Unclassified **NEA/CSNI/R(2002)2**

Organisation de Coopération et de Développement Economiques
Organisation for Economic Co-operation and Development

28-May-2002

English text only

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2002)2
Unclassified

SAFETY PERFORMANCE INDICATORS - WORKSHOP PROCEEDINGS

17-19 October 2000, Madrid, Spain

This document only exists in PDF format.

JT00127001

English text only

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14th December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concerned with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

* * * * * * * * * * * *

Requests for additional copies of this report should be addressed to:

Nuclear Safety Division
OECD Nuclear Energy Agency
Le Seine St-Germain
12 blvd. des Iles
92130 Issy-les-Moulineaux
France

Specialist Meeting on Safety Performance Indicators
Madrid, Spain, October 17-19 2000

INDEX

PROGRAMME OVERVIEW

PRESENTATIONS:

UTILITY INDICATORS
Chair: S. Floyd (NEI, USA) / Jürgen Schlegel (WANO)

- U.S. Industry Perspectives on Role of Indicators in the Regulatory Process – *S. Floyd (NEI, USA)*
- Performance Indicators in the USNRC's Revised Reactor Oversight Process – *D. Hickman (NRC, USA)*
- Can Safety be Measured? – *L. Dumont (EDF, France)*
- International Pls and the UK Nuclear Energy Generators – *C. Atkinson (B. Energy, UK)*

RISK INDICATORS
Chair: M. Khatib-Rahbar (ERI, USA) / U. Schmocker (HSK)

- An Approach to Development of a Risk-based Safety Performance Monitoring System for Nuclear Power Plants – *M. Khatib-Rahbar (ERI, USA)*
- Use of WANO Pl and living PSA in Okiluoto NPP – *R. Himanen (TVO, Finland)*
- Pls: Relationship to Safety and Regulatory and Inspection Programs? – *U. Schmocker (HSK)*
- Risk Indicators at Cofrentes NPP – *J. Suárez (IBERINCO, Spain)*

INTERNATIONAL PERFORMANCE INDICATOR SETS
Chair: L. Lederman (IAEA) / L. Carlsson (NEA)

- Indicators to Monitor NPP Operational Safety Performance – *L. Lederman (IAEA)*
- Results of PWGIP Baltimore Meeting Related with Pls – *J.J. Van Binnebeek (AVN, Belgium)*
- WANO Pls – *H. Hamlin, Y. Shimada (WANO)*

REGULATOR INDICATORS - PART 1
Chair: A. Gea (CSN)

- Pls at Bavarian NPP – *E. Seidel (Germany)*
- Development and Use of Safety Indicators at STUK – *P. Tiippana (STUK-Finland)*
- Experience in the Use of Pls in Korea – *Sae-Yul Lee (KINS, Korea)*

REGULATOR INDICATORS - PART 2
Chair: J.J. V. Binnebeek (AVN, Belgium)

- New Pls System in Spain – *M. Maroño (CIEMAT, Spain)*
- Development of Safety Pl of Regulatory Interest (SAFER) in Pakistan – *Khatoon (Pakistan)*
- The Development of Safety Indicators for NPP at the French Safety Authority – *M. Raymond (DSIN, France)*

REGULATOR INDICATORS - PART 3
Chair: P. Tiippana (STUK-Finland)

- Development of Safety Pl in Japan – *J. Tanaka (NUPEC, Japan)*
- Development of Safety Pl System at Ukrainian Regulator – *O.V. Pecherytsya (SSTC, Ukraine)*
- Safety Indicators in the Nuclear Regulatory Process – *T. Hill (CSN, South Africa)*
- Regulatory Body Experience with the Safety Indicator Use – *R. Rehacek (Czech Republic)*

ORGANISATION AND SAFETY CULTURE INDICATORS
Chair: J. Toth (Paks, NPP) / F. Calduch (Cofrentes NPP)

- Cofrentes NPP Indicators to Monitor Operational Safety Performance – *F. Calduch (Cofrentes NPP)*
- Indicators of Plant Performance During Events Identified by Recuperare Method – *S. Bardou (IPSN,France)*
- Assessment of Human Performance and Safety Culture at the Paks NPP – *J. Toth (Paks, NPP)*
- Pl at Daya Bay NPP – *C. Fang (Daya Bay NPP, China)*

LIST OF PARTICIPANTS

ANNEX - SUMMARY REPORT OF THE USE OF PLANT SAFETY PERFORMANCE INDICATORS [NEA/CSNI/R(2001)11]

Programme Overview

Specialist Meeting on Safety Performance Indicators
Madrid, Spain, October 17-19 2000

Tuesday 17 October 2000	
8:00 – 9:30	Registration
9:30 – 10:00	**OPENING** • *A. Martín (CSN),* • *A. Carmino (IAEA),* • *K. Shimomura (NEA),* • *F. Ynduráin (CIEMAT)*
10:00 – 10:30	*Break*
10:30 – 11:30	**UTILITY INDICATORS** Chair: *S. Floyd (NEI, USA) / Jürgen Schlegel (WANO)* • U.S. Industry Perspectives on Role of Indicators in the Regulatory Process – *S. Floyd (NEI, USA)* • Performance Indicators in the USNRC's Revised Reactor Oversight Process – *D. Hickman (NRC, USA)* • Can Safety be Measured? – *L. Dumont (EDF, France)* • International Pls and the UK Nuclear Energy Generators –*C. Atkinson (B. Energy, UK)*
11:30 – 12:00	*Coffee break*
12:00 – 13:30	**RISK INDICATORS** Chair: *M. Khatib-Rahbar (ERI, USA) / U. Schmocker (HSK)* • An Approach to Development of a Risk-based Safety Performance Monitoring System for Nuclear Power Plants – *M. Khatib-Rahbar (ERI, USA)* • Use of WANO Pl and living PSA in Okiluoto NPP – *R. Himanen (TVO, Finland)* • Pls: Relationship to Safety and Regulatory and Inspection Programs? – *U. Schmocker (HSK)* • Risk Indicators at Cofrentes NPP – *J. Suárez (IBERINCO, Spain)*
13:30 – 15:00	***Lunch***
15:00 – 16:30	**INTERNATIONAL PERFORMANCE INDICATOR SETS** Chair: *L. Lederman (IAEA) / L. Carlsson (NEA)* • Indicators to Monitor NPP Operational Safety Performance – *L. Lederman (IAEA)* • Results of PWGIP Baltimore Meeting Related with Pls – *J.J. Van Binnebeck (AVN, Belgium)* • WANO Pls – *H. Hamlin, Y. Shimada (WANO)*

Wednesday 18 October 2000	
8:00 – 9:00	Registration
9:00 – 10:15	**REGULATOR INDICATORS (Part 1 - 21)** Chair: *A. Gea (CSN)* • Pls at Bavarian NPP – *E. Seidel (Germany)* • Development and Use of Safety Indicators at STUK – *P. Tiippana (STUK-Finland)* • Experience in the Use of Pls in Korea – *Sae-Yul Lee (KINS, Korea)*
10:15 – 10:30	*Break*
10:30 – 11: 30	**REGULATOR INDICATORS (Part 2 - 22)** Chair: *J.J. V. Binnebeck (AVN, Belgium))* • New Pls System in Spain – *M. Maroño (CIEMAT, Spain)* • Development of Safety Pl of Regulatory Interest (SAFER) in Pakistan – *Khatoon (Pakistan)* • The Development of Safety Indicators for NPP at the French Safety Authority – *M. Raymond (DSIN, France)* •
11:30 – 12:00	*Coffee break*
12:00 – 13:30	**REGULATOR INDICATORS (Part 3 - 23)** Chair: *P. Tiippana (STUK-Finland)* • Development of Safety Pl in Japan – *J. Tanaka (NUPEC, Japan)* • Development of Safety Pl System at Ukrainian Regulator – *O.V. Pecherytsya (SSTC, Ukraine)* • Safety Indicators in the Nuclear Regulatory Process – *T. Hill (CSN, South Africa))* • Regulatory Body Experience with the Safety Indicator Use – *R. Rehacek (Czech Republic)*
13:30 – 15:00	***Lunch***
15:00 – 16:30	**ORGANISATION AND SAFETY CULTURE INDICATORS (24)** Chair: *J. Toth (Paks, NPP) / F. Calduch (Cofrentes NPP)* • Cofrentes NPP Indicators to Monitor Operational Safety Performance – *F. Calduch (Cofrentes NPP))* • Indicators of Plant Performance During Events Identified by Recuperare Method – *S. Bardou (IPSN,France)* • Assessment of Human Performance and Safety Culture at the Paks NPP – *J. Toth (Paks, NPP)* • Pl at Daya Bay NPP – *C. Fang (Daya Bay NPP, China)*

Thursday 19 October 2000	
8:00 – 9:00	Registration
9:00 – 11:30	**ROUND TABLE DISCUSSION ON OPPORTUNITIES FOR INTERNATIONAL CO-OPERATION ON PERFORMANCE INDICATORS (31)** Chair: *J. Johnson (NRC)* Table: *M. Raymond (DSIN), L. Carlsson (NEA), L. Lederman (IAEA), P. Baranowsky (NRC)* • Presentation of a proposal for discussion – J. Zarzuela *(CSN, Spain)*
11:30 – 12:00	*Coffee break*
12:00 – 13:30	**CLOSURE** **ROUND TABLE OF SESSION CHAIRMEN** **BRIEFING CONCLUSIONS (32)** Chair: *A. Alonso (CSN) / J. Zarzuela (CSN)* All session chairmen

**Papers of Specialist Meeting on
Safety Performance Indicators
October 17-19, 2000, Madrid, SPAIN**

A NEW REGULATORY OVERSIGHT PROCESS

Stephen D. Floyd, Senior Director, Nuclear Energy Institute

ABSTRACT

The purpose of this paper is to describe a new approach to regulatory oversight of the commercial nuclear power industry that is risk-informed and performance-based. While these concepts can, and have been, used in rulemaking and licensing activities, this paper focuses on the regulatory oversight activities of Assessment, Inspection, and Enforcement. The paper provides a discussion of the deterministic regulatory framework, why the time is ripe for a new paradigm, what is meant by “risk-informed, performance-based,” how a new regulatory oversight process would work, and what the licensee and NRC roles and responsibilities would be.

Section I addresses why the time is ripe for a paradigm shift toward the concepts of risk-informed, performance-based oversight.

Section II defines the concepts of risk-informed, and performance-based regulatory oversight.

Section III discusses the safety framework of the new regulatory oversight process. The objectives of the process are stated; the key success attributes are described; safety expectations and thresholds are defined; and the specific performance indicators and action thresholds are identified.

Section IV describes the implementation of the new risk-informed, performance-based regulatory oversight process. The roles and responsibilities of nuclear power plant licensees and the Nuclear Regulatory Commission are described, as well as reporting requirements.

PAPER CONTENTS

I. TOWARD A NEW PARADIGM OF REGULATORY OVERSIGHT

A. Deterministic Regulatory Framework

Since the advent of commercial nuclear energy in the early 1960s, regulation of the design and operation of nuclear energy plants has been based on various deterministic criteria. To obtain and maintain an operating license, a licensee must assure that its plant can be placed in a safe condition following a number of postulated design basis accidents. Given the minimal test data and operating experience that existed when these criteria were established, both the postulated accidents and the analytical methods used to evaluate a plant’s response were intentionally conservative. These deterministic criteria also provided the basis for identifying what plant structures, systems, components (SSCs) and activities were important from a safety perspective. Requirements were then established to regulate these “safety-related” SSCs and activities.

The implementation of regulations based on the deterministic framework has traditionally been accomplished through a detailed programmatic and prescriptive regulatory approach. This approach focuses on the process of how regulations are implemented, relies on licensee commitments to prescribed implementation methods (or programs), and uses inspection and enforcement to ensure compliance with specific processes and commitments, rather than on the safety intent or objective of the regulations themselves. The determination of compliance depends heavily on a review of records documenting the methods used by the licensee to implement the regulatory requirements. In short, the focus has been on the inputs to the program, and not on the outputs or safety results actually achieved.

In retrospect, the traditional regulatory framework, based on deterministic criteria to identify what is important to safety, and implemented through prescriptive regulations and regulatory guidance, has served its purpose in assuring the protection of public health and safety. It is widely acknowledged, and demonstrated by both NRC and industry performance indicators, that high levels of safety and reliability have been sustained by the U.S. operating plants.

Since 1984, however, when the NRC initiated a program to eliminate requirements marginal to safety, it was recognized that some of the regulatory requirements and guidance that had been issued were imposing burdens that were not commensurate with their safety benefits.

Initiatives by both the industry and the NRC have begun to improve the safety focus of regulations. These initiatives have identified areas where regulations or regulatory guidance are out of date, where operating experience or improved technology provide a better understanding of a source of risk, and where areas of marginal safety significance can be found that are highly resource intensive. In the course of these initiatives, it has been recognized that the traditional regulatory framework, deterministically-based and implemented prescriptively, can often lead to circumstances where NRC and industry resources are expended on matters that have little to do with the safe and reliable operation of a plant.

Two regulatory initiatives have contributed toward an improved focus on safety and risk insights. First, in response to NRC Generic Letter 88-20, U.S. commercial nuclear energy plants committed to producing plant-specific probabilistic safety assessments (PSAs). Increasingly, the insights from PSAs have been incorporated into the regulatory process as these studies advanced to their level one (core damage frequency) and level two (containment failure frequency, source term) results. PSA is a powerful analytical tool that provides a different means to evaluate the design and operational safety of a plant and complements traditional deterministic methods. Additionally, PSA insights can highlight which SSCs and activities are important to safety from a risk perspective.

The second initiative is the NRC's promulgation of 10 CFR 50.65, the maintenance rule. This rule relies on a risk-informed, performance-based approach as the means of regulatory oversight. The licensee is required to monitor the performance or condition of specific SSCs against licensee established goals or performance criteria to provide reasonable assurance that these SSCs are capable of fulfilling their intended safety functions. In this approach, the licensee is afforded great flexibility in implementation methods and in determining how it will comply with the regulation. In addition, regulatory oversight of implementation is based on monitoring the results of the licensee's efforts, rather than on the traditional review of programmatic compliance.

Risk-informed regulation, using PSA insights as a means of determining what is important, and performance-based regulation, where implementation methods are not prescribed and regulatory oversight focuses on the results of licensee activities, are concepts that can significantly improve the traditional regulatory framework. More and more, both industry and NRC activities aimed at regulatory improvements are relying on these new types of regulatory approaches to continue to improve plant safety and reliability. However, these concepts have largely been applied on an ad hoc basis to different technical areas or to areas where additional regulations are under consideration. In doing so, they are often interpreted differently for different applications. In other cases, they are not well understood by many individuals. Confusion over these concepts and their relationship to and distinction from traditional regulatory approaches can only detract from important initiatives that seek a more effective, efficient and stable regulatory framework and process.

It is important to make clear at this point that we do not intend to propose overthrowing the deterministic criteria, particularly defense in depth. As will be shown later in this paper, we believe that the deterministic criteria can be applied in setting the framework for the regulatory oversight assessment, but that the actual measurement of success in achieving safety should use objective and measurable performance indicators directly related to safety. Additionally, it should be made clear that this proposal does not require or envision any revision to the Code of Federal Regulations to make the regulatory oversight process more risk-informed. While there are other industry initiatives pursuing the safety benefit of various deterministic and probabilistic rules, this proposal does not.

B. We are ready for a new Paradigm

As this paper will discuss in succeeding sections, we believe the nuclear industry is ready for a new approach – a new paradigm which builds on the proven safety record of the commercial nuclear industry, the maturity of the technology and its application, and our ability to use risk analysis and operating experience to focus our attention and activities on truly significant safety indicators.

Nuclear electric generation is not a zero defect industry. Our approach has consisted of defense in depth. We have selected systems that are redundant, diverse, and single failure tolerant in order to both prevent and mitigate the consequences of potential events. The net effect of incorporating defense in depth into design, construction, maintenance, and operation is that the systems are more tolerant of defects. This was the deterministic foundation when we didn't have any operating experience upon which to base our regulatory structure.

But today nuclear electric generation is a mature industry with over 40 years experience. Our 2208 reactor years of operation reflect an experienced industry that can look at its operating experience and performance in an informed manner. We have learned and fixed a lot. We now know where to focus our management attention and resources to operate and maintain our power plants safely.

Industry has also matured in its ability to self assess and correct problems. The Institute for Nuclear Power Operations sets standards of excellence for operations, maintenance, engineering and other plant processes, and it has established performance indicator goals which the industry is exceeding. Human performance, self assessment and corrective action programs at nuclear plants are mature and aggressive in improving safety and production outcomes. Operating experience is shared within the industry and incorporated directly in self assessment and corrective action programs.

Our ability to use probabilistic risk assessment techniques has also expanded. Risk insights are now commonly used (e.g., the maintenance rule implementation and the use of Individual Plant Evaluations).

We now need to use the combination of operating experience and risk insights to establish objective safety and regulatory thresholds to better focus our resources and energies to achieve the desired safety significant performance results and to de-emphasize the regulator's focus on inputs, i.e., processes and procedures.

Before we describe the proposed new paradigm, it is worthwhile to discuss what we mean by "risk-informed, performance-based oversight." That discussion follows in Section II.

II. RISK-INFORMED, PERFORMANCE-BASED OVERSIGHT

In any regulatory regime with the aim of assuring the protection of public health and safety, there are two fundamental questions that must always be addressed. One question is, "What aspects of the licensee's facility and operation are important to safety and therefore merit regulatory oversight?" The next question that follows is, "What are the appropriate regulatory oversight activities for those aspects that are important to safety?" In short, these questions are "what's important," from the standpoint of assuring public health and safety, and "how does one regulate what's important."

In the previous section, it was noted that the traditional means of answering these questions were deterministic criteria for identifying what is important to safety and a prescriptive/programmatic approach for regulating licensee activities pertaining to the items important to safety. Risk insights offer a different means for identifying what is important to safety and performance-based regulation is a different means of regulating items important to safety.

The following subsections discuss risk-informed and performance-based regulatory concepts.

A. Risk-informed Oversight

Using risk insights as an aid to decision-making in the regulatory process is often referred to as risk-informed regulation. A more comprehensive definition of risk-informed regulation is:

> ***A regulatory approach in which operating experience and engineering judgment are used in***

concert with the analytical insights derived from probabilistic safety assessments to focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety.

The concept of risk-informed oversight is consistent and compatible with the overall goal of improving plant safety and reliability through a regulatory process that is more focused, objective and efficient. With respect to being efficient, clear and reliable, a risk-informed regulatory oversight approach offers a means to focus resources in a manner that effectively complements and improves the current deterministic approach.

It must also be noted that risk-informed methods are not new and have been in use by both the industry and the NRC for many years. While the use of risk insights has limitations, just like any other analytical tool, these limitations have been overcome by blending the insights derived from risk analysis with operating experience and engineering judgment. Operating experience includes that compiled and made available through codes, standards and guidance documents. The first part of the maintenance rule implementation is a good example of this blend. First, the scope of SSCs is defined in the rule itself deterministically. The risk significance of these SSCs is then initially derived from plant-specific PSAs by calculations using standard PSA importance measures. An expert panel is then utilized to review, adjust and finalize the list of risk significant SSCs. The NRC has recognized this approach as providing an effective means of establishing the risk significance of plant SSCs.

B. Performance-Based Oversight

Performance-based regulation is defined and characterized as follows:

A regulatory approach that focuses on results as the primary means of regulatory oversight, and that has the following attributes:

- ***Measurable parameters to monitor plant and licensee performance;***

- ***Objective criteria to assess performance based on risk insights, deterministic analyses and/or performance history; and***

- ***Licensee flexibility to determine how to meet established performance criteria.***

Performance-based regulatory oversight is consistent with the goal of continued improvements in plant safety and reliability through a more focused, objective and efficient regulatory process. By establishing objective criteria, clarity, consistency and stability in the regulatory process can be dramatically improved. In addition, a performance-based regulatory oversight approach helps to establish and maintain an appropriate distinction between NRC's regulatory oversight role and the licensee's responsibility to manage plant operations in a safe and effective manner.

It is important to note "performance-based" refers to safety results – outcomes – and not assessment of the work processes involved in achieving results.

C. Risk-Informed, Performance-Based Regulatory Oversight

A risk-informed, performance-based approach to regulatory oversight combines the "risk-informed" and "performance-based" elements described in subsections A and B above, and applies these concepts to NRC assessment, inspection and enforcement activities. Stated succinctly,

Risk-informed, performance-based regulatory oversight is an approach in which risk insights, engineering analysis and judgment, and performance results are used to:

- ***develop measurable and/or calculable parameters for monitoring safety performance,***
- ***establish objective criteria for evaluating safety performance,***
- ***establish objective safety and regulatory thresholds, and***
- ***focus on the results as the primary basis for regulatory oversight actions.***

D. Discussion and Implications for Regulatory Oversight

Under risk-informed, performance-based regulatory oversight, the focus is on the results, on assuring safety or the output of licensee programs rather than on the procedures and processes that make up licensee programs. A criticism of this approach is that it is reactive in waiting for failures to occur before any actions are taken. On the contrary, risk-informed, performance-based regulatory oversight provides a focus on those items important to safety and reliability and is a natural incentive to maintain high performance levels. Additionally, objective performance monitoring can provide early indicators of declining trends in safety performance.

A premise of risk-informed, performance-based oversight is that monitoring provides reasonable assurance that challenges to radionuclide barriers will be minimized and that safety functions will be fulfilled. This monitoring can also indicate the onset of problems which, if not addressed, could become more significant. To be effective, the objective performance criteria used to monitor performance must be set at a level that maintains safety margins above the standard of adequate protection of public health and safety. Licensee actions must be designed to preclude failures that threaten this standard. Many failures, however, that occur in the course of normal operations, do not significantly reduce the margin of safety due to the "defense-in-depth" principles reflected in the design and operation of each plant. For these failures, the key aspect is that appropriate cause determinations and corrective actions are taken so that performance is restored or maintained above the performance criteria. This ensures that adequate safety margins are maintained and are not allowed to degrade to a point that does not meet the standard of adequate protection of public health and safety.

The concept of a risk-informed, performance-based regulatory approach has often been confused with other concepts. For example, many people believe performance-based means inspecting and auditing work processes while they are occurring rather than reviewing paperwork documenting those processes after the fact. While real-time audits and inspections may be useful, they do not represent a risk-informed, performance-based regulatory approach that assesses the overall effectiveness of meeting the regulation. As another example, many people use the terms performance-based regulation and risk-based regulation synonymously. This confusion may stem from the fact that calculated values or assumptions (e.g., reliability and availability numbers) used in PSAs may also be used to establish performance criteria for SSCs in a risk-informed, performance-based regulatory approach. Risk insights may also be used to establish testing intervals for important plant equipment. Again, while these practices are advocated and are highly complementary, their effective implementation requires that the distinction between the two concepts be understood and maintained.

A risk-informed, performance-based approach focuses on objective safety outcomes and allows the licensee management the flexibility to determine how to achieve safety in an effective and efficient manner.

Figure 1 will be used to illustrate the risk-informed, performance-based approach. The licensee is responsible for all aspects of safely operating and maintaining the nuclear power plant. This responsibility includes providing the key inputs (plant, people, processes, and procedures), and exercising prudent management (shown here as effective human performance, robust self assessment, and effective corrective action) to ensure successful outcomes (safety performance and cost effective production).

In a deterministic regulatory regime, the regulator attempts to assess <u>all</u> aspects of the licensee's activities (with the exception of cost effective production), regardless of the nexus to safety. In this regime, the regulator specifies *what* the licensee must do, i.e., the requirements and also prescribes *how to* meet those requirements. Regulatory compliance is achieved by the licensee meeting its programmatic commitments to the prescribed methods or processes detailed in the regulatory guides and the interpretations of individual NRC staff. The regulator, without a framework in which to determine what is genuinely important for review, tries to review everything, including areas for which there are no regulations (such as human performance). The regulator also errs in viewing <u>any</u> error or deviation as a violation, even though it does not result in an unsafe outcome.

Under a risk-informed, performance-based approach, the regulations still specify *what* the requirements are; however, the licensee has the flexibility to determine *how* to meet the requirements. Achievement of the requirements is assured by monitoring performance relative to established safety performance goals.

Regulatory oversight using the risk-informed, performance-based approach should become more safety focused and succinct. The level of regulatory oversight should be commensurate with the degree of achievement of the safety performance criteria. For example, one would expect less inspection for those licensees who are maintaining high levels of safety performance above the regulatory threshold, and more inspections for those who are not meeting their safety performance criteria. Enforcement policy would be similar. If the licensee is meeting its performance criteria, then there should be no reason for enforcement actions in areas covered by those criteria.

As long as the licensee continues to meet the established safety performance criteria and takes appropriate actions to prevent recurring functional failures, the regulator should continue to allow the licensee flexibility in managing its implementation of the regulations. If performance degrades to the point where the licensee fails to meet safety performance criteria, this does not necessarily mean that the licensee is no longer in compliance with the regulation or is unsafe. Rather, it is a flag that increased regulatory oversight of licensee activities may be warranted, including focused inspection. However, should safety performance continue to decline and the corrective action is not providing reasonable assurance that the safety performance criteria or goals will be satisfied or the issue will be resolved, then more extensive regulatory interaction will occur. At this point, the licensee has lost much of the flexibility afforded when safety performance criteria were being satisfied, and corrective measures are likely to be reviewed in detail by the regulator. This additional regulatory attention may result in enforcement action to assure that appropriate corrective action is taken to comply with the regulations and satisfy the appropriate safety performance indicators.

In conclusion, regulatory oversight in an improved framework would be a graded approach based on safety significance and the safety performance results of the licensee. This approach provides an incentive for licensees to keep safety performance levels high, enables the NRC to focus its resources more effectively on safety significant matters when increased regulatory oversight is warranted, and enhances the ability of licensee management to achieve safety and cost effective power.

III. FRAMEWORK FOR RISK-INFORMED, PERFORMANCE-BASED REGULATORY OVERSIGHT PROGRAM

A. Purpose

The purpose of this framework is to define a safety focused regulatory oversight process for those activities that can be effectively monitored using risk-informed, performance-based approaches. The process acknowledges the need to preserve the current regulatory requirements (e.g., rules, regulations, operating license) that define the design and licensing basis of plants. It is recognized that those activities for which objective measures of safety cannot be provided, traditional oversight will be required. It is suggested, however, that this oversight should rely more on evaluating licensee self assessments as an alternative to NRC team inspections.

B. Objectives

The new risk-informed, performance-based approach is designed to meet the nuclear power plant stakeholders needs for an effective regulatory oversight program:

- Accurately and objectively measure the safety performance of nuclear power plants in protecting the public health and safety.

- Provide accurate and understandable safety performance information to the public, news media, and other stakeholders.

- Provide utility licensees and the NRC with objective indicators to assess safety performance and trends, to rationalize the NRC Enforcement Policy, and to allocate resources in a effective and efficient manner.

- Provide Congress with objective information to assist in performing its oversight and authorization responsibilities.

C. Program Attributes Necessary To Achieve Objectives

The following attributes are considered necessary to achieve the desired objectives:

- The program should be directly linked to the NRC's mandate to assure protection of public health and safety.

- The program should preserve current deterministic requirements of the regulations (e.g., defense in depth, single-failure, redundancy).

- The program should apply the concepts of risk-informed, performance-based oversight.

- Safety performance assessment should be based on public health and safety thresholds and regulatory thresholds, not on relative plant performance.

- Assessment conclusions should be supported by the direct measurement of the performance indicators.

- Attributes of appropriate indicators are:
  - a direct relationship should exist between the indicator and safety performance expectations
  - data necessary to measure the indicator should be available or capable of being generated
  - indicators should be capable of being expressed in quantitative terms that are not ambiguous
  - indictors should be meaningful, i.e., their significance is readily understood
  - indicators should be able to be validated

- Program implementation should:
  - provide clear roles and responsibilities of the NRC and licensees
  - communicate results to the public
  - include a decision model or criteria so that NRC actions are predictable
  - be simple, nonredundant, and resource efficient

D. Program Structure

The structure of the program necessarily must reflect the legal and regulatory obligations of the Nuclear Regulatory Commission (NRC). Under the Atomic Energy Act, the NRC is charged with issuing and enforcing requirements that are necessary to ensure adequate protection of public health and safety. While adequate protection is not defined in the Atomic Energy Act, NRC policy considers adequate protection to have been achieved if a plant is operating in conformance with the regulations. This position is reasonable because regulations are largely promulgated on the basis that they are necessary to establish adequate protection of public health and safety.

10 CFR Part 50 (and appendices) contains most of the technical regulations that apply to power reactors. The primary purpose of these regulations is to establish requirements that define:

1. the robustness of the barriers to radionuclide release,
2. the postulated plant events and accidents that must be considered and the methodologies for analyzing the events, and
3. the capabilities of the engineered safety features for mitigating postulated events.

Nuclear power plants were granted a license largely on the basis that a review of the design, construction and intended operation of the facility would comport with the requirements of 10 CFR part 50 and meet the guidelines of 10 CFR Part 100. Part 100 directs the NRC to consider "the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur." Public health and safety is not adversely impacted by nuclear plant operations unless radiation exposures exceed the limits imposed by 10 CFR Part 100.

Ensuring adequate protection of the public health and safety also requires capability to protect the public in the event of a radiological emergency, control of radiation exposure, control of radioactive materials, and physical security. The regulations addressing these areas primarily include: 10 CFR Part 50.47 (Emergency Plans); 10 CFR Part 20 (Standards for Protection Against Radiation); 10 CFR 50.36a (Technical Specifications on effluents from nuclear power reactors); 10 CFR 50, Appendix I (ALARA); 49 CFR 171-173, 10 CFR Part 71 (Packaging and Shipment of Radioactive Materials); 10 CFR Part 61 (Disposal of Low Level Radioactive Waste); and 10 CFR Part 73 (Physical Protection of Plants and Materials).

Based on these regulatory requirements, performance expectations that relate to the NRC's mission to protect public health and safety in the operation of commercial nuclear power plants can be grouped into three strategic performance areas and seven specific cornerstones of safety. (See Figure 2.)

E. Safety Performance Expectations

1. Reactor Safety

Within the performance area of reactor safety (avoiding accidents and reducing the consequences if they occur), there are four cornerstones: Initiating Events; Mitigating Systems; Barrier Integrity; and Emergency Preparedness.

Initiating events are those situations which upset plant stability and challenge critical safety functions, during shutdown as well as power operation. If not properly mitigated, and if multiple barriers are breached, a reactor accident could result which might compromise the public health and safety. Plant operators can reduce the likelihood of a reactor accident by maintaining a low frequency of these initiating events. Such events include reactor shutdowns due to turbine trips, loss of feedwater, loss of off-site power, and other significant plant transients.

Mitigating systems are designed to prevent an accident or reduce the consequences of a possible accident. Mitigating system equipment is maintained through a rigorous maintenance program, checked through periodic testing, and monitored during actual performance.

Barrier integrity provides reasonable assurance that the physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events.

Emergency preparedness ensures that the licensee is capable of implementing adequate measures to protect the public health and safety during a radiological emergency. Licensees routinely assess and refine their emergency plans through Emergency Response Organization (ERO) participation in drills, exercises, actual events, training, and subsequent problem identification and resolution. Employees are trained to ensure that the plan can be effectively implemented during an emergency. Drill and exercise performance, ERO drill participation and reliability of the alert and notification system contribute to reasonable assurance that the licensee has an effective emergency preparedness program.

2. Radiation Safety

Radiation safety encompasses both occupational exposure to workers, and protection of the public during normal plant operations. There are therefore two cornerstones:

Occupational radiation safety consists of programs and procedures to minimize exposure of workers to ionizing radiation and to ensure that doses are maintained at levels less than prescribed regulatory limits.

Public radiation safety monitors the effectiveness of procedures and systems designed to minimize radioactive releases from a nuclear power plant during normal operation and to keep releases within federal limits.

3. Physical Protection

Nuclear power plants are required to have well trained security personnel and a variety of protective systems to guard vital plant equipment, as well as programs to ensure that employees have received appropriate checks prior to employment, and are constantly fit for duty.

F. Safety Performance Indicators, Thresholds, And Performance Bands

This section will describe the performance indicators now in the program, how performance thresholds were set, and the four performance bands which inform the NRC staff's regulatory response to objective performance. During the development of the program NRC, industry and other stakeholders determined that it was not possible to develop indicators for all activities which constitute safe and effective operation and maintenance of nuclear power plants. These areas must of necessity continue to be assessed based on inspection. The next section will briefly discuss how inspection complements the performance indicator program.

1. Indicators

Each performance cornerstone has a set of specific safety performance indicators for objectively evaluating the achievement of each performance expectation. Each indicator is plotted over time to identify the trend in performance. Plotting the indicators also shows the available safety performance margin for each indicator. See figure 3 for a listing of the performance indicators and action thresholds.

2. Performance Thresholds

In general, three performance thresholds have been established for each indicator. (Exceptions will be discussed below.) The thresholds delineate four performance bands which assist NRC in determine the appropriate level of inspection activity. As performance decreases, more NRC attention and inspection resources are warranted.

The first threshold distinguishes normal high industry performance from results which, while still safe, represent outlier performance. In most cases, historical data was reviewed and the threshold was set at the 95th percentile. Below this level additional regulatory attention may be warranted.

The second threshold was set at a level that represented an approximate increase in core damage frequency of E-5. For example, an increase in scram frequency to greater than six per 7000 critical hours would increase the core damage frequency approximately E-5. At this point, while safety was still being maintained, more active involvement of the NRC would be appropriate.

The third threshold represents the point at which the indicator change would represent an increase in core damage frequency of E-4. Any further decrease in performance would be viewed as unacceptable and operation would not be allowed until the condition was corrected and actions were taken to prevent reoccurrence.

Where possible, historical data was used in establishing the first threshold. For the other thresholds, a variety of generic and plant specific risk studies were used to establish generic thresholds so that all plants were being treated in the same manner. Of course, some indicators required different thresholds based on plant equipment (BWR as opposed to PWR) or number of emergency diesel generators, for example.

Several of the indicators were completely new, or data had not previously been collected for them. Examples here include participation in emergency drills and occupational exposure control. In these cases, industry data was collected and thresholds established.

Several indicators do not have three thresholds. The reasoning in these cases was that it was not possible to determine changes in core damage risk based on a change in the indicator. For example, the performance of physical security equipment can not be related to core damage frequency.

The specific thresholds can be observed in figure 3. In general, the initiating event and mitigating system indictors were most amenable to establishing three thresholds; the other cornerstone indicators were more programmatic and consequently, less quantifiable.

3. Performance Bands

As discussed above, the three thresholds establish four performance bands: the utility response band, the increased regulatory response band, the required regulatory response band, and the unacceptable band. These bands have been color coded to provide a clear way of communicating plant performance. These colors, moving in the direction of increased risk, are green, white, yellow, and red. (See figure 4.)

Utility Response Band

This band recognizes and acknowledges that all manufacturing processes have a control band for performance. Utility management’s role is to maintain performance within the control band. The regulator performs a minimal core/baseline inspection program and monitors performance indicators. Performance within the control band provides an indication that human performance, self assessment, corrective actions, and key programs and processes related to the performance area are effective. The increased regulatory attention threshold is set at a value that provides an adequate margin to the required regulatory action threshold such that corrective action can be taken by the utility and reviewed by the regulator before a significant increase in potential risk.

Increased Regulatory Response Band

This band defines the point at which the regulator departs from a core/baseline inspection mode, and questions the adequacy of corrective actions, programs and processes related to the performance area. While performance is still acceptable, it represents a degree of reduction in safety margin that warrants increased regulatory actions.

Required Regulatory Response Band

This band defines the point at which the regulator shifts from reviewing the licensee’s assessment and corrective action, and makes its own independent assessment of the situation.

Unacceptable Band

This band defines the point at which plant operation is not normally allowed. Action to correct the condition and prevent reoccurrence are necessary to return to operations.

G. Inspection and Significance Determination

As stated above, performance indicators provide only a sample of plant performance, and cannot measure performance in several important areas necessary to ensure safe plant operation. Examples include fire protection programs, engineering analysis, and training. Therefore, a baseline inspection program is necessary.

During development of the new oversight program, the objectives of the seven cornerstones were developed and the necessary success attributes were determined. For those areas which could not be assessed using performance indicators, inspection procedures were developed.

However, the question remained, how to compare the performance indicator results and the inspection results? The answer was the significance determination process (SDP).

The SDP is a risk-informed process by which an inspection issue (a violation of a regulation, or a perceived deficiency) is assessed using risk techniques. In general, the process involves three phases. In the first, minor violations which do not affect cornerstone performance are screened out. In the second phase, risk techniques are used to characterize the finding based on the duration of the deficiency or violation, the frequency of initiating events which the deficient equipment was designed to help mitigate, and the degree of diverse or redundant equipment available. The results are expressed using the same bands and colors as the performance indicators. Green represents a core damage frequency (CDF) increase of less than E-6; White a CDF increase between E-6 and E-5; Yellow a CDF increase between E-5 and E-4; and Red a CDF increase of greater than E-4. Finally, a third phase may be conducted for findings greater than Green, so that plant specific PRA/PSA information can be used to sharpen the analysis.

Thus the oversight process provides both performance indicators and inspection findings which can be used in concert to determine performance and what regulatory oversight and action is appropriate.

IV. IMPLEMENTATION

A. Overview

This section describes how the risk-informed, performance-based regulatory oversight program can be implemented. The roles and responsibilities of the licensees and the NRC are outlined below. Three general concepts should be kept in mind. First, licensees and the NRC have individual, but complementary roles in the program in achieving adequate protection of the public health and safety. Second, the process is a continuing cycle (see figure 5) of assessment, inspection plan development, inspection, regulatory action, assessment, etc. Third, while these steps generally follow the order displayed, they are in continual interaction.

B. NRC Responsibilities

1. Overall Responsibilities

The NRC is responsible under the Atomic Energy Act for ensuring that nuclear power plants provide an adequate level of protection of public health and safety. This responsibility requires that margins to safety be maintained such that single performance problems do not result in adverse consequences to the public. At the same time, the NRC needs to exercise caution so as not to encroach on plant management's primary responsibility to safely operate and maintain nuclear power plants by trying to regulate to a zero defect threshold. The risk-informed, performance-based oversight process outlined in this paper recognizes this distinction.

2. Regulatory Oversight Responsibilities

a) Assess results

For areas covered by the safety performance indicators, the NRC would verify the completeness and accuracy of the indicators reported by the licensee and take appropriate action based on the results (as displayed in Figure 4.)

For areas not covered by the safety performance indicators, the NRC would review the results of the licensees performance based on inspections and corrective actions taken by the licensee in response to previously identified deficiencies.

The results would be assessed quarterly and annually on a cornerstone basis. Thus NRC actions would be dictated by a response to individual PI or inspection findings, and a response to multiple findings in a cornerstone area.

b) Develop Inspection Plans

The NRC would develop its inspection plan based on the results of its assessment of licensee performance in the safety performance indicators, its review of licensee corrective actions on previous regulatory actions, and its requirements to assess deterministic regulatory areas not covered by the safety performance indicators. There would always be a minimum, or baseline, inspection program.

For areas covered by the safety performance indicators, the scope of future inspection activities is determined by the performance results relative to the response bands. For example, performance in the utility response band would mean that the NRC would only conduct baseline inspections for that performance area. In addition, NRC could (but has not as yet) make expanded use of licensee self assessments and audits, in lieu of conducting its own inspections, which are often redundant to the licensees efforts. Performance in the increased regulatory response band or the regulatory required response band would warrant increased inspection activity to review licensee determination of the cause of performance problems and corrective action taken.

For areas not covered by the safety performance indicators, the NRC would plan to perform baseline inspections or opt to evaluate/participate in licensee self assessments and audits. By reviewing the licensee's self assessment and audit schedule for the next inspection cycle, the NRC could conserve resources by opting to evaluate licensee self assessments and/or audits rather than conduct redundant inspections. (A precedent for this approach already exists: Inspection Procedure 40501 "Licensee Self Assessments Related to Team Inspections.)

c) Conduct Inspections

Carry out inspection plans and document results in inspection reports. Issues identified by inspectors should be assessed using the SDP. Inspection reports should only describe factual results which rise to a level of significance of Green or greater (i.e., minor violations need not be documented in inspection reports).

d) Regulatory Actions

For areas covered by the safety performance indicators, regulatory actions would depend on the performance results. For example, performance discrepancies that did not cause the results to drop below the utility response band would be noted by the NRC without the need to conduct additional inspection. This would avoid the expenditure of NRC and licensee resources on matters of low safety importance. For performance within the increased regulatory response band, NRC would conduct a short inspection to determine the effectiveness of the licensee's actions to correct the problem. In the required regulatory response band, regulatory actions would depend on the available margin to safety. Actions could include increased inspection activities and, if necessary, confirmatory action letters. For performance which falls into the unacceptable band, it is possible that a shutdown order might be issued.

For areas not covered by the safety performance indicators, the degree of regulatory action should be commensurate with the safety significance or actual consequences of the discrepancy (the SDP result). Consideration should also be given to the overall performance of the plant. That is, if a plant is generally operating in the utility response band for a broad set of indicators, this provides confidence that the licensee has an effective corrective action program and further regulatory action may not be warranted. If the plant is operating in the increased regulatory response band for a number of indicators, this provides evidence that the corrective action program has weaknesses. NRC inspection of the corrective action program and performance areas are warranted.

e) Assess Results

The cycle continues with an annual assessment of licensee performance. This assessment considers the overall performance of the licensee in each of the cornerstone areas.

C. Licensee Responsibilities

1. Overall Responsibilities

The licensee is responsible for all aspects of safely operating and maintaining the nuclear power plant. Figure 1 provides the model depicting the key inputs (plant, people, processes, and procedures), management activities (human performance, self assessment and corrective action), and results/outputs of running the plant (safety performance and cost effective production).

In assessing the performance of inputs, management actions, and outputs, the licensee will develop its own unique self assessment methods and a set of indicators with which to monitor performance. These assessments and indicators are at the discretion of licensee management as they deem appropriate. (It will also have in place a Quality Assurance Program in accordance with 10 CFR 50 Appendix B.) The performance indicators will be created to monitor and assess performance in those areas deemed important to plant management to achieve its own goals and objectives.

The licensee will use its self assessment program and its own internal performance indicators to assess performance of plant equipment and systems, workforce, procedures and processes. When deficiencies or opportunities for improvement are identified, the licensee will use its corrective action program and other management actions to achieve improvement. The licensee will continue to report events and deficiencies as currently required in the regulations.

In addition to the licensee's staff, the industry has established an industrywide plant evaluation program under the Institute of Nuclear Power Operations (INPO). INPO's role is to assist utilities in achieving high standards for nuclear plant operations.

Utility management's role is to maintain safety performance within the utility response band. (See figure 4.) It is the responsibility of the licensee to monitor performance and correct individual errors or trends that are detected before dropping below the Green band by performing root cause analyses, taking corrective actions and monitoring the effectiveness of those actions to restore performance.

Performance within the utility response band provides strong indication that corrective action, self-assessment, and human performance are effective in operating and maintaining the plant.

If performance drops below the Green band, the licensee conducts an in-depth review of why its actions have been unsuccessful, and establishes an integrated plan to restore performance.

2. Regulatory Oversight Responsibilities

a) Assessment

The licensee will monitor and report the safety performance indicators described in Section IV to the NRC on a quarterly basis prior to the NRC assessment.

b) Develop Inspection Plans

The licensee should make the NRC aware of it's self assessment and audit plans that it intends to make available to the NRC to assist the NRC in planning its future inspection activities.

c) Inspection

The licensee provides the results of self assessments and audits in regulatory oversight areas to the NRC in advance of NRC inspections. This include regulatory areas not covered by the safety performance indicators such as security, emergency planning, etc.

d) Regulatory Actions

The licensee performs root cause analyses, identifies corrective actions and reports the status of corrective actions to the NRC prior to any NRC regulatory actions.

REFERENCES

SECY-99-007 Recommendations for Reactor Oversight Process Improvements (Washington, DC: USNRC, January 8, 1999)

NEI 99-02 Regulatory Assessment Performance Indicator Guideline (Washington DC: Nuclear Energy Institute, March 2000)

NUREG-1649 Regulatory Oversight Process (Washington DC: USNRC, July 2000)

Performance Indicators in the USNRC Revised Oversight Process

By
D. Hickman, USNRC

Probably the best summary of our PIs can be found at the following site:

http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/cornerstone.html

This document is about six pages long. It is what NRC is currently using in the initial implementation of our program at all plants, which began on April 2, 2000. Quite a few issues and concerns have developed since then and, although the basic definitions are as yet unchanged, NRC has made a number of changes to the details of what does and does not count. These changes are currently documented only in NRC guidance document, NEI 99-02, Rev 0, at:

http://www.nrc.gov/NRR/OVERSIGHT/NEI_9902.pdf

This is a large document. It is published by the Nuclear Energy Institute but has been endorsed by the NRC. Revision 1 of the document is currently being prepared to incorporate additional lessons learned.

NRC has learned a great deal in the past one-and-a-half years, beginning with the pilot program in June of 1999, and expect to incorporate these lessons in changes to the basic definitions, which could take place later this year, in Revision 2 to NEI 99-02.

Papers of Specialist Meeting on Safety Performance Indicators October 17-19, 2000, Madrid, SPAIN

CAN SAFETY BE MEASURED

Bernard FOUREST
Senior Executive Advisor
EDF / DPN
Site Cap Ampère
1, place Pleyel
93282 Saint-Denis Cedex - France
Telephone: 01 43 69 35 04
Fax: 01 43 69 21 47
e-mail: bernard.fourest@edfgdf.fr

Laurent DUMONT
Ingeneer
EDF / DPN / GSN
Site Cap Ampère
1, place Pleyel
93282 Saint-Denis Cedex - France
Telephone: 01 43 69 35 19
Fax: 01 43 69 21 47
e-mail: laurent.dumont@edfgdf.fr

ABSTRACT

All nuclear operators have indicators they use to mesure the performance. But the safety performance of a nuclear power plant can not be measured by a single number, because it is not a physical quantity.

Such indicators as the leaktightness of each of the three barriers, the number of reportable incidents, or the importance of their potential consequences, can contibute to a judgment of the safety level, but it is difficult to treat them all in the same way: the leaktightnesses of the barriers are physical quantities that can be measured and compared with respect to limits, but it is hard to fix a limit in the number of incidents, and it is necessary to develop a tool to measure the seriousnes of the consequences.

This finding as led EDF to develop indicators that belong to 3 different categories:

- **Indicators of power plant safety condition**, which measure compliance with safety requirements.
- **Safety managment indicators**, which measure the work done to achieve the safety improvement objectives.
- **Measurement of the potential consequences of incidents by probabilistic analysis**, which serves to identify priority improvement themes.

INTRODUCTION

All nuclear operators have indicators they use to measure the performance of their power plants in terms of cost, availability, dosimetry, radioactive release, production of wastes. But the safety performance of a nuclear power plant can not be measured by a single number, because it is not a physical quantity.

It is known on the other hand that such indicators as:

- the leaktightness of each of the three barriers (cladding, primary circuit, containment),
- the number of reportable incidents,
- the importance of the potential consequences of these incidents,

can contribute to a judgment of the level of safety of a nuclear power plant in operation. It can, however, be seen immediately that it is difficult to treat indicators such as those mentioned above all in the same way

- the leaktightnesses of the barriers are measurable physical quantities that can be compared with respect to limits that must not be exceeded (one speaks of safety criteria such as the level of unidentified leakage of the primary circuit, which must remain below 1 gpm),
- the number of reportable incidents can be used to detect a positive or negative evolution over the course of the years, but is hard to compare to a limit,
- the importance of the consequences of the incidents may be an indicator if it is

possible to develop a tool to measure the seriousness of the consequences.

So measuring safety is not simple!

This finding has led EDF to develop, in the last two years, indicators that belong to 3 different categories.

1) Power plant safety condition indicators

These indicators are used to assess the condition of the power plant on the basis of a certain number of measurable parameters for which limits are clearly defined by the safety report and the operating technical specifications. These parameters are regularly measured in operation and compared to the limits with which they must comply. It is possible to know the margin with respect to the limits for each of them and to define a sort of power plant "health report".

These indicators measure compliance with safety requirements.

2) Safety management indicators

These indicators are also measurable parameters, but they are not compared to limits defined by safety rules. They are compared to objectives defined by EDF's Nuclear Operations Division management, which has decided that improved safety must be reflected by the attainment of these objectives. The Nuclear Facilities management defines objectives that reflect its desire for improvement. Trade-offs, at the strategic level as well as the day-to-day level, influence these indicators.

These indicators measure the work done to achieve the safety improvement objectives defined by the management of EDF Nuclear Facilities.

3) Measurement of the potential consequences of incidents

EDF has decided to use the probabilistic analysis of incidents to measure their potential consequences. This analysis ranks incidents with respect to one another by calculating the conditional probability of damage to the fuel because of the occurrence of the incident. This yields an image of the relative importance of incidents with respect to safety. The incidents that appear the more serious according to this analysis are those whose causes must be treated first to avoid their recurrence.

Measurement of the potential consequences of incidents by probabilistic analysis serves to identify priority improvement themes.

PAPER CONTENTS

1 - SAFETY CONDITION INDICATORS

As stated in the introduction, these indicators are compared to limits defined:

- in the safety report that lays down the safety requirements applied when the power plants were designed,
- in the general operating rules that reflect the requirements of the safety report in rules applicable to the day-to-day operation of the power plants.

The safety condition indicators measure the compliance of the power plants with the safety requirements applied when they were designed.

They therefore measure the quality of operation, which must remain in conformity with the criteria defined by the design engineer, but they also measure the quality of construction and of design, which must be such that conformity with the requirements of the safety report can be maintained.

A fundamental principle

One of the fundamental safety principles applied to the design of power plants is the principle of defense in depth, of which one "physical" application is the placement of the three successive barriers between radioactive substances and the environment: fuel cladding, primary circuit, containment.

The studies of the safety report take account of the limiting values for the leakage rate in operation of these three barriers.

Three safety functions and a support function

To avoid failure of the barriers or to limit the consequences of their deficiencies, it is necessary to maintain three safety functions:

- control of reactivity,
- control of cooling,
- control of confinement.

These functions are performed by equipment and systems that must themselves be supplied by the systems necessary for their operation (electric power, compressed air, etc.). These latter systems are grouped under the term "support function".

General operating rules

These contain the operating technical specifications that:

- define the limits of the domains of normal operation of the reactor,
- require the availability of the equipment and systems that perform the three safety functions and the support function according to the domain of operation of the reactor,
- state what to do if equipment or a system performing a safety function or the support function is down, or if the limits of the domains of normal operation are exceeded.

The course to follow if a required item of equipment or system is down is to change to the fallback state (a state of the reactor where the non-availability has less impact on safety) when the non-availability has lasted a certain time (fallback time).

The operator must however strive to achieve the best possible availability of the equipment and systems that perform these functions; he must for example correct accidental down times as quickly as possible, and not wait to reach the fallback time limit.

The indicators proposed by EDF headquarters departments to the nuclear power plants correspond to surveillance of the leaktightness of the barriers and of the availability of the three safety functions and of the support function.

Surveillance of the 1st barrier

Indicator no. 01 : Quantity of iodine 131 from a corrected fault of UO_2 contamination deposited under flow (in Mbq/t) in the water of the primary circuit. This indicator is identical to the one proposed by WANO.

Surveillance of the 2nd barrier

Indicator no. 02 : Mean leakage flowrate of the primary circuit (in l/h)

Indicator no. 03 : Usage factor on the 2 most constraining zones of the primary circuit (this indicator will be developed with the gradual introduction of the automatic primary circuit situations counting system).

Indicator no. 04 : Mean primary/secondary leakage flowrate of the steam generators (in l/h)

Surveillance of the 3rd barrier

Indicator no. 05 : Mean leakage flowrate of the containment (in Nm 3/h referred to a delta P of 60 mbars with respect to the outside). This leak is measured at all times in operation by a system that takes account of the pressure and internal temperature of the containment and of the atmospheric conditions outside the containment.

Indicator no. 06 : Number of activity peaks measured at the stack of which the activity exceeds 4 E-05 Bq/m3.

Surveillance of availability of the three safety functions and the support function

Indicator no. 07 : Surveillance of the "reactivity" safety function

What is supervised is in fact the down time of the function, by the following ratio:

sum of the group 1 down time relative to the "reactivity" function divided by the time spent in the reactor's domain of operation.

Group 1 down times are the equipment and systems down times that have the most impact on safety because they raise questions about the design assumptions or the protection and backup systems.

Indicators nos. 08, 09, and 10 are similar to indicator no. 07 for the two other safety functions and the support function.

Indicators nos. 07 to 10 are numbers expressed as percentages.

Results

The indicators mentioned above are monitored and analyzed by each EDF nuclear power plant, which from them deduces what must be done to improve the safety condition of its installations.

At the level of the Nuclear facilities as a whole, the dispersion of these results is analyzed; this can reveal problems specific to some power plant that may be of technical origin or related to the operation of the power plant.

Figures 1 to 3 below are three examples of indicators tracked by a power plant.

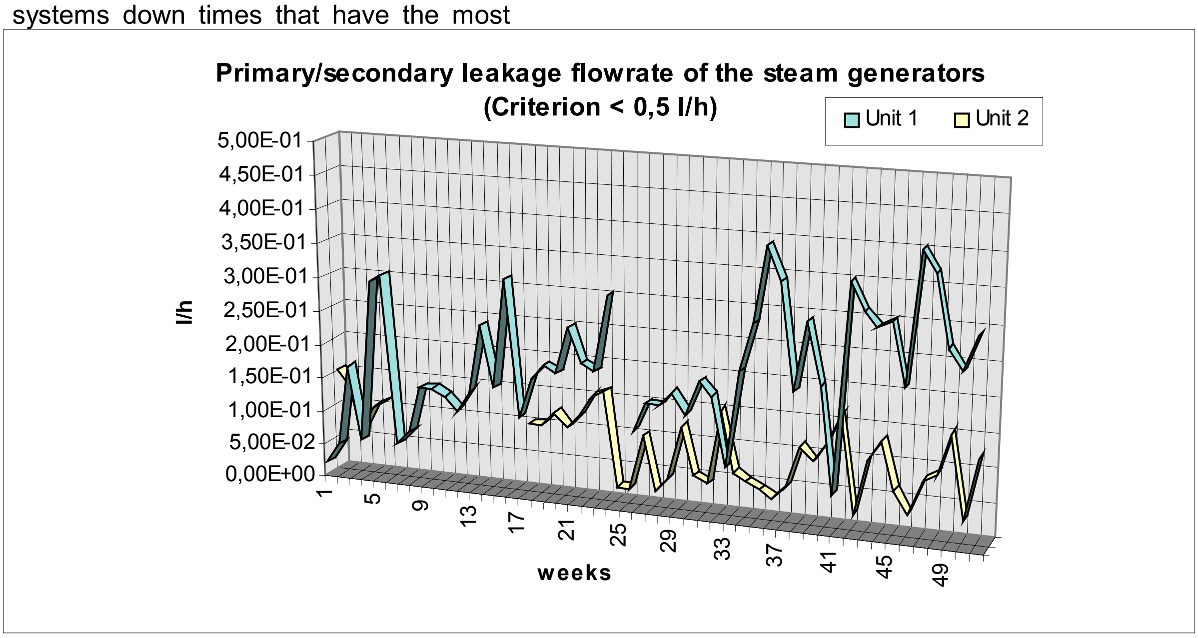


Figure 1

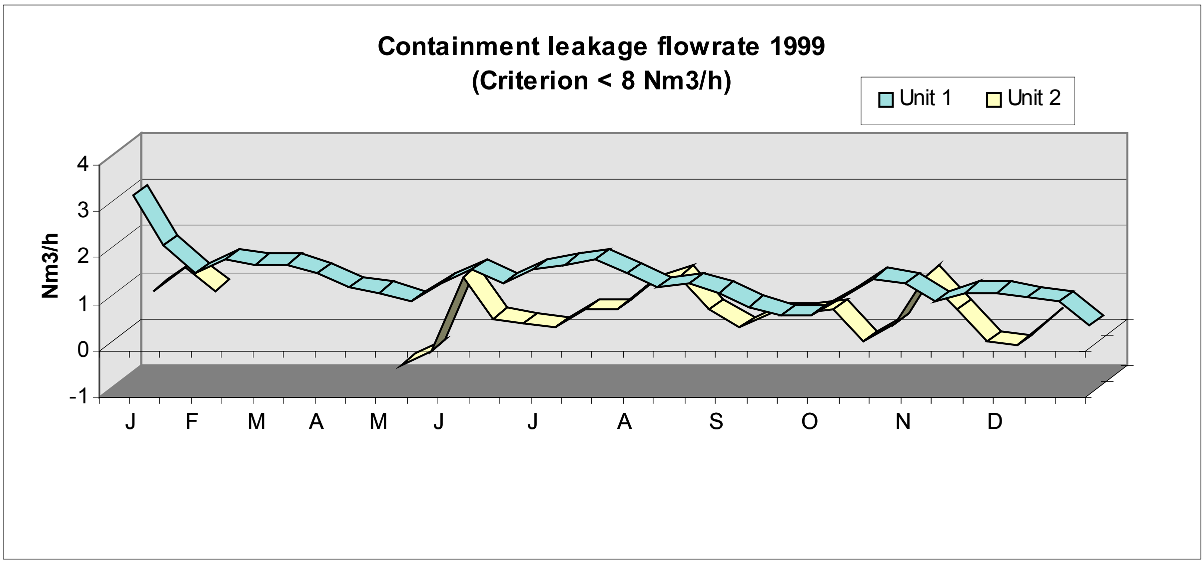


Figure 2

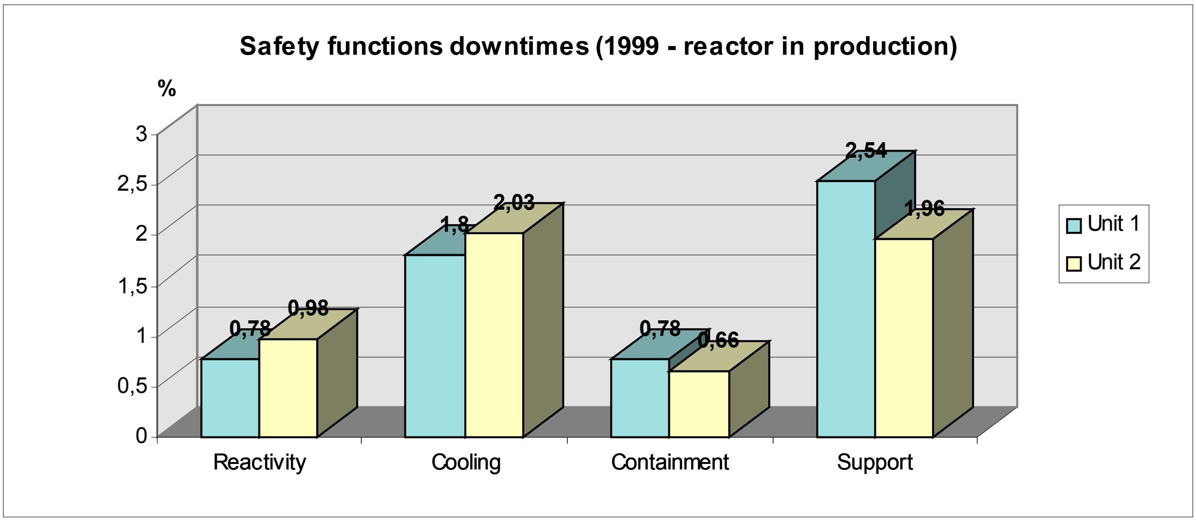


Figure 3

Figure 1 shows the primary/secondary leakage flowrates of the steam generators of the power plant's two reactors from January to December 1999. Although unit 1 is affected by microleaks (which were not detected during the decennial outage program), the value remains below the criterion of 0,5 l/h.

Figure 2 shows the leakage flowrates of the containments of the power plant's two reactors from January to December 1999. The values remain below the criterion of 8 Nm3/h.

Figure 3 shows the level of consumption of the fallback times allowed by the operating specifications. This is the sum of the time consumed (programmed and unexpected fallbacks)/time passed in the domain « reactor in production » ratios expressed as a percentage. This indicator is the image of the reactivity of the sites in restoring equipment that is down to availability: the lower the percentages, the faster the site has reacted.

The high level of consumption for the cooling function is due to frequent cleanings of component cooling heat exchangers, and the unavailability of two condenser steam dump valves in december.

The high level of consumption for the support function is due to the replacement of a terminal of unit 1 auxiliary transformer in june, and the repair of an oil leak in the unit 2 diesel generator set in february.

2 - SAFETY MANAGEMENT INDICATORS

Analysis of incidents reported according to the criteria defined by the French Safety Authority has shown, sometimes by comparison with the results obtained at the world level, that improvement objectives should be defined.

EDF's Nuclear Operations Division has fixed a 30% reduction of the number of declared incidents as objective for the 1996/2000 period. The Nuclear Operations Division has informed the managers of the power plants that this objective must more particularly be reached in the three domains that account for 2/3 of the incidents declared by the power plants.

1) Reactor scram

These incidents were 15 to 20% of all incidents in 1996.

One indicator is the mean number per nuclear site of days without reactor scram; it allows a stimulating comparison of sites.

Figures 3 and 4 show the evolution of the number of scrams per 7000 hours of criticality per unit per year (WANO indicator).

