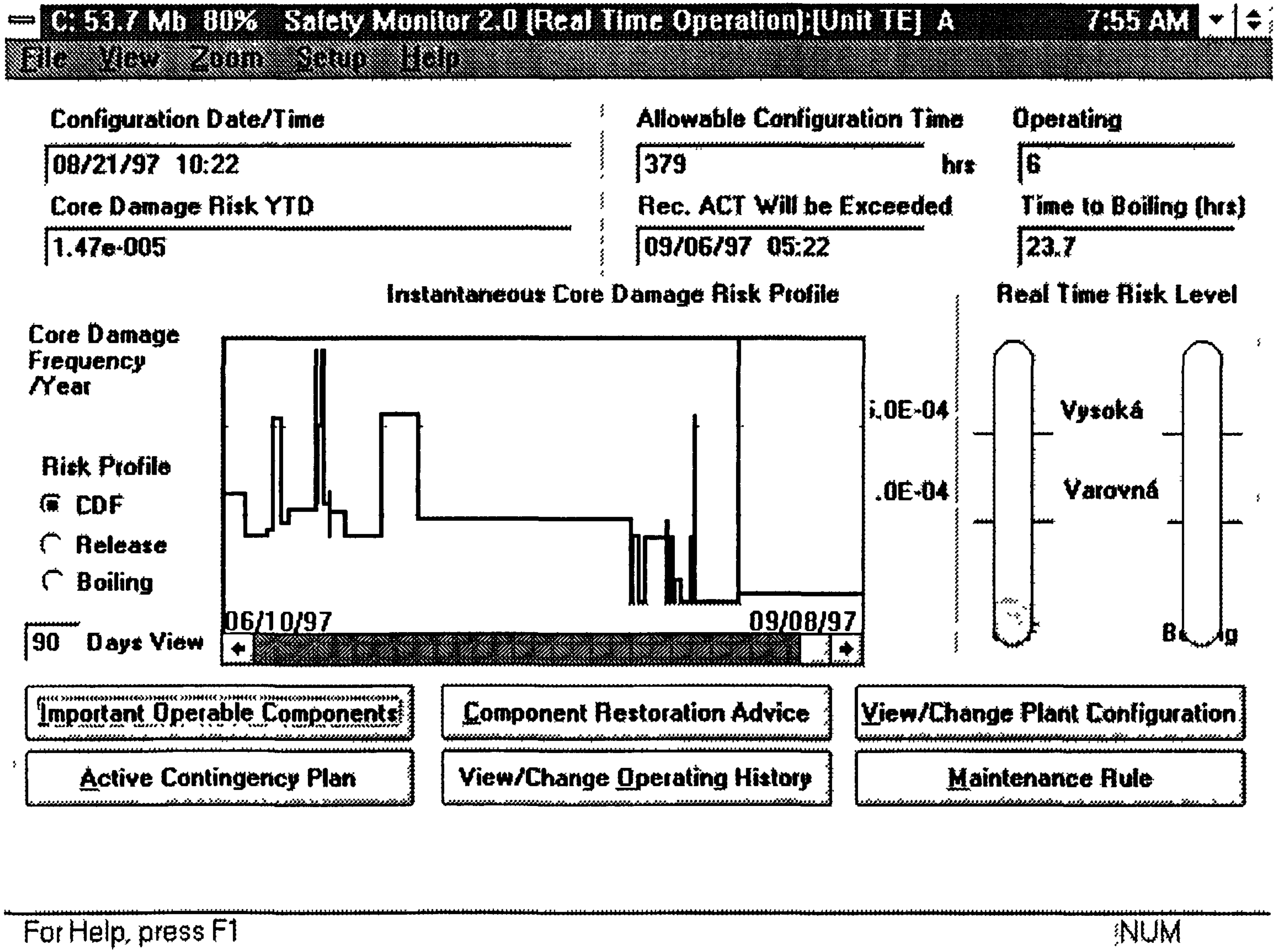
- Serve as a means to optimize planed maintenance activities through:

    - Import of maintenance schedule into the Safety Monitor
    - Risk profile calculation over the entire maintenance schedule
    - Schedule adjustment/editing from acceptable risk level point of view
    - Optimized schedule export back into the plant maintenance scheduler

- Provide history of plant configuration changes and component outages with associated risk levels

The following figures present the "Temelin Safety Monitor Screens".

SAFETY MONITOR MAIN SCREEN



C: 53.7 Mb 80% Safety Monitor 2.0 [Real Time Operation]:[Unit TE] A 7:55 AM
File View Zoom Setup Help

Configuration Date/Time: 08/21/97 10:22

Allowable Configuration Time: 379 hrs

Operating: 6

Core Damage Risk YTD: 1.47e-005

Rec. ACT Will be Exceeded: 09/06/97 05:22

Time to Boiling (hrs): 23.7

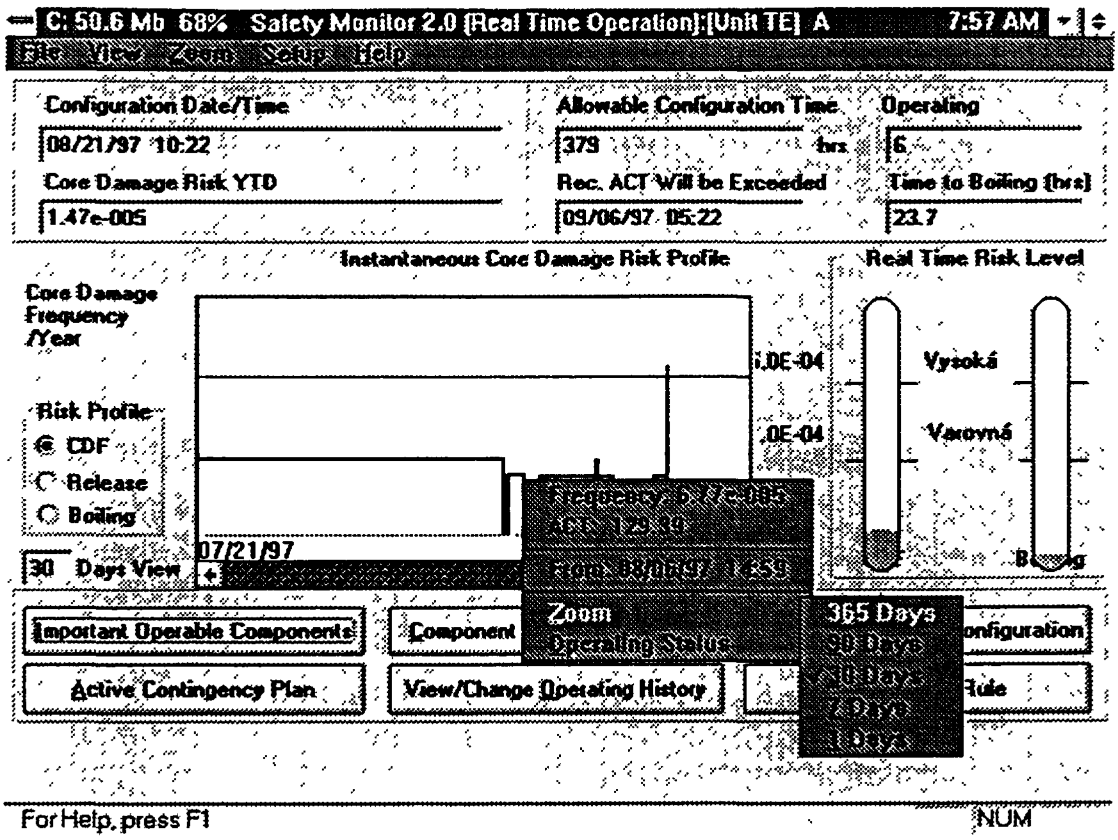
Instantaneous Core Damage Risk Profile
Core Damage Frequency /Year
.0E-04
.0E-04
06/10/97
09/08/97
Real Time Risk Level
Vysoká
Varovná

Risk Profile
● CDF
○ Release
○ Boiling

90 Days View

Important Operable Components
Component Restoration Advice
View/Change Plant Configuration
Active Contingency Plan
View/Change Operating History
Maintenance Rule
For Help, press F1
NUM

SAFETY MONITOR SCREEN - RISK PROFILE MENU



C: 50.6 Mb 68% Safety Monitor 2.0 [Real Time Operation]:[Unit TE] A 7:57 AM

Configuration Date/Time: 08/21/97 10:22

Allowable Configuration Time: 379 hrs

Operating: 6

Core Damage Risk YTD: 1.47e-005

Rec. ACT Will be Exceeded: 09/06/97 05:22

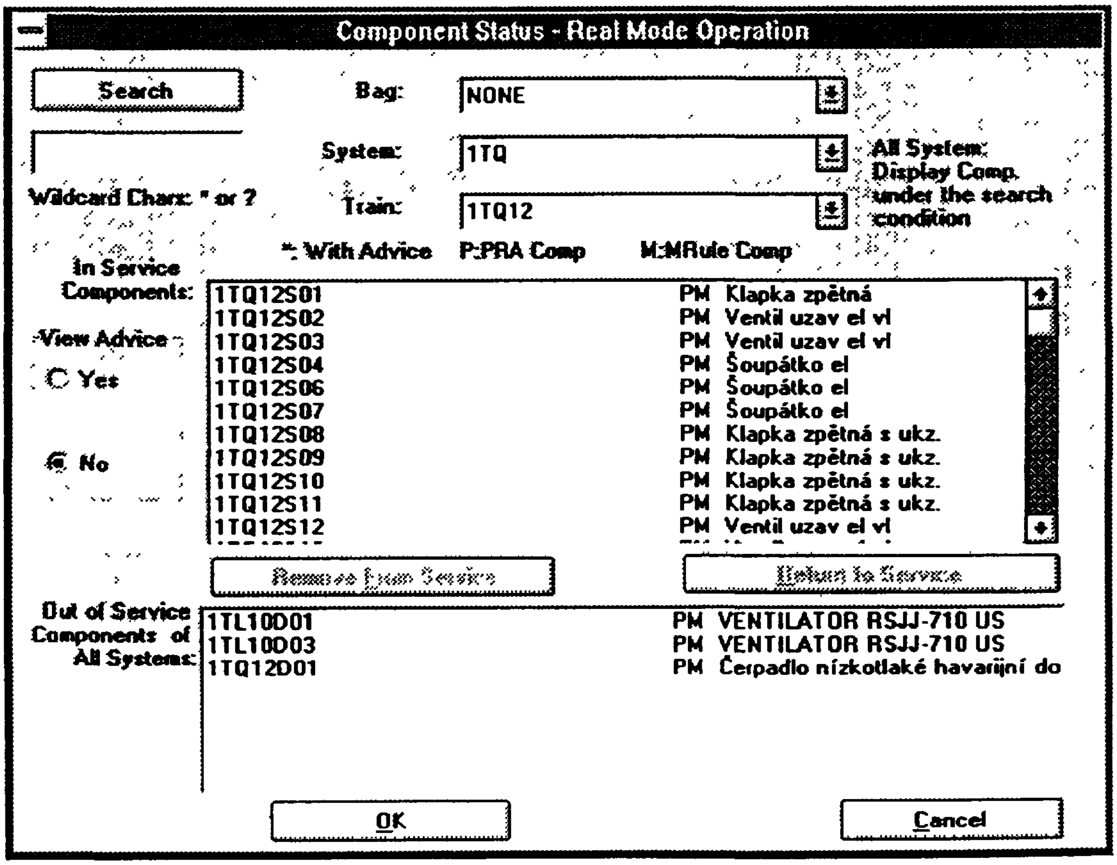
Time to Boiling [hrs]: 23.7

Risk Profile: ● CDF ○ Release ○ Boiling

30 Days View

For Help, press F1

OPERATING STATUS SCREEN



Component Status - Real Mode Operation

Bag: NONE

System: 1TQ

Train: 1TQ12

Wildcard Chars: * or ?

All System: Display Comp. under the search condition

*: With Advice P:PRA Comp M:MRule Comp

View Advice: ○ Yes ● No

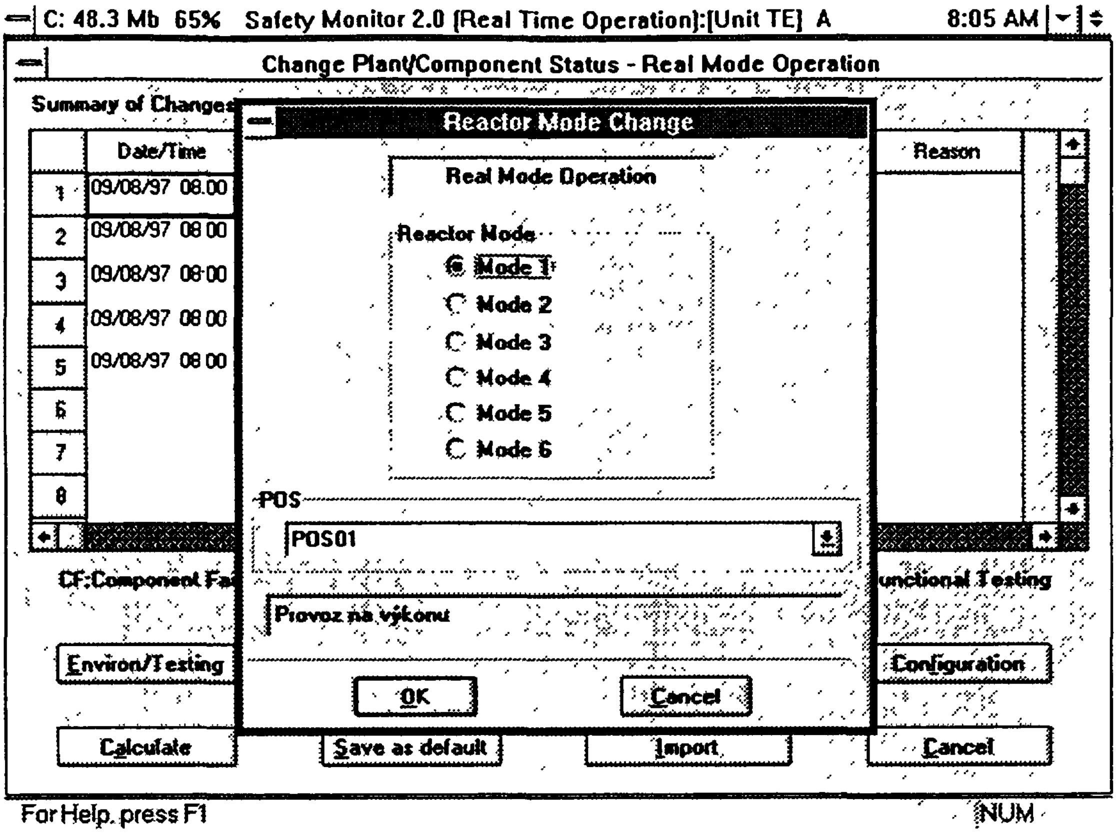
In Service Components:

```
1TQ12S01          PM  Klapka zpětná
1TQ12S02          PM  Ventil uzav el vl
1TQ12S03          PM  Ventil uzav el vl
1TQ12S04          PM  Šoupátko el
1TQ12S06          PM  Šoupátko el
1TQ12S07          PM  Šoupátko el
1TQ12S08          PM  Klapka zpětná s ukz.
1TQ12S09          PM  Klapka zpětná s ukz.
1TQ12S10          PM  Klapka zpětná s ukz.
1TQ12S11          PM  Klapka zpětná s ukz.
1TQ12S12          PM  Ventil uzav el vl
```

Out of Service Components of All Systems:

```
1TL10D01          PM  VENTILATOR RSJJ-710 US
1TL10D03          PM  VENTILATOR RSJJ-710 US
1TQ12D01          PM  Čerpadlo nízkotlaké havarijní do
```

REACTOR MODE CHANGE SCREEN



C: 48.3 Mb 65% Safety Monitor 2.0 [Real Time Operation]:[Unit TE] A 8:05 AM

Change Plant/Component Status - Real Mode Operation

Summary of Changes

	Date/Time
1	09/08/97 08.00
2	09/08/97 08 00
3	09/08/97 08·00
4	09/08/97 08 00
5	09/08/97 08 00
6	
7	
8	

Reason

CF:Component Fa… …unctional Testing

Reactor Mode Change

Real Mode Operation

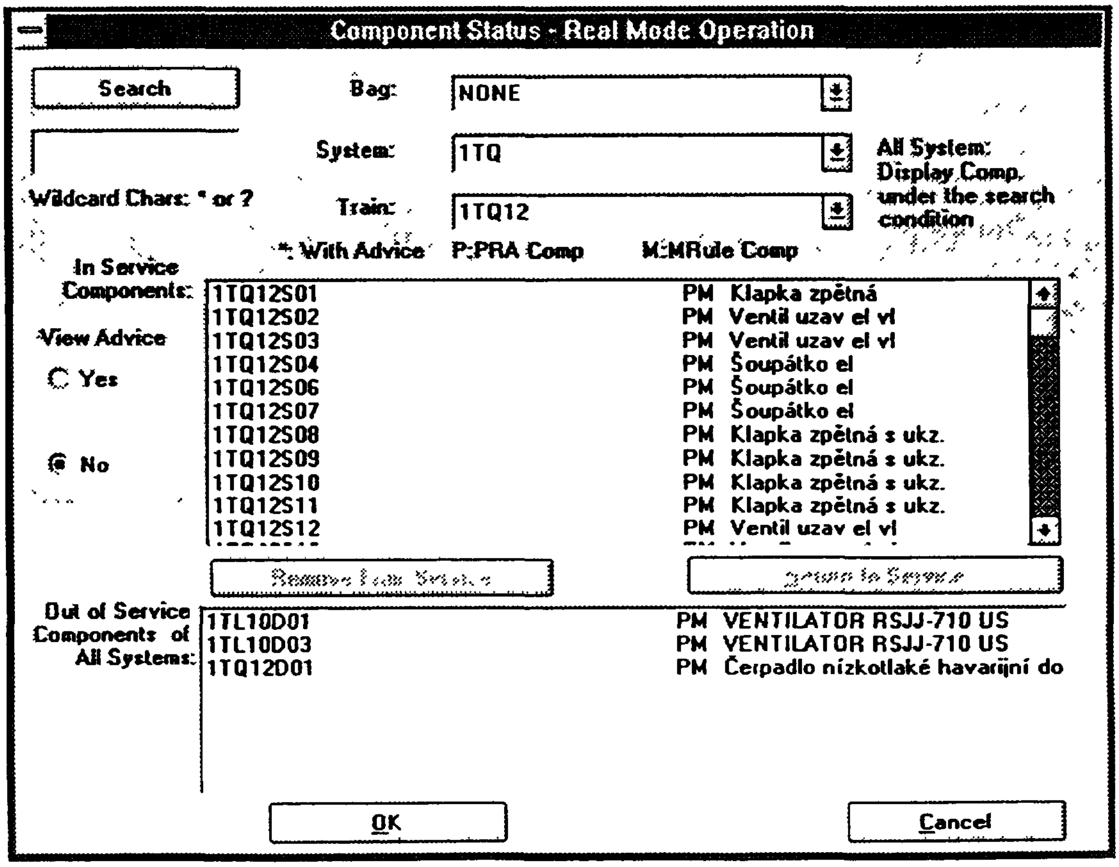
Reactor Mode

● Mode 1
○ Mode 2
○ Mode 3
○ Mode 4
○ Mode 5
○ Mode 6

POS: POS01

Provoz na výkonu

COMPONENT STATUS CHANGE SCREEN



Component Status - Real Mode Operation

Bag: NONE

System: 1TQ

Train: 1TQ12

All System: Display Comp. under the search condition

Wildcard Chars: * or ?

*: With Advice P:PRA Comp M:MRule Comp

In Service Components:

1TQ12S01	PM	Klapka zpětná
1TQ12S02	PM	Ventil uzav el vl
1TQ12S03	PM	Ventil uzav el vl
1TQ12S04	PM	Šoupátko el
1TQ12S06	PM	Šoupátko el
1TQ12S07	PM	Šoupátko el
1TQ12S08	PM	Klapka zpětná s ukz.
1TQ12S09	PM	Klapka zpětná s ukz.
1TQ12S10	PM	Klapka zpětná s ukz.
1TQ12S11	PM	Klapka zpětná s ukz.
1TQ12S12	PM	Ventil uzav el vl

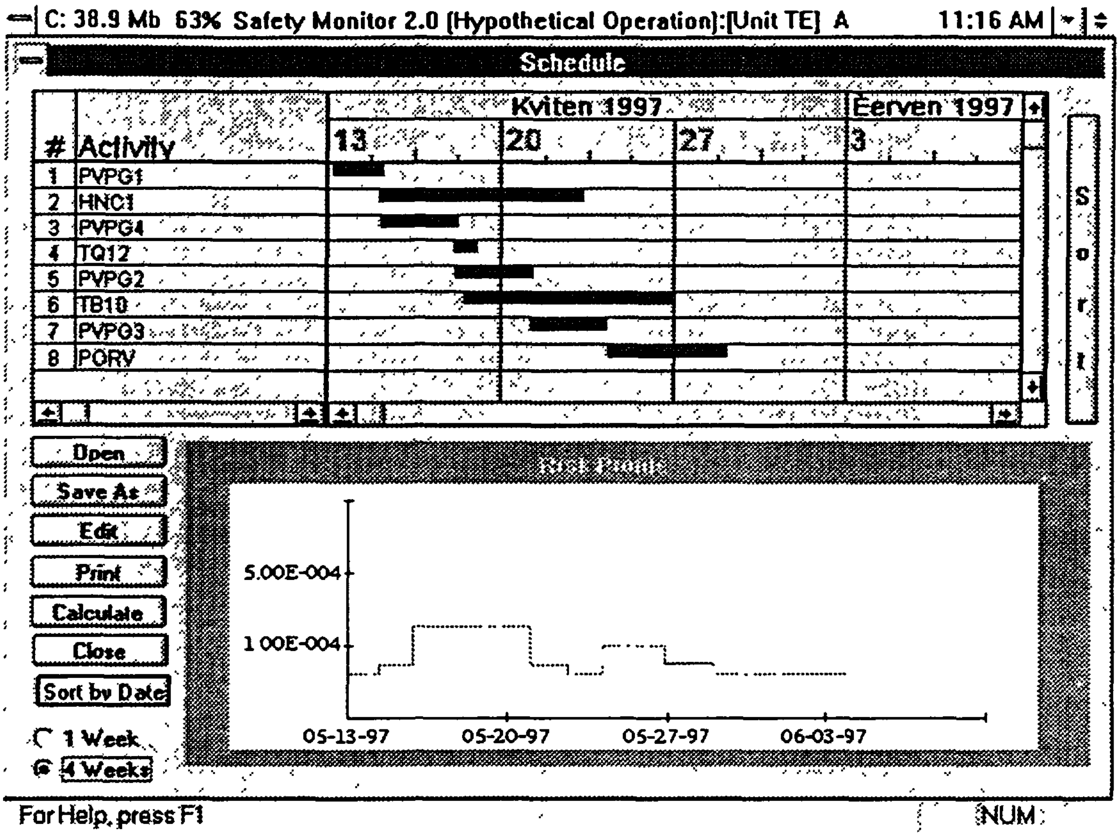
View Advice

○ Yes
● No

Out of Service Components of All Systems:

1TL10D01	PM	VENTILATOR RSJJ-710 US
1TL10D03	PM	VENTILATOR RSJJ-710 US
1TQ12D01	PM	Čerpadlo nízkotlaké havarijní do

SAFETY MONITOR SCHEDULER SCREEN

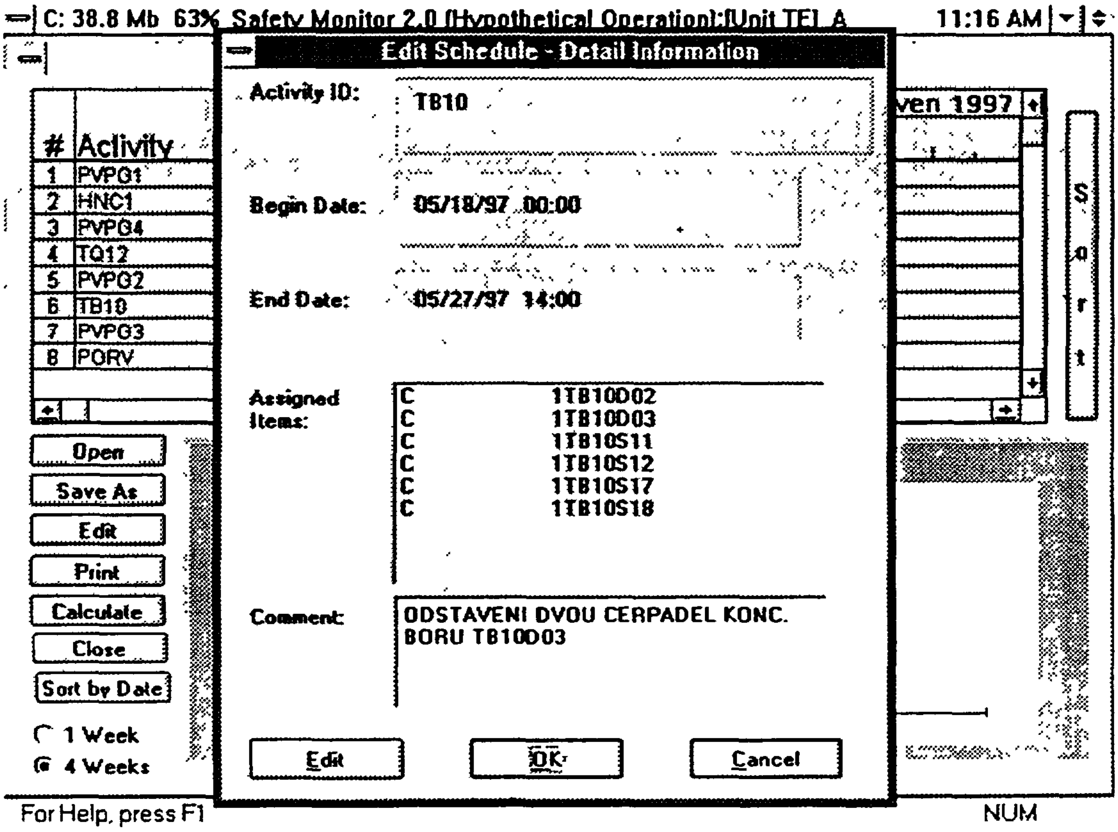


C: 38.9 Mb 63% Safety Monitor 2.0 (Hypothetical Operation):[Unit TE] A 11:16 AM

Schedule

#	Activity
1	PVPG1
2	HNC1
3	PVPG4
4	TQ12
5	PVPG2
6	TB10
7	PVPG3
8	PORV

SAFETY MONITOR SCHEDULER - EDIT ACTIVITY SCREEN



C: 38.8 Mb 63% Safety Monitor 2.0 (Hypothetical Operation):[Unit TE] A 11:16 AM

Edit Schedule - Detail Information

Activity ID: TB10

Begin Date: 05/18/97 00:00

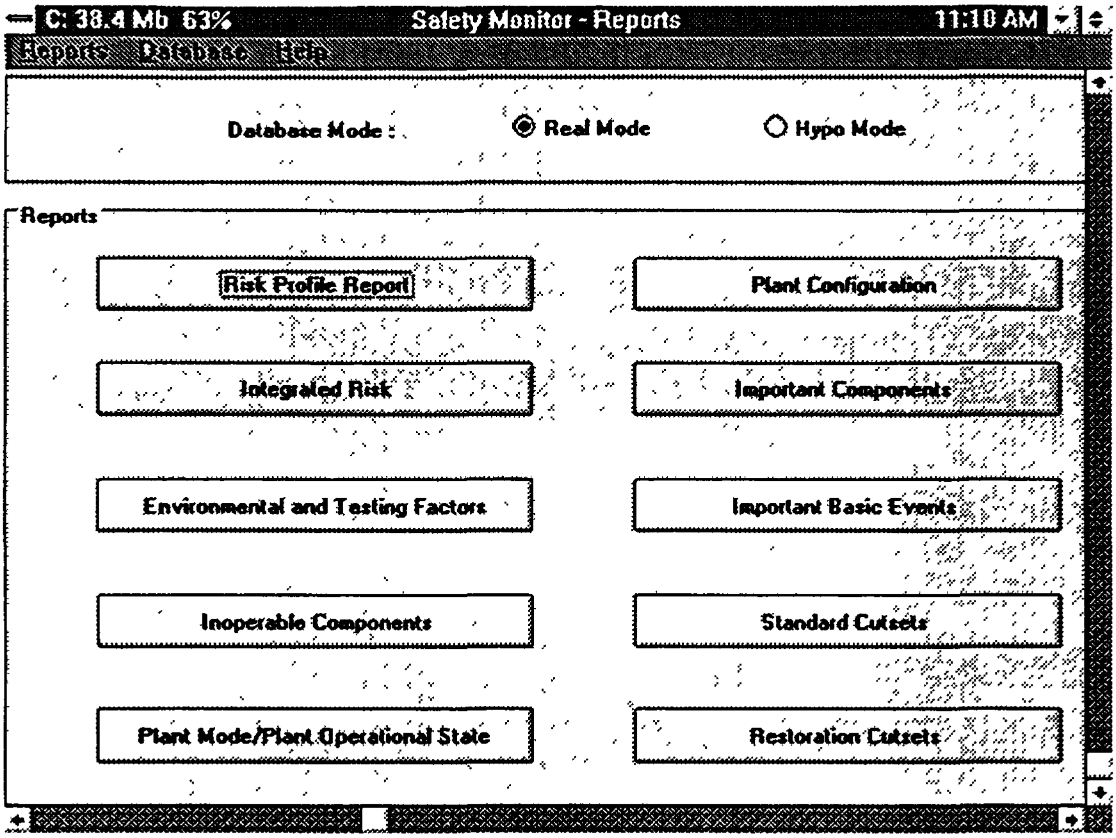
End Date: 05/27/97 14:00

Assigned Items:

C	1TB10D02
C	1TB10D03
C	1TB10S11
C	1TB10S12
C	1TB10S17
C	1TB10S18

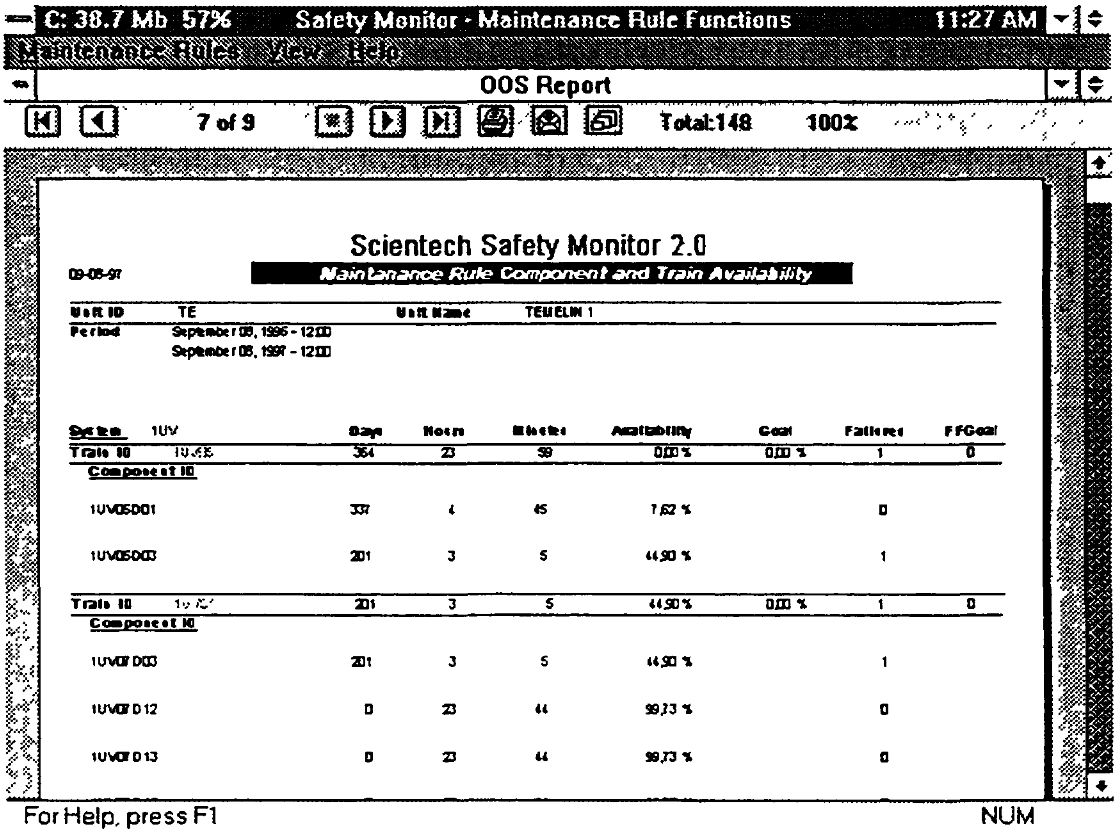
Comment: ODSTAVENI DVOU CERPADEL KONC. BORU TB10D03

SAFETY MONITOR REPORT SCREEN



Database Mode : ● Real Mode ○ Hypo Mode

SAFETY MONITOR MAINTENANCE RULE REPORT



Scientech Safety Monitor 2.0

09-08-97

Maintanance Rule Component and Train Availability

Unit ID TE Unit Name TEUELIN 1

Period September 08, 1996 - 12:00
September 08, 1997 - 12:00

System 1UV	Days	Hours	Minutes	Availability	Goal	Failures	FFGoal
Train ID 1UV05	364	23	59	0,00 %	0,00 %	1	0
Component ID							
1UV05D001	337	4	45	7,62 %		0	
1UV05D003	201	3	5	44,90 %		1	
Train ID 1UV07	201	3	5	44,90 %	0,00 %	1	0
Component ID							
1UV07D003	201	3	5	44,90 %		1	
1UV07D012	0	23	44	99,73 %		0	
1UV07D013	0	23	44	99,73 %		0	

IMPROVING THE RELIABILITY OF EMERGENCY DIESEL GENERATORS THROUGH SUSTAINED MAINTENANCE

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India

Abstract

In Nuclear Power Stations Emergency Diesel Generators are vital safety related equipment which ensures power supply to essential equipment during loss of power. In View of their importance Reliability of Diesel generators should be very high. Since these Diesel generators are standby equipment and operate only during demand or during surveillance checks, their demand failure probability should be very low and once they operate their operational availability should be very high.

Madras Atomic Power Station at Kalpakkam, India consists two pressurised heavy water Reactors each rated at 220 MWe. To supply standby power each unit has two Diesel Generators of I 500 kW capacity each. The Diesel Engine is 16 cylinder 'V' type engine and is cranked by Air starting motor and is connected to generator whose rating at 100% load is 1500 KW.

During commissioning and in the initial years of operation these Diesel Generators have encountered many problems. Major problem was Diesel Engine failing to start on demand. This was due to non engagement of air motor pinion with ring gear or continued engagement air motor even after the engine had picked up speed and failure of timer to initiate multiple starts after initial incomplete starts Apart from these there were problems like fuel oil leaks, high jacket water temperatures, low fuel oil pressure trips. Another major problem was with excitation system. How these problems were dealt with thereby reducing the demand failure probability and increasing the operational availability are discussed in this paper.

1. PROBLEMS AND SOLUTIONS

a) The engine is cranked with Ingersoll-Rand solenoid operated air starting motor. Brief description of this system is given below.

When a starting impulse is given, a solenoid valve in the air circuit energises and admits air into the air motor pinion chamber through a 3/8" dia tube which advances the pinion and once the pinion is engaged into the ring gear attached to the Diesel Generator the air supply from SV comes out of another port which operates a relay valve through which air at a pressure of about 11 Kg/cm^2 gets admitted into the motor through a 2" dia pipe and drives the air motor and hence the ring gear rotates thus cranking the Engine.

Initially, there used to be problem with engagement of pinion into the ring gear and engagement used to be incomplete

To avoid this, the profile of pinion teeth and ring gear teeth were tapered at entry point in such a way that probability of engagement is increased and incomplete engagement problem was sorted out

b) Air motor consists Hylam vanes These are 5 ply, 6 2 mm thick, phenolic resin bonded vanes, and these are driven by compressed air at a pressure of 11 Kg/cm^2. These hylam vanes swell due to moisture present in the compressed air and while in operation rub with the casing resulting in small pieces to break away from the vanes and air motor used to get seized due to the presence of this debris

Auto drain valves were introdu‹ ε l in the air receivers which drains moisture from air receivers periodically Quality of hylam vanes was also improved which had better bonding of laminations.

c) The compressed air line from Air receivers to Air motor is a carbon steel line. Due to presence of moisture corrosion used to take place and the corrosion particles apart from clogging Air motor used to get lodged in the solenoid valve which supplies and bleeds air supply to Air motor This was resulting in maloperation of solenoid valve and Air motor continued to get engaged with ring gear even after DG picks up speed, thus causing damage to Air motor

The CS lines were replaced with GI lines to prevent rust formation.

d) Lubricating oil, Diesel oil and jacket water lines were originally provided with victutalic couplings and dresser couplings and union joints These joints had become source of potential leaks and causing operational failures due to oil leaks apart from being fire hazard.