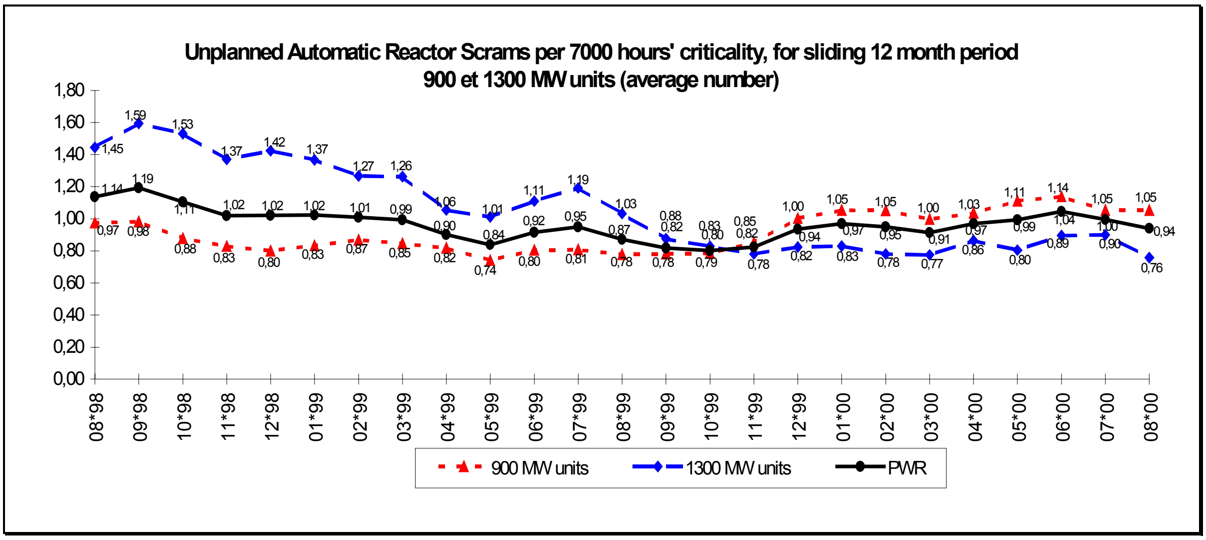


Figure 3: example of evolution included in the monthly safety board

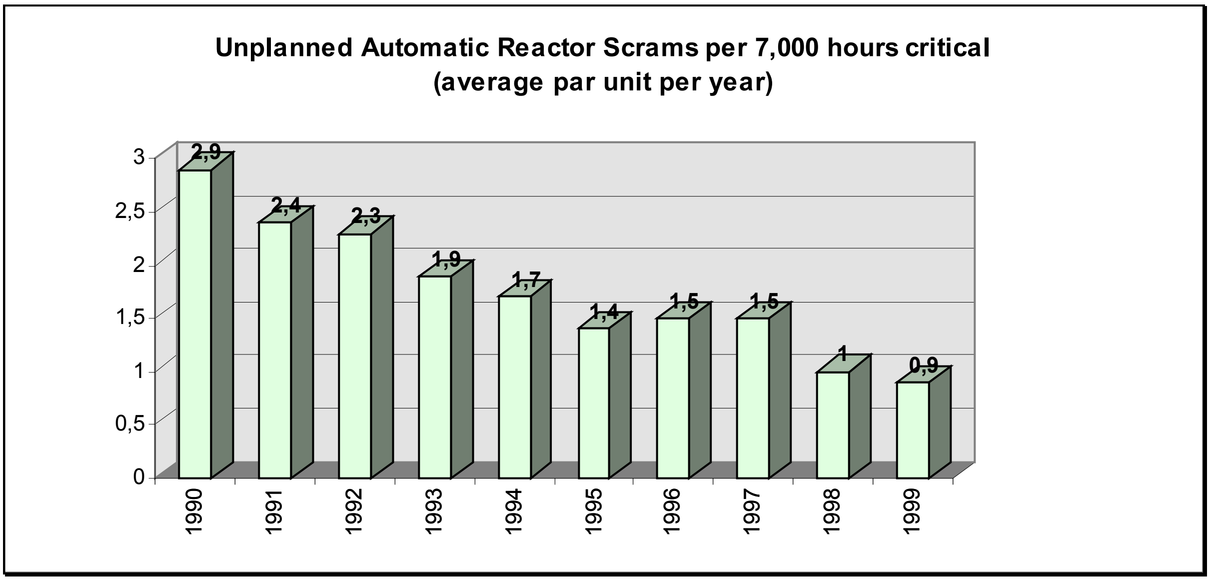


Figure 4 : evolution of ARS (1999 annual safety report)

2) Noncompliance with operating technical specifications

These incidents were approximately 40 % of all incidents in 1996.

The requirements contained in the operating technical specifications are defined in 1) above.

Noncompliance with operating specifications includes, for example:

- transgression of the limits of the domains of normal exploitation.
- failure to do what is required if a required item of equipment or system is down (overshoot of fallback time, etc.),
- the deliberate non-availability of a group 1 item of equipment or system when the equipment or system is required in the domain of operation the reactor is in,

Figure 5 shows the evolution between 1997 and august 2000 of the number of noncompliance with operating specifications per reactor per year.

Figure 6 shows an example of comparison between power plants.

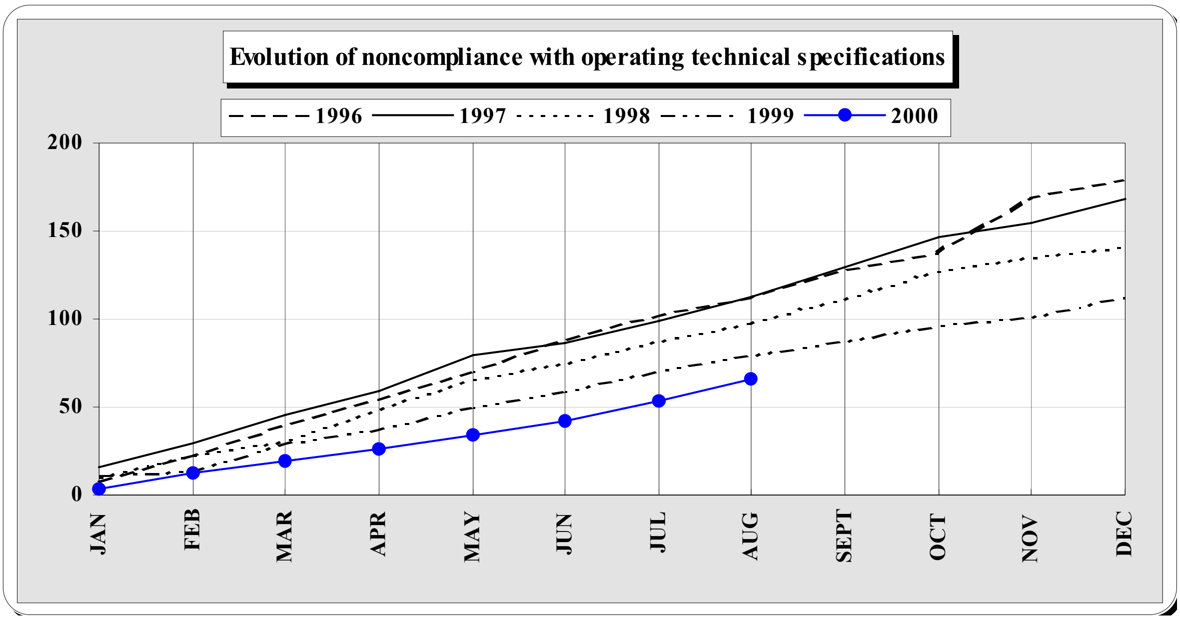


Figure 5 : example of evolution (monthly safety board)

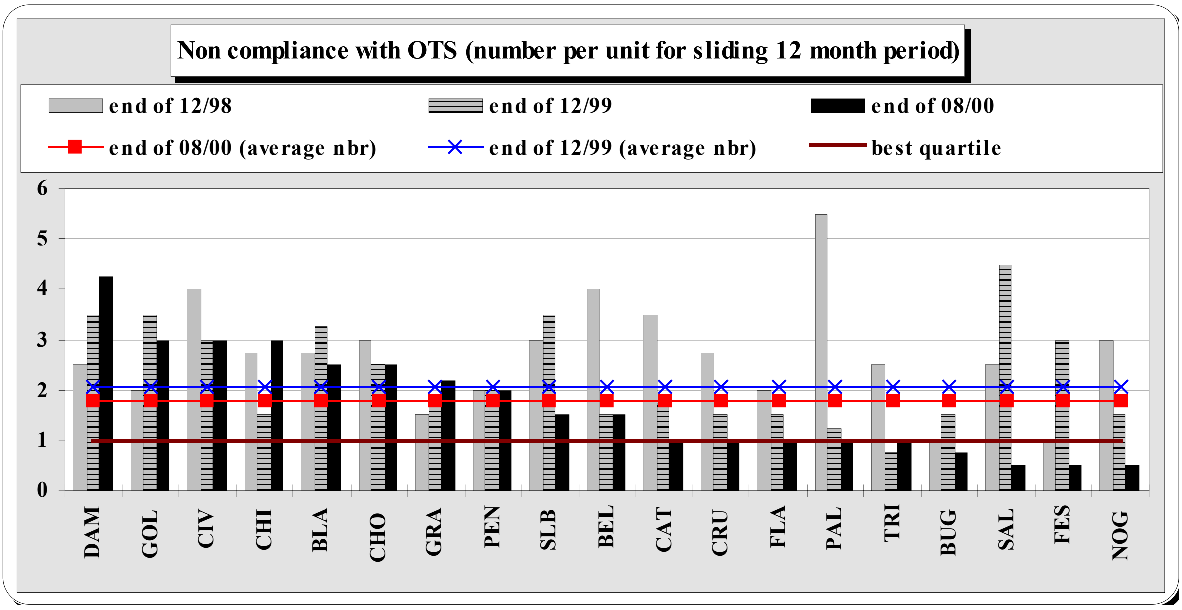


Figure 6 : example of comparison between units (monthly safety board)

3) Line up incidents

These incidents were 10 to 15% of all incidents in 1996.

These line up errors can make equipment or systems required by the operating specifications unavailable if, for example, manual valves have been set to the closed position when they should be open to ensure this availability.

Figure 7 shows the evolution of the number of incidents concerning line up errors per reactor per year.

Figure 8 shows an example of comparison between power plants.

Such indicators are used by local managment in their ways to reduce line-up incidents. They also serve the national level. For instance, EDF Nuclear Operations Division, which has decided to develop an experience sharing program, will focus on 3 or 4 units.

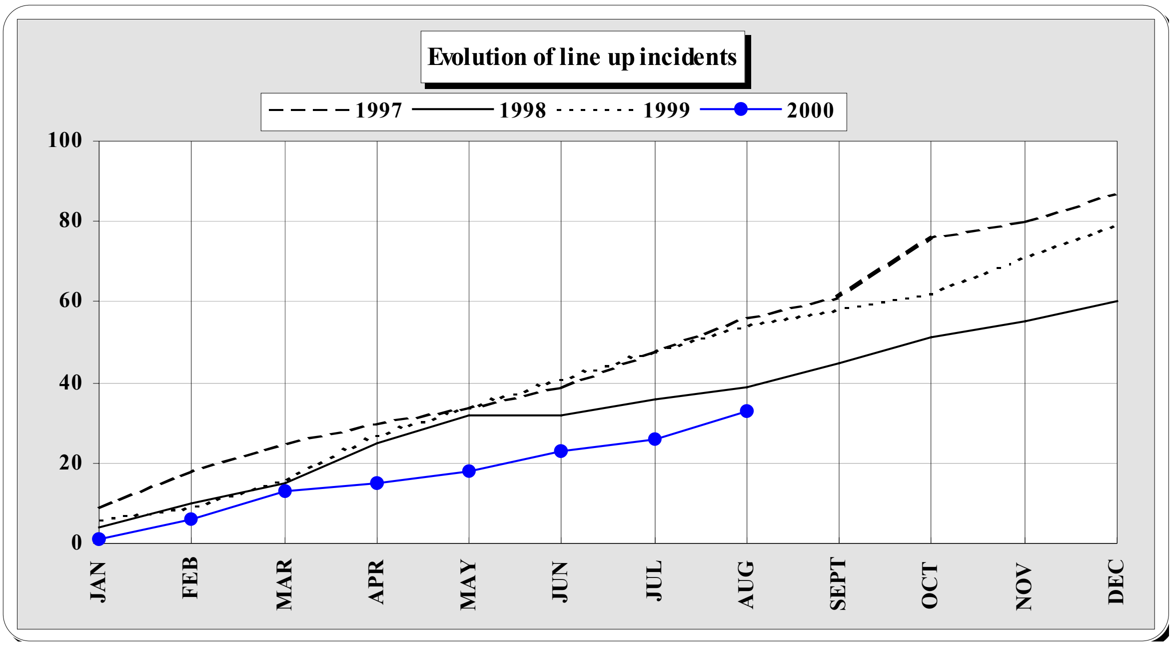


Figure 7 : example of evolution (monthly safety board)

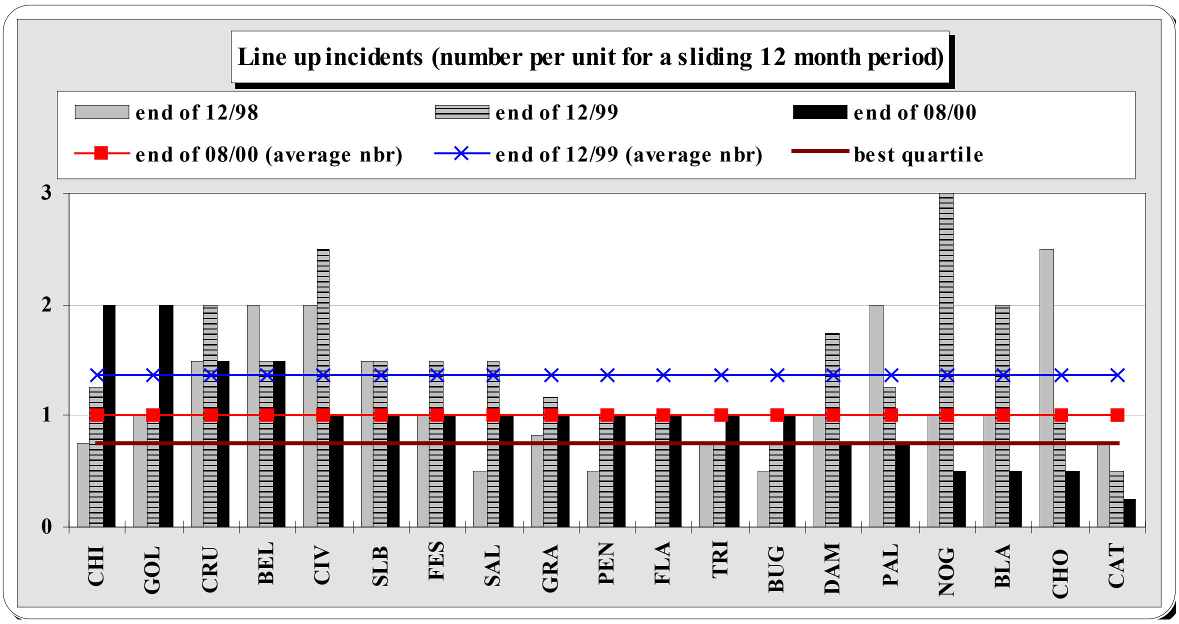


Figure 8 : example of comparison between units (monthly safety board)

3 - MEASUREMENT OF THE POTENTIAL CONSEQUENCES OF INCIDENTS

Probabilistic analysis of incidents consists of imagining, from the true situation of the power plant at the time of an incident, degradation scenarios that might lead to unacceptable consequences (core meltdown). Probabilistic quantification of these scenarios yields an evaluation of the protections remaining to avoid damage to the core during the incident. The conditional probability of damage to the fuel, given that the incident has occurred, is called the Potential Risk Index (PRI).

Incidents that have a PRI greater than 10 E-06 are called "precursor incidents" (precursors to accidents entailing damage to the core). These are the incidents of which the causes must be corrected first, to prevent their recurrence.

Among the incidents declared by the power plants according to the criteria defined by the French Safety Authority, some are the object of a probabilistic analysis.

There is a first selection of incidents "significant for safety". This selection is based on the potential consequences of the incident with respect to the risk of damage to the core or of significant radioactive releases. A probabilistic analysis of the incident is conducted whenever this is possible.

Two methods of analysis are used, and combined if necessary, according to the type of incident to be analyzed:

- if the incident is of the initiator type (sequences that can lead to damage to the fuel), the analysis consists of assessing the available lines of defense from the probabilistic viewpoint,
- if the incident is a degradation of defense in depth (situation where power plant systems are degraded), the analysis consists of evaluating the consequences if an initiator had occurred (whereas it did not).

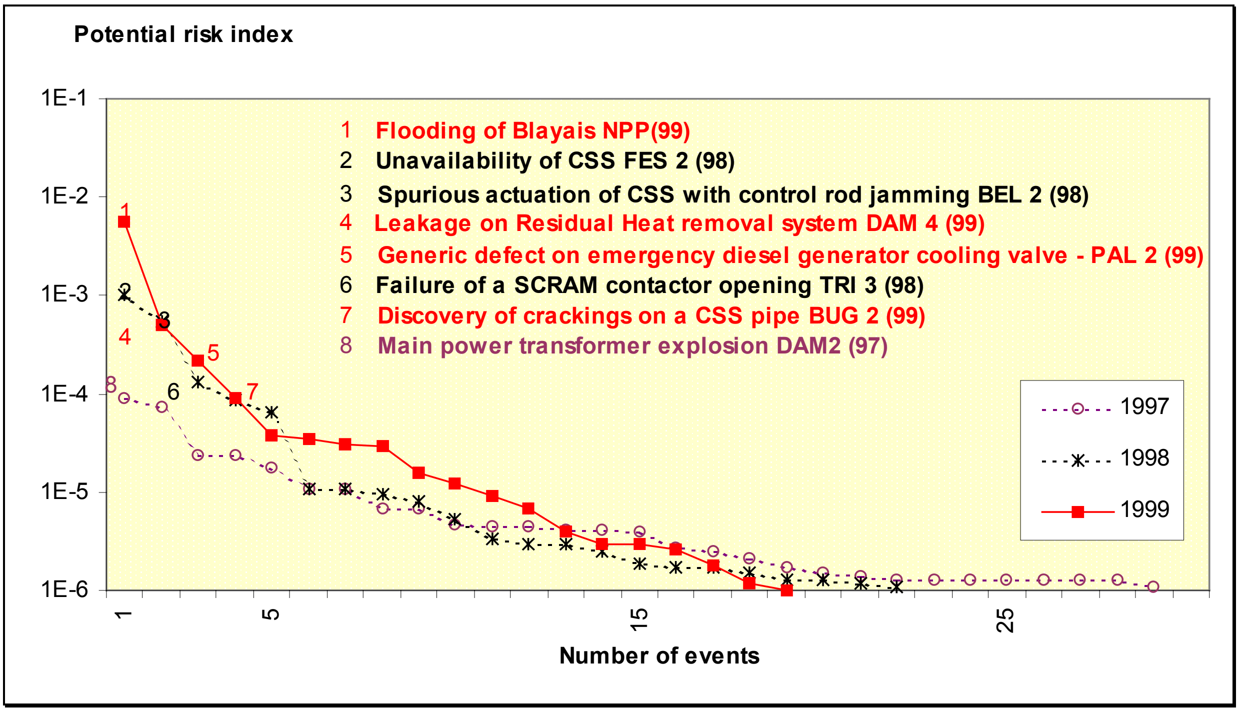


Figure 8

Number of precursor events by PRI intervals d'IRP:	1995	1996	1997	1998	1999
10^{-1} - 1					
10^{-2} - 10^{-1}					
10^{-3} - 10^{-2}	1			1	1
10^{-4} - 10^{-3}	2	5		2	2
10^{-5} - 10^{-4}	3	3	7	4	7
10^{-6} - 10^{-5}	27	25	22	15	9
Number of precursors events (PRI > 10^{-6})	**33**	**33**	**29**	**22**	**19**
Number of outstanding events with IRP < 10^{-6}	18	16	21	16	18
Number of outstanding events impractical to analyse	22	17	18	16	15
Total number of safety outstanding events	73	66	68	54	52
Total number of reported safety significant events	377	437	421	349	390

Table 1

Results

For the last five years (1995 to 1999):

- the number of precursor incidents has decreased although the number of reported incidents increased in 1996, 1997 and 1999 with respect to 1995.
- figure 9 and table 1 above show the distribution of precursor incidents for the last years and show that the importance of the most significant incidents has decreased every year.

These indicators show that safety has been improved in these last five years.

However, we must remain cautious in this interpretation because some incidents can not be the object of a probabilistic analysis, because probabilistic safety studies are limited to level 1, which covers only incidents related to damage to the fuel and does not deal with the behaviour of the systems participating in confinement and therefore with releases into the environment.

This probabilistic analysis is however a major factor in the a posteriori evaluation of the level of safety of a reactor population in that it makes it possible to have at least a partial image of the risk incurred.

4 - CONCLUSION

Measuring safety is no easy matter, and there is no simple indicator by which to do it.

It is only through a combination of a large number of indicators that it is possible to assess the level of safety of a reactor or of a set of reactors.

It is in effect necessary to measure simultaneously the physical condition of the installations and the dynamic of progress the operator applies. This dynamic of progress is a fundamental element.

In addition, probabilistic analysis of incidents can give an at least partial a posteriori image of the risk incurred by the operation of a large population of reactors.

Papers of Specialist Meeting on Safety Performance Indicators
October 17-19, 2000, Madrid, SPAIN

INTERNATIONAL PERFORMANCE INDICATORS And the UK Nuclear Electricity Generators

C M Atkinson
Performance Indicator and Event Analyst,
Operational Services

British Energy, Barnett Way,
Barnwood, Gloucester GL4 3RS
England

Tel: +44 (0) 14 52 65 40 40
Fax: +44 (0) 14 52 65 49 14
E-mail: carl.atkinson@british-energy .com

ABSTRACT

Since the beginning of the 1990's the UK electricity supply industry has undergone major changes moving from a wholly owned public utility to that of a multi company based activity, resulting in a fiercely competitive marketplace. In such a competitive market, where commercial pressures influence day to day decision making, it has been suggested that safety considerations could, in such an environment, be compromised.

Safety is, and always has been, the number one priority and commercial considerations always come second in this respect.

In the UK there are currently two companies generating electricity commercially by nuclear means, British Energy and BNFL Magnox Generation. The two companies co-operate and liaise with each other in a number of areas, one such area is in the field of performance indicators. Whilst this paper primarily presents a British Energy perspective, much of the content also applies to BNFL Magnox Generation and the principles described are essentially the same for both companies.

The paper describes how internationally comparable performance indicators form part of the UK nuclear electricity generators' performance enhancing measures to monitor and improve safety performance within the context of an increasingly competitive market.

The paper focuses primarily on five clearly comparable WANO Performance Indicators and shows how these indicators spearhead a suite of indicators that are used collectively in the companies' drive to improve their respective safety performance and as a consequence operational performance against the world's best performing reactors.

INTRODUCTION

In the late 1980's the British government announced its intent to move away from a publicly owned, fully integrated industry of electricity supply, preferring an industry in which competitive forces and commercialism prevailed. A number of companies were created and these new companies were moved into the private sector. Until 1996 the nuclear electricity generating stations remained in the public sector. The newer stations were then moved into the private sector in a new company, British Energy, whilst the remaining Magnox stations remained in the public sector operated by BNFL.

For some years now, therefore, the generation and supply of electricity within the UK has been within a competitive market. More recently, a review of the effectiveness of this competitive market was undertaken. It was concluded that the industry's customers could benefit from greater reductions in the price of electricity to those previously seen. The UK regulator of gas and electricity markets (OFGEM) considered what measures could be introduced to induce such benefits.

As a consequence new trading arrangements will shortly be introduced for the wholesale electricity market, to operate in England and Wales. The intent is to create a more commodity-like market to replace the existing pooling arrangements for England and Wales. With the new market arrangements there will be more active trading and participants will have a greater incentive to deliver planned generation and to forecast more accurately. Whereas in the past maximising output was the primary objective, reliability now becomes as important a consideration as that of maximised output.

World Class Operation

The progressive privatisation of the UK Electricity Supply Industry created a downward pressure on the price of electricity and a requirement to return a reasonable dividend to investors. The UK electricity industry, as in any other industry, therefore, has had to ensure cost effectiveness, whilst maintaining safe and reliable operation. In our industry particularly, safety is paramount and must remain our number one priority. Any measures employed in order to maximise commercial gain must <u>always</u> be underpinned by this requirement.

It has been questioned whether safety is potentially compromised in such a commercial world when the maximising of the profitability of a company carries such a high incentive. However, it has also been demonstrated that there is a direct relationship between safety and profitability and that the safest plants also tend to be the lowest cost generators. Many of these plants are essentially "World Class" when measuring their performance. In terms of the WANO performance indicators, many such plants are consistently within the top ten per cent or upper decile in a number of indiators.

In the UK BNFL Magnox Generation and British Energy are aiming towards emulating such "World Class" performance and actively support and participate in the WANO performance indicator programme. Performance is monitored against many of the indicators and the results are widely disseminated to staff at all levels on an ongoing basis. These indicators and others form an important and integral part of the companies' monitoring processes for safety and have the added benefit of supporting the objective of improving performance in terms of greater efficiency, output and reliability.

PAPER CONTENTS

Perception and acceptance of performance indicators and their results

In any modern company the complexities of the business require a vast number of processes using a variety of tools in order to function. A healthy company will question and review the use and value of such tools regularly to ensure that they remain focused and effective. This is an essential requirement in order to ensure that, not only is efficiency of the company maintained, but also that the workforce within the company remains focused and support enthusiastically the processes by using the tools in place.

Performance indicators as a tool are not exempt from this principle and it is essential that the workforce use such a tool effectively in order to maximise its benefit. However well intentioned, though, indicators can easily be viewed with some scepticism especially if the resulting data becomes meaningless or inaccurate, or can be seen as having little value or failing to meet their objective. Indicators can also be viewed with disdain if they are perceived as being ill-conceived or if the task of collecting the supporting data far outweighs the resulting benefit. It is therefore essential that performance indicators, whether internal or external to the company, are focused from their initial deployment onwards and remain so.

Regular consultation with those collecting or using the resulting data is clearly an important factor in encouraging "ownership" of any indicator, but the choice of indicator, its simplicity/complexity, and the ease of communicating the results clearly and effectively to those staff who can influence the outcome in an easily understandable format are also issues to address.

Consideration of these factors ensure that such “ownership” of the indicator tool within the workplace is secure. Without “ownership” the indicator tool becomes regarded as a bureaucratic chore and as such a meaningless, rather than a valuable, aid. With “ownership” the indicator tool acts as an aid to encourage an enthusiastic approach to the critical examination of the mechanisms within the company. Such critical examination supports the enhancement of both safety and operational performance.

The value of international indicators

For internal company indicators the principles outlined above are relatively easy to address. Indicators can be introduced, amended or deleted as appropriate and can be “tailored” to focus on particular requirements. These indicators can provide a clearly focused monitoring tool and tuned to indicate an improving trend, or, an early warning of a deteriorating and undesirable trend. Such indicators remain within the direct control of the company and “ownership” is relatively easily secured. For external, national and international indicators “ownership” is not so readily secured and other factors also need to be considered.

The main benefit, perhaps, of internationally used indicators is that they provide the potential for a company to draw comparison with its performance in any given indicator against that of other companies. This is an important additional factor not prevalent in internal indicators. International comparison is valuable not least because it reveals shortfalls and highlights the potential for improvement.

Whereas the previous “best” or a perceived standard becomes the benchmark to aim for in internal indicators, the world’s best performers become the benchmark with international indicators.

The main disbenefit of international indicators is, perhaps, that such indicators, to be of any real benefit, must be internationally supportable and directly comparable. Differences in reactor design, national regulatory body philosophy or even internal company processes can be problematical when considering the development and deployment of such indicators. Naturally such factors can cause difficulties in securing international agreement on a standard which can be applied equally to all, but it is essential that such agreement is achieved if the full potential benefit of an indicator is to be realised. Moreover, established indicators need to be reviewed at regular intervals to ensure that they remain focused.

In the UK the key international indicators used are derived from the WANO Performance Indicator Programme which currently supports ten internationally accepted indicators. These are:

- Unit Capability Factor
- Unplanned Capability Loss Factor
- Unplanned Automatic Scrams per 7000 Hours Operation
- Safety Systems Performance
- Thermal Performance
- Fuel Reliability
- Chemistry Performance
- Collective Radiation Exposure
- Volume of Solid Radioactive Waste
- Industrial Safety Accident Rate

The main benefit to the companies of external indicators is the potential for comparability. Companies can learn much from the performance of others and, importantly, recognise the limitations

With the exception of Sizewell B (PWR), the operating reactors in the UK have a design basis virtually unique, using CO_2 gas as the primary coolant. This aspect has an effect on comparability and therefore the primary benefit of some of the WANO indicators. The Fuel Reliability indicator, for example, is affected by the design differences of gas cooled reactors which can remove failed fuel more easily than other reactor types and do so. Moreover, improving the indicated values, of the Chemistry Indicator, for which the gas colled reactors appear are likely to appear poor when compared internationally, may actually cause boiler damage.

Two other indicators, the Solid Low Level Waste and Thermal Indicator are generally little used. For example, the Solid Low Level Waste Indicator is viewed in the UK as not being internationally comparable (one reason is because different countries have differing levels of compaction).

Five of the ten WANO indicators are clearly internationally comparable and are targeted to monitor against. They are:

- Unit Capability Factor
- Unplanned Capability Loss Factor
- Unplanned Automatic Scrams per 7000 Hours Operation
- Collective Radiation Exposure
- Industrial Safety Accident Rate

It should be noted that the new Forced Loss Rate Indicator to be trialled in the near future will also become one of those targeted by the company(ies).

How information is disseminated

Both companies in the UK have a desire to be amongst the best performers in the world, (“world class”). A ”world class” operator in these five indicators is generally deemed to be within the top ten per cent of the worlds reactors (Upper Decile). Moving towards “world class” does not happen overnight and is progressive. The aim, therefore, is to achieve upper quartile (top 25%) initially and then to progress towards upper decile. In support of this intention, it is important to make full use of the data available to the company. With regard to the indicators identified above, data is currently provided electronically to member utilities by WANO in the form of a spreadsheet. From this charts are created to show the respective positions of Upper Decile, Quartile and Median values together with the indicated values for each site or reactor. An example of one such chart is given in Figure 1 in the Appendix to this paper.

In this example a “snapshot” view is presented for Unit Capability Factor over the course of one year. The 420 or so reactor values are plotted from left to right, highest to lowest and markers are included to note Upper Decile, Upper Quartile and Median values. Against this the 15 British Energy reactors’ values are marked. It is important to note that the significance of the chart is not to provide rankings for the world’s population of reactors but to provide a clearly understandable and recognisable chart showing the degree to which any given indicated reactor must improve in order to achieve Median/Upper Quartile/ Upper decile (and hence World Class) status. These charts are revised as each new set of data is published and widely distributed within the company.

The chart in Figure 1, however, only shows a “snapshot” at an instant in time. The value of indicators is not necessarily only in the indicated value at one single point in time, but also in the change of that indicated value over time. Other charts are therefore also used that show this change. An example of such a chart is given in Figure 2.

Performance against these indicators is actively monitored within the company(ies) and they form part of the company internal acccountability process, which reviews performance against these indicators and others with those responsible for influencing them. The five WANO indicators identified above are essential to this process and spear-head the whole array of indicators used in this process.

Comparison is drawn against other nuclear generators external to the company and in so doing against world class performance. Supplementing these five WANO indicators are other internal indicators (some of which are used by other nuclear generators) covering a whole range of areas. Together, the whole suite of indicators forms a strong basis for monitoring both plant and personnel performance.

The indicators have been promoted within the company(ies) from the highest level and the interest from staff at all levels has been nurtured. Consequently, new data and charts are always keenly awaited. Display of charts by internal intranet is one way in which the latest results are disseminated, but they are also used in presentations to staff and team briefings.

As such the indicators form an integrated part of the company’s(ies) culture. Critical examination of the results encourages not only a review of the effectiveness of current practices but also provides an ongoing effective way of assessing the effectiveness of measures or initatives deployed to improve performance (identifying and addressing the root causes of unplanned losses or the personnel injuries, for example). Because they provide a very reliable and clearly communicable tool, the indicators are valued and form an essential and continual part of the company(ies) self-evaluation. Through the indicators, failings are quickly communicated and can be addressed and importantly, achievements can be rewarded.

Future Developments

Since December 1995 the IAEA has embarked upon a project to develop a framework for the establishment of operational safety performance indicators. A co-ordinated research project began towards the end of 1999 and is expected to be concluded in 2001/2.

The objective of the project is to assist nuclear power plants to develop and implement their own plant specific operational safety performance indicator programmes or to enhance existing programmes. Unlike the WANO indicators, the framework is not intended to produce an internationally comparable data source but is intended to pull together in a coordinated way the indicators (including the WANO indicators) used by a particular plant. The benefit of the framework is that it allows a “global” picture to be viewed capturing the performance on the whole spectrum of safety areas. The framework uses a heirachical structure and a simple diagram indicating how this might be structured is shown in Figure 3.

The framework is intended to highlight the change over time in the indicated values and significantly shows improving or deteriorating trends. In this way it acts as an early warning system indicating particular areas to be targeted for improvement.

The framework is essentially a self-evaluation tool, the development of which the company(ies) are actively participating in. The potential benefit from employing such a tool builds on work previously done within the company(ies) using WANO and internal indicators and is likely to enhance it, identifying any shortfalls in areas monitored.

In an increasingly competitive environment both companies value the range of international performance indicators available to maintain and enhance performance. They provide a standard to strive for, a target that both companies intend to meet.

Error! Not a valid link.

Unit Capability Factor - One Year Data Comparison between the Indicated Reactor and the Upper Decile, Quartile, and Median Values

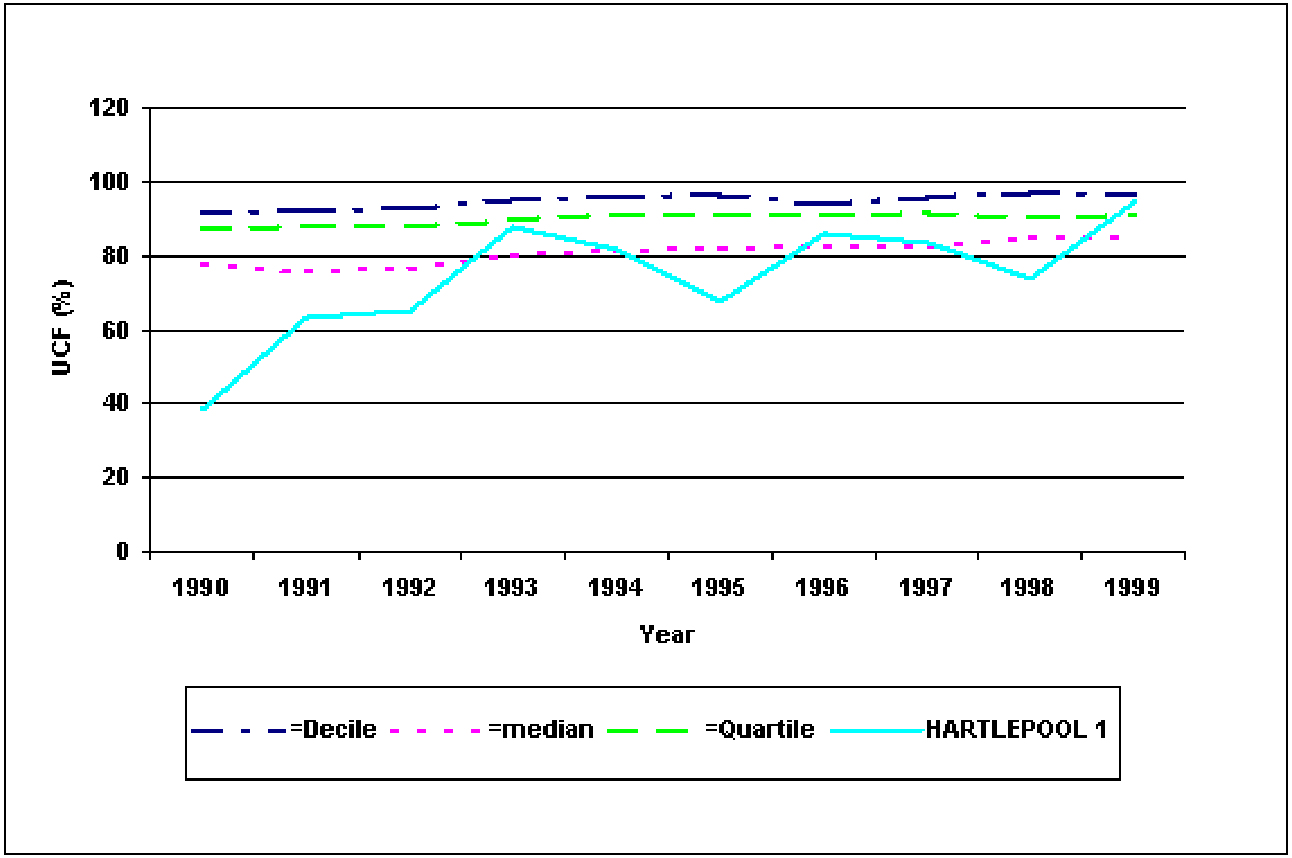
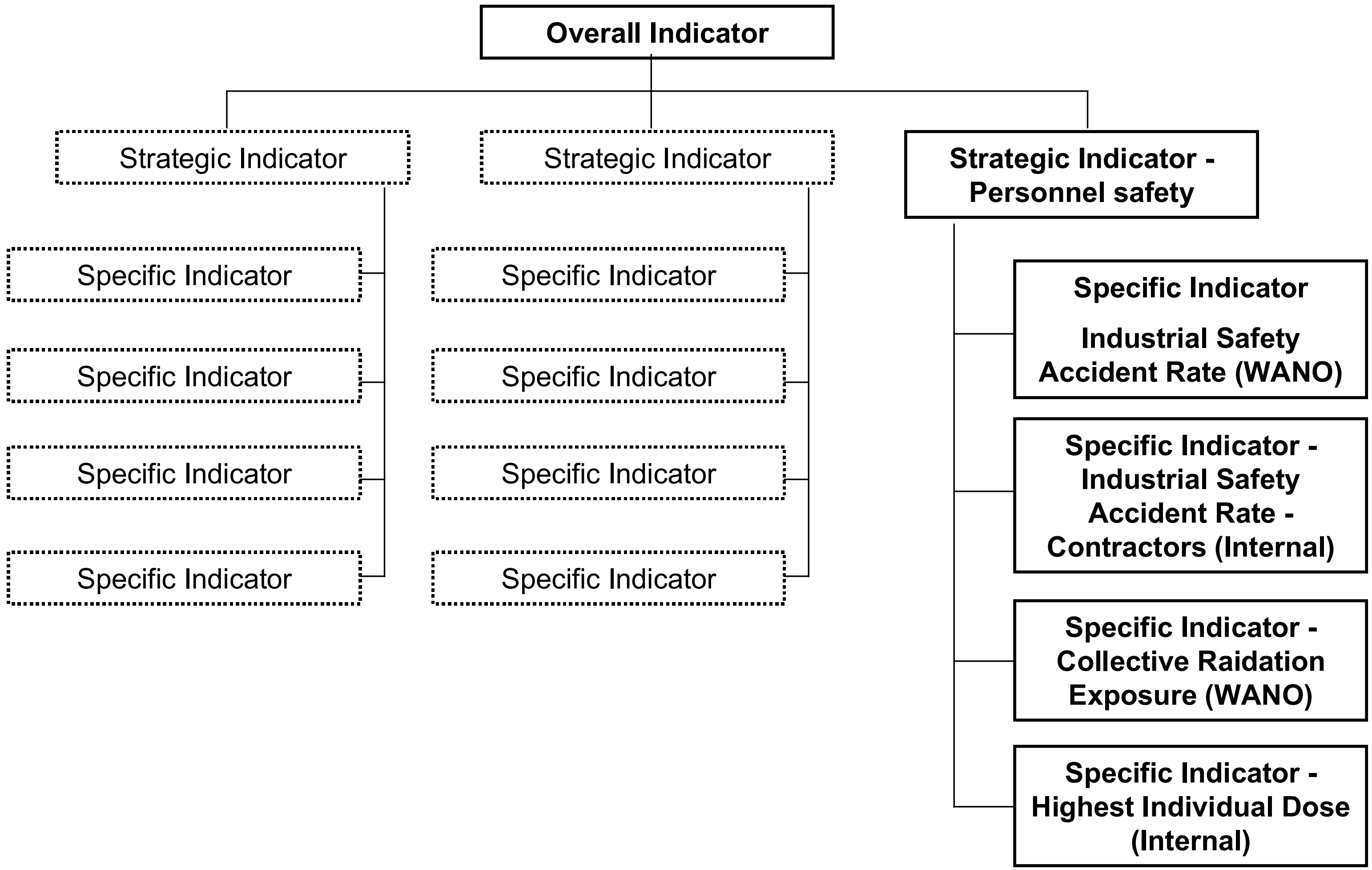


Figure 2



Overall Indicator
Strategic Indicator
Strategic Indicator
Strategic Indicator - Personnel safety
Specific Indicator
Specific Indicator
Specific Indicator
Specific Indicator
Specific Indicator
Specific Indicator
Specific Indicator
Specific Indicator
Specific Indicator
Industrial Safety Accident Rate (WANO)
Specific Indicator - Industrial Safety Accident Rate - Contractors (Internal)
Specific Indicator - Collective Raidation Exposure (WANO)
Specific Indicator - Highest Individual Dose (Internal)

Figure 3

Specialist Meeting on
Safety Performance Indicators
17-19 October 2000, Madrid, SPAIN

A NEW APPROACH TO DEVELOPMENT OF A RISK-BASED SAFETY PERFORMANCE MONITORING SYSTEM FOR NUCLEAR POWER PLANTS

M. Khatib-Rahbar
Energy Research, Inc.
6167 Executive Blvd
Rockville, Maryland 20852 USA
email: mkr@eri-world.com

R. T. Sewell
500 Orchard Drive
Louisville, Colorado 80027 USA

H. Erikson
Swedish Nuclear Power Inspectorate (SKI)
S-106 58 Stockholm, Sweden

ABSTRACT

A new hierarchical and risk-based approach to performance monitoring of nuclear power plants is proposed that is capable of accommodating several applications, including the identification and correlation of potential event forerunners that could signal deterioration in future safety performance. The basis and framework of the approach are presented, an example application and its results are discussed, and recommendations are made regarding actual plant data collection and analysis.

INTRODUCTION

A nuclear power plant (NPP) performance indicator (PI) is a basic parameter (whether described qualitatively or quantitatively) that is perceived as having potential meaning (or relationship) to a given figure of merit (e.g., economic value, power production, safety, etc.). For example, some PIs that have been proposed include: number of reactor scrams, number of safety system failures, the time between forced outages, the measured radiation exposure, etc.

If properly selected, PIs are useful in evaluating and comparing performance (e.g., over time for a given plant or group of plants, over a cross-section of plants at a given time, etc.), and in serving as the basis for making decisions that affect plant performance. Although PIs themselves are typically quantitative, their use in a decision process can be purely qualitative or, if appropriately conceived, based or guided (at least in part) on a more systematic, quantitative approach.

Currently, qualitative use of performance indicators is an essential element of efforts, by the nuclear industry and nuclear regulatory organizations, aimed at the activities such as: (1) monitoring performance; (2) identifying and rectifying potential degradations in performance; and (3) developing effective strategies for improving performance and enhancing operational safety of NPPs.

Various approaches to defining, collecting, monitoring, and reporting PIs have been proposed and/or implemented by various organizations, including the nuclear industry [1], regulatory authorities (e.g., see [2]), and the International Atomic Energy Agency [3].

The preponderance of such performance indicators can be characterized as "low-level" PIs. A "low-level" PI is described in terms of fundamental (and usually dynamic) plant events or conditions that are thought (or hypothesized) to ultimately impact overall performance (e.g., safety). Only to a lesser extent have industry and regulators proposed (at least within the performance indicator context) the development and monitoring of "high-level" performance indicators of plant safety. Such "high-level" safety performance indicators can be termed distinctly as safety indicators (SIs); they serve to more directly measure/quantify plant safety, or the safety significance of the low-level plant events or conditions, than just the PIs themselves.

Even though lesser attention has been given in the past to establishing a formalized PI and SI framework, the selection and implementation of a well-structured monitoring system for performance indicators and risk-based safety indicators is a crucial aspect of regulators' current endeavors to implement a meaningful process of risk-informed and performance-based regulations, and is equally crucial in corresponding industry safety management and decision processes.

OBJECTIVES

The principal objective of this paper is to present and recommend a new and advantageous system/process for risk-based safety performance monitoring of NPPs. The proposed approach and its development are substantially unique and versatile, and enable one to relate rates of occurrence of carefully chosen dynamic events and conditions (e.g., maintenance problems, safety-significant errors, etc.), that are potential accident forerunners manifest in safety culture and organizational procedures, to predicted quantitative and qualitative impacts on plant safety.

RISK-BASED APPROACH

In general, the worthiness of a risk-based performance monitoring system can be described in terms of the following desirable attributes:

1) Adequately represents plant <u>safety performance</u>, and <u>directly</u> relates such performance to risk and/or constituents of risk (reliability, availability, probability and frequency).

2) Identifies significant manifestations of <u>organizational</u> and other factors that could signal "deterioration" in safety performance before actual adverse safety impacts are realized.

3) Has small potential for spurious correlations.

4) Is implemented (defined and calculated) unambiguously and consistently across all plants.

5) Provides direct input to risk-informed and performance-based regulatory process.

6) Complements (and not replaces) other available means for assessing licensee performance (e.g., inspection activities).

In consideration of these desirable attributes, the focal point in developing the proposed approach has been on the effective use of probabilistic safety assessment (PSA), as PSA provides the formal and most logical means for quantifying the safety significance of operational events, corrective actions, design modifications, and configuration (plant condition) changes. In other words, PSA establishes a consistent framework for defining the most meaningful set of PIs, and for linking these with the most effective SIs. Risk-based PIs and SIs (i.e., those quantities collectively capable of relating failures or other plant occurrences to changes in risk) provide the most rational, quantitative, and uniform basis for comparing the safety significance of plant events and conditions (including their causes and trends) at a given plant, among a group of plants, or for the entire population of plants. (Stated differently, if any hypothesized low-level PI cannot be associated with a PSA input, or any proposed high-level SI cannot be related to a PSA output, its value to safety performance monitoring is substantially diminished since its relevance to safety cannot be clearly demonstrated or shown to be non-arbitrary.)

Accepting a living-PSA-based framework *specifically enhanced* for safety performance modeling facilitates the identification of those PIs and SIs that are important to track in a safety monitoring process. Simply stated, SIs are chosen indices of safety measured in a PSA (e.g., core damage frequency, containment failure frequency, etc.), and directly relevant PIs include only those conditions or events that act to rationally modify some PSA input parameter that (through re-quantification of the PSA model) has a consequential impact on the chosen SIs.

As already suggested, any PI that cannot be directly related, or at least meaningfully correlated, to a change in a PSA input parameter does not have a tractable effect on plant safety, and its use and implications are therefore unclear. Furthermore, it is important that a relatively small number of the best predictive PIs (e.g., those event forerunners that are related to, or affected by, safety culture and organizational performance) be used to augment the more obvious/explicit PSA-based PIs, so as to account for indicators at a still lower level than are normally considered in a conventional PSA that has not been enhanced with respect to performance monitoring.

At this point, it is important to distinguish the differences between the enhanced "performance monitoring system" described here, versus what has been recently referred among the nuclear profession as a "safety and risk monitor."

A "safety and risk monitor," typically implemented as a living PSA model, tracks the safety dynamics of a nuclear power plant due to changes in plant configuration and plant operational procedures.

On the other hand, the enhanced "performance monitoring system" is designed to track and/or predict the safety dynamics of a nuclear power plant due to the occurrence of (categorically described) safety-relevant events or conditions (including event forerunners manifest in safety culture and organizational influences that are not factored into a living PSA); and, as such, the objectives and implementation of this enhanced "performance monitoring system" differ importantly from a "safety or risk monitor."

Hierarchical Use of Performance Indicators

Some potential candidates for safety-relevant PIs include rate measures (e.g., number of events per reporting time period) of the following occurrences [4]:

- Initiating event (internal or external)
- Component unavailability or multiple related unavailabilities
- Safety function unavailability
- Safety system unavailability
- Train unavailability or multiple train unavailabilities
- Other significant events or collectively significant related occurrences
- Balance of plant (BOP) failures
- Personnel errors
- Non-power events
- Other potentially significant events or event forerunners

As will be introduced here, for performance monitoring purposes, safety-relevant PIs are most effectively implemented by categorizing events and conditions (i.e., the raw performance data) into a matrix of *"type"* of PSA impact and *"level"* of potential risk significance. The PIs of interest, then, are the number (or other quantitative description) of reported events/conditions that belong to a given category of occurrence. The matrix categories of PI types and levels are summarized, as follows.

PI Types

In the present approach, the proposed PI event types are defined as:

Type-A: Initiating event; or changes, in a known way, an initiating event frequency;

Type-B: Functional unavailability; system unavailability; train unavailability; or changes, in a known way, the failure rate of a safety function, system, or train (directly, without being attributable to a change in component failure rates);

Type-C: Component unavailability; or changes, in a known way, a component failure rate; and

Type-D: Correlates with a change in a failure rate (of a safety component, train, system, or function) or an initiating event frequency.

The purpose of categorizing PIs into these types is to map the myriad of possible events and conditions into a manageable set of groups; and to provide a means by which one can obtain greatest insight as to the source of changes in the relevant SIs.

In the foregoing classification scheme, the Type-D class of events is a very important and unique aspect of the proposed performance monitoring system that helps capture the ability to (a) predict *future performance*, and (b) identify manifestations of *safety culture and/or organizational influences* and other factors that could signal *deteriorating safety performance* before actual adverse safety impacts are realized. Additionally, Type-D events are expected to occur most frequently among the four types of events cited above.

To implement the Type-D indicators into the performance monitoring system, an approach that develops quantitative relationships between the Type-D PIs and changes in PSA input parameters, has been developed [5], as described later in this paper.

In the foregoing classification scheme, functional, system, or train (FST) failures are all embedded as Type-B events for the following reasons: (1) in general, few occurrences of such events are expected; (2) the number of PIs that are tracked becomes reduced; (3) the significance of such events are conservatively highlighted by treating them as potential functional failures; and (4) the data searching capabilities of an automated performance monitoring system will enable the analyst or decision maker to, if needed, readily extract the precise nature of the event/occurrence that led to the change in the PI. Also, to help simplify the PI system, any condition which results in a known change in the frequency of an initiating event, FST failure rate, or component failure rate is accounted, respectively, as a Type A, Type B, or Type C event, since it would have an impact on the PSA model that is similar to an actual unavailability. (In other words, the frequency in each such case changes, not to unity, but to some value between the baseline PSA value and unity.)

PI Levels

The proposed PI levels, defined in general terms of their potential risk significance, are:

Level-1: High potential relative increase in a SI,

Level-2: Moderate potential relative increase in a SI; and

Level-3: Low potential relative increase in an SI.

The purpose of categorizing PIs into levels is to enable one to be able to distinguish the significance of each PI. In this way, the source of a marked SI change can be readily tracked to either a single significant event, a number of events/conditions of lesser significance, or something in between.

Evaluation of PIs

The general procedure for evaluating safety-related PIs, and determining corresponding changes in SIs, on a periodic basis, consists of the following steps:

1. Collect raw data on the various occurrences (events and conditions) relevant to the PSA model, including the predictive Type-D events.

2. Perform an analysis of plant events, failure modes, and potential causes of observed failures (e.g., as in the NRC's Sequence Coding and Search System [SCSS]).

3. Perform the categorization of each event/condition into its respective PI matrix element. Accumulate changes to PIs based on all occurrences.

4. Re-evaluate each SI based on the cumulative effect of the changes in PIs.

Figure 1 provides an overview of this initial step in implementing the risk-based performance monitoring system. In this figure, it is seen that the raw data (plant events and changes in plant conditions) are mapped into a PI matrix element ij. Type A, B, and C events get directly assigned to the appropriate PI level. Each event/condition identified as a Type-D event (i.e., a potential forerunner of more serious events) gets indirectly assigned to a PI level (matrix element, 4j), by application of a specific event forerunner "correlation" with PSA inputs. (The development of such "correlation" is derived from a Bayesian approach, whereby each monitored realization of Type-D events is treated as a "sampling" of a plant's safety culture, and is used to "update" applicable reliability measures. The basis and implementation of this approach is discussed in more detail subsequently and in Ref. [5].) The effects of multiple events mapped into a given PI element are directly cumulative. In turn, the aggregate effects of all elements PI_{ij} are then accumulated for determining the change in SI_k, the desired high-level result of the performance monitoring system. (Stated more accurately, the proposed performance monitoring system uses a PSA-derived database to efficiently associate matrix values of PI_{ij} to their corresponding impacts on each SI_k.)

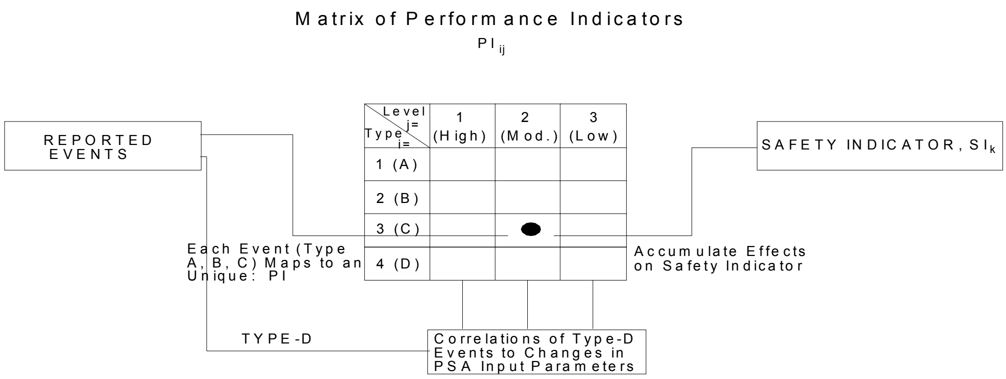


Figure 1 Schematic overview of the a risk-based safety performance monitoring process

As performance monitoring progresses, when an anomalous trend in any SI_k is observed, the anomaly can be traced back to a corresponding anomalous behavior in a given PI element or elements. The type and level of the PI provide key insights as to the ultimate source of the anomaly (which can, if desired, be further tracked back to its source[s]). It is anticipated that the frequency of raw events will increase, roughly, both with increasing PI type and with increasing level number. Much of the change in an SI may typically be associated with changes in event forerunners. Therefore, as an alternate way of using the performance modeling system, changes in demonstrably important event forerunners could also be tracked independently. By means of the event forerunner correlations (an enhancement that complements the PSA), combined with the inherent PSA-governed PI-SI relationships, any anomalous trends in the event forerunners can then be assessed as to their ultimate impacts on PIs and SIs.

Type-D PIs

Perhaps the most novel element of the proposed approach is the concept of identifying and relating Type-D event *forerunners* to changes in PSA inputs. The process for making use of event forerunner-PSA input relationships consists of the following steps [5]:

1. Collect raw data on the various occurrences of event forerunners.

2. Apply an appropriate event forerunner relationship to determine the implied change in PSA input (e.g., component failure rate, human error rate, etc.).

3. Categorize the change into its respective Type-D PI level. Accumulate changes to PIs based on all such changes derived from event forerunner relationships.

4. Re-evaluate each SI based on the cumulative effect of the changes in the Type-D PIs.

This process requires that the important (best predictive) set of Type-D events be defined and that relationships between the number of such events and changes in PSA reliability measures (i.e., component failure rates and initiating event frequencies) be developed. Initially, this requires an investigation to identify preliminary indicators and relationships to start the performance monitoring process. Over the long term, an important objective is to make use of performance monitoring experience to refine the selected events and predictive functional relationships. Correspondingly, there is a need to determine the long-term data collection strategies that will enable improvement of the Type-D event forerunner relationships. (These preceding points are emphasized to the reader. Advancement in performance monitoring depends critically upon first defining a meaningful stating point in data collection [e.g., what data to collect, as well as when and how to best collect that data], and then evolving the performance monitoring process through subsequent improvements suggested by the collected data. To that end, the proposed functional relationships and performance monitoring framework are intended to be viewed as a highly rational starting basis, as opposed to a final, unchanging set of relationships for performance monitoring.)

Based on an evaluation of several alternative methods, an expert elicitation approach was formulated and used [5] to develop the desired initial set of relationships between Type-D PIs and PSA inputs. This multiple-expert elicitation approach is summarized as follows.

The fundamental premises of the approach include:

- There exist a set of low-level PIs (i.e., descriptions of operational or organizational variables) that, given adequate data, could be shown to correlate with reliability (failure likelihood) at the component level.

- Lacking adequate data at the present time, the most informed basis for identifying such indicators is through multiple-expert opinion.

- The identified (from multiple-expert consensus) safety-relevant PIs can be used to characterize or measure a plant's attentiveness to safety, which (for convenience) can be referred to as "safety culture."

- Safety culture is thus a description that has a statistical relationship to reliability.

In other words, what is suggested here is that safety culture can serve as a convenient and effective link or "pinch point" between the quantification of well-selected Type-D performance indicators and the determination of reliability impacts.

In stating these premises, it is recognized that there have been various proposed measures or descriptions of safety culture that have served various purposes and that do not necessarily match the definition of safety culture presented here. For this study, however, categorization of safety culture is introduced for convenience, and safety culture is considered to have value to the extent that it relates to plant reliability.

Relationships to Safety Culture

The relationships between safety culture and reliability measures (i.e., component failure rates and initiating event frequencies) are described by conditional probability distributions that are characterized by a mean, standard deviation, and distribution type, as determined through Bayesian analysis. Plant-to-plant (population) data on component failure rates are used to define generic (prior) distributions or generic distribution parameters. Presumably, this data represents a composite experience for plants having varying safety cultures. It is thus reasonable to assume that, when conditioned on safety culture (e.g., as "sampled" by a recoded set of Type-D PIs for a given reporting time period), the failure rate distribution will not be equivalent to the unconditional distribution, but will be altered (updated) with respect to the unconditional distribution in a consistent way. For instance, conditional upon a poor safety culture, one would probably expect increased mean failure rates with respect to the unconditional case. Further, to roughly preserve the overall standard deviation of the unconditional distribution, reductions in the standard deviation for the safety-culture-conditional distributions (relative to the standard deviation of the unconditional distribution) would also be expected.

Consistent with the preceding concepts, the description of updated safety-culture-conditional probability distributions is facilitated by relating the conditional mean and standard deviation (μ', σ') to the unconditional mean and standard deviation (μ, σ). Further, it is convenient to model/describe the conditional means as:

$$\mu_i' = \alpha_i \, \mu \qquad (1)$$

and the conditional standard deviations as:

$$\sigma_i' = \lambda_i \, \sigma \qquad (2)$$

where α_i and λ_i are constants applying to the i^{th} category of safety culture.

In this formulation, it is important to note that a condition that the composite of moments of the conditional distributions should be precisely equivalent to the moments of the unconditional distribution would impose specific mathematical requirements on the (μ_i', σ_i'). However, our objective is not necessarily to strictly preserve the unconditional distribution, but rather, to use it as a rough guide to help ensure that the present determination of relationships of failure rates versus safety culture are tied to a reasonable and justifiable scaling basis.

Therefore, the several experts who participated in this study were not asked to directly identify the performance indicators that convey the notion of safety culture, but rather, to first select the indicators they believed had the most direct and significant impact on reliability measures. Subsequently, they were asked to relate values of the selected most-promising safety-significant performance indicators to meaningful designations of safety culture, and then to relate states of safety culture to quantitative influences on reliability measures.

Figure 2 illustrates the hypothesized influence of safety culture on a component failure rate or an initiating event frequency. This figure indicates that the (prior) probability distribution of a reliability measure, as used in a conventional PSA, is based on data reported from the experience of several plants, some having good safety cultures, some having average safety cultures, and perhaps some having poor safety cultures. Conditional probability distributions of the reliability measure for each category of safety culture, which collectively comprise the entire unconditional distribution, can be conceived. With information/sampling on safety culture (obtained from the values of the key performance indicators), therefore, the conditional (updated) probability distribution of the reliability measure can be refined.

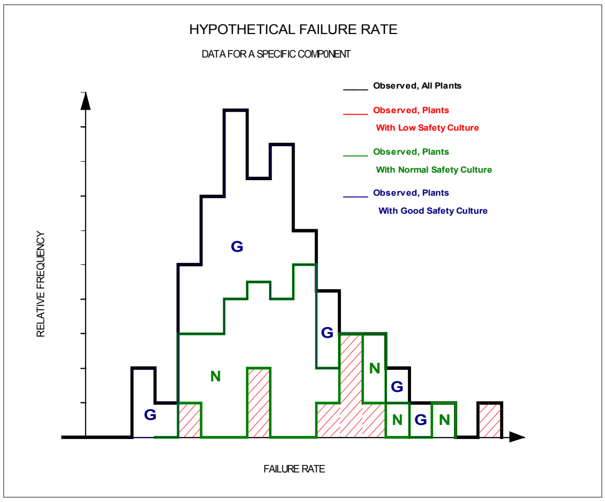
Expert Elicitation Process

The expert elicitation process was designed to consist of a combination of a multiple-phase questionnaire and workshop, summarized as follows:

1. A Phase-1 questionnaire was developed to identify the most promising independent variables, to assess the feasibility of

proposed methods, and to finalize the format of the relationships based on Type-D indicators.

(a)



(b)

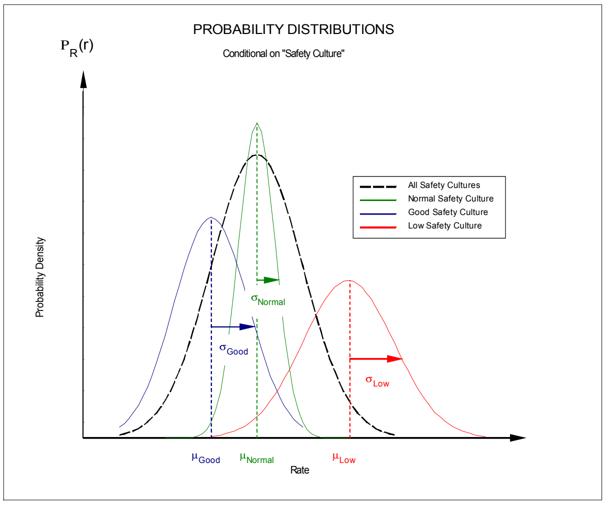


Figure 2 Hypothesized safety-culture conditional probability distributions comprising the unconditional distribution of a reliability measure (a) histogram form and (b) continuous form.

2. A Phase-2 questionnaire was developed to determine preliminary relationships between Type-D indicators and reliability measures.

3. A workshop was designed to enhance communication with experts; to ensure a meaningful technical exchange; to reach consensus on the format and results of the desired relationships between (a) performance indicators and safety culture, and (b) safety culture and impacts on reliability measures; and to note and address any caveats expressed on the part of the experts.

4. A revised Phase-2 questionnaire was developed based on the expert feedback received during the workshop, as the basis for final development of the desired relationships.

A total of 21 experts, representing Swedish licensees/plant staff, Swedish Nuclear Power Inspectorate (SKI), vendors, research organizations, consultants, and other non-Swedish licensees and regulatory authority staff participated in the study.

For each of eleven (11) key performance indicators selected and defined during the workshop, the experts provided their assessments, in the revised Phase-2 questionnaire, as to their degree of belief that the indicator has a relationship to changes in reliability measures. For each indicator, the mean value of the degree of belief was determined in order to characterize the relative worth of the indicators [5]. The results varied from the lowest-worth value of 53 to the highest-worth value of 87. Although a relative worth value in excess of 50 suggests at least some degree of agreement that the indicator correlates with reliability, values less than roughly 65 to 70 were not considered to constitute a strong endorsement of an indicator.

Based on the analysis of the final responses, a final list of five (5) PIs was derived, as summarized in Table 1.

Table 1 Final list of Type-D PIs

Type-D Performance Indicators	
1	Annual rate of safety-significant errors (i.e., reportable violations of technical specifications) by plant personnel, contractors, and others.
2	Annual rate of maintenance problems (defined as maintenance rework or overdue maintenance).
3	Ratio of corrective versus preventative maintenance work requests (MWRs) on safety equipment.
4	Annual rate of problems (deviations/failures) with repeated root cause (i.e., a cause previously identified by a vendor, the plant, another plant, the regulator, etc., for a similar plant or group of plants, or for similar components).
5	Annual rate of plant changes that are not incorporated into design-basis documents by the time of the next outage following the change.

As already discussed, reliability measures (component failure rates and initiating event frequencies) are generally described by means of a marginal (unconditional) probability distribution, whereas an underlying joint probability distribution function of reliability and safety culture can be used to refine the characterization of reliability through determination of distributions conditional on safety culture. Table 2 presents the expert-based assessment as to the change in mean value for the conditional distributions relative to the mean of the marginal distribution. Hence, the mean factors in Table 2 multiply the mean marginal reliability to obtain the refined conditional reliability. Without specific information regarding the influence on the conditional standard deviation, it is reasonable to assume the same standard deviation for the marginal distribution of reliability and the safety-culture conditional distributions.

Table 2 Relationship between safety culture (SC) and statistics (mean and coefficient of variation) of the change in mean frequency of failure or initiating event

Statistics	Safety Culture (SC)				
	Superior	Above Average	Average	Below Average	Inferior
Mean Factor Multiplying the Mean Frequency	0.4	0.6	1.0	1.75	2.5
Coefficient of Variation of Factor Multiplying the Mean Frequency	0.6	0.4	0.1	0.4	0.6

The expert-developed relationships can be refined through collection of data over a period of time, and re-application of the Baysian updating approach [5].

EXAMPLE RESULTS

The results of the example application described herein are based on:

- Simulated (not real) event data derived from examination of events in a real Licensee Event Report (LER).
- PSA impact table derived from an actual study.
- Use of expert-based Type-D PIs and correlations with reliability measures, as summarized in Table 2.
- Quantification of Type-D impacts derived from simulated PI data (i.e., a simulated "sampling" of safety culture).

There are two alternative approaches to development of a performance-impact database, for use in binning various plant occurrences into the PI-type/level matrix. The first approach involves using information obtained from extensive re-quantifications of the baseline PSA model. The second approach, which is utilized for this example, involves using a combination of component-level risk achievement worth importances (hereafter referred to as risk increase factors), initiating event core damage frequency (CDF) contributions, dominant accident sequence cutsets, and knowledge of plant design and operation. With this latter approach, only limited re-quantifications of the PSA model are necessary to construct the database, and then this database is used to actually implement performance monitoring.

The first step in developing the performance-impact database for the present case study involved constructing a table containing the different components, trains, systems, functions and initiating events (CSFIs) included in the PSA model, categorized by their PI-type. Each CSFI is represented by its event identifier from the PSA model.

The next step undertaken involved determining the numerical impact that each CSFI has on its associated PSA parameter. For initiating events, the associated initiating event frequency is set to one per year. For other CSFIs, the particular CSFI is assumed to be failed (i.e., its unavailability is set to a value of 1.0).

Using the information obtained in the previous step, the impact level for each CSFI was established.

Two methods are available for obtaining the impact level for initiating events. In the first method, the conditional core damage probability (CCDP) could be calculated for each initiating event. If core damage frequency (CDF) results are available for each initiating event, as for the case study, then the CCDP is simply obtained by dividing the associated CDF by the initiating event frequency (per year). If such results are not available, or easily obtainable, then the PSA model can be requantified by setting the subject initiating event frequency to 1.0 per year, and all other initiating event frequencies to zero. The resulting CDF will be equivalent to the CCDP for the subject-initiating event. Once the CCDPs are obtained for all initiating events, the CCDPs should be rank ordered, and engineering judgment can be used to subdivide the range of CCDPs into the three impact levels (i.e., high, moderate, and low).

In the second method, a qualitative level assignment can be made, as was done for the case study. All initiating events that result in a LOCA or significant reduction in the mitigating ability of the plant (e.g., loss of offsite power or loss of service water) were assigned to the high impact level category. These types of initiating events typically have a relatively low occurrence frequency. All initiating events that result in little or no reduction in the mitigating ability of the plant (e.g., general transients or turbine trips), were assigned to the moderate impact level category. These types of initiating events typically do occur with some regularity. Lastly, any initiating events considered in the PSA, which require some additional failure to occur in order to result in a reactor trip (e.g., a single turbine trip coincident with failure of the plant runback system), were assigned to the low impact level category.

The impact level for each safety function, system, train or component, was quantitatively assigned based on the corresponding risk-increase factor (RIF). The appropriate RIF was typically either obtained or estimated from the existing PSA importance results. A rank-ordered listing of the RIF for the top few hundred components was obtained directly from a PSA computer model. It was first determined that all components with RIFs below 1.01 have no quantifiable safety significance. Engineering judgment was then used to sub-divide the retained range of RIFs into high, moderate and low categories, as follows:

High (Level 1)	RIF $\geq$ 3.00
Moderate (Level 2)	1.30 $\leq$ RIF < 3.00
Low (Level 3)	1.01 $\leq$ RIF < 1.30

It should be noted that the boundaries selected for the different impact levels are somewhat arbitrary, and depending on the actual RIFs obtained for a given PSA, engineering judgment can be used to redefine these boundaries (based on, for example, a finding that a RIF clustering trend is noticeable).

Using these defined boundaries, the case study resulted in nearly an equal number of components being assigned to the high, moderate, and low impact categories.

The impact level for each safety system (or safety system train) was assigned based on correlation with the RIFs of components within the system or train. Typically, the RIF for a safety system or train was taken to be the highest RIF associated with a component which could fail, by itself, the subject system or train. (Due to the explicit modeling of common cause failures as basic events in the case study PSA model, single-point failure events were available for all systems modeled.)

The impact level for each safety function considered in the PSA was assigned to the high category, based on engineering judgment and knowledge of the PSA. For confirmation, the PSA model was requantified once for each safety function, with the unavailability for the subject safety function set to 1.0. The resulting total CDF for each requantification was divided by the baseline PSA total CDF to obtain the appropriate RIF. The RIFs associated with each safety function obtained by requantification ranged between 300 and 6000, thereby confirming that each safety function should be assigned to the high impact level category.

One of the most useful applications of the PI monitoring system is the trending of historical plant safety performance, by tracking changes in PIs and SIs. As performance monitoring progresses, when an anomalous trend in an SI is observed, the anomaly can be traced back to a corresponding anomalous behavior in a given PI element, or elements.

The plant events included in the case study typically involved Type-C PIs. Using the information provided in the event database, the PSA impact table, and the performance-impact table, the occurrences of the Type-C PIs were calculated [4] on a monthly basis, as shown in Figure 3. The SI considered for the case study is core damage frequency (SI_1), and is trended by plotting the RIF for each event occurrence. In Figure 3, the change in SI_1 (ΔSI) is tracked monthly through calculation of an averaged monthly RIF. The averaged monthly RIF is calculated by weighting the RIF associated with each event occurring in a given month by the event duration. Note, for functional failures of standby equipment, the average fault exposure time must be added to the event duration.

Review of Figure 3 shows, as expected, that PI has the greatest impact on ΔSI, while PI_{31} has the least impact. Obviously, the greater the PI impact level, the more closely it should correlate with ΔSI.

It should be noted that the correlation between the PIs and ΔSI in Figure 3 is not precise, since the tracking of PIs is only based on number of occurrences, and does not account for the duration of the event, which has a direct influence on ΔSI.

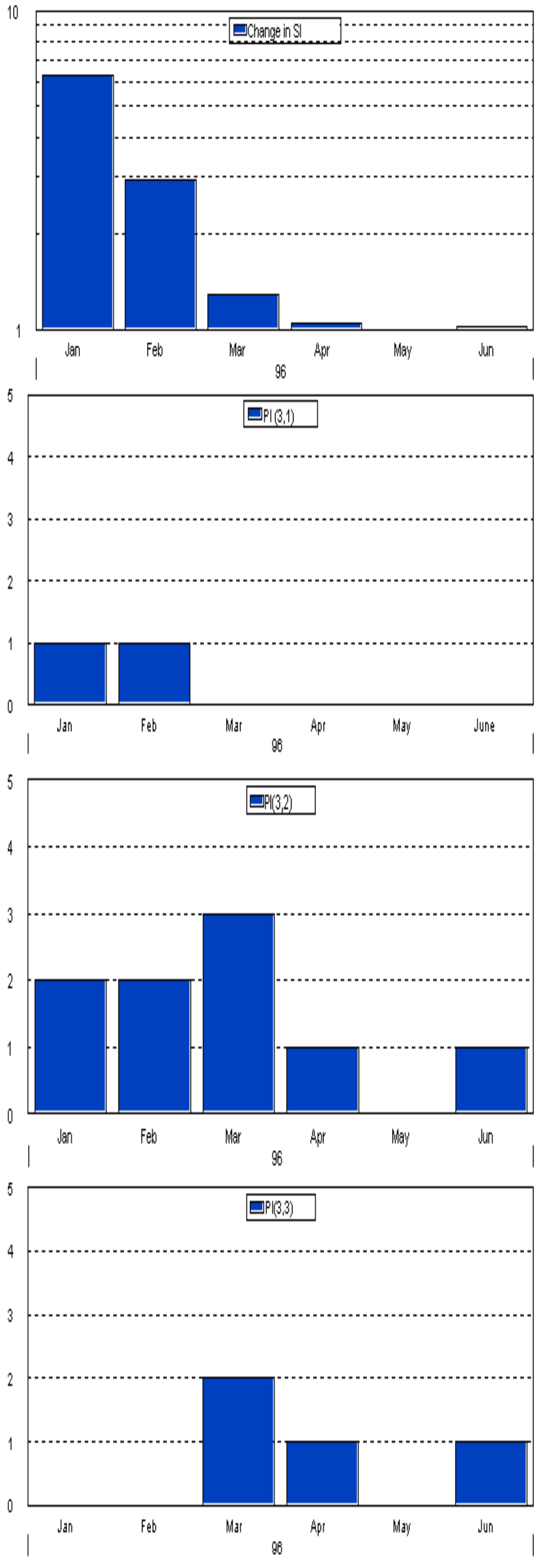


Figure 3 PI results (per monthly average)

Also, applying the Type-D correlations between safety culture and the changes in mean failure frequencies and initiating events, the impact on CDF is demonstrated in Figure 4 for the example plant considered here.

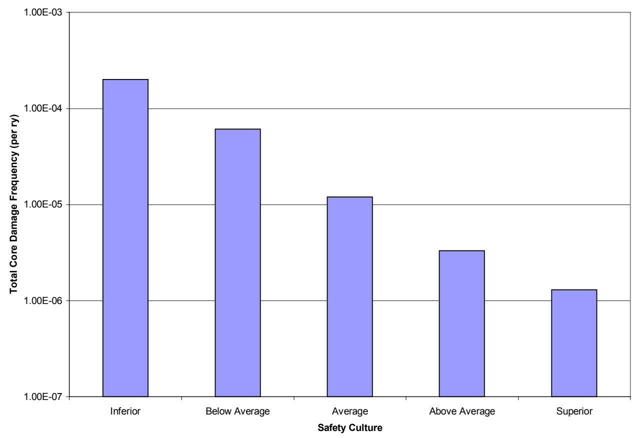


Figure 4 Impact of safety culture on core damage frequency (SI)

CONCLUSIONS AND RECOMMENDATIONS

The approach discussed here has established a consistent and logical basis for developing a risk-based performance indicator system, including event forerunners, used to detect early signs of deterioration in safety performance (i.e., Type-D PIs).

The relationships of the Type-D performance indicators developed as part of this research, enable various organizational, maintenance, and operational influences, that are manifested through key events that can be identified and reported at a plant, to be accounted for in terms of their impacts on safety. The relationships and the capability they pose are significant new and unique developments.

The relationships require that plant-specific data on the key performance indicators be acquired and analyzed. This, in turn, necessitates that a regular and systematic supplementary data collection program be implemented. Hence, it is recommended here that such a data program be developed and undertaken, specifically within the context of an overall PSA-based safety monitoring system. Plant licensees should be responsible for the supplemental data collection effort; however, the data collection requirements should not pose an undue burden on the licensees. To the extent possible, the data collection program should be coordinated, and possibly integrated, with existing licensee data collection and event reporting efforts. For instance, some of the indicators (Type A to C) used as a basis in the performance monitoring system will already likely be recorded in licensee event reports. Some licensees may also already be collecting data similar to the key Type-D indicators developed as part of this study.

The present study has required input from various knowledgeable experts for the purpose of developing an initial preliminary database from which to proceed. This approach has been necessary because scant empirical data exists and the few available theoretical approaches either have limited applicability to safety performance monitoring, or they are too onerous to implement for routine monitoring. Although the existing expert-based database does have some potential limitations with respect to robustness, it should be expected that the results of the overall performance monitoring system will improve significantly with time, as a history of relevant reliable data collection is realized. After a number of years of data collection, the expert assessments performed for this study can be replaced or refined by actual empirical statistics and the procedures for use of the data can be correspondingly improved.

ACKNOWLEDGMENTS

This work was performed under the auspices of the Swedish Nuclear Power Inspectorate (SKI). The authors wish to acknowledge the contribution of individuals who participated in the expert elicitation process, and provided substantial input to the implementation of the present approach. In addition, the authors acknowledge A.S. Kuritzky for performing the risk calculations in support of this project.

REFERENCES

1. “1993 Performance Indicators,” World Association of Nuclear Operators, London, UK (1994).

2. P. W. Baranowsky, S. E. Mays, and T. R. Wolf, “Development of Risk-Based Performance Indicators,” Proceedings of PSA 99, page 414 (1999).

3. “Indicators to Monitor NPP Operational Safety Performance,” Working Material, IAEA-J4-2883 (January 1999).

4. R. T. Sewell, M. Khatib-Rahbar, and A. S. Kuritzky, “Guidance on the Implementation of a Risk-Based Safety Performance Monitoring System for Nuclear Power Plants,” ERI/SKI-97-401, Energy Research, Inc. (1997).

5. R. T. Sewell, M. Khatib-Rahbar, and H. Erikson, “Research Project – Implementation of a Risk-Based Performance Monitoring System for Nuclear Power Plants: Phase II – Type-D Indicators,” Swedish Nuclear Power Inspectorate, SKI Report 99:19 (February 1999).

**Papers of Specialist Meeting on
Safety Performance Indicators
October 17-19, 2000, Madrid, SPAIN**

APPLICATION OF WANO-PI SYSTEM AND LIVING PSA AS SAFETY INDICATORS IN OLKILUOTO NPP

Risto Himanen
Teollisuuden Voima Oy
Nuclear Reliability
FIN-27160 OLKILUOTO, Finland
tel: +358-2-83813240, fax: +358-2-83813209, e-mail: risto.himanen@tvo.fi

ABSTRACT

Teollisuuden Voima Oy (TVO) operates two almost identical 840 MW_e BWR units on the Olkiluoto island on the west coast of Finland. The paper presents TVO's practices for WANO Performance Indicator program. Some experience is described concerning the weaknesses revealed by the indicators and actions taken to remove the root causes. The utility has maintained a living PSA since 1992. The paper ends with a short description on the development of core damage frequency.

INTRODUCTION

Teollisuuden Voima Oy (TVO) operates two almost identical BWR units on the Olkiluoto island on the west coast of Finland. The net electrical output of each unit is 840 MW. Consequently, these units have been named Olkiluoto 1 (OL1) and Olkiluoto 2 (OL2). The units have been designed and delivered by the Swedish company ABB Atom.

OL1 and OL2 produce energy at cost to the shareholders mainly consisting of the Finnish pulp and paper industry. The load factor of both units has increased steadily during the first half of the 1980's, reaching 90 percent. The average load factor of the past ten years (1988-1997) for both units has been about 93 percent. The unplanned energy unavailability has been during the last years below 0.5%. Despite efficient modernization of the units during 1994-1997 the indicator value for Unplanned Capacity Loss Factor was below 1.5%. The refueling outages, which are very short in the international comparison, are carried out in the spring when there is a lot of hydropower available in Finland.

WANO PI

Olkiluoto has begun to report six indicators to the WANO PI system in 1990, and all ten indicators one year later, in 1991. For each indicator there was named a responsible person from the organization, which is able to affect on the factors that the indicator measures. He submits the input data for the indicator, and performs analysis on the development of the indicator, when required. Inside the company the follow-up of indicators started at the same time, but the first reactions were rather mild, because the indicators showed rather good and stable values during several years. Comparisons were made to the median values of suitable reference groups, i.e. to ALL nuclear power plants in the world, or to BWR units. In addition the values reported by KSU in Sweden from plants delivered by ABB Atom, and the value of Industrial Safety Accident Rate from PWR reactors in Finland were used as close a references.

Olkiluoto has an own follow-up system based on four classes in comparison to the reference group (Figure 1). The green color means that Olkiluoto belongs to the best quartile in the reference group, and the red color the worst quartile, correspondingly. In addition a Weighted Generic Indicator (Figure 2) was developed. It is the sum of rankings of all 12 indicator values. The ranking gets value 2, 1, -1 or -2 if Olkiluoto belongs to the best, second, third or worst quartile in the reference group, correspondingly.

Figure 1 shows that several indicators of Olkiluoto started to place in worse quartiles in 1995. The management of the company was informed once a year about the status of the indicators, but they required a broad analysis on the worst indicators as late as 1998, when the comparative generic indicator in Figure 2 two years had showed that Olkiluoto is in average below the median value. Some progress can be seen during the later years, but the average level of the indicators in the reference group has improved drastically. Thus it seems to be rather difficult to reach the earlier good level.

TVO IN WANO PI QUARTILES **TVO IN WANO PI QUARTILES**

	Indicator	1994	1995	1996	1997	1998	1999	VII/99-VI/00
1	Unit Capability Factor	All(395)	All(407)	All(421)	All(419)	All(420)	All(422)	All(1999)
2	Unplanned Capability Loss Factor	All(404)	All(409)	All(421)	All(420)	All(420)	All(422)	All(1999)
3	Unplanned Automatic Scrams/7000 hours critical	BWR(85)	BWR(87)	BWR(91)	BWR(86)	BWR(89)	BWR(90)	BWR(1999)
4A	Safety: Auxiliary Feed Water	BWR(80)	BWR(78)	BWR(87)	BWR(86)	BWR(88)	BWR(90)	BWR(1999)
4B	Safety: Residual Heat Removal	BWR(82)	BWR(79)	BWR(88)	BWR(90)	BWR(91)	BWR(90)	BWR(1999)
4C	Safety: Emergency AC Power	All(161)	All(174)	All(179)	All(188)	All(185)	All(376)	All(1999)
5	Thermal Performance	All(296)	All(304)	All(300)	All(332)	All(357)	All(349)	All(1999)
6	Fuel Reliability	BWR(84)	BWR(86)	BWR(90)	BWR(80)	BWR(83)	BWR(88)	BWR(1999)
7	Chemistry Index	BWR(79)	BWR(81)	BWR(85)	BWR(82)	BWR(83)	BWR(85)	BWR(1999)
8	Collective Radiation Exposure	BWR(88)	BWR(88)	BWR(92)	BWR(90)	BWR(91)	BWR(90)	BWR(1999)
9	Volume of Low -Level Solid Radioactive Waste	BWR(79)	BWR(80)	BWR(77)	BWR(73)	BWR(90)	BWR(88)	BWR(1999)
10	Industrial Safety Accident Rate	All(191)	All(191)	All(197)	All(201)	All(202)	All(423)	All(1999)
	Weightened sum of rankings	2	1	3	-5	-5	-6	-4
		Final	Final	Final	Final	Final	Final	Prognosis based on VII/99-VI/00

Percentage of units/sites better than TVO:	**0-25%**	**25-50%**	**50-75%**	**75-100%**	
TVO in quartile:	**In best quartile**	**Above median**	**Below median**	**In worst quartile**	All = All types of NPPs BWR = Boiling Water Reactors only (Number of units or sites reported)
Weight:	**2**	**1**	**-1**	**-2**	
Weightened sum:	**13...24**	**1...12**	**-12...0**	**-24...-13**	

Figure 1 Annual ranking of Olkiluoto with respect of each indicator during 1994-1999. Colors are same as in NRC's Reactor Oversight Process , but the thresholds are relative to the ranking in the reference group.

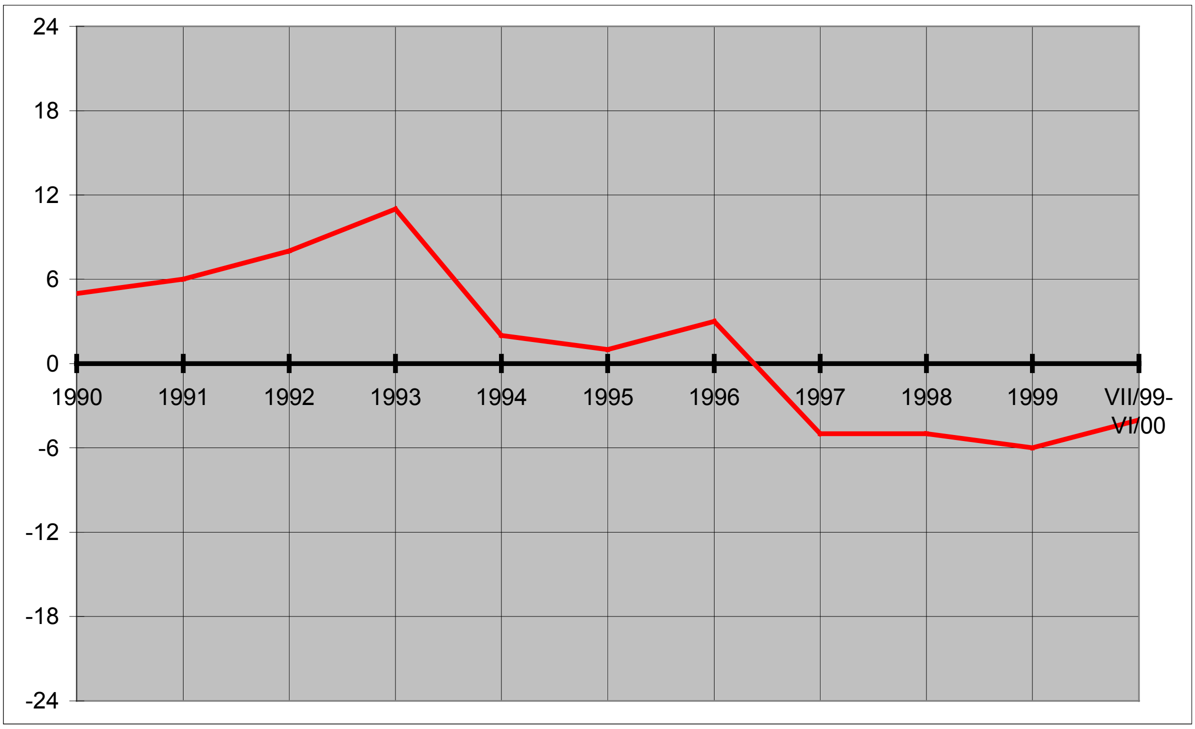


Figure 2 The comparative generic indicator is the weighted sum of rankings in Fig. 1

The following chapters describe first the good availability performance indicators, and give some examples on the analysis results and actions taken due to the broad analysis of the bad indicator values.

Good Availability Performance Indicators

The Unit Capability Factor (Fig. 3) of Olkiluoto has been in the best quartile compared with all units in the world, though the plant has been modernized during the years 1994-98, and the median value of the indicator has increased eight per cent in ten years.

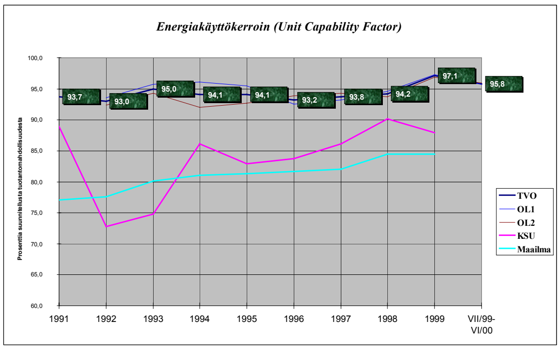


Figure 3 Unit capability factor

Similarly the Unplanned Capability Loss Factor (Fig. 4) has mainly been kept in the best quartile.

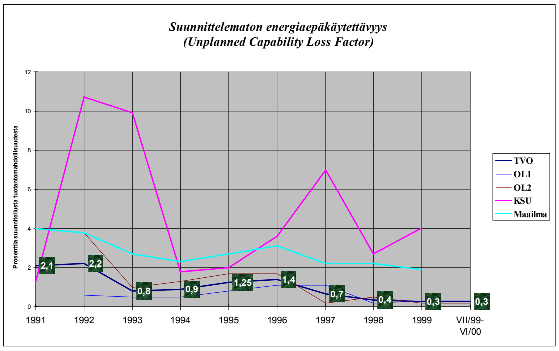


Figure 4 Unplanned Capability Loss Factor

Unplanned Automatic Scrams per 7000 Hours Critical

The number of automatic scrams has been a long time above the BWR median value. There has been made numerous studies in order to decrease the number of "erroneous" scrams.

During the modernization of the plant the instrumentation of the turbine plant was improved. Some single channel measurements were replaces with multiple measurements with 2/3 voting in the protection signal.

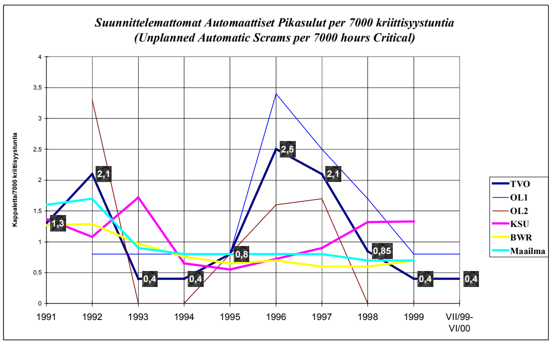


Figure 5 Plant modernization during 1995-98 slightly increased the number of scrams

However, immediately after the modernization the number of scrams increased due to new teething troubles. Now, two years after the modernization, the number of scrams seems to settle on a stable value. It has not been better than the mean value in the world, but not better than the median value, that has been zero during the last years. The scram reports go through thorough analysis continuously in order to prevent similar root causes to scrams.

Unavailability of Safety System Components

Reporting the unavailability time of the safety system components showed that maintenance personnel had difficulties to understand the safety importance of safety system component failure modes, especially those remaining latent until next test. Failure exposure time was difficult to define, and it was not reported. Although the maintenance personnel can affect on the condition of the components, the reporting responsibility was moved from the maintenance personnel to the plant operation personnel. A question was raised: Do the other plants report the failure exposure time honestly?

Before 1997 the indicator was reported on the basis of some selected components only. Same or corresponding components were selected in all ABB plants. This reporting method did not give reliable results, because important auxiliary components, like pressure dampers of piston pumps do not belong to the pump, but they have separate component codes, and were not reported.

The indicator is correctly reported on train basis since 1977, and it shows clearly higher values than earlier. However, if all critical components are selected to the component basis reporting, it should produce same result as the train based reporting.

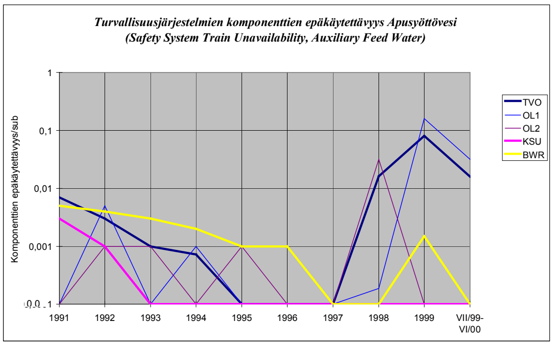


Figure 6 AFW component/train unavailability in logarithmic scale

The WANO indicators have expedited modifications in the safety systems. The problem does not lie in high failure rate, but in long failure exposure time in periodically tested stand-by system trains. Thus the early detection of failures should be improved with better condition monitoring methods and equipment. Design projects are underway to improve early detection of failures.

The maintenance department has started maintenance classification of all components on the plant. The classification is based on several factors, but the safety importance calculated from two importance measures (Fussel-Vesely and Risk Achievement Rate) is among them. Because the safety systems have four identical 50% trains, special emphasis is put on the prevention of common cause failures.

Another quantitative classification principle is the impact of component failure modes to the production of electricity. A large availability performance program (VARMA) has started having as a main goal to prioritize the tasks of different organizations, based on the importance of various components. This goal includes balancing of preventive maintenance among several other duties.

Fuel Reliability

At the beginning of 90'thies he value of the Fuel Reliability indicator was continuously high indicating on fuel leakages. The main reason were PCI (Pellet Cladding Interaction) failures in the fuel, but development of fuel more tolerable to PCI and more strict operational requirements have removed the PCI problem in 1994. Another problem seem to be the debris (foreign bodies) in the primary circuit. The new fuel elements have been equipped with debris catchers since 1999, but fuel leakages have been indicated again during the last two years.

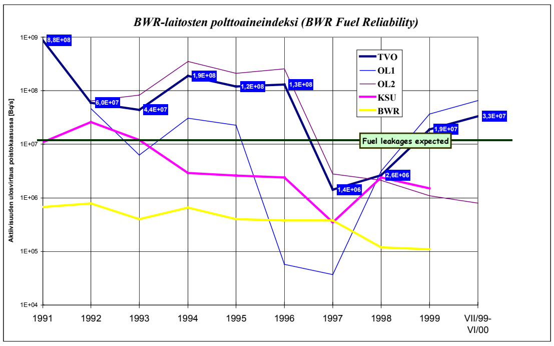


Figure 7 Fuel Reliability

The problem in the WANO Fuel Reliability indicator lie in its sensibility to small leakages, and its poor resolution on the size or number of simultaneous leakages. Most leakages in Olkiluoto have been very small, but the indicator shows results, that are clearly worse than median.

Chemistry Index

In the old Chemistry Index sulphate and conductivity were the parameters, which made the value of Olkiluoto not acceptable. The value decreased, but the median value of BWR decreased, too. When WANO introduced the new Chemistry Index in 1997, the value of Olkiluoto was double compared with the BWR median. The value has decreased during the last two years as a result of process modifications. It is meaning also costs for better chemistry.

The reason for sulphate in the reactor water is the degradation of ion-exhange resin catalysed by high water temperature. The temperature of the purification filters is kept below 60 centigrade with a bypass arrangement that decreases the thermal efficiency of the process. The duty period of the ion-exchange resin is only 30 days increasing the production of solid low active waste. Further investments to decrease the temperature of the resin are in the long term investment plan, but they require costly investments, and will be performed simultaneously with modernization of the intermediate superheater system.

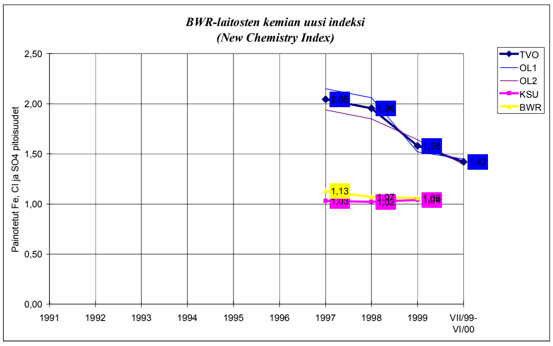


Figure 8 New Chemistry Index

Industrial Safety Accident Rate

One interesting indicator among WANO PI is the that has been high in the Finnish industry in comparison with other countries. Several studies have been performed to clarify the reasons to the difference, but no clear reasons have been found. The high rate can be seen also in the WANO PI of both Olkiluoto and the Finnish PWR units (Fig 9).

Trials to improve the Industrial Safety Accident Rate have been done in Olkiluoto, but the indicator value has reflected only small and temporary changes.

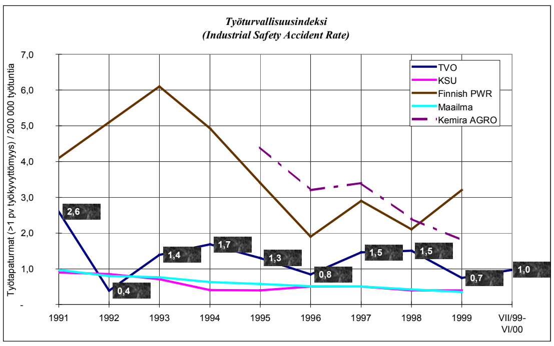


Figure 9 Industrial Safety Accident Rate

A comparison was made with the fertilizer plants of Kemira AGRO, having facilities in several countries in Europe. Figure 9 shows the impact of their industrial safety improvement program. The basic idea of the improvement of the indicator of Kemira AGRO was that the management, including also the highest managers, made themselves more visible among the staff, and continuous safety interviews performed by all managers and foremen.

PROBABILISTIC INDICATORS

Living PSA

Living PSA has been used in safety related decision making inside the utility since early 1990's. Living PSA covers today Power operation and refueling outages including shut-down and start-up periods. Besides the internal initiating events, fires, flooding, seismicity and numerous external weather and environment related phenomena are analysed as initiating events.

Living PSA continuously shows the value aroung $2\cdot10^{-5}$/reactor year despite plant modifications and complementation of the model. Regularly, model extensions have revealed new risks, and plant modifications have made them ineffective.

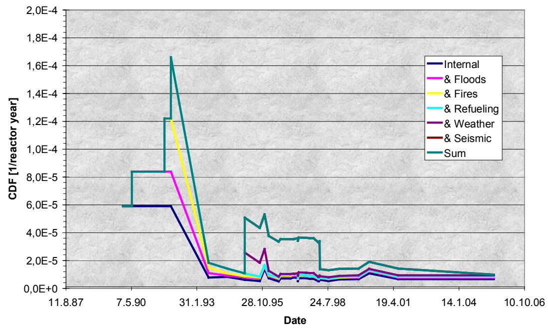


Figure 10 Core damage frequency from different causes of initiating events during 1987-2000, predicted until 2006.

Shut-down PSA

Shut Down Event PSA (SEPSA) complements level 1 PSA as a part of the living PSA model. The study was performed during 1990-1992. The first results showed that the contribution of the refueling outage on the annual core damage risk was of the same order of magnitude as the contribution of the full power operation. Modifications in maintenance procedures were adopted, and the analysis was thoroughly updated and included in the living PSA during 1995-1996. (Fig. 11).

Besides the severe nuclear risks the utility was interested in other risks, too, e.g. significant extension of outages. They were evaluated as a by-product of the event tree sequences, not severe enough to lead to core damage.

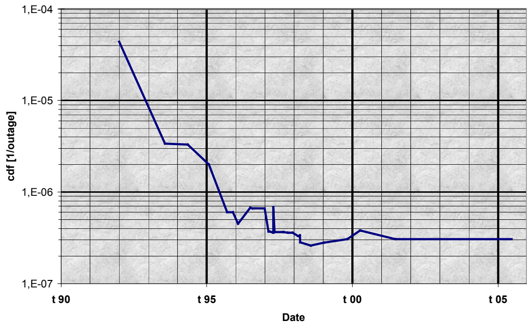


Figure 11 The core damage frequency per refueling outage has decreased drastically

OTHER INDICATOR SYSTEM TRIALS

Several trials have been made in selection of suitable indicators for safety management, but they have been died as unusable. One system was developed in co-operation with the PWR plant Loviisa, and the Technical Research Center of Finland, but the plants are very different, and identical indicators are not suitable for both. Another system was developed by the Technical Research Center of Finland, without other utilities, but it was too complex to update.

The only indicator system that has survived one decade is the WANO-PI system. Two main problems in the own indicator systems can be detected: 1) TVO has only two generating units, and there is no reference group having similar indicators. Follow up of own development of the indicators is not tempting enough. 2) all own indicator systems have been too complex. Too many parameters having only very little of information have been included in the system. Thus the updating of the system has been very laborious, and the management is not interesting in the results.

BONUS SYSTEM

The utility pays to the employees bonus based on exceeding goals on several areas, like electricity generation, environmental factors according to ISO 14001 goals, operation costs, and various TVO factors. Individual WANO indicators have been selected as a part of the bonus system, too, but their weight has been so low and the goals too difficult to reach or manageable by very small group of personnel only that the impact has been small.

The last attempt during 2000 is to improve the comparative generic indicator, because a bulk of personnel can affect to it through at least one of the indicators. Until today the generic indicator seems to stop decreasing, but the final results are available earliest in Mach 2001.

Specialist Meeting on
Safety Performance Indicators
17-19 October 2000, Madrid, SPAIN

PERFORMANCE INDICATORS: RELATIONSHIP TO SAFETY AND NUCLEAR POWER REGULATORY AND INSPECTION PROGRAMS

U. Schmocker
Swiss Federal Nuclear Safety Inspectorate – HSK
CH-5232 Villigen-HSK

C. Humbel
Swiss Federal Nuclear Safety Inspectorate – HSK
CH-5232 Villigen-HSK

G. Prohaska
Swiss Federal Nuclear Safety Inspectorate – HSK
CH-5232 Villigen-HSK

H. Wilhelm
Swiss Federal Nuclear Safety Commission – KSA
CH-5232 Villigen-HSK

ABSTRACT

Performance indicators can be categorized in several ways. In most cases determination of the indicator value is based on the direct observation of the licensee performance and the indicator values are “count-based” with an indirect implication to risk and safety. This complicates the interpretation of trends, and direct use of performance assessment results for regulatory decision-making. The HSK experience in the use of WANO-based indicators, and the need for development of a system that is more directly tied to safety and risk of severe accidents, is presented, including an assessment of the characteristic of the risk-based performance monitoring system that could be more effectively used for regulatory oversight and inspection activities.

INTRODUCTION

Over the last several years, performance indicators have been proposed by the nuclear power industry (e.g., World Association of Nuclear Operators [WANO]), and the nuclear regulatory authorities. These indicators have been intended, mainly, to provide the trends in the availability performance and safety, and for use as tools for management (by nuclear power plant operating organizations) to monitor performance and progress, to set goals for plant improvements, and to gain additional perspective on plant performance relative to that of their peers. In addition, performance indicators are also being used by some of the nuclear regulatory authorities (e.g., U. S. Nuclear Regulatory Commission, the Finnish Authority, etc.) to evaluate the performance of various plants in the areas of safety system reliability, plant and personnel safety. Recently, work has also been initiated under the auspices of the International Atomic Energy Agency (IAEA) [1] and by others [2,3], to develop new approaches to nuclear power plant performance monitoring.

The current approach to performance monitoring is based on a general expectation that there is an implicit relationship, between safety/risk and the selected indicators. However, experience to date, based on tracking and analysis of performance trends, shows that the use of the “count-based” indicators, in their current form, are of limited utility for assessing the safe operation of nuclear power plants. In addition, the present approach to performance monitoring cannot be used to identify early signs of deterioration in safety performance at nuclear power plants due to organizational and safety culture issues, before deteriorating conditions warrant regulatory actions.

OBSERVATIONS BASED ON THE CURRENT APPROACH IN SWITZERLAND

The Swiss nuclear power plants have been using the WANO indicators, supplemented by other indicators that are utilized by the Swiss Federal Nuclear Safety Inspectorate (HSK), for more than 10 years, even though the limitations of the WANO indicators to track safety performance of nuclear power plants are well known.

Figures 1 through 6 show the ten-year trend of a selected number of performance indicators (PIs) for various Swiss nuclear power plants. It is seen that a clear trend in the safety performance of individual Swiss nuclear power plants does not appear to emerge from the analysis of these results, as the safety relevance of the various tracked indicators is not unambiguously provided by the shown PIs, and as such, the tracked indicators do not show meaningful trends of plant safety. For instance, the number of reportable classified events (See Figure 1) does not show a safety significant trend, other than a general indication of the occurrences of only a few events per year. The definition of the so-called B events (events of minor relevance to safety) preclude events of risk significance and these events are of limited safety relevance (e.g., the indicator value does not provide any impact of the backfitted redundant shut-down and decay heat removal systems, in Mühleberg and Beznau nuclear power plants, in the early 1990s, which have been shown by plant-specific PSAs to reduce the core damage frequency for the respective plant, by more than an order of magnitude).

Another example is shown in Figure 2. For the Mühleberg nuclear power plant, where the indicator applicable to the diesel generators appears to indicate a higher degree of safety performance as compared to that as measured by the availability of the high-pressure safety injection (HPI) and heat removal systems (HRS). In reality, the situation is actually very different since the risk importance of diesel generators is typically, much greater than the HPI and the HRS. Experience shows that the diesel generators are often problem-prone (i.e., high rate of failure to start, and to run), and these failures have been shown to be risk-significant. While the “count-based” WANO-based indicators do not reflect this information.

Figure 3 shows that tracking the annual number of scram as an indicator, by itself, is not particularly insightful, as the number of unscheduled shutdowns have reduced over the years to a low level; therefore, making it difficult to associate any trends to safety and plant performance. Nevertheless, the number of unscheduled reactor trips, as a triggering initiating event for plant transients is still of interest.

Several of the existing WANO indicators are intended to provide an indication of the organizational culture and the effectiveness of quality control, as well as the means for tracking the various layers of engineered protection and defense-in-depth. These include the extent of fuel damage (Figure 4), the individual (Figure 5) and collective (Figure 6) radiation exposure.

Fuel integrity during normal plant operation is an excellent indicator of the attention to quality control, by fuel vendors and utilities, with respect to design, manufacturing and operation. The observed trend towards higher burn-up, optimized with respect to thermal-hydraulic use of the fuel, especially for BWRs leading to a much reduced fuel cladding thickness and fuel rod diameter, may result in a reduction of the historically excellent fuel performance, observed over the last few years. For instance, Figure 4 shows the trends of the relevant PI for the Mühleberg (KKM) and Leibstadt (KKL) nuclear power plants. Mühleberg uses a more conservative operational philosophy, thereby, it has only recently decided to introduce the 10x10 fuel elements, whereas, Leibstadt has been more aggressive in moving towards the adaptation of the 10x10 fuel design (i.e., it has already been operating with 10x10 fuel for several years). Experience with the latest operating cycle at Leibstadt has shown that for 10x10 fuel elements, even very small debris can become problematic, potentially resulting in fretting failure of the fuel rods. This example shows, that an optimization based on thermal-hydraulic performance alone, in this case, the fuel element, can result in recurrence of potential performance problems, which were believed solved many years ago.

Figures 5 and 6 show the trends in the individual and collective radiation exposure, respectively, for all the Swiss nuclear power plants. The improvements in the trends of these indicators are indicative of the effectiveness of improvements in the implementation of radiation protection measures by various power plants (e.g., temporary placement of about 50 to 80 tones of lead shielding during plant outages, effective worker training in radiation protection measures, etc.). These indicators are useful as an additional gauge of the attention by utilities, to quality and effective organizational control, thereby, the overall organizational safety culture.

The current PI system has not been found to provide the expected insights that could compliment the normal inspection activities, since, the relationship between the current PIs to actual safety, as measured by risk of severe accidents, is only implied (and in fact, sometimes this relationship can be easily obscured by the current PIs). On the other hand, risk-based inspections of power plants have been of greater benefit to regulatory activities.

Therefore, it is desirable to look into an alternative PI system that would provide a consistent and effective means of assessing the safety performance of Swiss nuclear power plants. Preferably, the new system should consider changes in the selection, reporting, and utilization of PIs, that are:

(1) More directly tied to plant safety performance as measured by quantitative risk measures, including the impact of human and organizational factor issues that could signal early degradation in safety performance before they actual occur.

(2) Less susceptible to ambiguities that could result from the present “count-based” system.

(3) A more direct input to risk-informed regulatory decision-making process, and are complementary to other regulatory implementation processes such as the regulatory inspection programs.

(4) Closely tied to utility and regulatory action plans, when pre-defined thresholds are reached and/or exceeded.

OUTLOOK

A new approach to selection, reporting and analysis of performance of Swiss nuclear power plants, should help:

(a) Focus regulatory inspection activities;
(b) Reduce regulatory burdens;
(c) Enhance all the interactions between HSK and utilities;
(d) Enhance communication to general public and the government.

Any changes to the current system should be aimed at enhancing the HSK’s regulatory effectiveness, while reducing the overall burdens on the utilities, especially, in light of a an emerging competitive and deregulated electric utility market. The necessarily elements of a risk-based PI system should be such that they:

- Focus attention on the most important risk-significant safety related activities during normal operation as well as plant outages;

- Establish objective criteria for evaluating safety performance of utility organizations;

- Provide a feed back mechanism for evaluation of direct and indirect influences of regulatory actions on maintaining and improving safety of nuclear power plants;

- Identify corporate/utility cultural problems affecting safety;

- Improve the authority's effectiveness; and

- Improve communication with utilities, general public and governments.

This type of an indicator system could serve as a tool, for the regulatory body, to direct inspections and safety review activities. Properly chosen and defined indicators can also provide an objective way for the regulators to assess nuclear safety and to evaluate regulatory priorities. Trends in safety performance or safety culture indicators can make possible an early detection in the potential degradation in safety performance.

A possible downside of using performance indicators that are more focused on gauging safety is that, the utilities may unduly concentrate their attention on just performance indicator values at the expense of real safety. To avoid this, indicators should never become the sole basis to assess safety performance. The future PI system:

- Should be sufficiently broad to cover all aspects of safety performance;

- Should have a direct relationship to safety preferable through quantitative risk measures, including the impact of human and organizational factors;

- Should not require extensive new data and should be easily and unambiguously measured and/or calculated;

- Should not be susceptible to numerical manipulation;

- Needs to be accompanied by adequate guidance for data collection, reporting, analysis and validation; and

- Should be tied to “thresholds” for regulatory and corrective actions.

In additions, these PIs should be complimentary to the current defense-in-depth basis for reactor design and operation. To date, most of the PIs are focused on the normal full-power operation, but they are not adequate for abnormal/accident conditions and/or low power, refueling and shutdown conditions.

Work is planned at HSK to initiate a program to define PIs that address the aforementioned attributes. It is planned to have the essential elements of a new performance monitoring system, in the near future.

Another project, which is currently under development and is expected to be complimentary to the overall performance monitory program, is the HSK's "Standard Inspection Program". This program plans a more systematic inspection program for all Swiss NPPs. Other elements this program include:

- Plans for a systematic inspection of all safety and safety-relevant systems (a risk-informed input is urgent to define the relevant systems and the details of the inspection);

- More detailed guidance on the performance of plant-specific regulatory inspection (already in use at HSK);

- Definition of the different type of inspections (e.g., specific issue-oriented inspections, software and human and organizational factor oriented inspections, team and/or individual inspections, etc.); and

- Analysis of trends in results of plant-specific inspections, including evaluations guided by the HSK MOSAIK system (to be discussed later).

Based on the evaluation of the trends in PIs, issue-specific inspections are foreseen. The intention is that at the beginning of each year, a complete inspection program (including both announced and unannounced inspections) is outlined, approved by the HSK Director and forwarded to the utilities for information. It is clear, that additional inspection can be made, if there is a need or if an abnormal event were to occur.

The Mensch, Organisation, Sicherheitskultur, Anlagen-Inspektions-Katalog[1] (MOSAIK), is the third program, which has already been completed, and is currently being implemented. The objective of MOSAIK is to provide HSK with an effective regulatory reporting tool in order to enable early detection of possible signs of deteriorating human safety performance. To optimize the resource management of the HSK Section for Human and Organizational Factors (MOS), MOSAIK is developed to be utilized during plant walk-throughs, meetings and component examinations in connection with engineering or radiation protection matters. Furthermore, this program is intended to reinforce the objective that during an inspection, inspectors should not only focus on technical details, but also concentrate on the adequacy of the details of operational, maintenance, and testing process at the plant (e.g., evaluation of the adequacy of specific testing programs such as planning, use of testing procedures, test setup, documentation, tag-out procedures, etc.). Table 1 lists the questions that are included as part of the MOSAIK process catalog, that address (a) work preparation, (b) work execution, and (c) housekeeping concerns.

The objective is to identify problems with repeated root-causes in a plant, as a way to detect signs of degradation or deficiency in safety culture.

The MOS section at HSK is responsible for evaluations of all of the collected information, in order to identify trends, and to bring to utilities attention, potential areas with trends toward poor performance.

As mentioned, the MOSAIK program has recently been put into practice, and some preliminary insights and results are expected to emerge by early 2001.

CONCLUSION

The difficulties in developing and implementing a risk-based performance monitoring system are not only in the development of a defensible and readily implementable process, but also, its proper use and applications within a risk-informed regulatory framework. Nevertheless, as the move to a streamlined risk-informed regulatory decision-making is essential in light of the greater knowledge that is gained from plant-specific probabilistic safety assessments (PSAs), it is inevitable that future regulatory and inspection activities can be made more effective through a system that focuses on safety-relevant issues. In addition, a focused performance monitoring system can help the communication process between utilities and regulatory authorities, following a more structured and technically sound basis. Furthermore, this would also benefit the

[1] Process catalog for inspections related to man, organization and safety culture issues.

overall risk and safety communication process with the general public, the press, and the governmental bodies.

A performance indicator system is a tool to direct regulatory inspections and safety review activities. Properly chosen and defined indicators can provide an objective way for the regulator to assess nuclear safety and to evaluate its priorities. Trends in safety performance and safety culture indicators can make possible an early detection of degradation in safety culture and organizational management with impact on plant performance.

On the other hand, it is very fundamental that every organization critically gauges its own performance. This is particularly true when dealing with an industry that is strictly regulated, especially, in light of an emerging free and competitive electricity market. Therefore, regulatory authorities need to be able to assess their effectiveness in devising and implementing rules and regulations, and to inspect the safety and operational practices effectively and efficiently in the regulated industry. This requires a method for assessing the performance history and potential future implications of the regulatory organizations to determine the overall regulatory effectiveness.

REFERENCES

1. "Operational safety performance indicators for nuclear power plants", IAEA-TECDOC-1141 (May 2000).

2. R. T. Sewell, M. Khatib-Rahbar, and H. Erikson, "Research Project – Implementation of a Risk-Based Performance Monitoring System for Nuclear Power Plants: Phase II – Type-D Indicators," Swedish Nuclear Power Inspectorate, SKI Report 99:19 (February 1999).

3. E. Lehtinen, "A concept of safety indicator system for nuclear power plants", VTT research notes 1646, 1995.

Table 1 List of questions included as part of the MOSAIK process catalog

PLANT	INSPECTION DATE	INSPECTION SUBJECT	
SECTION/S	INSPECTORS	NPP AT SHUTDOWN	(please cross)
		NPP AT POWER	

WORK PREPARATION

Work procedures clearly stated?

yes			no		
1	2	Safety tagout planned following procedures?	3	4	A
		Protection procedures established?			B
		Written step lists available?			C
		Plan design/sketches available?			D
		Responsibility in step lists clear?			E
		Work supervisions foreseen?			F

Work documents complete?

yes			no		
1	2	Do docs correspond to plant status?	3	4	A
		Is accuracy of procedures recognizable?			B
		Hand entries w/date and sign. of approval?			C
		Work documents followed up?			D
		Parameters identified with proper units?			E
		System/Room identific. numbers clear?			F
		Step lists sufficiently detailed?			G
		Work docs ergonomic?			H

Interfaces known?

yes			no		
1	2	Phases-/Net plans available?	3	4	A
		Maintenance coordinated with other dept.?			B
		Target oriented cooperation in trouble sit.?			C

Is work place properly equipped?

yes			no		
1	2	Materials/components ready?	3	4	A
		Tools/resources ready?			B
		Tools/resources proper for work?			C
		Radiation protection measures adequate?			D

Training adequate?

yes			no		
1	2	Are tasks familiar to workers?	3	4	A
		Work steps exercised in advance?			B
		Techniques tried on model/mockup?			C

Notes

WORK PERFORMANCE

Procedures available at the location?					
yes			no		
1	2	All necessary docs/procedures available?	3	4	A
		Referenced procedures available or known?			B

Compliance with procedures?					
yes			no		
1	2	Work/Instruction orders given from MCR?	3	4	A
		Safety tagout performed following process?			B
		Security tags available at plant?			C
		Security tags available in MCR?			D
		Depressurization of the secured systems tested?			E
		Communication to contr. room maintained?			F

Work controls following rules?					
yes			no		
1	2	Controls performed according to process?	3	4	A
		Do supervisors control workers?			B

Supervisor leads workers?					
yes			no		
1	2	Does supervisor give support in case of problems?	3	4	A
		Is supervisor present in area?			B

Deviations from schedule with reason?					
yes			no		
1	2	Influence of deviat. on other activities clear?	3	4	A
		Conformation to schedule/plans?			B
		Quality maintained despite schedule pressure?			C

Work/shift transitions properly executed?					
yes			no		
1	2	Is shift transition systematic?	3	4	A
		Do appropriate written rules exist?			B
		Do protocols of work transition exist?			C
		State of work kept in writing?			D

WORK PERFORMANCE
Continuation on next page

Notes

WORK PERFORMANCE
Cont.