These joints were eliminated by changing them into weld joints and where frequent dismantling is required joints were converted into flange joints

e) Originally there was only one pre lub oil pump for each DG set to provide lubricating oil to the bearings when DG is stationary. This pump has to operate 30 Minutes ON 50 minutes OFF cycle This was to ensure oil film in the bearings. Once Engine starts, shaft driven lub oil pump supplies the lub oil However it was noticed that bearings were getting damaged due to lack of lubrication

This problem was analysed and found that since lub oil filters were installed at a higher elevation oil was getting drained from the filters during pre lub oil pump OFF cycle and when engine starts. during this time, there was a time gap for oil to flow into the bearings from shaft driven pump as the pump has to fill the filters before oil flow is restored to the bearings To eliminate this problem, an additional pre-lub oil pump was installed and at any given time one pre-lub oil pump will be in service

f) Shaft driven lub oil pump is a gear pump In one of the pump, the Idler gear had failed and when Engine has started the failed idler gear damaged the pump casing causing all the lub oil to spill out

This failure of idler gear was due to lack of support of the gear. To prevent this sort failures idler gear was made integral with the shaft and supported on both ends by Journal bearings.

g) DG used to trip during operation on Jacket Water temperature high This was due to inadequate flow of Jacket water

To increase the flow, few elements in the Thermister valve were removed and tripping of DG on High Jacket Water temperature was eliminated

h) Diesel Generators in Unit-I have brushless Excitation system and we had hardly any problem with this system However in Unit-II excitation is by brushes. Initially excitation is done using batteries and once the machine is synchronised and connected to the Bus, excitation is done by the machine it self through rectifiers. This system had many problems Major problems were

i) low current sensing relay used to trip the machine even before Engine could pickup speed

This was attended by making the low current sensing relay effective only after closing the breaker

ii) Initially the batteries were of lead acid These were replaced with Nickel cadmium batteries which has more number of deep charging/discharging cycles

By making these changes, the number of failures due to excitation have been reduced from 12/year (average) to around 1 to 2/year

2. <u>SPECIAL TOOLS AND TACKLES:</u>

To reduce maintenance time many special tools and tackles were developed, thus improving the availability of Diesel Generator Some of the tools developed are

- main bearing removal tool
- piston/connecting rod lifting/lowering tool
- liner removing and inserting tool
- piston ring inserting tool
- tool for checking parallelism between connecting rod bores
- To prevent potential failure of rocker arm mechanism during failure of yoke, a clamp was installed which will retain the yoke in position in case of its failure during operation thus preventing damage to rocker arm Failed yoke can be replaced during the next available opportunity

The above are some of the modifications which have been carried out by analysing the failures, finding root cause for each failure and action taken to prevent recurrence of such failures

3. TRAINING:

Human factor is one of the important thing in improving the reliability of any equipment Hence developing the skills of personnel involved in maintenance of Diesel generators was given importance Five yearly overhauling of equipment was given on contract to the original manufacturer and during this time departmental personnel were associated thus becoming conversant with the job Few persons were also deputed to manufacturer's works for additional training All the persons involved with the maintenance of this equipment were made to follow the procedures so that a job once done does not get repeated and the maintenance time is kept to barest minimum

4 MAINTENANCE STRATEGY:

Since Diesel Generators are standby equipment and are expected to operate only on a demand, majority of the time they will be in operation only during surveillance check and hence running hours are very small So condition based maintenance is not of much relevance There wont be sufficient data to collect, analyse and to trend the deterioration and then to intervene to do maintenance to prevent failure Hence at MAPS the strategy adopted is time based maintenance Time interval for each component is fixed based on manufacturer's recommendations and past operating history, so that a component gets required attention before it can be fail

To keep the failure rates to minimum an exhaustive study was made based on manufacturer's recommendations and failure rates and checklists have been prepared for daily, quarterly, yearly preventive maintenance checks These check lists are enclosed herewith

5. ADHERENCE TO PROCEDURES:

A Maintenance Manual has been written giving exhaustive details of various maintenance activities This manual apart from detailed procedures for disassembly and assembly contains various clearances and values to be maintained, tolerances that can be permitted and the limiting values after which a component is to be replaced/reconditioned are indicated Check points during assembly and dis-assembly are also indicated These are the stages at which a supervisor has to inspect and clear only after which further work has to be taken up Maintenance Manual also contains spares, tools and tackles that are needed

Daily, quarterly, yearly preventive maintenance checks are carried out using the maintenance manual and any deficiencies noticed during these checks are attended immediately thus preventing major failures These preventive maintenance activities have brought down the failure rates Another feature is that Mechanical, Electrical and Instrumentation sections carry out their PM checks simultaneously thus reducing the planned downtime of the Diesel Generator sets

6. SURVEILLANCE FREQUENCY:

At MAPS each Diesel generator was being started for surveillance purpose on alternate days thus logging fifteen starts a month, though the Technical specification requirement was only once a week It was noticed that having more number of starts, the equipment failure rates were higher due to wear and tear

As per IEEE-387, Diesel Generator sets are generally expected to make 4000 starts and operate 4,000 hrs during their life time USNRC guidelines suggest the following guidelines for test frequency

No. of failures on Demand	**Test interval**
0 to 1	31 days
2	14 days
3	7 days
4 and more	3 days

Based on the above guidelines and to reduce demand failure probability the test frequency has been increased to once in 7 days for the last six months This increase in frequency has definitely reduced the demand failure probability

7. CONCLUSION:

The demand failure probability could be reduced and high operational availability could be achieved at MAPS by adopting the following methods

1. Preparation of Maintenance Manual, Procedures for each maintenance activity and adhering to them

2 Training of personnel to do a quality maintenance in quick time

3 Strictly following time based maintenance schedules

4. Finding the root cause of failure whenever it occurs and taking appropriate remedial action to prevent recurrence

8. PERFORMANCE STATISTICS:

The performance of Diesel Generators at MAPS for the year 96-97 is indicated below.

DEMAND FAILURE PROBABILITY:

	Unit I		Unit II	
	DG # 1	DG # 2		DG # 2
Total No of demands	274	257	225	204
No of Failures to start	NIL	4	1	4
Demand failure probability	O/D	1 5 x 10^{-2}/D	1 96x10^{-2}/D	
Standard value as per WASH-1400	3 x 10^{-2}/D			

OPERATIONAL FAILURE RATE

	Unit I		Unit II	
	DG # 1	DG # 2	DG # 1	DG # 2
Running Hours	455 5	235 75	228 25	201 5
No of Failures to run	6	5	2	4
Operational Failure rate	1 31 x 10^{-2}	2 12 x 10^{-2}	8 76 x 10^{-3}	1 96 x 10^{-2}

WANO PERFORMANCE INDICATOR

	Unit I		Unit II	
	DG # 1	DG # 2	DG # 1	DG # 2
Estimated unavailable hours	NIL	72	24	112
Known unavailable hours	145 91	115 63	16.5	24.25
Total unavailable hours	145 91	187 63	40.5	136.25
Performance Indicator	$\frac{145\ 91 + 187\ 63}{2 \times 24 \times 365}$ = 0 019		$\frac{40.5 + 136.25}{2 \times 24 \times 365}$ = 0 01	

After increasing the test interval to once a week, the demand failure probability and operational failure rates have come down as indicated below, during the last six months period

	Unit I		Unit II	
	DG = 1	DG = 2	DG = 1	DG ≠ 2
Demand failure probability	1 x 10^{-4}	0	0	1 3 x 10^{-2}
Operational Failure rate	9 3 x 10^{-3}	0	0	1.4 x 10^{-2}

By human intervention and increasing the maintenance alone cannot increase the reliability of an equipment than it is permitted by the design. We continue to have occasional problems with initial cranking of engine and excitation. Better alternate designs could have improved the performance further

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STUDY ON RISK-BASED OPERATION AND MAINTENANCE USING THE LIVING PSA SYSTEM

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Abstract

The objective of this study is to contribute to an improvement of fast reactor plant operation and maintenance from the standpoint of risk assessment. An effort was made to analyze a relationship between the valve failure probability and the standby time based on the component reliability database and statistical analysis system (CORDS). According to the analysis result, the following issues were examined: the surveillance test interval (STI), timing and the allowable outage time (AOT) of redundant valve system in a fast reactor model plant. An examination was performed based on the failure probability non-linearly dependent on the standby time using the risk importance measures and technique to optimize the AOT which are incorporated in the living PSA system (LIPSAS). The case study showed that consideration of non-linear time trend of the failure probability made the recommended STI and AOT longer under the same risk limitation. It is recommended to apply the non-linear expression of demand failure probability in estimating the STI and AOT based on the risk measures.

1. INTRODUCTION

PNC has been developing the "living" probabilistic safety assessment system (LIPSAS)[1] and the component reliability database and statistical analysis system (CORDS)[2] in order to improve the overall safety of a liquid-metal-cooled fast breeder reactor (LMFBR) plant system. The target of the LIPSAS is currently aimed at the Japanese prototype fast breeder reactor.

This paper describes preliminary an application of the LIPSAS and CORDS for management of LMFBR operation and maintenance (O&M). Discussion is focused on (1) optimization of surveillance testing interval (STI) of safety-related standby components and (2) limiting condition for O&M when a failure of the safety-related standby component is found during reactor operation. These issues were examined by using various analytical methods which are incorporated in the LIPSAS in cooperation with the component reliability data obtained from the CORDS.

2. STATISTICAL ANALYSIS OF COMPONENT OPERATING EXPERIENCE AND FAILURE DATA

We performed statistical analysis of the operating record and failure data on the component using the CORDS. The CORDS contains engineering-, operating- and failure-records of typical components used in the US and Japanese LMFBR and sodium test loops.

2.1 SELECTION OF COMPONENTS

Among the components compiled in the CORDS, the valve component was chosen in order to analyze the relationship between probability of demand type of failure and standby time period. Typical valves in the LMFBR plant system are valves exposed to sodium fluid or to inert gas of forming sodium free surface. Fig. 1 shows relative population related to time interval between open demands (or close demands) for each valve type. Four types of the valve are available for the analysis. They are the motor-operated valve (MOV) and air-operated valve (AOV) in sodium system, and the AOV in gas system and radioactive gas system. The population is counted in proportion to cumulative time between the demands. Total of relative population is normalized to unity. The following analysis result can be applied within the effective range where there is sufficient population in Fig. 1.

2.2 ASSUMPTION OF MATHEMATICAL FUNCTION

In order to examine dependency of demand failure probability on the standby time, it was assumed that the demand failure probability could be expressed as follows:

$$Q(t) = 1 - \exp(- H(t)), \text{ and } H(t) = a * (t + c)^b \qquad (1)$$

where "Q(t)" means the demand failure probability at "t",

"H(t)" is the cumulative hazard at "t",

"t" is the standby time which is reset to zero just after every same type of demands, and

"a", "b" and "c" are constant parameters.

This mathematical function has following characteristics.

(1) Where "t" is extremely greater than "c", "H(t)" is almost in proportion to "t^b".

(2) Where "t" is extremely less than "c", "Q(t)" becomes approximately independent from "t".

(3) When "b"=1, "H(t)" is expressed with summation of both the term in proportion to "t" and the constant. In such case, "a" becomes a constant failure rate, and the product of "a" and "c" is a constant demand failure probability. They are conventional model parameters.

2.3 QUANTIFICATION METHOD OF FAILURE PROBABILITY FUNCTION

Using the maximum likelihood estimation (MLE) method, the failure probability function, "Q(t)", is quantified. In this case, likelihood function, "L(a,b,c)", is defined as follows:

$$L(a,b,c) = \prod_{i=1}^{imax} \{1-\exp(-a(t_i+c)^b)\}^{m_i} \exp(-a(n_i-m_i)(t_i+c)^b), \qquad (2)$$

where "i" indicates the number of the valve,

"imax", "m_i" and "n_i" are the total number of the valve, of failure to open or close on the "i"-th valve, and of open demands (or close demands) on the "i"-th valve respectively, and

"t_i" is the mean time between open demands (or close demands) on the "i"-th valve.

The constant parameters are quantified so that the likelihood takes the maximum value in the MLE method.

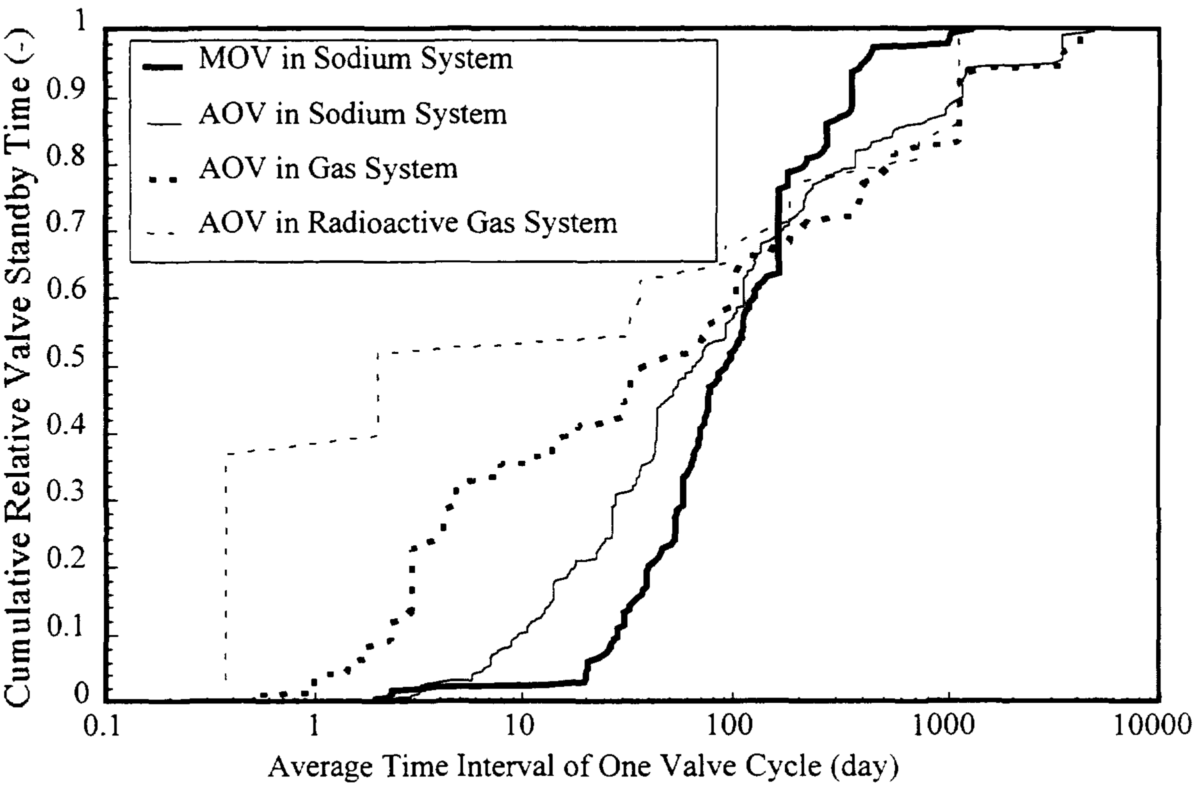

FIG. 1. Distribution of Statistical Population for Failure Probability Calculation

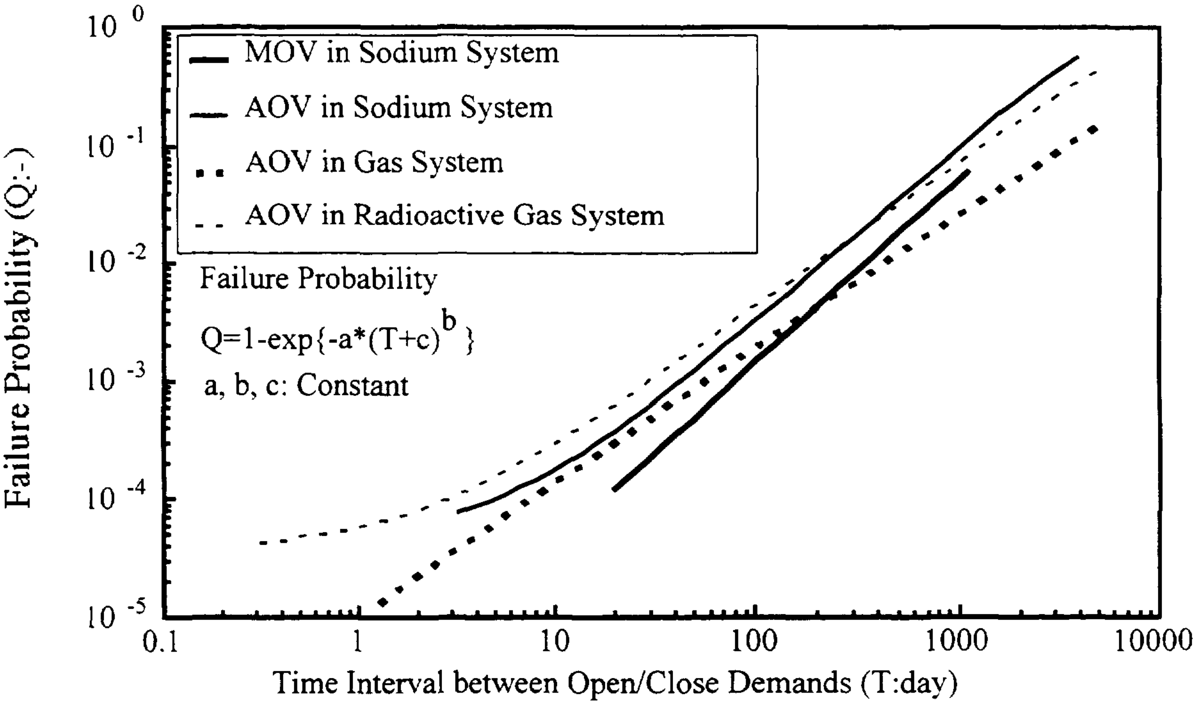

FIG. 2. Comparison of Failure Probability Curve Derived from the CORDS Data According to the Maximum Likelihood Estimation

2.4 PARAMETER QUANTIFICATION RESULT

As a quantification result according to the MLE, we obtained the value of parameters "a", "b" and "c" for each valve type. Fig. 2 shows failure probability curve, "Q(t)", for four types of the valve in the effective time range with sufficient statistical population. Both axis's are plotted in logarithmic scale. The quantitative result indicates that failure probability in all types of valve simply increases with time. Parameter "b" for each valve stays within the range between 1.0 and 2.0. There is indication of the positive asymptotic value in the range of less than about 10 days of standby time in the failure probability curves of the AOVs in sodium and gas system. This trend suggests that even if the standby time becomes close to zero, the probability of failure to open/close does not reduce to zero.

2.5 APPLICATION TO RELIABILITY ESTIMATION

Based on the probability considering the time trend, unavailability of a single valve was evaluated. Safety-related valves are periodically actuated in the surveillance test in order to detect failure prior to encounter with an accident which requires actuation of the valves. Failure probability at the surveillance test is obtained from the equation (1) by substituting the surveillance test interval (STI) to "t".

In safety analysis, we need the failure probability at the accident when valve actuation is required. Since the accident happens at random, the mean probability is given with the time-average over the STI. Fig. 3 depicts the mean probability curve for a single valve derived from "Q(t)" drawn in Fig. 2. Generally there is uncertainty in the prediction curve of the mean probability induced from averaging over both the effective and non-effective time range. The prediction curve in Fig. 3 had better be utilized only in case of the STI greater than about 30 days.

3. RISK-BASED EXAMINATION OF STI OF SAFETY-RELATED VALVES

3.1 EFFECT OF STI ON THE UNAVAILABILITY OF A SINGLE VALVE

Surveillance test of a single valve usually spends a couple of minutes. If the test needs 5 minutes and the STI is 30 days, unavailability due to test outage of the valve without test-override function becomes approximately 10^{-4}. The shorter the STI is, unavailability due to test outage increases. According to Fig. 3, it is possible to reduce the valve unavailability to 10^{-4} order of magnitude if test duration time should keep at most a few minutes. Otherwise the test-override function should be attached to the valve.

If a plant operator manually generates the valve actuation signal, we must consider human error probability in examining the valve unavailability. According to the references (3) and (4), it is difficult to reduce the human error probability to 10^{-4} even if action type is skill-base or rule-base action and if there is longer than one hour remained for diagnosis and operator action. Automatic actuation circuit is necessary in order to reduce the unavailability of the safety-related valve to 10^{-4} order of magnitude.

According to Fig.2, it is possible to attain to 10^{-5} of unavailability on the AOVs in gas system with both test-override function and automatic actuation signal if the STI is one day. So, it is assumed that both automatic actuation and test-override function are already incorporated into the valve design in the following examination of valve unavailability.

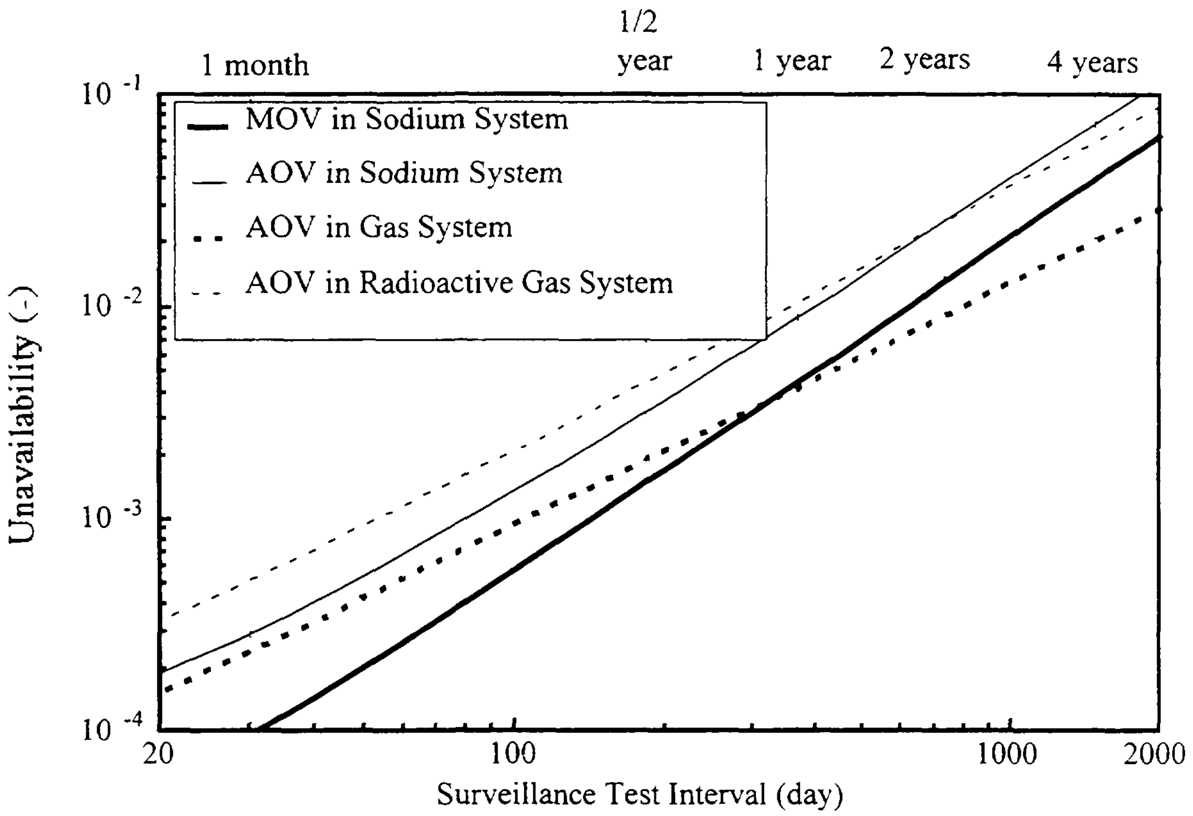

FIG. 3. Average Unavailability of a Single Valve on Random Demand

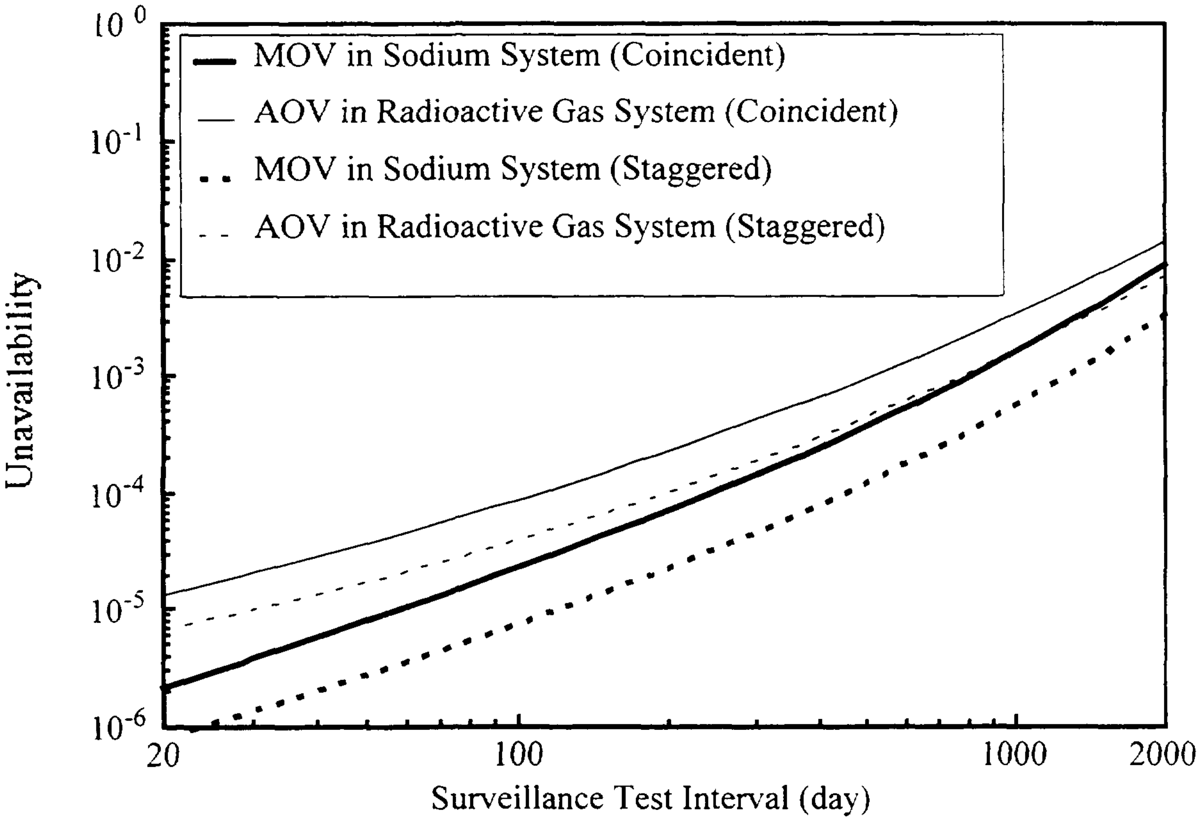

FIG 4 Average Unavailability of Double Redundant Valves on Random Demand (Considering CCF)

3.2 UNAVAILABILITY OF REDUNDANT VALVES

It is important to consider contribution of the common cause failure (CCF) in estimating the occurrence probability of failure to actuate at least one valve among two redundant valves. In this study we estimated the CCF probability using the beta-factor method. The value of beta-factor of the valve was quantified as 0.04 on the basis of the valve operating experience in the US light water reactor plant [(5)-(7)]. It is also important to consider timing of surveillance testing between redundant valves such as simultaneously or staggeredly. We computed the unavailability of a system consisting of two redundant valves as a function of the STI. The valves are periodically tested in a simultaneous or staggered way. Fig. 4 shows the calculation result of the MOV in sodium system and the AOV in radioactive gas system. If a limiting value for system unavailability is given, the STI to be recommended can be determined using Fig. 4 so that the system unavailability becomes less than the limiting value.

3.3 CASE STUDY ON RISK-BASED STI IN THE LMFBR MODEL PLANT

From the standpoint of safety, it is important to examine the degree of impact of the STI of safety-related valve whose failure affects the occurrence frequency of core damage. We studied the decision method for the surveillance test patterns of the valves to reduce the impact of each valve using the probabilistic safety assessment of an LMFBR model plant. The model plant is a loop-type and uses the liquid sodium as a reactor coolant. The accident sequence to be examined is focused on one sequence with relatively high occurrence frequency which result in the core damage and include failure of the valve component.

3.3.1 Plant description

In the plant to be examined, when the primary main cooling system leakage happens, it is necessary to stop pressurizing the reactor cover gas in order to make-up reactor sodium level. At least one of the two containment vessel isolation valves (see Fig. 5) in the cover gas supply system must be shut by the automatic signal. The valve lineup is shown in Fig. 5. There are two AOVs connected in series. In addition, there is another AOV connected in the same pipe line whose design is different from that of the isolation valves. This valve does not belong to the plant safety system. However, it is assumed that the operator remotely would actuate this valve as an accident management if the plant entered the accident sequence. Thus the core damage accident would happen if all of the three valves failed to close under the primary main cooling system leakage accident.

3.3.2 Examination of the STI based on the risk importance measures

The two kinds of risk importance measures with no dimension are introduced in examining the STI of these valves. One is the index which indicates how many times the total occurrence frequency of core damage increases when the specific component is unavailable. It is called "risk achievement worth (RAW)". The other is the ratio of the occurrence frequency of the specific core damage sequences which includes failure of the specific component to that of the total core damage sequences. It is called "Fussell-Vesely (FV) importance".

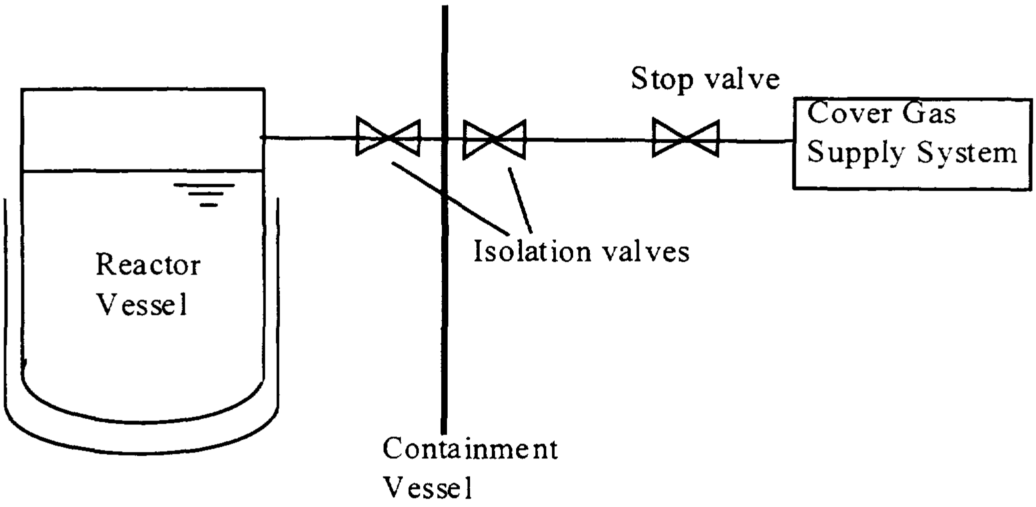

FIG. 5. Reactor Cover Gas Isolation System

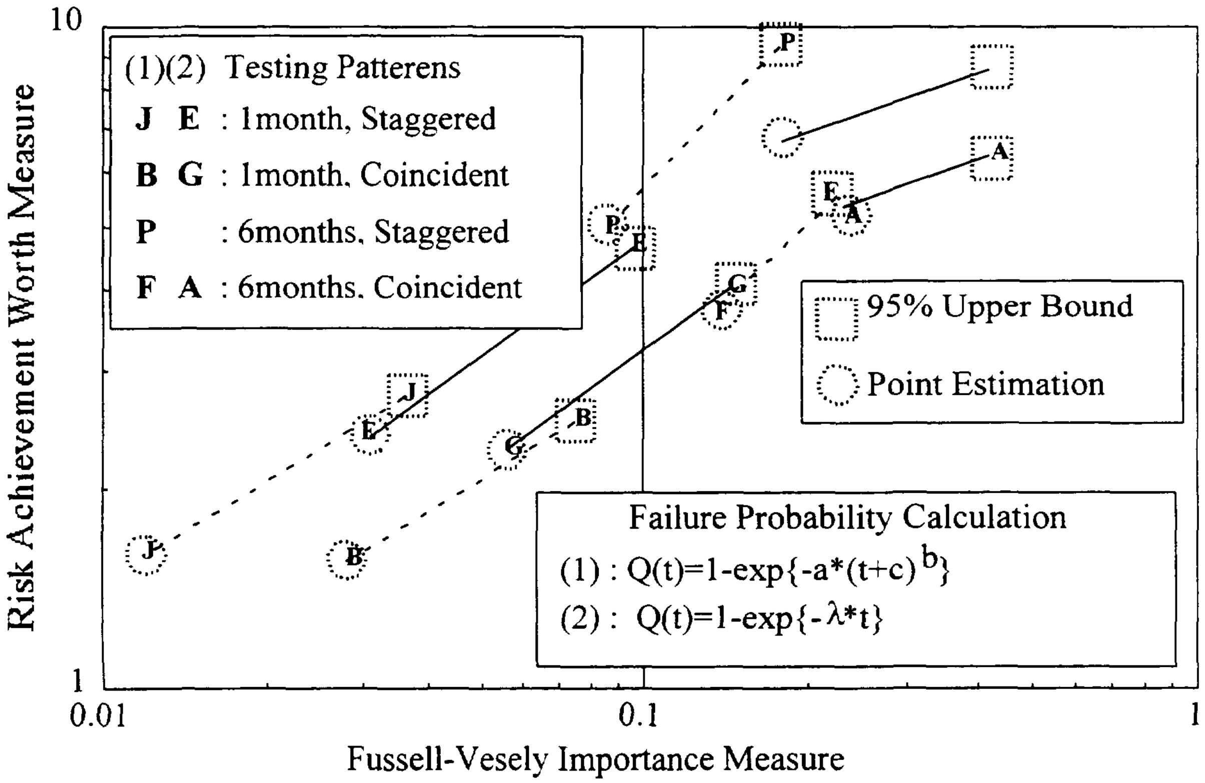

FIG. 6. Risk Importance Map of the Cover Gas Isolation AOV with Various Surveillance Testing Patterns

It can be proposed to restrict both the risk importance measures less than specific values from the standpoint of diversification of risks. We calculated the risk importance measures of the AOV with various surveillance testing patterns based on the unavailability quantified in the previous section. Not only point estimation but also 95% upper bound were computed. In estimation of the upper bound, we took it into account only the uncertainty of failure probability of the cover gas isolation AOV as an error factor of 5. These risk measures are compared with those calculated with the conventional method using the constant failure rate model in Fig. 6.

Difference between calculation methods is comparable to difference among surveillance testing patterns. However, these differences are also comparable to the difference between point estimation and the 95% upper bound. It is recommended to apply the failure probability derived in the previous section in the risk measure calculation considering the uncertainty. Assuming to keep the 95% upper bound of the FV importance less than 0.1 for each valve, we must select less than or equal to one month of STI for the two cover gas isolation AOVs. Furthermore if the 95% upper bound of the RAW must be restricted less than 2, none of them in Fig. 6 is selected. However the RAW of the valve means the risk impact only when the valve is unavailable. As to the valve with low FV importance and high RAW, it is recommended to determine the usual STI being associated with both the STI and allowable outage time (AOT) when the failure is detected, according to another limitation which is discussed in the following section.

4. RISK-BASED O&M MANAGEMENT WHEN A FAILURE IS DETECTED

There is a case where it meaningfully increases plant risk (i.e. core damage frequency) to shutdown the reactor soon after detecting a failure of the safety-related standby component. The risk increment in detecting the failure is expressed as summation of the following factors:

(1) "outage risk" which is a risk increment due to outage of the failed component, and
(2) "shutdown risk" which is a risk on reactor decay heat removal being accompanied with reactor shutdown.

In this case, it is possible to derive the allowable time to continue reactor operation with corrective maintenance activity on the failed component and the adequate STI of intact trains so that the risk increment becomes minimum or less than a limiting value. [8]

4.1 CASE 1: UNREPAIRABLE FAILURE

The "outage risk" is accumulated and increases with time. When the failed component can not repaired under reactor operation, the "shutdown risk" stays at a constant value independent from outage time. In this case, it is possible to determine the AOT so that the total risk increment does not exceed a limiting value. Fig. 7 shows a relationship between the total risk increment and the outage time when unrepairable failure of the cover gas isolation AOV is detected. It is assumed that soon after detecting failure of one valve, an operator must confirm the other valve available. According to Fig. 7, if the total risk increment is required not to exceed 100% of total risk before failure happens (i.e., total risk keeps less than twice) per one failure event, it is allowed to keep operating the reactor nearly half year with surveillance test periodically implemented at least once per month.