		Conformance to protection procedures?			
yes			**no**		
1	**2**	Are zone clothes worn correctly?	**3**	**4**	**A**
		Is obligation of helmet observed?			**B**
		Are protective devices carried?			**C**
		Is protection from falling used?			**D**
		Are ground openings closed?			**E**

		Concentration on Work?			
yes			**no**		
1	**2**	Little irrelevant discussion?	**3**	**4**	**A**
		Are work instructions seldom repeated?			**B**
		Does a person work only on one assignment?			**C**

		Work completion?			
yes			**no**		
1	**2**	Are work sheets signed?	**3**	**4**	**A**
		Are experiences evaluated?			**B**
		Are changes in docs provided?			**C**
		Are components/systems tested?			**D**

Notes

HOUSEKEEPING
Cleanliness and Order

Workplace kept clean					
yes			**no**		
1	**2**	Tools/components laid in order?	**3**	**4**	**A**
		Accessories laid in order?			**B**
		Leaks cleaned immediately?			**C**

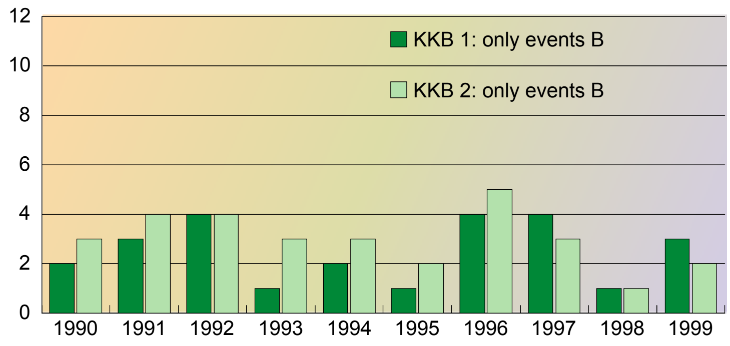
Plant clean and in order?					
yes			**no**		
1	**2**	Are escape routes free?	**3**	**4**	**A**
		Are rules of behavior in accidents clear?			**B**
		Are corridors free of trash?			**C**
		Is available equipment storage used?			**D**
		Is the plant free of burnable trash?			**E**
		Is only colored plastic foil used?			**F**
		Objects near Transmitter-Racks etc. secured?			**G**
		Are pressure bottles secured from falling?			**H**
		Is the plant tight? (no trace of leaks)			**J**
		Is lighting good everywhere?			**K**

Notes

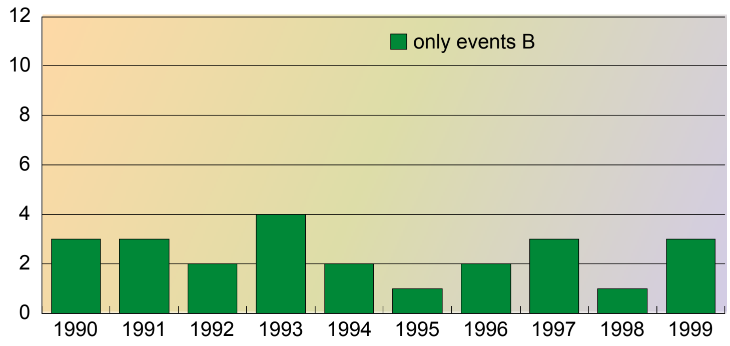
Meaning of the scale:

1: Expectations fully met
2: Within expectations
3: Improvements possible. Comments/directions by HSK
4: Improvements necessary. Weakness

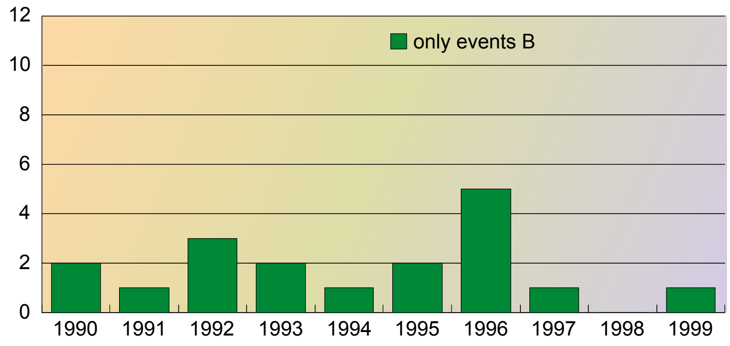
KKB 1, 2



KKM



KKG



KKL

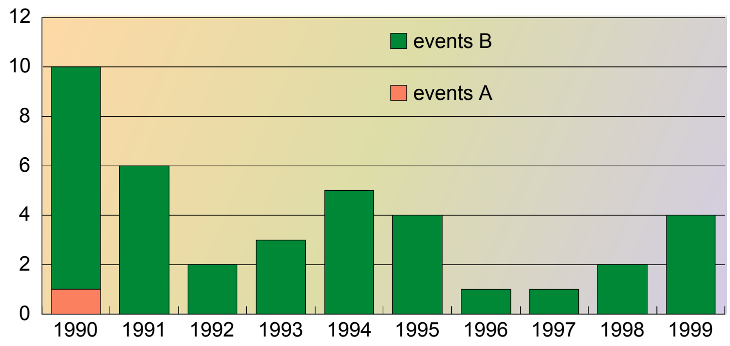


Figure 1 Notifiable, classified events for Swiss nuclear power plants

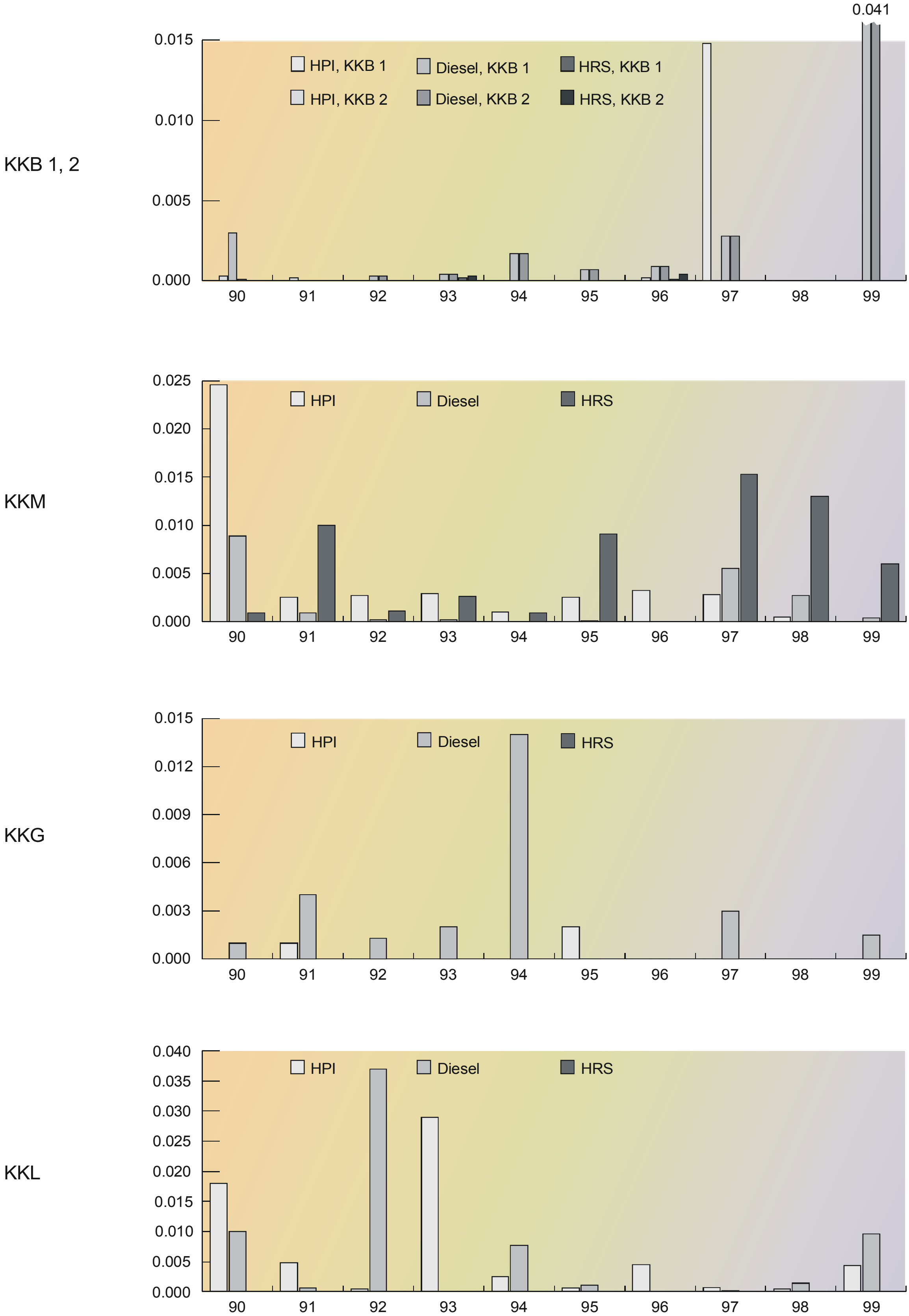
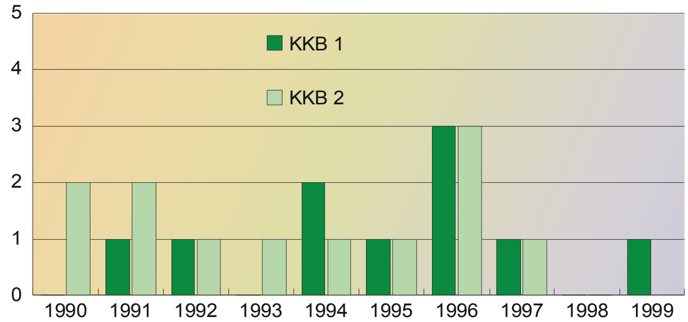
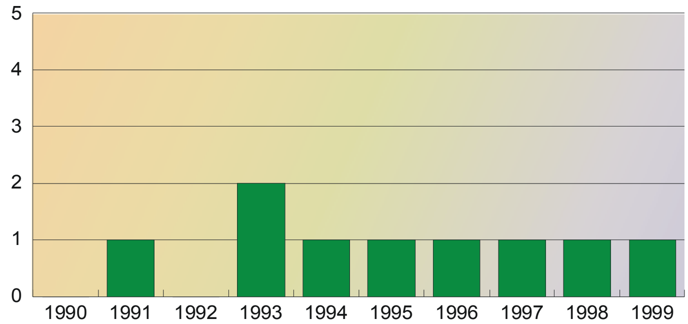


Figure 2 Trends in safety system unavailability for Swiss nuclear power plants

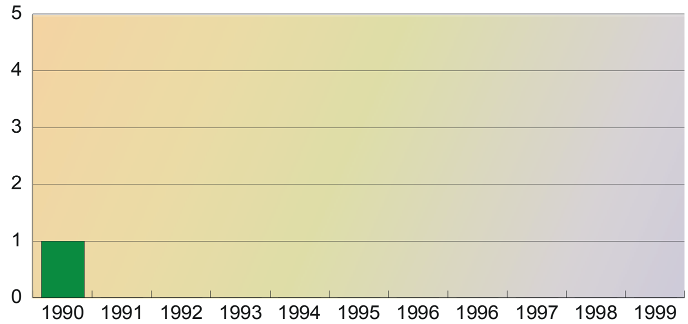
KKB 1, 2



KKM



KKG



KKL

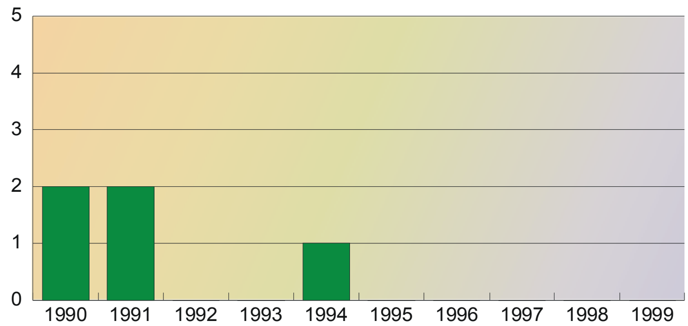


Figure 3 Annual number of reactor scram for Swiss nuclear power plants

KKB 1
KKB 2: no damages

KKM
no damages

KKG

KKL

[a] e.g. hair cracks

[b] e.g. larger damage or breaks in the cladding with fuel washout

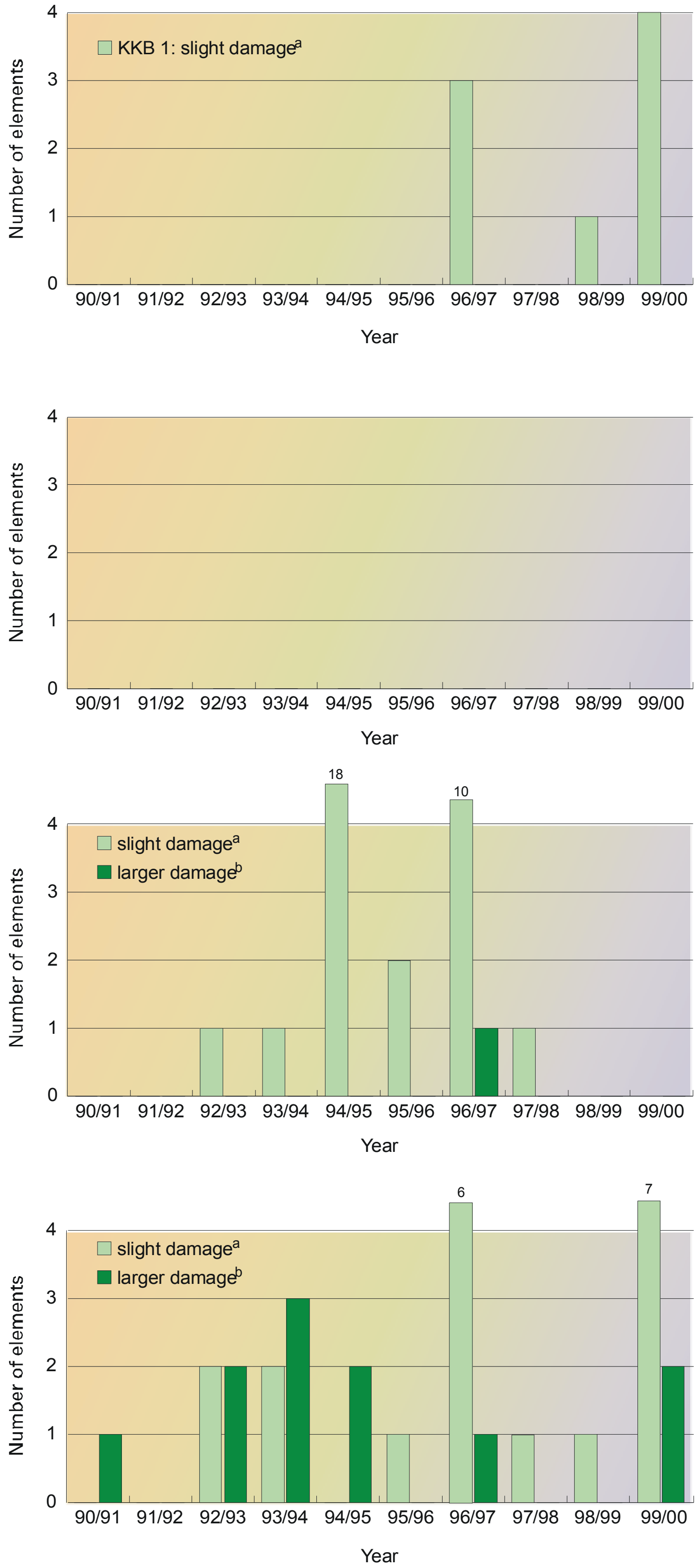
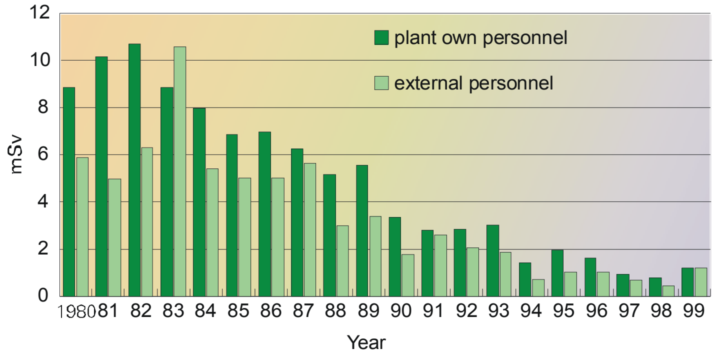
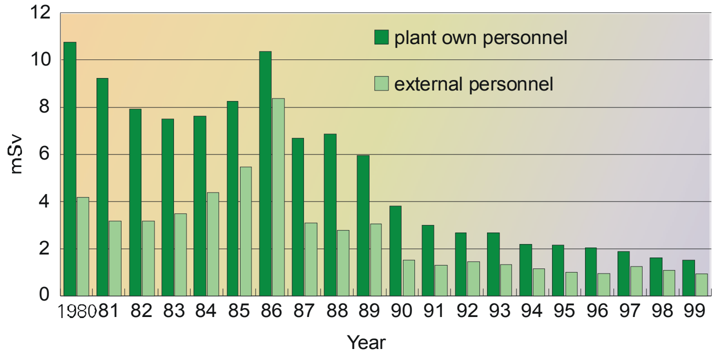


Figure 4 Fuel damage history for Swiss nuclear power plants

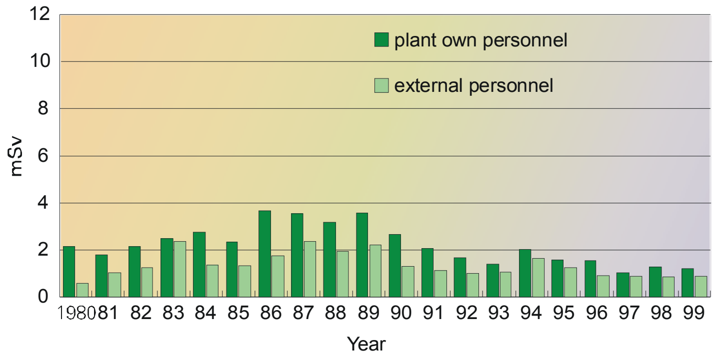
KKB 1, 2



KKM



KKG



KKL

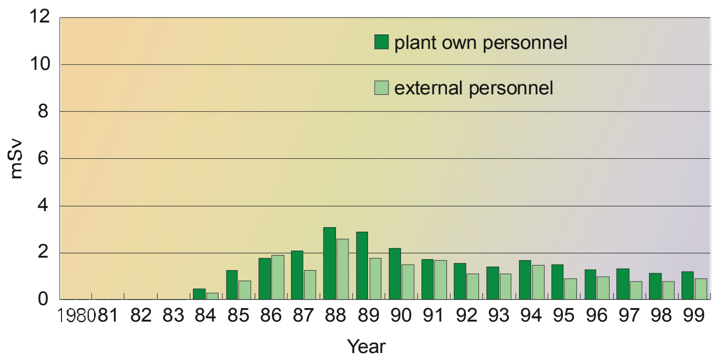


Figure 5 Individual exposure (dose to plant and other personnel) for Swiss nuclear power plants

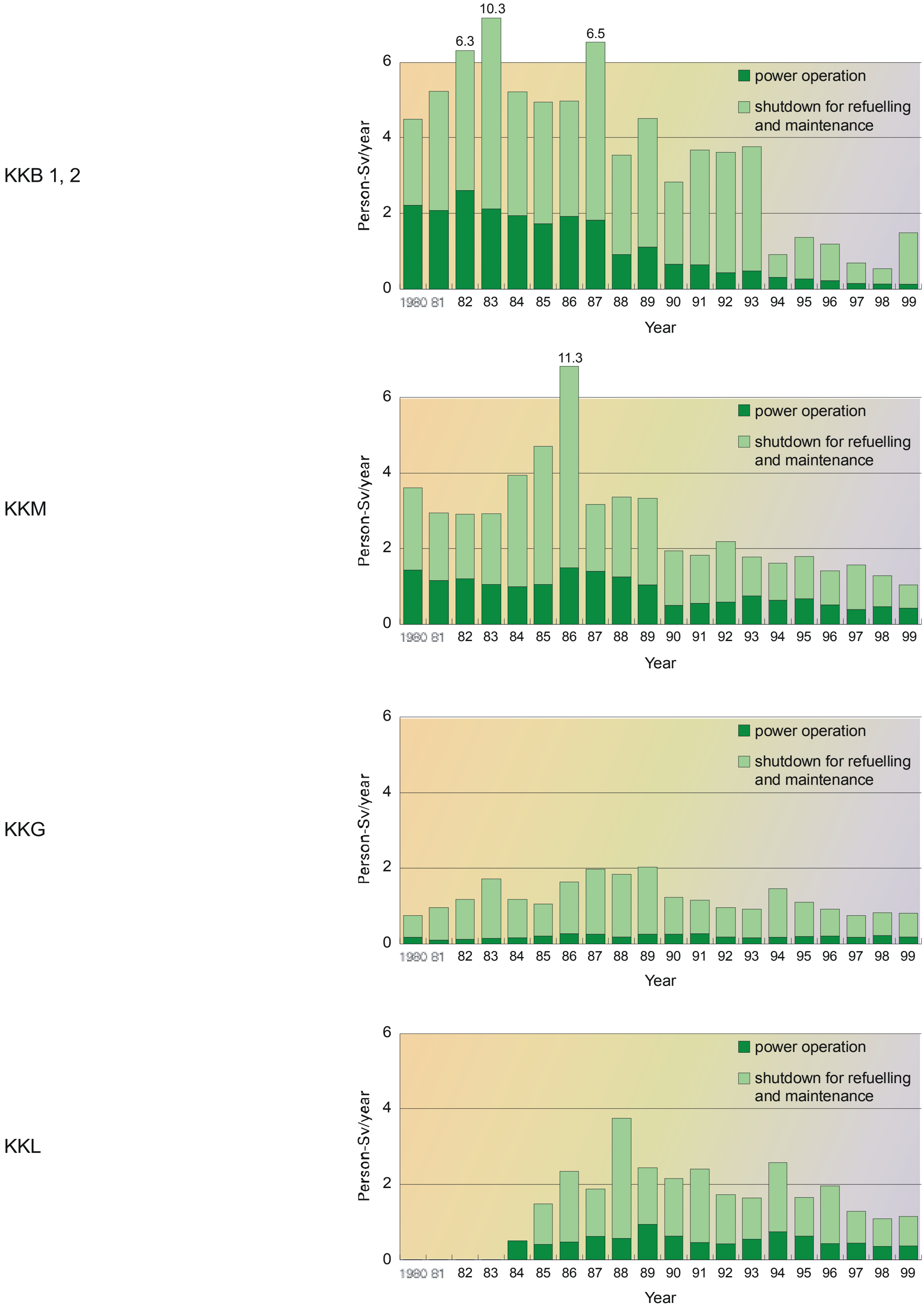


Figure 6 Annual collective doses (during power operation and refueling/maintenance shutdown) for Swiss nuclear power plants

Paper
Specialist Meeting on
Safety Performance Indicators
October 17-19, 2000 Madrid (SPAIN)

RISK INDICATORS AT COFRENTES NPP

Joaquín Suárez
IBERDROLA Ingeniería y Consultoría
Nuclear Generation Department
Av. Burgos 8b, 28035 Madrid (Spain)
+34 913833180
joaquin.suarez@iberdrola.es

Ignacio Hermana
IBERDROLA Ingeniería y Consultoría
Nuclear Generation Department
Av. Burgos 8b, 28035 Madrid (Spain)
+34 913833180
ignacio.hermana@iberdrola.es

ABSTRACT

Following the tend to try to find indicators to show the excellence in the performance where Nuclear Power Plants are currently involved, Cofrentes NPP are managing several indicators related with risk.

The concept of risk is classically associated with the product

*RISK = PROBABILITY * DAMAGE*

So what a risk based indicator will show is the probability of having a "damage". Speaking about a period of time, we will have frequencies of having "damages".

What is call "damage" can be differently interpreted depending of what we concern. In western NPP is very extended the concept of "core damage", meaning the loss of fuel integrity, as a final state to avoid. This have carried in most of western NPP to develope a Probabilistic Risk Assessment (PRA/PSA), that using technical based in fault trees and event trees models, looks for the frequency to reach core damage. The PSA in Cofrentes NPP has been deeply applied to find weakness in the design and procedures, prioritizations in maintenance activities, quality assurance requirements, justifications to continued operation, and others. A Risk Monitor based in PSA models (and so monitoring the Core Damage Frequency) has been developed and is currently installed in the Control Room to help operators to control the risk associated with each configuration of availability or unavailability of equipments. This PSA Monitor is the source for some indicators that Cofrentes NPP has defined and are sharing with IAEA trying to find an standard. Maximum Core Damage Frequency reached and accumulated annual probability is calculated and compared with expected values and with predefined limits.

As the PSA in Cofrentes NPP is only for at power Operations, there has been developed a methodology based on NUMARC 91-06 to measure and control the risk during shutdowns. The "damage" here is a concept related with the safety functions. Some coefficients are applied to each configuration according with how the safety functions are fulfilled (dephense-in-deep). Each outage is scheduled taking in care these indicators, and then a real following of the tasks shows the evolution of these index along the outage. It is notorious how this method has improved the safety during the outages in the last years.

Finally, another risk that is being measured in Cofrentes NPP is related with the loss of production, translated into money. Several elements are joined into this risk analysis: generation and prize, availability, regulation and environment. Each one of this elements is quantified through applicable factors, for example to quantify prize is taken into account cost of fuel, residues, salaries, operation and maintenance costs, prize of the generated kWh. The risk of having a core damage is also translated into money and taken into account in this analysis. Each factor is measured according with historical data when available or industry experience.

As a result Cofrentes NPP is creating a global methodology to learn the most important contributors in each aspect, looking for a way to improve the general performance of the plant.

RISK INDICATORS AT COFRENTES NPP

Cofrentes NPP manages several risk indicators, both at power and during shutdowns. These indicators began to be an initiative of the plant previous to any regulatory or industry requirement, but nowadays they are used to fulfill some regulator´s requirements, especially from 10CFR50.65, Maintenance Rule, which is a risk informed, performance based regulation.

In the countries where it applies, Maintenance Rule forces to fix a lot of risk based indicators to show the performance of structures, systems and components against maintenance effectiveness.

As a kind of final integration of all the previous indicators, the Core Damage Frequency is monitored inside a Risk Monitor available in the Control Room and in the Schedulers office with real data of unavailability of systems, trains and components.

For shutdowns, Cofrentes NPP has developed a program based on NUMARC 91-06 that monitorize the performance of each system inside each one of the safety functions, according also with the Technical Specifications requirements. All these indicators are also integrated in an index called "shutdown severity index".

Currently is under development a program to monitorize the performance of the important systems not for safety but for electricity production. It is expected that this project will improve the production and so the initiating events frequencies; this way the safety will also be improved.

The Maintenance Rule

Maintenance Rules applies in Spain since April 1999. Its requirements can be resumed in:

- The effectiveness of Maintenance activities must be monitorized to assure that the Structures, Systems or Components (SSCs) would perform its safety functions.
- Before performing maintenance activities (including but not limited to surveillance, post-maintenance tests, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities.
- All the process must be documented each operating cycle.
- It must be balance the improvement in reliability by performing maintenance with the objective of minimizing unavailability.
- The scope of the program must include all the safety related SSC, and those not safety related used in the Emergency Operation Procedures (EOP) or in the accident analyses of the Final Safety Analysis Report (FSAR) or in the Probabilistic Safety Analysis (PSA). Finally, all the SSC that has caused or may cause a scram or a safety related system actuation must be also inside the scope.

The NUMARC 93-01 describes a valid methodology to fulfill with the requirements of 10CFR50.65. This methodology has been revised and complemented in Spain in a Verification and Validation process developed between the CSN and Cofrentes NPP during the years 1996, 97 and 98.

In practice, all safety and non-safety related SSC that performs any relevant function in relation with safety must be inside the scope of the Rule. The performance of these SSC must be assured through the definition of indicators and its comparation with previously defined criteria.

The most common index used to monitorize the performance are the unavailability and the number of functional failures (FF).

The PSA becomes a key tool to define the scope, the risk significance, the criteria and the pursuit of the Rule, because is the only analysis that joins qualitative and quantitative aspects. In the PSA has already performed an analysis of failures and unavailabilities of the different systems of the plant, so it is very useful in order to know a "normal" performance of a system.

Maintenance Rule relation with safety indicators

Due to the concern expressed in 10CFR50.65 about that "goals shall be established commensurate with safety", and, "take into account industry-wide operating experience", the Maintenance Rule criteria must agree with:

- the risk significance of the function of the system,
- the historic performance,
- the strategic of the plant,
- the assumptions of the PSA.

Cofrentes NPP Maint. Rule indicators

A total of 124 systems of Cofrentes NPP are inside the scope of the Maintenance Rule, monitorizing 384 functions. 129 of these functions are considered as Risk Significant, (see NUMARC 93-01) and another 54 are considered non-risk significant. All of these 183 functions are monitored at a system-function level, so there are available indicators that describes the performance of the system in the fulfillment of each function. The most usual of these indicators are the number of functional failures (Fig. 1) and the number of hours unavailable (Fig. 2), but in some special cases they are used other indicators as the leakages from isolation valves.

The others 201 functions are non risk-significant and are normally in operation, and so its performance is controlled through plant level indicators, like unplanned capability loos factor (Fig. 3), number of scram, or number of licensee event reports. For these 201 functions, anyway, it is controlled the individual contribution to the total plant indicator (Fig. 4).

Normally, these indicators are monitorized in a “rolling window”, this means that is controlled the performance in the last, for example, 18 months. Each month attached means that one month’s data, 18 ago, is eliminated. With this method it is possible to control not only the performance against the criteria but also the tend of the indicator, in order to take compensatory measures before the criteria is overcomed (Fig. 3, 4, 7).

Cofrentes NPP Risk Monitor

The Risk Monitor of Cofrentes (Fig. 5) is an application that in function of the unavailability of systems, trains and components gives a measure of the Plant safety level based in calculating the Core Damage Frequency (CDF).

It is based in the At Power, Level 1 PSA. It has the same level of detail, and the same truncation level, than the original PSA.

It is installed in the network of the plant. All the Operation Shift has the possibility of write data, and a few set of personnel (Operation Manager, Operation Technical Staff, Maintenance Technical Staff) has the possibility of read data and use it in a simulation mode.

PSA staff supervise the running of the Risk Monitor and also use it in several PSA applications.

Normally the Risk Monitor automatically loads data from the Electronic Log. This is a database developed by Cofrentes’s Operators where they load all the data related with the day by day of the plant: unavailabilities of systems, tests performed, system alignments, etc. Loading data this way Operators has a current view of the risk of the plant without burden them with more work.

Linking Maintenance Rule (MR) with the Risk Monitor

The first necessity covered by the Risk Monitor is that refereed to know the global impact over the safety of performing a maintenance task (MR [a][4] paragraph).

A new revision of the NUMARC 93-01 dated on February 2000, and that will become effective on 28 November 2000 in the USA, accept risk monitors as a reasonable way to fulfill this requirement.

Cofrentes NPP controls the Core Damage Frequency reached by each maintenance task executed (Fig. 6). This control sometimes has served to avoid risky situations, and a lot of times has served to know the most significant contributor to the risk.

The other main use of the Risk Monitor is to assure the long-time well performance of the systems and components:

The Maintenance Rule criteria must be fixed, in order to be easily managed, related to a short period (1-3 cycles) and so have an answer to the required cyclical report.

The criteria are also defined with the necessary margin over the historical performance of the system, in order to avoid falses alarms because of random failures.

Spanish’s regulator was very concern about a possible degradation of the system that were not detected by this cycle by cycle monitoring. So they asked about the possibility of having some indicators related to a longer period.

There were also a concern about the global effect in the plant safety of having such a margins over the systems performance. May be that if a given system reach, but not overcomes, its criteria, the plant safety will not be affected, but could it be affected if a lot of systems reach theirs criteria limits?.

Statistical data needs bigger amount of data to be significant. Translated inside our problem, this means that we need longer periods to obtain real answers to see potential degradation of safety. This is consistent with the data analysis of PSA, that takes into account big periods of time, making sets of similar components to increase the experience.

But this must be mixed with the fact that it is necessary to assure the performance of the system in the fulfillment of a function. So it is not possible to avoid of the control of a few set of components, not similars at all, but that combines to perform a function.

The solution given by Cofrentes NPP was:

- The annual Core Damage Probability Increase will be monitorized. Here all the effects of failures and unavailabilities will be combined to see the global impact over the plant safety.
- Cycle by cycle, the real data will be introduced inside the PSA, changing the event´s probabilities where necessary, and so showing the impact over the Core Damage Frequency.

For these solution, a key feature is the PSA updating period. A working group mixing CSN and utilities expert has been formed to define how and when PSA must be updated.

On-Line Maintenance

Cofrentes NPP has a program to perform At Power preventive maintenance in systems that its unavailability imply an entry of a Limited Condition of Operation (LCO) of the Technical Specifications.

The objective of this program is to improve the reliability of systems without increasing the risk due to the unavailability increase, improving at the same time the safety during shutdowns.

The first step is a feasibility study performed with the PSA to show which system's unavailabilities has a very low or null impact in the risk (Core Damage Frequency and Core Damage Probability). Only the systems that have a few impact in the Core Damage Frequency will go inside the program. Additionally, several administrative controls are performed (not simultaneous unavailabilities, a change in the operation mode is not expected, etc.).

Anyway this program has given that the at power unavailability of some systems have increased comparing with their historic performance. This has carried that the Maintenance Rule unavailability criteria is greater for these systems (Fig. 7).

In order to avoid that the risk would be significantly increased due to these tasks, a set of controls are defined:

- Each six months, a preliminar On-Line program is defined for the following six months, analizing the Core Damage Probability (CDP) increase planned.
- This program is compared with the current CDP increase due to real maintenance data of the last year (Fig. 8).
- Just before performing a task, the expected increase in CDP is again compared with the accumulated CDP due o real data of the last year.

The experience is that the inclusion of this program has not increased the Core Damage Frequency.

Shutdown Risk Control

Because in Cofrentes NPP it has not been developed a Shutdown PSA, the safety control during shutdowns is performed based in qualitative criteria foolowing the NUMARC 91-06 recommendations.

An specific Plant Procedure to control this has been performed. The outages are scheduled according with the results of this analysis, and during the outage a real-time following of the tasks shows how the reality is adapted to the scheduled.

Each one of the safety functions during shutdowns (Reactivity Control, Residual Heat Removal, Inventory Control, Spent Fuel Pool Cooling, Power Supply, and Containment Control) are monitored through the control of the unavailability of the systems capable of fulfill the specific function.

It is defined a quantitative measure of the safety through taking in care by expert judge:

- the Technical Specification requirements,
- the hours the plant is in a configuration,
- the capacity of a given system to fulfill a function.

All of these are combined inside a indicator called "Severity Index", which is monitorized both during the scheduled and during the day by day of the outage (Fig. 9).

Since the implementation of the procedure, a significant improve of the safety during outages has been achieved (Fig. 10).

Unavailability Risk Management

Currently a new project is under development to improve the capability factor of the Plant. It is called GERDIS, spanish acronym for Unavailability Risk Management.

Using PSA technique of system's modelling and failures probability estimation, it will try to avoid risky situations that could carry to loos of production by causing a trip, a manual shutdown, power reductions, etc.

As a result of this work it is expected a better control of the unavailability risk due to the valuable additional information for the operators in their daily decision-making process.

Conclusions

Cofrentes NPP has developed a big amount of risk indicators due to the implementation of Maintenance Rule.

The plant is monitoring Core Damage Frequency through a Risk Monitor that automatically loads data of unavailabilities of systems and components.

It has been implemented a Safety during Shutdown Procedure that has achieved sitgnificant improvements in the safety during outages.

Is under development a new project that would increase plant capacity and reduce initiating event frequencies, improving this way the safety.

ACKNOWLEDGMENTS

Thanks to the Maintenance staff of Cofrentes NPP, specially to Jerónimo Roldán and Antonio Serna. They has given us all the graphics and in every moment they have deeply collaborate.

The Risk Monitor installed in Cofrentes NPP is EOOS, developed by Data Systems and Solutions (DS&S) sponsored by the Risk and Reliability Workstation Project of EPRI.

REFERENCES

- Equipment Out Of Service (EOOS). User´s Manual. *EPRI / DS&S*.
- 10CFR50.65: Monitoring the Effectiveness of Maintenance at Nuclear Power Plants. *NRC*.
- NUMARC 93-01: Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants. *NEI*.
- NUMARC 91-06: Industry Guidelines to Evaluation of Outages Management. *NEI*.
- PGCENCO-13: Procedimiento General para aplicación de Seguridad en Paradas. *Cofrentes NPP*.
- PGCENCO-14: Procedimiento para la ejecución de Mantenimiento a Potencia. *Cofrentes NPP*.
- CNCOF/COF/ISAM/99-17: Instrucciones complementarias al permiso de explotación de C.N. Cofrentes relativas a la implantación de la Regla de Mantenimiento. *CSN*.

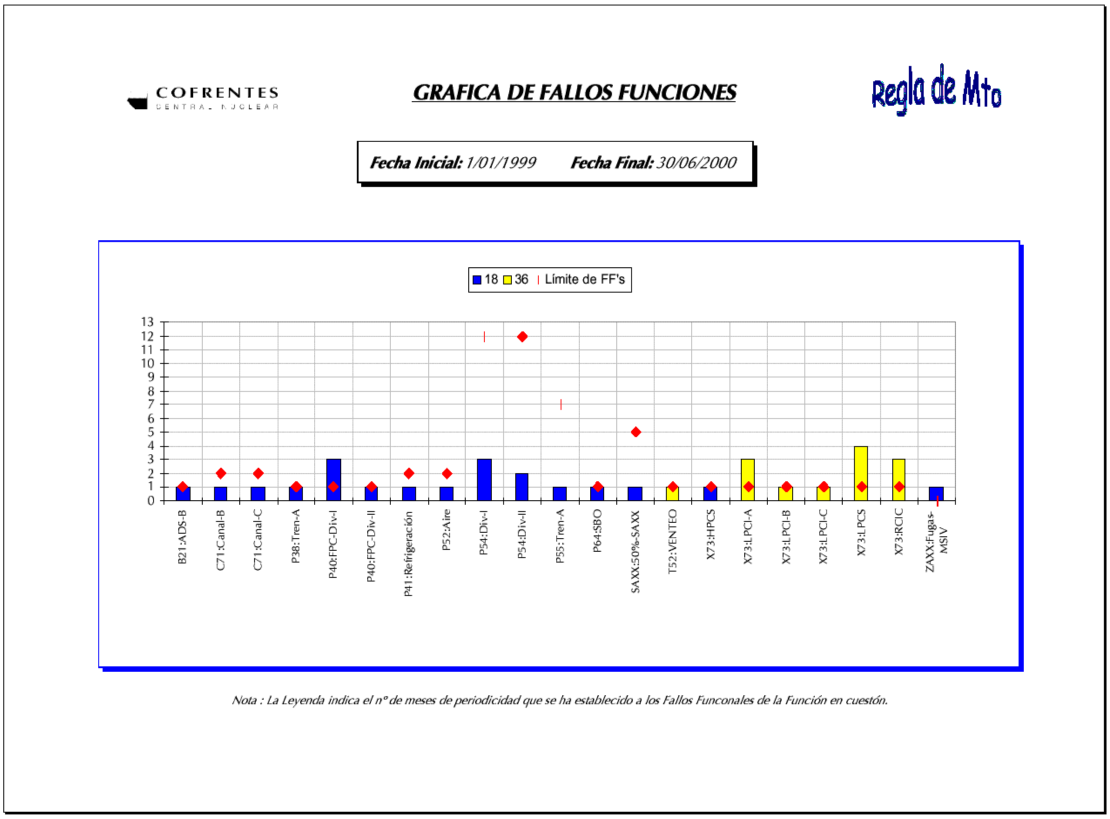


Fig.1: Functional Failures of systems and limits fixed (Maintenance Rule)

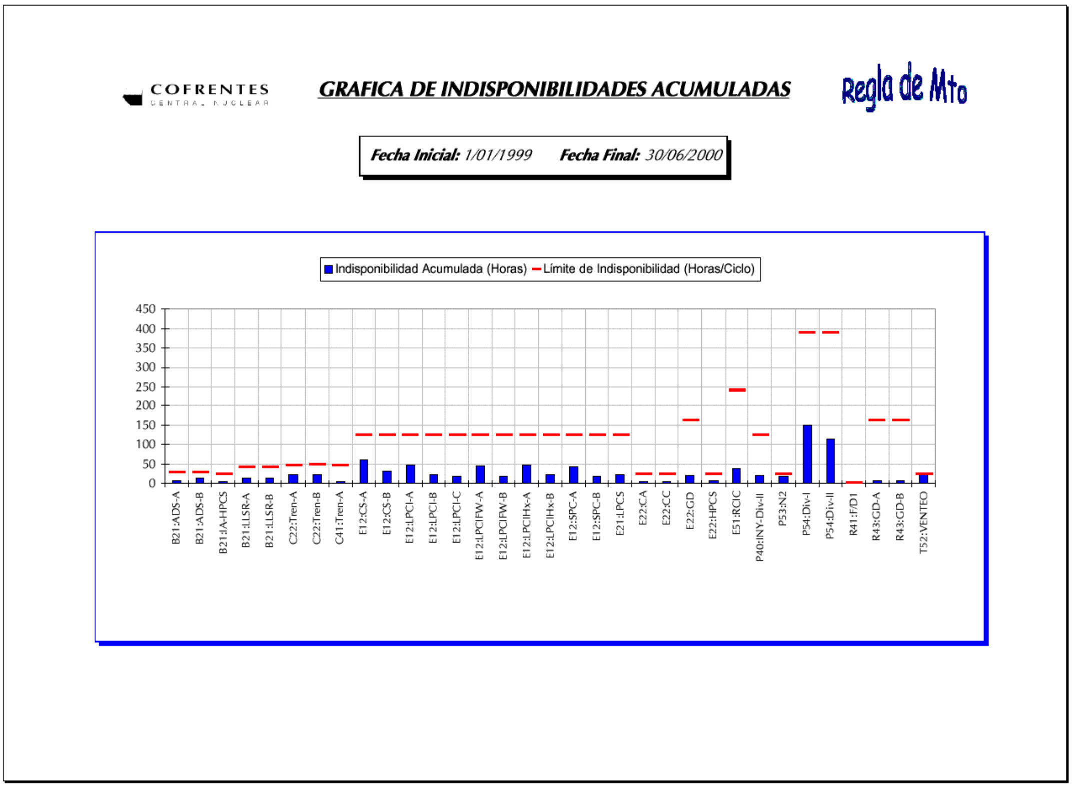


Fig.2: Unavailabilities of systems and limits accepted (Maintenance Rule)

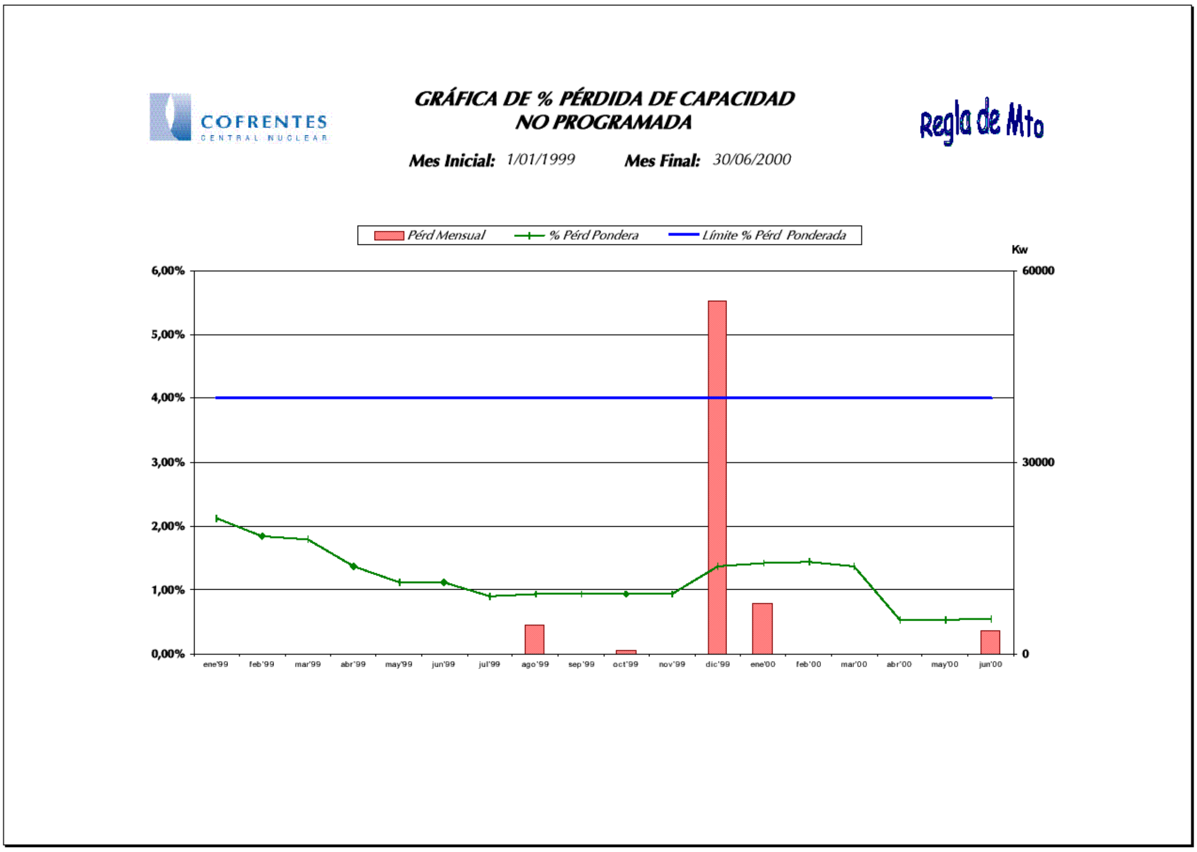


Fig.3: Global Loss of Capability

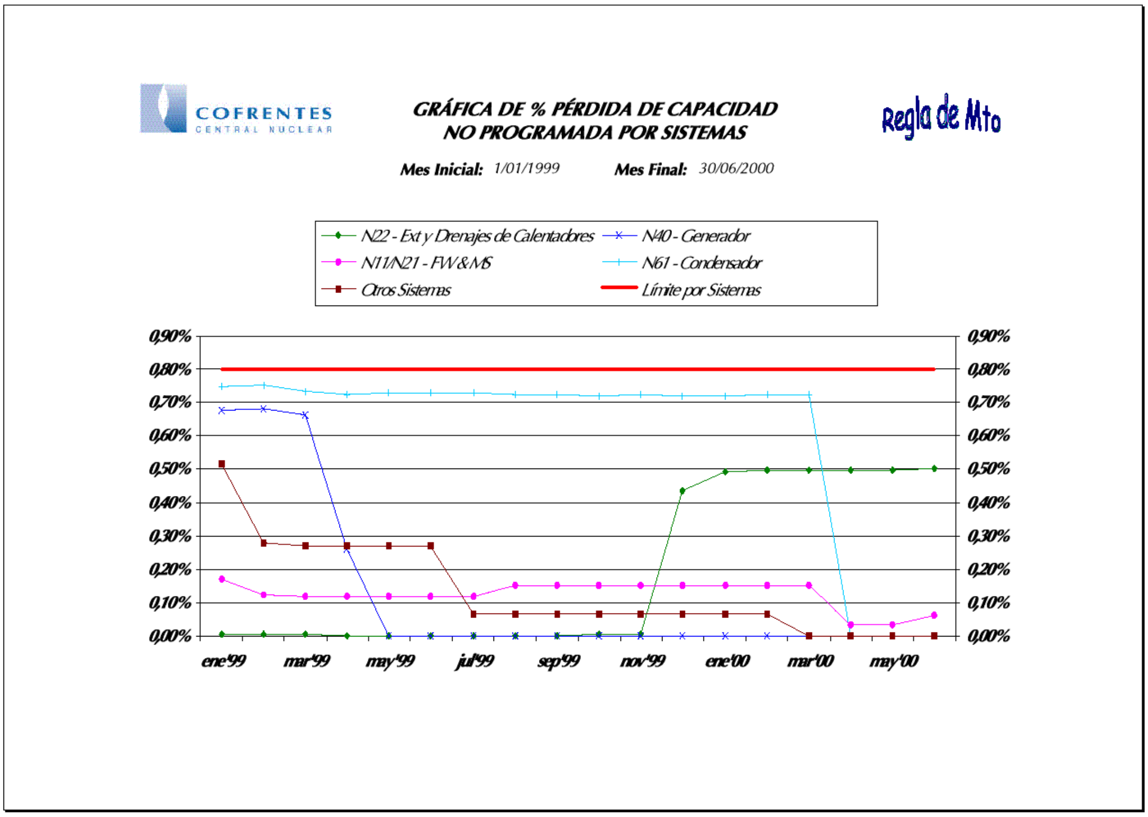


Fig.4: Systems contribution to Global Loss of Capability

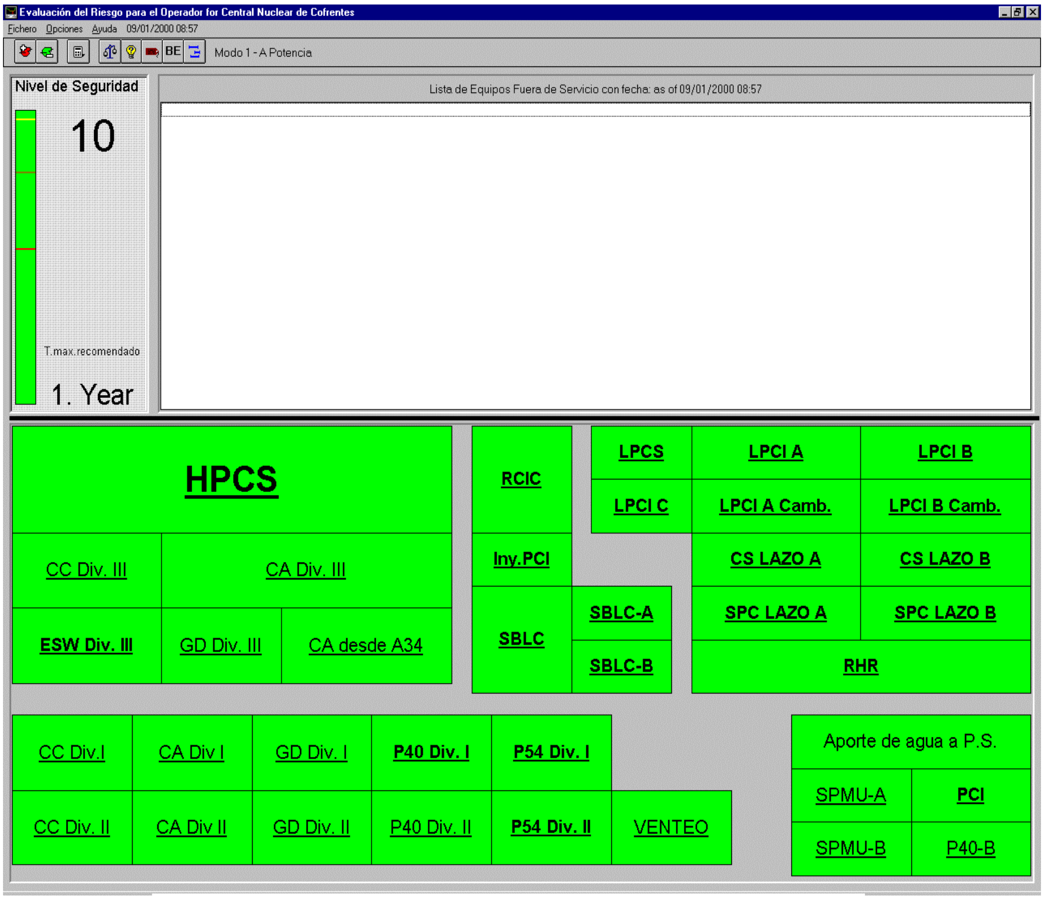


Fig.5: Cofrentes NPP Risk Monitor (Operators screen)

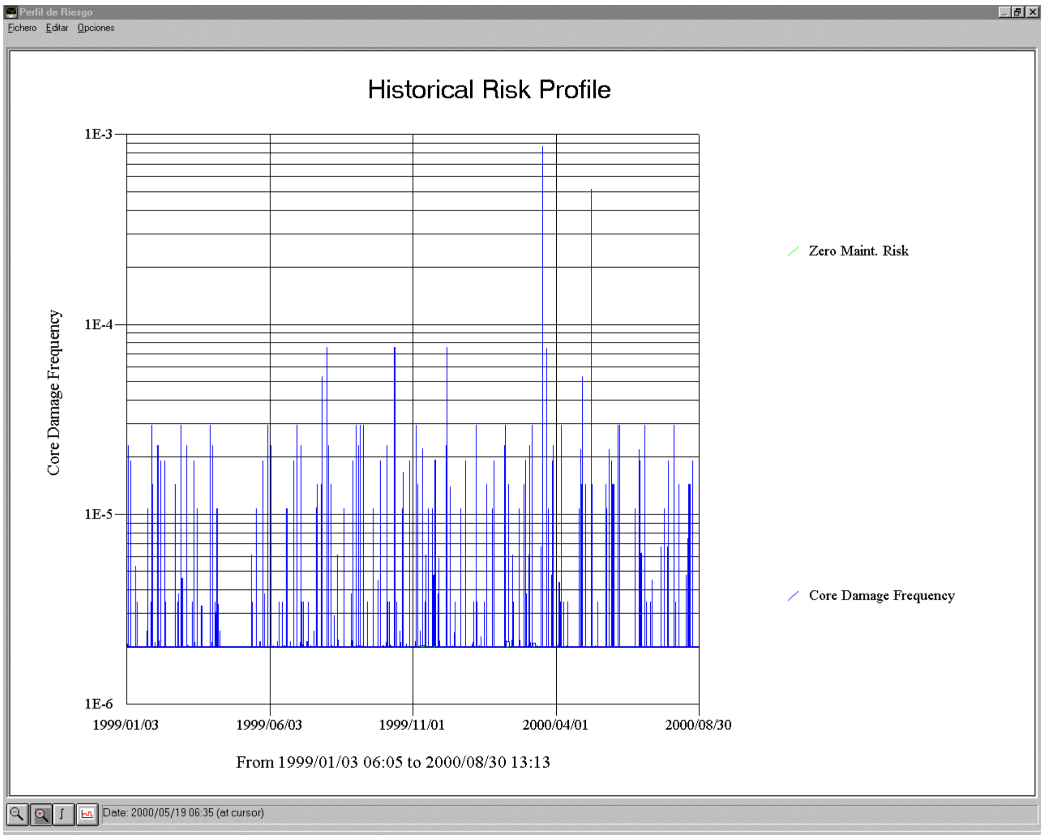


Fig.6: Core Damage Frequency of Cofrentes NPP since January 1999

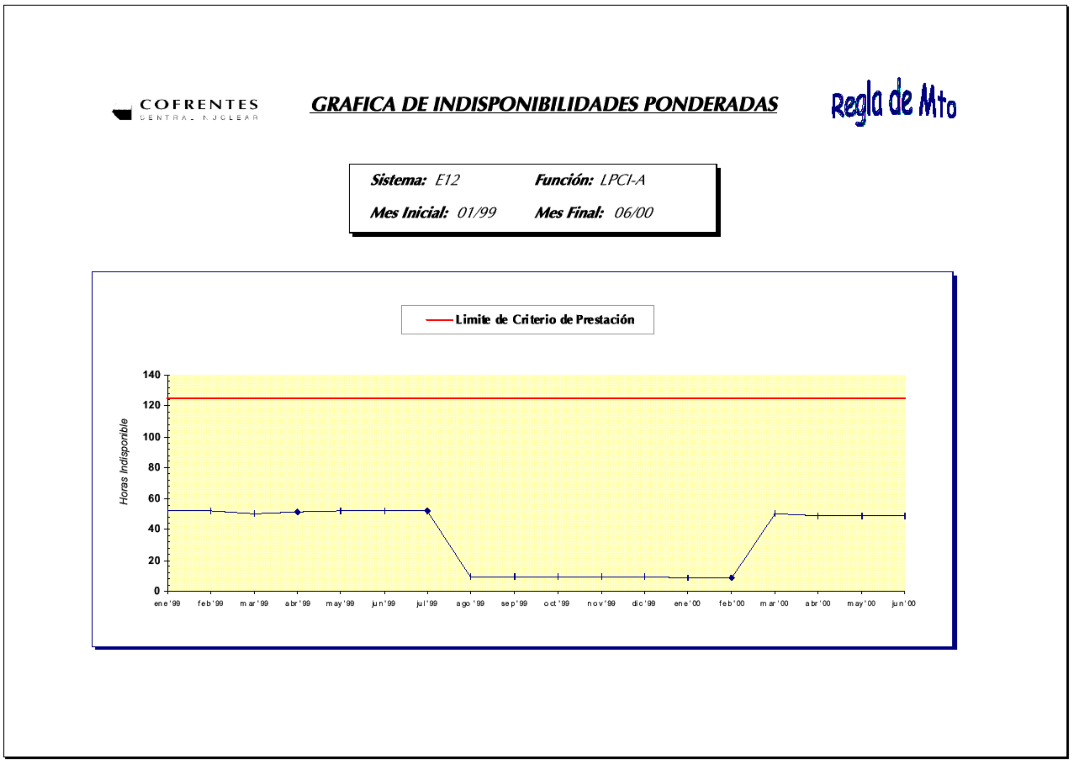


Fig.7: Unavailability rolling window for a system under the On-Line Maintenance

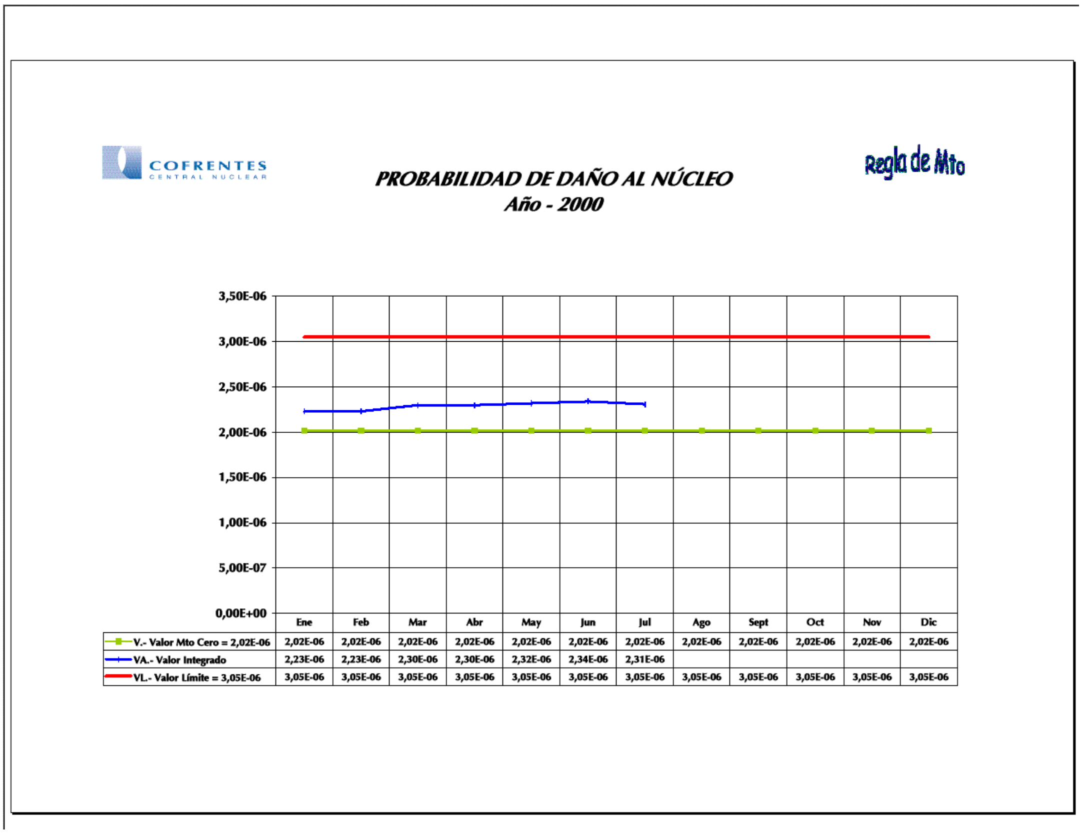


Fig.8: Core Damage Probability yearly rolling window

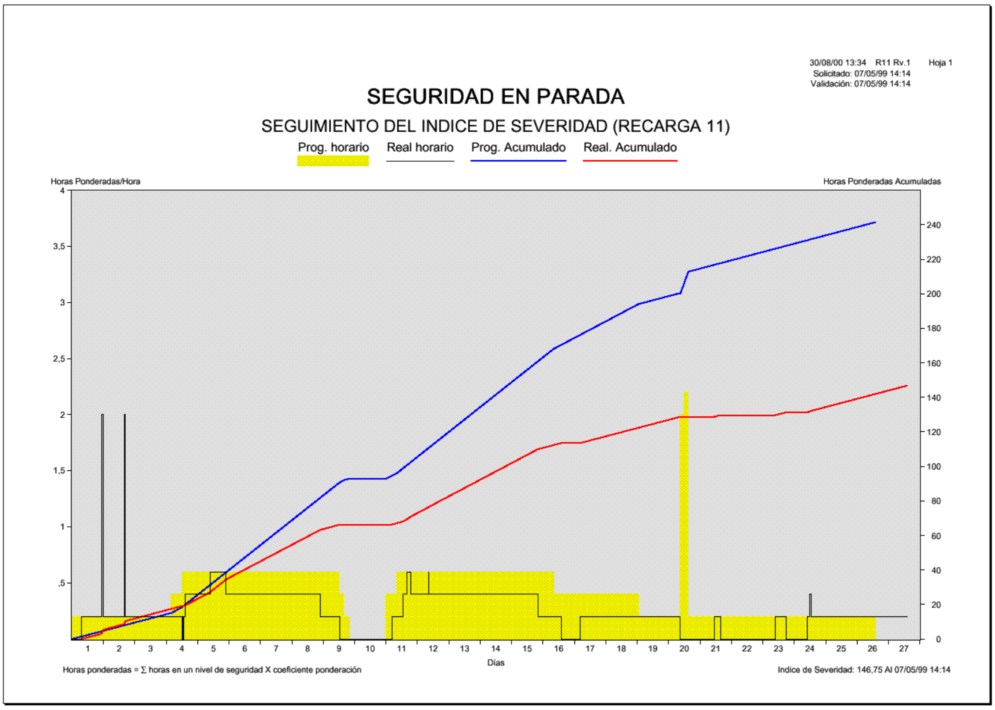


Fig.9: Severity Index during outages: Scheduled and real

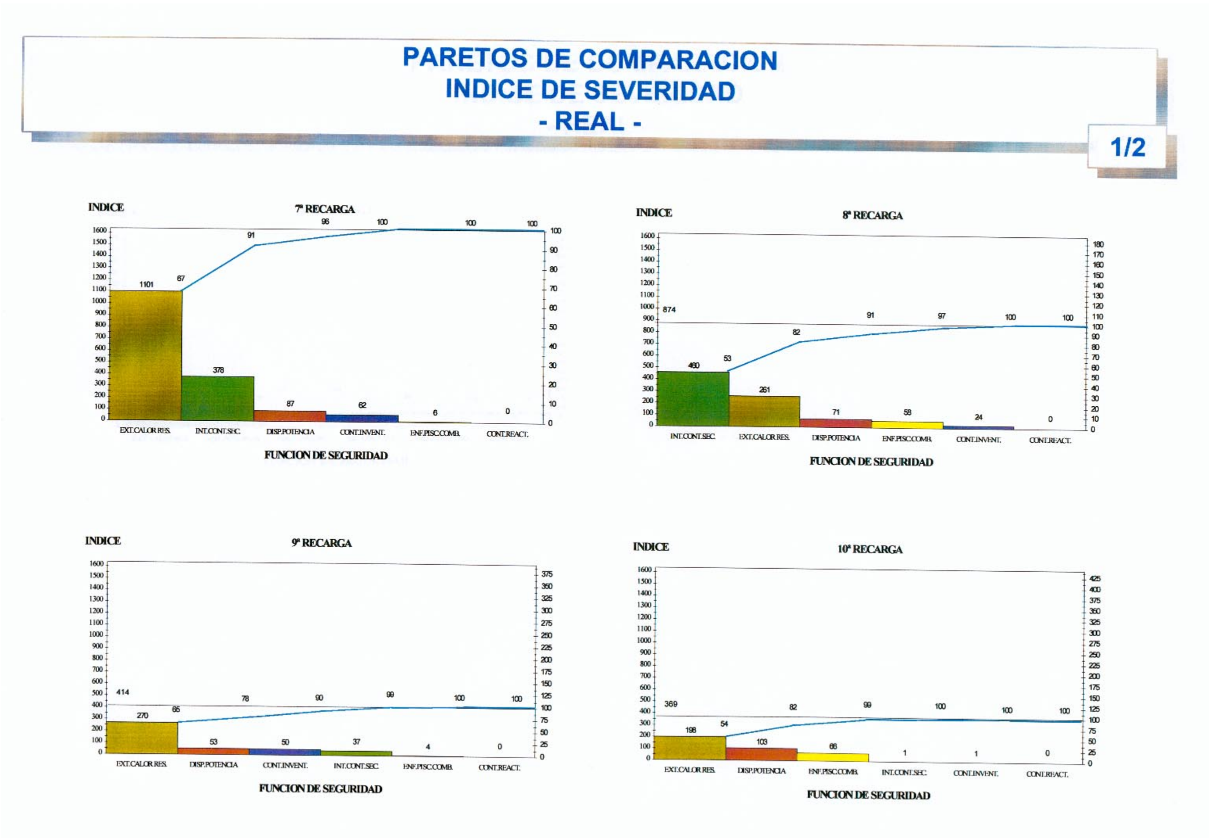


Fig.10: Severity Index during several outages

Specialist Meeting on
Safety Performance Indicators
October 17-19, 2000, Madrid, SPAIN

INDICATORS TO MONITOR NPP OPERATIONAL SAFETY PERFORMANCE

Ana Gómez-Cobo
International Atomic Energy Agency
Department of Nuclear Safety
Wagramerstrasse 5
A-1400, Vienna
Austria
Tel.: +43 1 2600 22516
Fax: +43 1 2600 7 22516
E-mail: a.gomez-cobo@iaea.org

ABSTRACT

Since December 1995 the IAEA activities on safety performance indicators focused on the elaboration of a *framework for the establishment of an operational safety performance indicator programme.* The development of this framework began with the consideration of the concept of NPP operational safety performance and the identification of *operational safety attributes.* For each operational safety attribute, *overall indicators,* envisioned as providing an overall evaluation of relevant aspects of safety performance, were established. Associated with each overall indicator is a level of *strategic indicators* intended to provide a bridge from overall to specific indicators. Finally each strategic indicator was supported by a set of *specific indicators*, which represent quantifiable measures of performance. The programme development was enhanced by pilot plant studies, conducted over a 15 month period from January 1998 to March 1999. The result of all this work is compiled in the IAEA-TECDOC-1141, to be published shortly. This paper presents a summary of this IAEA TECDOC. It describes the operational safety performance indicator framework proposed and discusses the results of and lessons learned from the pilot studies.