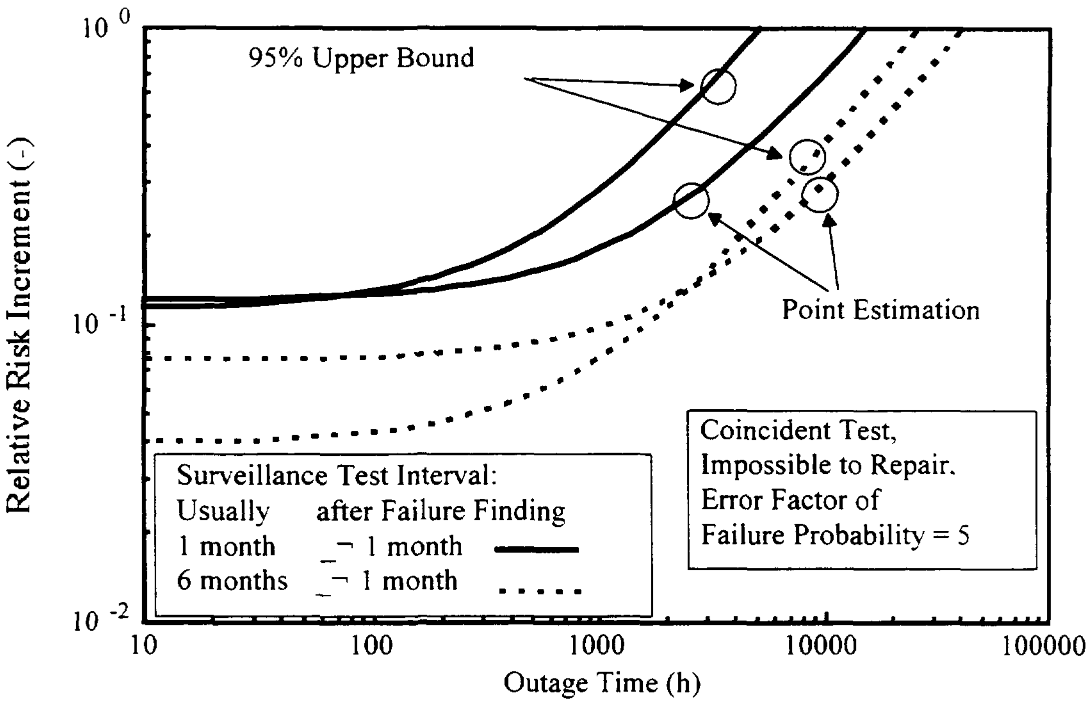

FIG. 7 Relationship between Relative Risk Increment and Outage Time When Detecting Unrepairable Failure of the Cover Gas Isolation AOV

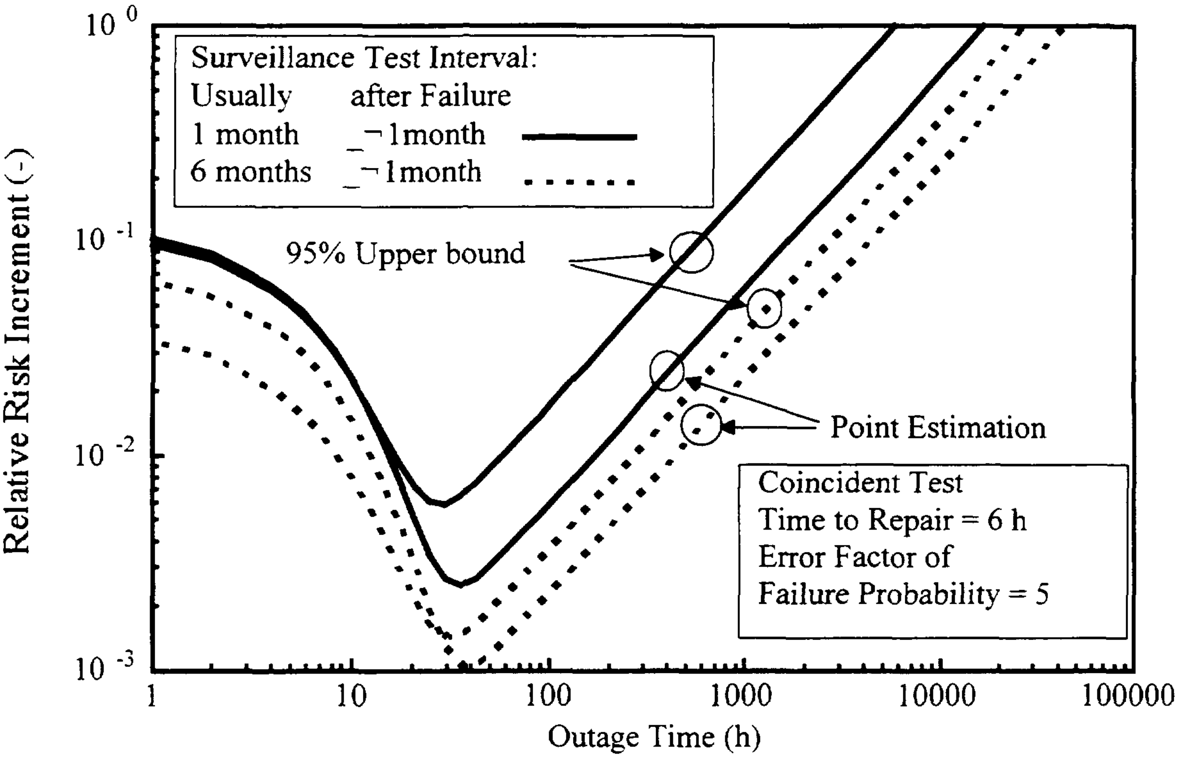

FIG. 8. Relationship between Relative Risk Increment and Outage Time When Failure of th Cover Gas Isolation AOV Is Detected

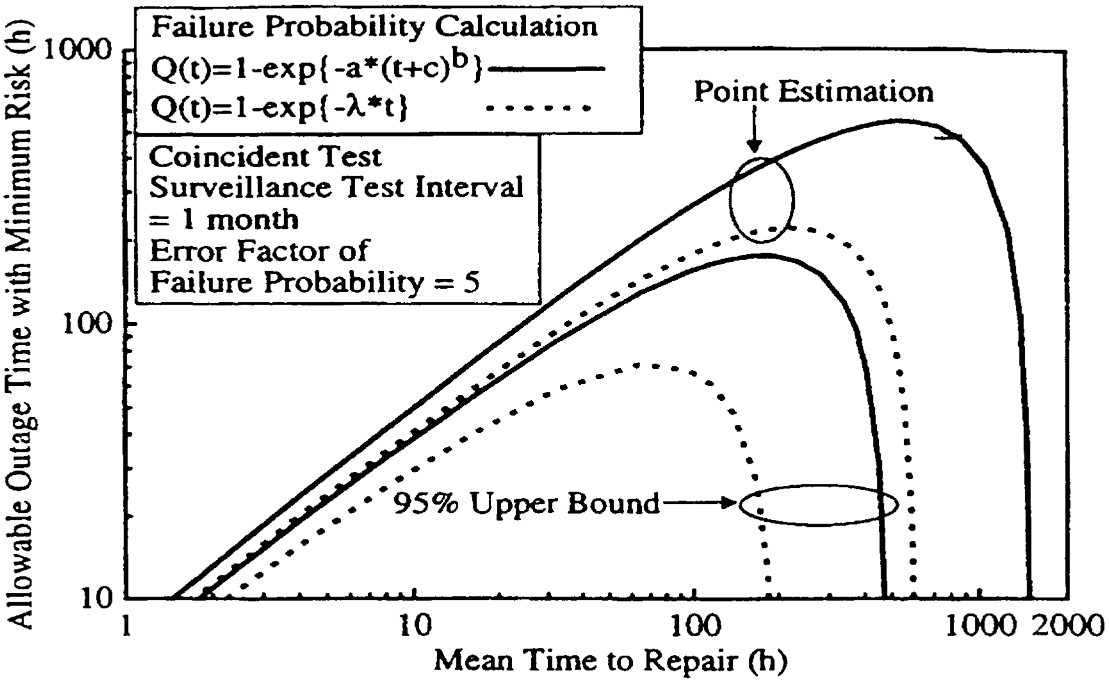

FIG. 9 AOT after Detecting Failure of the Cover Gas Isolation AOV

4.2 CASE 2: REPAIRABLE FAILURE

On the other hand, some of the failed component may be repairable in a short time under the reactor operation such as failure of electric parts in accessible area. If the AOT is longer than mean time to repair, the probability of restoration without reactor shutdown increases. Considering this effect, the "shutdown risk" decreases with time. It is possible to determine the AOT so that the total risk increment is the minimum.

Fig. 8 shows the total risk increment curve in case that the mean time to repair of the failed valve is 6 hours. In the one-month STI before failure, the AOT which gives the risk increment minimum becomes about 30 hours. The total risk increment becomes less than 1% of normal risk.

Fig. 9 depicts a relationship between the AOT with minimum risk increment and the mean time to repair of the cover gas isolation valve. In comparison of failure probability calculation, there is little difference within a factor of two in the AOTs corresponding to the mean time to repair shorter than several tens of hours. However in this case, the conventional method gives the AOT shorter than the detailed calculation. It can be said that we should apply the demand failure probability non-linearly dependent on the standby time into risk-based examination of the AOT.

5. CONCLUSION

The time trend of the demand failure probability of the valve was quantified based on the CORDS data and was applied to risk-based O&M examination of the safety-related isolation valves in the LMFBR model plant. As a result of the application, it was found that both the STI and AOT derived from the failure probability considering non-linear time dependency were longer than those from the conventional failure rate model under the same risk limitation. We concluded that it was better to apply the demand failure probability considering non-linear dependency on the standby time into the rational risk-based examination of the STI and AOT.

It became possible to examine management of the STI and AOT based on the operating experience in the real plant from the standpoint of risk assessment. It results from both accumulation of operation-/failure-history of individual component and development of various analytical techniques of risk considering the STI and AOT, that are the CORDS and LIPSAS.

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PSA AND METHODS TO OPTIMIZE THE MAINTENANCE OF SAFETY RELATED EQUIPMENT AT LAGUNA VERDE NPP

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Abstract

In this paper the most significant advances reached in the development of a project[1] to optimize the maintenance and surveillance tests of safety related equipment at Laguna Verde Nuclear Power Plant (LV NPP) are described.

The first part of this process consists of the application of risk-based results and techniques using the specific PSA Level 1 of LV. Before the use of these risk-based techniques in the generation and interpretation of results, it was necessary to perform a review of the risk criteria used in PSA applications for prioritization of maintainable components, determination of risk-based allowed outage times and surveillance test intervals, definition of component risk classes, determination of risk significances for systems and components within the scope of the maintenance rule, besides criteria used in a first approach to evaluate simultaneous inoperabilities impacting the risk. Partial results of the application of risk methods for AOTs, STIs and simultaneous inoperabilities are presented.

Also some results of the application of a selected method for maintenance optimization focused on reliability are explained. The results involve descriptions about the determination and prioritization of risk significant components, advances in the review of operational experience of a selected system to determine dominant failure modes and the comparison with those failures modeled in the LV PSA, operational experience review to assess the real times spent in maintenance activities for different types of components and, review of maintenance activities addressed to minimize the identified failure modes.

Within the regulatory assessment of the maintenance rule implementation in LV by the utility, some results of the independent work done by our regulatory staff to select those structures systems and components (SSCs) to be under the scope of the maintenance rule are presented. Also a discussion of our perspective about the establishment of performance criteria and performance goals of SSCs based on the specific LV operating experience is summarized. Finally this document includes some suggestions to make an initial evaluation of the maintenance and its effectiveness, and to define requirements of data for collecting plant data for accomplishment of the maintenance rule by the utility.

1.0 INTRODUCTION

The Laguna Verde Nuclear Power Plant (LV NPP) is located in the coasts of the Gulf of Mexico. This nuclear plant has two reactor units of the GE BWR/5 type with Mark II containment. The production capacity at full power is 654 MWe each unit. At this moment

[1] Project [1] under contract (RO-9294) of a Coordinated Research Program organized by the IAEA on *"Development of Methodologies for Optimization of Surveillance Testing and Maintenance of Safety Related Equipment at NPPs"*

the Unit 2 is being refueled to begin its 3rd operational cycle and the Unit 1 is operating on its 6th cycle. For the Unit 1 a PSA level 1 has been completed and the level 2 is in regulatory assessment phase. The PSA level 1 will be applicable to the Unit 2 after a comparative analysis to account for design differences between both units.

The following section contains a description of the methodology that is being structured to improve the LV operation through the maintenance optimization. The methodology is conformed by three major parts. The first one includes the use of risk-based techniques to analyze allowed outage times for equipment out of service (AOTs), surveillance test intervals (STI) and an approach to assess risk critical plant configurations due to simultaneous component inoperability. The second part is the application of a method to optimize the maintenance focusing on reliability techniques. The third part deals with the implementation process of the maintenance rule at LV by the utility and the regulatory point of view. The enrichment of this work is given by the inclusion of LV operational experience.

2.0 METHODOLOGY DESCRIPTION

The Figure 1 shows schematically the methodology to follow in the application of the techniques to optimize the maintenance and surveillance tests, including risk information, a Reliability Focused Maintenance (RFM) method and Maintenance Rule interactions. Each part is briefly described in the next subsections.

2.1 Allowed Outage Times (AOTs)

Basically the AOTs included in technical specifications (TS) are employed to perform activities of preventive maintenance (PM) or corrective maintenance (CM). During the AOT for a component, the lost of function of the component has a risk impact. The quantification of the AOT risk contribution (r) gives the accumulated risk over the allowed time d [2],

$$r = \Delta R \times d$$

where: ΔR is the risk increase associated to the unavailable component,
d is the time duration of the unavailability and,
r is the AOT risk contribution

Instead of perform only a ranking of the AOT contributors, a separation into classes may be done: Class 1 or Unacceptable risk contribution; Class 2 or Medium risk contribution; and Class 3 or Unimportant risk contribution.

With the risk classification it is possible to identify class 1 AOTs, to search for interactions with other component AOTs, tests, alignment checks, or a detailed analysis to modify the AOT with high risk contributions. The class 3 AOTs can be used to relax AOTs for the associated components in conditions of exception to TS or even for TS permanent changes. The class 2 can be considered to interact with class 1 or class 3. This strategy for the management of AOTs has a strong relationship with configuration control. By other side, imposing a limit to the risk contribution of an AOT, a risk-based AOT can be calculated, $d_{max} = r_{max}/\Delta R$. Then the risk-based AOT can be compared to the real time duration of maintenances and to the AOT established in the TS.

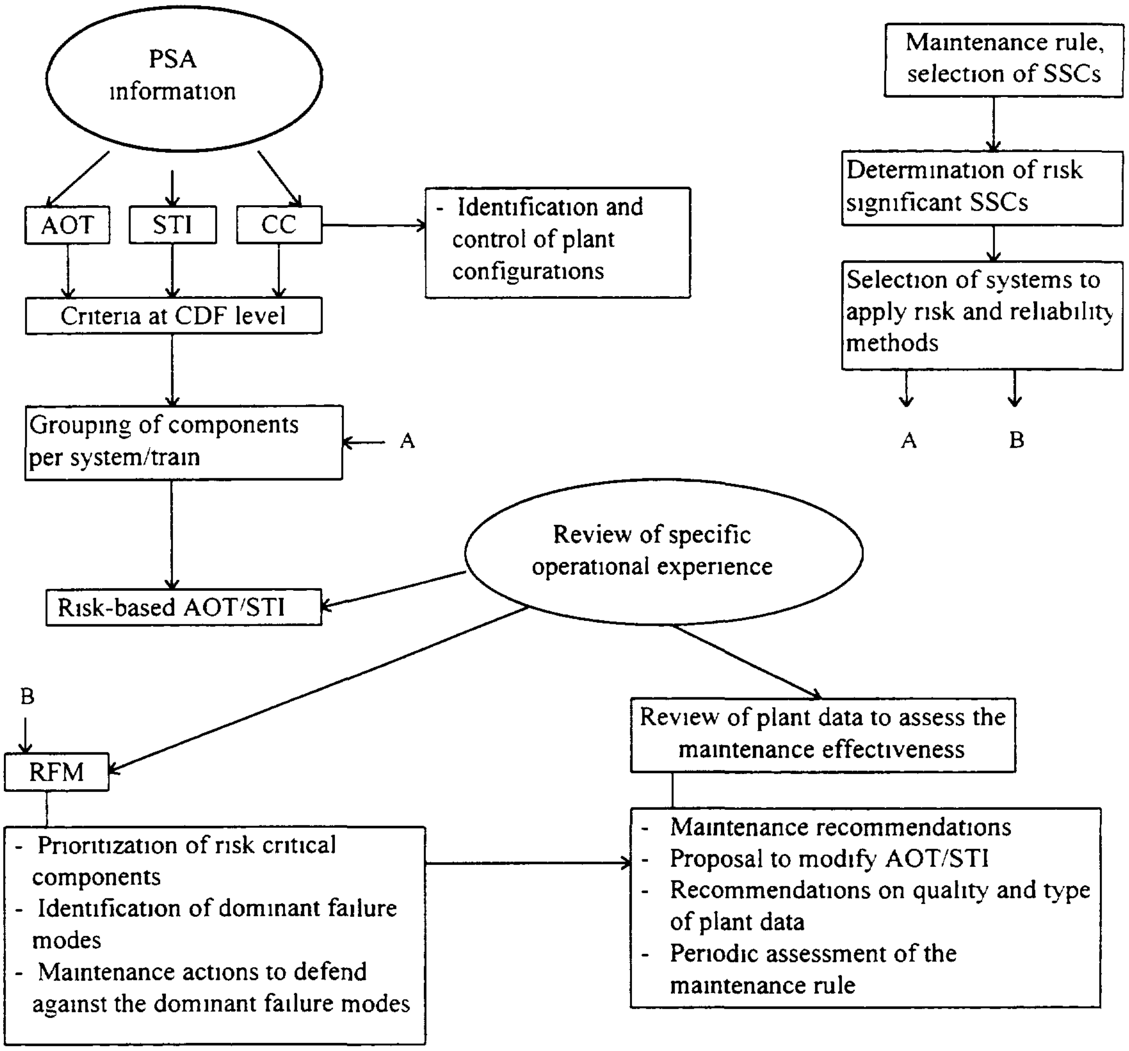
PSA information
AOT
STI
CC
- Identification and control of plant configurations
Criteria at CDF level
Grouping of components per system/train
A
Risk-based AOT/STI
Maintenance rule, selection of SSCs
Determination of risk significant SSCs
Selection of systems to apply risk and reliability methods
A
B
Review of specific operational experience
B
RFM
Review of plant data to assess the maintenance effectiveness
- Prioritization of risk critical components
- Identification of dominant failure modes
- Maintenance actions to defend against the dominant failure modes
- Maintenance recommendations
- Proposal to modify AOT/STI
- Recommendations on quality and type of plant data
- Periodic assessment of the maintenance rule

FIG 1 Integrated strategy to optimize maintenance and surveillance testing

2.2 Surveillance Test Intervals (STIs)

The analysis of STIs is based on the risk contributions arising from failures occurring between tests and detected at the moment of the test [2]. The STI risk contribution of a component is given by

$$R_D = \tfrac{1}{2}\lambda T\ \Delta R$$

Where ΔR is the risk increase when the component is found failed at the moment of the test, λ is the standby constant failure rate and T is the STI. Similar to the AOT risk contributors, the STIs can be classified and setting a limit to the risk contribution,

$$T_{max} = \frac{2 \cdot R_{Dmax}}{\lambda \cdot \Delta R}$$

In the determination of risk contributions it is possible to extend the analysis to include all the components tested in a specific test of a system. In this case the risk contribution for n components involved by the test is

$$R_D = \sum_{i=1}^{n} \tfrac{1}{2} \lambda_i \cdot T \cdot \Delta R$$

2.3 Configuration Control (CC)

Configuration control is the management of combinations of components simultaneously inoperable, to control the risk and assure a safety plant operation. Similar to the concepts used in AOTs, in configuration control the risk contributions r=ΔR×d are considered [2], [3], where ΔR is the risk increase caused by the simultaneous inoperability and d is the time duration for the configuration. Also the configurations can be classified in terms of their risk contributions as Unacceptable, critical and marginal, where marginal is a term used to identify risk insignificant configurations that can be significant depending on the time duration allowed to exist. A maximum time duration can be calculated if a limit to the risk contribution of a configuration is imposed, $d_{max} = r_{max}/\Delta R$.

2.4 Reliability Focused Maintenance

The RFM method for maintenance optimization is in essence a risk-based method to identify critical components to focus maintenance activities. The method begins with the identification of risk-critical components and continues with the determination of dominant failure modes of the critical components. Finally a detailed assessment of the maintenance is needed to address maintenance activities to avoid or minimize the dominant failure modes. The Figure 2 shows a summary of the general process of the RFM method. More details about variants of the method can be found in [4]. The shown method was selected because the convenience to cover maintenance rule aspects addressing the specific operational experience, besides the interactions with the risk significant failure modes from the PSA modeled components.

As it will be seen in section 3, the determination of the risk critical or significant components involves only the systems and components modeled in the LV PSA level 1 using the typical risk importance measures, and for purposes of the maintenance rule, other systems not modeled in PSA can be classified as risk significant through an analysis to search for a relationship to systems modeled in PSA. The determination of dominant failure modes is performed from a review of the specific operational experience of a selected set of components, and the maintenance recommendations are reviewed to conclude if the present maintenance programs tend to avoid the identified failure modes, or those programs need to be improved.

2.5 Maintenance Rule

The Maintenance Rule regulation [5] has been required to the utility for its implementation to the LV NPP. With the aid of the NUMARC 93-02 [6] guidelines and [7], our regulatory staff made a selection of structures, systems and components (SSCs) to be under the scope of the rule. This selection was used to evaluate the selection made by the utility. As established in the 10CFR50.65, the required SSCs are the safety related SSCs and those non safety related that help to mitigate transients and accidents, are within emergency operating procedures, those SSCs that can cause a scram or actuation of a safety related SSCs, and those that avoid a safety related SSC to fulfill its intended function. The scope of the present project is to cover only safety related SSCs. After the selection, specific criteria are applied to determine risk significant SSCs.

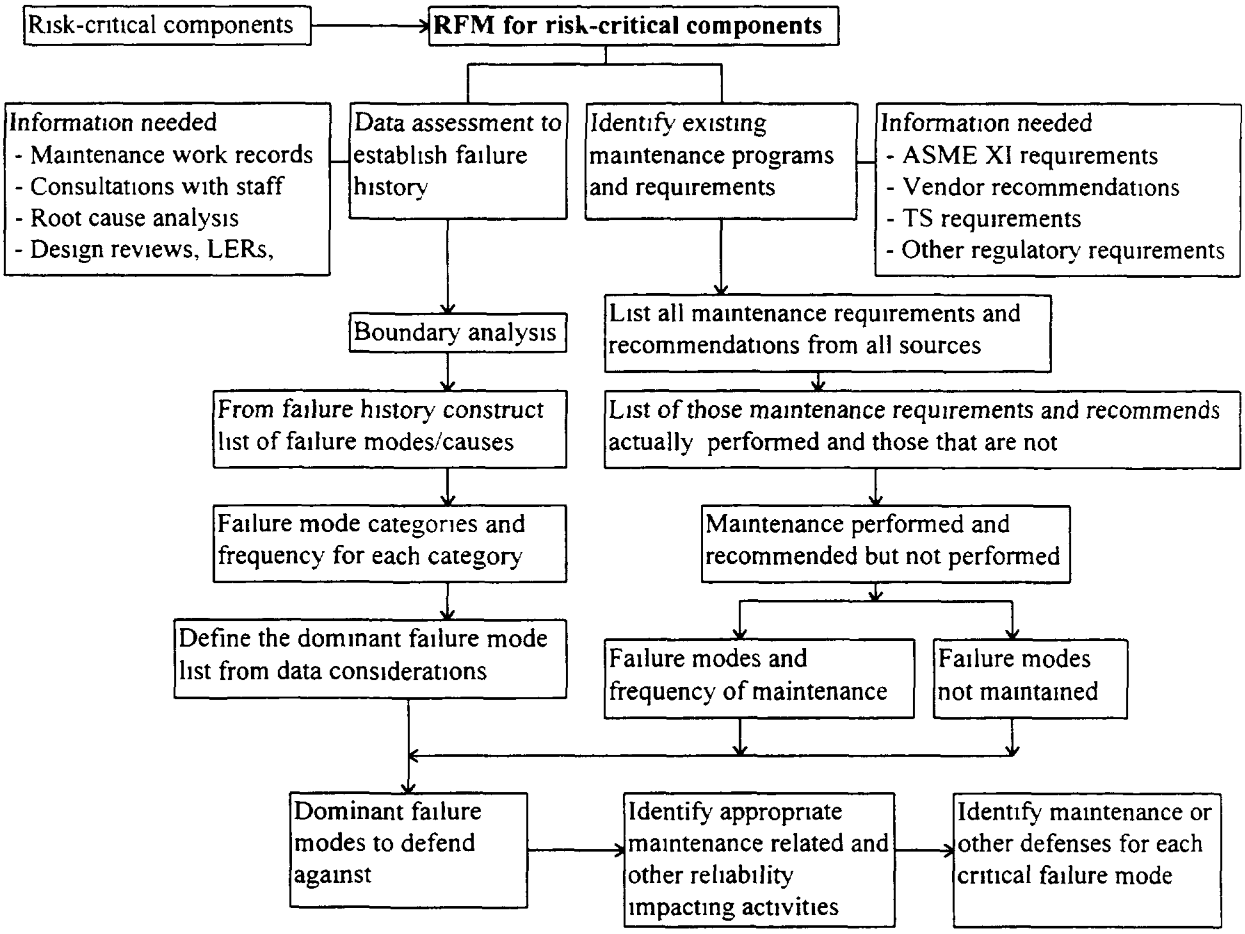
Risk-critical components
RFM for risk-critical components
Information needed
- Maintenance work records
- Consultations with staff
- Root cause analysis
- Design reviews, LERs,
Data assessment to establish failure history
Identify existing maintenance programs and requirements
Information needed
- ASME XI requirements
- Vendor recommendations
- TS requirements
- Other regulatory requirements
Boundary analysis
List all maintenance requirements and recommendations from all sources
From failure history construct list of failure modes/causes
List of those maintenance requirements and recommends actually performed and those that are not
Failure mode categories and frequency for each category
Maintenance performed and recommended but not performed
Define the dominant failure mode list from data considerations
Failure modes and frequency of maintenance
Failure modes not maintained
Dominant failure modes to defend against
Identify appropriate maintenance related and other reliability impacting activities
Identify maintenance or other defenses for each critical failure mode

FIG 2 Summary of the general process for Reliability Focused Maintenance.

3.0 PRELIMINARY RESULTS

Before the generation of results based on risk techniques for AOTs, STIs, configuration control and identification of risk significant SSCs, the following risk criteria were selected to use in the PSA applications:

Table 1. Criteria for classification and maximum risk contributions [8].

	AOT	STI	Configuration Control	
Unacceptable	$1E\text{-}06 \leq r$	$1E\text{-}06 \leq R_D$	Unacceptable	$1E\text{-}02 \leq \Delta R$
Medium	$1E\text{-}08 \leq r < 1E\text{-}06$	$1E\text{-}08 \leq R_D < 1E\text{-}06$	Critical	$1E\text{-}04 \leq \Delta R < 1E\text{-}02$
Unimportant	$r < 1E\text{-}08$	$R_D < 1E\text{-}08$	Marginal	$\Delta R < 1E\text{-}04$
	r_{max}=1%CDF	R_{Dmax}=1%CDF	r_{max} =1%CDF	

Table 2. Criteria to select risk significant structures, systems and components [6].

A SSC is risk significant if:

- its risk importance RRW contributes to the 99% of the accumulated RRWs, or its RRW exceeds the 0.5% of the baseline CDF.
- appears in the ranked minimal cutsets accounting for the 90% of the baseline CDF
- its risk achievement worth RAW shows at least the doubling of the baseline CDF

Table 3. Risk results for the AOTs of RHR-A equipment.

Residual Heat Removal System - A					
Equipment	**ΔR**	**Class**	**TS AOT (days)**	**d_{max} (hours)**	**d_{max} (days)**
Heat exchanger	4.37E-06	3	3*, 7	2982	124
Moto-pump	6.29E-06	3	3, 7	2070	86
HX & moto-pump	7.75E-06	3	3, 7	1680	70
HVAC pump room	1.35E-06	3	7	9616	401

** 3 days for the shutdown cooling mode.*

Table 4. Results of components grouping for systems risk analysis with IMPOSUB

HPCS	AOT: 14 days.			RCIC	AOT: 14 days		
Equipment	**ΔR**	**r**	**d_{max} (hrs)**	**Line**	**ΔR**	**r**	**d_{max} (hrs)**
Electrical	5.742E-04	2.202E-05	23	Steam	3.279E-04	1.258E-05	40
CV'S & XV'S	5.895E-04	2.261E-05	22	Suction/Pool	2.940E-04	1.128E-05	44
Pump	5.795E-04	2.223E-05	23	Suction/CST	2.490E-04	9.551E-06	52
Maintenance	5.752E-04	2.206E-05	23	Common line	1.300E-04	4.986E-06	100
MOVs	5.213E-04	2.000E-05	25	Injection	3.970E-04	1.523E-05	33
O. Restr.	5.765E-04	2.211E-05	23	Condensate	1.596E-04	6.122E-06	82
HVAC	5.514E-04	2.115E-05	24	Turbine-Pump	2.501E-04	9.593E-06	52
Suppression Pool	3.723E-04	1.428E-05	35	Testing line	2.160E-05	8.285E-07	603
Filter (CST)	1.900E-06	7.288E-08	6856				

Table 5. Preliminary results for STI risk calculations.

HPCS			RCIC		
Components	**TS STI (days)**	**Risk-based STI (days)**	**Components**	**TS STI (days)**	**Risk-based STI (days)**
Pump, Valves V2 & V3, MOVs 8187 & 8169, Strainer.	92	91	Pump, Valves V1, V2 & V44, MOVs 8132 & 8100, Strainer.	92	83
LPCI-A			**RHR-A**		
Components	**TS STI (days)**	**Risk-based STI (days)**	**Components**	**TS STI (days)**	**Risk-based STI (days)**
Pump, MOVs 8221, 8204, 8226 & 8219, Strainer,	92	368	Pump, MOVs 8226 & 8219, Strainer.	92	366

3.1 Allowed Outage Times

In order to support the analyses for the risk impact due to the inoperability of components, the package R&R [9] was used for the development of a computer program to calculate risk importances of equipment and components. The code was named IMPOSUB [10] because of its particular ability to subsume non-minimal cutsets resulting when an unavailability is set to a true state.

The Table 3 above presents the results for the analysis of the AOTs for the RHR system. The TS AOT for this system is 7 days, but particularly for the suppression pool cooling mode the TS limit the AOT to 3 days. The most restrictive risk-based AOT of 70 days is quite longer than the TS AOTs. The long risk-based AOT for the heat exchanger results because of a redundancy in the cooling modes and two redundancies in low pressure injection. Jointly with deterministic analysis these results are being used to evaluate a solicitude of the utility to change the 3 days AOT to 7 days. Table 4 shows the results obtained for unavailability of emergency system components as pumps, valves, filters, etc., and for types of lines in a system, i.e., suction, injection, etc. In the same table appear the risk-based AOTs associated with the inoperability of each component.

The computed risk-based AOTs need to be improved by effects like the risk decrease effect due to testing or verification of operability of redundant components and common cause failure considerations

3.2 Surveillance Test Intervals

Table 5 shows the results for STI risk contributors, mainly for tested components in trains during power operation not providing water to the reactor vessel. Also the table contains the results for risk-based STIs and the present STIs required by TS.

For the HPCS and RCIC systems, practically the risk-based STIs are the same as those required by TS. Because of the lower risk importance of the LPCI-A and RHR-A systems their corresponding STIs are greater than the times required by TS.

Table 6. Different classes of risk configurations.

CONFIGURATION	ΔR(/yr)	d_{max} (hours)	CLASS
DG-1A *unavailable for maintenance* RRA-FN-002 there is no signal to start	*1.02E-02*	*1.3 hours*	*Unacceptable*
Battery set 1A125 lost of function RRA-FN-002 unavailable for maintenance	1.00E-02	1.3 hours	Unacceptable
AC-I-141A1C unavailable for maintenance SLC manual valve V-9 open	1.06E-04	5 days	Critical
RCIC MV-8100 fails to remain open SLC manual valve V-9 open	1.01E-04	5 days	Critical
RRA-FN-001 unavailable for maintenance SLC manual valve V-9 open	9.81E-05	6 days	Marginal
AC-I-141A1C unavailable for maintenance SLC-P-001 unavailable for maintenance	3.92E-05	14 days	Marginal

3.3 Configuration Control.

Through the application of a simple process to generate possible combinations of unavailable equipment and verification of existence of such a combinations, the results of some existing configurations were obtained and they appear in table 6.

In table 6 it is observed the maximum time to allow a configuration to be present. If this time is exceeded, the configuration risk contribution will exceed the 1% of the baseline CDF, taken as risk criterion. The configurations identified have not been analyzed from an operational standpoint because it is necessary first to review the criteria to select components as candidates to form configurations.

3.4 Reliability Focused Maintenance

The determination of risk critical systems and components was carried out with the application of the criteria shown in Table 2 to the main results of the LV PSA. The process consisted with the obtainment of a list of components which were used to generate a list of risk significant systems. This first list covered the systems modeled in the PSA in an explicit way. An analysis of the initiating events was perform to include some systems not considered in an explicit way. In order to finalize the list for maintenance purposes some others methods were used by the utility to assign risk significances to the systems not explicitly modeled in PSA. The Table 7 shows some risk-critical safety related systems that are related to the PSA whether explicit or implicitly. This type of list contains systems selected for this project and for maintenance rule purposes.

Table 7. Examples of risk significant systems.

SYSTEM	DESCRIPTION	TYPE OF MODELING
R62	125 VDC Power Distribution	Implicit
E12	Residual Heat Removal System	Explicit
B35	Reactor Recirculation Control System	Implicit
E51	Reactor Core Isolation Cooling	Explicit
X60	DG Area Vent. and Air-conditioned system	Implicit
E22	High Pressure Core Spray System	Explicit
C72	Reactor Protection System	Implicit
P41	Nuclear Service Water	Implicit
N61	Removal of Air from the Main Condenser	Implicit

From the SSCs selected by CNSNS during the assessment of the maintenance rule a safety related system was selected to apply the method of RFM. This system is the Residual Heat Removal system (RHR). Nevertheless the RHR is a system of medium risk importance in the classification. it was selected for a first approach in the application of the defined RFM method because it has several operational mode, with two redundant loops and a loop dedicated to low pressure injection.

An initial technical visit was carried out to LV just to know the type, quality, location and control of the information generated by the distinct plant areas. A second visit had the intention to get information about of failure data for the RHR system. For this purpose the surveillance and maintenance records were reviewed taking into consideration the impact of the failures on the functionality and operability of the system. In order to classify the failures they were categorized as

Primary Failure (PF): Failures leading to a loss of the safety function,
Secondary Failure (SF): Failures leaving the components in a degrade state, but the safety function is not lost.
Maintenance Failures (MF): Errors occurred during the execution of maintenance.
Maintenance Recommendations (MR): Deficiencies identified in procedures, practices, recommended frequencies and tests.

For convenience in the analysis of the identified failures, these were separated by engineering disciplines: mechanical, electrical and instrumentation and control. The Table 8 includes the most important failure events found in the review and analysis of maintenance and surveillance records since 1991 up to the first third of this year. The failures shown are mainly those considered PF and the last two events classified as MF and MR.

Table 8. Dominant failure modes found in the review of the maintenance history of the RHR system.

DATE	COMPONENT	AREA	TIME (H)	DESCRIPTION
92-09-02	DPIS-N012B	I&C	4	Spurious Signal to initiate the shutdown cooling mode
92-10-03	B35-PS-018A	I&C	3	Pressure switch out of tolerance, high signal causes system isolation.
92-10-19	MV-8248	Mec	8	Failure to open in shutdown cooling mode.
94-03-10	MV-8202A	Elec	8	Minimum flow valve, failure to open.
95-09-28	PS-N019A	I& C	4	Pressure switch out of tolerance, high signal causes system isolation.
96-01-22	MV-8294B	Mec	2	Failure to open totally (40%).
96-03-11	PS-N033A	I&C	1	Pressure switch out of tolerance.
96-06-19	PS-N033A	I&C	1	Pressure switch out of tolerance.
96-10-23	1-MCC-1B1-C	Elec	10	Thermic relay damage. Failure to control MV-8202B
95-09-23	AV-8206B	Elec	8	Valve actuator, lack of lub. Failure to open.
96-12-01	MV-8210B	Mec	?	Limitorque switch misinstalled giving failure in the position indication.

In the final step for classification of failures, the analysis of the information was completed with consultation of other sources like the licensing event reports, plant databases and periodical reports from the plant's systems engineering area. The identified dominant failure modes were compared with failure modes modeled in the PSA for this system. The only conclusion from this comparison is that at least for this system the failure mode "Permanent signal to close" is not considered in the system's fault tree model for the failure modes of the valves MV-8248 and MV-8247. This type of failure cause a lost of shutdown cooling mode because the failure of pressure instruments.

The events classified as secondary failures are in evaluation phase because almost all these events are from internal and external leakage. The criteria to select them as failures were to consider failures in components modeled in PSA and that the failure cause a system inoperability, or violation of the technical specifications.

The next task is to perform an evaluation of maintenance programs and surveillance requirements related to the identified critical components of the RHR system. It is intended that this evaluation will cover maintenance activities carried out and those not performed but recommended by the vendor. The corresponding results will serve to identify maintenance activities needed for minimization of the found critical failure modes.

3.5 Maintenance Rule

This section describes the relationship between the present project and the implementation of the maintenance rule required to the utility. The maintenance rule covers the systems mentioned in section 2.5 while this project intends to prove its own methodology on safety related systems.

It is considered that for the satisfactory accomplishment of the maintenance rule it is necessary to have an adequate establishment of performance goals and performance criteria for the SSCs to be evaluated through the effectiveness of their maintenance. An adequate performance goals setting can be achieved from the specific plant data and compared to the wide industry to establish goals and criteria. In this way, for the maintenance rule not working at full implementation can give some benefits to make an initial evaluation of the maintenance by means of a plant data review for the determination of the initial status and setting of the initial performance goals. Then in a following evaluation of maintenance the initial settings can be compared.

Another objectives of the two technical visits to LV were the determination of the usefulness of existing plant data to require the licensee an initial assessment of the maintenance for its effectiveness, to start the maintenance monitoring and periodic assessments; and the demonstration of the usefulness of existing plant data to define performance goals and criteria, for instance, number of failures, reliability, and unavailability for risk significant components. Although it is true that it is necessary to have a compilation data system to fulfill the compromises with the rule by the utility, it is considered that the present plant data information is spread but it has enough quality to be used for an initial maintenance assessment. For maintenance rule purposes, also a series of requirements and recommendations are being prepared about the level of detail and type of data to have a satisfactory accomplishment of the rule by the utility. In this sense it is strongly suggested to review the document referenced in [11] to get insights about what type of plant data are the most recommended for maintenance rule objectives.

4.0 CONCLUSIONS

Although the integration of risk and reliability methods is in its first approach and in experimenting phase through its practical application, it is noted that it is possible to optimize maintenance and surveillances mixing risk results of AOTs and STIs with data from operational experience used in the application of the RFM method to risk significant systems. Then, it is expected to integrate risk information to the results from RFM and maintenance rule.

The risk analysis of AOTs, STIs, and plant configurations need to be improved in the use of the computational processes and to consider assumptions like testing and alignment of redundant components and modifications for common cause failure events.

In order to conclude the definition of strategies to follow in the RFM method, it is necessary to complete the application of the method to the selected system. The final strategy will serve to improve the application of the method to other systems or groups of components.

The deficiencies found in the quality and type of existing data in LV NPP do not make impossible the application of the present methodology to another systems, to the contrary, they give the opportunity to define data requirements to accomplish successfully with the maintenance rule regulation. In a future this data requirements will serve for reassessment of system reliability analyses.

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INDUSTRY COMPARISON THROUGH SYSTEMATIC SELF ASSESSMENT OF THE MAINTENANCE FUNCTION OF EACH OPERATION COMPANY OF THE NORWEGIAN CONTINENTAL SHELF

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Abstract

In this paper The Norwegian Petroleum Directorate (NPD) will describe a method for internal evaluation or self-assessment of the maintenance function of each operating oil company in the Norwegian offshore petroleum industry.

Due to the current trends of cost reduction, demanning and downsizing, the industry has started to use RCM and other risk based maintenance techniques.

Judged by the experience gained so far in this industry, the successful implementation of RCM and other risk based maintenance optimisation techniques seems to be closely linked with the existence of a efficient and professional maintenance management system.

NPD is therefore now developing a guideline for systematic self-assessment of the maintenance function and the maintenance management system. The guideline has been developed based on a method developed by SKI. NPD has in close co-operation with the industry adapted SKI's guideline to fit the problems and challenges of the petroleum industry.

It is NPD's intention to highlight different strategic, organisational and administrative issues linked with the introduction of risk based maintenance methods in the guideline. NPD also try to reflect "best practices" in the industry in the guideline in order to provoke the oil companies to review their own practices in different areas.

The results so far from using this method for self-assessment will be presented, and both the advantages and disadvantages of such an approach will be discussed.

1. Introduction

1.1 Norwegian Petroleum Industry

First a few words about the Norwegian petroleum industry. Norway is today the world`s second largest oil exporting country and we supply Europe with approximately 20% of the gas needed. The amount of energy produced on each platform varies very much, but to give an idea, the Troll A platform, produces equivalent of 40 000 MW/D.

All of our petroleum production comes from offshore fields, we have both subsea installations, and various types of platforms, varying from large integrated platforms with both drilling, production and accommodation to minor rather simple platforms performing one major function, e.g. drilling or gas compression. Among the main contributors to risk are blowouts from the wells, which can cause both major environmental damage and if ignited, to the loss of a platform with substantial loss of lives. Another major risk contributor is process leaks leading to fire and explosion and thus impairing vital safety functions such as escape, evacuation and control.

Among the predominant external risk factors are collisions, helicopter crashes and dropped objects.

1.2 Norwegian Petroleum Directorate

The Norwegian Petroleum Directorate (NPD) is supervising the safety and working environment of the petroleum industry in Norway based upon the following main principles:

- focus on each individual operating company's own responsibility for prudent operations through regulations focusing on management and different types of administrative and organisational requirements
- objective or goal oriented regulations, moving away from prescriptive regulations towards risk oriented regulations
- regulations based on the defence-in-depth concept (no single failure shall lead to ---)

In the maintenance area, the regulations have been very functional, but because of their general nature, they have not been very predictive for the industry. This will be changed in the ongoing revision of the regulations.

2. Maintenance – a challenge to the Norwegian petroleum industry

The actual lifetime of Norwegian oil and gas fields are longer compared to other nations both due to large reservoirs and extensive programs for increased oil and gas recovery. The Troll field has a planned lifetime of 50 – 70 years; other major fields have an expected lifetime of 40 – 50 years.

In such a perspective with a production period often extending the original design lifetime, maintenance becomes an important issue. Since the oldest installations now are around 30 years various ageing problems have emerged. Various types of corrosion, erosion, fatigue and embrittlement have lead to undesired events. This is an increasing problem, and existing inspection programs and inspection methods are not able to fully predict where this will happen, when it will happen and how often this will happen.

Likewise the current trends with cost reduction, demanning and use of different maintenance optimisation techniques such as reliability based maintenance (RCM); reliability based inspection (RBI) new demands have been put on the maintenance organisations.

3. RCM applications in Norway – some experience

3.1 Data and methods

The first RCM adaptations in this industry came in the early nineties. The industry used consultants from the nuclear industry. The consultants encountered several challenges, for instance in the area of data collection. The industry has a joint database called OREDA, but OREDA data doesn't cover all of the platform equipment. The use of platform specific data requires a lot of work, because data have to be analysed manually. The probabilistic risk analysis (PSA) is not as detailed as in nuclear and therefore can't be used directly.

There was no screening process so a full RCM was done for all of the platform equipment, and therefore creating a lot of work.

3.2 Follow up and management support

Classifications and calculations was paper based, and therefore quickly outdated because of the frequent modifications on the platforms. Not enough resources were spent on building in-house competence to carry on the work, like updating and adjusting the PM programs in accordance with experience (analysis of failure data).

Management support were weak, and as a consequence not enough was done to make necessary changes in data collection, and preparing new computer based maintenance management systems for the use of RCM and so on (data on failure modes and failure causes). Analyses of maintenance data were not improved. This list could be made much longer. But the point to be illustrated is that implementation of these kind of techniques must work hand in hand with a professional upgrade of the whole maintenance function.

3.3 New approaches

Since a big part of the industry was scared off by the amount of work doing a proper RCM study, various consultants now saw a marked offering different "RCM-light" versions. Briefly explained, different consultants made versions without all the work of a proper RCM study. But with all the conservatism built in the models, the outcome was poor in terms of actual reduction in PM programs.

But it also should be mentioned that some of the operators have worked very hard applying RCM both on new and old platforms and with satisfying results both from a safety and economical point of view. One of the operators has also done a very promising experiment of reliability based operations.

3.4 Authority follow-up in the RCM/RBI area

A national standard for criticality classification was made last year. NPD has several comments to this standard, and a report was sent out to the industry with our comments. Among our comments to the standards were:

- In the consequence matrix most critical items are critical both to production and safety. To mix criticality for production and safety in the same category is not recommendable.

  Safety critical items should be clearly defined.

- Criticality classification will be done on a functional level, and items on a sub-function level will be given the same criticality as the sub-function they are part of. This approach will be conservative in operation.

- Some safety related sub-functions may have a low probability of failure, but still is very safety critical. In such cases a risk analysis must be done to assess the probability of the consequence of the failure of the sub-function.

- This standard doesn't from a safety point of view, reflect "best practices" in the industry.

- Various inconsistencies in the use and interpretation of data were commented on.

NPD will follow the development of standards in this area very closely, and the standards will be carefully reviewed by NPD (with the use of consultants). And in parallel NPD will try to influence the industry to work on the necessary organisational improvements.

4. RCM - a organisational challenge

One of NPD's main concerns is that the use of these kinds of optimisation techniques require what we have chosen to call an "administrative infrastructure" that is more advanced and complex than what is required for more conventional maintenance management methods. This is especially true when applying RCM/RBI on old installations. NPD realised that we needed to develop methods to assess the quality of the different maintenance organisations. These methods had to be more effective and give us the possibility to address all relevant operators instead of dealing with them one by one.

NPD learned that SKI had been developing methods that was very much in line with what we were looking for. SKI had developed what they called a baseline study, which is a self-assessment of the maintenance function in each operating company. SKI very generously allowed us to use their material to start a similar process in Norway.

NPD wanted to use this material and methods in order to underline the responsibilities of the line management for safe operations and maintenance, and also to serve as a basis for improving the control function in the operating companies. This is an opportunity for maintenance people to describe their own problems and challenges and get top management's attention and focus.

5. Baseline study

5.1 Principles

The model in the baseline study have been based on the principles guiding most quality assurance or quality management programs:

- Quality systems shall contribute to continuous improvement

- Problems should be identified and solutions standardised. The problem handling should be processoriented, and integrating across organisational boundaries. Another important aspect Is that it should be preventive.

Different parts of the quality system should be taken care of by a specific set of work processes (that could be documented by flow diagrams, procedures etc). The work processes should also be designed as Quality loops and be oriented towards problem solving.

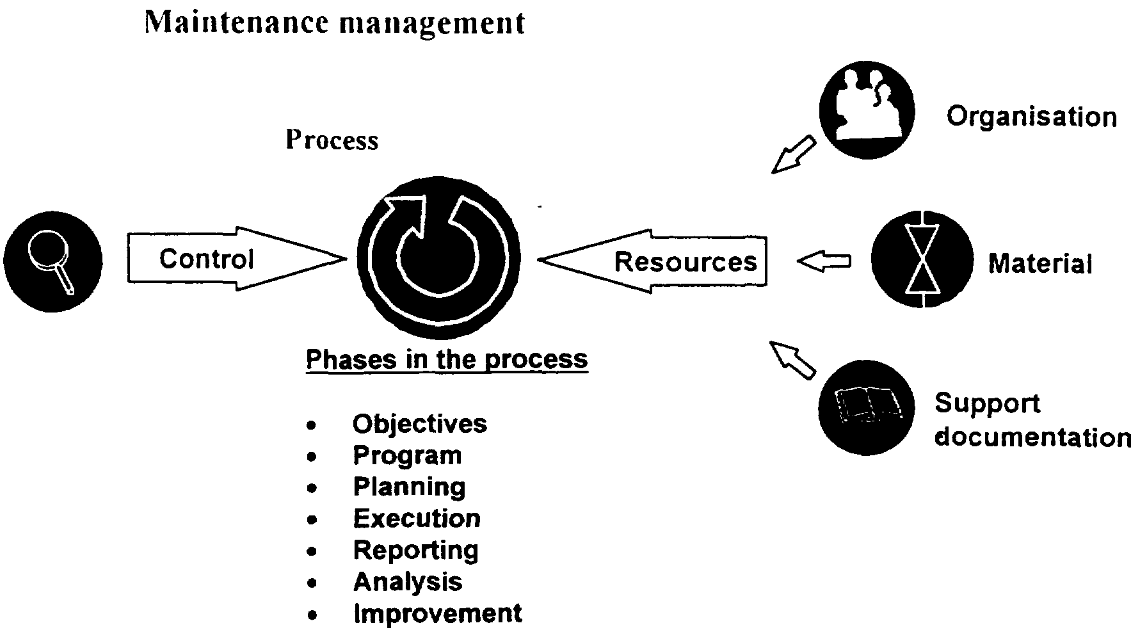

Figure 1

5.2 Objectives of a baseline study

For the operators the baseline study should a part of continuous improvement process of their systems for safety related maintenance.

The baseline study should enable the authorities to prioritise between operators and fields and to prioritise certain problem areas or focal areas.

The baseline should also give both the operators and the authorities a mutual understanding of the weaknesses, strengths and improvement areas of the maintenance management system and thus form a basis for further communication and follow up.

5.3 The focus of the baseline study

The focus of the baseline study should be directed towards the maintenance management systems quality with regard to:

- technical condition
- safe operations

5.4 Status of the baseline study

The report from the baseline study should clearly express the management's statement regarding:

- the quality of the maintenance management system (holistic view)
- which improvements must be done where, by whom and when

6. Guidelines for performing a baseline study

NPD has developed a guideline for performing a baseline study. The guideline will try to reflect the principles described under item 6.1 with emphasis on work processes and possible improvements.

Figure 1 describes the model NPD use in the guideline. Based on the quality management philosophy in our regulations NPD has focused on the management loop and the different stages in this loop. In addition, both resources needed and necessary control functions are included.

The guideline is a 60 - 70 pages document-containing questions related to the different elements in the model. In addition, where relevant, comments from other NPD reports have been included. The questions and comments are supposed to reflect todays and future safety challenges and problems.

The report from the operators is expected to be 25 - 30 pages. One page for each element, with only the most important information regarding background/history, status of today, problems identified and action plans.

7. Experience so far

The idea of performing baseline studies was welcomed by the industry and three of the operators volunteered to do the first pilot studies. Two pilot baseline studies have been performed, and the third is starting this week.

Performing a baseline study requires around 500 man-hours. NPD has received the reports from the two first pilot studies. The feedback from the pilot studies is that the concept is useful and can be applied in other areas. Therefore one of the operators has taken an initiative to include perform a baseline of operations as well.

The holistic view seems to be useful, and due to this approach, some improvement areas not earlier recognised, have been identified. Identifying work processes, defining process-owners and improving processes related to the different elements is seen as useful.

NPD intends to use "Best practices in the industry – and from other industries" as a way of transferring experience to other operators. The baseline is now under continuous update and will be so for at least another half year. Then there will be at least an annual update.

Among the disadvantages are that this self-assessment can be done in a superficial way, and be a means of concealing problems, not identifying and solving problems. Also a baseline study can be seen as an authority requirement and therefore something that has to be done, but without any real commitment from top management. In that case, the operator is not fooling the authorities, but themselves.

An interesting question is how long such an approach can be effective? There is no clear answer to that, but the self-assessment approach should last for some years.

RELIABILITY CENTERED MAINTENANCE (RCM) PROGRAM FOR CHASHMA NPP (CHASNUPP)

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Abstract

This paper describes the proposed Reliability Centered Maintenance (RCM) program for Chashma Nuclear Power Plant (CHASNUPP). Major steps are the identification of risk critical components and the implementation of RCM procedures. Identification of risk critical components is based upon the CHASNUPP level 1 PSA results (performed under IAEA TC Project PAK/9/019) which is near completion. The other requirements for implementation of RCM program is the qualitative analysis to be performed for identifying the dominant potential failure modes of each risk critical component and determination of the necessary maintenance activities, required to ensure reliable operation of the identified risk critical components. Implementation of RCM program for these components will lead to improvement in plant availability and safety together with reduction in the maintenance cost. Development / implementation of RCM program at this stage will help the CHASNUPP Maintenance department who is now developing the maintenance program / procedures for CHASNUPP.

1. INTRODUCTION

Chashma Nuclear Power Plant is going to be commissioned in 1999. A level 1 PSA of CHASNUPP is being performed under an IAEA TC project PAK/9/019, first set of the quantification results has been obtained and reviewed. Based upon the first set of quantification results, a list of risk critical components has been developed . This list of risk critical components will serve as the basis for implementation of reliability centered maintenance (RCM) program for CHASNUPP in future. The objective of this paper is to outline a reliability centered maintenance program for CHASNUPP risk critical components, identified from PSA level 1 results.

The selection of these risk critical components is based upon the fact that these enable the plant system to fulfill their essential safety function and the failure of these components may initiate challenges to safety systems. RCM for these components may lead to improvement in plant availability and safety together with reduction in the maintenance cost.

As right now the maintenance program for CHASNUPP is under preparation, this paper helps the Maintenance department to provide a criteria and guidance for establishing a reliability centered maintenance program for the risk critical components that accounts for the unique reliability characteristics of each component.

One major purpose of RCM is to provide a systematic set of criteria. based upon risk. for identifying which of the components considered in the process are to be defined as critical to risk (risk critical components) and which are not. Only risk critical components are included within the scope of RCM program.

The proposed RCM program applies to a portion of the total plant maintenance program. Plant equipment receives and should continue to receive maintenance for reasons other than the RCM program described herein. Use of RCM program will not preclude other maintenance activities, the maintenance people considers necessary for proper maintenance of the equipment.

The reliability centered maintenance program, therefore, consists of following two major steps based upon the above description:

1. Identifying risk critical components

2. Determining the necessary maintenance activities, required to ensure reliable operation of the identified risk critical components

The overall process and the first step are "risk focused"; the program for individual components is "reliability focused."

The implementation of the top level program for RCM is illustrated in figure 1-1. The first major step is to determine if the component is critical. If a component is not risk critical, it is not included in the domain of the overall RCM program. If the component is determined to be critical to risk then it is incorporated into a RCM program.

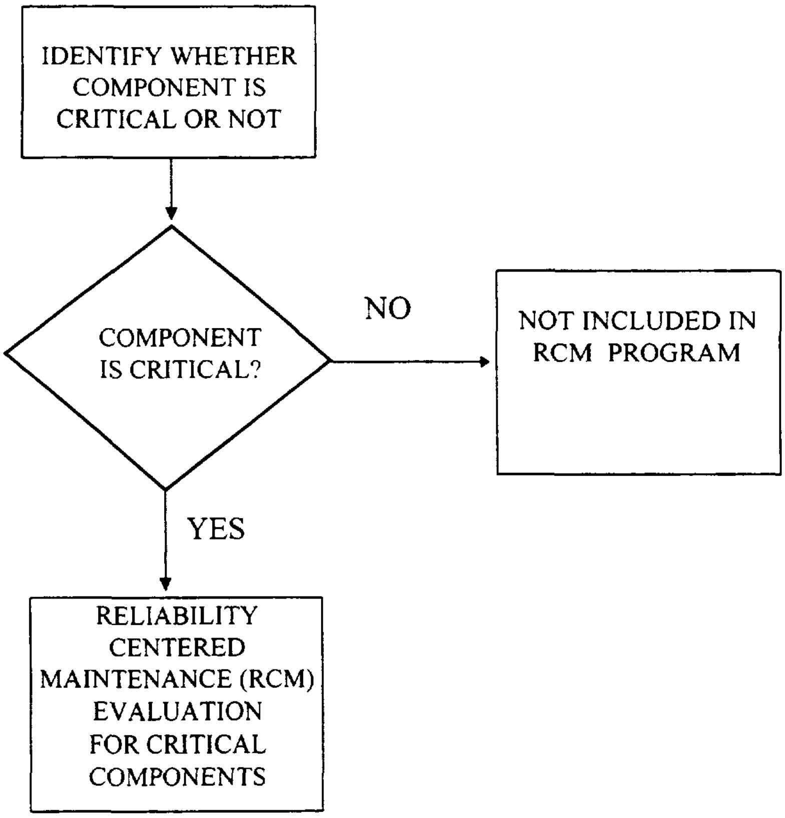

IDENTIFY WHETHER COMPONENT IS CRITICAL OR NOT
COMPONENT IS CRITICAL?
NO
NOT INCLUDED IN RCM PROGRAM
YES
RELIABILITY CENTERED MAINTENANCE (RCM) EVALUATION FOR CRITICAL COMPONENTS

FIGURE 1-1 TOP LEVEL RELIABILITY CENTERED MAINTENANCE (RCM) PROGRAM APPROACH

The process for identifying the risk critical components begins with consideration of the functions that must be performed for safe operation of nuclear power plant. Next step is to identify major systems that provide essential safety functions including mitigation of accidents and the components that enable each such system to perform its safety functions. Then the support systems for the essential system providing the essential safety function and the components that enable these support systems to provide their support functions are identified. RCM program also identify normally operating systems and components whose failures could initiate an accident or transient which challenges safety. For CHASNUPP since level 1 PSA is near to completion, the above described risk critical component identification process based upon the first set of quantification results, has been completed.

After the identification of risk critical components, RCM program determine what maintenance activities are required to ensure reliable operation of the risk critical components identified. The methodologies evaluate failure modes of the risk critical component identified in the first step and identifies maintenance activities required to defend against those failures and then to be incorporated into a RCM program.

Figure 1-2 illustrates the maintenance evaluation process for risk critical component. RCM methodology is further described in section 3.0.

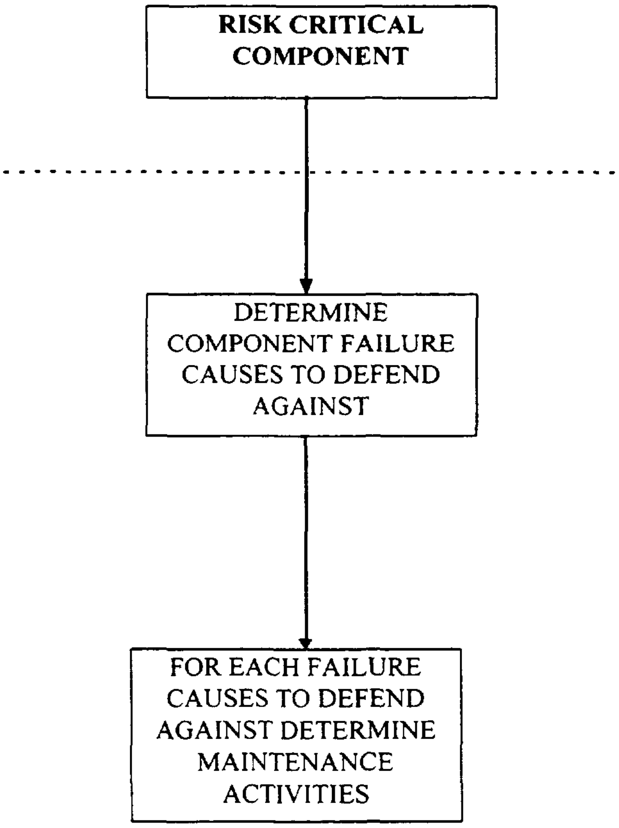

RISK CRITICAL COMPONENT
DETERMINE COMPONENT FAILURE CAUSES TO DEFEND AGAINST
FOR EACH FAILURE CAUSES TO DEFEND AGAINST DETERMINE MAINTENANCE ACTIVITIES

FIGURE 1-2 MAINTENANCE EVALUATION PROCESS FOR RISK CRITICAL COMPONENTS

2. IDENTIFICATION OF RISK-CRITICAL COMPONENTS BASED UPON THE PSA RESULTS

An approach for identifying risk critical component based on using the level 1 PSA result is illustrated in figure 2-1. In order to identify the risk critical component from PSA's accident sequences, the first step is the selection of the core melt frequency that represents the most likely accident scenarios. The next step is to identify the components whose failure modes are represented in this set of accident scenarios. Passive components whose failure would violate the technical specifications success criteria or could result in offsite dose comparable to 10 CFR 100, "Reactor site criteria", would also be designated as risk critical. Determination of risk critical passive components should center on the identification of failure modes that can, or will impact safety. If the failure of a component could initiate an accident or if the component is required to mitigate consequence of any accident, given that it has occurred, it should be considered a risk critical component. Finally any standby component for which aging or common cause failure is a concern, from plant specific experience, should also be added to the list of risk critical component.

In order to identify risk-critical components from accident sequences, only the most likely accident sequences (90% contributors to total CD frequency) are considered. The initiating events associated with those sequences are then identified. Finally, all BOP or other equipment having failure modes that could result in these transients or accidents are identified. The components experiencing the most frequent failure for each of the "dominant" initiating event are kept as ***risk critical components***.

Another approach that can be used to identify the risk critical component is based upon the importance measures or sensitivity analysis results. However for CHASNUPP the approach based upon the core melt frequency results has been used to identify the risk critical components.

From CHASNUPP PSA level 1 results, the initiating events contributing about 90% to the total core melt frequency are selected as a basis for identifying the risk critical components. For CHASNUPP PSA a list of 27 initiating events have been developed comprising of LOCA's, transients and support system initiating events. Out of these 27 initiating events, 10 initiating events appeared to contribute 90% to the total core damage frequency, table 2-1 lists these ten initiating events. Table 2-2 lists the systems whose post initiating events failures dominates the analysis results of core melt frequency for these initiating events. The initiating events listed in table 2-1 are selected for identification of risk critical components. From the analysis result of these initiating events, accident sequences contributing about 90% to the core melt frequency are examined for recognition of risk critical components. Table 2-3 represents the ***risk critical components*** for CHASNUPP identified from the dominant accident sequences. This list of risk critical components is however preliminary at this stage, as this is based upon the first set of quantification results. There may be slight changes in this list after performing the necessary refinement to PSA model.

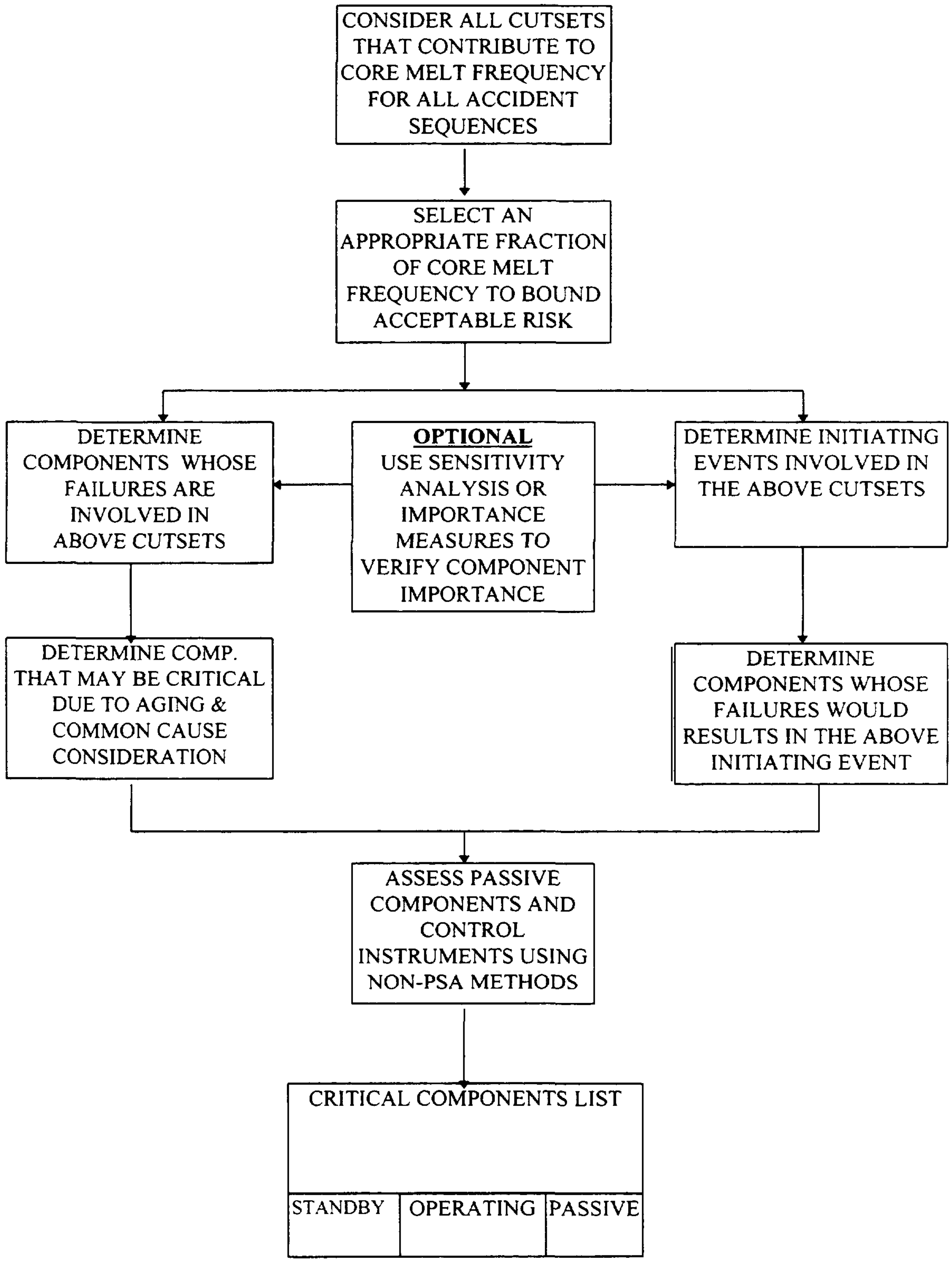

CONSIDER ALL CUTSETS THAT CONTRIBUTE TO CORE MELT FREQUENCY FOR ALL ACCIDENT SEQUENCES
SELECT AN APPROPRIATE FRACTION OF CORE MELT FREQUENCY TO BOUND ACCEPTABLE RISK
DETERMINE COMPONENTS WHOSE FAILURES ARE INVOLVED IN ABOVE CUTSETS
OPTIONAL
USE SENSITIVITY ANALYSIS OR IMPORTANCE MEASURES TO VERIFY COMPONENT IMPORTANCE
DETERMINE INITIATING EVENTS INVOLVED IN THE ABOVE CUTSETS
DETERMINE COMP. THAT MAY BE CRITICAL DUE TO AGING & COMMON CAUSE CONSIDERATION
DETERMINE COMPONENTS WHOSE FAILURES WOULD RESULTS IN THE ABOVE INITIATING EVENT
ASSESS PASSIVE COMPONENTS AND CONTROL INSTRUMENTS USING NON-PSA METHODS
CRITICAL COMPONENTS LIST
STANDBY
OPERATING
PASSIVE

FIGURE 2-1 PSA BASED PROCESS FOR RISK CRITICAL COMPONENTS IDENTIFICATION

TABLE 2-1 INITIATING EVENTS - DOMINANT TO CORE DAMAGE

NO	INITIATING EVENT	NAME
1.	L4	Steam Generator Tube Rupture
2.	T-LOP2	Total Loss of Offsite Power
3.	T-SMF	Loss of Main Feedwater
4.	T-SCW	Loss of Component Cooling Water System
5.	T-CND	Loss of Main Condenser
6.	T-VWE	Loss of Essential Chilled Water System
7.	T-EMS1	Loss of 1E 6 KV EMA Power Supply
8.	T-LOP1	Loss of Offsite Power (220 KV)
9.	T-GT	General Transients
10.	L3	Small LOCA

TABLE 2-2 SYSTEMS - DOMINANT IN CORE DAMAGE ACCIDENT SEQUENCES

NO.	SYSTEM CODE	SYSTEM NAME
1.	SAF	Auxiliary Feedwater System
2.	SRC	Reactor Coolant System
3.	CRP*	Reactor Protection System
4.	SAF & VWE	Auxiliary Feedwater System and Essential Chilled Water System
5.	TG & SAF	Total Grid Loss and Auxiliary Feed Water System

* Note: ATWS are not separately modeled at this stage.

In Table 2-3 the nomenclature used is as follows:

VBC: Pump Room Ventilation System
VER-A: 6 KV IE Electrical Building Ventilation System
SIS: Safety Injection System
SRH: Residual Heat Removal System
CES: ESF Actuation System
VO: Motor Operated Valve
PO: Motor Driven Pump
PD: Diesel Driven Pump
EC/BR: Electrical Breakers
IND.: Independent Failures
CCF: Common Cause Failures

TABLE 2-3 RISK CRITICAL COMPONENTS FOR CHASNUPP

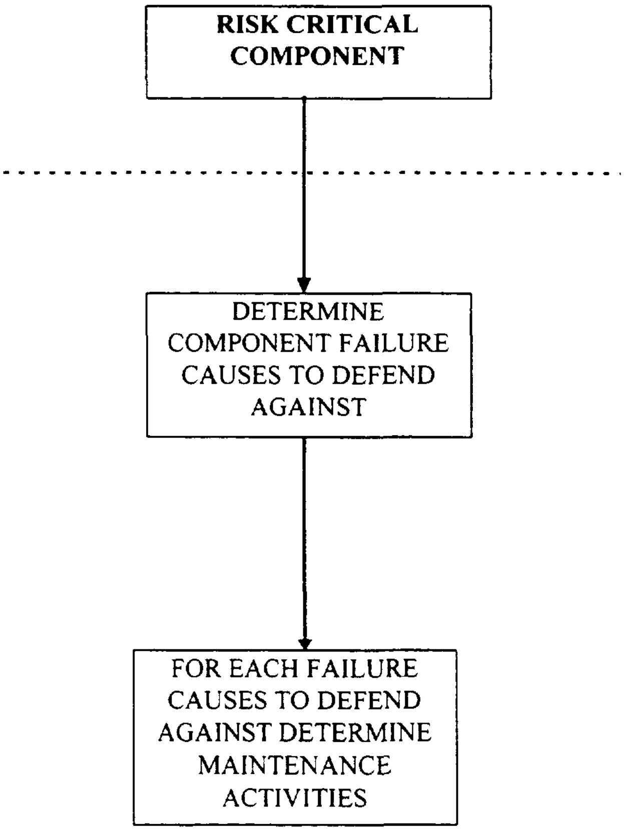
NO.	SYSTEM	COMPONENT	DESCRIPTION	FAILURE MODES	
1	**SAF**	V10A/B/C/D-VO	ISOLATION VALVES	IND	CCF
		V12A/BC/D-VO	RECIRCULATION LINES VALVES	IND	CCF
		V19A/B-VO	DIESEL DRIVEN PUMP COOLING SIDE	IND	CCF
		P01A/01B-PO	MOTOR DRIVEN PUMPS	IND	CCF
		P02A/02B-PD	DIESEL DRIVEN PUMPS	IND	CCF
2	**SRC**	P01A/01B-BR	SRC PUMP BREAKERS	IND	
		TT01A/01B-TT	TEMPERATURE TRANSMITTER	IND	
		V02A/02B-VR	PRESSURIZER RELIEF VALVE	IND	
3	**CRP**	QF-EC	BREAKERS	IND	CCF
		KD-ER	RELAYS	IND	CCF
4	**WES**	P01A/02A-PO	MOTOR DRIVEN PUMPS	IND	CCF
		V07B/08B-VH	PUMP DISCHG LINE CHECK VALVES	IND	
		FT01A-SR	SUCTION STRAINER	IND	
5	**VWE**	101/201-CH	CHILLERS	IND	CCF
		101/201-PO	MOTOR DRIVEN PUMPS	IND	CCF
		V207-VH	CHECK VALVE	IND	
6	**SCW**	P02BCD-PO	MOTOR DRIVEN PUMPS		CCF
		FT09A/B-WF	WATER FILTER	IND	
7	**VBC**	106/206RU-FN	FANS FOR VWE VENTILATION		CCF
		V150/250-VO	SRH PUMPS VENT FAN SUCTION SIDE VALVES		CCF
		V266-VO	WES COOLING LINE VALVE	IND	
		V169/269-VO	SCW PUMPS VENT FANS SUCTION SIDE VALVES		CCF
		V141/241-VO	SIS PUMPS VENT FANS SUCTION SIDE VALVES		CCF
8	**SIS**	P01A/B-PO P02A/B-PO	MOTOR DRIVEN PUMPS		CCF
9	**SRH**	P01A/B-PO	MOTOR DRIVEN PUMP		CCF
		V01A/B-VO	SUCTION SIDE VALVES		CCF
		V01C/D-VO	SUCTION SIDE VALVES		CCF
		V09A/B-VO	HEAT EXCHANGER DISCHG VALVES		CCF
10	**CES**	TRA/B-AR	ACTUATION RELAY	IND	CCF
11	**VER-A**	111FT-WF	WATER FILTER	IND	
		HEPA FTA-AF	HEPA FILTER	IND	
		SDFTA-AF	SAND FILTER	IND	
		HXA-HX	HEAT EXCHANGER	IND	

3. RELIABILITY CENTERED MAINTENANCE PROGRAM METHODOLOGY

This section describes the methodology for developing a reliability centered maintenance program for risk critical components. This methodology is appropriate for establishing a reliability-centered maintenance program for risk critical components identified by the PSA approach described in the preceding sections.

Establishing a reliability centered maintenance program for a risk critical component involves determining the preventive or predictive maintenance actions (e.g. surveillance, condition monitoring, overhaul) or other maintenance related activities such as redesign or reconfiguration, which are responsive to the reliability needs of that component.

Figure 3-1 indicates the two steps that should be addressed by a reliability centered program for a risk-critical component. The first step is to determine the dominant component failure modes that should be defended against. The second step is to determine maintenance activities for these dominant failure modes that will be defend against. Methodologies for completing each step are discussed below.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DETERMINE COMPONENT FAILURE CAUSES TO DEFEND AGAINST
FOR EACH FAILURE CAUSES TO DEFEND AGAINST DETERMINE MAINTENANCE ACTIVITIES

FIGURE 3-1 MAINTENANCE EVALUATION PROCESS FOR RISK CRITICAL COMPONENTS

3.1 IDENTIFICATION OF DOMINANT FAILURE MODES FOR RISK CRITICAL COMPONENTS

Figure 3-2 shows an expanded version of a reliability centered program for identifying the most important component failure modes. Three assessment paths are shown in that figure:

- Identifying the failure modes of the sub-components (elements) of the risk critical components using qualitative. analytical methods
- Identifying failure modes of sub-components (elements) of the risk-critical component from failure history, and
- Identify existing maintenance related activities and requirements.

These three assessment paths are denoted assessment path A. assessment path B. and assessment path C. respectively.

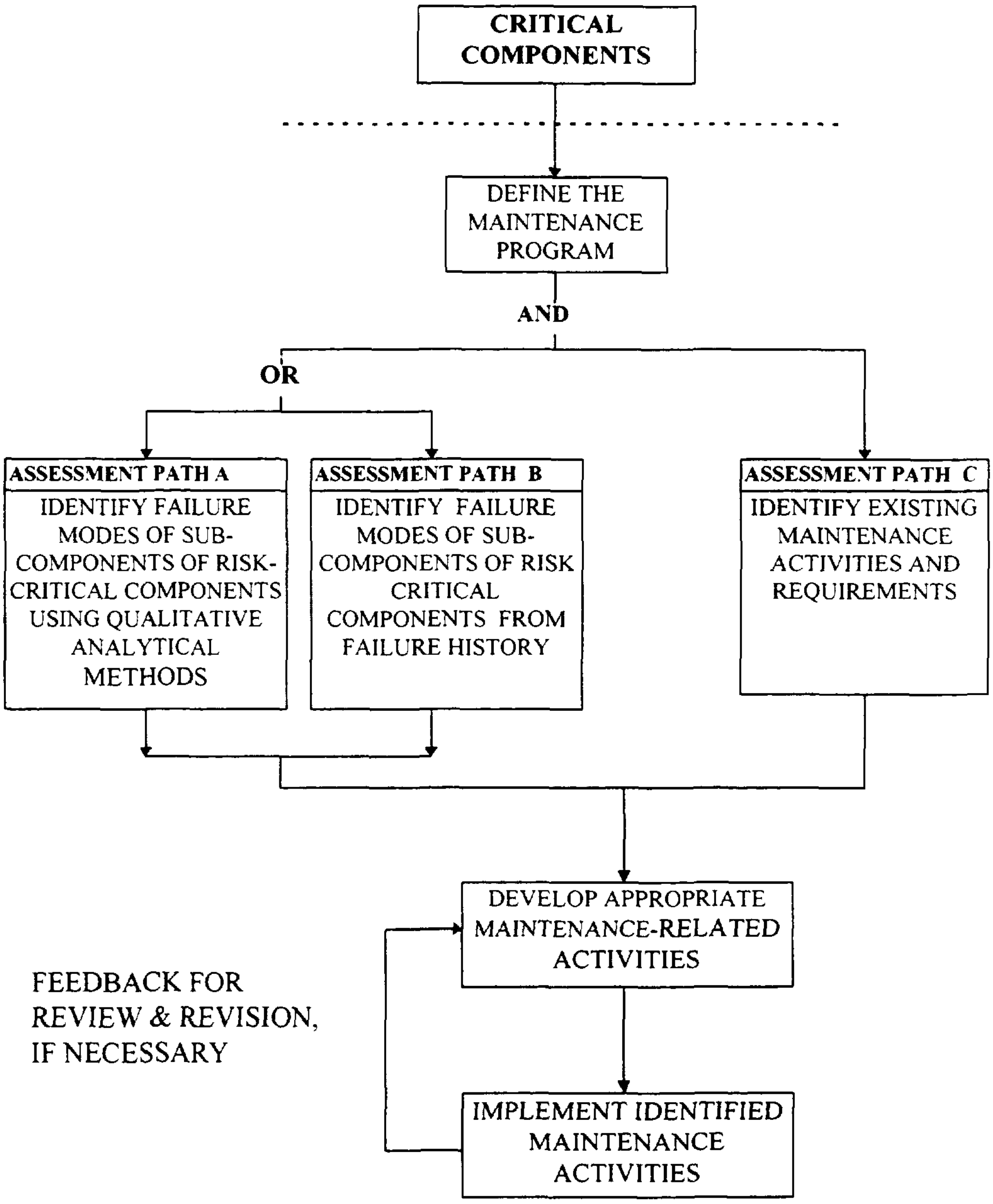
CRITICAL COMPONENTS
DEFINE THE MAINTENANCE PROGRAM
AND
OR
ASSESSMENT PATH A
IDENTIFY FAILURE MODES OF SUB-COMPONENTS OF RISK-CRITICAL COMPONENTS USING QUALITATIVE ANALYTICAL METHODS
ASSESSMENT PATH B
IDENTIFY FAILURE MODES OF SUB-COMPONENTS OF RISK CRITICAL COMPONENTS FROM FAILURE HISTORY
ASSESSMENT PATH C
IDENTIFY EXISTING MAINTENANCE ACTIVITIES AND REQUIREMENTS
DEVELOP APPROPRIATE MAINTENANCE-RELATED ACTIVITIES
FEEDBACK FOR REVIEW & REVISION, IF NECESSARY
IMPLEMENT IDENTIFIED MAINTENANCE ACTIVITIES

FIGURE 3-2 RCM EVALUATION PROCESS FOR OPERATING & STANDBY EQUIPMENT

Assessment Paths A and B are options for identifying the dominant failure modes.