1. INTRODUCTION

The safe operation of all nuclear power plants is a common goal for all involved in the nuclear industry. However, as a concept, safety is not easy to define. Even more difficult is the establishment of a clear definition of an adequate level of safety. Nonetheless, there is a general understanding of what attributes a nuclear plant should have in order to operate safely. The challenge lies in measuring the attributes.

This paper presents the work done during the IAEA project on "operational safety performance indicators". This project focused on the development of a framework for identification of performance indicators which have a relationship to the desired safety attributes, and therefore to safe plant operation. The actual indicators are not intended to be direct measures of safety, although safety performance can be inferred from the results achieved

2. FRAMEWORK FOR THE ESTABLISHMENT OF PLANT SPECIFIC OPERATIONAL SAFETY PERFORMANCE INDICATORS

The development of the IAEA framework began with the consideration of the concept of nuclear power plant safety performance. To ensure a reasonably complete set of operational safety indicators, a decision was made to work down a "structure" in which the top level would be *operational safety performance* and the next level would be *operational safety attributes*, from which a set of *operational safety performance indicators* could be developed.

Three key attributes were chosen that are associated with plants that operate safely:

Plants ***operate smoothly***.
Plants ***operate with low risk***.
Plants ***operate with a positive safety attitude***.

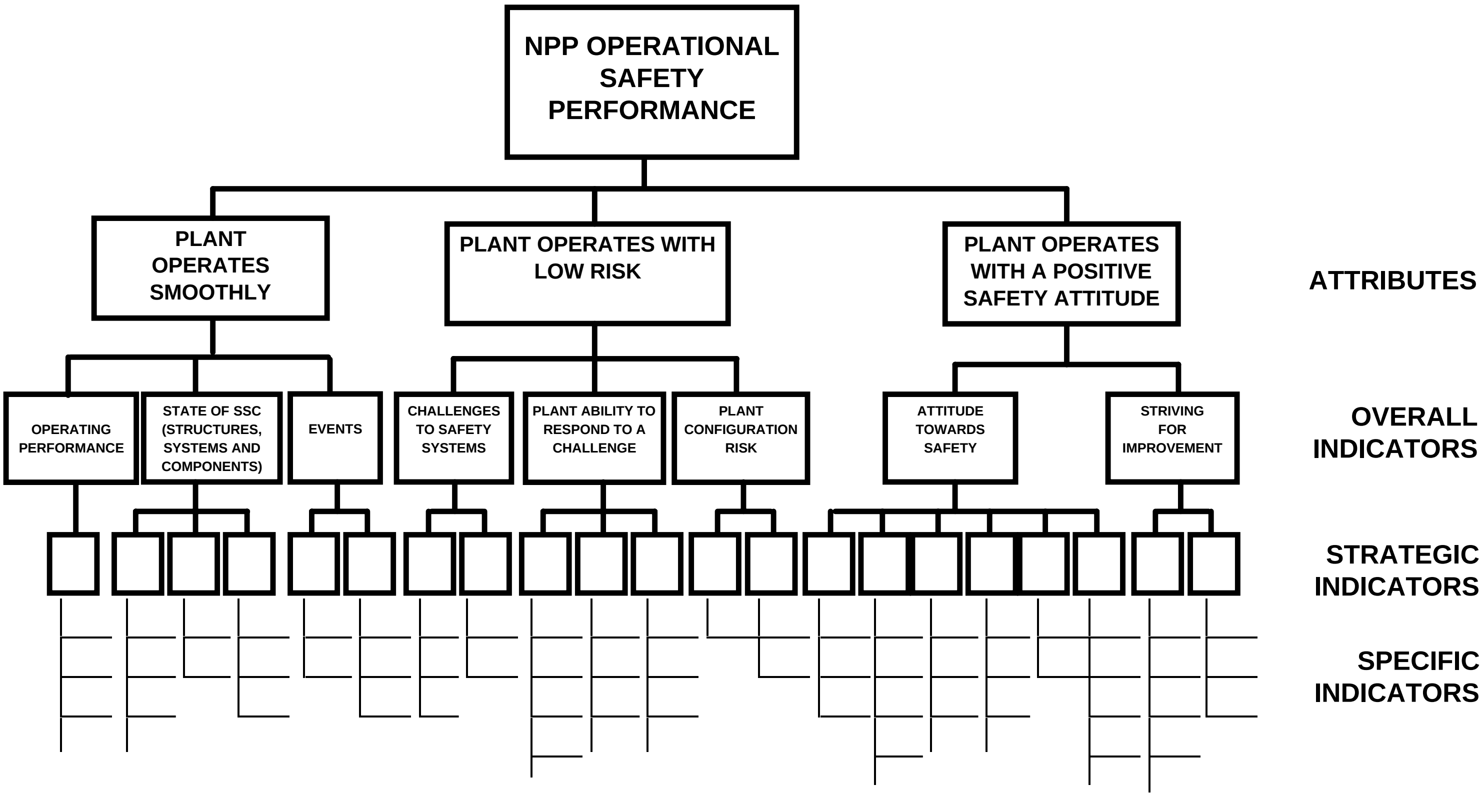


FIGURE 1. Operational safety performance indicator framework.

Using the above attributes as a starting point for indicator development, a set of *operational safety performance indicators* were identified. Below each attribute, *overall indicators* were established. Associated with each overall indicator was a level of *strategic indicators*. Finally, each strategic indicator was supported by a set of *specific indicators*, most of which are already in use in the industry. Figure 1 presents the proposed framework.

The *overall or key indicators* were envisioned as providing an overall evaluation of relevant aspects of safety performance. *Strategic indicators* were intended to provide a bridge from overall to specific indicators. *Specific indicators* represented quantifiable measures of performance. Specific indicators were chosen for their ability to identify declining performance trends or problem areas quickly so that after proper investigation, management could take corrective actions to prevent further performance degradation.

The following sections describe, for each safety attribute, the related overall and strategic indicators, and provide some examples of specific indicators. The IAEA-TECDOC-1141 on "Operational safety performance indicators for nuclear power plants" provides additional examples of specific indicators associated to each proposed strategic indicator.

2.1. PLANT OPERATES SMOOTHLY

The overall indicators chosen to represent the degree of smoothness with which the plant operates are *'operating performance', 'state of SSC (structures, systems and components)',* and *'events*'.

2.1.1. Overall Indicator: Operating Performance

The first means of preventing accidents is to strive for high quality plant operation with infrequent deviations from the normal operational state.

Strategic Indicator: *forced power reductions and outages*

This measure addresses forced power reductions of some predefined percentage or more and forced outages. The minimum power reduction that is reasonable to measure should be considered specifically by each plant.

Example of specific indicator: number of forced power reductions and outages due to internal causes.

Example of specific indicator: unplanned capability loss factor (WANO performance indicator).

2.1.2. Overall Indicator: State of Structures, Systems and Components (SSC)

Measures of the status of the SSC reflect the contribution of the maintenance programmes to the plant safety performance through the reliability of plant components, systems and structures. In addition, a good control of the chemistry in the plant will help to ensure that the life of safety related equipment will be as long as expected by the equipment design.

Strategic Indicator: *corrective work orders issued*

Usually a corrective work order is issued for all troubleshooting, corrective maintenance and minor modifications. A large amount of corrective maintenance may reflect potential reliability problems, but, also, maintenance deficiencies.

Example of specific indicator: number of corrective work orders issued for risk important BOP systems.

Example of specific indicator: Number of pending work orders for more than 3 months.

Strategic Indicator: *material condition*

A good control of the plant chemistry and the ageing will help to ensure equipment life according to the design.

Example of specific indicator: Chemistry Index (WANO performance indicator).

Example of specific indicator: ageing related indicators (condition indicators).

Strategic Indicator: *state of the barriers*

Defence in depth is one of the basic principles of nuclear power plant safety. In order to avoid contamination of the environment and radioactive doses to the public, the source of the risk needs to be isolated by concentrically located barriers: cladding, primary coolant boundary and containment. Therefore, it is very important to establish indicators that help to monitor the state of these barriers.

Example of specific indicator: fuel reliability (WANO).

Example of specific indicator: containment leakage.

2.1.3. Overall Indicator: Events

Every event is an indicator of some plant deficiency. There are different types of events with causes of various nature and different level of safety impact. The safety significance of an event can be minimal (e.g. the failure of a single fuse, leading to no consequence) or significant, as e.g. the failure of an entire safety system.

Strategic Indicator: *reportable events*

The intent of this strategic indicator is to monitor those events that are considered to have higher safety significance, namely those of interest to other organizations, such as the regulatory body or other nuclear operators through WANO, events in IAEA-INES scale of level 1 or higher, etc.

Example of specific indicator: significant reportable events.

Example of specific indicator: licensee event reports.

Strategic Indicator: *significant incidents*

The intent of this strategic indicator is to account for those events that, even though they are not necessarily reportable (externally), are still significant according to plant specific selected criteria.

Example of specific indicator: significant incidents due to hardware/design related causes.

Example of specific indicator: significant incidents due to human related causes.

2.2. PLANT OPERATES WITH LOW RISK

This safety attribute considers the overall risk of the plant and can be monitored using the traditional deterministic approach and the probabilistic approach. It should be noted that the probabilistic and deterministic approaches are not mutually exclusive, but rather, complementary. In this paper, only the deterministic approach is discussed. The IAEA-TECDOC-1141 on “Operational safety performance indicators for nuclear power plants” provides information on indicators based on PSA (probabilistic safety assessment).

The safety attribute ‘plant operates with low risk’ can be monitored by three overall indicators, the number of *‘challenges to safety system’,* the *‘plant ability to respond to such challenges’* and the *‘risk associated to the plant configuration’.*

2.2.1. Overall Indicator: Challenges to Safety Systems

This overall indicator is directly related to plant safety. A low number of challenges translates into a lower possibility of having nuclear transients and/or accidents due to a reduced number of accident initiators.

Strategic Indicator: *actual challenges*

Example of specific indicator: unplanned automatic scrams per 7000 hours critical (WANO performance indicator).

Example of specific indicator: number of demands on RPS/ECCS/RHR/Emergency Power Supply systems.

Strategic Indicator: *potential challenges*

Looking at the actual challenges to safety systems may not provide a very useful measure, since, in general, the number of challenges to safety systems is very small. More and more plants look at low level events in order to get an early warning of future challenges.

Example of specific indicator: number of RPS/ESFAS failures.

Example of specific indicator: number of incipient or partial failures in safety significant BOP systems.

2.2.2. Overall Indicator: Plant Ability to Respond to a Challenge

When a challenge to the plant occurs, the plant should respond in such a way as to prevent any damage to the reactor core, and in the event that some damage occurs, the plant should mitigate the consequences to prevent radioactive releases to the environment. Furthermore, in the event that some radioactive releases to the environment occurs, it is necessary to protect public health and safety.

Strategic Indicator: *safety system performance*

Safety system performance is of obvious importance to plant safety. Safety system unavailabilities can arise from different sources such as equipment failures, performance of maintenance and surveillance tests, and it can also be due to human errors during the performance of tests or maintenance activities.

Example of specific indicator: number of times a safety system is unavailable.

Example of specific indicator: safety system performance (WANO performance indicator).

Strategic Indicator: *operator preparedness*

The operator actions during the course of an abnormal event can be such that they can exacerbate the progression of an accident. Therefore, indicators that monitor this domain can potentially detect areas of deficiency before they become a problem.

Example of specific indicator: errors due to deficiencies in training.

Example of specific indicator: operator errors during accident scenarios in the simulator.

Strategic Indicator: *emergency preparedness*

Emergency management is the last barrier to protect the public if an external radioactive release cannot be avoided. Therefore, the level of preparedness of the plant in order to cope with an emergency also provides a measure of the plant ability to respond to the challenges.

Example of specific indicator: findings during emergency drills.

Example of specific indicator: number of staff receiving training on the emergency plan.

2.2.3. Overall Indicator: Plant Configuration Risk

Different plant configurations happen due to planned and unplanned maintenance activities, operational requirements and occurrence of operational events. It is well known that the risk associated to some plant configurations can be very high. Therefore, it is important to establish the means to monitor this parameter.

Strategic Indicator: *risk during operation*

The most appropriate way to monitor the risk during operation at power is the implementation and use of a PSA based risk monitoring system. However, such a tool is still not available in many nuclear power plants. Even if a PSA or a risk monitor are not available and because of the safety significance of this parameter, it is necessary to find deterministic or engineering based indicators to monitor the risk of the plant during operation at power.

Example of specific indicator: number of technical specification violations.

Example of specific indicator: number of LCO (limiting conditions for operation) entries.

Strategic Indicator: *risk during shutdown*

During shutdown the large amount of maintenance tasks performed and the combinations of system unavailabilities may lead to high risk configurations.

Example of specific indicator: Risk index during shutdown.

2.3. PLANT OPERATES WITH A POSITIVE SAFETY ATTITUDE

The overall indicators chosen to monitor the attitude of the plant staff towards safety are *'attitude towards safety'* and *'striving for improvement*'.

2.3.1. Overall Indicator: Attitude towards Safety

This overall indicator covers implementation and attitudes toward managerial programmes necessary to operate the plant in a safe manner, respecting administrative limits, with low impact on the health and safety of the plant workers.

Strategic Indicator: *compliance with procedures, rules and licensing requirements*

The purpose of the indicator is to assess how well personnel maintain the plant within licensing requirements and comply with other procedures and rules.

Example of specific indicator: number of violations of the licensing requirements.

Example of specific indicator: technical specification exemptions.

Strategic Indicator: *attitude towards procedures, policies and rules*

This is an indication of the attitude of the personnel as a consequence of administrative control policies, level of safety culture, and/or adequacy of training.

Example of specific indicator: ratio of downtime to allowed outage time (AOT).

Example of specific indicator: number of findings in configuration management.

Strategic Indicator: *radiation protection programme effectiveness*

These measures are directed towards control of the sources of radiation, to the provision and continued effectiveness of protective barriers and personal protective equipment, and to the provision of administrative means for controlling exposures of the personnel and contamination of materials and areas in the plant. fdasjdkdkt in the plant.

Example of specific indicator: Collective radiation exposure (WANO performance indicator).

Example of specific indicator: Percentage of controlled area that is contaminated.

Strategic Indicator: *human performance*

The purpose of this indicator is to monitor the influence of human factors on different safety related activities in the plant. It indicates the degree of importance of human errors in these activities.

Example of specific indicator: percentage of events due to human error.

Example of specific indicator: number of human related incidents during testing, maintenance, or restoration.

Strategic Indicator: *Backlog of safety related issues*

This indicator provides a measure of the problem solving capacity of the organization.

Example of specific indicator: number of safety issues in the backlog (analysis phase).

Example of specific indicator: number of safety issues in the backlog (implementation phase).

Strategic Indicator: *safety awareness*

The purpose of this strategic indicator is to assess the level of interest in improving the knowledge of the staff in safety related matters, the openness towards external new ideas and in particular the interest in improving staff attitude towards nuclear safety.

Example of specific indicator: percentage of plant staff trained in safety management/safety culture.

Example of specific indicator: number of seminars on safety related matters.

2.3.2. Overall Indicator: Striving for Improvement

Striving for improvement means the plant has established a strong positive safety culture where continuous improvement is the expected behaviour and a commitment of all employees.

Strategic Indicator: *self-assessment*

Internal safety reviews and audits are very important part in the framework of the plant self-assessment activities. They are performed to assess effectiveness of the plant programmes and procedures, to assess the effectiveness of controls and verification activities, to verify that corrective actions have been planned, initiated, or completed, etc.

Example of specific indicator: number of independent internal safety and QA inspections and audits.

Example of specific indicator: number of external review findings not previously identified by internal reviews.

Strategic Indicator: *operating experience feedback*

Operating experience feedback (OEF) results from reviews of actual events which have happened either at the plant or at other installations.

Example of specific indicator: number of similar or repeated deviations and failures.

Example of specific indicator: number of events at other plants that undergo review/analysis.

3. RESULTS OF THE PILOT STUDIES

The programme development described above was enhanced by pilot plant studies, conducted over a 15 month period from January 1998 to March 1999. The objective of this pilot study was to validate the applicability, usefulness and viability of the approach for implementation at nuclear power plants. A secondary purpose was to obtain feedback regarding the difficulties encountered in implementing the programme and to identify recommendations for adjustments to the framework based upon pilot plant experiences and perceptions. Four NPPs from different countries and with different reactor designs participated in this study.

The activities carried out by the pilot plants were: selection of indicators, review of definitions of indicators, establishment of the necessary organizational support, data collection and analysis, development of support software, and preparation of reports.

The IAEA-TECDOC-1141 on “Operational safety performance indicators for nuclear power plants” provides information on the experience of each participating plants in the selection of indicators, establishment of indicator definitions, identification of specific indicator goals, indicator display and interpretation, logistics and resources required, management involvement and insights and lessons learned. It also presents the operational safety performance indicator systems adopted by each plant.

The four participating plants recognized the inherent value of the concept and framework, and maintained the overall hierarchical organization of indicators. However, each participating plant needed to introduce plant specific adaptations to suit individual data collection systems, plant characteristics, etc.

The pilot plants agreed that the selection of indicators, definitions and goals was an important step in creating a plant specific tool. The process of developing indicator goals and definitions helps to focus the organization on the critical elements of nuclear safety performance that should be measured. However, this process introduces significant variation in how the indicators are defined and measured. This implies that comparison of data and benchmarking among plants utilizing plant specific definitions should be approached with extreme caution. Invalid comparisons can lead to the establishment of inadequate goals and forfeit the benefit that this tool can provide.

4. FINAL REMARKS

This paper has presented a framework for the development of a programme to monitor nuclear plant operational safety performance developed at the IAEA from 1995 to 1999. The framework was derived from the concept that, while safety is difficult to define, it is easy to recognize.

In January 1998, a pilot study was started in order to validate the applicability, usefulness and viability of the approach for implementation at nuclear power plants. The participating plants concluded that the proposed framework provides a good approach.

The efforts described in this paper are documented in more detail in the IAEA-TECDOC-1141 on “Operational safety performance indicators for nuclear power plants”, to be published shortly.

Despite the efforts described, it is clear that additional research is still necessary in areas such as plant-specific adaptation of proposed frameworks in order to suit individual data collection systems and plant characteristics, indicator selection, indicator definition, goal setting, action thresholds, analysis of trends, indicator display systems, analysis of overall safety performance (i.e., aggregation or combination of indicators), safety culture indicators, qualitative indicators, and use of additional indicators to address issues such as industrial safety attitude and performance, staff welfare, and environmental compliance. This is the rationale for a new IAEA Coordinated Research Project on “Development and application of indicators to monitor NPP operational safety performance” that was started in 1999. The objective of this project is to foster the co-ordination of efforts and the exchange of information and experience among NPPs world-wide in the field of operational safety performance indicators.

ACKNOWLEDGMENTS

The author wishes to acknowledge the contribution of all experts and IAEA staff members who participated in the development of the work presented in this paper. In particular, the author gratefully acknowledges the efforts made by the four nuclear power plants that participated in the pilot study.

RESULTS OF THE WGIP BALTIMORE WORKSHOP SESSIONS RELATED TO PIS

J. J. Van Binnebeek

Director Operational Projects & Inspections

AVN

Avenue du Roi 157

B-1190 Brussels (BELGIUM)

Tel.: +32 25368355

Fax.: +32 25368585

e-mail: vbk@ avn.be

ABSTRACT

Regulators attempt to evaluate the safety performances of the licensees by using objective indicators.

The Baltimore workshop organised by the WGIP (working group of Inspection practices) of the CNRA, OECD addressed such a usage of objective indicators and discussed in details their regulatory usage: scope, outcome, objectives, advantages/disadvantages, criteria for good PI, use of PI for regulatory assessments.

The overall results are presented here, namely the intent of the regulatory use of objective safety indicators, the practical usage of the indicators, a discussion on their benefits and limitations, some criteria to define an adequate indicator, as well as considerations on the combination of indicators with other data. This will be followed by some general considerations and conclusions.

INTRODUCTION

Safety indicators have always been used informally by regulators to trigger investigations or initiate regulatory actions.

The last years a tendency developed itself to obtain "objective safety indicators" to evaluate , and if possible to measure, the safety of the regulated nuclear power plants.

This tendency seems to have appeared from the need the regulators, as many others in different industries, are in to be transparent and objective. Hence the necessity to show or prove allegation concerning the safety of NPPs.

Such a use of indicators is based on the postulate that everything is numerically measurable so that the indicators can be used to define clear objectives and to introduce improvement feedbacks. This is also a key issue in the development of quality systems, which are very comparable to safety systems.

When the WGIP first addressed the PI issue a few years ago, it collected the practices of its member countries. Besides the overall informal use of PIs by everybody for day to day inspection tuning, two groups of countries actually emerged: the countries with a large number of plants, which wished tools to be consistent or to define comparable policies, and the countries with a smaller number of plants, which could evaluate the safety of their plants on a more informal way.

The WGIP continued afterwards to try to come up with a consensual approach of indicators. As a step in that directions, and with in mind the US policy of development of public safety PIs to modulate the inspection programme, the WGIP found useful to collect information on a broader way through its biannual workshop. This paper summarises the main consideration made by two discussion groups on the subject.

INTENT OF THE REGULATORY USE OF OBJECTIVE SAFETY INDICATORS

It appears from the countries' practices that there is a spectrum of regulatory usage of Performance Indicators. Usage ranges from

- concentration on "outcomes" which tends to result in "high level" PIs;
- concentration on "process", which can result in finely focussed PIs.

The former "measures" licensee's performance outcomes, leaving monitoring of processes to licensees.

The latter allows regulators to monitor trends in licensee performance at a low level as a means to determine the effectiveness of the licensee's safety management system and licensees can use them in their improvement processes.

Usage by individual countries varies, and individual regulators use various approaches along the spectrum.

A fundamental principle is that, regardless of the way in which PIs are used, there must be a reasonable assurance, supported by evidence, that the PIs are valid for that purpose.

OBJECTIVES OF DEVELOPPING PIs

The various identified possible uses of PIs are:

- As part of a structured, formal process for communication within and between the RB and licensees.
- To identify off-normal conditions to trigger regulatory actions.
- In combination with other information processes, to improve the focus of the regulator's activities.
- To provide information to stakeholders (the degree of information & the stakeholders informed depend on the country's practice).
- To provide a "measure"[1] of effectiveness of other regulatory tools.
- To facilitate efforts of licensees to improve their safety performance (e.g., through appropriate benchmarking).

The first three items are clearly a formalisation of the informal use of indicators by the regulator. A systematic definition of a set of indicators may allow a structured approach by the regulator.

The next two items are more communication related, allowing third parties to get an idea of the plant safety. They might however be the subject misinterpretations.

The last objective could be a help for the licensee, but one has to be careful not to divert the PIs from their use.

BENEFITS AND LIMITATIONS OF THE SAFETY INDICATORS

The analysis of the benefits and the limitations of the use of PIs. This led to the following enumerations:

Potential benefits

- The PIs are objective, auditable, and not disputable
- When used as a set, PIs provide insights regarding what is important for safety
- A structured set of PIs provide can provide information that is understandable to all stakeholders
- The PIs provide additional bases for investigation by the regulatory body
- The PIs are relatively low cost (i.e., easy to report) and easy to evaluate
- The PIs enable comparisons or benchmarking
- The PIs encourage licensees to monitor performance at a lower level
- The PIs can promote licensees own improvement processes

The validity of these benefits are depending on the honest and rational use of the PIs. Deviations from such a philosophy may impair the adequacy of the conclusions drawn from PIs.

Potential limitations

- PIs cannot to be used alone
- PIs are difficult to define without ambiguity
- PIs may be misconstrued as providing a measurement of safety level rather than an indication of a particular aspect of performance
- PIs may be subject to misuse or manipulation
- Data collected by the utility must be verified to ensure its accuracy
- PIs may not provide timely indication of trends in safety performance
- PIs are not effective unless used as part of a full set of PIs that provide information regarding a spectrum of activities or attributes

- It is difficult to develop and collect PIs for non quantifiable issues (e.g. program effectiveness, management effectiveness)

- PIs may be of limited value in comparing plants where differences between plant types are great

- Regulators may lack legal powers to require licensees to collect PIs

- PIs cannot replace qualitative judgements

The preceding reservations must be kept in mind when developing as set of PIs. The limitation with respect to non quantifiable issues shows that PIs cannot completely replace human judgment.

Cautions when developing and using PIs

The following pitfalls were identified when developing a set of PIs:

- PIs are most effective when they provide information that is sufficiently timely to allow the regulator to identify adverse trends in safety performance before a significant degradation has occurred.

- It is important to have as complete a set of PIs as possible

- PIs Should not be used alone. Rather, they should be used along with other performance insights.

The first aspect is particularly important, as a present degradation of safety may be masked by still good PIs. It is however essential for the regulator to be able to anticipate a safety degradation. Hence the use of other safety evaluation techniques to supplement the use of PIs.

CRITERIA FOR GOOD PI

A good PI is one that meets the following criteria:

- It is resistive to manipulation, misuse, and misunderstanding.

- The definition of PI is clear, concise, and precise to make sure different observers given the same input are able to produce the same results.

- There exists a clearly defined, logical relationship to the safety regulator’s objectives.

- In combination with other information, it enables timely indication of safety degradations.

- It is measurable and quantifiable to the extent possible.

- It is relatively easy to define, report, and evaluate.

- It does not result in licensees taking action contrary to safety.

The previous conditions form a set which is not obvious to be met, let alone to be proven to be met. This shows the challenge behind the definition of a coherent set of safety PIs.

COMBINING PI AND OTHER INSIGHTS

As a final touch it appears that PIs must be used in combination with other insights. The following elements give guidance on the possible ways to use PIs:

- PIs should always be combined with other objective and subjective inputs (such as inspections, investigations into events and risk informed data), and be evaluated on a collegial manner.

- The way in which such insights are combined depends on the purpose of each particular use.

- When information (including PIs) is combined for a particular use it should be incorporated into the normal, systematic, process of preparing for regulatory action or decision taking.

Accordingly the use of PIs should be integrated in a complete regulatory inspection framework, where PI's and various other data should confirm their respective conclusions.

CONCLUSIONS

The WGIP workshop sessions on objective PIs has issued recommendations on the overall definition, use and integration of safety related PIs, pointing out conditions which must be met in order that the regulator is not misled and keep a sensible approach on the use of PIs.

This is particularly important when the so-called "objective indicators" are used and interpreted by stakeholders too far from the field, who might take indicators or investigation trigger as a true and absolute measure of safety.

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

WANO Performance indicators

Kent W. Hamlin, P.E.

World Association of Nuclear Operators

Kings Buildings

16 Smith Square

London SW1P 3HQ

United Kingdom

Tel: +44 (0)207 828-2111

Fax: +44 (0)207 828-6691

e-mail: hamlin@wanocc.org

Yuichi Shimada

World Association of Nuclear Operators

Kings Buildings

16 Smith Square

London SW1P 3HQ

United Kingdom

Tel: +44 (0)207 828-2111

Fax: +44 (0)207 828-6691

e-mail: shimada@wanocc.org

Abstract

The WANO performance indicator programme supports the exchange of information by collecting, trending, and disseminating nuclear plant performance data in key areas. The data is gathered for a set of quantitative indicators of plant performance in the areas of nuclear plant safety and reliability, plant efficiency, and personnel safety.

WANO has completed a review of this set of indicators with a few changes as noted in this paper.

Overall, performance at WANO member nuclear power plants continues to improve, as it has for the past ten years.

PAPER CONTENTS

The WANO performance indicators are intended principally for use as a management tool by nuclear operating organisations to monitor their own performance and progress, to set their own challenging goals for improvement, and to gain additional perspective on performance relative to that of other plants.

WANO published and distributed the first performance indicator report in April 1991. By the end of 1992, several programme milestones had been reached. A Performance Indicator Working Group, with representatives from each region, had completed its initial work: agreement on the definitions for each indicator for each reactor type. The frequency of report preparation and distribution was changed to yearly in 1996, and improvements were made to allow individual plants to compare their performance more easily with industry average values.

In 1993, reporting of data began for all reactor designs. The level of reporting has grown to 100 percent of the operating nuclear power plants reporting at least four indicators, 98 percent reporting at least seven indicators, and nearly 60 percent of the plants reporting data for all 10 indicators.

It is expected that the use of WANO performance indicators will encourage emulation of the best industry performance. It should also further motivate the identification and exchange of good practices in nuclear plant operations.

Characteristics of a Good Indicator

The following are characteristics of an ideal performance indicator, as seen from WANO's perspective. It is recognised that no indicator has all of these traits, but groups should approach with caution an indicator definition that falls short of too many of these characteristics:

- The indicator is useful to plant and utility management as a device for encouraging comparison, emulation, and goal setting (as appropriate) amongst WANO members.
- The indicator has a strong and obvious tie to the WANO mission (e.g., safe and reliable power plant operation) for all reactor designs. Indicators not clearly tied to the WANO mission should be left to others to develop.
- The indicator is easily understood—the definition as simple as possible without impairing the technical and operational significance of the indicator.
- The indicator does not, by its nature, encourage plant staffs to take actions that are non-conservative from a safety or reliability perspective in order to achieve a good indicator result.
- The indicator has meaning for all different reactor types.
- The indicator represents a practical approach that is not likely to be misconstrued by indicator users.
- The indicator is not easily manipulated.
- Indicator data collection and reporting are straightforward.
- The indicator should have limited risk of misuse outside the industry.

Revisions to Performance Indicators

With the above characteristics in mind, over the past two years WANO has conducted a review of the set of WANO performance indicators with an eye for any necessary changes. This process relied on extensive input and involvement of WANO members and multiple review groups. The changes will take effect with data reporting for 2001. As a result of this review, WANO will no longer use the Thermal Performance Indicator and the Volume of Solid Radioactive Waste Indicator. Additionally some clarifications to definitions were developed. These review groups also examined development of a risk-based indicator for safety systems, and an event-based indicator. Although satisfactory indicators were not developed for these two areas, they remain of interest for future improvements to the set of WANO performance indicators.

Performance Indicator Results

For several years, WANO members have agreed to share performance indicator data for individual nuclear power plants amongst members in keeping with WANO's Confidentiality Policy. This allows open and candid exchange of information amongst WANO members in keeping with the spirit of WANO's mission.

Additionally, WANO has made public the overall performance indicator results on an annual basis. The following are the overall WANO results through year end 1999.

Unit Capability Factor

Unit capability factor is the percentage of maximum energy generation that a plant is capable of supplying to the electrical grid, limited only by factors within control of plant management. A high unit capability factor indicates effective plant programmes and practices to minimise unplanned energy losses and to optimise planned outages.

Unplanned Capability Loss Factor

The unplanned capability loss factor is the percentage of maximum energy generation that a plant is not capable of supplying to the electrical grid because of unplanned energy losses, such as unplanned shutdowns or outage extensions. A low value indicates important plant equipment is well maintained and reliably operated and there are few outage extensions.

Unplanned Automatic Scrams per 7,000 Hours Critical

The unplanned automatic scrams per 7,000 hours critical indicator tracks the mean scram (automatic shutdown) rate for approximately one year (7,000 hours) of operation. Unplanned automatic scrams result in thermal and hydraulic transients that affect plant systems.

Collective Radiation Exposure

The collective radiation exposure indicator monitors the effectiveness of personnel radiation exposure controls for boiling water reactors (BWRs), pressurised water reactors (PWRs), pressurised heavy water reactors (PHWRs), light-water-cooled graphite reactors (LWCGRs), and gas-cooled reactors (GCRs). Low exposure indicates strong management attention to radiological protection.

Industrial Safety Accident Rate

The industrial safety accident rate tracks the number of accidents that result in lost work time, restricted work, or fatalities per 200,000 work-hours. The nuclear industry continues to provide one of the safer industrial work environments.

Volume of Solid Radioactive Waste

This indicator monitors the volume of solid radioactive waste produced per unit for boiling water reactors (BWRs), pressurised water reactors (PWRs), pressurised heavy water reactors (PHWRs), light-water-cooled graphite reactors (LWCGRs), and gas-cooled reactors (GCRs). Minimising radioactive waste reduces storage, transportation, and disposal needs, lessening the environmental impact of nuclear power.

Thermal Performance

Thermal performance monitors how efficiently a plant converts thermal energy into electrical output. A high thermal performance indicator reflects high thermal efficiency. Efficient, well-tuned plants produce more electrical energy and enable operators to detect abnormal trends and correct them early, contributing to more reliable operations.

WANO monitors three additional performance indicators: safety system performance, fuel reliability, and chemistry performance. These indicators are defined in a manner that reflects differences in plant-specific designs, configurations, or operational practices. As a result, data cannot be meaningfully summarised across reactor types.

Safety System Performance

The safety system performance indicator monitors the availability of three important standby safety systems at each plant. Safety systems that are maintained in a high state of readiness have a high probability of being capable of mitigating off-normal events.

Fuel Reliability

The fuel reliability indicator monitors progress in preventing defects in the metal cladding that surrounds fuel. Maintenance of fuel cladding integrity reduces radiological impact on plant operations and maintenance activities.

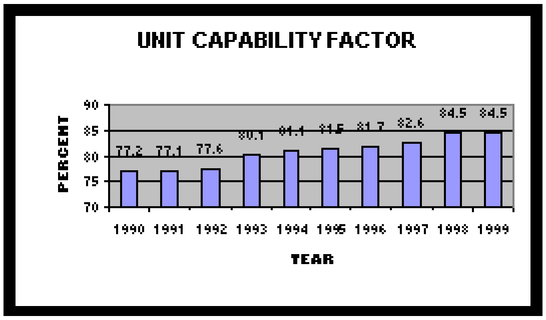
Chemistry Performance

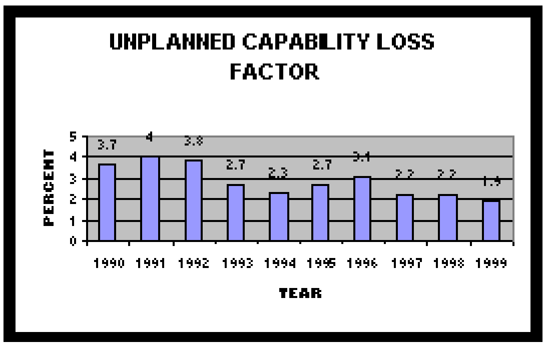
The chemistry performance indicator provides an indication of progress in controlling chemical parameters to retard deterioration of key plant materials and components. These parameters are already being maintained within strict guidance developed by the industry.

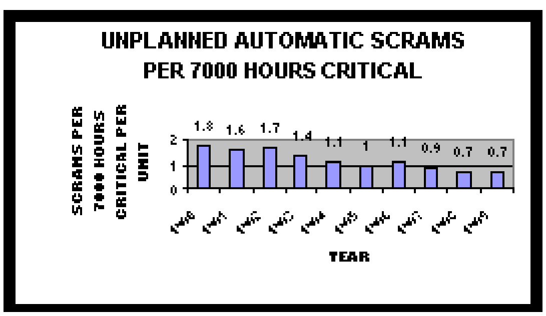
Summary

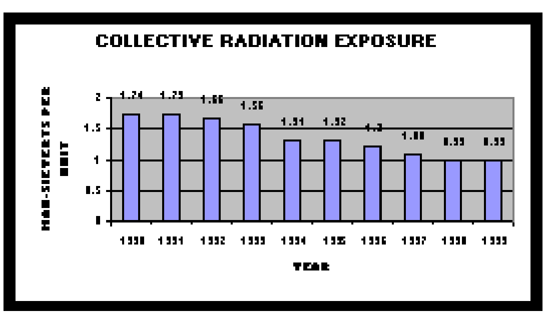
WANO performance indicators provide WANO members with a means of exchanging plant performance data to support communication, comparison, and emulation of industry best practices.

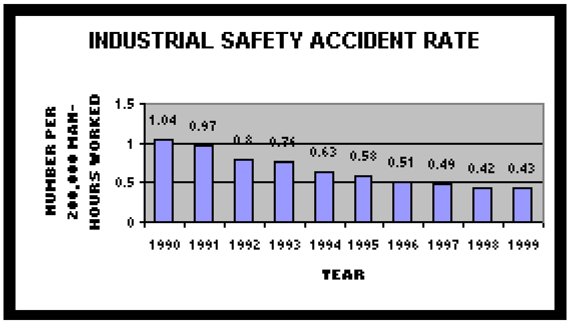
Although these indicators recently underwent an extensive review for potential changes, the majority have withstood the test of time and continue to be a valid means of comparison.

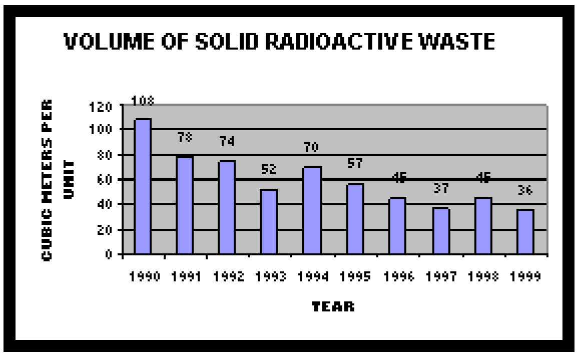

UNIT CAPABILITY FACTOR
PERCENT
90
85
80
75
70
77.2
77.1
77.6
80.1
81.1
81.5
81.7
82.6
84.5
84.5
1990 1991 1992 1993 1994 1995 1996 1997 1998 1999
YEAR

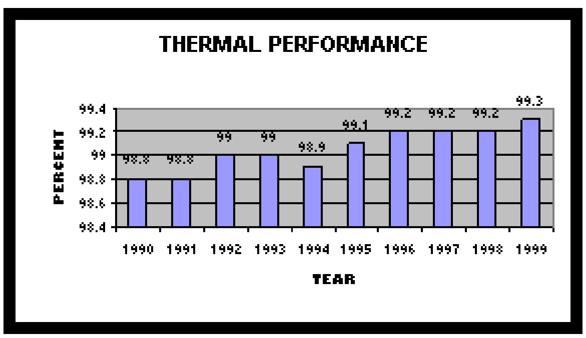

UNPLANNED CAPABILITY LOSS FACTOR
PERCENT
5
4
3
2
1
0
3.7
4
3.8
2.7
2.3
2.7
3.1
2.2
2.2
1.9
1990 1991 1992 1993 1994 1995 1996 1997 1998 1999
YEAR


UNPLANNED AUTOMATIC SCRAMS PER 7000 HOURS CRITICAL
SCRAMS PER 7000 HOURS CRITICAL PER UNIT
2
1
0
1.8
1.6
1.7
1.4
1.1
1
1.1
0.9
0.7
0.7
YEAR


COLLECTIVE RADIATION EXPOSURE
2
1.5
1
0.5
YEAR







Notes: (1) The median of plant values is displayed for all indicators except unplanned automatic scrams per 7,000 hours critical, where the mean of plant values is shown, and industrial safety accident rate, which is an overall industry value (summation of plant values).

(2) Half of the plant values are above and half are below the displayed median values. The mean is the arithmetic average of the plant values. The median value is normally displayed rather than the mean value, because the median value is less susceptible to the influence of outliers and is, therefore, more representative of overall performance.

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

Reportable Events as Safety Indicators for Bavarian Nuclear Power Plants?

Ernst R. Seidel,
Bavarian State Ministry for State Development and Environmental Affairs
Rosenkavalierplatz 2, 91925 Munich, Germany
Phone: +49-89-9214 2448
Fax: +49-89-9214 2286

e-mail: ernst.seidel@stmlu.bayern.de

Günter Straub
TÜV Süddeutschland Bau und Betrieb GmbH
Westendstr. 199, 80686 Munich, Germany

Phone: +49-89-5791 1179
Fax: +49-89-5791 2696

e-mail: guenter.straub@tuevs.de

Abstract

Integral analysis of reportable events by the supervisory authority using selected safety indicators and the analysis of the results provide important information on the significance in safety-engineering terms of the events and the current technical safety status of the nuclear power plant and its mode of operation. The assessments determined for the individual safety indicators form the basis for further investigations, examinations and the enforcement of necessary improvements by the supervisory authority. An investigation is currently being conducted to examine the extent to which it is possible to deduce the overall safety assessment of the event by linking the assessments for the individual safety indicators using fuzzy logic.

Introduction

The peaceful use of nuclear energy has a long tradition in the Free State of Bavaria, one of the 16 Länder (federal states) of the Federal Republic of Germany. As early as 1957, the first research reactor in the Federal Republic of Germany went into operation in Garching near Munich. It is almost forty years ago since the first nuclear power station in the Federal Republic of Germany, the Versuchsatomkraftwerk Kahl (VAK) in the Bavarian town of Kahl, began producing electricity. Today, five nuclear power plant units with a total electrical output of around 6400 MW produce 2/3 of the

total electricity consumption in Bavaria. These plants were put into operation between 1977 and 1988. Together with hydroelectric power, around 80 % of the electricity produced in Bavaria is CO_2-free.

The high operating availability of the Bavarian plants during the past 10 years is shown in Fig. 1. The number of reactor scrams and reportable events as well as the collective radiation doses of the internal and external personnel show a slight downward tendency at a low level. This development is due above all to the fact that in the past, the operators have continuously improved the safety of their plant by extensive backfitting measures, by further increasing the quality assurance, through consistent orientation towards elements of safety culture and through periodic safety reviews including probabilistic safety analyses.

Due to these measures carried out by the operators, the safety standard of all Bavarian nuclear power plants was continuously brought up to the latest state-of-the-art in science and technology. Therefore, the safety of the plants has increased to a very significant degree since the time at which their nuclear operating licenses were issued. This has been confirmed by the Internationale Länderkommission Kerntechnik (International Nuclear Technology Commission, ILK), jointly appointed by the Free State of Bavaria, and the Länder Baden-Württemberg and Hesse, in July 2000 in a report on the safety of the utilisation of nuclear energy in Germany[1].

However, the high safety standard of the Bavarian nuclear power plants is also due to the strict yet appropriate supervision by the Bavarian Ministry for State Development and Environmental Affairs (BStMLU). This supervisory procedure is performed with the support of TÜV Süddeutschland (TÜV) as expert organisation and uses a number of instruments which consider aspects of man, technology and organisation in an integrated approach (s. Table 1).

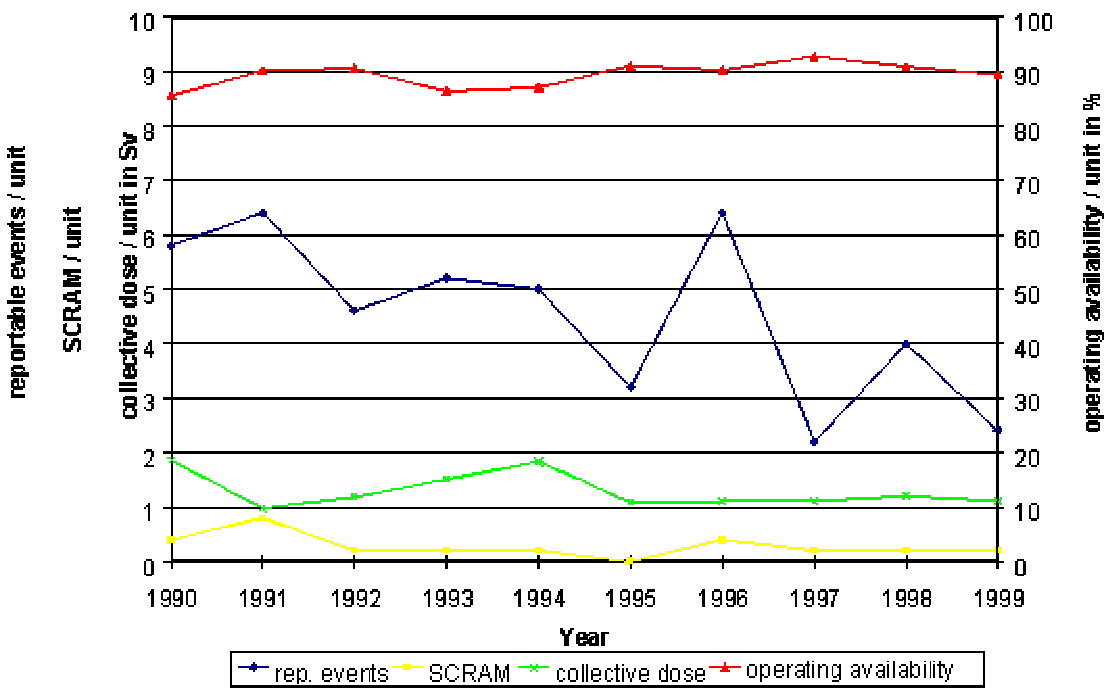


Fig. 1:

Performance indicators for Bavarian nuclear power plants

1	Inservice inspection of systems, structures and components
2	Site inspection of the plant and its personnel during operation and shut down periods
3	Quality assurance measures during maintenance
4	Assessment of modifications to the plant and its mode of operation
5	Evaluation of reportable events
6	Independent monitoring of radioactive emissions and their impact on the environment
7	Assessment of operator reports
8	Periodic safety reviews

Table 1:

Integrated approach for the supervision of nuclear power plants

The recent liberalisation of the electricity market within the European Union has put considerable pressure on the companies operating the Bavarian nuclear power plants with respect to competitiveness and costs. It is therefore expected that in future, the plant operators will apply strict criteria on their willingness to carry out voluntary additional safety improvements, which will also depend on the remaining time of operation for the respective plant. Even if the respective plant continues to formally fulfil the nuclear safety regulations and the conditions of the license, there is a risk that over the course of time, the overall safety level of the plant will fail to keep up with the latest state-of-the-art in science and technology. Due to technological ageing of the plants and the cost-induced reduction of the plant staff, the BStMLU as the competent supervisory authority will have to face new challenges.

Since the regulatory supervision quite naturally cannot exercise 100% control over the technical, organisational and personnel situation of a nuclear power plant, the supervisory authority increasingly has to rely on safety indicators which can deliver a sufficiently reliable and fast assessment of the safety level of a plant and its operating mode on the basis of incomplete information. These safety indicators should also help the supervisory authority to distinguish between essential and non-essential safety improvements.

The title of the paper at first suggests that the number and reporting category of reportable events in nuclear power plants can be considered as indicators for the safety of these plants. However, this approach does not enable a systematic and objective safety analysis because the low number of events, for example, is subject to strong statistical variations. Nevertheless, as is shown below, the reportable events can be evaluated using selected safety indicators in such a way that important information about the current safety condition of the respective plant can be obtained.

Reportable events in Bavarian nuclear power Plants

According to the Ordinance on Reportable Events[2], the nuclear power plant operators are obliged to report to the supervisory authority

accidents and any other significant occurrences. The aim of the reporting procedure is to enable the supervisory authority to identify possible deficiencies already at an early stage and, if necessary, to enforce preventive actions. Reporting of events has to follow specific reporting criteria. For plants in operation, these reporting criteria are assigned to the categories N (normal), E (urgent) and S (immediate), which determine type and deadlines for reporting. The categories are a measure of the significance of the event in safety-engineering terms and are associated with different obligations for preventive actions of the supervisory authority. The reportable event, its causes, effects and removal as well as preventive measures against a recurrence must be described in an official reporting form. In case of events which were caused by a damage to systems or components or in the course of which damages to safety relevant systems or components occurred, the plant operators are legally obliged to take measures to preserve evidence in order to enable the clarification and subsequent inspection of the exact causes and consequences of the event at a later date. In addition, the plant operators are also obliged to categorise every reportable event according to the seven levels of the International Nuclear Event Scale (INES)[3].This scale is used to inform the general public about the significance of a particular event with special regard to the safety of the plant and to whether or not it had or could have had any radiation impacts on the public or the environment.

The reportable events that have occurred at the Bavarian nuclear power plants during the past 5 years are shown in Table 2 as well as in Fig. 2 and 3. 98% of the reported events were category N. Only two events were reported according to category E. In each case, one of two containment isolating valves did not close properly during an inservice inspection. Since only the necessary number of safety installations – instead of the usual double redundancy - was available, the event was classified according to category E.

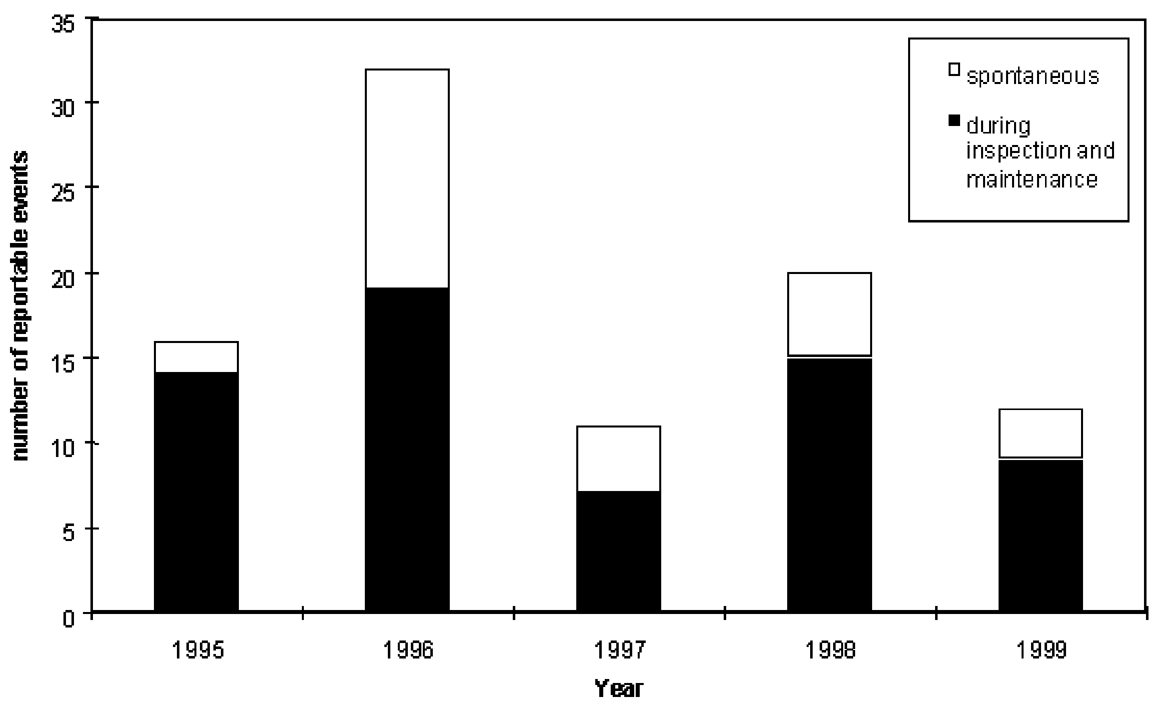


Fig. 2:

Number of reportable events in Bavarian nuclear power plants according to the kind of occurrence

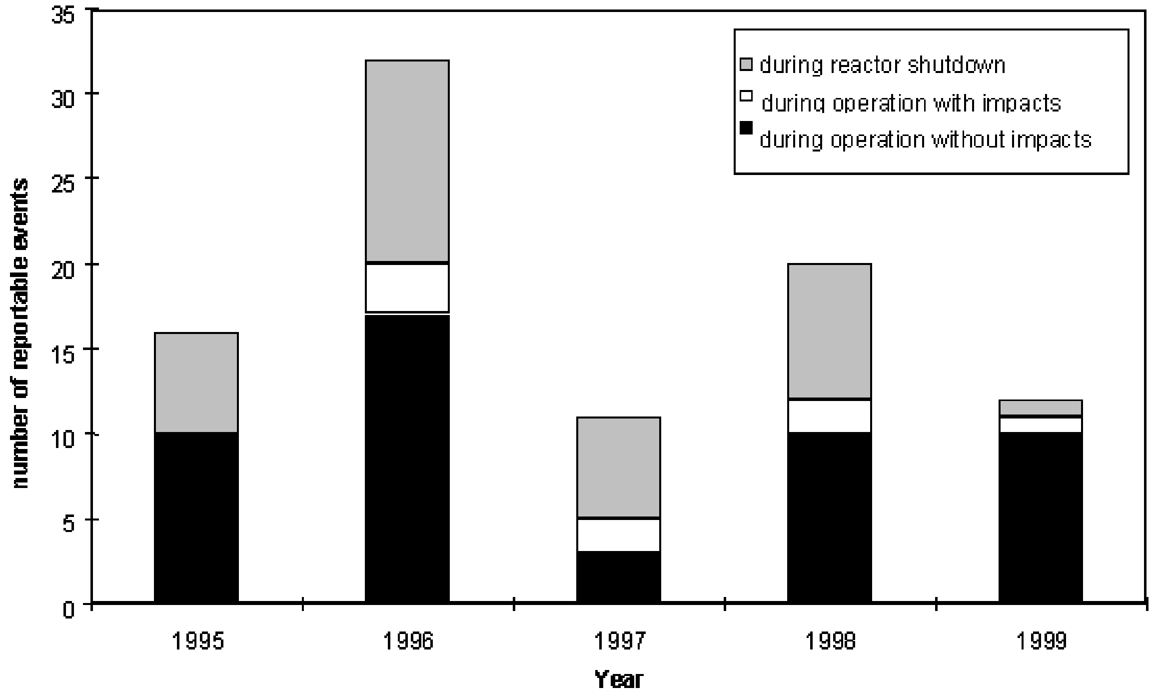


Fig. 3:
Number of reportable events in Bavarian nuclear power plants according to mode of and consequence on operation

		Reporting Category			INES-scale		
		S	**E**	**N**	**0**	**1**	**≥ 2**
1995	16	0	0	16	16	0	0
1996	32	0	0	32	31	1	0
1997	11	0	1	10	11	0	0
1998	20	0	1	19	20	0	0
1999	12	0	0	12	12	0	0

-

Table 2:
Number of reportable events in Bavarian nuclear power plants according to the different reporting categories

All events had no effects on the operating personnel or the environment. Only one event was classified according to INES level 1. The reason was a deficiency in the quality assurance system identified during the investigation of the event.

A copy of the reporting form is sent by the BStMLU to the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU). BMU centrally collects and documents the nuclear events reported from all the five Länder in which nuclear power plants are operated. If the in-depth analysis shows a significance and applicability to the safety of other German nuclear power plants, BMU will distribute an information notice to the supervisory authorities and expert organisations. As a consequence, the plant operators have to submit a comment on each information notice to the competent supervisory authority with special emphasis on the implementation of recommendations.

Investigation and analysis of the operator’s Event report by the supervisory authoritY with the HELP of safety indicators

After having been informed about a reportable event by the operator, the BStMLU usually entrusts the TÜV as expert organisation to carry out a comprehensive safety assessment of the event. Based on the completed reporting form and preliminary verbal information from the operator, TÜV personnel carry out further examinations on site. TÜV and – if necessary – also representatives of the BStMLU interview the responsible plant personnel about causes, development and the impact of the reportable event on safety systems, the operating personnel and the environment. Usually, the operator has to submit a detailed nuclear event report. This report must also include details about removal of the consequences of the event, restoration of a plant status to the necessary safety requirements and, if applicable, improvements required to prevent a recurrence of the event. In some cases, the operator is obliged to make material and activity samples available for investigation by an independent expert organisation.

In addition, the remote surveillance system for nuclear reactors owned by the authority delivers data about certain plant parameters and especially on the radioactive emissions of the plant at the time the event occurred.

In order to enable a comprehensive and integral evaluation of the event, the investigations cover the whole system man – technology – organisation. Six safety test areas are considered in more detail for the safety assessment of the event. The individual safety test areas are assigned to a total of 15 safety indicators (s. Table 3), which experience has shown to have a high degree of significance for the safety level of the plant and the operative management. By investigating a specific event with respect to these safety indicators, it is possible to generate a systematic and objective integral assessment of the event. In order to make the assessment reconstructable for the operator in an optimal way, the importance of every safety indicator can be characterised according to a scale consisting of five levels. These levels are:

"insignificant",
"fairly significant",
"significant",
"very significant",
"extremely significant".

Safety test area	Safety indicators
1 Initiating events	1 Events which have made the activation of safety installations

	necessary
2 Safety systems	2 Malfunctions, damage or breakdown of safety installations which are necessary for the control of design basis accidents
	3 Identification of a common cause failure
3 Integrity of activity	4 Activity of the reactor coolant
barriers	5 Leak tightness of the containment
	6 Integrity of the primary circuit and other systems carrying radioactive materials
	7 Activity in intermediate cooling circuits or in the secondary circuit
4 Radiation exposure of the operating personnel and	8 Compliance with dose limits for radiation exposure as provided by the Radiation Protection Ordinance
the general public	9 Radioactive contamination within radiation protection areas above limiting values
	10 Dispersion of radioactive material beyond plant boundaries, also by shipping casks transporting spent fuel elements
	11 Uncontrolled release of radioactive material
5 Operative management	12 Compliance with the operating manual regulations, the regulations of the license and with supplementary and other requirements imposed by the supervisory authority
	13 Deficiencies in safety culture (including human factors)
6 Plant security	14 Unauthorised intervention; damage or failure of security equipment
	15 Incorrect action by the security service personnel

Table 3:
Safety test areas and assigned safety indicators for the integrated safety analyses of reportable events

These safety indicators are used to investigate all aspects of an event relevant to safety engineering and plant security. The safety test areas 1-4 (safety indicators 1-11) refer to the impact of the event. They are based on the well-known protection goals

- Control of reactivity

- Cooling of fuel elements

- Confinement of radioactive material

- Limitation of the radiation exposure

The safety test areas "Initiating events" and "Safety systems" concern the protection goals "Control of reactivity" and "Cooling of fuel elements". The following questions are subject for investigation:

- Was it an initiating event? Can the event be assigned to an initiating event? How significant is this initiating event with respect to the plant-specific probabilistic safety analysis (PSA)? What is the expected failure rate? (safety indicator 1)

- Was there a malfunction, damage or a breakdown in the safety system? How far did the malfunction extend? Was the safety function of a component, train or system still guaranteed? How many redundancies were affected? Is there enough diversity in the safety system? (safety indicator 2)

At the same time, special consideration is given to whether a common cause failure occurred or whether the event can be associated with such aspects as:

- Was there a common cause failure? Are there any hints indicating a common cause failure? (safety indicator 3)

The safety test areas "Integrity of the activity barriers" and "Radiation exposure of the operating personnel and the general public" are based on the protection goals "Confinement of radioactive material" and "Limitation of radiation exposure". The following questions are of interest:

- Was the integrity of activity barriers destroyed, endangered or diminished? (safety indicators 4-7)

- Were statutory limits for radiation exposure or radioactive contamination within radiation protection areas exceeded? To what extent? Was there a risk that limits could have been exceeded? (safety indicators 8-9)

- Was radioactive material dispersed outside of the plant or subject to uncontrolled release? To what extent? (safety indicators 10-11)

Apart from these indicators, further aspects of the event are also considered for the safety assessment. These aspects include operative management and plant security issues. Consequently, an event can be assigned to a higher safety significance than would be the case if the event were analysed on the basis of protection goal-oriented safety indicators alone. The following questions have to be answered:

- Were the requirements of relevant regulations fulfilled? (safety indicator 12)

- Was the event caused by organisational problems? Was the event triggered by erroneous action by plant staff? Which circumstances facilitated the error? Was there a deviation from operating instructions? Did mistakes play a role in the course of the event? (safety indicator 13)

- Was there any unauthorised intervention; was there a damage to or breakdown of the security equipment? To what extent?

(safety indicator 14)

- Did the security service personnel fail to act correctly? (safety indicator 15)

The evaluation of the individual safety indicators is carried out on the basis of many years of engineering experience as well as a detailed knowledge of the plant and finally leads to the overall assessment of the event. For the categorisation of the safety importance of the event according to the scale mentioned above, further aspects such as the frequency of similar types of events can play a role. The importance category of the overall assessment must, however, be at least as high as the highest individual classification.

Measures taken by the superviso-ry authority

The BStMLU is deeply convinced that the acceptance of the peaceful use of nuclear energy by the public primarily depends on a completely open information policy of both the operators and the supervisory authority. In this way, rumours, misconceptions and unjustified fears can be very effectively overcome. Therefore, it was voluntarily agreed between the operators of the Bavarian nuclear power plants and the BStMLU that each reportable event will be published in the Internet on the homepage of the operator only a short time after it occurs. The BStMLU also reports on the event on its homepage a short time after the TÜV has submitted its preliminary safety assessment. The internet report of the BStMLU deals in particular with the question of whether the event had an impact on the operating personnel or the environment. If necessary, the internet reports are adjusted to the current situation.

In case of particularly safety relevant events, the public is informed by the plant operator and the BStMLU additionally through press releases and - if necessary - press conferences.

The basis for measures to be taken by the BStMLU as the supervisory authority over the operator, is the overall assessment of the reportable event established by TÜV according to the procedure described. The BStMLU uses it to check

- if the operator has categorised the reportable event in the correct reporting category and

- if the operator’s measures for restoration of the plant safety status are sufficient to meet all necessary requirements and to prevent a recurrence of the event

(s. Table 4).

Assessment of causes, effects and measures to prevent recurrence	**Publication of the reportable event and measures for enforcing the safety requirements of the supervisory authority**

• Performance of additional examinations in the plant and questioning of the responsible personnel • Enhanced supervision of other parts of the plant • Safety assessment of the reportable event with respect to cause, the development of the event and its effects • Evaluation of the operator's measures for restoration of a plant status which meets all necessary requirements • Evaluation of the operator's measures to prevent a recurrence of the event	• Rapid publication in the Internet • In significant cases press releases and press conferences • Injunction to take measures for restoration of a plant status which meets all necessary safety requirements • Permit of restart after refuelling • Injunction to prevent risk to life, health and property (e.g. shut-down) • Imposition of additional licensing conditions • Revocation of the license

Table 4:
Measures of the supervisory authority in the case of reportable events

The BStMLU is primarily interested in a trustworthy dialogue between the supervisory authority and the operator. The aim is to reach a solution which is mutually acceptable to both sides. The assessment of the event using safety indicators offers the best prerequisites because the individual steps of the supervisory assessment can be clearly reconstructed by the operator.

In the rare case that this procedure fails due to irreconcilable differences of opinion, the supervisory authority can order special safety measures and safety inspections as well as an interruption of operation if deviations from legal provisions or the licensing conditions or dangers for life, health and ownership of third parties are detected. The supervisory authority can impose additional licensing conditions or – as the final possibility – revoke the license if the operator is not willing to follow the orders of the supervisory authority (s. Table 4). The operator can have the lawfulness of these administrative acts examined by a court. In this context, it is of particular significance that the Bavarian plant operators need a supervisory permit for restart after each refuelling. This permit is only granted by the BStMLU, if there are no doubts regarding the safety of the plant.

Practical Example

During an inservice inspection performed in a pressurised water reactor in the course of a planned outage for refuelling, it was observed that all shut-off valves of the measuring transducer of one of the hot-side accumulators were closed and the equalising valves had been left open. All the valves were found sealed in this wrong setting. During the subsequent control of the measuring devices of the other accumulators, a further case of a wrong setting was discovered. The wrong setting of the measuring transducer valves had existed since the previous refuelling outage, during which maintenance works had been carried out on these measuring transducers.

The accumulators are passive components of the emergency cooling system. Their task is to make large quantities of water available quickly for cooling the reactor core in the low-pressure range in the case of a large loss of coolant accident. When required, the water is fed in by a nitrogen gas blanket located above the water column.

The water level meter concerned is used to trigger the shut off of the accumulator in the case of the water level falling below the lower limit. Because of the wrong valve setting, this function was deactivated. Therefore, if the accumulator had been actuated, the nitrogen could have entered the reactor cooling circuit .

The analysis of the event using safety indicators provided the following result (s. Table 5):

Safety test area	**Safety indicator**	**Assessment of the safety indicator**
1 Initiating events	1	Insignificant
2 Safety systems	2	**Fairly significant**
	3	**significant**
3 Integrity of	4	Insignificant
activity barriers	5	Insignificant
	6	Insignificant
	7	Insignificant
4 Radiation exposure of the	8	Insignificant
operating personnel	9	Insignificant
and the general public	10	Insignificant
	11	Insignificant
5 Operative management	12	**Fairly significant**
	13	**significant**
6 Plant security	14	Insignificant

	15	Insignificant

Table 5:
Evaluation of the safety indicators for a specific event (Example)

Due to the event, the components of the safety system were restricted in their proper function. However, the safety oriented functionality of the accumulator was not affected. In the case of a possible transfer of nitrogen into the reactor cooling circuit, the heat transfer from the reactor would have been affected, but nevertheless adequately guaranteed.

The event was caused by wrong action of the plant personnel and indicates some weaknesses in the organisational procedures for internal tests and maintenance. In addition, due to the fact that the same wrong settings were detected at several other redundant measuring devices, the event has to be classified as a common cause failure.

Thus, the event is considered to be significant with respect to safety.

Before the BStMLU permitted the restart of the plant, the operator was asked to carry out in the presence of TÜV a control of all safety relevant valves in the area of the reactor protection system, the electrical power supply and process engineering. In this connection, the correct setting of the equipment shutoff valves and the equalising valves was also checked.

According to the experience gained from the analysis of the event, various improvements were deduced for the organisational procedures (work permit procedure, issue of sealing pliers). In addition, the checking of safety relevant equipment shut-off valves (correct setting of the valves, check of leaktightness and seals) were introduced as a new inservice inspection which has to be performed in the presence of TÜV immediately before the reactor protection system is switched on.

The operators of the other Bavarian nuclear power plants were requested by the BStMLU to check the applicability of the event to their plant. Where necessary, these checks have led to additional improvements.

Conclusion

The integral safety assessment of reportable events in Bavarian nuclear power plants by the supervisory authority has proven its worth and is fully recognised by the nuclear power plant operators. By analysing the causes, the development and the effects of a particular event using safety indicators, a very high degree of objectivity and verifiability can be achieved. The BStMLU, together with the TÜV, is currently investigating whether the assessment process can be further standardised and therefore simplified by linking the assessment levels of the individual safety indicators using fuzzy logic.

Acknowledgements

The authors would like to thank Dipl.-Ing. R. Hero and Dipl.-Phys. H.-J. Rauh (both TÜV Süddeutschland) for stimulating discussions and the critical reading of the manuscript.

References

1. Stellungnahme der Internationalen Länderkommission Kerntechnik - ILK - zur Sicherheit der Kernenergienutzung in Deutschland vom 09.07.2000, Bayerischer Staatsanzeiger Nr. 34/2000, S. 5 (siehe auch www.bayerische-staatszeitung.de)

1. Verordnung über den kerntechnischen Sicherheitsbeauftragten und über die Meldung von Störfällen und sonstigen Ereignissen vom 14.10.1992 (BGBl. I S. 1766)

1. Kotthoff, K.: „Internationale Bewertungsskala für bedeutsame Ereignisse in kerntechnischen Einrichtungen, Benutzerhandbuch", GRS-111, Juni 1994, ISBN 3-923875-61-4

.

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

DEVELOPMENT AND USE OF SAFETY INDICATORS AT STUK

Petteri Tiippana

Radiation and Nuclear Safety Authority, STUK

P.O.Box 14, FIN-00881 Helsinki, Finland

Tel. +358 9 7598 8654

Fax. +358 9 7598 8382

E-mail: petteri.tiippana@stuk.fi

Abstract

This paper presents general outline of the use and development of YTO Indicator System at the department of Nuclear Reactor Regulation (YTO) in the Radiation and Nuclear Safety Authority, STUK. The purpose of the YTO Indicator System is to be a complementary tool in the nuclear safety regulation in addition to inspections and safety reviews. The development project of indicators and indicator system has shown that properly chosen and defined indicators provide an objective tool for the regulator to control nuclear safety and also to evaluate it's own activities and effectiveness. Development of indicator system has been a long project and has not been finished yet.

YTO Indicator System is divided into two main areas; safety of nuclear facilities and regulatory activities. Safety of nuclear facilities is divided into 3 areas based on the concept of defence in depth; safety and quality culture, operational events and physical barriers. Regulatory activities is also divided into 3 areas; working processes, resource management and regeneration and ability to work. These areas are measured using several indicators. At the moment some of these indicators are included in YTO's management system to measure whether internal goals are achieved or not.

General

The Radiation and Nuclear Safety Authority, STUK, is the nuclear regulatory body in Finland. STUK's regulatory activities comprise all safety review and all safety related inspections at the Finnish NPP's, as well as drafting of safety regulations and issuing of regulatory guides. To form an overview of the safety level at NPPs, and to have a measuring tool, the department of Nuclear Reactor Regulation (YTO) at STUK decided to start developing safety indicators.

Development project for safety related indicators

The project to develop a set of indicators for nuclear safety regulation was established in the middle of 1990. The project was included in the Finnish contribution to the research programme "Collection and classification of human reliability data for use in PSA" co-ordinated by the IAEA.

Goal of the project

The goal of the project was to develop an indicator system, which could be used:

- To illustrate levels and trends of nuclear safety in a quantitative manner
- To identify weaknesses at nuclear power plants
- To focus and optimise the use of YTO's resources
- To evaluate and develop YTO's review and inspection activities
- To develop co-operation between STUK and other organisations

Development project

Development of a set of safety related indicators has included several steps, such as:

- Determination of areas to be monitored
- Determination of existing data sources
- Nomination of candidate indicators for each interest area
- Data collection
- Data validation and test calculations
- Screening and updating of candidate indicators
- Development of an information system for indicators

Objectives and areas to be monitored were defined during the winter 1995 - 1996. Initial data collection, data analysis and test calculations were performed during summer 1996. A decision to adopt the indicator system as a managerial tool was made in 1997. Some new projects were also initiated in 1997 to develop additional indicators for certain areas. These are discussed later in the text. During 1998 and 1999 the system has changed; some new indicators has been included into the system and some has been dropped out based on the information obtained from test calculations.

Experience from the development project

Development of the indicator system has given experience and ideas. Some of them are listed below:

- Documents published by the IAEA and other international organisations have been a useful tool to get familiarised with the concept of an indicator system as well as to organise the development project.

- The limited number of data sources already in existence restricts the possibilities for determination of specific indicators. This should be noted at an early stage of the project. In practice, the areas to be monitored may be examined on a theoretical basis, whereas the specific indicators should not be nominated before getting familiarised with the data sources.

- The acceptability and usefulness of the indicator system within the regulatory body can be improved by asking for needs and opinions of the staff. Furthermore, participation of the staff in the data collection and analysis should improve the commitment throughout the organisation.

- Interpretation of the results (figures) should be carried out carefully. The focus of the analysis should lie on the trends and reasons for changes instead of numbers.

DESCRIPTION OF THE PRESENT INDICATOR SYSTEM

At the moment system consists of more than 20 indicator areas in which there are about 150 indicators. The large number of indicators can be explained by the fact that most of them are calculated separately for each plant unit. The number of indicator areas and indicators as well has been bigger during the development project. Based on the test calculations and data validation the most descriptive indicators has been chosen and included in the system. Also the structure of the system has changed during the development project. The next attempt is to try to combine balanced score card system and indicators describing regulatory activities.

At the moment 10 indicator areas are included in YTO's internal goals and these areas are followed annually. Some of these indicators are presented in chapter 4. In other areas follow-up has not been so systematic. Time scale of data gathered for indicators varies from 2 up to 10 years so there is enough information to make decisions whether the indicator is descriptive or not.

Different type of indicators

Indicators can be categorised in several ways. In most cases determination of the indicator value is based on the direct observation of the object. In that case the calculation of the actual value is often simple whereas the interpretation of trends can be complex. Examples of this kind of indicators are "number of scrams", "number of LERs" and "number of regulatory inspections". In some cases the correlation between an indicator and an object can be indirect. For example "overall number of equipment failures" and "respective volume of preventive maintenance" represent this type of indicators, object being maintenance resources. Some indicators are "predictive" whereas some are "corrective". "Risk-based" indicators and "safety and quality culture" indicators are evident representatives of predictive indicators while "collective doses" and "radioactive releases" describe an actual situation that may require corrective measures.

Nuclear safety and Regulatory activities

The indicator system developed at STUK is divided into two main areas: NPP safety, from regulator's point of view "outcome" of activities and regulatory activities, direct "output" indicator

Consideration of the area of "NPP safety" is based on the adoption of the concept of "Defence in Depth". The areas (or layers) under consideration are:

- Safety and quality culture

- Operational events
- Physical barriers

These indicators reflect mostly the achievements of the operating organisations, but STUK can also make some contribution on them.

Regulatory activities concentrate on the review and inspection activities, and on some supporting areas:

- Main working processes
- Resources
- Renewal of the regulatory activities and working abilities

Each of the areas mentioned above is going to be monitored using one or several indicators. More detailed description of the system and indicator areas are presented in appendix 1.

EXAMPLES OF RESULTS AND USE OF INDICATORS

Numerical values for the majority of all indicators have already been calculated for a period of several years. The figures (or trends) clearly demonstrate the usefulness of quantitative indicators, not only for the nuclear safety regulation but also for illustrating the quality of the regulator's work.

YTO has recently paid attention to improve the quality of its own activities within the nuclear safety regulation. For several areas of activities, a positive development can be observed by use of indicators. However, according to the figures continued inadequate performance is seen in some areas, and there is an obvious need for improvements.

As mentioned earlier a sub-set of indicators are also being used for setting goals and measuring YTO's own performance on annual basis. Examples of those goals and respective results are shown in the in the following chapter. These indicators are calculated and reported to the management at the end of each year. Changes in indicator values are analysed generally. Based on indicator values further activities have been started. For example the increase in the number human originated common cause failures and in the number of deviations from Technical Specification resulted in investigation that started at the beginning of year 2000.

Outcome of the regulatory work

YTO's internal goals are presented inside quotation marks before the figures.

"The number of technical and human originated CCFs does not increase remarkably."

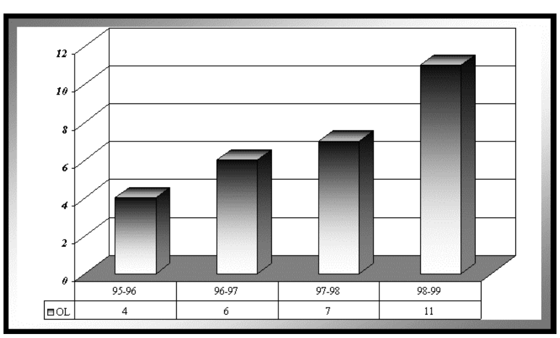


Figure 1. Annual numbers of human originated CCFs at a two-unit Olkiluoto site.

"Collective radiation doses do not exceed 5.78 man Sv / 4 NPP units. Annual doses for each individual do not exceed 20 mSv, considering the average value for 5 years period."

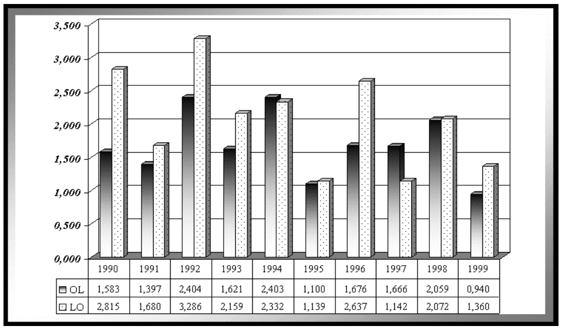


Figure 2. Annual collective doses recorded at each of the two-unit site. No cases of individual dose exceeding the limit have occurred.

"NPPs are operated in compliance with Tech. Specs."

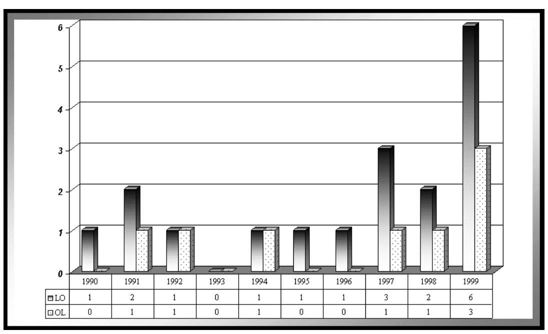


Figure 3. The indicator value is the number of recorded deviations.

"The dose of the most exposed person living near the NPP is below 0.005 mSv/year." Note that this is the performance goal. Limit in the license is 0.1 mSv/year.

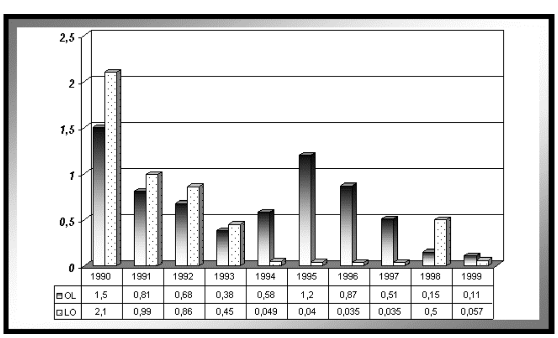


Figure 4. Annual doses (μSv/a) based on the releases from the NPPs, and calculations with a conservative dose model.

"Core damage risk contribution from actually occurred events does not exceed 5%."

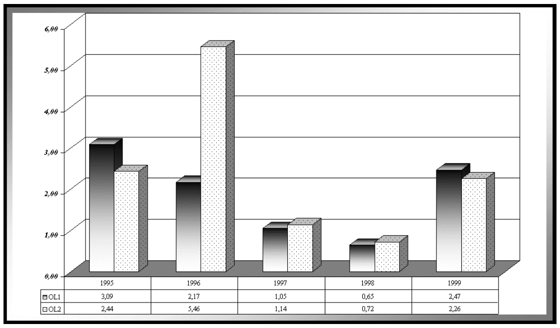


Figure 5. As-occurred risk contribution at Olkiluoto NPP, presented as a percentage of the average annual core damage risk, which is estimated in the PSA study (see more explanation later in the text).

"Fuel integrity, integrity of primary circuit and integrity of containment fulfil requirements, and no significant negative changes are seen."

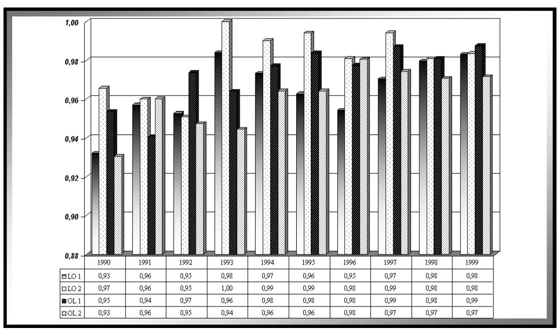


Figure 6. The proportion of isolation valves, which passed the first leakage test.

Output of the regulatory work

"All inspections included in STUK's annual inspection programme are performed and reported."

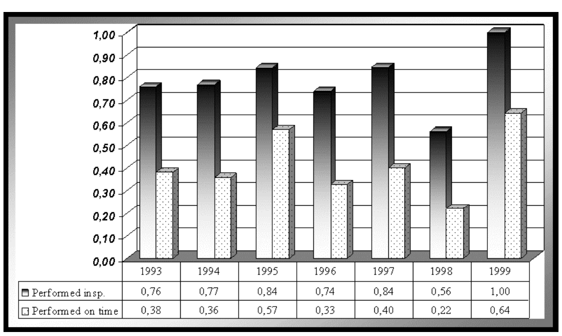


Figure 7. The proportion of planned inspections made in a given year and the proportion of inspections performed on time (Olkiluoto NPP)

"All regulatory reviews are conducted according to their priority, and if a longer review time is not specified in an exceptional case, decisions on reviewed items are made within 3 months."

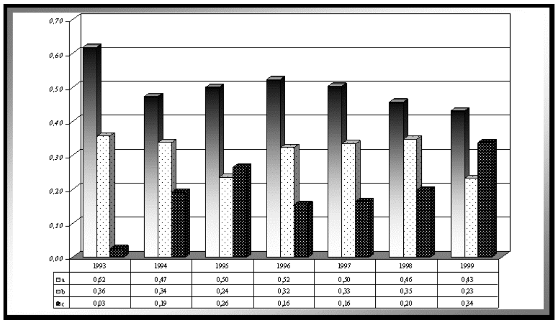


Figure 8. The proportion of different review times concerning Olkiluoto NPP: a < 1 month, 1 < b < 3, c > 3 months

"Regulatory guides are updated according to the annual plan."

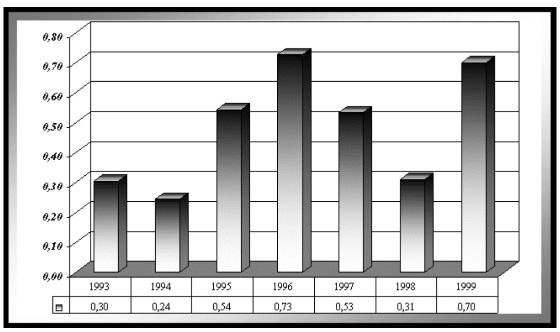


Figure 9. The proportion of updates, as compared with the plans.

"R&D funded by STUK support effectively regulatory activities. Research programs are reported by the contractor according to the contract. Research reports are analysed and commented during one month by STUK experts."

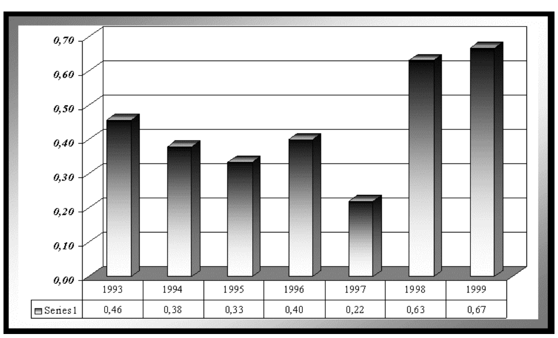


Figure 10. The proportion of comments issued on time.

Future activities

To finish this indicator project and to form a well functioning indicator system, following issues should be determined and decided:

1. Each indicator should have a responsible person ("owner") who calculates values and evaluates the indicator against the following targets:

- Practicality
- Relevance to NPP safety / regulatory activities
- Absence of negative impacts
- Accuracy

1. Requirements on the licensee reporting should be assessed against the information needed for calculation of indicators. At the moment, the information needed for some indicators is not regularly submitted to STUK.
2. Information and reporting system for the indicators should be developed. The goal is that indicators are available for all workers for example in the internal net.
3. Reporting methods and criteria for reporting should be established for each indicator. Also the reporting period should be decided. For example it has been considered that responsible persons should analyse the reasons for the changes in the indicators in the following two cases:

- The value of the indicator does not meet the given target.
- The value of the indicator deteriorates during two consecutive years even though it does meet the given target.

1. Development of an overall safety index based on all other indicators. For example, if the indicator value deteriorates it could get value –1 and if it gets better +1. The sum of all these values could be the safety index.
2. Collection of data for indicators that measure regulatory activities should be included into the main working processes (instructions/procedures), so that these processes produce either directly the indicators or the data needed for the calculation of these indicators.

Development of specific safety related indicators

PSA-based indicators

A few PSA-based indicators are set up in order to identify the safety significance and to follow up and monitor the risk development of specific events in NPP operation as follows:

1. Exemptions from the Technical Specifications

1. Failures of devices covered by the Technical Specifications
2. Preventive maintenance and other disconnection of devices covered by the Technical Specifications
3. Operating events

Each indicator is given as the annual sum of core damage frequency contributions from respective type of events, divided by the average annual core damage frequency from the PSA study. Each sum contains all respective events that reduce the reliability of some safety function, and thus cause a temporary risk increase above the basic risk level. Basic risk level prevails when no deviations from faultless plant condition are known to exist. One should recognise that the basic risk level already contains the risk contribution from majority of the aforementioned events that reduce the safety systems reliability. At a plant performing properly, a low indicator value demonstrates that the risk contributors which can be measured have a minor impact to the total risk. The majority of risk comes from infrequent significant initiators such as LOCAs, Loss of offsite power etc.The figure 5 of this paper presents a sum of all four indicators (exemptions from Tech. Specs., failures, maintenance, other events).

The associated plant configurations necessary for PSA based indicators are calculated using plant specific living PSA-programs. The last application was carried out as a follow-up to an earlier study and it covered years 1995-1998. The total risk of the aforementioned issues was found rather small, and it corresponds to the earlier studies. The first application in 1992 covered the operating history range 1985 through 1991 at the OL1. The second application was carried out in 1994 and covered the operating history range 1985 through 1994 at the OL2. In these studies, unavailabilities of tens of component and subsystem failures, and also few transients were analysed in the Living PSA framework.

While developing the risk based indicators, we are aware of the limitations of PSA such as completeness problem, modelling uncertainty, shortages in human error analysis and CCF analysis etc., which result in uncertainty into the PSA figures. These uncertainties however are found rather insignificant as concerns the use of indicators.

The main problem within the PSA based indicators is that some issues are difficult or even impossible to model with the current PSA-model. Hence it is required that a sophisticated Living PSA system including extensive and detailed system models, with a well established data collection and processing system to provide plant specific data, and an efficient, user friendly PSA code are available. If these conditions are met, the determination of PSA based indicators is quite straightforward.

Among the individual indicators, the risk importance of exemptions from the Technical Specifications and of preventive maintenance are the most straightforward ones. Most of the deviations in the process are modelled with PSA-programs. Indicators describing the risk importance of failures are also applicable in most cases but all devices are not modelled in detail. Indicators describing operating events often need further development of the PSA model.

Indicators based on plant specific fault data statistic

Useful indicators can also be extracted from fault data records, such as indicators for the common cause failures and the quality of maintenance. The idea is to examine the usability of fault data records in calculation and screening of different types of failures.

The analysis of common cause failures was based on a method jointly developed by STUK and VTT (STUK's main contractor for nuclear safety research). These indicators have been developed and defined only for Olkiluoto nuclear power plant. The indicators are simply the numbers of different failure types. The screening of the plant specific fault data covered years 1995-1996 (about 2800 cases). Failures at Loviisa NPP are currently being analysed in a new project contracted to VTT.

Common cause failures were divided to two categories - to human or technical failures. These were further divided to critical or non-critical failure classes according to their influence on system or devices. In the screening of failures, also individual human errors and multiple technical failures were identified.

The results of this part of the study showed clearly that hidden CCFs can be found from detailed examination of fault data history. The number of occurred human originated CCF’s (2,3/y) corresponded well to earlier studies done at STUK and VTT.

It was concluded that there are many good ways to utilise the information in failure records as indicators. Based on the study carried out for Olkiluoto nuclear power plant, the indicators from failure statistics are already applicable for safety assessment. The monitoring of before mentioned indicators continues, and the aim is to focus on safety related systems.

Safety Culture indicators

Evaluation of the current level of safety culture by quantitative means is a complex task. Instead of direct measurement of safety culture, the evaluation could be carried out by identification of features of safety culture and measuring their values. A project to develop such a methodology was started at the beginning of 1999 in co-operation with VTT.

As another project, it was decided to study the NPP staff opinions on quality and outcome of work carried out by the regulatory body. These opinions or attitudes depend on individuals' experience and also on the overall atmosphere within the NPP organisation. Based on the assumption that the perception by the NPP staff corresponds to the real situation, an attempt was made to find out factors causing a certain attitude. Furthermore, it would be interesting to clarify what are the assumptions of the regulatory staff about the same factors and finally make a comparison between those two. This project was initiated in 1997 by interviewing NPP staff. Quantitative indicators are so far not developed.

YTO INDICATORS

General

The YTO Indicator System developed by the Nuclear Reactor Regulation (YTO) illustrates the safety of the nuclear power plants and performance of the regulatory control. The system is composed of more than 20 indicator areas which spread out further to about 150 separate indicators.

This paper presents a proposal for the YTO Indicator System to be taken up from the beginning of

2000 and practical actions for maintaining the system. The paper describes

- structure and contents of the YTO Indicator System
- responsibilities and methods for determining indicators.

A report concerning a separate development project considers more thoroughly how to bring the strategy plan closer to the practice. The project in question touches the YTO Indicator System and therefore the results of the project may in due cause affect needs to update the YTO Indicator System, too.

Structure and contents of the YTO Indicator System

The YTO Indicator System has been, first of all, intended for information system which different functional sectors within YTO can utilise when willing to do it. Based on the former consideration the YTO Indicator System is applicable for assessing success of the strategy plan as well as for focusing an review and inspection programme.

The YTO Indicator System is divided in two principal groups that are the safety of a nuclear facility and the regulatory activities. Indicators for describing the safety of nuclear facilities can also be utilised to assess effectiveness of STUK. Other relevant arguments like, among other things, achievements in societal impact and in international co-operation shall be additionally noted in assessing the effectiveness. Indicators concerning regulatory activities are applicable for assessing work processes, resource management and personnel viewpoints.

The principal groups A, "Safety of nuclear facilities" and B, "Regulatory activities" of the YTO Indicator System are divided into sub-groups and further into indicator areas as follows:

A. Safety of nuclear facilities	**B. Regulatory activities**
A1 Safety and quality culture A1.1 Failures and their repairs A1.2 Number of TTKE deviations A1.3 Availability of safety systems A1.4 Radiation doses A1.5 Radioactive releases A1.6 Documentation	- **B1 Working processes** B1.1 Fulfilment of outcome targets B1.2 Timely decision making B1.3 Maintenance of regulations B1.4 Implementation of inspection program B1.5 Steering of contracted safety research B1.6 Actions in abnormal situations
A2 Operational events A2.1 Number of events A2.2 Significance of events	B2 Resource management B2.1 Resources for regulatory control of nuclear safety B2.2 Distribution of work load

A2.3 Causes of events A2.4 Number of fire alarms	
<u>A3 Structural integrity</u> A3.1 Integrity of nuclear fuel A3.2 Integrity of primary circuit A3.3 Integrity of containment	<u>B3 Regeneration and ability to work</u> B3.1 Maintenance of YTV Quality Manual B3.2 Execution of development projects B3.3 Execution of training program B3.4 Work satisfaction B3.5 Compliance with values

The review period of the indicators relating to the safety of nuclear facilities will be mostly the operating cycle from the beginning of the refuelling outage to the beginning of the next refuelling outage. The indicators relating to the regulatory operations are determined every calendar year.

Definitions of indicators and responsibilities for calculating indicators

1. A Safety of nuclear facilities

1. A1 Safety and quality culture

1. A1.1 Failures and their repairs

Scope of the indicator area:

The following data relating to failures and their repairs are followed under this indicator area:

- failures of TTKE equipment (number of failures that caused unavailability); TTKE is the Technical Specifications
- preventive maintenance of TTKE equipment (a number of preventive maintenance works in relation to a total number of failure repairs and preventive maintenance works)
- repair of TTKE equipment (a real repair time in relation to a repair time allowed in the TTKE)
- failure types (human single failures, human common cause failures, technical common cause failures, multiple failures).

Purpose of indicators:

The indicators are used to follow number and type of failures, elimination of failures by the means of preventive maintenance and attitudes towards repair times allowed in the TTKE.

Source of data:

Data for indicators is collected from the daily reports submitted by the utilities, from failure data bases and from maintenance reports whose files are up to now unofficially submitted to STUK. Failure data related to operating cycles is available from operating cycle and monthly reports of the Maintenance Unit.

Responsible unit:

Operational safety (KÄY)

1. *A1.2 Number of TTKE deviations*

Scope of the indicator area:

A number of plant occasions against the Technical Specifications (TTKE) as well as a number of exemption orders granted by STUK are followed within this indicator area.

Purpose of indicators:

The indicators track and describe a number of plant occasions against the Technical Specifications as well as a number of exemption orders granted by STUK.

Source of data:

Data for the indicators are collected from the event reports issued by utilities and from applications for the exemption orders.

Responsible unit:

Operational safety (KÄY)

2. *A1.3 Availability of safety systems*

Scope of the indicator area:

The plant unit -specific WANO indicators are used as the indicators. At the Olkiluoto NPP the items of follow-up are the systems 321, 327 and diesel generators and at the Loviisa NPP, respectively, the high pressure safety injection system (HPSI), auxiliary feed water system (AFW) and diesel generators. The review was started in 1990.

Purpose of indicators:

The indicator illustrates an unavailability of the safety systems. By the means of the indicator it is possible to supervise condition of safety systems and changes of condition.

Source of data:

Data for the indicators are collected from the utilities. It is not delivered officially to STUK.

Responsible unit:

Operational safety (KÄY)

3. A1.4 Radiation doses

Scope of the indicator area:

The following indicators are tracked within this indicator area

- an amount of collective radiation doses
- an average of ten highest amount of annual radiation doses

Tracking is utility-specific and was initiated in 1990.

Purpose of indicators:

The purpose of the indicators is to supervise and to chart radiation doses. The average of ten highest annual personal doses illustrates how close the allowed limit of 20 mSv there is.

Source of data:

Data for indicators is collected from the utilities. Data for cumulative radiation doses is received from the annual reports but the average value of the ten highest annual personal doses are not officially submitted to STUK.

Responsible unit:

Radiation safety (SÄT)

1. A1.5 Radioactive releases

Scope of the indicator area:

The radioactive effluents and emissions (TBq) of the plants are tracked within this indicator area. Tracking is utility-specific.

Purpose of indicators:

The purpose of indicators is to supervise the amount and trend of radioactive releases.

Source of data:

Data for indicators is collected from monthly and annual reports submitted by utilities.

Responsible unit:

Radiation safety (SÄT)

2. A1.6 Documentation

Scope of the indicator area:

A number of modifications whose plant documentation was not annually updated by the next refuelling outage.

Purpose of indicators:

The purpose of indicators is to supervise quality management of the utilities and their ability to maintain plant documentation.

Source of data:

Data for calculating indicators is collected from the plant modification register and from on site inspections conducted by STUK.

Responsible unit/person:

Operational safety (KÄY)/Co-ordinator for plant modifications.

1. A2 Operational events

1. A2.1 Number of operational events

Scope of the indicator area:

A number of operational events reported in accordance with the Guide YVL 1.5 is tracked within this indicator area. Tracking is plant unit –specific and it was initiated in 1990. An objective is to move to tracking in accordance with operating cycles.

Purpose of indicators:

The purpose of indicators is to track a number of operational events. It illustrates a number of safety significant events.

Source of data:

Data for indicators is collected from the data base Nuclear Safety Register (YTR) on the basis of reported operational events.

Responsible unit:

Operational safety (KÄY).

2. A2.2 *Significance of operational events*

Scope of the indicator area:

A calculated risk significance based on probabilistic safety assessment (PSA) of operational events is followed as the indicators. The indicator is the summed risk of each follow-up area. Tracking is plant unit –specific. The follow-up areas to be tracked are as follows:

a. applications for TTKE exemptions

b. failures of TTKE equipment

c. preventive maintenance and other planed separations of TTKE equipment

d. operational events according to the Guide YVL 1.5.

Purpose of indicators:

The purpose of indicators is to track a risk significance of selected operational events and parallel to it to monitor lengths of planned separations and preventive maintenance actions.

Source of data:

Data for indicators is collected from the utility reports and applications for TTKE exemptions.

Responsible unit:

Operational safety (KÄY).

1. A2.3 Causes of operational events

Scope of the indicator area:

Based on their origin the direct causes of operational events are roughly divided to technical and human failures. The indicator is defined by calculating the mutual proportions of the aforementioned cause types within the reported operational events. Basically, the direct cause in assessment is assumed to be either technical or human. When needed, an influence of the aforementioned factors can be assessed in percentages.

Purpose of indicators:

The purpose of indicators is to track changes in mutual proportions of operational events that cut across the reporting limit.

Source of data:

Data for indicators is determined on the basis of the operational events that are reported according to the Guide YVL 1.5.

Responsible unit:

Operational safety (KÄY).

2. A2.4 Number of fire alarms

Scope of the indicator area:

A number of fire alarms is tracked within this indicator area. Tracking is utility–specific. Based on their origin the fire alarms are divided as follows:

a. automated failures

b. actual automated alarms

c. actual fires

d. other alarm operations.

Purpose of indicators:

The purpose of indicators is to supervise and to track operations of fire alarm systems and fire brigades.

Source of data:

Data for indicators is collected from the utilities. Olkiluoto NPP reports the data in its annual report but Loviisa NPP does not do it and therefore no official reporting does not exist from Loviisa NPP.

Responsible unit:

Risk assessment (RIS).

1. A3 Structural integrity

1. *A3.1 Integrity of nuclear fuel*

Scope of the indicator area:

A maximum activity of the primary circuit equivalent to I-131 (Loviisa NPP, at the Olkiluoto NPP solely I-131) (kBq/m^3) during the operating cycle. Concerning the Loviisa NPP, I-131 and I-134 are included in calculations.

Purpose of indicators:

The purpose of the indicator is to describe the integrity of the nuclear fuel during operating cycle.

Source of data:

Data for indicators is collected from the monthly and annual reports issued by the utilities.

Responsible unit:

Operational safety (KÄY).

2. *A3.2 Integrity of the primary circuit*

Scope of the indicator area:

An overall amount of identified and non-identified leakage of the primary circuit is tracked within this indicator area.

Purpose of indicators:

The purpose of indicators is to track amount of identified and non-identified leakage that describe the integrity of the systems inside the plant containment.

Source of data:

Data for indicators is collected from the utilities. The data is not submitted officially.

Responsible unit:

Operational safety (KÄY).

3. A3.3 Integrity of the containment

Scope of the indicator area:

The following matters are tracked within this indicator area:

a. overall leakage of isolation valves compared with the highest allowed overall leakage of the isolation valves

b. percentage of isolation valves at each plant unit that passed the leakage test at the first attempt

c. an overall leakage of containment's entrance and other holes in relation to the highest allowed overall leakage of these holes at each plant unit.

Purpose of indicators:

The purpose of indicators is to track tightness of isolation valves, penetrations and entrance holes.

Source of data:

Data for indicators is submitted officially to STUK.

Responsible unit:

Reactor and systems engineering (REA).

1. B Regulatory control

1. B1 Working processes

1. B1.1 Fulfilment of outcome targets

Scope of the indicator area:

Fulfilment of department's outcome targets on a scale from 0 to 1 is used as the indicator. The target-specific evaluation is made and the indicator is the average of these numerical values.

Purpose of indicators:

The purpose of the indicator is to track the fulfilment of outcome targets. The indicator makes it also possible to evaluate action planning from the "challenges/realistic objectives" point of view.

Source of data:

In connection with the superior/subordinate discussions of department

Responsible unit:

Management.

2. B1.2 *Timely decision making*

Scope of the indicator area:

In this indicator area there are three sub-items. The indicators are developed in accordance with the new YTV guides. The indicators are tracked by the duration of decisions prepared in STUK by following classification:

a. A portion of decisions made by STUK within one month from all decisions.

b. A portion of decisions made by STUK within a period from one month to three months from all decisions.

c. A portion of decisions made by STUK in the time more than three months from all decisions.

Purpose of indicators:

The purpose of the indicator is to track the duration of YTO's document handling.

Source of data:

The information for the indicators is collected from database YTR. The annually sent decision letters are used as reference term of the search.

Responsible unit:

Management.

1. B1.3 Maintenance of regulations

Scope of the indicator area:

The indicator is calculated according to the Guide YTV 3.1 and compared with the Annual Action Plan. The indicator prefers to evaluate the amount of work instead of the final completion.

Distribution of work amount (formation of added value) in different draft and preparation stages:

Assessment of updating need; drawing up a guide preparation plan	+5% ⇨ 5%
Draft 1 drawn up by the Task Force	+45% ⇨ 50%
Draft 2 drawn up by the Task Force (internal comments taken into account)	+20% ⇨ 70%
Draft 3 drawn up by the Task Force (external comments taken into account)	+10% ⇨ 80%
Approval by the departmental meeting, document technical and legal review conducted	+5% ⇨ 85%
Draft 4 drawn up by the Task Force (request for comments sent to the Advisory Committee)	+5% ⇨ 90%
Comments of the Advisory Committee taken into account	+5% ⇨ 95%
Approval of the STUK Management received (JL, AN, LR/TVa), Guide completed	+5% ⇨ 100%

Performance indicator is calculated in the following manner by using these percentages as weights (it has been assumed that the Annual Action Plan - just as it does at the present - includes targets for the assessment of updating needs, for Guides that are to be drawn up to stage "draft 1" and for Guides that are about to be completed):

$$\mathbf{S = \{\Sigma\ Q_i/q \times 5 + \Sigma\ P_i/p \times 50 + \Sigma\ K_i/k \times 100\} / n}$$

where (weights as presented above),

Q_i = 5 x number of Guides, whose

updating need has been assessed

q = number of Guides, whose updating need has to be assessed (target)

P_i = number of Guides x weight (5 or 50), from which a plan or draft 1 as been drawn up

p = number of Guides, from which draft has to be drawn up (target)

K_i = number of Guides x weight (5, 50,…,100) in accordance with the preparation stage

k = 100 x number of Guides, which has to be completed (target)

n = number of different objective levels, here 3.

Purpose of indicators:

The purpose of the indicator is to track the effectiveness and follow-up of schedules as well as commitment to the guide updating within YTO.

Source of data:

The necessary information to calculate the indicator is gathered from the annual plan of the YVL guides and from preparation status of individual guides.

Responsible unit:

Co-ordinator for Rule making.

2. *B1.4 Execution of the Periodical Inspection Program*

Scope of the indicator area:

Indicators are divided in four sub-items. The indicators cover the following three utility-specific items:

a. A proportion of annually performed inspections from all inspections of the Periodical Inspection Program

b. A proportion of inspections performed during planned inspection month from all inspections of the Periodical Inspection Program

c. A proportion of performed inspections from all inspections of the Periodical Inspection Program where the inspection memorandum has been prepared within one month according to the Guide YTV 4.1.

Purpose of indicators:

The purpose of the indicator is to track and supervise the execution of the annual plan of the Periodical Inspection Programme.

Source of data:

The data for indicators is collected from the inspection protocols and memorandums of the Periodical Inspection Programme. A systematic approach of collecting information has to be developed in connection with commissioning the new Periodical Inspection Programme.

Responsible unit/person:

Operational safety (KÄY) / KTO Co-ordinator

1. *B1.5 Steering of contracted safety research*

Scope of the indicator area:

A proportion of research memorandums prepared within one month from all prepared research memorandums required by the Guide YTV 8.1 is used as the indicator. Other indicators representing contracted safety research are being developed.

Purpose of indicators:

A purpose of the indicator is to track the fulfilment of the requirements specified in the YTV Quality Manual.

Source of data:

The data for the indicator is collected from the database TTR concerning contracted safety research.

Responsible person:

Co-ordinator of contracted safety research.

2. *B1.6 Actions in abnormal operating*

events

Scope of the indicator area:

Fulfilment of obligations in compliance with the YTV 4.6 is used as the indicator concerning among other things the following YTO operations:

- communications with the utility
- immediate reporting to the department's management and in needed extent to the rest of the department
- initiation of the emergency response if necessary
- reporting to TYK and drafting of a press release

To define the indicator, the YTO's operations are assessed on a scale from 0 to 1 and the average of these values is calculated. The results of communication experiments and indicators describing participation in emergency preparedness exercises are being developed.

Purpose of indicators:

The purpose of the indicator is to track execution of YTO's actions in abnormal operating events.

Source of data:

The data for indicators is collected from event memorandums prepared in STUK and from minutes of the departmental meetings. The data (success) that is needed to calculate the indicator, is evaluated right after the event. The indicator is formed on the basis of self assessment.

Responsible unit:

Operational safety (KÄY)

1. B2 Resource management

1. B2.1 Resources for regulatory control of nuclear safety

Scope of the indicator area:

Distribution of work hours at different action areas within YTO is followed as the indicators:

a. regulatory control

b. administration

c. contracted services

d. maintenance and development of professional knowledge and skills

e. other obligations

Purpose of indicators:

The purpose of the indicators is to track an allocation of human resources within YTO. By the means of the indicator, a focus of YTO operations can be followed and directed.

Source of data:

The data for indicators is collected from STUK's working hour reports.

Responsible unit:

Management.

1. B2.2 Distribution of work load

Scope of the indicator area:

The indicator is composed of two factors:

a. the proportion of overtime hours and cut working hours of the follow-up balance from the total working hours

b. the proportion of the staff members whose working hours have been cut more than twice a year from the entire personnel.

Purpose of indicators:

The purpose of indicators is to track sufficiency of normal working hours and distribution of work load among personnel.

Source of data:

The data for indicators are collected from working hour balances.

Responsible unit:

Management.

1. B3 Regeneration and ability to work

1. B3.1 Maintenance of YTV Quality Manual

Scope of the indicator area:

The amount of work used for maintaining the YTV Quality Manual is followed by the means of a indicator, which compares the amount of performed work to the total amount of work assumed by the Annual Action Plan of the department. In order to assess the amount of performed work, the completion rates and ratings of the guides have been chosen as follows:

- preparing a new draft guide to a stage, where it can be sent to an internal review round ⇨ completion rate 50% ⇨ rating index 50
- completion of a new guide from the aforementioned draft stage ⇨ completion rate 100% ⇨ rating index 50
- small-scale review of an old guide and its completion ⇨ completion rate 100% ⇨ rating index 50
- extensive review of an old guide and preparing it to a draft stage, where it can be sent to an internal review round ⇨ completion rate 50% ⇨ rating index 50
- extensive review of an old Guide and its completion from the aforementioned draft stage ⇨ completion rate 100% ⇨ rating index 50
- translation of a YTV-Guide into English by using department's own human resources ⇨ completion rate 100% ⇨ rating index 50.

The indicator describing the maintenance of the YTV Quality Manual can be calculated by dividing the cumulative rating index at a moment of review by the cumulative rating index calculated according to the Annual Action Plan.

Purpose of indicators:

The purpose of the indicator is to track efficiency and compliance with the targets for updating internal guides within YTO.

Source of data:

The data for indicators are collected from the Annual Action Plan of the YTV guides and from the follow-up of their execution.

Responsible person:

Management

1. *B3.2 Implementation of development projects*

Scope of the indicator area:

A proportion of the completed/progressed development projects from all development projects specified in the Annual Action Plan is used as indicator. The indicator is formed on the basis of self assessment.

Purpose of indicators:

The purpose of the indicator is to track execution of development operations within YTO.

Source of data:

The data for indicators are collected from the Annual Action Plan and from the Annual Outcome Report of YTO.

Responsible unit:

Management

2. B3.3 Implementation of training program

Scope of the indicator area:

Implementation of the training programme attached in the Annual Action Plan is used as indicator. The indicator is formed on the basis of self assessment on a scale from 0 to 1.

Purpose of indicators:

The purpose of indicator is to track implementation of training operations.

Source of data:

The data for indicators is collected from the Annual Action Plan and from the Annual Outcome Report of YTO.

Responsible person:

Training manager.

3. B3.4 Work satisfaction

Scope of the indicator area:

Results of enquiry of work satisfaction barometer performed in the department are followed as the indicator. Four elements are followed as indicators on the department level:

a. contents of work

b. leadership

c. operability of work community

d. possibilities to develop personal knowledge and skills

Purpose of indicators:

The purpose of the indicator is to track a progress of work satisfaction.

Source of data:

The data for indicators is collected from work satisfaction enquiries. Determination of the indicator in connection with outcome discussions requires, that enquiry is scheduled to right after turn of the year.

Responsible unit:

Human and organisational factors (INH)

1. B3.5 Implementation of values

Scope of the indicator area:

Development of department's organisational culture is used as the indicator. Assessment is based on enquiry, in which the personnel is asked to evaluate, on a scale from 0 to 1, implementation of seven values defined by YTO for the practical work. Result is presented separately for each value as the average of department.

Purpose of indicators:

The purpose of indicator is to track the progress in department's values.

Source of data:

Enquiry is carried out in connection with work satisfaction enquiry.

Responsible unit:

Human and organisational factors (INH)

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

Experience in the use of performance indicators in korea

Sae Yul Lee
Korea Institute of Nuclear Safety
Regulation Analysis Department
P.O. Box. 114, Yu-Sung
Taejon, Republic of Korea
Tel: +82-42-868-0157
Fax: +82-42-863-3381
e-mail : K272LSY@KINS.RE.KR

Koo Hyun Bae
Korea Institute of Nuclear Safety
Regulation Analysis Department
P.O. Box. 114, Yu-Sung
Taejon, Republic of Korea
Tel: +82-42-868-0210
Fax: +82-42-863-3381
e-mail : K217BKH@KINS.RE.KR

Abstract

PIs for Korean nuclear power plants were developed through a government funded project in 1997 by KINS with the cooperation of the Korea Electric Power Corporation (KEPCO). The PIs, currently used after one year trial application and modification, are composed of 8 indicators for PWR plants. The 10 years trend graphs of each indicator from 1990 to 1999 are presented and trend analysis of PIs in three different ways including average values of all PWRs, of different reactor capacity groups and of different reactor supplier groups are shown.

introduction

Many operating organizations, regulatory bodies, and international organizations have developed and used performance indicators for the quantitative assessment of NPP operation. The performance indicators can be used to monitor and to gain perspective on performance and progress of a nuclear power plant. The PIs also provide an indication of the possible need to adjust priorities and resources to achieve improved performance.

The importance and usefulness of performance indicators were recognized also in Korea and PIs for Korean nuclear power plants were developed through a government funded project in 1997 by KINS with the cooperation of the Korea Electric Power Corporation (KEPCO). The PIs, currently used after one year trial application and modification, are composed of 8 indicators for PWR plants. Separate set of PIs for

CANDU reactors are under development because CANDU reactors have different characteristics with PWRs and relatively less operating experience than PWRs in Korea. The project for development of CANDU PIs will be completed in the end of 2000.

This paper contains brief definition of each indicator with its background, 10 years trend graphs of each indicator from 1990 to 1999 and the analysis of the trend graphs. The trends of PIs are illustrated in three different ways including average values of all PWRs, of different reactor capacity groups and of different reactor supplier groups. Three reactor capacity groups are 600 MWe group (2 units), 900 MWe group (6 units) and 1000 MWe group (3 units). Westinghouse (6 units), Combustion Engineering (3 units) and Framatome (2 units) are three reactor suppliers.

Definition of PIs

1. Unit Capability Factor (UCF)

This factor reflects how efficiently the plant has been operated and maintained. It is defined as the ratio of actual electricity generated to the plant's design capacity for a given period, expressed as a percentage.

 Formula

actual electricity generation

(MWe- hour/year)

UCF = ------------------------------------ x 100 %

plant design capacity(MWe) x time (hour/year)

* If the UCF is greater than 100%, the UCF is considered as 100%

 Background

Since the unit capability factor is an important indicator for assessing the effective operation and maintenance, this factor was selected as a specific indicator to evaluate the stability of a plant. There can be a little difference between the design capacity and the maximum capacity which may make UCF greater than 100%. To make this indicator more reasonable, the plant design capacity will be replaced with the maximum generation capacity in the near future.

2. Unplanned Outage Rate (UOR)

This indicator is defined as the ratio of the reactor subcritical time caused by unplanned occurrences to the summation of reactor critical time and subcritical time excluding planned outage.

 Formula

unplanned outage time

UOR = ---------------------------------- x 100 %

(reactor operating time) + (unplanned outage time)

* Planned outage period for scheduled refueling and maintenance is not included.

 Background

Reactor criticality is used as a criterion to determine the unplanned outage rate because the incident reporting criteria of Korea defines reactor outage based on reactor criticality rather than the connection to grid.

3. Unplanned Scrams for Critical Period (USCP)

This indicator is defined as the number of unplanned automatic scrams (reactor protection system actuations) during critical period.

□ **Background**

Manual scram for maintenance and scram that is planned to occur as a part of test (e.g., a reactor protection system actuation test) are not included.

4. Safety System Actuation (SSA)

This indicator is defined as the summation of the number of safety system actuation and the number of automatic actuation of emergency diesel generator.

□ **Background**

Both manual and automatic actuation of safety system are included. Planned actuation for a test is not included and multiple actuations due to a single signal are considered as one actuation.

5. Primary System Boundary Integrity (PSBI)

This indicator is defined as the average unidentified leak rate from the Reactor Coolant System as defined in the Technical Specifications of the plant. Only the average value of the leak rate during full power operation is considered. The unit is m3/hr.

□ **Background**

The trend of the unidentified leak rate is considered to represent indirectly the integrity of major equipments and the piping in primary system pressure boundary.

6. Fuel Reliability (FR)

This indicator is to monitor plant progress in achieving and maintaining high fuel integrity, and to foster a healthy respect for preservation of fuel integrity.

□ **Formula**

$$FR = \square(A131)N - (k) \times (A134)N\square x\ \square(Ln/LHGR) \times (100/Po)\square^{1.5}$$

$(\lambda 131 + Ba)$

$(A131)N = (A131)actual\ x$ -------------------

$(\lambda 131 + Bn)$

$(\lambda 134 + Ba)$

$(A134)N = (A134)actual\ x$---------------------

$(\lambda 134 + Bn)$

where,

$(A131)N$: I-131 activity (Bq/g or μCi/g)

$(A134)N$: I-134 activity (Bq/g or μCi/g)

k : the tramp uranium correction

coefficient (a constant with a value of

0.0318)

Ln : the linear heat generation rate (kw/m)

LHGR : the average linear heat generation rate (kw/m)

Po : the average reactor power (%)

Ba : the reactor coolant purification rate (second-1)

Bn : a common purification rate constant (second-1)

$\lambda 131$: the decay constant of the I-131 (second-1)

$\lambda 134$: the decay constant of the I-134 (second-1)

- **Background**

WANO's calculation formula is selected and the calculation is based on the radioactivity of I-131 in the reactor coolant.

7. Radiation Collective Dose (RCD)

This indicator is to monitor the efforts to minimize total radiation dose at each facility and to measure the effectiveness of the radiation protection program which minimizes radiation dose to plant personnel. The unit is man-rem or man-Sv.

- **Formula**

$$RCD = \frac{\text{total plant radiation collective dose}}{\text{number of units in plant}}$$

- **Background**

This indicator reflects the total external whole-body dose received by all site personnels including contractors during a given period. For multi-unit stations that do not track radiation collective dose separately for each unit, unit values are estimated by dividing the station data by the number of operating units at the station.

8. Low-Level Solid Radioactive Waste (LSRW)

The volume of solid radioactive waste indicator is to monitor the progress toward reduction of the volume of waste generated. The unit is the number of 200 L (55 gal) drums generated a year.

□ **Background**

The amount of solid radioactive waste is counted by the number of drums managed by the licensee. For multi-unit station that does not track solid radioactive waste separately for each unit, unit values are estimated as dividing the station data by the number of operating units at the station. This allows more meaningful comparisons among single and multi-unit stations. The volume of radioactive liquids and gaseous effluents are not included.

Review of Performance Indicators for Last 10 Years

The trends of all 8 developed performance indicators using the performance data of operating nuclear power plants from 1990 to 1999 are shown in chapter □. The graphs are in three categories. In the first category (Fig. 1) each graph shows 10 year trend of performance indicators averaged over all operating PWR plants. In the second (Fig. 2) and third (Fig. 3) categories, the performance indicators are averaged over three reactor capacity groups and three reactor supplier groups respectively. The trend of each PI and the unusual values in the PI graphs are explained in this chapter.

1. Unit Capability Factor

The unit capability factor of Korean NPPs continuously increased from the year of 1988, and has maintained high levels, over 80%, since 1991. It shows that all of the operating nuclear power plants have been operated at stable states. We can notice that the stable trend was maintained even after 1995 when the CE type NPP, Yonggwang unit 3, started commercial operation. (Fig. 1)

The unit capability factor for the CE type reactors maintained high levels in 1995, but then dropped sharply below average in 1996, because of the decrease in the unit capability factor of the Yonggwang unit 4 which started commercial operation in 1996 and experienced more unstable condition at the initial stage of commercial operation than Yonggwang unit 3. (Fig. 2)

A noticeable result in the unit capability factor analysis is that the unit capability factor of Framatome type reactors are higher than those of Westinghouse type reactors, as shown in the supplier group averaged graph (Fig. 3). The overall increasing trend of the unit capability factor is evaluated as the result of licensee's efforts to improve performance in operation and maintenance.

2. Unplanned Outage Rate

The increase of UOR in 1990 was caused by maintenance job to repair the damaged low pressure turbine at Ulchin unit 2 from November 1989 to the end of January 1990.

Another increase in the unplanned outage rate in 1994 resulted from the long duration required for maintenance and repair of the steam generator tube leak at Kori unit 1 from November 8, 1994 to January 2, 1995.

It is certain that the major contributors to the increase in the unplanned outage rate are the combined problems of the main generator and turbine which are parts of the secondary system. These problems happened in both Ulchin units 1 and 2 which were in the initial stage of commercial operation in 1990 and in Kori unit 1 which has relatively longer operating years.

Increase of UOR in 1999 was caused by a series of reactor shutdowns (4 times) at Yonggwang unit 2 in March.

3. Unplanned Scrams for Critical Period

The average shutdown rate of more than 3.0/reactor year in 1990, was a direct result of the aging effects of the instrument and control system in Kori unit 1. Four shutdowns of Kori unit 4 in 1992 and three shutdowns of Kori unit 3 in 1993 were the major contributors to the increase of shutdown rate.

In 1997, Framatome type reactors, Ulchin units 1 and 2, showed high shutdown rates because of reactor shutdowns by natural phenomena such as falling down of a transmission tower due to heavy snow and strong storm, inflowing of swarms of shrimp into intake, and so on.

Four successive reactor shutdowns at Yonggwang unit 2 contributed the increase of shutdown rate in 1999.

However, overall trend of shutdown rate was evaluated decreasing because of licensee's continuous efforts to improve the performance of equipments and quality of operators.

4. Safety System Actuation

The highest average of safety system actuations of all NPPs was recorded in 1990, due to one actuation of safety system and three abnormal starts of the emergency diesel generators at Ulchin units 1 and 2. The actuation of the safety injection signal was initiated by a malfunction of the MSIV, and the starts of the emergency diesel generators were caused by the failure of a disconnect switch on the 345kV Bus.

In 1996, emergency diesel generator started two times at Yonggwang unit 1 and one ESF acuation signal was generated by carelessness of test personnel at Yonggwang unit 3.

There were one ESF actuation signal and one abnormal diesel generator start at Yonggwang unit 2 in 1999.

5. Primary System Boundary Integrity

The unidentified leakage rate through the primary system boundary indicates a decreasing trend. That is evaluated to be the result of the licensee's maintenance efforts to prevent potential leakage. The decrease of the leak rate in CE type reactors, Yonggwang units 3 and 4, indicates that the overall systems have become more stable since they began commercial operation.

6. Fuel Reliability

The Iodine concentration peaks in 1992 and 1993, which represent low fuel reliability, were the results of fuel damages in Kori unit 2. After the replacement of the damaged fuels, the overall fuel reliability returned to normal levels.

7. Radiation Collective Dose

The trend of radiation collective dose for the last 10 years shows a decreasing trend in general due to the improvement of the radiation protection management program.

Fuel defect at Kori unit 2 affected the increase of collective dose in 1992 and 1993. RCP maintenance jobs at Kori units 3, 4 and at Yonggwang units 1, 2 were the contributor of the increased radiation collective dose in 1995.

Steam generator replacement job at Kori unit 1, which is the oldest NPP in Korea, was the major reason of increase in 1998.

8. Low-Level Solid Radioactive Waste

Over all trend of solid waste generation for the last 10 years indicates a decreasing

trend. However, solid wastes generated during the repair of the steam generator tube leak of Kori unit 1 in 1990 and the decontamination following a contamination event in a controlled area of Kori unit 1 in 1995 caused the increase of this indicator. Steam generator replacement job at Kori unit 1 in 1998 also increased the generation of solid waste.

Discussions

The PIs have been used as a quantitative measure of performance trend for Korean nuclear power plants and the performance trend and analysis of it have been published as an annual report since their development.

However, the PIs were not used for direct regulatory purpose because current PIs have limitations.

With the 3 years experience and increasing demand of public for the information on the safety performance of nuclear power plant, discussions on improving PIs are carefully undergoing in Korea. Topics include completeness of PIs, regulatory applicability, PIs for different reactor types, international cooperation and public open through internet etc.