- Assessment Path A would be used for complex components such as diesel generator systems or feedwater systems or where failure history data is not available.
- Assessment Path B would be used for less complex components when failure history data is available.

Both of the above paths should be used to provide substantiating evaluations of failure modes to defend against when this is appropriate. Identifying the dominant failure modes of sub-components is assumed to be synonymous with identifying the risk critical components. For CHASNUPP right now no risk critical component failure history data is available so RCM program proceeds with the assessment path A. However later on when the plant starts operation and the risk critical components failure history data will be available, assessment path B may also be used.

Assessment path C is compulsory, should be done for each risk critical component (after or in parallel with assessment path A or B) and is not to be considered optional.

The activities using qualitative, analytical methods to identify dominant failure modes of the risk critical components are characterized by the left most column of figure 3-2. In this option, a qualitative analytical reliability tool such as fault tree, Failure Modes and Effects Analysis (FMEA), or reliability block diagram will be used to identify elements (sub-components) of risk critical components whose failures are of the types:

- Single element (sub-component) failures that fail the component`s function and that are likely to occur
- Latent element (sub-component) that are not detectable through ordinary component demand testing
- Element (sub-component) failures that, though internally redundant, have common cause potential
- Element (sub-component) failures that have large consequences in terms of repair resources required, or that could cascade to more serious failures. The element failures that will be defended against by preventative maintenance or by other means should be chosen from this set.

A failure history assessment option for determining dominant failure modes of the risk critical components is characterized by the box representing assessment path B of figure 3-2. Though at present this option is not being used for CHASNUPP, a brief description about this assessment process is outlined here.

Since a reasonably long failure history is necessary for most components to determine the dominant failure modes from failure and repair data, it may be useful to combine components into categories that would allow pooling or mixing of the failure histories from several components. One appropriate option would be to combine the failure histories of components of the same type in the same environment, such as large motor operated valves that see borated water environments. Thus, the first step in this option will be to develop the analysis boundary in terms of categories of equipment whose repair and failure data would be pooled.

The next step in this option will be to construct a list of failure modes found in the particular data. This should be accomplished in terms of sub-component failures using, if available, sub-component failure cause data. If sub-component failure cause data is not available, the list should be constructed by major sub-component failures (e.g. "valve driver," valve gate binding", etc.).

The occurrence frequency of each category is then computed and the categories ranked by occurrence frequency, with the most frequently occurring sub-component failures indicated as the prime candidates for inclusion as the dominant failure modes.

3.2 IDENTIFICATION OF EFFECTIVE MAINTENANCE ACTIVITIES FOR RISK CRITICAL COMPONENTS

The steps to assess existing maintenance requirements and recommendations for each risk critical components are characterized by the box representing assessment path C in Figure 3-2. This assessment will be conducted after, or in parallel with, the assessment in path A or B; it is not considered an option.

In overview, the proposed assessment process will be to collect and review all maintenance requirements and recommendations for the component from all relevant sources, and then divide these into maintenance actions that are part of the existing maintenance plan for the component and those that are not being performed.

Reasoning will be defined for both sets of maintenance actions. That is, a basis will be developed for each maintenance action that is included in the existing maintenance plan, and a basis will be developed to explain why each recommended performance is in the 'not performed' category. This explicit set of steps will serve as a starting point for the assessment of maintenance needs for the component.

The dominant failure modes which should be defended against and for which maintenance strategies should be devised will be those identified in assessment path C, plus those identified using a reliability assessment similar to assessment paths A and/or B.

The process of determining effective maintenance to defend against the dominant failure modes of a component is mainly based upon the engineering judgment. However, there are some options based upon the information about sub-component's failure mode, its impacts, occurrence frequency and failure type. Such information tables can aid systematic completion of the task of effective maintenance determination. Table 3-1 represents one configuration that may assist the process of determining effective maintenance. All dominant failure modes for a single risk critical component are listed in the left most column of the matrix, usually as individual sub-component failures. Succeeding columns, from left to right, list:

- Consequences of these sub-component failures in terms of resources for repair, impacts on risk, impacts on technical specifications (if any), potential for cascading or common cause failure, etc.

TABLE 3-1 CRITICAL FAILURE MODE DETECTION MATRIX (RCM MATRIX)

CRITICAL FAILURE MODE	FAILURE MODE IMPACTS			OCCURRENCE FREQUENCY	INSTRUMENTATION	LATENT OR ANNOUNCED	POTENTIAL DETECTION OR DEFENSE	CRITICAL FAILURE MODES
	REPAIR OUTAGE TIME	IMPACT ON RISK	IMPACT ON TECHNICAL SPECIFICATIONS					

- The estimated occurrences frequency for each sub-component failure, estimated either from historical failure data, or as a category such as high, medium, or low.

- Instrumentation, if any, that would provide an indication that the sub-component has failed or is likely to fail.

- Whether the sub-component failure is latent or announced.

- Potential maintenance defenses such as preventive or predictive maintenance, surveillance, etc. that could be used to detect the sub-component failure or a precursor to sub-component failure or prevent failure.

The last column represents a final assessment as to whether or not the failure mode will be defended against.

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BALANCING PREVENTIVE AND CORRECTIVE MAINTENANCE IN CERNAVODA UNIT 1 NPP

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Abstract

The paper presents a short reminder of Romania's Cernavoda NPP entering commercial operation and a brief description of the CANDU-6 project on which Unit 1 is based. The short term objectives of the maintenance management, the status of the existing maintenance programmes as well as future predictable maintenance programmes are outlined together with the Government plan to complete the balance of NPP.

0. INTRODUCTION

A new chapter in the Romanian nuclear power sector began last December, when Cernavoda Unit 1, the first CANDU in Europe, entered commercial service. In the first nine months of operation the plant showed remarkable performance (90% availability factor for a net capacity of 635 MWe, compared to 78% target). Owned by the Romanian state utility RENEL, the first full year of NPP operation will give,. in 1997, an anticipated output of 4.840 TWh. This is equivalent to 1.25 million tonnes of imported petrol, thus cutting Romania's oil import bill by US $ 160 million.

1. ROMANIA'S LARGEST SINGLE POWER UNIT, CERNAVODA 1, THE ONLY NUCLEAR FACILITY IN EASTERN EUROPE EFFECTIVELY BASED ON WESTERN TECHNOLOGY AND INTERNATIONALLY RECOGNISED SAFETY CRITERIA

Cernavoda Unit 1 provides a reliable, cost - effective and clean source of electricity to support the Romanian economy, delivering about 9% of the country's average annual requirement to the national grid. The successful start-up milestones of the first from five identical 700 Mwe CANDU units originally planned for Cernavoda NPP (located on the Danube, about 160 km east of Bucharest) included 16.04.1996 (first criticality), 11.07.1996 (first connection to the grid), 02.10.1996 (first full power operation), 02.12.1996 (commercial operation) and 31.07.1997 (AECL/ANSALDO Consortium handing over management responsibility to Romanian staff).

The option of a Western technology for our first NPP was mainly based on the following reasons:

- the great attention paid to safety matters (i.e. containment, redundant reactor control, seismic design, environmental qualification a.s.o.)
- the use of natural uranium as fuel and heavy water as coolant and moderator, which were possible to be manufactured in Romanian facilities.

- the chosen technology was a well-proven one by the large experience gained in construction and operation of other CANDU stations around the world.
- the process equipment for CANDU NPP did not required as large an investment in sophisticated manufacturing plants as that for other types of nuclear stations. thus allowing utilization of Romanian manufactured equipment.

2. CANDU - 6 PROJECT BRIEF DESCRIPTION

CANDU is an abbreviation for CANadian Deuterium Uranium, and means a pressurised heavy-water moderated and cooled reactor (PHWR) design. The Cernavoda CANDU-6 project is based on:

a) the reference plant Point Lepreau Canadian design for NSP
b) the Italian ANSALDO design for BOP
c) the American GENERAL ELECTRIC design for the turbine - generator
d) the Romanian design fbr some specific systems and general activities coordination (except 1991 to mid. 1997 when AAC took over coordination)

The reactor "burns" unenriched uranium (found in Romania) thus requiring a highly efficient neutron economy (which demands the use of heavy water D_20 as neutrons moderator) and also on-line refuelling, with frequent replacement of small amounts of spent fuel with fresh ones. The latter is done remotely and automatically by two refuelling machines, without having to shut down the reactor - an important element in 0 & M cost reduction.

This on-line refuelling dictates another PHWR specific design feature: horizontal arrangement of fuel assemblies into the pressure tubes, centered within calandria tubes. All fuel in CANDU-6 cores is contained inside the calandria - a cylindrical low-pressure tank filled with the heavy-water moderator at low temperature and near-atmospheric pressure. The calandria is positioned in a steel-lined concrete vault filled with light water. which provides external radiation shielding and emergency cooling. Heat developed from fission within fuel channels (2,064 MWt) is removed by PHT pumps circulating D_20 coolant through two separate loops to four steam generators (SG). each loop servicing half of the fuel channels. Demineralized light water circulated through the SG secondary side is the steam source.

The BOP regenerative cycle is characterised by a high thermal efficiency. enhanced thermal recovery of feedwater and condensate extraction systems and an improved main equipment design.

3. MAINTENANCE MANAGEMENT SHORT-TERM OBJECTIVES

The main objective of Maintenance management is to ensure. that all systems, structures and components are in adequate state to fulfil their functions, i.e.:

- provide equipment operability and power maneuverability
- achieve greater economy in power production reducing overall 0 & M costs.
- reduce occupational radiation exposure (ALARA)
- maintain or enhance safety and reliability levels for safety systems as well as for production systems.

Actual Maintenance Department organisation includes around 350 personnel divided into four sections: Mechanical; Electrical, Instrumentation & Control; General Services; and Maintenance Engineering Support. Main specific activities are shown in appendix 1.

Till present the man-hours distribution of maintenance activities (outage, backlog, completion of SDIDCN (station dispositions/design change notices)) was 60 % corrective maintenance (a high percentage specific to NPP's first year of operation and enhanced by the non-existence of external specialised services), 30% preventive maintenance including inspections, and 10% modifications implementation. Maintenance's short-term objective (appendix 2) is to reduce the unplanned (corrective) maintenance share, while improving the preventive maintenance one. The long-term target is to move towards modern developments in safety-related maintenance increasing the predictive maintenance share.

4. DEVELOPMENT AND IMPLEMENTATION OF EXISTING MAINTENANCE PROGRAMS

Corrective maintenance program. The goal is to lower the increased “breakdown maintenance" percentage away from 60% to a more convenient value of 10-20% from total maintenance man-hours while strengthening work quality and introducing new cost-effective techniques (as FURMANITE on-line under-pressure leak sealing, used to prevent unit shutdown).

Preventive maintenance program. Completion of call-ups (about 70% are produced), revision of those already implemented, data input and specific database updating are current activities. Equipment selection and scheduled preventive maintenance frequency adjusting are also considered. Identification of all U1-needed oil and grease types and of their original equipment manufacturer's approved equivalents are main steps of lubricants program.

Master Equipment List program. Main target is to complete identification of all U1-installed equipment and it's main technical and QA characteristics, while loading infos into MEL database.

Spares systematic review program. Main objectives are Ul-needed consumables and common stock items catalogues preparation, as well as equipment spare parts technical identification and stock assessment. First two NPP operation years critical spares were already identified and procured.

Equipment maintenance history program. The aim is to record all applicable data for maintenance and related test activities into a special dedicated database.

Obsolete items program. This program's intention is to collect in due time all necessary data in order to identify a substitute for replacement once there are indications that a particular type of equipment will no longer be made, and to be prepared for when obsolete equipment has to be replaced.

Maintenance tools program. Tools inventory review and tools replacement plan are main objectives.

Personnel training program. Up to now, most training has been developed by the "on the job" training process. While part of employees have gone through systems-knowledge oriented training, and safety culture in maintenance, for some specific areas (like metrology, welding, pressure boundary work a.s.o.) personnel were specifically trained, qualified and authorized. JRTR's job related training requirements) have been prepared. Special facilities for skill-oriented training are to be ready as soon as possible. Design and construction of a unique maintenance training center building are also being considered.

Other on-going main activities include: maintenance documentation updating and specific library organizing; maintenance databases designing and their update; maintenance-related performance indicator system implementation; improving coordination and interface with other maintenance-related groups inside the NPP, as well as with external organizations (Regulatory Body a.s.o). Identification of critical equipment, taking into consideration the manufacturers' recommendations or operation point of view (plant specific experience feedback) and analysis of external experience are only a few of the difficulties to keep maintenance programs alive or establish management decision criteria.

5. PREDICTIVE MAINTENANCE

Efforts are made to purchase and implement a new integrated licenced program, transforming the existing U1 maintenance programs through RCM philosophy into a complete analysis of equipment health thus allowing to predict failures, identify root-cause of problems and affording to accurately schedule the maintenance downtime. Main directions of this program are vibration analysis, infrared thermography, motor diagnostics, alignment, balancing and continuous online equipment condition monitoring.

Another last-generation licenced program to be purchased is dedicated to the on-line valves testing and accurately measuring of different characteristics.

6. CONCLUSIONS

While developing a more proactive maintenance policy and monitoring the overall maintenance process, Cernavoda must reevaluate the tuning/design philosophy of all existing/future maintenance programs, expecting appropriate 0 & M tasks being carried out at cost-effective time intervals, and a long-term improvement in plant performance.

Unit 1 has decided to have a long-term relationship with other nuclear utilities and to receive support from international organizations (IAEA, COG, WANO, INPO) to ensure we keep-update with new maintenance developments, as well as to benefit from experience gained by other power plants. This is vital for our facility in today's conditions of very tight budgets and a very competitive market-place.

The Romanian Government has recently declared its nuclear energy program a national priority for the country's economy over the next five years. This program foresees:

a) completion of NPP Unit 2 till 2001 (construction is 26% complete, with about 60% of equipment and materials on site, and estimated cost of completion raising to $750 milion);
b) commissioning of Unit 3 before 2005
c) commissioning of Units 4 and 5 after 2005.

With all future-needed nuclear fuel and heavy-water inventory being produced exclusively in Romania, Cernavoda NPP is expected to have an important role in assuring a safe electricity supply not only in Romania but even to neighbouring areas by replacing old and obsolete fossil and non-Western designed nuclear power plants. Now Cernavoda is a key step for the socio-economic transition to a new future in Romania and Eastern Europe.

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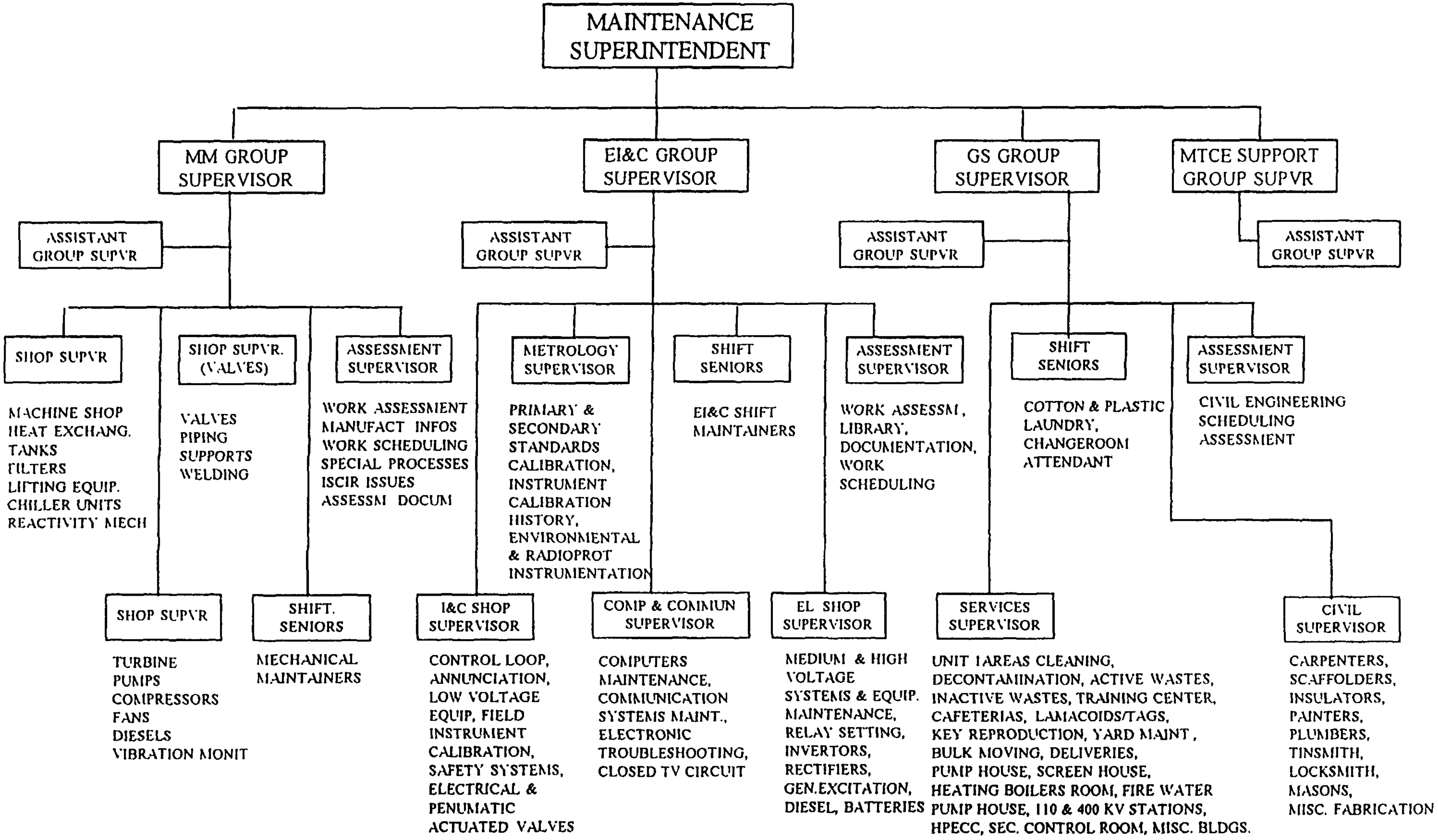
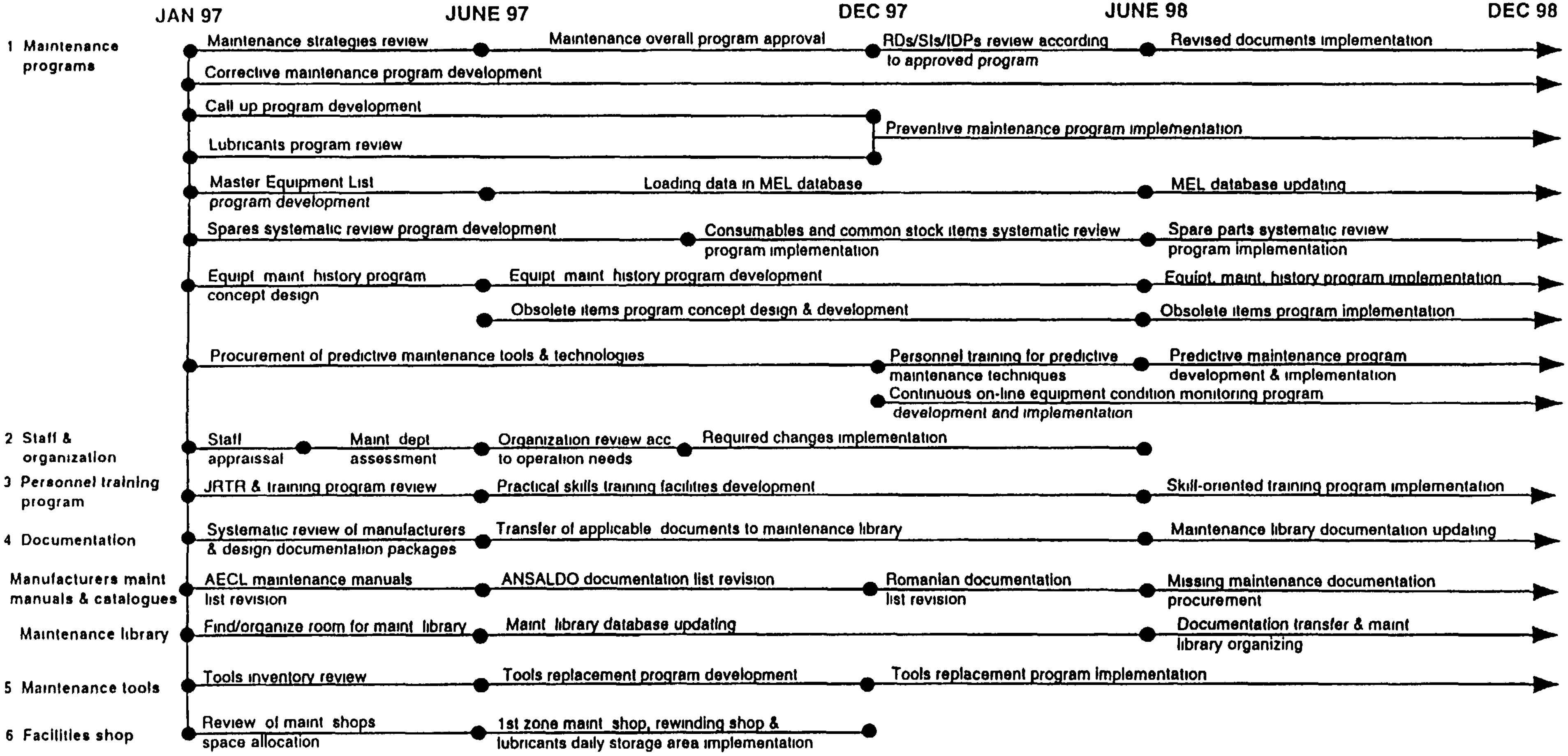

MAINTENANCE SUPERINTENDENT
MM GROUP SUPERVISOR
EI&C GROUP SUPERVISOR
GS GROUP SUPERVISOR
MTCE SUPPORT GROUP SUPVR
ASSISTANT GROUP SUPVR
ASSISTANT GROUP SUPVR
ASSISTANT GROUP SUPVR
ASSISTANT GROUP SUPVR
SHOP SUPVR
MACHINE SHOP HEAT EXCHANG. TANKS FILTERS LIFTING EQUIP. CHILLER UNITS REACTIVITY MECH
SHOP SUPVR. (VALVES)
VALVES PIPING SUPPORTS WELDING
ASSESSMENT SUPERVISOR
WORK ASSESSMENT MANUFACT INFOS WORK SCHEDULING SPECIAL PROCESSES ISCIR ISSUES ASSESSM DOCUM
SHOP SUPVR
TURBINE PUMPS COMPRESSORS FANS DIESELS VIBRATION MONIT
SHIFT. SENIORS
MECHANICAL MAINTAINERS
METROLOGY SUPERVISOR
PRIMARY & SECONDARY STANDARDS CALIBRATION, INSTRUMENT CALIBRATION HISTORY, ENVIRONMENTAL & RADIOPROT INSTRUMENTATION
SHIFT SENIORS
EI&C SHIFT MAINTAINERS
ASSESSMENT SUPERVISOR
WORK ASSESSM, LIBRARY, DOCUMENTATION, WORK SCHEDULING
I&C SHOP SUPERVISOR
CONTROL LOOP, ANNUNCIATION, LOW VOLTAGE EQUIP, FIELD INSTRUMENT CALIBRATION, SAFETY SYSTEMS, ELECTRICAL & PENUMATIC ACTUATED VALVES
COMP & COMMUN SUPERVISOR
COMPUTERS MAINTENANCE, COMMUNICATION SYSTEMS MAINT., ELECTRONIC TROUBLESHOOTING, CLOSED TV CIRCUIT
EL SHOP SUPERVISOR
MEDIUM & HIGH VOLTAGE SYSTEMS & EQUIP. MAINTENANCE, RELAY SETTING, INVERTORS, RECTIFIERS, GEN.EXCITATION, DIESEL, BATTERIES
SHIFT SENIORS
COTTON & PLASTIC LAUNDRY, CHANGEROOM ATTENDANT
ASSESSMENT SUPERVISOR
CIVIL ENGINEERING SCHEDULING ASSESSMENT
SERVICES SUPERVISOR
UNIT 1AREAS CLEANING, DECONTAMINATION, ACTIVE WASTES, INACTIVE WASTES, TRAINING CENTER, CAFETERIAS, LAMACOIDS/TAGS, KEY REPRODUCTION, YARD MAINT, BULK MOVING, DELIVERIES, PUMP HOUSE, SCREEN HOUSE, HEATING BOILERS ROOM, FIRE WATER PUMP HOUSE, 110 & 400 KV STATIONS, HPECC, SEC. CONTROL ROOM, MISC. BLDGS.
CIVIL SUPERVISOR
CARPENTERS, SCAFFOLDERS, INSULATORS, PAINTERS, PLUMBERS, TINSMITH, LOCKSMITH, MASONS, MISC. FABRICATION

Appendix 1 - **Cernavoda NPP Maintenance Department Organization**

Appendix 2 ***Maintenance management short-term objectives***



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1 Maintenance programs

- Maintenance strategies review — Maintenance overall program approval — RDs/SIs/IDPs review according to approved program — Revised documents implementation
- Corrective maintenance program development
- Call up program development; Lubricants program review — Preventive maintenance program implementation
- Master Equipment List program development — Loading data in MEL database — MEL database updating
- Spares systematic review program development — Consumables and common stock items systematic review program implementation — Spare parts systematic review program implementation
- Equipt maint history program concept design — Equipt maint history program development — Equipt. maint. history program implementation
- Obsolete items program concept design & development — Obsolete items program implementation
- Procurement of predictive maintenance tools & technologies — Personnel training for predictive maintenance techniques — Predictive maintenance program development & implementation
- Continuous on-line equipment condition monitoring program development and implementation

2 Staff & organization

- Staff appraissal — Maint dept assessment — Organization review acc to operation needs — Required changes implementation

3 Personnel training program

- JRTR & training program review — Practical skills training facilities development — Skill-oriented training program implementation

4 Documentation

- Systematic review of manufacturers & design documentation packages — Transfer of applicable documents to maintenance library — Maintenance library documentation updating

Manufacturers maint manuals & catalogues

- AECL maintenance manuals list revision — ANSALDO documentation list revision — Romanian documentation list revision — Missing maintenance documentation procurement

Maintenance library

- Find/organize room for maint library — Maint library database updating — Documentation transfer & maint library organizing

5 Maintenance tools

- Tools inventory review — Tools replacement program development — Tools replacement program implementation

6 Facilities shop

- Review of maint shops space allocation — 1st zone maint shop, rewinding shop & lubricants daily storage area implementation

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RISK BASED DEFINITION OF TS REQUIREMENTS FOR NPPS WITH VVER-1000 TYPE REACTOR

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Abstract

The main regulations in safety related maintenance for NPPs in Russia are defined as a part of Technical Specifications (TSs). It includes limiting conditions for operation (surveillance requirements, allowed outage time, et.). In Russian practice the two levels of TSs are presented: general TSs that have been established as a master documents for similar designed NPPs and plant specific based on operation practice of each NPP unit.

This paper presents a brief review of submissions to TS changes for NPPs with VVER type reactor were issued by AEP PSA team since 1988 year. Besides it provides an approach allows to estimate the complex affect on plant risk for both Limiting Conditions of Operation (LCO) and Surveillance Test Intervals (STI) based on relevant probabilistic tool (Minimal Cut Sets method and Marcov Chains methods).

1. INTRODUCTION

Since 1988 AEP has performed a number of probabilistic risk studies for different VVER type NPPs. The results of these studies in addition to the purpose of design and operation improvement were used also as a base for definition of limiting conditions for both general and plant specific TSs.

Quantification of the risk probabilities associated with loss of critical safety functions for different test interval and allowed outage time values provided the base for choice of limiting conditions in general TSs for VVER-1000/320 (this document was developed by AEP, VNIIAES, ODB Hydropress and Kurchatov institute).

Mentioned risk calculations used conservative generic reliability data and model assumptions to obtain conservatism in results which is important for such type of documents.

The best estimate results could be obtained by performing more detail studies for specific NPP units. For this studies the risk measure usually associates with core damage frequency. It means that plant specific PSA models and data base should be used. AEP performed this type of studies for Kola Unit 3,4 and Kalinin Unit 1 and 2 NPPs. The periodicity of component testing and AOT as well as repair strategies were under consideration. For decision making regarding TS optimization an acceptance of risk increase over 10% of nominal level for the TS changes was assumed. However regarding to Kalinin NPP, the risk level was demonstrated even to improve. Such results was achieved by extending of AOT in exchange for reducing a number of long-term surveillance test intervals as well as implementing staggered testing strategy.

It should be underlined that above mentioned studies based on PSA approach and used corresponding computer codes. An experience shoes, however, that for complete optimization, including changes in surveillance requirements, AOT and repair strategies together, existing PSA codes (RISK SPECTRUM, IRRAS, PSA PACK, et.) can not provide an adequate model response to all possible variables, as they suggest a limited number of fixed fault tree and component reliability models that do not have enough flexibility to consider the actual operation history.

Taking this into account an original approach was developed to solve the task in a complex form. According to it, quantification procedure includes calculation of a set of two concurrent characteristics: core damage frequency and frequency of unplanned unit shutdowns as well as comparison of values obtained for different alternatives. For this purpose a method that summarize the advantages of Minimal Cut Set methodology and Marcov Chains can be used.

The paper in addition to description of mentioned studies and results that have been performed using traditional PSA tools also presents a basis of methodology seems to be able to provide complex optimization process for TS decision making.

2. KALININ NPP SPECIFIC STUDY

In 1988 Kalinin NPP requested a study to be performed by Atomenergoproekt institute to resolve safety system testing and AOT issue. The problem was that it was required that safety systems had to fulfill the single criterion during maintenance action as well. If not, a plant was forced to shutdown due to a Technical Specifications requirement which were based on deterministic analysis and engineering judgment.

The only method to eliminate latent failures was to run available trains of safety systems during the whole repair of failed component. In this case failures of available components were supposed to be directly revealed by instrumentation or process symptoms. However, such procedure led to overheating of emergency cooling water caused by a long-term pump operation in the recirculation mode as well as useless losses of diesel fuel.

Another restriction of TSs was to limit AOT by 24 hours. It was not suitable for operators because the component restoration times should include detection plus waiting times as well as post-repair test time. It should be noted that delay time during which repair is unlikely to be performed because of the time required for detection and repair initiation may be considerable. As a matter of fact, repair initiation time can include administrative time, component cooldown time, decontamination time, and time waiting for tools and spare parts needed for repair.

Thus, to meet TSs requirements, occurrence of frequent unscheduled reactor trips, followed by cooling down, would be evident that could give itself an additional contribution into the core damage frequency.

It was decided that new requirements would be justified and provided to the NPP which allowed to the operator more flexibility and which removed the pessimism from the previous requirements.

Impact of different testing strategies and AOT values on core damage frequency was studied using VNF computer code based on event tree / fault tree linking method. That code makes it possible to take into account the time dependent effects such as staggered testing scheme, repair time distribution censored by AOT value, etc.. The study was limited by full power reactor operation mode and internal initiating events.

Data collated from NPPs of so-called "small series" (Novovoronezh unit 5, South Ukraine NPP and Kalinin NPP) were used to derive input reliability values such as failure rates and mean times to repair. Initiating event frequencies used were generic.

With regard to allowable outage time for repair of safety system component failed in reactor power operation mode, additional time duration was taken into account. This time window was necessary to bring NPP into safety state given unsuccessful repair of failed component. Such time duration was estimated to be ten hours. Thus, to assess impact of allowable outage time on the core damage frequency, plus ten hours should be also taken into consideration.

Risk level in terms of core damage frequency was demonstrated to improve from 1.6E-3 per year to 6.8E-4 per year in case of implementing technical specification modification recommended. Such result was achieved by extending of AOT in exchange for reducing a number of long-term surveillance test intervals, implementing surveillance tests of untested motor- and air-operated valves as well as the fact that staggered tests over redundant trains were arranged for availability benefit.

The calculation results demonstrate that application of staggered testing strategy with extraordinary tests may reduce unavailability considerably (1.5-6 times). On the other hand, extending of the allowable outage time of a safety system train accepted at Kalinin NPP was not so important from the safety point of view.

PSA results were used to reissue Technical Specifications. At present, there is the following requirement to safety system tests & maintenance at Kalinin NPP:

- each safety system train must be tested once a month. The trains are tested at staggered intervals, once every ten days, and, if there is a failure, the rest of trains are to be tested in 4 hours;
- allowable outage time of failed train may be 72 hours including the above-mentioned 4 hours.

3. KOLA UNIT 3 AND 4 SPECIFIC STUDY

In the early 90's, reliability analysis of safety systems for Kola unit 3 and 4 was performed to validate STIs and AOTs. The impact of STIs and AOTs on safety function performance was estimated. Those safety functions were to:

- maintain the reactor subcriticality
- maintain primary reactor coolant inventory
- remove residual heat via the secondary circuit at high and low pressure in the primary circuit
- remove heat from containment
- scrub radioactivity from containment atmosphere

Kola plant specific reliability data was used for the study. The component reliability data base for all mechanical and electrical components in safety systems at Kola Units 3 and 4 covered 6 reactor-years of operational experience. A total of 613 components such as pumps, motor-operated valves, check valves, safety valves, relief valves, air-operated valves, control valves, fans, diesel generators, invertors, rectifiers, circuit breakers were under consideration. Over 230 events were collated from early 1986 through 1988.

Both front-line and support systems which should perform the above-mentioned safety functions were analyzed. Study was performed using APRA computer code package developed by Atomenergoproekt. APRA uses success path diagram linked with fault tree models, which makes it possible to perform modularization followed by intermediate screening. VNF computer code is a part of APRA. APRA makes use of minimal cut set (MCS) method for Boolean reduction.

For decision making regarding TS optimization, risk values in terms of probabilities of unfulfillment of safety functions were derived. An acceptance of risk increase over 10% of nominal level for the TS changes was assumed.

It was concluded that the increase of AOT from 24 to 72 hours would not effect significantly on the probability of safety function fulfillment, given an extraordinary test of the other two trains would be carried out and their availability would be confirmed. According to Technical Specifications implemented in Kola NPP based on the reliability study, a functional test of safety system trains is to be staggered among the three trains. The procedure calls for testing all redundant components in case of any failure discovered.

4. RECENT ACTIVITY

Mentioned above plant specific studies were performed using a traditional tool for Risk measure evaluation - fault tree/event tree computer codes. For Kalinin and Kola studies VNF and APRA codes, derived by AEP were applied. Currently a new work, aimed to justify the extension of STIs for unit 4 Balakovo NPP is under way. Within this task a modern software for risk calculation - Risk Spectrum PSA computer code is used.

Regarding to evaluation of maintenance input to risk these codes as well as the others have the following peculiarities:

- component unavailability models (average or time depended) are represented by a number of formulas based on base-line unavailability function, assuming no detected failures and maintenance during an operation;
- unavailabilities due to maintenance to be introduced to the fault trees by hand and normally placed at the train top event level;
- to define unavailability due to single maintenance a user should calculate both the probability to entry the corrective maintenance over the test interval and mean down time of the train given AOT.

Described above calculations become complicated when multiply maintenance events are also considered (if they are not prohibited by LCO), as well as when staggering effect and extra tests should be taken into account. So, typical PSA computer codes are not convenient tool for the complex risk analysis associated with maintenance. That is why an approach based on importance measure (uses only base line fault tree model) is used for the purposes of finding the individual optimal or acceptable TS parameters. This, of course, can provide the AOTs and STI values that can be consider as candidates for optimum ones. However, the justification of the choice an a complex form is still a task to be solved.

Doing this, a special method that allows to estimate unplanned maintenance contribution to plant risk and availability have been recently derived. To insert a flexibility to the model in accounting of LCO alternatives (staggering effect, different strategies of testing and repair) as well as considering unplanned outages caused by AOT limitations an approach based on Marcov Chain theory have been used. This approach also uses PSA results as an input data.