References

1. Numerical indicators of nuclear power plant safety performance, IAEA-TECDOC-600, IAEA

2. Operational safety performance indicators

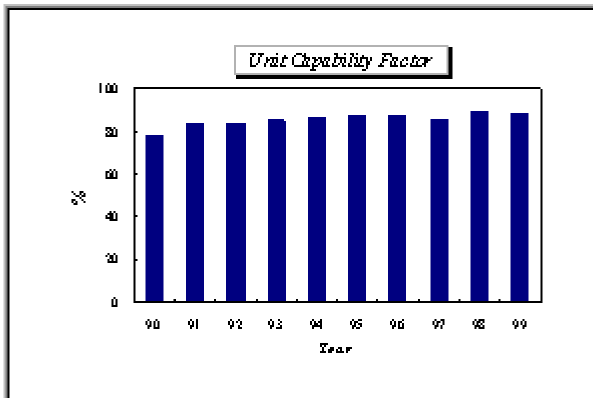
for nuclear power plants, IAEA-TECDOC-1141, 2000, IAEA

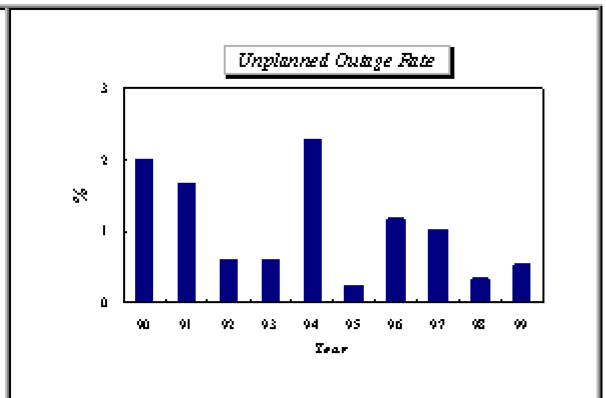
3.Sae Yul Lee et.,al, ‘Performance Indicators of Korean Nuclear Power Plants’, KINS/AR-729, 2000,KINS

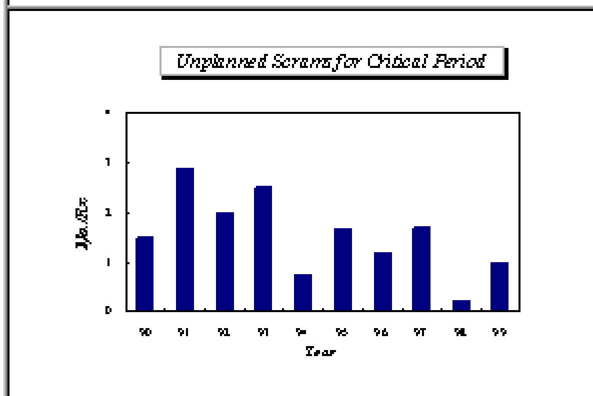
Appendix: Illustration of PIs

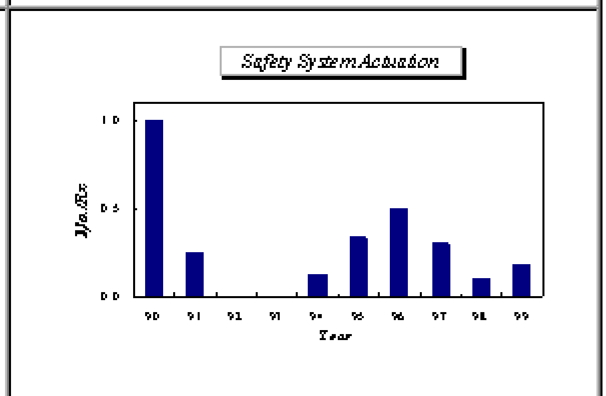
(1990 □ 1999)

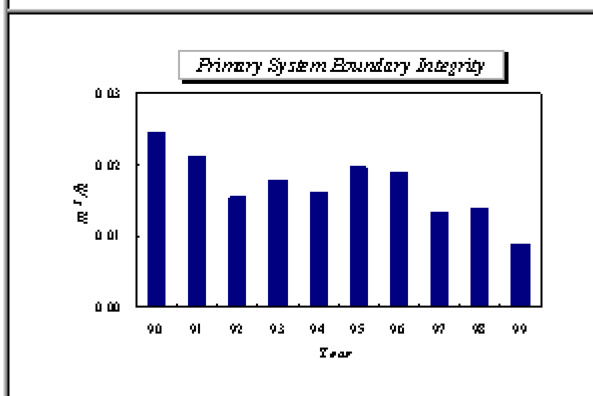
Fig. 1. Average Trend of Each Indicator

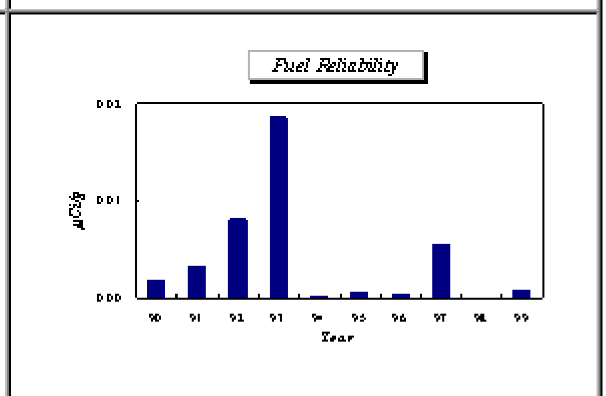

Unit Capability Factor
100
80
60
40
20
0
%
90 91 92 93 94 95 96 97 98 99
Year

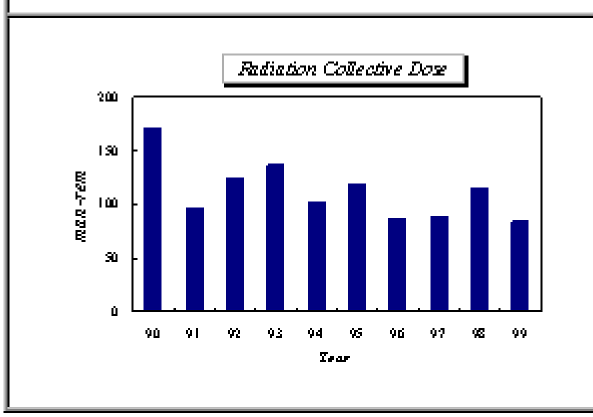

Unplanned Outage Rate
3
2
1
0
%
90 91 92 93 94 95 96 97 98 99
Year


Unplanned Scrams for Critical Period
No./Rx
Year


Safety System Actuation
1.0
0.5
0.0
No./Rx
Year


Primary System Boundary Integrity
0.03
0.02
0.01
0.00
90 91 92 93 94 95 96 97 98 99
Year


Fuel Reliability
μCi/g
Year


Radiation Collective Dose
200
150
100
50
0
man-rem
90 91 92 93 94 95 96 97 98 99
Year

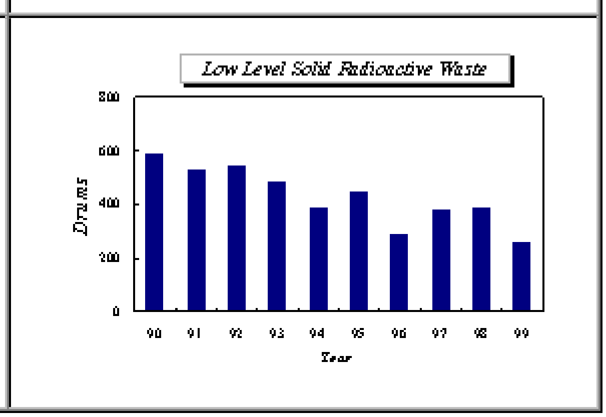

Low Level Solid Radioactive Waste
800
600
400
200
0
Drums
90 91 92 93 94 95 96 97 98 99
Year

Fig. 2 . Trend of Indicators Categorized by Reactor Capacity

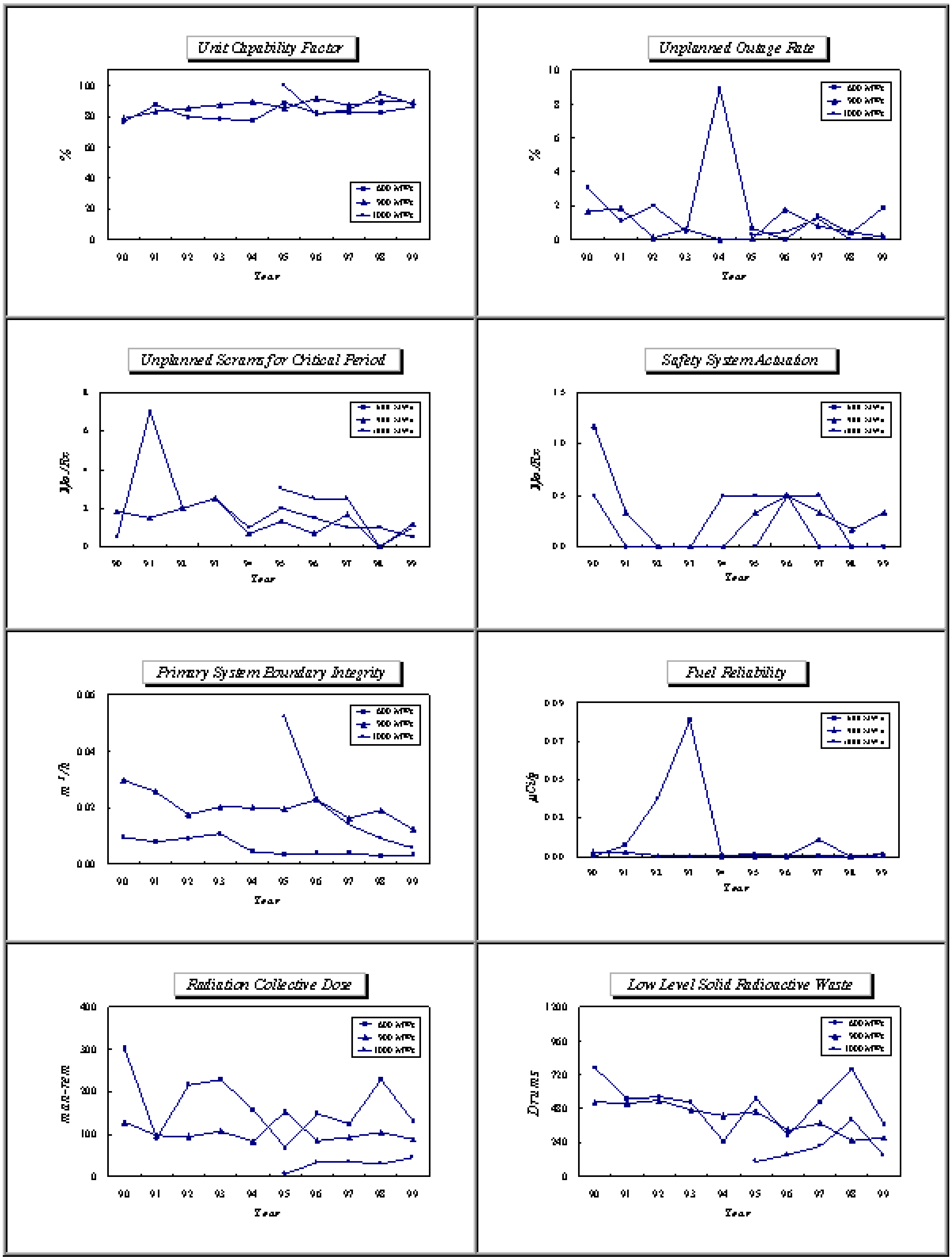
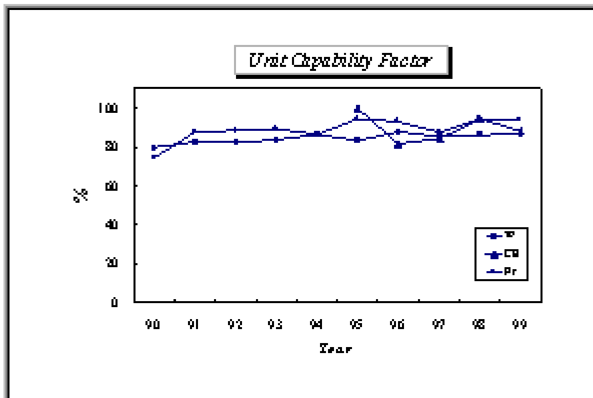
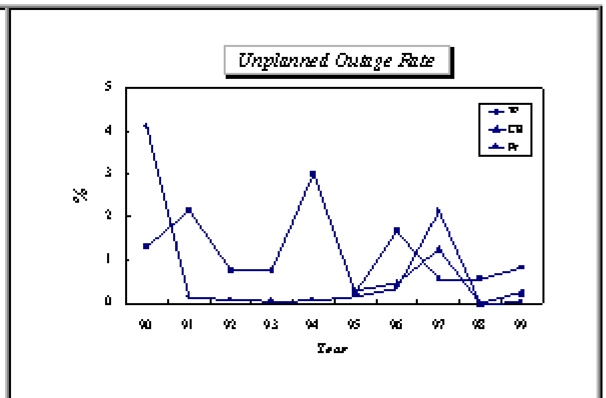
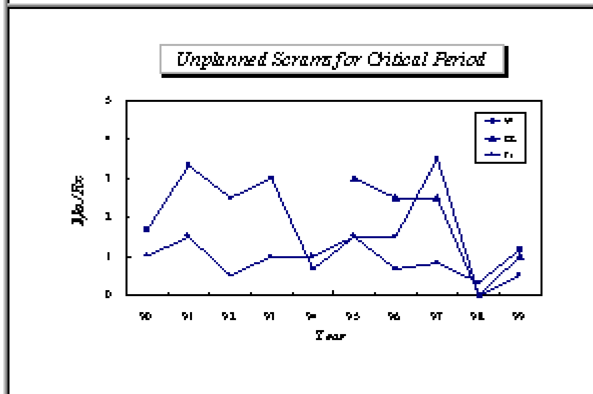
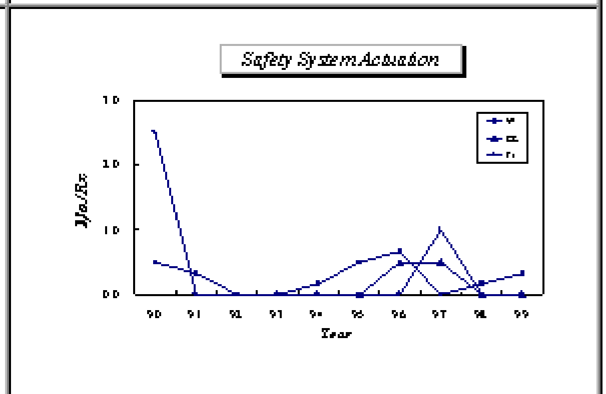

Unit Capability Factor
%
Year
Unplanned Outage Rate
%
Year
Unplanned Scrams for Critical Period
No./Rx
Year
Safety System Actuation
No./Rx
Year
Primary System Boundary Integrity
Year
Fuel Reliability
μCi/g
Year
Radiation Collective Dose
man-rem
Year
Low Level Solid Radioactive Waste
Drums
Year

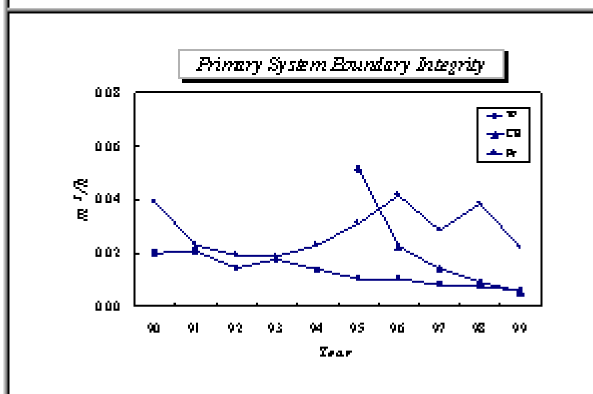
Fig.3. Trend of Indicators Categorized by Reactor Supplier

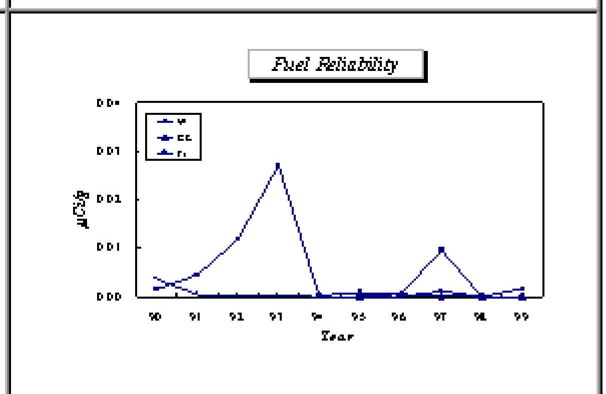

Unit Capability Factor
%
Year

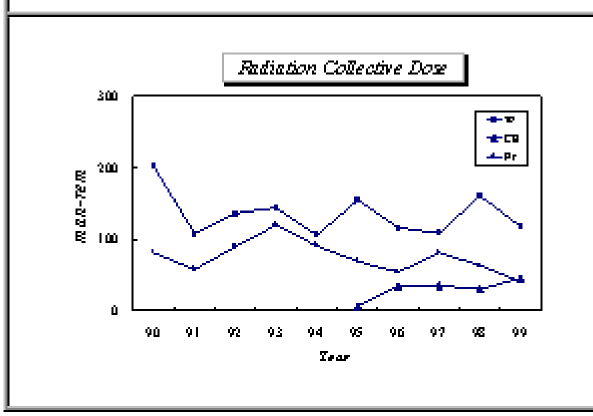

Unplanned Outage Rate
%
Year

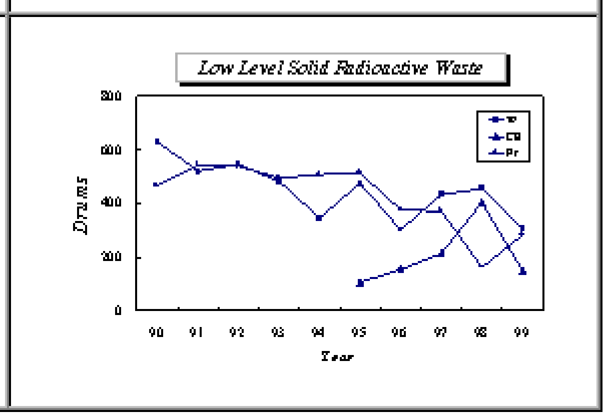

Unplanned Scrams for Critical Period
No./Rx
Year


Safety System Actuation
No./Rx
Year


Primary System Boundary Integrity
Year


Fuel Reliability
μCi/g
Year


Radiation Collective Dose
man-rem
Year


Low Level Solid Radioactive Waste
Drums
Year

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

NEW PERFORMANCE INDICATOR SYSTEM IN SPAIN

Maroño, M.,

CIEMAT (Research Center for Energy, Environment and Technology), Av. Complutense, 22, 28040 Madrid. Tel: +34 91 3466382; Fax: +34 91 3466005; e-mail: marta.marono@ciemat.es

Abstract

This paper describes the work being performed by CSN (Nuclear Safety Council) and the Utilities (UNESA) to develop a new system of Performance Indicators (PI) for the regulator and the Spanish Nuclear Power Plants (NPP´s).

For the last six years, the CSN has been using the 8-indicator Nuclear Regulatory Commission (NRC) PI system. However, some reasons have recently promoted a review of the Spanish system and the starting up of a new development.

To meet these objectives, an *ad hoc* working group, made up of representatives of CSN, Utilities and CIEMAT as technical support, has been formed.

Up to now, the main activities carried out by the working group involve: (1) a definition of criteria for selecting indicators, (2) a review of main performance indicator systems currently in use in different countries and organisations, (3) a proposal of a suitable set of indicators for the Spanish NPP´s and regulator and (4) a proposal of definitions and calculation algorithms for each indicator.

A description of all these activities and those still under development is presented here.

introduction

Since 1994 the CSN has been using the NRC 8 - performance indicator system.

For the last six years, the CSN has been collecting data and calculating the 8 indicators for the Spanish NPP´s. Since 1996 CIEMAT has been the contractor that gathered and processed the PI data, developed the software and prepared the annual report of Performance Indicators.

As most of the Spanish NPP´s have the same technology as the US NPP´s, this system allowed CSN to take advantage of the experience already gathered by the NRC in the use of the indicators and the possibility of making comparisons between PI results.

The CSN annual PI report which brought together the Spanish and US results has been very useful in order to identify significant deviations in performance.

Figure 1, included in Appendix 1, shows the CSN and NRC PI results for this 6-year period. As can be seen there, some indicators (e.g. Significant Events or Safety System Failures) show differences in performance while other (e.g. Collective Radiation Exposure) are very similar. The comparison of results allowed, in some cases, the identification of the causes of the differences in some PI results.

However, the NRC has recently changed their whole oversight process and, since April 2000, a new PI system is being used at the US Nuclear Power Plants.

Without data to make comparisons, the NRC PI system loses part of the interest for the CSN. Besides this reason, some indicators of the current CSN system are considered subjective or not risk significant. This is the case, for example, for the Significant Events and the Cause Code Indicators respectively.

These facts have motivated the development of a new system of indicators for the Spanish NPP´s and the regulator.

It is worthy mentioning that this development is in agreement with an international tendency to review the plant safety measurement tools used by regulators and licensees, shown by the increasing number of international conferences and workshops that include this topic.

The paragraphs below describe the steps already followed to develop the new system of performance indicators in Spain and the scheduled activities until the implementation of the system in the whole Spanish industry.

DEVELOPMENT OF A NEW SYSTEM OF PERFORMANCE INDICATORS

The reasons that have promoted this new development can be summarised in two: (a) the need to define more objective and risk significant indicators and (b) the changes of the NRC PI system that prevents the comparisons of PI results.

Representatives from CSN, Utilities and CIEMAT have formed a working group to carry out this development.

This makes a significant difference with the process followed six years ago, when the former system of indicators was implemented. While then the decision was taken by the CSN, now a consensus between licensees and regulator is desired.

This working group has already carried out the following activities: a definition of objectives and criteria, a review of the current PI systems and the preparation of a draft proposal of indicators.

All these activities are detailed below:

1. Objective Definition

The objective of this development is the design and implementation of a new system of Performance Indicators for the Spanish NPP´s and the CSN with the following characteristics:

∗ The performance indicator system should cover three significant areas of safety: Stability of Operation, Reliability of Mitigating Systems, Barrier Integrity and Radiological Impact.

∗ The annual report that contains the results of the indicators must be open to the public. This implies a significant difference with the current PI system which results were only for CSN internal use.

∗ The new system should allow its integration into future international programs.

2. Criteria Definition

The working group has established the criteria that indicators should fulfil in order to reach the defined objectives. They have been proposed taking into account the previous experience with the former system of indicators.

Thus, in the new proposal, the indicators should meet the following criteria:

- To be significant for the plant operational stability and plant risk.
- To be objective and not redundant (the same event should not be counted in more than one indicator).
- To be obtained with information already available at the plants.
- The results of the indicators must be open to the public.
- A reduced number of indicators should be defined.
- The indicators must be easily understood by the public.
- The indicators should allow the establishment of trends.
- Some of the indicators should belong to international PI systems in order to allow comparisons and the tracking of historical trends of results.
- The PI should be opened to be integrated into international programs.
- The indicators should be not susceptible to tampering.

Some of these criteria were already met by indicators of the current system so their inclusion in the new proposal could be considered.

3. Review of current PI systems

Once the objectives and criteria were established, the working group proposed the review of the main PI systems currently in use, in order to identify those that could fulfil the criteria.

International and National systems of indicators (such as WANO, NRC, NUPEC, EU, EDF, etc.) have been reviewed and discussed in the working group.

This activity showed that many countries use PI and some of them are present in almost every system.

Some differences appear related to the period of calculation or the scope of the indicators.

Based on this review it was possible to identify some indicators that fulfil the established criteria. Some examples are: The Unit Capability Factor (WANO) or the SCRAMS indicator which have been included in the new system.

4. Draft Proposal of Performance Indicators

Taken into account the defined objectives, the proposed criteria, the Performance Indicator systems currently in use in another countries and the own experience of performance of the Spanish plants, the working group prepared a draft proposal with 12 indicators grouped in four Areas of interest.

Although a summary of the indicators is included in appendix 2, a brief description of the new PI system is detailed here.

The draft proposal includes the following indicators:

AREA 1: Stability of Operation

This area is covered by four indicators related to plant performance.:

- Unit Capability Factor
- Number of SCRAM´s/7000 critical hours
- Number of Non Scheduled Shutdowns (excluding SCRAMS)
- Number of Forced Safety Systems Actuation

AREA 2: Reliability of Mitigation Systems

Two indicators that inform about the reliability and availability of the mitigation systems are included in this area:

- Safety System Failures
- Safety System Unavailability

The Safety Systems considered for each reactor design are as follows:

PWR.	BWR
Emerg. AC Power system	Emerg. AC Power system
HPSI	HPCI/HPCS
RHR	IC/RCIC
AFW	RHR

AREA 3: Barrier Integrity

This area has two indicators that give information about the integrity of the barriers:

- Reactor Cooling System (RCS) specific activity
- Reactor Coolant System Identified Leakage Rate

AREA 4: Radiological Impact

This area includes three indicators that inform about the radiological impact to workers (personnel and contractors). The indicators are:

- Collective Radiation Exposure
- Volume of low and medium level solid radioactive waste
- Activity of liquid radioactive release
- Activity of gas radioactive release

At this point it might be worthy highlighting the differences of this new proposal with the current 8-indicators system. The new proposal has the following characteristics:

- The results of the indicators will be open to the public. The current 8-indicator system is not open to the public. To now, licensees submitted data to CSN, an annual PI report was delivered with the PI results and the results were only for CSN internal use.
- All the indicators included in the new proposal are related to safety systems or they are risk significant (all the indicators belong to a safety relevant area).
- The new system contains indicators of international use that will allow comparison and historical track of the results.
- The indicators are objective.
- The calculations will have an annual base in accordance with many other PI systems.

NEXT ACTIVITIES

The definitions and calculation algorithms for each of the 12 Performance Indicators included in the draft proposal are now under discussion and development.

The working group is also involved in the definition of the following activities and specific tasks of the development. A brief outline of the schedule is as follows:

- development of a specific software to collect the data and obtain the new PI results,
- design and performance of a pilot experience with the participation of two Spanish NPP´s,
- analysis of results and lessons learned from the pilot experience,
- implementation of the new PI system in the whole industry.

A detailed description of these next activities is included:

1. Development of a specific software

The new system of indicators requires a specific software that simplifies data collection and indicator calculation. The development of this new software package involves the following tasks:

- Detailed definition of the required technical specifications.
- Design and implementation of database structure, calculations, desired outputs, etc.
- Software review after pilot experience in order to include proposed improvements.

2. Pilot Experience

Prior to the implementation of the proposed system of indicators in the whole industry, a pilot experience should be performed in order to check the availability of data, the suitability of scope and calculation periods defined for the indicators, the performance of the software, etc.

In a recent meeting of the working group it was proposed that two NPP´s participate in this pilot experience, one BWR type and one PWR type.

Although this activity is still under development, two tasks can be outlined:

- Pilot experience design: during this task two pilot NPP´s will be selected and the main steps for the implementation of the draft PI system will be defined.
- Implementation: the new PI software will be installed at the pilot plants and supporting meetings will be hold to solve issues related to data compiling, indicator calculations or software problems.

3. Analysis of results

The insights gathered during the pilot experience and the results of the PI obtained after the implementation of the new system of indicators in the pilot plants would allow the improvement of the new system of indicators.

A report will contain the main results and lessons learned from the pilot experience. The definitions, calculations, and the scope of the indicators will be reviewed and a final proposal of the system will be delivered.

4. Implementation in the whole industry

As the starting up of this activity is strongly dependent on the ending dates of previous steps, it has not been scheduled yet. However, it is expected that the new system will be operative at the end of 2001.

CONCLUSIONS

Although the work presented in this paper is still under development, some conclusions can be given in advance.

Firstly, the collaboration between regulators and utilities in the development of the new Spanish PI system allows to achieve results agreed by consensus and profitable for both. The participation of CIEMAT adds the necessary technical support to carry out this new development.

The new PI system is designed to be objective and risk significant and contains indicators that could be integrated into international programs. Also, it is important to highlight that the results of the new PI system will be open to the public.

Finally, in a wider perspective, the new PI system presented in this paper could serve as the basis for the definition of an international system of performance indicators for the regulator.

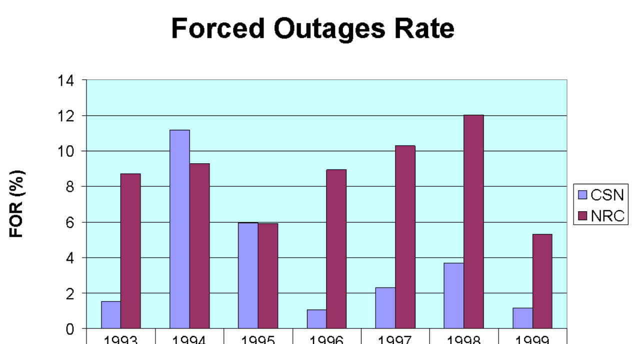
ACKNOWLEDGMENTS

This paper describes the work carried out by the Performance Indicator working group, led by Mr. Palomo (UNESA) and Mr. Zarzuela (CSN), as part of the activities defined in the CSN-UNESA R&D contract. The participation of Ciemat is possible through the CSN-CIEMAT research contract n. 211/96.

APPENDIX 1

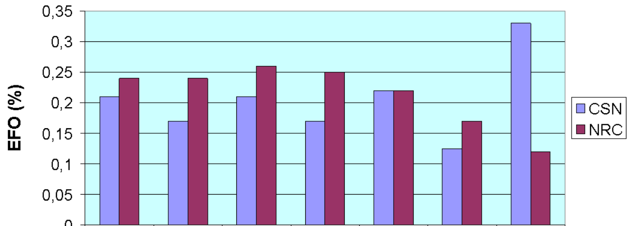
Figure 1: ANNUAL INDUSTRY (CSN AND NRC) PERFORMANCE INDICATORS AVERAGES

(1999: only 3 quarters



	1993	1994	1995	1996	1997	1998	1999
CSN	1,54	11,19	5,96	1,07	2,31	3,7	1,14
NRC	8,73	9,3	5,9	8,94	10,3	12,04	5,3

Forced Outages rate for 1000 hours of critical commercial operation

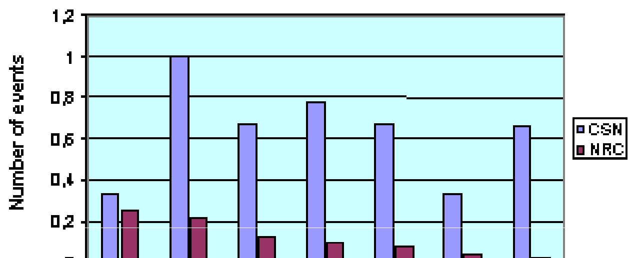


	1993	1994	1995	1996	1997	1998	1999
CSN	0,21	0,17	0,21	0,17	0,22	0,125	0,33
NRC	0,24	0,24	0,26	0,25	0,22	0,17	0,12

Year

Collective radiation exposure 1999: only 2

Significant Events

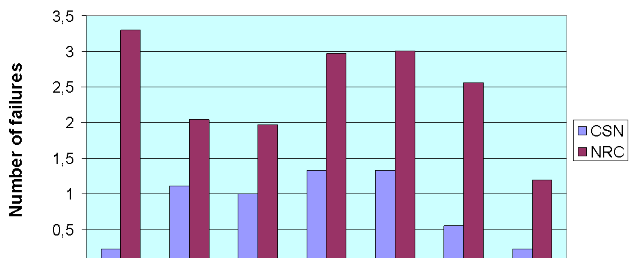


	1993	1994	1995	1996	1997	1998	1999
CSN	0,33	1	0,67	0,78	0,67	0,33	0,67
NRC	0,26	0,22	0,13	0,1	0,08	0,04	0,03

Year

quarters)

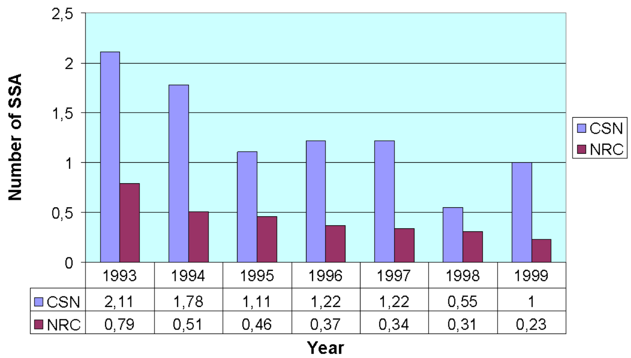
Safety system failures



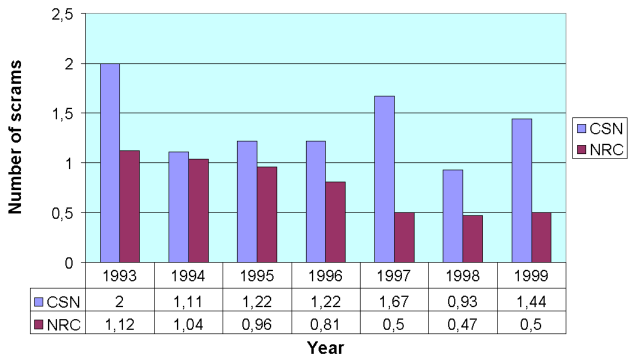
	1993	1994	1995	1996	1997	1998	1999
CSN	0,22	1,11	1	1,33	1,33	0,55	0,22
NRC	3,3	2,04	1,97	2,97	3,01	2,56	1,19

Year

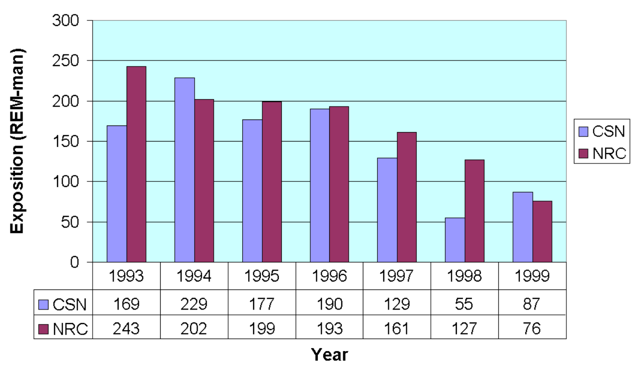
Safety Systems Actuations



Scrams while critical



Radiation exposition



APPENDIX 2

Future Performance Indicators of Spanish NPP

PERFORMANCE INDICATOR	*AREA*
Unit Capability Factor (%)	PERFORMANCE STABILITY
Non scheduled shutdowns/year (excluding SCRAMS)	
SCRAMS/7000 critical hours (auto + manual)	
Non scheduled Safety System Actuation [(1)] /year	
Safety System Failures[(2)] / year	RELIABILITY OF MITIGATING SYSTEMS
Safety System Unavailability / year	
RCS Activity (% TS limit)	BARRIERS INTEGRITY
RCS Identified Leakage (% TS limit)	
Collective Radiation Exposure (Sv-year)	RADIOLOGICAL IMPACT
Volume of Low and Medium Level Solid Radioactive Waste (m^3/year)	
Activity of Gas Radioactive Release (GBq/year): Total w/o Tritium/ Tritium	
Activity of Liquid Radioactive Release (GBq/year): Noble gas/Halogens/Particles/Tritium	

[(1)] Safety System Actuation: It is counted as long as the challenged System fulfils its function: to inject water, to supply power.

[(2)] The Safety Systems considered vary at different reactor design:

PWR.	BWR
Emerg. AC Power system	Emerg. AC Power system
HPSI	HPCI/HPCS
RHR	IC/RCIC
AFW	RHR

Paper of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

DEVELOPMENT OF SAFETY PERFORMANCE INDICATORS OF REGULATORY INTEREST (SAFPER) IN PAKISTAN

ABIDA KHATOON

Principal Scientific Officer

Pakistan Nuclear Regulatory Board

Directorate of Nuclear Safety & Radiation Protection

P.O. Box No. 1912, Islamabad, PAKISTAN

Tel: ++92 051 9205917 Fax::++92 051 9204112

E-mail: safety@paknet2.ptc.pk

Abstract

Safety performance indicators provide a very useful tool for monitoring operational safety of a nuclear power plant. Utilities in many countries have developed plant specific indicators for the assessment of their performance and safety. Regulators can make use of some of these indicators for their regulatory assessment. In addition to these regulatory bodies in some countries have also developed programs for the formulation of safety performance indicators which are used in monitoring operational safety and regulatory decision making. Realizing its usefulness Directorate of Nuclear Safety and Radiation Protection (DNSRP-the regulatory body in Pakistan) has also initiated a country specific program for the development of Safety Performance Indicators (SAFPER) based on data provided by the utility and that collected during the course of regulatory inspections. Selected areas of NPP operation to be monitored are:-

- Significant events
- Safety systems performance
- Barriers integrity

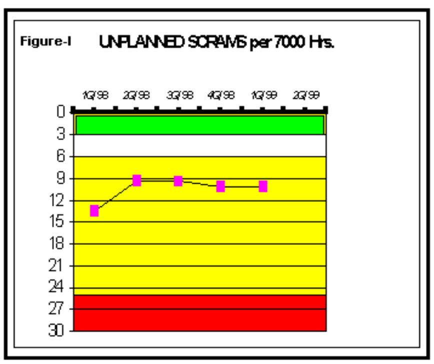
- Environment protection
- Workers radiation safety and
- Emergency Preparedness

One of the objectives of this program is also to monitor the effectiveness of DNSRP regulatory activities. IAEA framework is taken as one of the bases for our program. Safety performance will be assessed on the basis of Performance Indicators and inspection findings. DNSRP program as shown in (A**ppendix-I)** includes the indicators in use and under development

introduction

The Pakistan Nuclear Regulatory Board (PNRB) was established in October 1994 and is responsible for carrying out Nuclear Regulatory Functions in the country. Directorate of Nuclear Safety and Radiation Protection (DNSRP) is the executive arm of PNRB, functional chart of PNRB is shown in Appendix-II. DNSRP has been carrying out regulatory surveillance of the Chashma Nuclear Power Plant (CNPP) during its design, construction and commissioning phases. The Karachi Nuclear Power Plant (KANUPP) is also under regulatory surveillance. Routine regulatory inspections are carried out in addition to special inspections. The Pakistan Nuclear Regulatory Authority Act (PNRA Act) which envisages PNRA as a complete independent nuclear regulatory body in the country is in the final stages of legislative approval.

As the regulatory body (DNSRP) licenses the construction and operation of NPPs, develops, implements and enforces the rules and regulations that govern nuclear activities, inspects facilities to ensure compliance with legal requirements. It also stations inspectors at KANUPP and CNPP sites and supplements their inspection activities with special inspections by the staff from the Headquarters at Islamabad. After award of an operating license DNSRP's statutory obligation is to require sufficient information from the licensee to enable it to assure itself that adequate protection is being provided for the health and safety of the public.



KANUPP was permitted to operate in early '70s without a formal license. This deficiency will be rectified when a revised full scope FSAR is submitted, and a formal license issued. At present PNRB has issued a fuel load permit to CNPP which is being commissioned. For the purpose of the Safety Performance indicator Program in Pakistan CNPP will be taken into account after it starts commercial operation.

Current Program and Experience in the use of safety performance indicator program (SAFPER).

For the past many years KANUPP was using WANO indicators which are primarily performance based indicators. As all of these indicators are not useful for the regulators, some other indicators have been proposed and used from the regulatory point of view. In order to measure and monitor the operational safety of KANUPP, DNSRP generated a trend data for the following indicators already in use. The data has been taken from KANUPP Special Technical Reports on Safety Aspects of KANUPP Operation 1995-1999.

Unplanned Scrams per 7,000 Critical Hours

Automatic Scrams/Trips While Critical.

The number of unplanned automatic scrams that occurred while the reactor was operating. (An automatic scram is a condition under which the reactor shuts down automatically as a result of being programmed to do so under certain conditions.) This results in thermal and hydraulic transients and represent challenges to plant safety systems. The following data is reported

•the number of unplanned automatic and manual scrams while critical in the previous quarter

•the number of hours of critical operation in the previous quarter

The indicator is determined using the values for the previous four quarters as follows:-

value =

(total unplanned scrams while critical in the previous 4 qtrs) 7,000 hrs

(total number of hours critical in the previous 4 qtrs)

The value of 7,000 hours is used because it represents one year of reactor operation at an

80.0% capacity factor. If there are fewer than

2,400 critical hours in the previous four

quarters the indicator value is computed as

N/A because rate indicators can produce misleadingly high values when the denominator is small. KANUPP remained shutdown for the first three quarters of the year 1999 and remained critical only for 1045.94 hours having 3 scrams out of which two were unplanned and one was planned.

Figure-1 shows that all the scrams are in the yellow band of Regulatory Response Band, This indicator has provided a good correlation with plant performance in the past and is considered to be a leading indicator of the more risk-significant indicators . In Figure-I the GREEN area indicates the acceptable performance in the licensee respond band, WHITE indicates the acceptable performance in the Technical Specification limits, YELLOW indicates acceptable performance in the Regulatory Response Band exceeding Technical Specifications Limits, while RED indicates Unacceptable Performance and plant performance is outside the design basis .

Unavailability of Safety System

The following KANUPP safety-related systems were chosen for monitoring

- Emergency Injection Systems
- Dousing Water Systems
- Emergency AC power

These systems were selected for the safety system performance indicator based on their importance in preventing reactor core damage

or extended plant outage Figure-2 only shows the trend and not the indicator value.

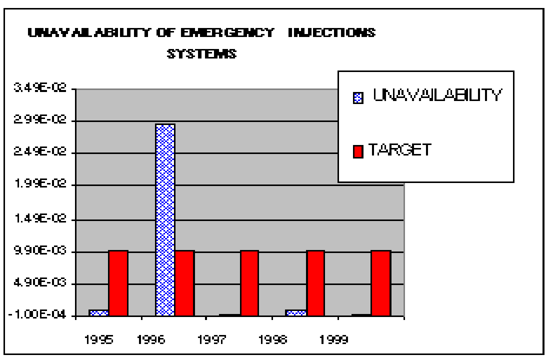


Figure-2 a

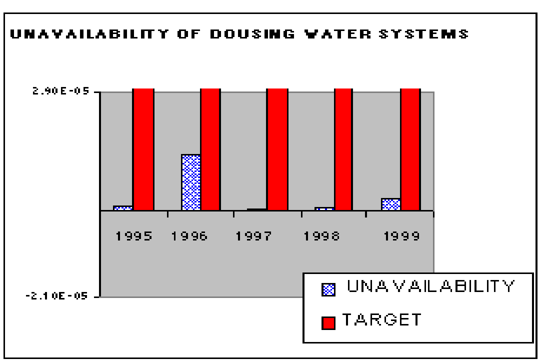


Figure-2 b

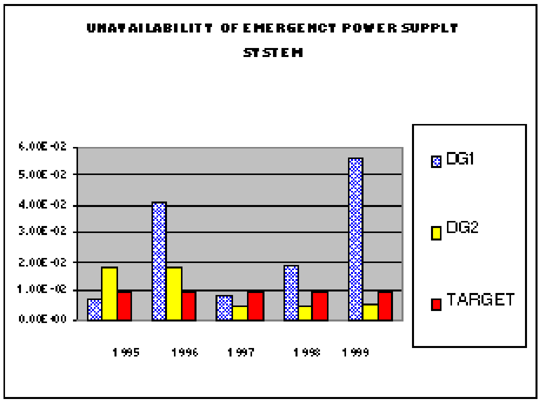


Figure-2 c

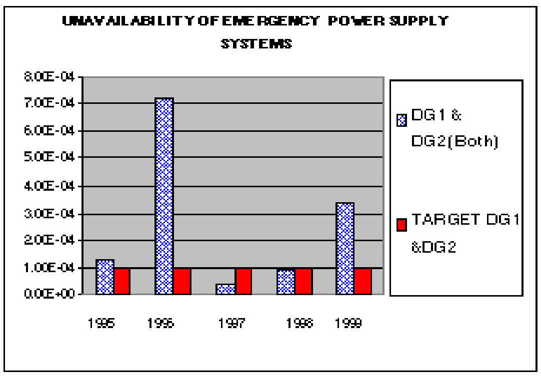


Figure-2 d

This safety performance indicator can only be accurately and objectively measured by establishing safety performance indicator values or the decision thresholds. In order to calculate the indicator value we have to have the data for the following data elements:-

- Planned unavailable hours — These hours include time the train was out of service for maintenance, or any other time equipment is electively removed from service and the activity is planned in advance.

- Unplanned unavailable hours — These hours include corrective maintenance time or elapsed time between the discovery and the restoration to service of an equipment failure or a human error that makes the train unavailable.

- Fault exposure unavailable hours— These are hours that the train was in an undetected, failed condition

Unavailability =

$$\frac{\Sigma \text{ (Planned, unplanned, and fault exposure unavailable hours.)}}{\text{hours train required}}$$

The unit or station indicator value is the sum of the train unavailabilities for that system divided by the number of system trains. To acquire this data we have to formulate a data entry form which chould be utilized by the licensee for easier data entry. We are under the process of developing this.

Radiation Protection Program Effectiveness:

Occupational Exposure Control Effectiveness

The purpose of this strategic indicator is to monitor efforts to minimize total radiation exposure at the facility. This parameter is a measure of the effectiveness of radiological protection programs in minimizing radiation exposure to plant workers. Collective radiation exposure, is the total external and internal whole body exposure determined by primary dosimeter (TLD or film badge), and internal exposure calculations. **Figure-3 a to f** represent the trend of effectiveness of radiological protection program at KANUPP.

Collective Radiation Exposure.

The total radiation dose accumulated by KANUPP employees for the year 1999 is about 17% less than the station dose of previous calendar year. **Figure-3a**

WANO KANUPP value is 2.04 man-Sv while median =1.00 manSv

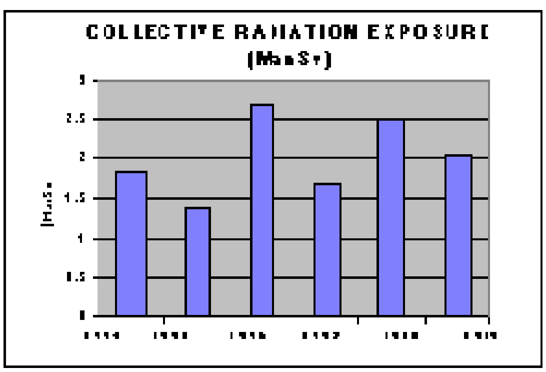
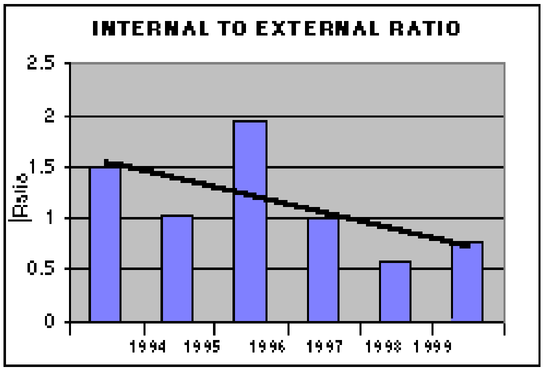


Figure-3 a

Radiation Doses Internal to External

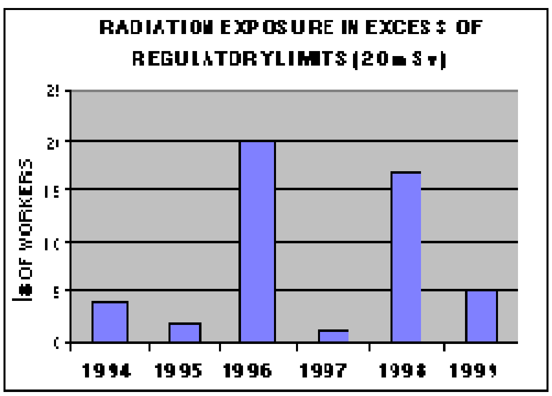


Ratio.

Figure-3 b

Due to the implementation of ALARA Program

major efforts have been taken to reduce the internal exposure due to uptake of Tritium. **Figure-3b** shows the decreasing trend.



Number of Workers Receiving Doses above 20 mSv.

Figure-3c

The indicator indicates risk-informed dose criteria and encompasses events that represent a substantial potential for exposure in excess of regulatory limits. Figure-3c shows a trend graph for the period 1994-1999. In 1999 Five persons received the annual dose greater than 20 mSv. Radiation doses received by them in excess of 20 mSv will be compensated during next year to keep the average below the regulatory limit during the current five year segment (1998 – 2002) in conformity with ICRP-60.

On-Power Entries into the Boiler Room

Figure-3d and Figure-3e show a good trend in the specific indicators for the period 1994-1999. During the year 1999, two on-power entries were made into the Boiler Room. Five persons who were involved in these entries received total dose of 1.24 Man-mSv consuming about 0.2 Man-hrs. The average dose per worker per entry is 0.248 mSv.

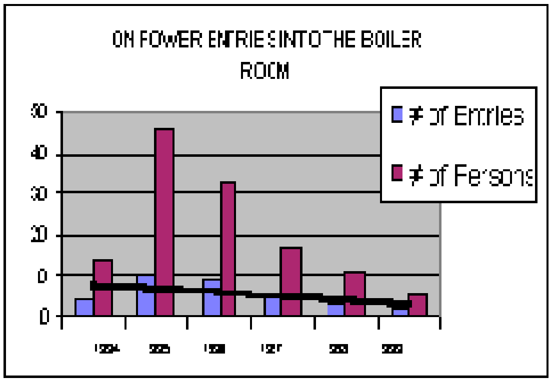


Figure-3d

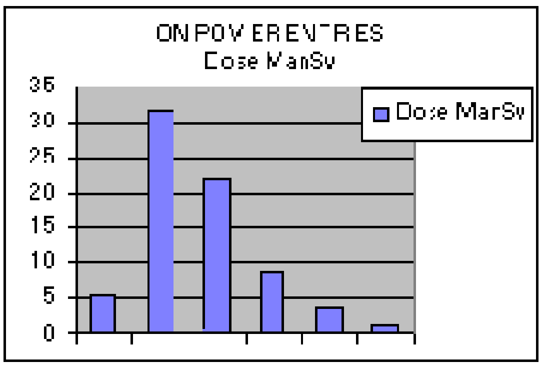
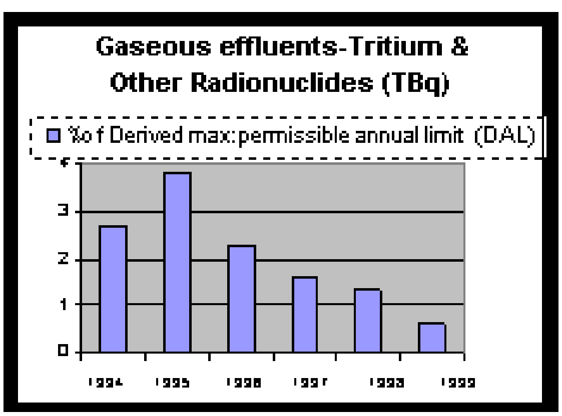
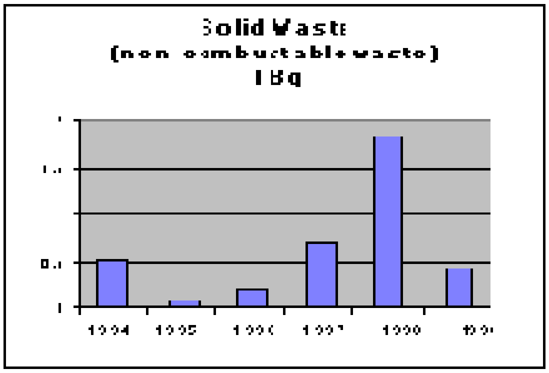


Figure-3e

It has now been felt that DNSRP must also develop safety performance indicator values (decision thresholds) for Radiation Protection Effectiveness Program at KANUPP. Thresholds have to be identified for the Required Regulatory Response Band or the Unacceptable Performance Band because the indicators trends cannot be directly tied to risk data. These values will be important for prompt decision from regulatory point of view. The available data is not sufficient for the required objective. Moreover the data should be easily available to the regulatory body via the licensee or through inspections in a regular and timely manner.



Public Health and Safety.



To assess the performance of the radiological effluent **control** program this indicator shows the bases for protecting public health and safety from exposure to radioactive material released into the environment as a result of NPPs operations. These releases include routine gaseous and liquid radioactive effluent discharges,. The indicator uses as its bases, the dose limits for individual members of the public specified in PNSRP Regulation-1990, which defines that doses to members of the public from effluent releases be kept "as low as reasonably achievable" (ALARA).

Station Effluents

Figure-4 a to d show the trends in the radioactive releases into the environment. A good trend is observed while looking at the trend graphs.

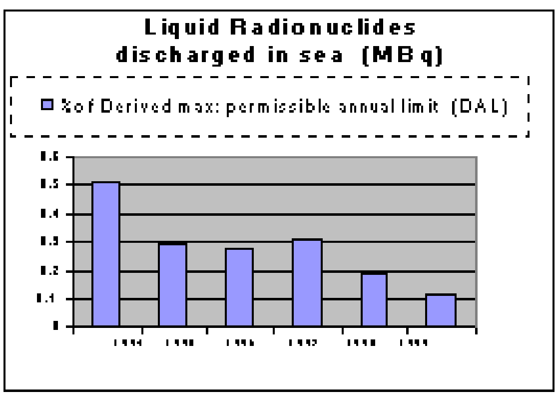


Figure-4 a

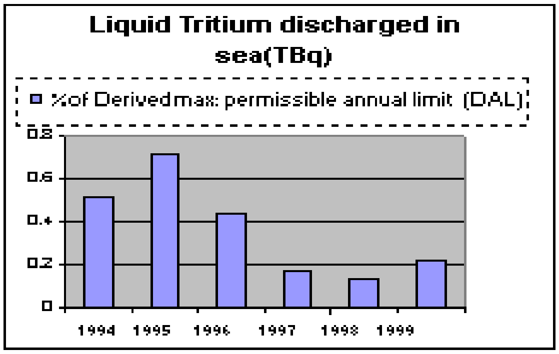
Figure - 4 b

Figure-4 c

Figure-4 d

Experience:-

It is found that there was a positive correlation between the performance levels as indicated by the indicators and the DNSRP's evaluations of the reactor, the observed trend of the safety performance indicators also signifies the ageing of the plant. However the safety performance indicators are only one of the tools, DNSRP uses to measure performance safety.



If plants safety performance indicators and other data show a pattern of deteriorating safety performance an additional oversight and more intensive inspection activity will be carried out by DNSRP. Since at present there is only one operating plant in the country a comparison of safety performance trend cannot be generated. A comparative study with the other good performer plants is thus suggested, This can be done by exchange of information on safety performance indicators in the open International programs and projects.

Advantages of Safety Performance indicators :

- Accurately and objectively measures the safety performance of NPP in protecting the public health and safety.
- Provide accurate and understandable safety performance information to the public and news media.
- levels and trends of nuclear safety can be illustrated in a quantitative manner
- weaknesses at nuclear power plants can be identified.
- Trend result can be utilized for the allocation of resources in an effective and efficient manner.
- Regulators can develop inspection plans.
- to evaluate and develop its own supervision
- to develop co-operation between regulatory body and the utility

Limitations:

- The safety P.I. provide result in a more quantitative manner and in some instances does not provide meaningful information.
- It has also been observed that there has been no concerted effort to verify the date for completeness and accuracy.

In view of the above, **some additional indicators** to the current set of safety performance indicators has to be considered to provide the management with a more objective basis for monitoring the safety condition of a reactor. Apart from the performance safety indicators some regulatory effectiveness indicators have to be dealt with for a complete picture of safety.

2. Regulatory Efficiency Indicators

Regulatory efficiency is a measure of the performance of the regulatory system which exists in a country to assure the safety of the public and workers from nuclear activities. Achievement of Safety levels at nuclear facilities is not an exclusive indicator of the Regulatory Body. There are various other factors which are prerequisite of an effective Regulatory Organization eg.:-

1. Independent body-- the most important indicator of Regulatory Efficiency is if the regulatory body has enough powers to make and implement its own decisions as for safety is concerned.

2. Well defined Safety Policies and Objectives.

3. Organization Size and Structure.

4. Allocation of Resources.

5. Trained Manpower.

6. Reliability

7. Internal Quality Assurance.

8. Some other possible indicators to assess the regulatory efficiency are:-

- The ratio of time spent on planned inspections to time spent on reactive inspections (should be high).

- An average time from the identification of the poor results to the decision to update the regulation or not to update the regulation, and further, to the issue of the revised regulation.

- Percentage of established changes within one year from the issue of new regulation.

- Number of safety issues not reported by the utility but discovered by the regulatory body.

- Number of non-conformances identified in event analysis but not discovered in inspections (not discovered by the utility nor the regulatory body)

- Average number of rates in inspection protocols (non-conformances not discovered by the utility but found by the R.B).

- Number of delayed corrective actions.

- An average time from results identification to the decision to update internal procedures and further, to the issue of the revised regulations.

- Number of internal corrective actions to be taken after an incident.

- Number of changes in the regulatory statutory requirements after an incident.

Regional Nuclear Safety Inspectorate (RNSI) was established, at CNPP in 1994 and a resident inspector has been stationed at KANUPP in 2000. The function of RNSI at CNPP was to perform the regulatory activities during the design, construction and commissioning of CNPP. Figure-5 shows Regulatory

Inspection activities by RNSI at CNPP, Table I gives a better view.

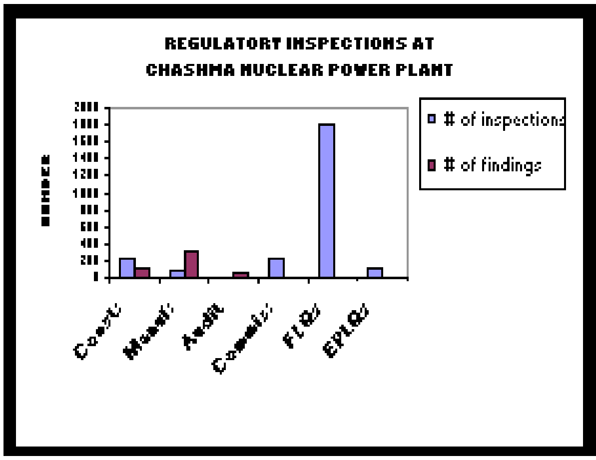


Figure-5

DNSRP is now under way to perform full scale independent regulatory activities at KANUPP. At present we do not have sufficient data for this indicator so that a meaningful trend can be generated. **Appendix-II only shows some** findings by DNSRP inspectors and corrective actions taken by KANUPP during regulatory inspections with special reference to **Workers Radiation Safety. The inspection findings are in conformation with the trend** analysis of the safety performance indicator for **Workers Radiation Safety. It** can now be concluded that the Regulatory Body in Pakistan is performing its responsibility in an effective manner with the co-operation of the Management of the Utility.

2. Risk Indicators in use and under development

Risk indicators can be divided as:-

A. Public health and safety

a. ***barier integrity*** (indicator under development -UD)

Fuel cladding, reactor coolant system and containment should be designed as such to assure the protection of the public from radionuclides, releases caused by events or accidents. These barriers are important elements and the indicators to be measured are:-

- *Reactor coolant system activity*

  At present we do not have data for this parameter.

- *Reactor coolant system boundary*

  Primary charging system (CPH) Concentration of I-131 remained within limits for the period 1995-1999 Table II shows the values for this period.

- *Containment integrity*

  Containment leakage rate is measured once/2 years. In 1996 and 1998 it was 50.38% and 51.72% of the allowable Technical Specification limit at 2 psi.

a. ***control of exposure and radioactive materials*** (indicator in use)

- *emergency preparedness* (# of drills per year)
- *radioactive material control* (gaseous and liquid effluents in the environment)
- *exposure control* (occupational exposure

A. Safety Performance Margin

a. Operating challenges

- *Unplanned Automatic Scrams* (in use)
- *Safety System Actuations* (UD)
- *Shutdown operating margins* (UD)
- *Unplanned operating transients* (UD)

a. ***Mitigation Capability*** (UD)

- *High risk significant SSC performance*

C. **Overall plant performance** (UD)

- *Plant Performance Trend* (overall trend displays by color windows)

This can only be measured when a full scope Safety Performance Indicator data has been studied, analyzed and assessed keeping in view the decision thresholds. At present we are not in a position to do this evaluation.

2. Indicators for Organization Evaluation, Safety Culture Performance Indicator.

A good regulatory body has the capability to enhance safety culture and has to make sure that

- An open interface between the utility and the regulatory body should exist.
- Utility Management must not tolerate an atmosphere that accepts degraded conditions, rather, than, establish the atmosphere of a high quality operating environment.
- The management must not take delayed decisions on whether or not equipment is operating as required.
- Inadequate management oversight is considered to be a cause of adverse quality events, which can also lead to operators and engineers not having sufficient knowledge of the design basis of structures systems and components to recognize problems and take timely corrective actions.
- Number of senior management meetings to aid early intervention.

- Contractors safety culture – should be the responsibility of the utility but may be assessed by the regulatory body

5 Indicators for regulatory usage:

It is a fact that not all the indicators developed by the utilities are useful for the regulators for regulatory decision making, some of the indicators which can be of usage to them are:-

- number of equipment failures causing unavailability of the plant.
- Ratio of preventive maintenance actions to corrective maintenance actions.
- Ratio of corrective actions of the equipment specified in the technical specifications to the all corrective actions.
- An average unavailability time of all failed equipment.
- Distribution of failures in different main systems.
- Number of human related common cause failures.
- Number of un-availabilities due to common cause failures.
- Number of common cause failures which do not cause unavailability of the equipment and system.
- Number of multiple failure (same failure causes consequences in several equipment and systems).
- Number of human single failures.
- Number of permits to deviate from the technical specifications.
- Collective radiation doses.
- Average of the ten biggest personal doses.
- Radioactive releases to the water in TBq.
- Radioactive releases to the atmosphere in TBq.
- Number of unsatisfactory utility functions noted in inspection protocols.
- Number of electric power reductions.
- Number of fire alarms
- Integrity of fuel elements (maximum activities of the primary circuit).
- Integrity of the primary circuit.

- Unidentified leakage in the primary circuit.

- Integrity of the containment (proportion of isolation values which passed the first leakage test).

- Operation of the reactors control room (statement-NRC ordered the Peach Botton Plant in Pennsylvania to shut down in 1987 after finding that personnel in the control room were sleeping on the job).

6 Opportunities and difficulties for exchange of information on safety performance indicator in the open International programs and projects

Exchange of information on safety performance indicator among regulators has not been done in the past due to the fact that we have not as yet formulated a well defined program in this area. We would like to have an opportunity for the exchange of information on Regulatory Practices in other countries, specially in formulating the indicator values and thresholds for a prompt evaluation of Nuclear Safety of a Plant. A Peer Review in this area would be a good idea for exchange of information within the region.

7. **Indicators in which the public may be interested:**

From the public perspective it is the safety of the nuclear facilities that is their primary concern. It is the performance and reliability of the engineered systems, the software and the licensee staff that together with their processes operates and maintains the facility at an acceptable level of risk. Over the years the only nuclear power plant in Pakistan has operated well within the safety limits. The operating personnel have never been subjected to excessive radiation doses and adequate safety measures have remained available to monitor and counter any potential hazardous situation.

INES constitutes a valuable service for the prompt reporting of incident to the media and public. Pakistan joined the INES information system in 1994 and under its obligation is committed to the prompt communication of the Nuclear Events significant for safety (level 2 and above) or significant for the public interest (level 1 and above).

A Pre-IRRT Mission was carried out in April 1997 to

- Review the written material

- Interviews with personnel and

- Direct observation of organization, practices and activities both at DNSRP and RNSI.

Recommendations and suggestions by the Mission were given for improvement.