The operation NPP Circle between two refueling outages can be represented by the state graph, that defines the possible plant configurations and directions of their transfers during operation.

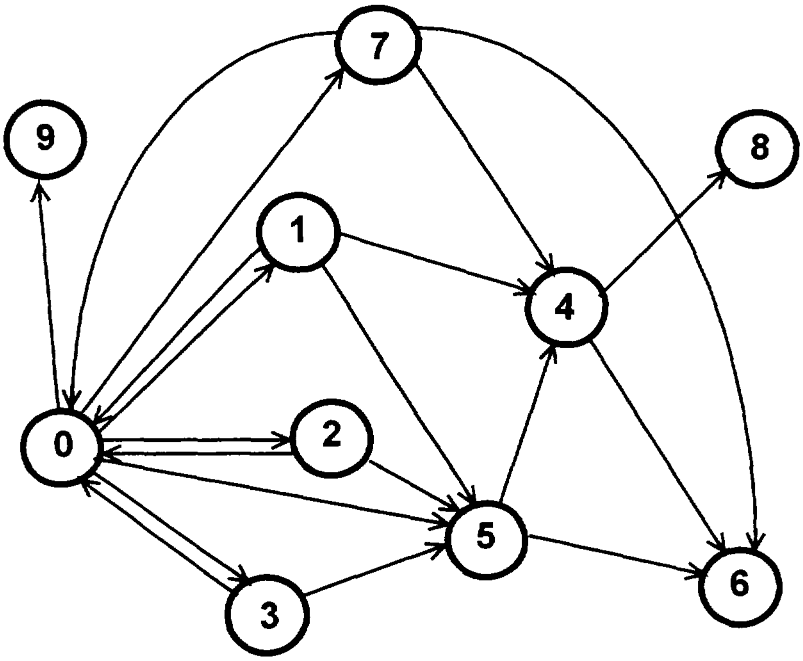

Fig.1. Typical NPP state graph

An illustrative example of shush state graph is given on Fig. 1, where:

0 - normal operation

1,2,3 - unplanned corrective maintenance (restoration of single failures in safety system)

4 - plant shutdown and coolingdown under accident conditions

5 - unplanned shutdown under normal conditions

6 - unplanned outage

7 - planned maintenance

8 - plant damage state (core damage, etc.)

9 - refueling outage.

Unplanned maintenance (state #1,2,3 on the graph) refers to corrective maintenance required to restore equipment to service following a critical failure that makes it unavailable. Planned maintenance period (state #7 on the graph) is used to conduct both preventive maintenance and minor corrective maintenance items on noncritical faults that can be deferred. Both are constrained by the AOTs in the Technical Specifications.

Consider the consequence of different plant configurations during the operation: E_i, i=1,2....,N_c at the moments of their changes θ_k (k=1,...2). Then for Markov Chain $\{E^k=E(\theta_k)$, $\theta_k\}$ the following equation, define state probability distributions can be written:

$$P\left\{E_j,\theta_{k+1},d\theta\right\}=\int_0^{T_r}\left[\sum_{i=1}^{N_c}P\left\{E_i,\theta_k,d\theta\right\}\cdot P\left\{E_j,\theta_{k+1},d\theta\ /E_i,\theta_k\right\}\right],$$

where Tr - time period between the two refueling outages.

In order to define transfer conditional probability density functions P(E_j, θ_{k+1},dθ/ E_j, θ_k) the state space of the random process being considered should be extended by insertion of specific events A_l (l=1....M), that correspond to conditions define the state transfers according to LCO. These events reflect the results of component testing. Pairs of (E_j, A_l) j=1,2,....N; l=1,2,...M make it possible to consider the chain (E_j, A_l, θ_k) as a random process of Markov type.

For purpose of quantification of such events a method which is similar to Minimal Cut Set approach is applied. So, each event A_l is represented by the sum of cut sets (not always minimal). The list of such cut sets is derived according to a special procedure. It uses minimal cut set data base that coming from PSA. Then the number of cut sets is extended in order to harmonize cut set terms which contribute to different A_l and to CD at the same time. The purpose of this is to provide possibility of the calculation of conditional probability for A_l or any CD MCS given A_m just before θ_k. Therefore knowing P(E_j,A_l,θ_{k-1},dθ/E_j,A_m,θ_k) it is possible to obtain distribution functions for each configuration.

Hence, the method described above seems to be able to resolve all the problems related to complex TS optimization. It keeps advantages of both Marcov Chain and MCS approaches. The completion of method elaboration is assumed during the next year.

Conclusion

The problem related to TS definition has been under consideration by AEP PSA team during several years. The plant specific study was performed for two type of VVER reactor VVER-1000 (Kalinin NPP, unit 1,2) and VVER-440 (Kola NPP, Unit 3,4). The main purpose of studies was to extent AOT and justify monthly STIs. This was done on a base of risk calculation for different TS alternatives including staggered testing effects, extra tests etc. The risk level for Kalinin NPP was demonstrated to improve from $1.6*10^{-3}$ per year to $6.8*10^{-4}$ per year after implementation of recommended TSs.

For Kola NPP the increase of AOT from 24 to 72 hours would not effect significantly on the probability of safety function unfulfillment. All recommended TS changes were implemented and used in operation practice.

For the purpose of a complex TS optimization taking into account risk due to unplanned maintenance as well as plant availability and risk due to unplanned shutdown caused by LCO, a special approach that summarizes the advantages of Marcov Chain and MCS methodology have been developed. This work suppose to be completed during the 1998 year.

ON TEST AND MAINTENANCE — OPTIMIZATION OF ALLOWED OUTAGE TIME

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Abstract

Probabilistic Safety Assessment is widely becoming standard method for assessing, maintaining, assuring and improving the nuclear power plant safety. To achieve one of its many potential benefits, the optimization of allowed outage time specified in technical specifications is investigated.

Proposed is the risk comparison approach for evaluation of allowed outage time. The risk of shutting the plant down due to failure of certain equipment is compared to the risk of continued plant operation with the specified equipment down. The core damage frequency serves as a risk measure.

1. Introduction

Test and maintenance plays an important role in assuring safe and reliable operation of nuclear power plants (NPPs). Results and conclusions of probabilistic safety assessment (PSA) are increasingly being used to improve both.

To achieve one of many potential benefits of probabilistic safety assessment, the optimization of limiting conditions for operation (LCO) in technical specifications (TS) is investigated.

Limiting conditions for operation define allowed outage times (AOT). Optimization of AOT bases on risk comparison of two scenarios: plant shutdown and plant continued operation, in case if some equipment becomes unavailable.

2. Recent research

Much work has been done in recent years in evaluating the Surveillance Requirements (SR) [1 2 3 4,5 6 7], LCO [8,9 10,11], and integration of both[12], to agree that improvement of TS [13 14 15 16] leads to safer NPPs.

Besides results of PSA [17,18,19,20], most of the methods are based on risk calculations and risk comparisons. One can also find: the use of Markov processes for maintenance optimization, the use of nonlinear programming [21] and the use of dynamic programming approach.

Majority of the methods use probabilistic criteria such as component and system unavailability, core damage frequency/probability, risk factors [22], some of them use costs in addition [2 23].

3. Risk Comparison Approach

Proposed is the risk comparison approach for evaluation of allowed outage time. The risk of plant shutdown due to failure of certain equipment is compared to the risk of continued plant operation with the specified equipment down.

The risk comparison of two scenarios was the initial idea of Mankamo at al. in ref. [8]. Their approach considered the results of the shutdown probabilistic safety assessment and was used for the example of the auxiliary feedwater system.

Our approach [24] assumes the existence of the risk monitor as an already implemented tool in the nuclear power plant. The core damage frequency (CDF) serves as a risk measure. If risk monitor is not implemented yet, the time dependent core damage frequency is replaced by its constant value for the plant power operation and constant value for the shutdown.

In the case of plant shutdown it is assumed that the shutdown occurs immediately after certain component fails/equipment becomes unavailable.

Calculation of the core damage frequency assuming continued plant operation $CDF_{co}(T)$ and core damage frequency assuming plant shutdown $CDF_{sh}(T)$ is the prerequisite for the calculation of CDF_{co} and CDF_{sh} which are the mean values calculated by equations:

$$CDF_{co} = \frac{1}{Ta} \int_{T_o}^{T_o+Ta} CDF_{co}(T)dT \tag{1}$$

$$CDF_{sh} = \frac{1}{Ta} \int_{T_o}^{T_o+Ta} CDF_{sh}(T)dT \tag{2}$$

where T_a is the examined time interval.

Due to relationship between the T_{outi} (outage time of the equipment i) and other input parameters such as:

T_{csh} ... time of shutdown; from power operation to cold shutdown,
T_{st} ... time of startup; from cold shutdown to power operation,
T_{oc} ... minimum time for the plant to stay in a shutdown after the shutdown has occurred,

the problem is divided into three cases:

1. $T_{outi} > T_{csh} + T_{oc}$ $\quad T_a = T_{outi}+T_{st}$
2. $T_{csh} < T_{outi} < T_{csh} + T_{oc}$ $\quad T_a = T_{csh}+T_{oc}+T_{st}$
3. $T_{outi} < T_{csh}$ $\quad T_a = T_{csh}+T_{oc}+T_{st}$

Equations for the CDF_{co} and CDF_{sh} are normally different in each case.

For those T_{outi} which result in $CDF_{co} < CDF_{sh}$ the risk of continued plant operation is less than the risk of plant shutdown. For those T_{outi} which result in $CDF_{co} > CDF_{sh}$ the risk of plant shutdown is less than the risk of continued plant operation.

The limiting T_{outi} where both risks are equal, is the outage time which is proposed to be an optimal allowed outage time (AOT_{iopt}) for the equipment i.

4. Analytical Solution

The three separate cases exist because some limiting conditions for operation include T_{oc} (minimum time for the plant to stay in a shutdown after the shutdown has occurred). Besides, the time of shutdown transient - time from power operation to cold shutdown T_{csh} is included in the analysis.

The following parameters are included in the mathematical model:

AOT_i ... allowed outage time of component i (or equipment i)

CDF_n ... core damage frequency - nominal value of the plant power operation (result of PSA Level 1)

CDF_{pi} ... core damage frequency - if the specified component i is down (result of PSA Level 1)

CDF_u ... core damage frequency - nominal value of the plant at shutdown

$CDF_{maxi} = CDF_n \cdot F_1 \cdot F_2$

... core damage frequency at the transient e.g. shutdown (with component/equipment i down)

F_1 ... risk increase factor due to shutdown of the plant

$F_2 = I_{mi}$ ... importance factor of component/equipment i (result of PSA Level 1)

F_3 ... risk increase factor due to startup of the plant

T_o ... time of the failure of the component i on time scale

T_{csh} ... (time of shutdown transient) time from power operation to cold shutdown

T_{st} ... (time of startup transient) time from cold shutdown to power operation

T_{oc} ... minimum time for the plant to stay in a shutdown after the shutdown has occurred

T ... time

F_1 ... ratio between the average value of failure rates of components with failure mode: failure to start and average value of failure rates of components with failure mode: failure to run. Or, ratio between the average value of failure rates of components with failure mode: failure to change position and average value of failure rates of components with failure mode: failure to remain in position.

F_3 ... ratio between the number of failures resulted in plant trip during startup over certain time period and number of failures resulted in plant trip during power operation over the same time period.

For each of three cases, equations for CDF_{co} and CDF_{sh} are developed for linear and exponential decrease of CDF. For each of three cases and for both assumed decreases of CDF optimal outage time for analysed equipment $T_{outiopt}$ is calculated.

To ilustrate the proposed approach, mathematical model for assumed linear decrease of CDF for the first case ($T_{outi} > T_{csh} + T_{oc}$; $T_a = T_{outi} + T_{st}$) is shown.

Assumed Linear Decrease of CDF

Case 1: condition 1: $T_{outi}>T_{csh}+T_{oc}$

Equation 3

$$CDF_{co}=\frac{1}{T_{outi}+T_{st}}\left[\int_{T_o}^{T_o+T_{outi}} CDF_{pi}(T)dT+\int_{T_o+T_{outi}}^{T_o+T_{outi}+T_{st}} CDF_n(T)dT\right]$$

Equation 4

$$CDF_{sh}=\frac{1}{T_{outi}+T_{st}}\left[\int_{T_o}^{T_o+T_{csh}}\left(\frac{(CDF_u-CDF_{maxi})(T-T_o)}{Tcsh}+CDF_{maxi}\right)dT+\right.$$

$$\left.+\int_{T_o+T_{csh}}^{T_o+T_{outi}} CDF_u(T)dT+\int_{T_o+T_{outi}}^{T_o+T_{outi}+T_{st}} CDF_n(T)\,F_3\,dT\right]$$

Equation 5

For $CDF_n(T)$, $CDF_{pi}(T)$, $CDF_u(T)$ as constant values:

$$CDF_{co}=\frac{1}{T_{outi}+T_{st}}(CDF_{pi}T_{outi}+CDF_nT_{st})$$

Equation 6

$$CDF_{sh}=\frac{1}{T_{outi}+T_{st}}\left[\frac{CDF_u+CDF_{maxi}}{2}T_{csh}+CDF_u(T_{outi}-T_{st})+CDF_nF_3T_{st}\right]$$

From equations for CDF_{co} and CDF_{sh} (for $CDF_n(T)$, $CDF_{pi}(T)$, $CDF_u(T)$ as constant values) $T_{outiopt1}$ is expressed:

Equation 7

$$T_{outiopt1}=\frac{\frac{CDF_{maxi}-CDF_u}{2}T_{csh}+CDF_nT_{st}(F_3-1)}{CDF_{pi}-CDF_u}$$

If $T_{outiopt1}$ meets condition 1, it is an optimal allowed outage time for the first case.

Developed equations for other two cases of linear decrease of CDF and three cases of exponential decrease of CDF are presented in ref. [25].

In the third case for exponential decrease of CDF the equation: CDF_{co} - CDF_{sh} = 0 from which $T_{outiopt}$ is calculated, is transcendental (even if the values $CDF_n(T)$, $CDF_{pi}(T)$, $CDF_u(T)$ are constant) and it can't be analytically solved. Therefore in parallel the numerical solution is developed.

5. Numerical Solution

Numerical solution includes all three cases from the chapter analytical solution. All three parameters: $CDF_n(T)$, $CDF_u(T)$, $CDF_{pi}(T)$ are functions of time. Here are the steps of the numerical solution:

1. Small Δt is chosen (Δt ... time increment, $\lim \Delta t \rightarrow 0$).
2. At each time point T: $T=\{t_1,t_2,t_3,...t_n\}$; $t_1=T_o$, $t_n=t_{n-1}+\Delta t$; $CDF_{co}(T)$, $CDF_{sh}(T)$ are input values.
3. For both functions: $CDF_{co}(T)$, $CDF_{sh}(T)$, the cumulative functions are calculated:

Equation 8

$$CDF_{cocum_i}(T) = \sum_{n=0}^{\frac{T-T_o}{\Delta t}} CDF_{co}(T_o + n \cdot \Delta t)$$

Equation 9

$$CDF_{shcum_i}(T) = \sum_{n=0}^{\frac{T-T_o}{\Delta t}} CDF_{sh}(T_o + n \cdot \Delta t)$$

4. Both cumulative functions are functions of T and T_{out_i}. Expressed are:

   $CDF_{cocum}(T_{out_i})$ = cumulative CDF_{cocum}; time interval: ($T=T_0 \ldots T=T_o+T_{csh}+T_{co}+T_{st}$),

   $CDF_{shcum}(T_{out_i})$ = cumulative CDF_{shcum}; time interval: ($T=T_0 \ldots T=T_o+T_{csh}+T_{co}+T_{st}$).

5. For those T_{out_i} which result in $CDF_{cocum} < CDF_{shcum}$ the risk of continued plant operation is less than the risk of plant shutdown. For those T_{out_i} which result in $CDF_{cocum} > CDF_{shcum}$ the risk of plant shutdown is less than the risk of continued plant operation. The limiting T_{out_i} where both risks are equal, is the outage time which is proposed to be optimal allowed outage time (AOT_{iopt}) for the equipment i.

   $\Delta CDF_{cum}(T_{out_i}) = CDF_{cocum}(T_{out_i})- CDF_{shcum}(T_{out_i}) = 0, \quad \Rightarrow \quad T_{out_iopt}$;

   $AOT_{iopt} = T_{out_iopt}$

Results of analytical solution may be compared to the results of numerical solution.

6. Results of an Example

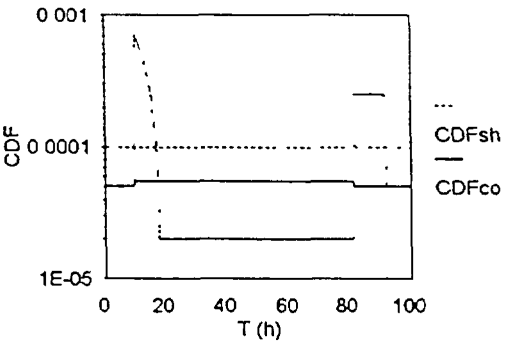

Figure 1: Functions: CDF_{co} and CDF_{sh}

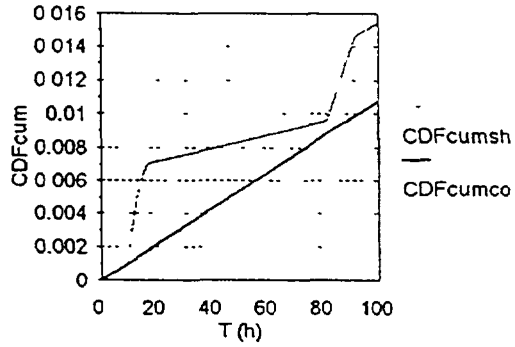

Figure 2: Cumulative functions: CDF_{cumco} and CDF_{cumsh}

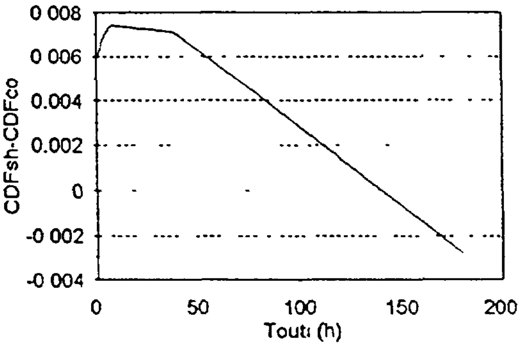

Figure 3: Function CDF_{sh} - CDF_{co}

Figure 1 shows functions $CDF_{co}(T)$ and $CDF_{sh}(T)$ for an example (data for the example is presented in ref. [9]; AOT_i = 72 hours). Figure 2 shows respective cumulative functions $CDF_{cumco}(T)$ and $CDF_{cumsh}(T)$. CDF_{cumsh} on Figure 2 is higher than CDF_{cumco}. A plant shutdown results in higher risk than the continued plant operation. Figure 2 shows that original AOT_i =72 h should be relaxed. Increased AOT_i would allow the equipment i to be in outage longer.

Risk based allowed outage time for the equipment i is calculated from equation:

$CDF_{cocum}(T_{outi})$- $CDF_{shcum}(T_{outi})$ = 0; $\Rightarrow T_{outiopt}$.

Figure 3 shows the function of CDF_{sh} - CDF_{co} versus T_{outi}. CDF_{sh} - $CDF_{co}(T_{outi}$ = 144 hours) = 0

The optimal allowed outage time (AOT_i) for equipment i in the example equals to T_{outi}=144 h.

7. Conclusion

Optimization of limiting conditions for operation has shown the results which are dependent the most on assumed core damage frequency during shutdown and startup. Nevertheless the results are uncertain, they give us the information on risk based allowed outage time which may serve as an additional information in decision making process of improving technical specifications.

For application of the proposed approach for optimization of limiting conditions for operation as one of the criteria for real examples in the nuclear power plant, the future work is needed, which is to be focused to decrease the uncertainty of the risk at the plant startup and shutdown.

Conducted probabilistic safety assessment for the plant operation and for plant shutdown are the prerequisites for the proposed model, therefore the application of this model extends the usefulness of the PSA results in a new way.

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MAINTENANCE RELATED TO LIFE MANAGEMENT: SURVEY AND CONTROL OF EQUIPMENT AGEING

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Abstract

The aim of this paper is to review relevant objectives and aspects of the Maintenance Evaluation and Improvement Programmes for Nuclear Power Plant Life Management.

Recent experience shows that current maintenance practice often fails to directly address long-term degradation that affects singular plant components and equipment populations. Instead, delayed attention to the consequences makes good Life Management unfeasible. This has brought about the need for specific Maintenance Evaluation and Improvement Programmes to adjust to the basic objective of Life Management which is to protect against, mitigate and/or monitor ageing that affects the safe, profitable life of the facility.

The paper analyses the methodologies used, incidents during their application and the main conclusions reached from the implementation of these programmes in Spanish nuclear power plants. Special attention is paid to recommended solutions for improving the efficiency of the utility's contributions, its leadership in task development and integration, and its interfaces with organisations specialised in providing services that support Life Management Programmes.

The coexistence of these and other similar maintenance programmes make it necessary to integrate tasks to optimise effort and tools. The paper analyses the guidelines to be considered when integrating these Programmes with other maintenance optimisation programmes (economy and feasibility, RCM) and with tasks derived from the application of Maintenance Rule regulatory requirements.

Lastly, the paper reports on the state of these Maintenance Evaluation and Improvement Programmes, their development, what prospects they have, and the Industry's initiative and actions concerning the matter.

1. INTRODUCTION

Around the world, power station owners are increasingly concerned to optimise Plant Life Management. In response, they are setting up Life Management programmes, of more or less ambitious scope and depth.

Strategic, economic and security concerns and the close link between life extension work and the improved maintenance practices that are so important today, will increase and globalise these programmes for monitoring and conservation or mitigation of ageing.

These programmes are all based on knowledge of the precise condition of all components and populations with the greatest effect on the economics and safety of the plant, and trends in changes in their condition.

The technical support for these programmes is:

- Methodologies and knowledge required to identify degradation mechanisms as a function of the characteristics of the components or populations, and service conditions

- Techniques for determining condition and trends over time

- Analysis of the efficiency of maintenance practices based on the above knowledge, techniques and methodologies

- Improvement of maintenance practices for adequate mitigation and monitoring of ageing

- Techniques and tools for collecting and ordering data about ageing and for condition assessment

The following sections describe the structure and content of these programmes, with special emphasis on engineering tasks that support them.

2. BASIC OBJECTIVES AND STRUCTURE OF REMANENT LIFE MANAGEMENT PROGRAMMES (RLMP)

2.1 OBJECTIVES OF AN RLMP

The basic objectives of an RLMP translate into information about the condition of the installation and forecasting of its possible change over time. This knowledge enables the selection of adequate measures for monitoring, conservation, mitigation, repair, replacement or modification of the installation and the process, compatible with the owner company's strategy and a cost-benefit ratio favourable for the installation.

An RLMP is a continuously repeating cycle of evaluating condition and taking corrective and/or monitoring measures.

The frequency of evaluation of condition varies with each plant, component or population. It can range from continuous monitoring of components used in very harsh conditions, those that may present unpredictable change and/or those with more weight in management, through to re-evaluation over extended periods for components and populations in which degradation is slower or better understood.

2.2 STRUCTURE OF AN RLMP

On the basis of the objectives described, RLMPs are structured as shown in Figure 1. The activities that support these programmes are:

- Selection of important components and populations, according to economic and safety indicators, and the establishment of priorities for the RLMP on the basis of a rigorous and formal application of the methodologies based on weighted criteria

- Analysis, during the initial evaluation, of the characteristics of the components and their service conditions to identify potential ageing. These are complemented with study of history of incidents during operation and maintenance, and with the definition and execution, where appropriate, of additional tests and/or inspections. Periodic re-evaluations are fed back to the same sources, where they are added to the data from monitoring of ageing trends

- Evaluation and optimisation of maintenance and monitoring practices, to mitigate or survey the effects of ageing

- Analysis, selection and implementation of remanent life management measures, decided on the basis of the tasks described above. These measures are assigned to the following different areas:

  - Repair, replacement and modification in component populations especially affected

  - Modifications to operating procedures to reduce harshness where appropriate

  - Modifications to maintenance practices, to make them more effective for mitigating the effects of ageing

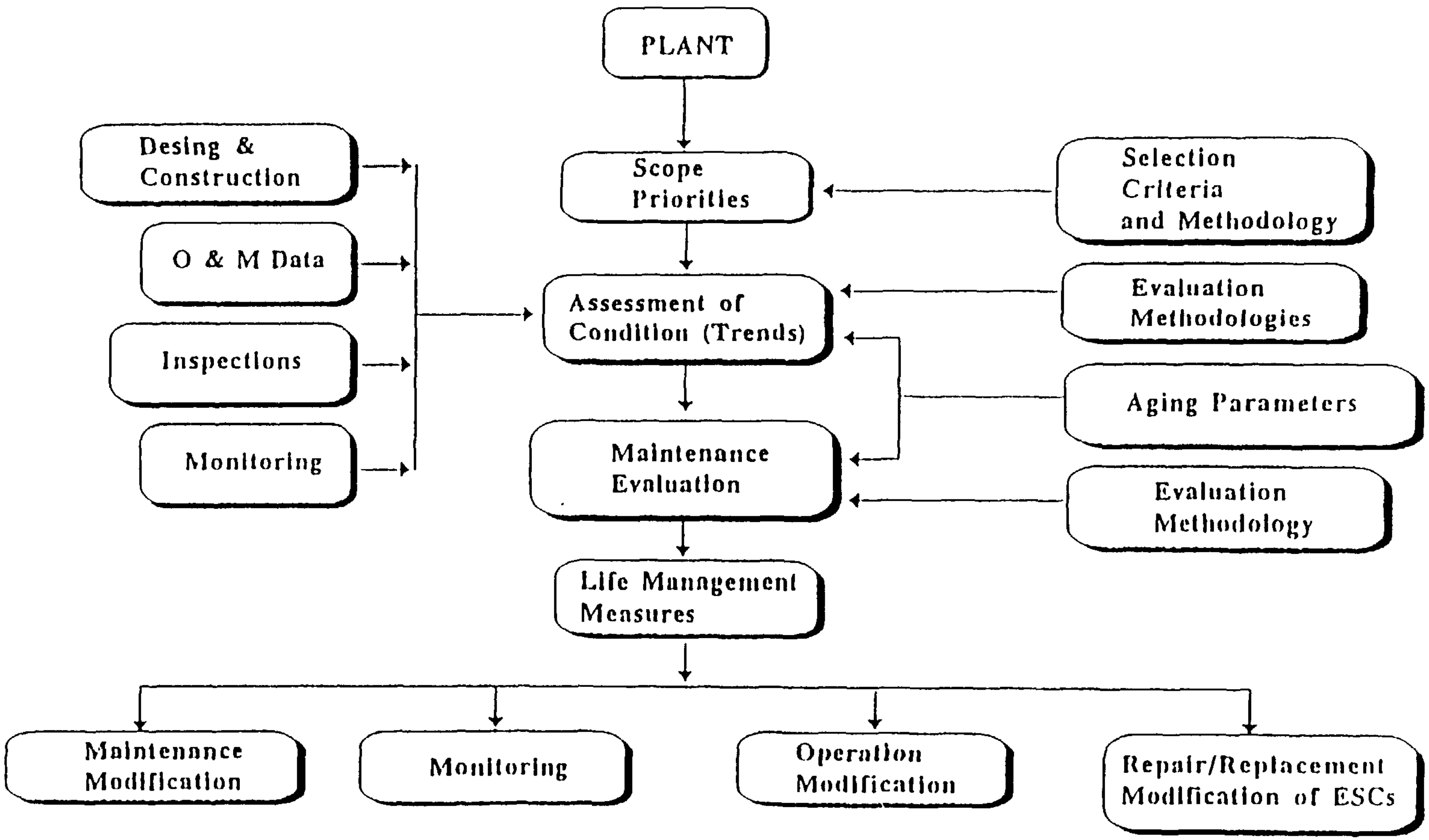
  - Implementation of additional monitoring necessary to obtain a more accurate picture of change in degradation mechanisms of most severe ageing effects or where uncertainty of evaluation is greatest

3. DESCRIPTION OF ENGINEERING TASKS FOR RLMP

All stages of the activities involved in an RLMP require heavy support from specialised engineers. The prediction of potential ageing and evaluation of the degree to which this affects the different components, and especially, the monitoring of change in their condition and/or prediction of the change, as well as the definition of corrective or monitoring measures, require massive specialised engineering support in these fields.

There follows a brief description of the main tasks and methodologies required for an RLMP.

STRUCTURE OF LIFE MANAGEMENT PROGRAMMES



PLANT

Scope Priorities

Desing & Construction

O & M Data

Inspections

Monitoring

Selection Criteria and Methodology

Assessment of Condition (Trends)

Evaluation Methodologies

Aging Parameters

Maintenance Evaluation

Evaluation Methodology

Life Management Measures

Maintenance Modification

Monitoring

Operation Modification

Repair/Replacement Modification of ESCs

Figure 1

3.1 SELECTION AND PRIORITISATION OF COMPONENTS WITHIN THE RLMP STRATEGY

The first requirement for adequate plant life management is to avoid dispersion and waste of the RLMP resources. These resources are always limited, and should not cover the whole population, indiscriminately. Prioritisation is necessary and needs to be slightly adjusted periodically, to adapt to the margin of uncertainty of all predictions. This prioritisation uses a weighted criteria methodology.

The strategy of each plant affects the methodology through adjustments in the plant-unique weighting of each of the criteria.

The filter criteria for the selection of components are grouped in three types:

- Safety criteria
- Availability criteria
- Replacement and cost criteria

Compliance with any of the criteria above means that the component is important for remanent life management. The criteria of the Weighting Methodology, are grouped to apply them in a more homogenous way.

Each of the criteria are assigned their own waiting factor. The plant should participate with the engineering panel in the definition of some of the factors as far as these factors are a reflection of its operating strategy.

The conclusion of this process is a prioritised list of components, which is used as the subject of the RLMP.

3.2 IDENTIFICATION OF DEGRADATION AND EVALUATION OF CONDITION

This task consists of clearly differentiated stages. The RLMP begins with an initial condition evaluation, which serves as the basis for establishing the main corrective and monitoring actions, and for preparing the first cost/benefit analyses for Life Management. The RLMP continues to progress with periodic re-evaluation of condition to confirm the corrective measures are the right ones and to adopt new measures, if necessary, as a result of the monitoring established.

Appendix A provides additional information regarding the ageing processes affecting main components and component groups in nuclear power plants.

The initial evaluation begins with a determination of potential degradation mechanisms and of the level of harshness of these on the selected components. This requires a study of the characteristics of the components relative to their design, materials, manufacture, process and service conditions.

This analysis is complemented by a rigorous study of the history of the operation and maintenance, and the results of diagnosis and monitoring, to detect incidents that might have affected the condition of the plant, or for evidence of degradations. Uncertainty about the severity of some of these ageing effects may require extra inspections or tests, to provide more precise data.

Condition evaluation requires collection and ordering of the documentation and records of manufacturer, operation and maintenance that contain information needed for the analysis. This collection requires application of procedures that establish the data and records, with the periodicity of their acquisition clearly identified for successive re-evaluations, and the screening requirements for easier collection and analysis. In this area, the Events Log is of special interest. It organises the selected records of Plant Events that have a significant effect on component life and that are being used, both for the initial evaluation and for periodic re-evaluations.

The periodicity of successive re-evaluations varies for each plant, component and population, depending on age, management strategy and the severity of the ageing. The purpose of re-evaluation is to confirm or modify the corrective action taken on the basis of monitoring of the ageing. The benefit of an RLMP is based on the precision of the determination of the condition and especially its trend, to make possible the calculation of residual life required to support any management decisions.

For this purpose, a programme is established for monitoring parameters that represent the progress of ageing. This together with the results of the inspections, testing and maintenance work at the plant constitute the raw material of the residual life analysis.

The need for this monitoring and the prediction tools described, have led to development work, which in the case of Spain, are referenced to the Project for Development of a Remanent Life System for Nuclear Power Plants[1] (SEVR) that arose from a life extension management initiative by UNESA and owners' groups. The project has completed functional specification and architecture of the system, and is ready to begin developing the Pilot application.

Chapter 4 describes in more detail, the engineering tasks involved in the installation and use of specific monitoring and global systems as described above.

[1] *Proyecto de Desarrollo de un Sistema de Gestión de Vida Remanente de Centrales Nucleares*

3.3 MAINTENANCE PRACTICES EVALUATION TO OPTIMISE RLMPs

In addition to the improvements in operation and service conditions, a substantial part of the causes and effects of ageing mechanisms have to be mitigated by maintenance work. The nature of these long-term ageing mechanisms has meant that, in certain cases, current maintenance practices do not prevent them. This requires these practices to be evaluated and modified where necessary to improve their efficiency in conservation and the mitigation of degradation.

Appendix B describes the methodology for maintenance evaluation and provides details regarding the lessons learned during its application in the two pilot plants.

The engineering activities followed in the evaluation process are:

1) DEFINITION OF SCOPE OF EVALUATION OF MAINTENANCE

The tasks described above produce the component-degradation mechanism pairs that it is considered necessary to evaluate.

2) PRODUCTION OF COMPONENT DEGRADATION SHEETS (CDS)

A component degradation sheet (CDS) is completed for each component selected.

The data to be filled out on the CDSs are: component description; functions; design parameters; operating experience; degradation mechanisms; and the part of the component affected by ageing.

3) PREPARATION OF MAINTENANCE PRACTICES DATA SHEETS

For each of the programmes, practices and procedures that affect each component/degradation mechanism pair, a data sheet is prepared, showing the following information about the practice: limitations on performing it, time when corrective action is taken, the data necessary, action to be taken to mitigate, detect and monitor the degradation and finally comments and experience resulting from the practice application.

The purpose of this task is to take and inventory of all practices current at the Plant and to discover details of the application, to exploit them and improve their efficiency for life extension.

4) EVALUATION

Each Component Data Sheet is attached to all the Maintenance Practice Data Sheets that affect the component. With the information from both sources, the Maintenance Evaluation Checklist is completed.

The evaluation shows the possible deficiencies in control of ageing of the maintenance of each component. When necessary, improvements to maintenance are proposed, documenting the details of the improvements using the tool developed for that purpose, the Maintenance Evaluation Proposed Improvement.

3.4 ANALYSIS, SELECTION AND APPLICATION OF MEASURES FOR IMPROVEMENT OF LIFE MANAGEMENT

The tasks described above provide information about ageing and trends, and the degree of uncertainty in their evaluation, and also a determination of the efficiency of maintenance practices and their shortcomings. On the basis of this, it is possible to decide on the life extension measures to be applied. These measures fall into the following categories:

- Repairs, replacements or modifications and most efficient programming, of the components most severely effected and/or for which the improvement in availability or performance justifies the investment. It is important to remember that Remanent Life is only considered as such if it is safe (reliable) and economically viable
- Modifications to operating procedures and/or in service conditions to make them less harsh
- Improvements to Maintenance Practices, to achieve full efficiency, for safe and economically viable life extension
- Implementation of additional monitoring with some of the following criteria:
    - Improve precision of condition evaluation and trends, for those component/degradation mechanism pairs for which forecasting is more uncertain
    - Allow for continuous condition monitoring, or at least to reduce the effort required for collection and analysis of the information required during re-evaluation

    This improves the flexibility and solvency of life management decisions

    The type of monitoring and the parameter to represent ageing should be selected with realistic criteria of accessibility and efficiency

4. REMANENT LIFE EVALUATION SYSTEMS. ENGINEERING TASKS FOR ADAPTATION AND USE

As described in Section 3, the need for an increasingly precise and up to date understanding of the condition of the components and the evaluation of their state over time for good Life Management, has created a need for tools and methodologies. This situation has led to development of specialised systems for monitoring of certain types of ageing (fatigue, erosion-corrosion, stress corrosion cracking, vibration, degradation of electrical insulation, electrical machines, etc.) and more ambitious general systems, such as the SEVR developed in Spain by UNESA and that concerns the acquisition, storage and processing of data for significant parameters and evaluates the conditions and trends in conditions over time of the main single components and component populations.

The application and use of this type of system requires a substantial amount of engineering, which translates into tasks such as those described below:

- Definition of the scope of application suited to management of the plant and its configuration and characteristics. This task includes, as described above, prior evaluation of significant ageing and the components affected, and analysis of the effectiveness of maintenance practices to mitigate them

- Analysis of information generated by operation and maintenance, that may be used by the Remanent Life Evaluation System and communication lines, acquisition modes and interfaces

- Analysis of available signals that are useful to the Remanent Life Evaluation System, data lines and process and pre-processing requirements

- Analysis of specialised monitoring and diagnosis systems, available at the Plant and communications lines and interfaces with the Remanent Life Evaluation System

- Definition, location and characteristics of new monitoring sensors, signal pre-processing and data acquisition processing

- Adaptation of Remanent Life Evaluation System to specification of each plant (communications, acquisition, storage and processing of data, algorithms for evaluation of conditions and trends, incorporating coefficients and factors specific to each component, definition of admissible limits for ageing-significant parameters, etc.)

The nature of these engineering tasks makes it essential that the system be adapted and installed by persons or organisations that are experts in ageing of installations and monitoring and diagnosis tools.

Appendix A

AGEING OF MAIN COMPONENTS IN NUCLEAR POWER PLANTS

A1 INTRODUCTION

The tasks of surveillance, evaluation and control of ageing, and the research efforts of the majority of countries have provided several valuable lessons to be taken into account in Plant Life Management Programmes.

Plant Life Management Programmes carried out by Electric Utility Owners in practically all countries —and pressure from regulatory bodies to monitor the ageing of NPPs and its resulting potential impact on their safety— have resulted in a profound knowledge of critical degradations and represented an economic and technical effort from which future power plants should benefit.

This paper examines serious degradations that affect main, single components and component populations.

A1.1 Ageing of Main Components in NPPs

NPP systems, components and structures undergo degradations of different types even before they are installed. Some of the degradations affecting main components or groups of components in varying degrees are listed below:

- Fatigue
- Stress Corrosion Cracking (SCC)
- Corrosion
  - General corrosion
  - Local corrosion
  - Microbiologically influenced corrosion (MSC)
- Erosion and Erosion/Corrosion (E/C)
- Creep
- Wear
- Stress relaxation
- Embrittlement
  - Thermal
  - Strain age
  - Neutron
- Fouling
- Cracking/spalling
- Electronic drift
- Vibration of electronics
- Electrical component design factors
- Thermal/irradiation ageing of nonmetallic materials

Obviously the effects of these vary from component to component, depending on the variables of design, materials, manufacture, process, fluid chemistry, environment, stresses associated with operating modes, maintenance, etc.

A few of the more severe degradations affecting some of the most significant components or structures are listed below.

Main Degradations in Materials Used in the Reactor Coolant System and Related Systems

- Wrought Austenitic Stainless Steels

  Main Problems: Sensitisation and cold work from forming and bending make the material susceptible to IGSCC and IASCC

  Solutions: Use materials resistant to sensitisation (low-carbon types such as 304L, 316L, 304NG, 316NG and modified 347). Materials in the solution heat-treated condition (≈ 2000°F) and grinding and cold work control

  Ensure average ferrite content in the welding materials in the range of 5 to 13 FN (Ferrite Number)

  For welded designs of internals, crevices, fillet welds and dissimilar metal welds should be avoided

  Prevent corrosive environments by limiting halogens to < 5 ppm and $O_2 < 10$ ppb

  Reduce neutron fluence for vessel and internals ($< 10^{20}$ MeVn/cm^2)

- Martensitic Stainless Steels

  Main Problems: These materials are used chiefly in pumps and valve components and are susceptible to SCC

  Solutions: Control the heat treatment (normalised and tempered) to limit hardness

- Ni, Cr, Fe Alloys

  Main Problems: These materials, used in specific applications due to their strength and low thermal expansion characteristics, present SCC (Alloy 600 and Alloy X-750 because of improper heat treatment) at temperatures greater than 600°F

  Solutions: Reduce the neutron fluence ($< 10^{20}$ MeVn/cm^2) to avoid IASCC

Use Alloy 690, but only for specific components (till proven by experience)

Carry out special treatment (1300°F, 12-20 hours) to improve resistance to SCC

Restrict the use of Alloy X-750 with tough material specification

- Austenitic Stainless Steel Castings

Main Problems: Embrittlement after long periods of exposure to high temperature ($>$600°F), due to the transformation of delta ferrite to a sigma phase. Molybdenum contributes to degradation

Solutions: Solution heat treatment ($>$2000°F), use of centrifugal casts, ferrite control ($5 > FN < 14$) and molybdenum control

- Carbon and Low Allow Steel

Main Problems: Corrosion, erosion/corrosion, pitting and crevice corrosion, and environmentally-assisted fatigue

Solutions: Use austenitic stainless steel cladding for PWR vessels, BWR vessels with hydrogen water chemistry and piping in contact with the reactor coolant

For BWR vessels without H_2 chemistry and the remaining components, establish corrosion allowances on the basis of experience in the industry and a plant life of 60 years. Limit the sulphur content to obtain a high resistance to environmentally-assisted fatigue crack growth

The use of 1% Cr alloys or, wherever possible, piping designed for low fluid velocities ($<$ 5 ft/sec) will reduce the erosion-corrosion rates

Single Components. Specific Problems

- Reactor Pressure Vessel. Fast Neutron Embrittlement

Experience shows that changes occur in the ductility properties of the RPV material due to the effect of fast neutron exposure. Decreased notch-ductility is a function of neutron dose, irradiation temperature and content of copper, phosphorous, vanadium and nickel in the welding materials used for joining the ferritic base materials of the RPV. The parameter used to express ductility reduction is the reference nil ductility transition temperature (RT_{NDT}) and is used to define pressure and temperature transients and limits during heatup, cooldown and pressure tests.

The design and manufacture of new RPVs should diminish the effect of embrittlement in zones directly surrounding the active core which are exposed to the highest neutron radiation.

To reduce this degradation, penetrations and nozzles should be avoided in the beltline region and, in general, it is advisable to decrease the number of welds in these zones, use base and welding materials with a low content of copper, phosphorous, vanadium and nickel, and reduce the levels of neutron flux affecting the shell.

The current reactor vessel material surveillance programme based on test data should be maintained for the new RPV even though the RT_{NDT} shift predicted on the above basis is conservative.

Fatigue Failures in LWR Components

Field failures have identified several sites susceptible to damage from fatigue which were not considered vulnerable to fatigue in the original design.

Failures of components on which fatigue analyses were performed have resulted from stressors which were not accounted for in the design analysis. These stressors include low- and high-cycle fatigue due to thermal stratification, high-cycle thermal fatigue from thermal stripping and thermal mixing, mechanical fatigue from flow-induced vibrations and low-cycle environmentally-assisted fatigue (Tables 1 and 2 show examples of fatigue failure areas in LWR components).

Table 1 Areas of Fatigue Failures in BWR Plants

Location	**Mechanical Stress[2]**		**Thermal Stress**	
	High-Cycle	**Low-Cycle**	**High-Cycle**	**Low-Cycle**
Reactor vessel				
Feedwater nozzle			x	x
CRDRL nozzle			x	x
Reactor internals				
Feedwater sparger	x			
Jet pump	x			
Steam dryer	x			
Recirculation system				
Pump internal welds	x			
Thermowell	x			
Small piping				
Branch connections	x			
Instrumentation lines	x			
Control rod drive system				
Insert/withdraw lines	x			

[2] Not including pressure stress

Table 2 Areas of Fatigue Failures in PWR Plants

Location	Mechanical Stress[3]		Thermal Stress	
	High-Cycle	Low-Cycle	High-Cycle	Low-Cycle
Steam generator				
Feedwater nozzle area			x	x
Girth weld area				x
Tubes	x[4]			
Pressuriser				
Lower head				x
Diaphragm welds (B&W)				x
Reactor internals				
Flux thimble tubes	x			
Bolts	x			
Holddown ring	x			
Core barrel	x			
In-core instruments	x			
CRDM penetrations in RPV head			x	x
Reactor coolant pump				
Shaft	x			
Piping				
Thermal sleeves	x			
ECCS and RHR piping			x	x
Feedwater piping			x	x
Surge line			x	x
Small piping				
Branch connections	x			
Instrumentation lines	x			

Steam Generators

The main degradations affecting these components are:

- IGSCC affecting the inner surface (PWSCC) of U-bends and roll-transition zones, and the outer surface of hot-leg tubes in the tube-to-tube sheet crevice zone

- Pitting in cold-leg tubes where scale contains copper deposits

- Wastage on the outer surface of the tubing above the tube sheet

- Denting affecting the tubes in the tube-support zones

[3] Not including pressure stress

[4] Environmentally-assisted fatigue caused stress concentrations for crack initiation in once-through steam generators, and anti-vibration bar problems caused failures in recirculation steam generators

- Fretting due to flow-induced vibrations which affect contact points between the tube and the antivibration bar

The solutions for limiting stressors are put into practice by one or several of the following lines of action:

- Use of heat-treated Alloy 690 for tubing material to provide more resistance to SCC in an alkaline environment

- Use of tube support plates manufactured from chromium ferritic stainless steel and new designs with broached holes which direct the flow along the tubes reducing dryout in the crevices

- Inclusion of blowdown arrangement and support plate geometry to improve flow distribution above the tubesheet, and to minimise potential local concentrations of impurities (in the tube-to-tubesheet intersection zones)

- Tube expansion procedures which eliminate the tube-to-tubesheet crevice

- Strict control of water chemistry on the secondary side and use of titanium condenser tube material

Emergency Diesel-Generators

The analysis of EDG failures shows that more than 50% of them may be attributed to ageing.

Some of these ageing mechanisms are:

- Vibration
- Thermal and mechanical shocks and excessive operating loads
- Corrosion
- I&C set points drift
- Chemical attack from fuel and lube oils
- Environmental conditions and fouling
- Microbiologically influenced corrosion

The solution to such ageing lies in implementing certain improvements in the equipment, and a degradation detection and mitigation programme right from commencement of operation.

Improvements could include the incorporation of prelubrication and preheating of the diesel.

The degradation detection and mitigation programme should include:

- Plant Maintenance oriented to preventive maintenance based on trending of significant parameters, rather than overhauls on a strictly periodic basis
- Testing procedures which exclude harmful practices. The test programme should include prelubrication, slow loading, longer run times, and post-test gradual load reduction and cooldown
- Vibration monitoring/signature analysis
- Lube-oil analysis and ferrography to detect metal wear
- Specific governor maintenance based on the manufacturer's recommendations
- MIC control programme

Instrumentation and Control Equipment

The following are degradation mechanisms that affect I&C subcomponents:

- Corrosion (diaphragms, bellows, bourdon tubes, electronics, switches, linkages) (Note: corrosion also includes IGSCC, MIC)
- Thermal ageing (solenoid valve operators, high temperatures in the process and in the environment)
- Fatigue (diaphragms, bourdon tubes, bellows, linkages)
- Wear (elastomer seats, linkage mechanisms, switches)
- Electronic drift (circuits)
- Vibration (design and installation practices)
- Radiation (nonmetallics, O-rings, seals, insulation)
- Setpoint drift (mechanical/electrical interaction)

The significance of any of these degradation mechanisms to the safety functions is limited by the programmes used to detect and mitigate such degradation. Plants should therefore establish programmes of such frequency and attributes as to ensure early, adequate identification and mitigation of any ageing parts and their replacement (by plugging technology).

Typical attributes of these programmes include calibration, functional testing, visual and thermographic inspections, and operator logs and checklists.

Electrical Equipment

The potentially significant ageing degradation mechanisms for these components are:

- Corrosion of buses, transformers, contacts, operating mechanisms, electronics, relays, switches, cables, motor bearings, connectors, batteries, battery chargers and panel components

- Fatigue (including vibration) in connectors, contacts, operating mechanisms, relays, circuit breakers, cables, motor bearings, battery grids and case, chargers and inverters

- Wear of contacts, operating mechanisms, relays, circuit breakers and motor bearings

- Electronic drift in electronic components and devices, relays, inverters, chargers and panels

- Design factors in contacts, arc chutes, operating mechanisms and circuit breakers

- Fouling of motor winding insulation and bus insulators

- Loss of mechanical and electrical properties (e.g., insulation resistance power factor or loss factor) in cable jackets and insulation due to radiation and thermal effects and dependent upon the materials used in the cable

It may generally be said that degradation associated with electrical equipment can be detected and mitigated by means of adequate plant maintenance and testing activities based on vendor recommendations and plant/industry operating experience. Special attention should, however, be paid to cables since their replacement is complicated and requires extensive outage time.

A2 MAINTENANCE FOR AGEING MANAGEMENT

An Effective Maintenance Programme is the cornerstone of the Ageing Management Programmes which they are designed to support.

A programme or combination of programmes meeting the following criteria is considered to be effective for detecting and mitigating ageing, and for monitoring performance throughout plant life:

1. The programme and implementation procedures ensure that component functions are properly addressed, considering the effects of age-related degradation and performance criteria

2. The programme and implementation procedures are kept up to date with expected significant changes and/or tendencies in ageing processes

3. The programme establishes specific acceptance criteria to determine the need for corrective actions

4. The programme provides for adequate monitoring of process and physical parameters used in performance evaluations

These Maintenance and Ageing Control Programmes are based on the timely definition or identification of any significant ageing, its evolution, causes, location and representative parameters and its permanent comparison with maintenance practices, monitoring, testing, inspection, housekeeping, etc. Assessments are carried out on such specific attributes of these practices as adequacy, frequency, action levels, acceptance criteria, corrective measures, documentation requirements, etc.

Appendix B

MAINTENANCE RELATED TO LIFE MANAGEMENT

B1 MAINTENANCE ENGINEERING OF LIFE MANAGEMENT PROGRAMMES

B1.1 Introduction

An adequate plant Life Management requires the establishment of maintenance aimed at identification, monitoring and control of the ageing processes that affect plants. The characteristics of single equipment items and component populations in NPPs and the severity and peculiarity of their service conditions have produced specific forms of degradation which are not always covered by current Maintenance practices which tackle, solely and belatedly, the consequences of these ageing processes.

This has led to the need to assess maintenance practices and adapt them to the basic objective of Life Management which is conservation, mitigation and/or monitoring of ageing processes that affects plant safety and profitability.

New regulatory requirements, regarding monitoring of ageing processes and ensuring efficient maintenance as a safety guarantee (Maintenance Rule), have reinforced the need for Maintenance Evaluation to optimise resources and tools.

Maintenance Evaluation and Optimisation activities aimed at efficient Life Management and their integration into those associated with the aforementioned regulatory requirements are essential for plant operation safety and profitability.

The following is a summary of the methodologies and contents of NPP maintenance evaluation and improvement based on their application to Garoña and Vandellós II NPPs as part of the UNESA Project for the development of a NPP Residual Life Evaluation System.

B1.2 Objectives of Maintenance Evaluation and Improvement Programmes for Life Management

Maintenance evaluation and improvement aimed at optimising Life Management is integrated in terms of scope, priorities and cost/profit balances into Life Management Programmes.

In other words, both the scope of the maintenance practice evaluation process and the objectives of this evaluation are focused on determining, for component populations important to Life Management, the efficiency of prevailing maintenance practices to prevent, mitigate and/or monitor ageing and correct its consequences, by providing resources for safe and profitable plant management.

The purpose of these programmes is to question existing practices regarding their efficiency in ensuring adequate reliability and immediate availability. The experience of Spanish NPPs in this respect confirms that these practices are efficient and there are optimisation programmes under way to reduce activities and their concentration on components with greater responsibility.

The evolution of degradation mechanisms is in most cases slow and easily detectable through normal inspection and monitoring. Lack of knowledge about these ageing processes and their evolution precludes the adoption of mitigation and/or monitoring measures and that is why it is necessary to assess these practices in order to adapt them to the objectives of the Life Management Programme.

The main objective of Maintenance Evaluation for Life Management, therefore, is to detect additional tasks or changes in the frequency or scope of some existing ones aimed at identifying and mitigating long-term degradations and their evolution, which are the vital variables to be considered in Management decisions. The basic contents of this main objective are summarised below.

- Identification of all degradation mechanisms that affect components or structures and cause failures or malfunctions. Knowing the cause allows adoption of more efficient, short- and long-term maintenance and/or operation measures saving labour and the cost of corrective actions or replacements

- Following up ageing processes using the parameters defined in the Maintenance Evaluation is a fundamental contribution to predictive Maintenance activities, with their attendant advantages

- The updated information on plant condition generated by Maintenance for Life Management, and more important still, trends in the development of this condition are essential for any operation strategy

- Maintenance improvement for life management constitutes the basis for compliance with regulatory requirements on the control of ageing and its impact on nuclear safety. It also contributes to keeping open the option of license renewal, even beyond design life

- The basic objective of maintenance evaluation and improvement for life management is to maintain safety in the facility throughout its life. This concurs with the philosophy of the Maintenance Rule (10 CFR 50.65), so that the improvement of maintenance practices must translate in the medium and long term in the upkeep of the good performance levels required by the Maintenance Rule

- This makes maximum use of the set of specific programmes established in the power plants while avoiding unnecessary or duplicated work through the knowledge of the internal causes of degradation. Hence the importance of integrating these programmes with maintenance optimisation ones based on statistical reliability levels

B2 METHODOLOGIES OF THE MAINTENANCE EVALUATION AND IMPROVEMENT PROGRAMMES FOR LIFE MANAGEMENT

These programmes have been applied in Spain at the nuclear power plants of Garoña and Vandellós II, in the framework of a project of the Spanish electric power sector aimed at developing a life evaluation system.

The evaluation methodology covers the activities indicated in Figure B1 and briefly described below:

a. Determination of the population covered by the evaluation, following selection criteria established in accordance with the life management strategy of each plant. The application of weighted criteria and of the Delphi methodology also allow the establishment of management priorities

b. Identification of the significant degradations to be evaluated. Said identification came as a result of the studies of degradations that could significantly affect, directly or indirectly, the population selected. In any event, this process was carried out through the condition evaluation tasks performed in the corresponding life management programme. From these sources is obtained the information contained in the Component Degradation Sheets, which provide the basic data regarding materials, design, construction, configuration, operation conditions, design and fabrication codes, degradation factors, affected subcomponents, operating history and relevant incidents. The Degradation Data Sheet must contain complete information on each component/degradation for the efficiency analysis of maintenance practices with respect to ageing control. The proper preparation of these data sheets was based on the participation of experts in the evaluation of ageing processes; they integrate all the information required to make them self-sufficient in the evaluation

c. Inventory of current maintenance practices in the power plant. The practices to be evaluated cover all testing and inspection tasks required in the technical specifications, the inspections imposed by applicable codes and standards, as well as preventive and predictive maintenance activities with their surveys. They also include all the tasks covered in the specific programmes (motor-operated valves, erosion/corrosion. SCC, electrical engines, calibrations, environmental qualification, leaks, vibrations, etc.) of each power plant

   An inventory of activities is established for each maintenance practice, and laid out in the corresponding Maintenance Instructions Sheet, indicating the subject, scope, acceptance criteria, collected data, frequency, as well as the events and incidents of interest that are related to the practice

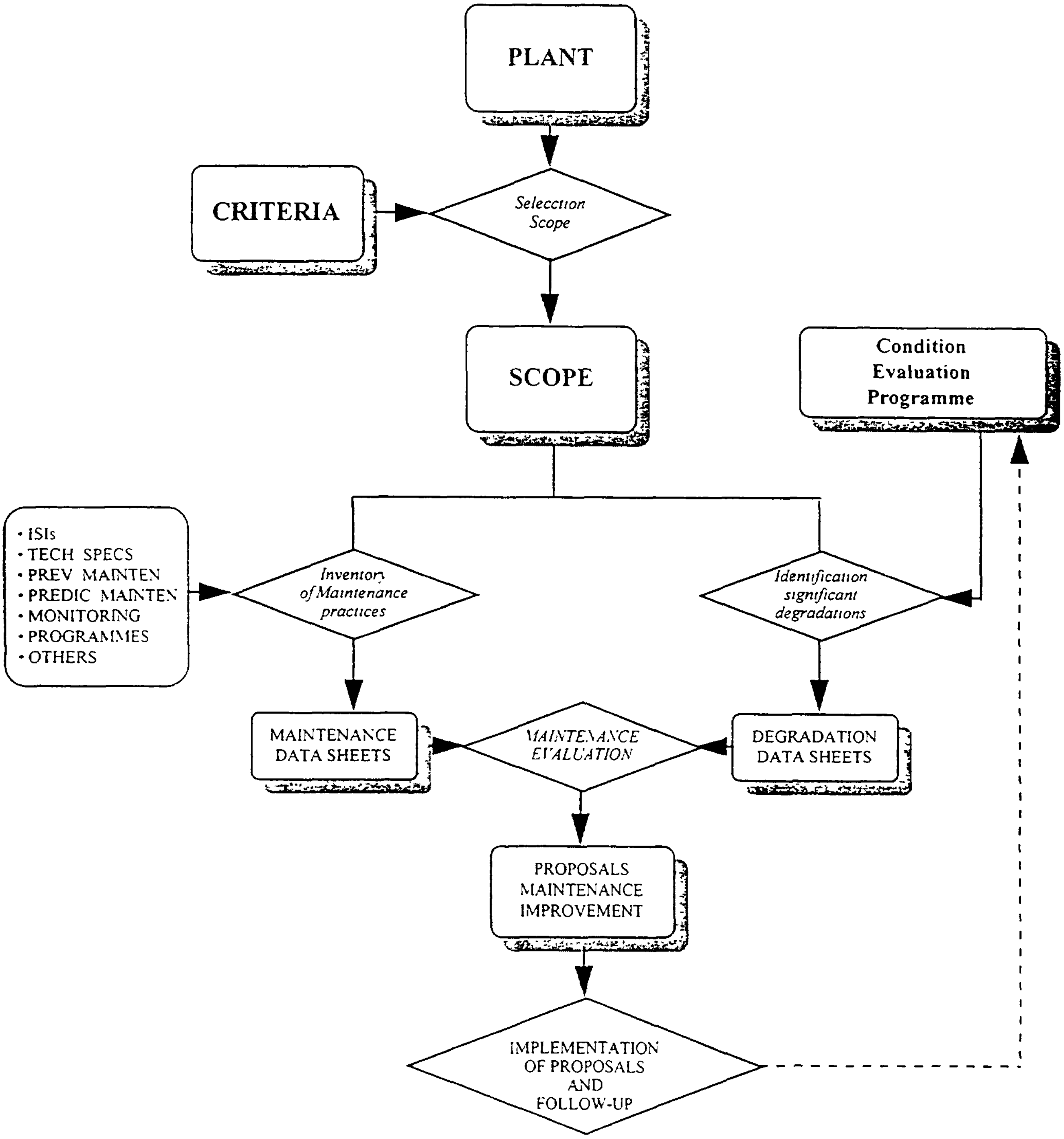

PLANT
CRITERIA
Selecction Scope
SCOPE
Condition Evaluation Programme
• ISIs
• TECH SPECS
• PREV MAINTEN
• PREDIC MAINTEN
• MONITORING
• PROGRAMMES
• OTHERS
Inventory of Maintenance practices
Identification significant degradations
MAINTENANCE DATA SHEETS
MAINTENANCE EVALUATION
DEGRADATION DATA SHEETS
PROPOSALS MAINTENANCE IMPROVEMENT
IMPLEMENTATION OF PROPOSALS AND FOLLOW-UP

Figure B1

The participation in the inventory activities of personnel specialised in the different components and their degradation mechanisms led to orienting the consultations on site to gather information regarding each practice in a selective and efficient way, allowing the reduction of time required for the process and the improvement of inventory quality for the purpose of the evaluation

d. Evaluation. The methodology was complemented with relevant evaluation guides to determine in detail the weaknesses of each maintenance practice in some of the following areas:

- Appropriate scope and depth to detect the degradation
- Suitability of the frequencies and acceptance criteria
- Information generated sufficient for evaluation purposes
- Formal procedures and data

Experience Obtained

The performance of the aforementioned engineering tasks and their results have provided useful experience, whose main conclusions are laid out below:

- The availability of proven methodologies and the contribution of personnel with experience in the tasks described make it possible to improve the efficiency of current maintenance practices in order to optimise plant life management

- The cost of implementing the recommendations arising from the evaluation process is of little significance, and is always compensated by the benefits resulting from the safe and profitable management of the facilities

- There are obvious benefits in integrating these programmes with the maintenance rule implementation programmes or in their complementarity with other programmes such as maintenance optimisation

Here are some of the advantages of such an integration:

- The implementation of maintenance improvements to optimise life management should translate into improved performance of the maintenance rules in the medium and long term

- The condition monitoring imposed in the framework of life management is the valid performance parameter that can be used in the Maintenance Rule for structures, cables. active components in standby and passive components

- The justification of inherently reliable structures and components within the Maintenance Rule is supported by the follow-up on the condition of life management programmes

- The information generated in life management programmes regarding operation records, population statistics, trends in condition evolution, etc., are basic materials often shared with the Maintenance Rule. Hence the convenience of integrating the different information systems (databases, information supports and sources, and tools)

- These maintenance improvement programmes for life management go hand in hand with the optimisation programmes, as they are complementary in their objectives (short-term v medium- and long-term), in their scopes (mainly oriented to integrity in the one, and to operability in the other), in the identification of critical points (through evaluation of condition in the first, through statistics in the other), and in surveys (condition v performance)

  For these reasons, these programmes, far from being mutually exclusive, require on the contrary to be fully integrated

Just as the early implementation of a Life Management Programme is recommendable to reap the most benefits possible, so is maintenance evaluation to such an end, in order to ensure early knowledge of degradations and performance of conservation, mitigation and/or monitoring actions as soon as possible, which should contribute decisively to better life management.

DEVELOPMENT OF THE MAINTENANCE RULE IN SPAIN

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Abstract

In response to a request from the Spanish Nuclear Safety Council (CSN), the Spanish Nuclear Utilities developed a comprehensive plan to comply with the provisions of the Maintenance Rule, 10CFR5O.56. This paper discusses the objectives of the programme, some organizational and methodological aspects and some results, insights and lessons learned from the implementation of the Maintenance Rule in Spain.

1.0 INTRODUCTION.

In response to a request from the Consejo de Seguridad Nuclear (CSN), the Spanish Nuclear Utilities, developed a comprehensive plan to comply with provisions of the Maintenance Rule, 10CFR5O.56 To assure consistency and efficiency, the Spanish nuclear plant operators, formed a Working Group, the GPEMR, to develop, guide, and implement the program while presenting and discussing the Maintenance Rule with the CSN. The Detailed Methodology Plan is based on the principles set forth in the NUMARC Guides 93-01, developed by the U.S. nuclear industry as an acceptable method of compliance. Additionally, a Test and Application process is included in the program to exercise the methodology at one PWR and BWR each, to facilitate utility interaction, training, and feedback of experience. This again, is assuring consistent application and effective and timely compliance.

2.0 PROGRAM OBJECTIVES

The Program developed by the GPE-MR, is based on the following objectives.

1. Promote and reach consensus among the Spanish nuclear plant owners to adopt a single acceptable methodology for the implementation of the Maintenance Rule.
2. The GPE-MR will seek acceptance of the methodology on a generic basis from the CSN, to assure consistent application in the Plants.
3. The Program must result in real plant improvement for performance, safety and operation and must be flexible to accommodate plant-unique considerations.
4. This basic methodology, will closely follow the U.S. utility program, plant implementation experience and evolution of the Maintenance Rule.
5. Coordinate on-going efforts. Aging management, maintenance optimization, etc.

3.0 ORGANIZATION

The GPE-MR Working Group consists of two representatives of each of the seven Spanish Nuclear Sites, with one being from de plant Maintenance Organization. Persons from the licensing and reliability organizations are also involved.

The main function of this group, is to reach technical consensus among the Plants, according with the defined objectives.

In addition, the GPE-MR has retained a U.S. consultant to assist the group with technology transfer and U.S. experience in the MR appl4.0 PROGRAM STRATEGY

The strategy being pursued by the GPE-MR consists of two distinct phases:

- Phase 1 - A detailed methodology plan is prepared. The methodology plan is further elaborated by a set of Application Guides to provide the generic details for consistent MR implementation.

- Phase 2 - A test and application program is conducted to exercise and apply the application guides at a BWR and PWR pilot plant. This step is intended to identity any specific difficulties in data gathering, scope determination, risk significance assessment, performance criteria, and goal setting, and any other aspect of the program. Experience form this validation process, U.S. implementation experience, and the evolving nature of the Maintenance Rule are then incorporated into the process of implementation at each Plant.

4.0 SUMMARY OF METHODOLOGY

Consistent with the Maintenance Rule the methodology consists of the following major elements:

- Identification of the systems, structures and components within the scope of the MR

- Determination of the risk significant systems and standby safety systems.

- Establishing performance criteria for risk significant systems, standby safety systems and the remaining SSCs within the scope.

- Establishing methods for, root cause analysis, functional failures determination, and monitoring and trending of SSCs.

- Consideration of risk significance when taking SSCs out of service for corrective or preventive maintenance, including monitoring.

- Conducting periodic assessment of the plant maintenance programs, for the optimization of availability and reliability and other performance criteria.

5.0 MAIN CONCERNS RELATED WITH THE BASIC STEPS

The discussions and meetings held within the GPE-MR, with the CSN, and the audits to the pilot plants performed by the Regulatory Body, have allowed the detection of discrepancies and divergences of opinion in the interpretation of the different documentation to be applied.

By other hand, there are some results, that due their interest, are shown in the next paragraphs.

5.1 Identification of the systems, structures and components within the scope

- 50% to 60% PLANT SYSTEMS IN SCOPE
  - 150 systems in the plant.
- COULD CAUSE VS DID CAUSE SCRAM/SSA
  - Difficulty in screening BOP systems.
- SCOPING OF SOME NON SAFETY-RELATED SUPPORT SYSTEMS
  - Lightning arrestors
  - Grounding system
  - Vibration monitors, etc.
- DIFFICULTIES IN THE DEFINITION OF SYSTEM LIMITS.
  - Electric yards.
- DEFINITION OF TRAINS, SPARES, REDUNDANCY
  - Lack of definitions.
- HISTORICAL PERFORMANCE OF OTHER PLANTS
  - Difficulty of capture and compare.
- SYSTEMS EXCLUSION.
  - Must be perfectly documented.

5.2 Determination of the risk significant systems and standby safety systems

Engineering maintains a computer model which calculates the potential for core damage based on probability of equipment failure or personnel errors. This computer model, along with a multi-disciplined panel called the" Expert Panel", determined which systems are risk significant.

- TWENTY TO THIRTY SYSTEMS RISK-SIGNIFICANT.
  - Similar in many plants.
- RISK-SIGNIFICANCE AT FUNCTION LEVEL.
  - Additional work to be done.
- DIFFICULTY TO RISK-SIGNIFICANCE DETERMINATION
  - For systems not included in PRA
- NON-RISK SIGNIFICANT VERSUS LOW- RISK
  - Agreed that it has the same meaning.

5.3 Establishing performance criteria

There are two types, **Plant Level** criteria and system, train or component level **specific criteria**.

Maintenance Rule systems are categorized into risk significant systems, standby systems and non-risk significant systems.

In general, risk significant systems and standby systems have specific criteria, while non risk significant systems have plant level performance criteria.

Examples of Plant Level Criteria

No more than 2 trips/ 2 cycles
No more than 5 safety systems actuations/2 cycle.
Less than 5% unplanned capability loss/cycle.

Examples of specific criteria

Reliability: No more than 3 Maintenance Preventable Functional Failures (MEFF) over a 2 cycle rolling period
Availability: No more than 150 unavailability hours over a 2 cycle rolling period.

- USING AVAILABILITY AND RELIABILITY.
  - Must be monitored for all risk-significant systems to facilitate subsequent balancing
- PRA NUMBERS ARE TOO CONSERVATIVE.
  - Must be applied to medium range term
- PERFORMANCE CRITERIA ,AND OPERATING EXPERIENCE.
  - Not easy to document..
- RELIABILITY CRITERIA.
  - Establishment of performance criteria for component families.
- AVAILABILITY CRITERIA.
  - Establishment of performance criteria for functions.
- MONITORING OF STRUCTURES.
  - Needed to develop industry guidance.
  - Further training anticipated.
- UNAVAILABILITY CRITERIA
  - Has to leave good margin for unplanned equipment problems.
- RELIABILITY CAN NOT BE MEASURED AS MPFF
  - It does not depend on the number of demands.

5.4 Compliance with the performance criteria. Root cause analysis and goal setting

- FAILURE VERSUS FUNCTIONAL FAILURE.
  - Address all MPFs, not only MPFFs.
- CHANGE FROM MPFF TO FF.
  - It affects not only maintenance

- MONITORING OF STRUCTURES
  - Monitoring needs to be preventive
- MONITORING OF NON-RISK SIGNIFICANT TRAINS
  - Redundancy can mask poor performance.
- UNAVAILABILITY ACCOUNTING FOR SUPPORT SYSTEMS.
  - Will not be added to the supported systems.
- MPFF IN NON-RISK SIGNIFICANT SYSTEMS.
  - Conduct cause determination after reactor trip or SSA.
  - Consider goal setting if plant level criteria is exceeded

5.5 Considerations when taking SSCs out of service

A new procedure is being developed that will establish controls for evaluating and managing risk associated with removal of SSC's from service to conduct PM or CM work.

Existing and future risk assessment tools will be used to quantity the risk associated with certain plant/system configurations that will be encountered during the conduct of planned activities.

Unavailability and reliability will be closely monitored and controlled over time so that we can decrease the probability that the system would be out-of-service when needed.

Operation, supported by Technical Support, is going to develop detailed, specific guidelines, that will ensure the plant is not placed in adverse configurations.

The scheduling process will ensure adverse configurations are avoided.

5.6 Periodic assessment, optimization of availability and reliability

- A(1), A(2), DETERMINATION.
  - This is the key point.
- DONE PLANT BY PLANT.
  - In a multiple units site.
- BALANCING AVAILABILITY AND RELIABILITY.
  - Not easy to be done.
- SYSTEMS IN A(l).
  - Initially 5 to 10 systems.
- STRUCTURAL MONITORING.
  - Visual inspection, tendon monitoring, engineering walkdown, non-destructive-test.
- MUST BE PERFECTLY DOCUMENTED.
  - Will take long time and effort.

6.0 RESOURCES NEEDED.

IMPLEMENTATION PHASE:

- HISTORICAL DATA COMPILATION AND ANALYSIS.
  - 2 man-year equivalent
- SCOPING + RISK DETERMINATION + PERFORMANCE CRITERIA DEFINITION.
  - Expert panel meetings: 5 to 7 people - 6 months. (PSA, Licensing, Operations, Maintenance, etc.).
- TRAINING.
  - Affect 30%-40% plant personnel. One week course aprox.

MONITORING PHASE:

- MONITORING PERFORMANCE / GOAL SETTING
  - 1/2 man-year. (Maintenance).
- CAUSE DETERMINATION / ROOT CAUSE ANALYSIS.
  - 2 man-year equivalent. (Operations-Maintenance)
- PERIODIC ASSESSMENT.
  - 4 weeks-year. (Expert panel).

7.0 HOW DOES THE RULE IMPACT PLANT OPERATION?

Work control:

- Improves the control related to the amount and duration of clearances.
- Clarifies system and function boundaries.
- Consistent in-house data bases

Maintenance:

- Focuses PM's on critical components, with an overall effect to reduce CM's.
- Improves cause determination and corrective actions.

Operations: Aids in evaluating and reducing plant risk

Technical Sup.: Focuses priorities on SSC's that cause risk impact or effect Maintenance Rule performance criteria.

PRA Group (Engineering): Evaluates risk scenarios and historical data to aid in maintaining suitable risk configurations

8.0 DEFINITIONS

Condition (a)(1) - A Maintenance Rule SSC which has failed to meet performance criteria due to an incorrect maintenance activity. This condition requires evaluation of maintenance activities increased management attention, and the identification of specific performance goals and monitoring.

Condition (a)(2) - A maintenance Rule SSC meeting its performance criteria and having effective maintenance.

Maintenance - The aggregate of those functions requires to preserve or restore safety, reliability and availability or plant structures, systems and components. This term also includes the supporting functions.

Functional Failure (FF) - The failure of a system or train such the system or train is not capable of performing its intended function.

Maintenance Preventable Functional Failure (MPFF) - Failure of a MR SSC to perform its intended function that should have been prevented by the performance of appropriate maintenance actions.

Risk Significant SSCs - Those SSCs that are significant contributors to risk, based on their importance and contributions to nuclear safety as calculated in the Individual Plant Examination (IPE), and as determined by the Expert Panel.

ON-LINE MAINTENANCE AT COFRENTES NPP

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Abstract

Cofrentes NPP (CNPP) has developed a Level 1 PSA with the following scope: analysis of internal events, with the reactor initially operating at power; internal and external flooding risk analysis; internal fire risk analysis; reliability analysis of the containment heat removal and containment isolation systems.

Level 1 CNPP-PSA results reveal that total core damage frequency in CNPP is less than other similar BWR/6 plants.

The CNPP-PSA related activities and applications being carried out currently are: prioritization of motor operated valves related to GL-89/10; complementary analysis for exemption to some 10CFR50 App. J requirements; Q-List grading; risk-informed IST program; reliability-centered maintenance; maintenance rule support; on-line maintenance support; off-line risk-monitor development; PSA applicability to the 10CFR50 App. R requirements, analysis of the frequency of miss-oriented fuel bundle event, adjusting of MAAP 3.0B, revision 10, on VAX and PC; acquisition of MAAP 4; development of Level1/Level2-PSA interface; seismic site categorization for the IPEEE; etc.

INTRODUCTION

Cofrentes Nuclear Power Plant, a GE BWR/6-Mark III of 990 MWe owned by IBERDROLA, is located in Valencia (in South-eastern Spain). It has three electrical divisions, with three emergency diesel generators. Its commercial operation began on May 1985.

In 1989, IBERDROLA began the Cofrentes NPP Level 1 Probabilistic Safety Assessment (PSA) activities, with a large participation of own personnel. The acquired knowledge was useful for improving some punctual aspects of design and procedures concerning generation and maintenance of the plant. Likewise, the availability of probabilistic tools allowed their use on licence and operational support.

In this paper, a global view of IBERDROLA's activities related to Probabilistic Safety Assessment, Individual Plant Examination and risk Applications is outlined with special mention for the On-line Maintenance.

COFRENTES NPP PROBABILISTIC SAFETY ASSESSMENT

The scope of the Cofrentes NPP Level 1 Probabilistic Safety Assessment developed so far covers:

- Analysis of possible scenarios of core damage accidents, with the plant operating at power, caused by internal initiating events.
- Fire risk analysis, with origin inside the plant, and which could lead to reactor trip and mitigation systems degradation through loss of equipment, cables, etc.
- Flooding risk analysis: pipe ruptures that could lead to reactor trip and mitigation systems degradation, as a consequence of water effects.
- External flooding risk analysis, as a consequence of heavy rain and dam rupture upstream from the plant, that could cause degraded safety situations.
- Analysis of sequences that could constitute loss of cooling accidents (LOCAs) in systems that traverse the containment.
- Reliability analysis of containment heat removal and containment isolation systems.

The overall core damage frequency obtained in the internal event analysis is 2.1 E-6/year (Ref. 1). This is lower than the core damage frequency of similar BWR/6 plants like Grand Gulf NS (1.7 E-5/year), River Bend (1.5 E-5/year) and Perry NPP (1.2 E-5/year).

ON-LINE MAINTENANCE APPLICATIONS AT COFRENTES NPP

Safety is the first priority in Nuclear Industry, so it is the requisite that all the optimizations of operations and maintenance must accomplish.

At present, it is normal at Spanish NPPs for only corrective maintenance to be performed during operation at power on systems whose unavailability is limited by the Technical Specifications. Consequently, a large number of work orders is generated for tasks to be performed during refueling outages.

This policy implies an extension of the out of service time for systems during refueling, and may lead to undesirable risk configurations.

During outages thousands of tasks are performed, generally by contractors, with poor planification and supervision of works. By performing tasks at power, with own people and detailed planification, the maintenance is better performed, and it carries to a better reliability of the equipments.

In addition, delays in the performance of certain maintenance tasks may decrease system reliability, especially in those cases where there is a progressive deterioration that might lead to failure.

In many cases it is possible to demonstrate that maintenance of the system may be accomplished at power, assuring that the reliability and the safety of the plant are accomplished or even improved.

Cofrentes NPP is developing an analysis procedure making it possible to determine, in a structured manner, which maintenance activities may be carried out on-line, with the plant at power, optimizing the effectiveness of the maintenance and improving the plant safety.

Comparison between the plant situations existing during power operations and refueling outages shows the following advantages in favour of on-line maintenance:

- Through adequate task preparation it is possible to perform the work in less time than during the refueling outage.
- It is possible to schedule performance of maintenance when the status of the plant is more favourable.
- It allows the attention of all the personnel involved to centre on the performing tasks.
- It allows the most suitable personnel to be selected for each task.
- It improves the application of ALARA criteria by a better scheduling of the tasks.
- It reduces the industrial accident risk by a better preparation of the tasks.

Scope of the Cofrentes NPP On-Line Maintenance Program

System selection begins with the identification of those systems which have the greatest volume of performance tasks during refueling outages.

Initially, it only has been considered, for preventive maintenance at power, those systems with an unavailability allowed by the Technical Specifications longer than 72 hours, before the requirement of the plant outage.

The systems in the scope of the Cofrentes NPP On-Line Maintenance Program, by now, are:

- Low Pressure Core Spray System (LPCS),
- Residual Heat Removal (RHR),
- Essential Compressed Air System,
- Fire Protection System (PCI),
- MSIV´s Leakage Control System
- Stand-by Gas Treatment System (SGTS),

- Control Room HVAC and,
- Drywell/Containment Atmosphere Mixture System.

Feasibility analysis for performing the On-Line Maintenance

It is performed a system or train level analysis with the next phases:

- Qualitative justification that the safety functions are accomplished during the Limiting Condition for Operation (LCO) and that exists sufficient capabilities for mitigating the consequences of an accident or a transient.
- For the systems included in the Probabilistic Safety Assessment, an evaluation, by means of the PSA models, of the increased risk arising as a result of a system/train being left out of service for the maximum time established in the limiting conditions for operation (LCOs).

  First, a standard configuration with the plant in power operation is evaluated. If the risk is not negligible, a more detailed analysis will be required, justifying that the risk could be lowered by reducing the unavailability or setting a compensatory program.

  The criteria for accepting the increase in the risk are the ones established by the EPRI-"Probabilistic Safety Assessment Applications Guide" (Ref. 5).
- Verification that there is enough time available for the performance of the tasks during power operations.
- Confirmation that inoperability of the system or train does not imply a significant increase in the risk of a trip or of the non-desirable actuation of an emergency system.
- Confirmation that the working conditions in the plant areas where the components of the system or train are located allow the work to be performed during power operation (ALARA).

If any of the conditions analysed are not entirely satisfactory, contingency plans containing specific actions will be studied, and the analyses will be repeated under the new conditions.

Control of simultaneous system unavailabilities. Risk Configuration Matrix

In order to control the risk associated to the on-line maintenance tasks, Cofrentes NPP has set, in a matrix format, the configurations of simultaneous system unavailabilities. This matrix will not let certain configurations to be performed.

This matrix sets the system availability requirements to take into account before and during the on-line maintenance works, in order to avoid:

- Configurations forbidden by the Technical Specifications which could require the immediate plant outage or could be allowed during short periods of time (<72 hours, based on criteria of defence in depth).

- Configurations allowed by the Technical Specifications that could lead to risk significant scenarios (by means of Probabilistic Safety Assessment risk analysis using the EPRI TR-105396 (Ref. 5) quantitative criteria).

Limitations to be taken into account when scheduling the On-Line Performance Works

Keeping the NRC (Nuclear Regulatory Commission) recommendations the following limitations are proposed:

- Basically, no on-line maintenance tasks will be performed in more than one system/train at the same time. In some cases, with the goal to reduce the total unavailability of front systems, it could be convenient to combine works with their supports systems.

- The tasks must be completed preferably with a single entry into each limiting condition for operation. All the tasks to be performed on the component, system or train should be grouped.

- The annual accumulated unavailability of the train or system due to preventive (< 60% AOT- Allowed Outage Time) and corrective maintenance should not exceed the Allowed Outage Time by the Technical Specifications.

- The time involved in performing the work within a Limiting Condition for Operation (LCO) should be minimized by an accurate task preparation and with a non-stop performance of the works.

Verification of the requirements to consider before starting the On-line Maintenance Works

The objective of this task is to ensure that an acceptable level of risk is maintained during performance of the work, within a previously analysed plant configuration, and that alternative/redundant systems capable of responding to the unforeseen conditions exist and are identified.

In this respect, before getting into a Limiting Condition for Operation (LCO) it will be verified that:

- All the systems (redundant and alternative) required by the configuration matrix are available.

- The plant configuration is the same that the one quantified by the risk analysis (with the same electric and mechanical alignments. etc).

- No other simultaneous work/test will be scheduled which could increase the unavailability or the risk of trip of systems in the configuration matrix.

- A modification of the plant mode will not be scheduled during the Limiting Condition for Operation (LCO).

- If the risk analysis has a compensatory program, this will have to be prepared.

- The preparation of the tasks will be reviewed for assuring that the maintenance can be completed in the scheduled time (tools availability, stocks, required and trained personnel. etc.)

- No extreme meteorological conditions are expected (not modelled in the Probabilistic Safety Assessment).

Prior to initiating the work, a verification should be performed to ensure that all the checks have been performed and that the tasks have been prepared, and management approval will be obtained.

On completion, the process will be evaluated to identify potential areas for improvement.

CONCLUSIONS

Cofrentes NPP is decided to apply the Probabilistic Safety Assessment to improve operational activities like maintenance, QA, inspections, Technical Specifications, etc. using the risk-informed approach. The expected benefits can be summarized as follows:

On-Line Maintenance

- It improves system availabilities. More human resources. better schedules, improvement of the maintenance performance.

- It reduces works to perform in the refueling outage, therefore can reduce the outage time and improve the safety during outages.

- It controls certain configurations, currently allowed by the Technical Specifications. which may lead to plant damage.

- It reduces the risk of industrial accidents by a better task scheduling.

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THE SAFETY MONITOR AND RCM WORKSTATION AS COMPLEMENTARY TOOLS IN RISK BASED MAINTENANCE OPTIMIZATION

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Abstract

Reliability Centred Maintenance (RCM) represents a proven technique for rendering maintenance activities safer, more effective, and less expensive, in terms of systems unavailability and resource management. However, it is believed that RCM can be enhanced by the additional consideration of operational plant risk. This paper discusses how two computer-based tools, i.e., the RCM Workstation and the Safety Monitor, can complement each other in helping to create a living preventive maintenance strategy.

1. Introduction

In the United States, the Nuclear Regulatory Commission have introduced the Maintenance Rule to encourage the raising of the minimum levels of systems availability, and the reduction of operational plant risk. Reliability Centred Maintenance (RCM) represents a proven technique for rendering maintenance activities safer, more effective, and less expensive, in terms of systems unavailability and resource management. However, in order to comply with the principles inherent in the Maintenance Rule, it is necessary to ensure that maintenance strategies become risk-aware. This paper describes the way in which two computer-based approaches can complement each other in achieving this goal effectively.

It is not the intention here exhaustively to describe the RCM procedure. The emphasis is on the way in which this procedure is enhanced by the additional consideration of operational plant risk, and on the need to introduce the Safety Monitor (Ref. 2) as a method of evaluating this risk in a controlled and informative manner. In this way, these two complementary tools for addressing maintenance practices and requirements are shown to be highly effective in helping to create a true living Preventive Maintenance Strategy.

At present, with the express objective of enhancing the way in which the USNRC Maintenance Rule is applied and tracked, installation of the Safety Monitor is almost complete in the United States at Wolf Creek, Callaway and Comanche Peak. In addition, the Safety Monitor has been in continuous use at San Onofre since 1994, for which it won the Top Industry Practice Award in 1996. Installation is also in the initial stages at six other power stations in the United States, and is nearing completion at Temelin in the Czech Republic.

2. Reliability Centred Maintenance

The RCM process first identifies the failed states of a system, or ways in which a system can fail to live up to its expectations. This is followed by a Failure Modes and Effects Analysis

(FMEA), to identify all the events that are likely to cause each failed state. Finally, the RCM process seeks to identify a suitable failure management policy for dealing with each failure mode, given its consequences and technical characteristics. Failure management policy might typically include:

- Preventive maintenance
- Failure-finding
- Changing the design and/or configuration of the system
- Changing the way in which the system is operated

The RCM Workstation (Ref. 1) provides an efficient means of developing powerful rules for deciding whether any failure management policy is technically appropriate. In addition, it enables the development of precise criteria for deciding how frequently routine tasks should be performed. The decisions taken in this process take into account the actual operating environment experienced by equipment, and the functional performance requirements of individual systems and components.

2.1 What does RCM achieve?

RCM does far more than simply produce new maintenance schedules. Applied correctly, RCM achieves all the main objectives of maintenance by improving:

Safety and environmental integrity through:

- Systematic review of the safety and environmental implications of every failure
- Clear strategies for preventing failures that can affect safety and the environment
- Improved maintenance of protective devices
- Widespread understanding of the importance of protective systems
- Provision of new protective devices
- Fewer failures caused by unnecessary maintenance

Operating performance through:

- Greater emphasis on the maintenance of critical items
- Extended maintenance intervals, and in some cases the complete elimination of invasive maintenance
- Shorter shutdown work schedules leading to shorter, less costly and more easily managed shutdowns
- Fewer maintenance-induced problems after shutdowns
- The elimination of unreliable components
- Quicker fault diagnosis
- Fewer failures caused by operator errors

and by helping to ensure cost effectiveness, longer asset life, and better teamwork and motivation.

Using the RCM Workstation periodically to reassess the maintenance requirements of existing equipment, RCM transforms maintenance schedules and the way the maintenance function as a whole is perceived in the organisation. Such reassessments would be carried out taking account of actual operational experience, and benefitting from feedback concerning equipment and systems unavailability. The result is maintenance that is safer, more effective, less expensive, and carried out with the agreement of production and maintenance personnel.

The next section describes the way in which the feedback obtained from operational experience is enhanced and complemented by the consideration of operational risk.

3. The Safety Monitor and Risk-informed Feedback

The requirement to perform preventive maintenance (PM) during on-line plant operation receives increasing attention as fuel cycles are extended. This fact lends weight to the argument, in practical terms, that PM becomes risk-aware for maximum effectiveness. By constructing a real-time model for predicting the risk of day to day operation of the plant, it becomes possible to generate information that can be used to evaluate the effectiveness of adopted maintenance strategies and schedules. Armed with this knowledge, the RCM procedure can be iterated so that the principal safety, reliability and production objectives of plant operation are progressively optimised.

Naturally, the more realistic the risk model, the greater the confidence in its use for this, or for any other purpose. With the advent of yet more capable high speed solution algorithms, aided by ever increasing computer processing speeds, the requirement to optimise the full PRA models is relaxed. This retains compatibility between the results of risk calculations for the full models and for the optimised models constructed for evaluation by the Safety Monitor. The result is that information of the highest integrity is available for feeding back into the RCM procedure.

Typically, maintenance scheduling covers periods of several months, involving a wide range of extensive activities with highly complex interactions. The risk implications of such activities cannot be envisaged without the aid of a capable risk calculation methodology, and even with such a technique available, the task of compiling the risk profile over the planned maintenance period can be formidable. Worse still, the day to day activities on the plant, involving small but necessary changes to the schedule, and encountering random equipment failures, can cause severe problems in determining the effectiveness of the schedule carried out, as opposed to the schedule that was planned. It is, of course, important to ensure that such variations are adequately accounted for in assessing the quality of the feedback information.

The next section describes the advanced features of the Safety Monitor that have been developed both to solve these problems, and more generally to enhance the range and quality of information that can be derived from its use in monitoring operation of the plant.

3.1 The Safety Monitor in Action

One of the features of the Safety Monitor that makes it unique amongst risk calculation tools is its ability to estimate plant risk by completely solving the plant risk model for every risk calculation. Pre-solved cut set lists are NOT used, and this removes problems associated with numerical inaccuracies being introduced through premature truncation. The full PRA model, expanded to include all anticipated operating alignments, is calculated giving the most accurate point risk estimates available. In addition to the calculation of Core Damage

Frequency (CDF), the Safety Monitor also calculates the frequency of Large Early Release (LERF) to account for containment systems that are important to risk, even though they do not directly affect the CDF. To complete the flexibility of the risk monitoring capability, the Safety Monitor can also handle shutdown modes of operation, these being integrated with the full power mode in a single generic model, modified seamlessly as the need arises.

Originally. the Safety Monitor was designed to enable single calculations of point estimated risk for real or hypothetical plant status and configuration. In Version 2.0 (Ref. 3), this has been extensively revised to enable calculations of entire schedules of proposed activities. and to generate a time-based risk profile for each such calculation. In this way, the maintenance schedules derived through the RCM procedure can be tested and refined before receiving final approval.

Periodically, the risk profile of the real operating plant may be reviewed to determine its accuracy with regard to the actual plant operating conditions, as compared with what might have been known at the time of the calculations. In this way, revisions to the conservative assumptions that might have been made in real time can be applied. and the risk profile for any period in the plant history recalculated in a single step. Such reviews would then generate the feedback information that would be taken into the next application of the RCM procedure for further enhancement of the proposed maintenance schedules.

4. Conclusion

This paper has shown that the RCM process, given the additional requirement to incorporate risk awareness. requires the complementary use of a suitable risk calculation methodology. The complexity of the tasks associated with the RCM procedure can be greatly assisted through the use of a software tool that formalises and organises the decision making process. of which the most capable example is the RCM Workstation. The Safety Monitor represents the most advanced tool of its kind. and provides the best means available of complementing the RCM Workstation in its application for the development of progressively optimised, risk-based maintenance procedures and schedules.

As real experience of the use of the Safety Monitor, at an ever increasing number of locations. develops, it becomes more and more evident that the greatest benefit is afforded to Maintenance Planning Personnel. The questions most frequently asked of the Safety Monitor are of the "What if?" type, as the maintenance planners focus upon obtaining optimal proposed maintenance schedules. This is. of course, to be expected if we assume a responsible approach to the management of operational risk, and is concrete evidence of the major role that the Safety Monitor can play, in conjunction with the RCM Workstation to develop a closed cycle of activities designed to optimise maintenance on a fully risk-informed basis.

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RISK BASED MAINTENANCE TO INCREASE SAFETY AND DECREASE COSTS

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Abstract

Risk-Based techniques have been developed for commercial nuclear power plants for the last eight years by a team working through the ASME Center for Research and Technology Development (CRTD). System boundaries and success criteria is defined using the Probabilistic Risk Analysis or Probabilistic Safety Analysis developed to meet the Individual Plant Evaluation. Final ranking of components is by a plant expert panel similar to the one developed for the Maintenance Rule. Components are identified as being high risk-significant or low risk-significant. Maintenance and resources are focused on those components that have the highest risk-significance. The techniques have been developed and applied at a number of plants. Results from the first risk-based inspection pilot plant indicates safety due to pipe failure can be doubled while the inspection reduced to about 80% when compared with current inspection programs. Pilot studies on risk-based testing indicate that about 60% of pumps and 25 to 30% of valves in plants are high safety-significant The reduction in inspection and testing reduces the person-rem exposure and resulting in further increases in safety. These techniques have been documented in publications by the ASME CRTD which are referenced.

Introduction

Risk-Based In-Service Inspection (ISI) and In-Service Testing (IST) methods have been under development within the ASME Center for Research and Technology Development for about eight years. A series of documents have been written by a multi-disciplinary ASME research task force and published by the ASME. These documents define a four-part process for managing the inspection and testing of nuclear power plant components.

Risk-Based Process

The four major elements of the process are:

1. Definition of system boundaries and success criteria using a plant probabilistic risk assessment (PRA) or probabilistic safety assessment (PSA) that has been developed to meet the Individual Plant Examination (IPE) and Maintenance Rule requirements of the U.S. Nuclear Regulatory Commission,

2. Ranking of components or piping segments by a plant expert panel that makes the final selection of where to focus ISI or IST resources by considering risk importance measures, consequences of failures, and other deterministic measures,

3. Determination of effective ISI or IST programs that define when and how to appropriately inspect or test the two categories of more-safety-significant or less-safety-significant components, and,

4. Performing the ISI or IST program to verify component reliability and then updating the risk rankings based on the inspection or test results.

RBI Pilot Studies

Pilot tests of the risk-based ISI methodology have been accomplished. A major study has been completed at Millstone-3 Power Station by Northeast Utilities with support from the Westinghouse Owners Group (WOG) and Westinghouse. The results from this effort have been forwarded to the U.S. Nuclear Regulatory Commission via the Nuclear Energy Institute (NEI) as a Westinghouse Owners Group topical report, WCAP-14572 (1996). This work adapted the ASME research methods in order to accomplish this full scale study.

A project is underway to perform a verification and validation (V&V) of the risk-based process through industry and NRC participation in an ASME research project. This verification and validation project uses Virginia Power's Surry plant for the evaluation. The use of Surry is significant because of the extensive initial risk-based ISI work performed there under previous research efforts: a favorable comparison of those previous results with those produced by the enhanced process is anticipated to assist the acceptance of the process by the NRC for generic industry use.

Millstone-3 was selected for the pilot study because of the support of the WOG and Westinghouse. Surry was selected for the V&V effort because of the previous research efforts performed there. Although it was not a consideration in the selection process, the fact is that both of the studies were conducted on Pressurized Water Reactors (PWRs). An application of the developed technique to a Boiling Water Reactor (BWR) has not been attempted. An application study has been initiated at the Browns Ferry Plant to addresses plant type differences.

Application Study of RBI at a BWR

Basic differences between PWRs and BWRs that would affect the risk-based process exist in several areas:

- Some of the more safety-significant systems on a BWR (RCIC for instance) are currently exempted from Section XI requirements based on size; therefore, scope is different.

- BWRs have a PSA Core Damage Frequency that can be as much as an order of magnitude less than a PWR. The amount of CDF attributable to piping failures could be such a small number as to be considered below the cut-off point for significance.

- BWR chemistry and pressure have potential impact on Structural Reliability and Risk Analysis.

- BWRs are subject to different significant failure mechanisms than PWRs: for instance. IGSCC.

Another basic difference that affects BWRs is the applicability of Generic Letter 88-01. which defines requirements for IGSCC programs. A risk-based selection technique could potentially optimize the inspections performed under these programs.

The Browns Ferry project will assist industry in the validation of the ASME Research risk-based in-service inspection approach on a BWR. Currently, the NRC is developing a draft Standard Review Plan and Regulatory Guide to be submitted for public comment. A pilot application of the technique at a BWR will provide valuable insight to assure the SRP and Regulatory Guide are applicable to all the major reactor types. The ASME is also seeking individuals that will help support this project.

The project is being performed with a team from ASME Research, Tennessee Valley Authority, and other industry participants. This team would apply the current ASME Research Risk-Based In-Service Inspection approach to a BWR plant and compare the results to the previous pilot studies. Comparisons would be made along the way and any technical issues would be resolved during the course of the project. The project started in July 1997 and is scheduled to be completed in mid-1998.

Benefits of the BWR RBI Program

Risk-based inspection program development has benefits for the industry, BWR owners, the owners of the plant being studied (Tennessee Valley Authority in this case.), the NRC. and the Code writing Body. These benefits are as follows:

a. Industry
   - Assure applicability of the approach to all major reactor types.
   - Provide a better understanding of the risk-based ISI technology.
   - Increase success of risk-based in-service inspections.

b. BWR owners
   - Provide a documented basis for potential optimization of the inspection process mandated by Generic Letter 88-01

c. Tennessee Valley Authority
   - Provide a risk-based ISI program for the plant based upon an updated IPE model.
   - Lead to earlier consideration for program approval.

d. ASME Code
   - Provide support for Code changes.

e. NRC
   - Provide insights to the applicability to all major reactor types of the techniques outlined in their draft SRP and Regulatory Guide.
   - Provide insights to the potential optimization of the inspection process mandated by Generic Letter 88-01 for inclusion in their draft SRP and Regulatory Guide.

e. All parties
   - Provide a mechanism to resolve issues in a nonregulatory setting (ASME Research-CRTD) as they occur during the process (i.e., reduce review cycle time).

Results of the RBI Pilot Studies

The Risk-Based ISI project at Millstone-3 has been completed using the ASME Research methodology described in WCAP-14572. A total of 119 elements have been selected for some type of examination under the Risk-Based ISI program as compared to 753 welds now scheduled under the current ASME Section XI program, representing an 84% reduction in the raw number of examinations to be performed. In addition, examination of the current ASME Code locations addresses 44% of the Core Damage Frequency attributable to piping, while examination of the Risk-Based elements addresses 98%, representing a 122% improvement. Although total Core Damage Frequency attributable to piping is a small fraction of the total plant CDF, safety is enhanced with fewer examinations being performed. While the Surry pilot has not been completed, it is estimated that the number of Risk-Based examinations will be approximately 40% of the number now scheduled under the current Section XI program.

In economic analysis, these pilots represent a direct cost savings of 60-84% of the current costs of examination per outage. Additionally, Millstone-3 estimated an exposure savings of 15 man-rem each outage. Other indirect cost savings are expected from items such as reduction in costs associated with evaluating flaw indications which may not really exist (i.e., false calls).

Results are not available for the Surry Verification and Validation Project, but, a reduction in inspection of 60% is expected.

These results indicate that a risk-based program can be successful in greatly reducing costs, both dollars and exposure, while improving safety; however, they have only been done on PWR nuclear steam supply systems. Validation of the process on a BWR in the Browns Ferry Application also has the potential to provide a path for optimization of the inspection process mandated by Generic Letter 88-01, and as such makes this a worthwhile project.

The objective of the program is to further validate the ASME Research Risk-Based In-Service Inspection approach when applied at a Boiling Water Reactor.

Advantages of the Quantitative RBI Approach

The advantages of the ASME Research Risk-Based In-Service Inspection quantitative approach when compared with the less effective qualitative approaches are:

- Provide a quantitative approach to measure risk reduction
- Provide risk trade off--active components can be inspected or operation changes can be made to take the place of inspections
- End the subjective percentage of components inspection criteria
- Probabilistic Fracture Mechanics Calculations for inspection and frequency evaluation

- Augment the generic data and plant specific sources
- Project failure probability into the future to evaluate conditions that have not occurred.

The quantitative approach to risk-based inspection should be as efficient and should be no more costly than the qualitative approaches which do not offer the advantages.

Risk-Based Testing

The recommend process applying risk-based methods to inservice testing of active components in nuclear plant systems (Reference 1 ASME RBT Book) is centered on three major areas. These categories are ranking of component importance, development of the inservice testing program, and implementation of the testing program.

Ranking of Component Importance. Components are ranked in two groups, those that are high safety-significant and those that are less safety-significant. The identification of risk/safety ranking of IST components involves the application of the plant PRA to the IST populations of components. Risk ranking is accomplished using the Fussell-Vesely (FV) measure for the core damage frequency end state. The FV threshold value used for this risk-rankings is 0.001. Components that exceed this value are initially considered risk-significant, those below it, less-risk-significant. Components not modeled or truncated out of the PRA model results are initially placed in the latter group.

Several quantitative mapping and sensitivity studies are performed to determine the effect of large, early-release frequency, the risk achievement worth measure, the risk-reduction worth measure, and external events on the IST population of components, as required. The quantitative analysis is blended with the deterministic analysis in the form of a plant expert panel.

Development of Inservice Testing Programs for High Safety Significant Components. Components are evaluated using a component IST team of plant experts (and other personnel, as required) that considers the following steps.

(1) Component Review and Failure Modes and Causes Analysis. Key characteristics are identified that could influence the determination of effective testing methods(component type, design features, configuration, application, service duty, component age, industry experience, and plant specific experience). Potential failure causes are identified for the failure modes shown to be critical from the component importance ranking process. Results of data bases for component failure are useful to identify the ways components actually fail in service.

(2) Test Effectiveness Assessment. A qualitative or quantitative assessment of the effectiveness of each test, based on the components ability, is performed to detect a failure and any significant conditions that are a precursor to failure. An assessment of a level of confidence in the testing methods is also made to determine whether the component will function correctly should a real demand occur at any time during the operation interval before the next test.

(3) Strategy Formulation and Evaluation. For each component, a definition of some schedule of tests is made. An assessment of a level of confidence for each strategy and an evaluation of the value-impact for the various IST strategies, in terms of core damage frequency and testing costs (direct costs and person-rem exposure) are also performed. An appropriate strategy is terms of safety and cost is selected from these results for each. component of interest.

Implementation. The above two steps are used for implementation of the IST program and feedback of the IST program results into the prior steps of the process. The effect of this revised IST program must be predicted in order to ensure that a program that could have an adverse effect on safety is not implemented. The combined quantitative impact of these changes must be assessed by requantifying the base PRA used for risk ranking. The potential for initiating test-related transients at the plant and the effect of taking equipment out of service should also be considered in determining the overall effect of testing on the performance of system safety functions.

Periodic assessments should be performed to establish the effectiveness of the IST program and to feed back changes to the prior steps of the process. The IST program review could be addressed in conjunction with periodic updating of the plant PRA, industry operating experience programs, and the Maintenance Rule program.

Each of the above areas includes equipment performance considerations using both plant-specific and industry experience, as appropriate. The process can be integrated with other risk-based and performance-based applications (e.g. the Maintenance Rule). While the focus of the process is on safety/risk, risk-based IST can also minimize plant investment and risk.

RBT pilot studies have been completed or are in progress at the Idaho National Engineering Laborites, Shearon Harris Nuclear Power Plant, the Palo Verde Nuclear Power Plant and Commanche Peaks Nuclear power Plants. These pilot studies indicate that about 60% of the pumps and 25 to 30% of the valves are high safety significant.

Conclusions

Techniques have been develop that focus scarce resources on components that most affect risk. Risk-based approaches focus maintenance activities on components where failures can occur and have high consequences at plants. Results indicate that safety can be increased and inspection of piping components and testing of active components can be decreased. These techniques have been developed by teams working with the ASME Center for Research and Technology Development. The results of this work is published in documents which have been referenced.

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RISK BASED MAINTENANCE: RESOURCE REQUIREMENTS AND ORGANIZATIONAL CHALLENGES

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Abstract

10 CFR 50.65 "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" required licensees to monitor the performance or condition of structures, systems, or components (SSCs) against licensee established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. The goals were required to be commensurate with safety significance and operating experience. Northeast Utilities relied upon PRAs to implement 10CFR 50.65, which is also referred to as the "Maintenance Rule." The Maintenance Rule changed some aspects of maintenance of structures, systems, and components (SSC) at nuclear power plants. One objective of the rule was to focus the maintenance resources based on risk significance of components. This paper will discuss the organizational challenges and resource requirements associated with implementation of the Maintenance Rule at nuclear facilities that are supported by the Northeast Utilities Services Company (NUSCo). The paper will discuss (a) how these challenges were addressed, (b) the resources required for ongoing efforts to support the Maintenance Rule, and (c) several key safety benefits derived from the implementation of the Maintenance Rule.

A. PURPOSE

10 CFR 50.65 "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" required licensees to monitor the performance or condition of structures. systems, or components (SSCs) against licensee established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. The goals were required to be commensurate with safety significance and operating experience. Northeast Utilities relied upon PRAs to implement 10CFR 50.65, which is also referred to as the "Maintenance Rule." The Maintenance Rule changed some aspects of maintenance of structures, systems, and components (SSC) at nuclear power plants. One objective of the rule was to focus the maintenance resources based on risk significance of components. This paper will discuss the organizational challenges and resource requirements associated with implementation of the Maintenance Rule at nuclear facilities that are supported by the Northeast Utilities Services Company (NUSCo). The paper will discuss (a) how these challenges were addressed, (b) the resources required for ongoing efforts to support the Maintenance Rule, and (c) several key safety benefits derived from the implementation of the Maintenance Rule.

B. DEPARTMENTS SUPPORTING SAFETY RELATED MAINTENANCE

The Maintenance Rule was published in July 1991. The licensees were given five years to fully implement this rule. The rule implementation required interfaces among several functions within the licensees to work closely. They are:

- System Engineers
- PRA Engineers
- Operations
- Work Planning and Outage Management (WP&OM)
- Unit Maintenance Rule Coordinator
- Expert Panel

At NUSCo. the plant system engineers have ownership of specific plant systems. They are cognizant of day-to-day activities associated with those systems. The system engineers are located onsite and they are expected to possess an in-depth understanding of operating history and characteristics of the systems which they take ownership. At NU, the PRA engineers support all of the nuclear units from a central office. The Operations Department has the primary responsibility for the control of the operating equipment. The WP&OM Department is responsible for planning and scheduling test and maintenance activities. The unit Maintenance Rule Coordinator is a position created to support the implementation of the rule. Finally, an Expert Panel was established to support the Maintenance Rule. This panel consisted of at least five members with representatives from Operations, PRA, Plant Engineering, Safety Analysis (Design Basis Analysis), and Work Management. In addition to this, during the Maintenance Rule initial development phase, a Maintenance Rule Working Group supported the resolution of issues and maintained consistency among the nuclear units.

C. PRA's ROLE AND INTERFACES WITH OTHER DEPARTMENTS

PRA's primary role in implementation of the Maintenance Rule are represented by its contribution to the following tasks:

- Determination of Risk-Significance
- Participation in the Expert Panel
- Development and Review of Unavailability Performance Criteria
- Development and Review of Reliability Performance Criteria, and
- Risk Assessment of Work Schedule

In order to support each of the above tasks, PRA interacted with other function within the organizations. While task completion (e.g, Development of a list of Risk-Significant systems) was necessary for initial implementation, the long-term success relies upon an effective exchange of information between PRA engineer and the interfacing functions. Management endorsement of PRA training activities is a key contributor to the long-term success of the Maintenance Rule implementation.

C.1 System Engineering and PRA

Prior to the Maintenance Rule, there was no regulatory requirement to motivate the system engineers to understand the reliability models (Fault trees) associated with their systems.

However, the Maintenance Rule required the system engineers to gain an understanding of the reliability models and the risk significance of the systems that they own. Management support was obtained to transfer the PRA knowledge to the system engineers. Information exchange sessions between PRA engineers and system engineers enhance the system engineers' understanding of the reliability aspects of their systems. These information exchanges also benefit the PRA engineers by enhancing their understanding of system operations.

C.2 Operations and PRA

Prior to the Maintenance Rule, PRA had provided periodic training to the operators primarily focused on keeping them informed of dominant core-damage sequences and risk-significant operator actions. With the Maintenance Rule, the focus of operator training shifted to understanding the risk associated with removing equipment from service and the monitoring process implemented to manage risk. The need for this training had to be communicated to the Training Department which is responsible for scheduling operator training. Even though the risk management concept provided to the operators was in some respect additional requirements which operators needed to follow, there was near unanimous agreement on its value and contribution to public health and safety. The credibility of the risk assessment and monitoring principles (e.g, prohibition to perform high risk test with key decay heat removal system trains out of service) led to full acceptance by the operators.

C.3 WP&OM and PRA

10CFR 50.65 (Maintenance Rule) requires the licensee to assess the overall risk impact associated with on-line maintenance. The PRA function provides the expertise to assess this overall impact. However, prior to Risk Monitoring and Maintenance rule, PRA had almost no experience or involvement in the area of Work Planning. Similarly, prior to the Maintenance Rule the work planners had almost no experience with the concept of risk assessment. In order to meet part (a)(3) of the Maintenance Rule, PRA and WP&OM had to start working together and share information. WP&OM personnel had to be informed of the subset of systems and tests which are of interest to PRA. WP&OM had to expend resources to accommodate PRA information needs. They had to summarize the detailed work plans so that the PRA engineers could interpret the schedules. There was a significant resource burden on the PRA since the weekly work schedules had to be reviewed in order to provide timely feedback to the units.

D. RESOURCE REQUIREMENTS

D.1 PRA Resource Requirements

During the initial implementation phase, the PRA engineers provide critical input to the following activities: Risk Significance determination and Performance Criteria development. In addition, PRA engineers participated in the development of the risk management process by generating risk matrices, reviewing the work planner's schedules, and transferring the risk monitoring knowledge and principles to operators. As PRAs were periodically updated, some of these activities had to be repeated.

While the Rule was put into place between 1990 and 1996, PRA resources per nuclear unit was approximately 6 person-months (PM) per unit per year. Even though the rule is fully in place, PRA spends significant resources to adjust methods and processes based on lessons learned. The ongoing PRA support for the rule include the following tasks:

(i) Periodically update PRA Models
(ii) Review risk significance and performance criteria decisions when PRA models are updated
(iii) Participate in Expert Panels
(iv) Provide periodic training to System Engineers, Operators, and Work Planners, and
(v) Review maintenance and test schedules to support risk management process

Item (i) above requires significant resources. However, those resources are required to support all PRA related efforts. Item (v) can be resource intensive unless proper state-of-the-art tools (computer software) are used. Items (ii)-(v) can be supported with about 3 PM/unit/year.

D.2 System Engineering Resource Requirements

During the implementation stage of the Maintenance Rule, the system engineers expended a significant amount of resources to perform a multitude of tasks such as examining the past operating history, preparing system basis documents, establishing performance criteria, learning the Rule, and failure definitions. The unit Maintenance Rule coordinator provides extensive support to the system engineers during this phase The total effort to support these activities were estimated at 6PM/Unit/Year. Now that the Rule is fully implemented, the system engineer is expected to continue to track and trend the system performance. If the system is a bad performer (referred to as an (a)(1) system in the Maintenance Rule space), then additional ongoing support is needed for goals setting and monitoring. If one assumes, 20 systems at 2 hours per month per system, the ongoing resource burden is approximately 3 person-months/unit/year.

D.3 Operations Resources

The operators need to be cognizant of risk-management guidance. In order to accomplish this, additional training was incorporated into the operator training curriculum. If 1 hr/operator/year is estimated as the training burden to the operators, assuming 60 operators/unit, the total ongoing burden is approximately 0.5 person-month/Unit/Year.

D.4 Expert Panel

The resource requirements to support the expert panel are estimated as follows. During the Maintenance Rule implementation phase, the Expert Panel met approximately once a week. The panel consists of at least 5 people. If each panel meeting has a 3 hour burden per participant and one assumes 6 participants per meeting, this equates to approximately 6 PM/unit/year. For on-going rule compliance, it is likely the expert panel will only meet monthly or quarterly. The Expert Panel burden is estimated to be about 2 PM/year/Unit.

D.5 Unit Maintenance Rule Coordinator

From 1991 to 1996, until the Maintenance Rule was fully in place, a new full time position was created to support all Maintenance Rule related tasks. In addition, 4 full-time contractors and a program manager provided support for 2 years to support 4 nuclear plants. That is, during this phase, the resources expended was approximately 18 person-months/Unit/Year. However, now that the rule fully implemented, the Maintenance Rule Coordinator will likely be able to assume other responsibilities as well. Assuming a 50% reduction, the resource needs to support the ongoing efforts associated with the Maintenance Rule by the unit coordinator is estimated to be 9 PM/Unit/Year.

D.6 WP&OM

In order to meet paragraph (a)(3) of Maintenance Rule to assess risk when removing equipment from service, NUSCo relies upon WP&OM. During the implementation stage, the PRA and WP&OM had to learn the capabilities of their respective disciplines. PRA had to understand the processes and procedures applicable to work planning and identify their information needs to WP&OM. WP&OM had generate the information that was needed by PRA in a format that was usable by PRA. The implementation burden associated with WP&OM is estimated at 1 PM/Unit/Year based on a weekly work load of 3 hours for WP&OM to produce the necessary documentation for PRA review. In addition, 1 PM/Unit/Year was added to accommodate the learning curve. Since the WP&OM and PRA information exchange has to continue on a daily basis, the WP&OM resource will continue to be needed, however, at a slightly reduced levels.

D.7 Total Resources

Function	**Resources needed to Support Implementation of Rule (Person-months per unit per year)**	**Resources to support Ongoing Tasks (Person-months per unit per year**
PRA	6	3
Expert Panel	6	2
Maintenance Rule Coordinator	18	9
System Engineers	6	3
WP&OM	2	1
Operations	0.5	0.5
Total	**38.5**	**18.5**

E. SIGNIFICANT SAFETY BENEFITS

Maintenance Rule implementation has resulted in reductions in risk associated with nuclear plant operations. Perhaps the most profound safety benefits will be gained from risk assessment and monitoring when removing equipment from service. The Technical Specifications did not always prevent the operators from entering high risk configurations. However, the shortcomings of the Technical Specifications in this area were eliminated by the implementation of risk assessment processes. The Maintenance Rule highlighted the need to control unavailability for several critical safety systems. In addition to meeting the specific objectives associated with the Maintenance Rule, the activities associated with the Maintenance Rule set a strong foundation for a future risk-informed safety culture at nuclear utilities.

F. CONCLUSIONS

Implementation of the Maintenance Rule, 10CFR 50.65, at nuclear plants has many organizational as well as technical challenges. The organizational challenges had to be overcome by generating appropriate procedures and processes that requires interactions among different organizations and processes. In the short-term, the ability to create these processes and procedures relied upon management support. The long-term success will rely on continuing training and information exchanges between functions (e.g, work planning and PRA, system engineering and PRA). Since the Maintenance Rule is performance based rather than prescriptive, technical challenges were addressed by generating internal guidelines. In addition to providing the benefits of improved maintenance, the Maintenance Rule has set a strong foundation for future risk-informed regulations. The implementation burden was extensive at the beginning. However, if the state-of-the-art tools are used, the ongoing burden minimized.

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Technical Committee Meeting
Vienna, Austria: 15–19 September 1997