The indicators of public interest in addition to the above are

- The annual average dose per worker

- environmental monitoring of both the possible routes i.e. gaseous as well liquid effluents.

DISCUSSION AND CONCLUSION

It is felt that the term Safety Performance Indicators may be termed as "**SAFPER Indicators**" to be used by the Regulators, as it is clear from this presentation that utility safety performance indicators together with the regulatory effectiveness indicators constitute the measure for the adequate safety to the public and the environment. Additional research is still necessary for

- indicator definition for the proposed and under developed indicators
- data collection systems
- thresholds
- trend analysis
- goal setting (benefit from the trend can be enhanced only if meaningful goals and targets are established)
- analysis of overall plant performance
- safety culture indicator (qualitative indicator)

Some of the indicators, like Sudden outages, unavailability of Safety Systems, Collective Radiation Exposure, Station Effluents, are analysed under Operating Performance parameter of KANUPP Report for the period 1994-1998.

As observed from the findings by DNSRP inspectors and corrective actions taken by KANUPP during regulatory inspections **Appendix-III** and the trend analysis of the safety performance indicators for **Workers Radiation Safety (Figure 3) it** can be concluded that the Regulatory Body in Pakistan is performing its responsibility in an effective manner with the co-operation of the Management of the Utility.

It is concluded that the general trend observed during this study is expected from an old plant like KANUPP. The encouraging aspect which matters for the Regulators is the trend observed for the radiation dose internal to external ratio for the period 1994 to 1998. This ratio has started decreasing now after doing some maintenance work. Another important parameter of safety concern is the number of forced outages which have been decreasing from 1994 to 1998.

Kanupp is facing operational problems mainly due to its Ageing and some other maintenance activities. These conditions can cause safety concerns that, if not appropriately addressed, would require the licensee to shut down the plant.

This deficiency will be rectified to a large extent when the effort to produce a revised full scope PSAR is complete, and a new license issued.

Acknowledgments

The author is grateful to Director General DNSRP for his valuable guidance and contribution in the preparation of this paper. I am also thankful to Head Nuclear Licensing Division (NLD) who has reviewed this document and provided technical guidance in many areas especially information regarding inspection data from the regulatory inspection of Nuclear Power Plants.

I am also thankful to my colleagues at DNSRP for their co-operation and help.

References

1. IAEA-TECDOC-1141, Operational Safety Performance Indicators for Nuclear Power Plants.

2. NEI-99-02 Revision-0, March-2000 Regulatory Assessment Performance Indicator Guideline

3. Peer Discussion on Regulatory Practices, Assessment of Regulatory effectiveness.

Appendix - I

NPP OPERATIONAL SAFETY

PERFORMANCE

• Ratio of time spent on planned inspections to time spent on reactive inspections • An average time from identification of poor results to the decision to update the regulation or not to update the regulation, and further to the issue of the revised regulation • Percentage of established changes within one year from the issue of new regulation. • Number of safety issues not	• **O**perating License Renewal Period	• **V**iolations of PNSRP Regulation. • **T**echnical Specifications violations • **N**o. of outstanding Regulator Actions • **N**on Compliance frequency • **V**iolation of physical security Reg.	• **W**orkers Radiation Safety: • *Collective Radiation Exposure.* • *Radiation Exposures in Excess of Regulatory Limits.* • *Unplanned Exposures* • *Average of ten biggest personal doses.* • **W**orkers Industrial Safety. • *Loss Time Accident Rate* • *Industrial Safety Accident frequency rate.*	• **S**tation Effluents • *Total Emissions of Radioactivity* • *Total Liquid Radioactive Discharge (Bq).* • *No. of times % derived emission limit target exceeded* • **E**nvironmental Monitoring and Population Dose. • *Maximum annual Radiation dose to members of public* • *Average boundry external dose rate* • *Volume of low level Radioactive waste stored* • **S**afety Systems Performance • *Safety System unavailability* • *Containment leak rate* • *No. of Safety Support System Events*

reported by the utility but discovered by the regulatory body. • Number of non-conformances identified in event analysis but not discovered in inspections (not discovered by the utility nor the regulatory body) • Average number of rates in inspection protocols (non-conformances not discovered by the utility but found by RB) • Number of delayed corrective actions. • An average time from results identification to the decision to update internal procedures and further , to the issue of the revised regulations				• **S**ignificant Event • *Reactor Scrams* • *Safety Systems Actuations* • *Reactor Cooling System Leakage* • *No. of Reportable Events/ precursor events.* • *Human performance/ repeated events.* • *No. of fires* • **O**perations & Maintenance (plant reliability) • *Net Capacity Factor* • *Unplanned Capability Loss factor* • *Fuel Reliability.* • *Operating Preventive Maintenance call up compliance percentage* • *Corrective maintenance backlog* • *No. of jumpers outstanding* • *Software temporary modifications* • *Chemistry index-stem generator leakage exceeding limits* • **T**raining • *Candidates passing regulator exams*

• Number of internal corrective actions to be taken after an incident. • Numder of changes of the regulatory statutory requirements after an insident.				• *Scheduled drills completed* • **S**ecurity • *No of reportable security events*

NPP OPERATIONAL SAFETY PERFORMANCE INDICATORS

Appendix-II

Functional Chart of PNRB

Appendix-III

RADIATION CONTAMINATION AWARENESS

Observation based on regulatory safety inspections

1996

The containment leak rate test was due on 10.6.1995 (tolerance time 4 months) but it was not performed so far the test should be performed in the next long shut down.

Recommendation agreed.

RPT re-qualification of all radiation workers as per station policy had to be done after every two years which was not observed during the inspection period.

Recommendation enforced.

1997

16 persons received doses more than 20 mSv and one received whole body dose of 36 mSv due to H^3 environment about 70% of the dose received to workers is due to internal radiation

More stringent radiation protection measures should be adopted. Use of masks/respirators, reduction of residence time in high radiation areas special effects should be made in the case of unskilled workers exposed to internal uptake.

Recommendation/suggestion put into effect. Use of unskilled persons was minimised and regulated. They should be supervised by a qualified plant personnel.

Radiation survey meters in the control room were calibrated in 1986 and in the Health Physics Division the radiation monitors were last calibrated in 1996. Acc. to call up card calibration is required after every three months.

Calibration should be done acc. to call up cards.

.

During the inspection, an increasing trend(40 times) of MPCa in the Boiler Room was noted.

It was also noted that the protective breathing suits have a reduction factor of about 150 which has probably deteriorated with age.

Remedial action must be taken to reduce the MPCa to reasonably low level:-

Should look into the possibility of reducing H3 con. In the Boiler Room

Necessary measures should be taken to reduce the internal exposure of the workers.

KANUPP was asked to look into the procurement of more efficient protective suits

It was observed that, the TLDs of some radiation personnel working in mechanical sections were not regularly changed and monthly exposure record of a number of workers was not available since the TLDs issued to them were not regularly monitored on monthly basis

Personnel dosimetry of all radiation workers may be ensured on regular basis and upto date exposure record should be maintained.

List of TLDs should also be updated

KANUPP informed that the record belonged to daily wages personnel generally or to employees who have been transferred or posted.

KANUPP apprised DNSRP team that all KANUPP personnel are regularly trained/retrained(rec. March '96) in the handling/protection from radiation. It was noted that operational personnel are more serious towards the said training while the attitude of maintenance personnel to RPT looks never to be desired as the failure rate is >50%

Should take vigorous steps so that all the workers are available to attend the course and should qualify the same.

The management agreed to improve the situation

The absorbed dose record showed substantial difference between the readings of Direct Reading Dosimeters (DRDs) and TLDs. This was attributed to irresponsible attitude of worker toward the use of DRDs.

Eg. Ref. Case no 2627:

DRD showed 430 mRem

TLD showed 150 mRem during Sept.'97

This deficiency should be overcome through administrative measures and training of the workers. Quality assurance of TLD system should also be ensured.

should ensure that the workers while entering the airlock areas wear the DRDs and TLDs simultaneously.

More efforts should be made in the maintenance of exposure record

The inconsistency in the two readings should also be removed

Scanning of personal dose records revealed that Mr. X had exceeded the 20mSv/year limit in November 1998. However he was deputed to perform dose intensive work on the fuelling machine in December 1998.

Keeping in view the ALARA principle such practice should be avoided.

TABLES

REGULATORY INSPECTIONS	# OF INSPECTIONS	# OF FINDINGS
Construction & Installation Phase • GSR-General Surveillance • NPSR*-Control Point Inspections	---- 223	 107 ----
Manufacturing Phase: • Record Point • QA Inspection	32 ~40	No finding 321
Audit Inspection	1	48
Commissioning:- • J-Point * • Random Surveillance	213 58	
FLQs*	1808	
EPLQs* (Emergency Plan)	~108	

Table I --- REGULATORY INSPECTIONS AT CHASHMA NUCLEAR POWER PLANT

NPSR - Notification Point Surveillance Report

J-Point - Joint Inspection (ie. In the presence of regulatory body)

FLQs - FSAR List of Questions

EPLQs - Emergency Plan List of Questions

Reactor Coolant System Boundry *Primary Charging System (CPH)* (Concentration of I-131)	January-December	January-June	June-July	July-December
1995	<74 KBq/Litre 2 u Ci /Litre			
1996	<111 KBq/Litre 3 u Ci /Litre			
1997	< 74 KBq/Litre 2 u Ci /Litre			
1998		< 74 KBq/Litre 2 u Ci /Litre	1.11 MBq/Litre 30 u Ci /Litre	2.59 MBq/Litre 70 u Ci /Litre
1999	< 3.7 MBq/Litre 100 Ci /Litre			

Table-II --- REACTOR COOLANT SYSTEM BOUNDRY ACTIVITY

Note: Technical Specification limit = scanning of the system at 1 mCi/ litre

Shutdown of the plant at 5 mCi/ litre

Abstract

Even if, due to the development of an implementation program of standardised plant by an unique operator, French Nuclear Power Plants are similar, since several years with an increase of Plant responsibilities, differences in operation and ageing of NPP, it has been noticed an increase in disparities between plants. This is the reason why the French Nuclear Safety Authority develops tools to evaluate the performance of each nuclear plant.

This evaluation is performed every year on the basis of recorded data from the plant, results of inspection and outage supervision and Safety related incidents or events declaration and analysis. The synthesis of the evaluation is performed in a document called "monograph" for each NPP. The first part of this communication will deal with the actual content of these monographs.

These monographs contains both quantitative indicators directly measured and qualitative information about the plant. A good evaluation of the "safety level" of an operating plant, at given time, if it is possible, will require the development of a complex set of indicators which have to be evaluated with an accurate methodology. These methodology is stressed by the necessity to perform this evaluation frequently (every year) with a minimum of resources. The second part of this paper will describe the structure and principles of the set of indicators under development at the French Nuclear Safety Authority.

1. INTRODUCTION

The nineteen French Nuclear Power Plant in operation are equipped with standardised reactors designed by the same vendor (FRAMATOME) and operated by an unique utility (EDF). The different series of reactors are the following.

Power	Series	Number of reactors	Plants
900 MWe	CP0	6	Fessenheim, Bugey
	CP1	18	Dampierre, Gravelines, Blayais, Tricastin
	CP2	10	Cruas, Chinon, St. Laurent
1300 MWe	P4	8	Paluel, Flamanville, St. Alban
	P’4	12	Cattenom, Penly, Nogent, Belleville, Goldfech
1450 MWe	N4	4	Chooz, Civaux

Table 1 : French nuclear power plants

Despite the standardisation of the French Nuclear power reactors, differences have been introduced in the design of systems and equipment. The evolution of safety requirements and risk reduction programs have led to system modifications and improvements in normal and accidental plant operations. The second ten yearly periodic safety review is also a period for the introduction of new modifications improving the safety of the plant. All these improvements are not implemented at the same time in all the reactors, and these result in the finding that at a given time, reactors of a same initial design may have different design, different fuel management and different operating and/or accidental procedures.

Reactor Power	Fuel management	Modification batch	Accidental procedures
900 MWe	CP0 : • Garance (12 months) • Cyclades (18 months)	Lot 93 Lot VD2	Event-based procedures Symptom-based procedures
	CPY : • Garance (12 months) • Hybride MOX (12 months)		
1300 MWe	• GEMMES (18 months)	Lot 93 Lot VD2	Symptom-based procedures Generalised symptom-based procedures

Table 2 : Main differences in fuel management, modification of design and accidental procedures encountered in 900 MWe and 1300 MWe reactor series

One may point out that since several years, the French utility (EDF) has decided to decentralised responsibilities towards nuclear plants and this leads to large differences in organisation.

If one consider that a high level of safety results in a good design and high quality in plant operation, despite the initial standardisation of the French nuclear power plant program, one cannot consider that all the reactors have actually the same level of safety in operation. These are the main reason for which the French Nuclear Safety Authority has decided to develop tool to evaluate individual plant safety performances.

The main objectives of the safety performance evaluation of NPP by the Nuclear Safety Authority are first, to have a brief and complete view of the yearly safety performance and evolution for all the plants based on a standardised method then, to give some rational in the orientation of its safety policy towards operators. The analysis of these indicators will permit the definition of priorities of the nuclear Safety Authority control activities for each plant. The synthesis for each reactors series may also lead to define generic priorities at a national level.

The development of a complete set of indicators which provides a valuable evaluation of the safety performance of a plant is a complex task. Requirements for this need to quantify the status of the nuclear plant for various relevant aspects of safety performance from a technical aspect to operation performance and also impact on the environment. This means that this indicators will include both quantitative information and qualitative indication of plant performance related to established goals.

To be efficient, a second requirement is that this evaluation have to be frequently performed (each year), by nuclear inspectors and with a minimum of resources.

The evaluation is synthesised by Nuclear Inspectors in a document called "monograph" for each NPP. The first part of this communication will deal with the actual content of these monographs. These monographs contains both quantitative indicators directly measured and qualitative information about the plant. The second part of this paper will describe the structure and principles of the set of indicators under development at the French Nuclear Safety Authority, which will improve the contain of these monographs and make the safety evaluation more objective.

1. NUCLEAR POWER PLANT MONOGRAPH

1. Source of the information for monograph

Information for monograph are collected through plant documents, operator reports asked by the Safety Authority like incident reporting, waste production, radioactive and chemical release, observations and finding of refuelling outage supervision, planned or reactive control activities performed by nuclear inspectors and, if necessary, by specific demand to the plant operator.

Acceptation of results is principally performed on the basis of safety requirement reference for quantitative indicators and on legally binding for the quality in plant design, construction and operation for more qualitative parameters.

Monographs are documents elaborated by the safety Authority for its own use. More than a collection of information, they intend to give the point of view of the inspectors on the safety performance of a plant and to deduce priorities for the Safety Authority control activities for the plant. These documents need to be elaborated with quality by using a standardised process inside the safety Authority with a control and verification process. In order to avoid any self-censorship or biased evaluation of the redactor, these documents are not externally diffused.

2. Content of NPP monograph

The monograph contains nine chapters which deals with the various aspects of the plant operation during the previous year. Quantitative information associated with the nuclear reactor safety, general information on the plant and qualitative information on the plant operation are collected . The standardised content of a monograph is given in table 3.

Chapter	Items
1 : Synthesis – conclusion of the monograph	
2 : General information & regulation	• Regulatory authorisations • Safety case • Technical characteristics
3 : Environmental data	• Release • Waste • Radioactive material confinement • Underground water
4 : Plant and plant operation safety	• Status of the three barriers • Protection and safety systems status • Irradiated fuel storage • Operation
5 : Security	• Dosimetry • Work inspection • Pressure vessel regulation
6 : Action of the regional staff of the Nuclear Safety Authority	• Operation Technical Specifications waivers • Incidents • Outage supervision • Inspections
7 : Plant organisation	
8 : Communication	

9: emergency preparedness	

Table 3 : Content of an annual Nuclear Power Plant monograph

The chapter 1 : synthesis, is an executive summary which gives :

- the strong and weak points of the plant found in the various aspects evaluated for the plant
- The results obtained for that previous year by the regional Safety Authority staff for this plant and the priorities for the next year.

This synthesis contain also various data related to the status of each reactor of the plant as :

- Annual electrical power output
- Outage duration
- Radioactive and chemical releases
- Waste production
- Status of the three barriers
- Status of the spent fuel storage pool
- Dosimetry.

Inside the chapter two the technical and safety reference for the various reactor is given (Safety Analysis report, General Operating Rules and technical characteristics as the last integrated modification batch, fuel management).

The Environmental data given at chapter 3, will show and comment the pluri-annual evolution of liquid and gaseous radioactive release for the plant and also for non radioactive products release, the amount of high and medium activity waste which are stored in the spent fuel storage pool, the amount of short live, medium and low activity waste with and without available processing capabilities. The plant radioactive cleanliness is evaluated through indicators as the maximum of activity detected and the number of point having an activity greater than 100 kBq and 1MBq. The activity of underground water and its pluri-annual evolution are also reported and commented.

The forth chapter deals with the safety of the plant and plant operation. Quantitative indicators permit easily to estimate the status of the three barriers and to comments on their time evolution, Table 5 gives the set of indicators which are used.

	Indicators
First barrier	• reactor cooling system activity • Number of unsealed fuel

	assemblies
Second barrier	• Primary – Secondary leakage rate • Primary leakage rate • Number of cracked SG tubes
Third barrier	• Leakage rate • Containment penetrations leakage rate

Table 4 : Barriers status indicator

For the other elements of the evaluation, the actual monographs doesn't refer to specific indicators. The status of the protection and safety systems are evaluated qualitatively through information obtained during inspections, plant reports and maintenance operations. The operation evaluation is commented taking into account the way the operator takes into account the safety requirements mainly those resulting of the ministerial order on quality in design, construction and operation of Nuclear Installations.

The sixth chapter will evaluate the plant through the Safety Related Incidents, it contains some quantitative indicators as :

- the number of incidents for each reactor of the plant
- the number of incidents with radioactive release
- the mean delay for the incident declaration and the analysis report and the distribution of delays
- the number of incidents which have been found by the Safety Authority
- the number of Safety Related Events reclassify in Safety Related incidents.

The analysis of significant incidents takes into account the reactor status, the distribution of the events according to a list of declaration criteria and the concerned safety functions. More qualitative features related to Safety Relevant Incidents as relationships with the plant staff, quality of plant analysis and the respect of commitments taken by the plant staff after an incidents, are also included in the analysis performed by Nuclear Inspectors.

1. DEVELOPMENT OF SAFETY PERFORMANCE INDICATORS

As seen in the previous paragraph, the actual safety evaluation of Nuclear Power Plant performed by the safety Authority will include both quantitative indicators and qualitative analysis to be an efficient tool to identify strong and weak points of NPP and for the improvement of the control strategy of each plant but also at the national level for generic findings.

In order to improve this process DSIN is developing a new set of indicators allowing to better take into account qualitative information from the plant and to associate to each item of the evaluation a quantitative indicator in order to obtain overall safety indicator for the plant.

The structure of monographs will also be slightly modify (table 5).

Chapter	Items
1 : Synthesis – conclusion of the monograph	
2 : General information & regulation	• Regulatory authorisations • Safety case • Technical characteristics
3: Overall safety Indicator	
4 : Plant safety	• Status of the three barriers • Protection and safety systems status • Irradiated fuel storage • Operation
5 : Safety of plant operation	• Organisation • Technical engineering and transverse technical support • Quality system • Human Factor & organisation consideration in operation • Training, enabling and competency development
6 : Environmental data	• Release • Waste

	• Radioactive material confinement • Underground water
7 : Security	• Dosimetry • Work inspection • Pressure vessel regulation
8 : Action of the regional staff of the Nuclear Safety Authority	• Operation Technical Specifications waivers • Incidents analysis by the regional staff • Outage supervision • Inspections
9 : Communication	
10: Emergency preparedness	
11 : Other	

Table 5 : Proposal for the content of an annual Nuclear Power Plant monograph

Most of the quantitative indicators used in the previous monographs are still used in this improved document. Indicators for all the items of the chapter 4 to 7 are under development, this paper will describe mainly the proposed indicator evaluation methodology for chapter 4 and 5. The overall safety operator will results of a combination of all the individual indicators and will describe the Safety Plant Status according to five level

- the safety of the plant is not acceptable
- the safety of the plant is low but acceptable with improvements
- the safety of the plant correspond to the standard level of safety
- the safety of the plant is good
- the safety of the plant is excellent

1. : Indicators for the plant safety

If it is possible to measure different parameters characterising the status of any equipment important for the safety, these measurements are often insufficient to take into account important parameters as the knowledge of construction deficiency, insufficient equipment maintenance program, modifications, or effect of ageing on components. For instance if it is possible to characterise the integrity of a barrier to a measured leakage rate, this is not sufficient to characterise the increase of risk due to the status of the barrier. It is the reason why it is necessary to develop a set of indicators which permit to evaluate the real status of material components and equipment and to evaluate the importance of indications or non conformance on the overall risk. For such evaluation results of Probabilistic safety analysis are required. The methodology which is adopted for these indicators are based on a two level analysis.

- level 1 : qualitative – existence or not of indications relevant for the safety
- level 2 : quantitative – combination of the indications by taking into account their contribution to the frequency of an initiator

The following table resume this methodology for the integrity evaluation of the second barrier.

Material	Indications	Yes/No	Overall importance calculation for the risk
Vessel	Δ RT_{NDT}		
	Shell ring coating defects		
	Nozzle coating defects		
	Penetration tubes defects		
	Head adapter defects		
	Fabrication anomaly		
Steam generators	Inconel I 600 MA		
	Inconel I 600 TT		
	Circumference cracking		
	Fast secondary side corrosion		
	Fabrication anomaly		
Pressuriser	Delayed control of lower head		
	Fabrication anomaly		
Primary coolant pumps	Old thermal barrier		

	Casing gasket leakage		
	Casing-nozzle welding defect		
	Fabrication anomaly		
Primary coolant piping	LBM		
	Elbow sensitive to ageing		
	Fabrication anomaly		
Auxiliary piping	"Farley Thihange " modification		
	Line tap modification		
	Fabrication anomaly		
Valves	Presence of martensitic steel		
	Fabrication anomaly		
	In-service anomaly		
Safety devices			
TOTAL			

Table 6 : methodology for the second barrier status indicator evaluation

1. **: Indicators for safety of plant operation**

Plant operation safety is evaluated through the evaluation of the efficiency of the organisation and the ability of plant employees to operate safely the reactors. Indicators for plant operation safety use a five level scale. Each level is defined by a minimum of requirements which have to be satisfied. The requirements of lower level have be also fulfilled.

Indicator

Level 1	Level 2	Level 3	Level 4	Level 5
Non acceptable	Acceptable	Standard	Good	Excellent

1. The plant doesn't perform any action to develop the related item
2. : The plant has defined a policy and an organisation to implement the item
3. : The policy is implemented in all departments
4. : The policy is implemented by all the employees
5. : The policy is included in an iterative process periodically evaluated

Table 7 : generic definition of indicators for plant operation safety

The requirement to evaluate the level of performance of the plant is that the rating between the first to the forth level should be easily obtained by using information collected by routine control performed by inspectors. Nevertheless, the rating between the 4th and 5th levels ask for a larger knowledge of the plant and the related domain.

The following tables describe for various indicators the requirements for each level.

Item	Level	Requirements
Organisation, Safety management	1	• Safety is not a priority of the plant
	2	• Safety is a priority clearly notified by the plant directorate

	3	• A process for the acceptation, the analysis and the implementation and the evolution of safety related prescriptions is implemented • The presence of the management staff at the operation level is observed
	4	• Decisions are taken at the good level of responsibility and are tracked
	5	• The presence of the management staff at the operation level is a natural component of management practices

Table 8 : Indicators for Organisation and safety management evaluation

Item	Level	Requirements
Quality system	1	• The plant has no Quality Manual and specific organisation
	2	• The plant has a Quality Manual. • Commissions and responsibilities are defined for all departments and employees. • The Quality department is operational. • A process for non-conformance treatment is defined
	3	• Quality documents are implemented in all departments • The Quality department is sufficiently designed and its audit activities are planned and are deduced from a defined policy • Phases and content of technical control are clearly identify in operational documents

	4	• The verification program of the Quality Department covers all the requirement of the central staff and the plant • The Quality department is an effective support to operational department • The employees adhere to the necessity for technical control and this control is effective
	5	• Practices for verification and non-conformance detection allow an efficient correction process • Each employee is involved in the quality obtaining for its daily activities

Table 9 : Indicator for Quality System evaluation

Item	Level	Requirements
Human factor and organisation	1	• Human factor or organisation items are systematically ignored in the technical framework
	2	• The plant has a Human factor policy and a structure is involved for the consideration of human factor and organisation in the social-technical framework • At least, ¼ of incidents having a human origin have a deep human factor analysis • The human factor is not restricted to Human error
	3	• The human factor structure has human sciences competencies • The analysis of incident is not the main work of the human factor structure • The human factor structure is implied in social-technical context modifications

	4	• The human factor structure is sufficiently designed for its missions • Human features are taken into account for each modification of the social-technical context • A deep analysis is performed for incident with human factor implication and correctives measures are taken
	5	• The plant implement a voluntarist and controlled human factor policy • The human factor structure is a support for the management staff and operational departments • Dispositions are taken in order to guarantee the efficiency of the process for human factor and organisation features

Table 10 : Indicators for Human factor and organisation consideration evaluation

Item	Level	Requirements
Traning, enabling and competency improvement	1	• Periodicity of training is not respected, the enabling process is insufficient
	2	• The enabling process is performed on time and tracked, it takes into account the realisation of periodic enabling training courses
	3	• The evaluation of training needs is performed for each people • Each trainee is evaluated after an enabling training course • The plant define a policy for the competency development • The plant has a sufficient training structure according to its needs

	4	• Each profession has a competency reference. • Competency of employees is evaluated by the hierarchy • Employees training and competency are tracked • The adequacy of training courses is evaluated
	5	• Enabling is given according to the employees competency • The management participate to the development of employees competency • Activities are shared between employees according to their competency • Dispositions are taken in order to guarantee the efficiency of the competency improvement process

Table 11 : Indicators for Training, enabling and competency improvement evaluation

4 : CONCLUSION

The evaluation of the Safety performance of Nuclear Power Plants is a necessity for Safety Authorities in order to provide the best objective as possible measure of the plant compliance with the safety requirements. It allows also to focus the attention on the main weakness of the plant and to improve the control activities of the authority.

Despite the standardisation of the French NPPs, the French Safety Authority has a need to annually evaluate each power plant. This is done by the mean of monograph which synthesised the main safety features of the plant, both quantitative measurements and qualitative elements are considered to obtain a judgement on the safety of the plant and to define priorities for the control policy.

Nevertheless, most of measurements cannot be interpreted as a real measure of the impact on the reactor safety of the system, equipment or component status. A more detailed process which include the main deficiencies encountered, maintenance and modification programs is under development. The construction indicators on plant safety will need the use of PSA.

The evaluation of the safety operation performance of a plant will take into account various items which are difficult to quantify as organisation, quality systems, human factor consideration or competency improvement. For this, the French safety authority is developing a set of indicators based on five level scale. The evaluation of an indicator can be performed quite easily by inspectors by comparing the status of the plant with a set of requirements.

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

DEVELOPMENT OF SAFETY PERFORMANCE INDICATORS IN JAPAN

Jun Tanaka

Nuclear Power Engineering Corporation, Safety Information Research Center

Fujita Kanko BLDG. 8F, 17-1, 3-Chome Toranomon, Minato-Ku, Tokyo 105-0001, Japan

Tel ;+81(3)3435-3406 Fax:+81(3)3435-3410 E-mail: j-tanaka@nupec.or.jp

Abstract

In Japanese safety regulations to operating plant, the regulatory authorities utilize two types of measures. One is the direct regulation such as periodical inspection through which they inspect the electrical systems to verify that the function and performance of important safety related equipment conform to the approved construction plan and the technical standards, the other is audit type regulation such as preservation inspection through which they inspect the utilities to verify that plant operation conform to the safety preservation rules.

NUPEC are studying a effective set of overall operational performance indicators for the audit type regulations in the frame of present safety control regulation system.

We have classified various performances required by regulation, and developed the performance indicator framework.

We have started the investigation of the evaluation methods and the judgment criteria. In this paper, I report the result of completed study and status of ongoing item.

1. Status of Performance Indicators and Regulations at Operational Stage in Japan:

Safety Information Research Center has performed, periodically in every year, the overall performance analyses of the nuclear power plants in Japan to evaluate the safety and operational performance of the plants in the aspects of the capability factor, incidents, radiation exposure, and radioactive waste. The analyses have been performed with the performance indicators on the capability factor, frequency of unplanned shutdowns, unplanned automatic scram frequency during critical, forced shutdown frequency per critical hours due to equipment failure, the number of incident and failure reports, the number of unexpected releases of the ECCS waiting conditions, total dose equivalent and volume of the low level solid waste, and have been reported and distributed to the regulatory authorities, related organizations and utilities including foreign organizations. The performance indicators were developed by Safety Information Research Center with reference to WANO and/or INPO's overall performance indicators.

Regarding the performance indicators, the regulatory authorities in Japan have no intention to use the performance indicators or safety indicators directly as a tool of regulations for regulating the operational control or safety assurance at present, because they are ensuring the plant safety by hearing and by receiving report from each utility with regard to the safety programs, operational controls for specific unit, and the periodical inspection results etc, in detail from planning to the results thereof.

The safety regulations to the operating plants has been performed with direct regulations enforcing utilities to have the obligation to take the periodical inspection for the plant equipment and function by the regulatory authorities accompanied with some audit type regulations (ex. audit of the self controlled safety preservation activities).

Under these situations, the critical accident occurred due to the deviation works from Government licensed procedures at JCO uranium manufacturing facility. The accident enforced to strengthen the nuclear safety control system in Japan, and the Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors was revised. Then, the inspection system relating to conform the safety preservation rules was established and nuclear safety inspectors from the regulatory administrative office have been stationed in all nuclear facilities to inspect the conformance of safety preservation rules etc.

We are investigating a set of overall operational performance indicators to develop the effective evaluation method for the safety preservation activities in the audit type regulations.

Compiling the investigation results, we are aiming at establishing the appropriate tool for the audit type safety control regulations by developing the effective use of performance indicators and by clarifying the applicable scope of utilization, and by setting the overall operational performance indicators available for future use in Japan.

2. Investigation Procedures

The establishment of the set of overall performance indicators should be consistent with the regulation system in Japan, considering that it is used by the regulatory authorities, and aim at establishing the stepwise system.

The investigation procedures, accounting for the above, are shown as follows, and Fig. 1 shows the actual time schedule:

(1) Investigation and reflection of domestic and foreign status:

Confirmation of the Japanese safety regulations and safety control system (confirmation of the existing law system, and the existing operational safety regulations);

Investigation and reflection of the domestic and foreign trend;

(2) Classification and systematization of performances on safety control in Japan:

(3) Development of the overall performance indicator framework and extraction of the expecting performance indicators for each performance:

(4) Investigation of the performance evaluation methods by using the performance indicators and judgment criteria:

(5) Trial evaluation by using the performance indicators, evaluation of the applicability effectiveness to safety regulations, and extraction of the problems:

(6) Development of the overall performance indicator set (draft) applicable in Japan:

(7) Extraction of the subjects to be investigated for performance indicator application in audit type safety regulation system:

The investigation is performed from 1999 through 2001.

3. Confirmation of Japanese safety regulations and safety control system

3.1 Confirmation of Japanese safety regulations

Fig.2 shows the basic safety laws and regulations system in Japan.

The intentions of these laws and regulations are rearranged with the viewpoint of nuclear power plant operational safety regulations as follows:

	Investigation items for the set of overall performance indicators	Time schedule		
		1999FY	2000FY	2001FY
(1)	Investigation and reflection of domestic and foreign status			
(2)	Classification and systematization of the performances on safety control in Japan			
(3)	Development of the overall performance indicator framework			
(4)	Extraction of the performance indicators for each performance (Except for the emergency measures and safety during shutdown period)			
(5)	Extraction of the performance indicators for the emergency measures and safety during shutdown period			
(6)	Investigation of the evaluation method and judgment basis			
(7)	Trial evaluation (including PSA evaluation except for shutdown period)			
(8)	Evaluation of the applicability effectiveness to the safety regulations, and identify the problems to be solved			
(9)	Extraction of the performance indicators for each reactor type, and trial evaluation			
(10)	Trial evaluation applying PSA evaluation for shutdown period			
(11)	Development of the overall performance indicator set (draft) applicable in Japan			
(12)	Extraction of the subjects to be investigated for performance indicator application in audit type safety regulation system			

Fig.1. Time Schedule

□The Law for the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors : Organizational management and operational safety insurance to protect public safety.

□Electricity Utilities Industry Law : Ensure the public safety, and prevent the environmental pollution as well.

□Industrial Safety and Health Law : Ensure the safety and health of occupational radiation workers.

Special Law for Nuclear Disaster Measures : Protect the life, body and property of the people from the nuclear disaster.

These laws and regulations are rearranged with the standpoint of the plant operation as follows; the safety regulations for nuclear power plant in Japan are mainly controlled by nuclear material safeguards rule, safety preservation rules, and periodical inspection. And recording and reporting, the qualified reactor engineer, and the responsible engineer for operation are covered by the safety preservation rules.

There is "Comprehensive Investigations of Safety Preservation Management" in order to perform the audit on the safety control system by the regulation side. In addition, the periodic safety review, and the periodic check performed by utility parallel with the periodical inspection are the regulatory measures to review the results of utility review by the regulation side.

Inspection system relating to conformability of safety preservation rule was established this year and the nuclear safety inspectors from the regulatory office are stationed in all nuclear facilities to inspect the conformance of safety preservation rule etc.

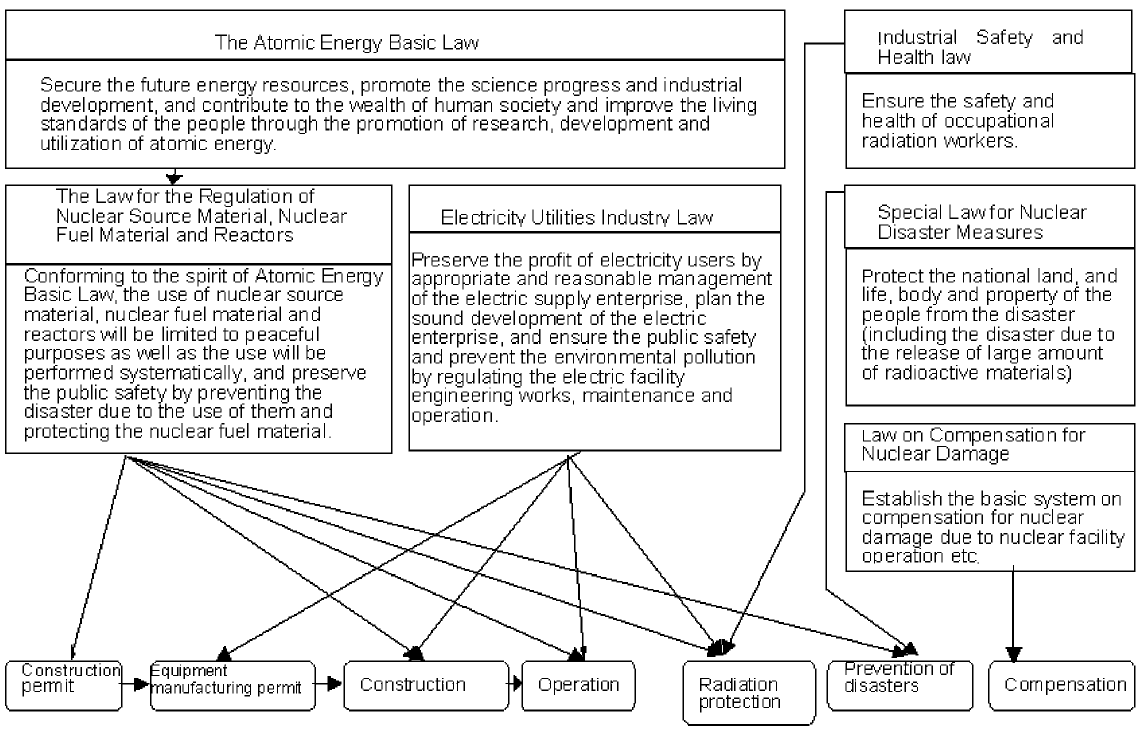


Fig. 2 The laws and regulations system for nuclear power station in Japan.

The operational safety control measures discussed above are shown in Fig.3. which includes items regulated by laws and regulations, items not regulated by laws and regulations but reportable to the Government, and items positioned to be the audit methods for control measures.

3.2 A general view of major operational safety regulations

Concrete safety control measures for the operating nuclear power plants are as follows:

Set and obtain the approval of the nuclear material safeguards rule as measures to protect from specific nuclear material;

Establish the safety preservation rule on concrete operation methods, operational limits and conditions, and obtain the approval;

Nominate the qualified reactor engineer to supervise the plant operational safeguards, and notify the competent Minister;

Station the engineer responsible for operation who has the right and responsibility to judge and cope with emergency;

Undergo inspection (periodical inspection) on reactor facilities at regular intervals by the competent Minister;(Electricity utilities industry law specifies periodical inspection)

Prepare records for the reactor and other nuclear facilities operation, and keep them for a specified period of time. An abnormal event occurred at the nuclear facilities should be notified immediately, and the accident /failure conditions and the measures to cope with the accident should be reported as soon as possible.

Outline of these laws and regulations are as follows:

a. Safety preservation rules:

Plant licensee is obligated to establish the concrete measures to ensure safety preservation and protect specific nuclear material for each nuclear facility, prescribe in the safety preservation rule, and observe it. Ruling side verify that the licensee is preparing the appropriate safety preservation system through the approval of the safety preservation rules, and the nuclear safety inspector inspects the observation status. The safety preservation rules is prescribed to regulate the operation related items, walk down and check related items, and items to deal with them by the competent Ministerial ordinance.

b. Periodical inspection:

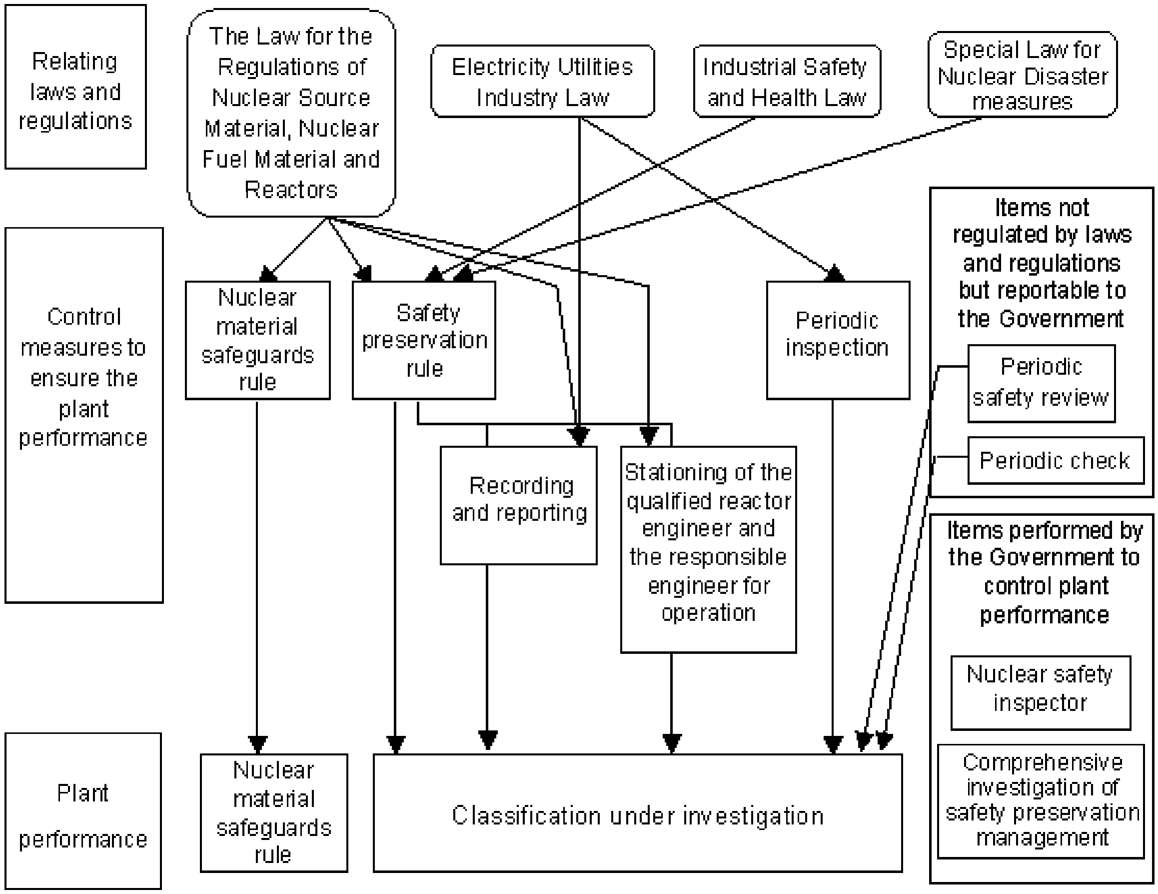


Fig. 3. Safety regulations and control measures

The reactor constructor is obligated to undergo the inspection at regular intervals by Ministry of International Trade and Industry (MITI) for the important electric equipment to ensure the safety preservation and smooth supply of electric power of the reactor power station. Periodical inspection is the inspection periodically performed by the Government on the Electricity Utilities Industry Law. Electric equipment to be inspected on nuclear facilities are steam turbine, power reactor and its related facilities. Time of inspection for steam turbine is specified to be the day after one year and not over 13 months from the start of operation or preceding inspection completion date, and for power reactor and its related facilities is specified within 13 months from the start of operation or the preceding inspection completion date. Inspection term is specified from the date of disconnection from grid for inspection to the completion date of full load test.

Main purposes of the inspection are:

a. Verification of the reactor coolant pressure boundary;

b. Verification of the integrityof fuel assembly;

c. Verification of the functions of the major safety related equipment by overhaul inspection;

d. Verification of the functions of reactor shutdown system;

e. Verification of the overall functions of nuclear installations:

During the period of periodical inspection, utilities perform their own inspection as a part of self-safety preservation. The purposes of this inspection are the prevention of accident and/or failure, the prevention of recurrence, and the prevention of performance decline by deterioration due to aging, and utilities perform the comprehensive check and inspection using plant overall programs on the operating conditions of each facility and past operation and maintenance experiences. Furthermore, based on the basic concept of maintenance for nuclear facilities in Japan as preventive maintenance which prevent the defective conditions by taking previous measures before the failure occur, main facilities are inspected, maintained, improved, or replaced on regular interval basis or operating condition basis by planned maintenance procedures. These results are reported to the Government with the regulatory inspection results.

c. Others:

Besides the laws and regulations based safety control, the comprehensive investigation of safety preservation management, as regulatory safety control, is prepared to verify the status of operation control, safety preservation control, and self safety preservation. In addition, safety and reliability enhancement have been evaluated by requesting reactor constructors the periodic safety reviews.

4. Classification and systematization of performance on safety control in Japan

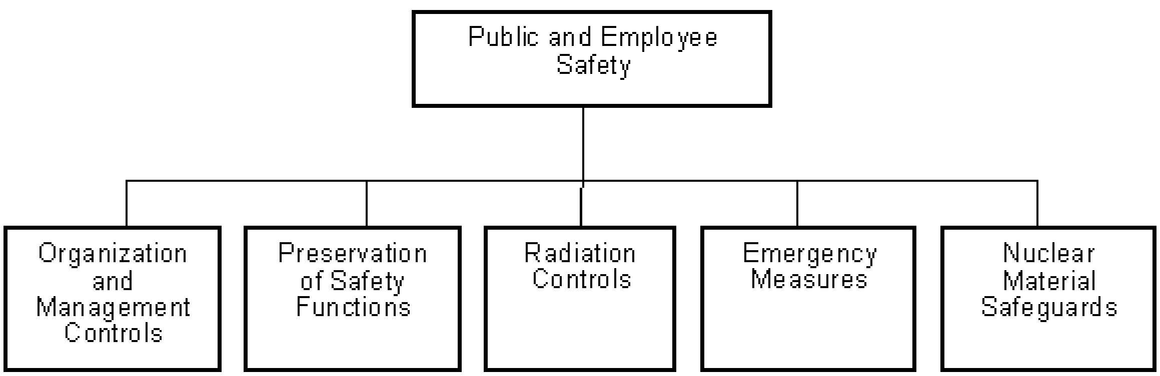


Fig.4. Basic performance indicator framework

In order to establish the stepwise regulatory system, we performed the development and classification of various performances so as to apply the regulatory control based on the requirements of "Control Measures to Ensure Plant Performance". Although basic purpose of regulation for the operational plant safety control is "Public and Employee Safety" based on laws and regulations system, it is considerable to classify into "Organization and Management Controls", "Preservation of Safety Functions", "Radiation Controls", "Emergency Measures", and "Nuclear Material Safeguards" as a stepwise major classification to achieve the purpose.

Definitions of the classification are as follows:

a. Organization and Management Controls□Performances relating to the organizations based on reactor and fuel safety, management of procedures etc. Preparation of procedures, education and training implementation etc, are included in this category.

b. Preservation of Safety Functions: Performances relating to preserve facility functions necessary to ensure the reactor and fuel safety. Specific management aimed at facility maintenance (inspection, check etc.) is included in this category.

c. Radiation Controls: Performances relating to occupational radiation workers and public radiation protection on organization, facilities, and manpower.

d. Emergency Measures: Performances relating to abnormal, accident, and emergency measures.

e. Nuclear Material Safeguards: Performances relating to nuclear material safeguards.

We had confirmed that the above classification was applicable for extracted performances in our safety control regulations.

Fig.4 shows the basic overall performance indicator framework in Japan. "Nuclear Material Safeguards" is excluded from the investigation subject, since that performance is not open to the pubic and difficult to investigate. Development of "Emergency Measures" will be performed this year.

In the next step, we had investigated the applicability of performance indicators to the following specific characters that are considered to be possessed as safety regulations controlled in Japan

- Quantification: Performance should be quantifiable. (Deterministic/risk information application)
- Measurement objectivity: Performance should be measured objectively. (Plant specific objectivity)
- Evaluation objectivity: Performance should be evaluated objectively.(Plant common objectivity)
- Representativity□Performance representing indicators are to be defined.
- Accuracy, Verifiability: Data should have sufficient accuracy and verifiability.
- Corrective Action Feasibility:Effective and appropriate corrective action should be picked up by auditing as performance indicators.
- Predictability: Potential safety defectives should be predicted by auditing performance indicators, and corrective action should be indicated timely.
- Easy Acquisition:Necessary data should be taken easily.
- Easy Handling: Data rearrangement and evaluation should be handled easily.

Table.1. shows the results of applicability investigation.

The results indicate that there are many safety control items which are not applicable or only partially applicable, and large scope of area of them are should be depend on inspection of safety preservation performed by nuclear safety inspectors, periodical test, comprehensive investigation of safety preservation management, or audit of periodic safety reviews etc.

Table.1. Investigation result of performance indicator applicability

Feasibility of performance indicator applicability	Number of items	Deterministic theory or Risk Evaluation theory applicability	Deterministic Theory	Risk Evaluation theory
Yes	88	Both feasible	8	
		Either feasible	50	30
		Total	(58)	(38)
No	111		141	161
Total	199			

Notation: In this table "feasible" includes partially

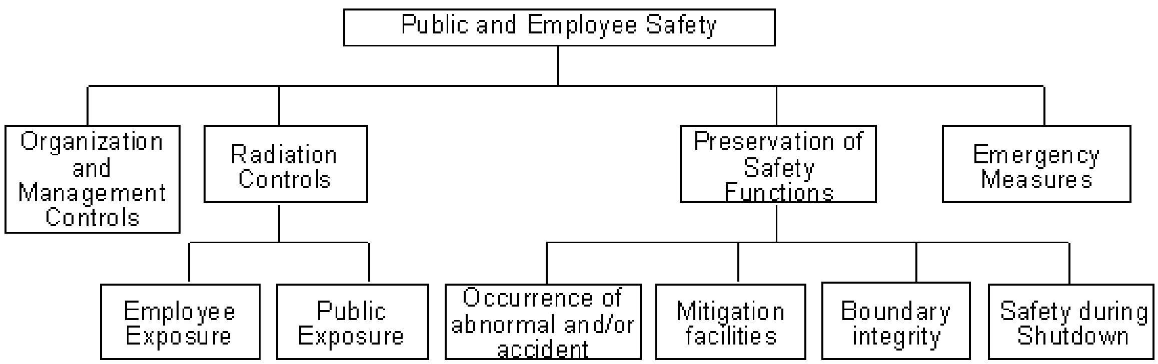


Fig.5. Overall performance indicator framework.

5. Development of the overall performance indicator framework

In developing the overall performance indicator framework, "Preservation of Safety Functions" and "Radiation Controls" are divided into more detail classifications since they have many corresponding performances.

a. Preservation of Safety Functions: With standpoint of an essential foundation of the concept of defense in depth, 3 categories were adopted:

- Occurrence of abnormal events and/or accidents;
- Mitigation facilities;
- Boundary integrity.

In addition, as "Safety during Shutdown " also becomes to be regard as important, it is added to the developing items. "Safety during Shutdown" will be investigated this year reflecting the changes of safety preservation rules now under review, because under present safety control regulations "Safety during Shutdown" performance is not detailed as in operation.

b. Radiation Controls: Radiation Controls will be developed in "Employee Exposure" and "Public Exposure", because "Employee Exposure" is differently dealt with The Law for the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors, Electricity Utilities Industry Law and Industrial Safety and Health Law.

Fig.5. shows the overall performance indicator framework.Fig.6. and Fig.7. show the results of investigation on the proposed performance indicators applicable for

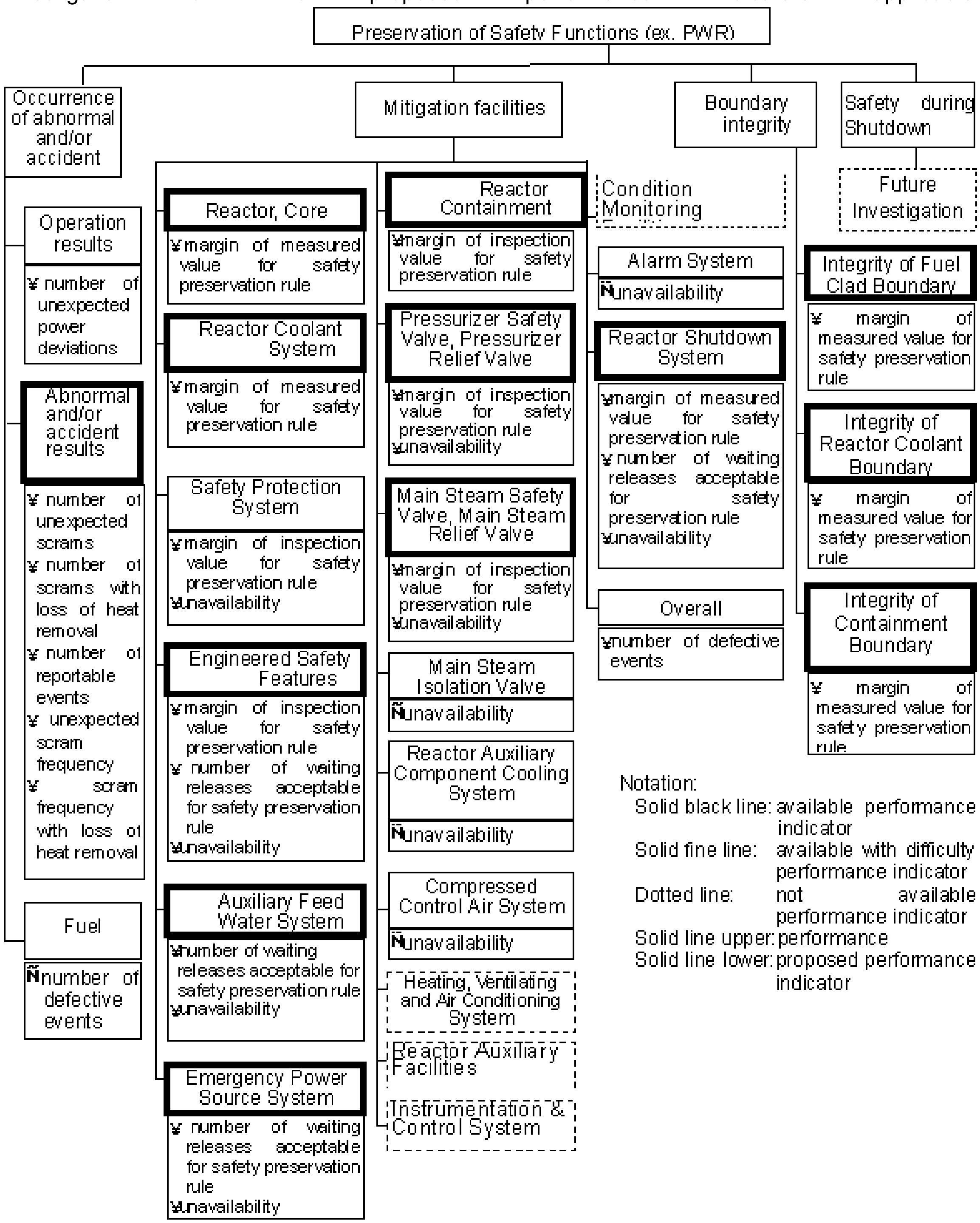


Fig.6. "Preservation of Safety Functions"(ex. PWR) performance and corresponding proposed performance indicators

performance with developing into small classifications. Table.2. shows the classification of performance and the number of relating indicators. Fig.8. shows the overall performance indicator framework filled with the proposed performance indicators.

As a result of above discussions, it can be say that too many performance indicators based on deterministic theory are extracted, because, risk information application methods do not take a definite form in Japan. While, NRC reduced sharply the number of performance indicators especially for the mitigation facilities by representing with risk information applied performance indicators. That is the main difference between NRC and us. Too many indicators complicate the measures and evaluations, and the inclusive evaluation and reduction of umber of indicators are necessary hereafter.

It is possible to get hold trends for some specific items to apply the performance indicators in safety control regulations at this stage, which make clear the points of audits and expect to enhance accountabilities.

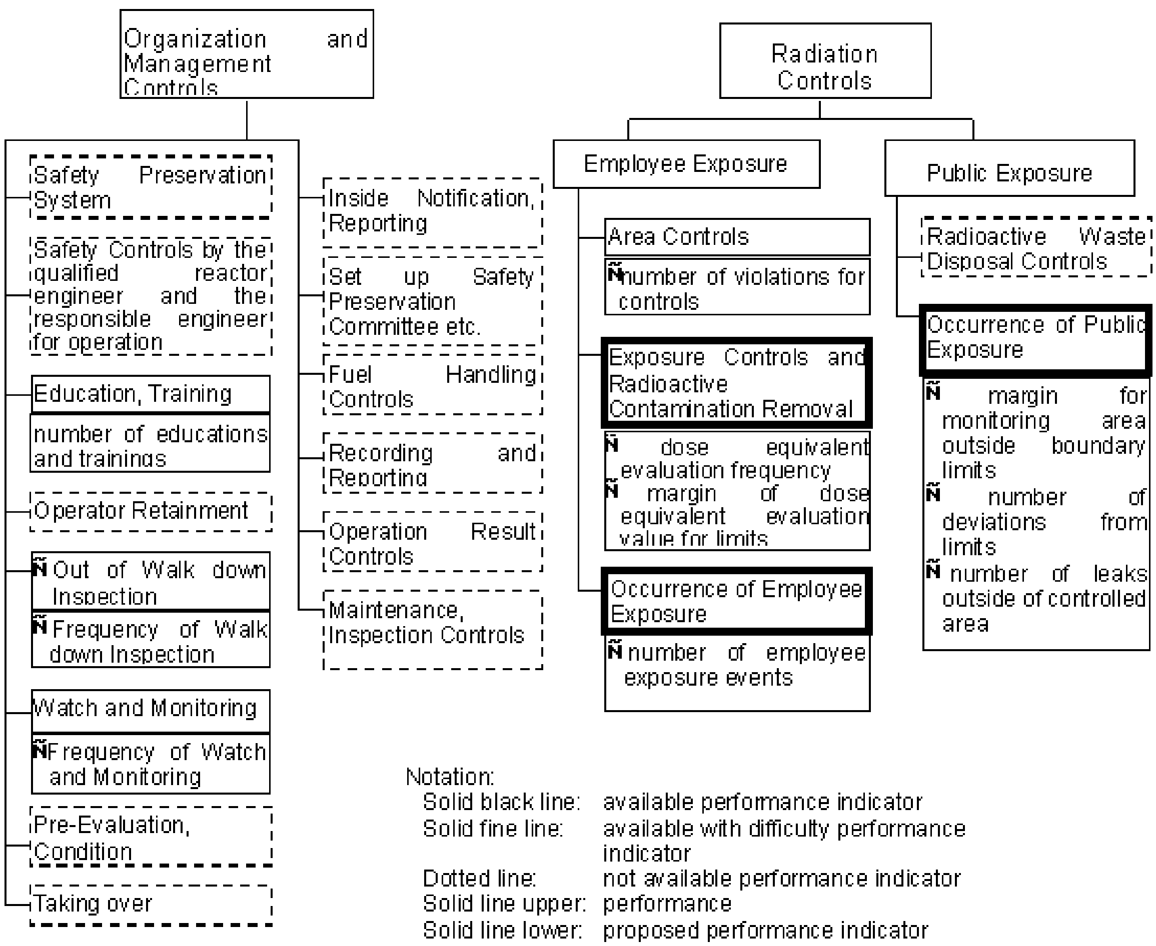


Fig.7. "Organization and Management Controls", "Radiation Controls" performance and corresponding proposed performance indicators

In above discussions, although we accounted for both indicators based on the deterministic and risk information theories, some indicators on deterministic theory may be represented by occurrence frequency of causing events or unavailability applying risk information. In addition, applicable indicators may be changed depend on the way of application of risk information. For example, in case of "Occurrence of Abnormal and/or Accident", we considered the performance indicators on occurrence frequency for risk evaluation cause events, but depend on the classify method, it is available to take off the cause events with loss of heat removal as NRC, or deal with all causes inclusively etc. In this investigation, we will follow the way of NRC but reevaluation on risk information application methods will be required in the future.

6. Investigation of the performance evaluation methods by using performance indicators and the judgment basis

Regarding the evaluation methods to audit by using performance indicators, we have started the investigation of the evaluation methods more appropriate for Japanese present regulation system, including the evaluation method feasibility to apply the quantitative threshold value setting, or trend using method etc.

Although present safety control regulation system is basically deterministic theory basis, we will include the investigation on probabilistic approach applying method using risk information at this stage, problem extraction for application, performance indicator evaluation method using risk information, and the trial setting evaluation of the concrete judgment basis such as system unavailability etc.

(1) Basic concept using performance indicators to regulations:

Our immediate plan is to establish the measures to effective use of performance indicators in the frame of present safety control regulation system. Overall safety control regulation system applying the performance indicators will be evaluated in next stage.

Practically, the performance indicators may be used as supportable indicators in the audit to the utilities on the safety control activities such as safety preservation rules. The effective practical applying methods in the area of supportable indicators are considerable as follows:

Apply draft 1: Evaluation using trend

Determine the control conditions are on the better or worse trend to make clear the significant points to be audit.

Apply draft 2: Evaluation comparing with the other plants (or plant mean values).

Determine the better or worse performance comparing with the other similar plants to make clear the significant points to be audit.

Table.2. The Classification of Performance and the Number of Relating Performance Indicators

Purpose	Public and Employee Safety													
Performance, major classification	Organization and Management Controls		Radiation Controls				Preservation of Safety Functions							Emergency Measures
Number of performances	39		37				111							12
Performance, intermediate classification			Employee Exposure		Public Exposure		Occurrence of abnormal and/or accident		Mitigation facilities		Boundary integrity		Safety during Shutdown	
Number of performances			24		13		22		77		10		2	
Number of performances after groupings (minor classification)	14		3		2		3		19		3		Not yet investigated	Not yet investigated
Number of performances feasible for performance indicator applicability	3		3		1		3		15		3			
Number of performance indicators	Deterministic theory	Risk theory	Deterministic theory	Risk theory	Deterministic theory	Risk theory	Deterministic theory	Risk theory	Deterministic theory	Risk theory	Deterministic theory	Risk theory		
	3	0	4	0	3	0	5	2	13	11	3	0		

Apply draft □3: Evaluation of safety margin for quantified limits

Evaluate the safety margin for quantified limits of performances to make clear the significant points to be audit.

We discuss the feasibility of each performance indicator for safety control regulations on the way of applications in the followings:

(2) Performance indicator evaluation methods based on the deterministic theory:

One of the ways to evaluate the performance indicators based on the deterministic theory is to set the stepwise target values as far as possible noticing the safety margin of regulatory limits such as the limits for safety preservation rules, and provide useful information to the safety control regulations. In this case the target values are not the threshold values to tighten control, but have the object to provide information to the safety controls (ex. to support the safety inspector to find the significant items during the audit etc.). Therefore the area exceeding the limits of regulations is excluded from the subject of performance evaluation.

Take "Margin of dose equivalent evaluation value for limits " as a practical example. The limitation value of dose equivalent for the employee is specified as 50 mSv/yr or under, but it is advised to maintain as

Notation:
Dotted Line: Future Investigation Items
Double Frame : Risk Information Application Items

low as

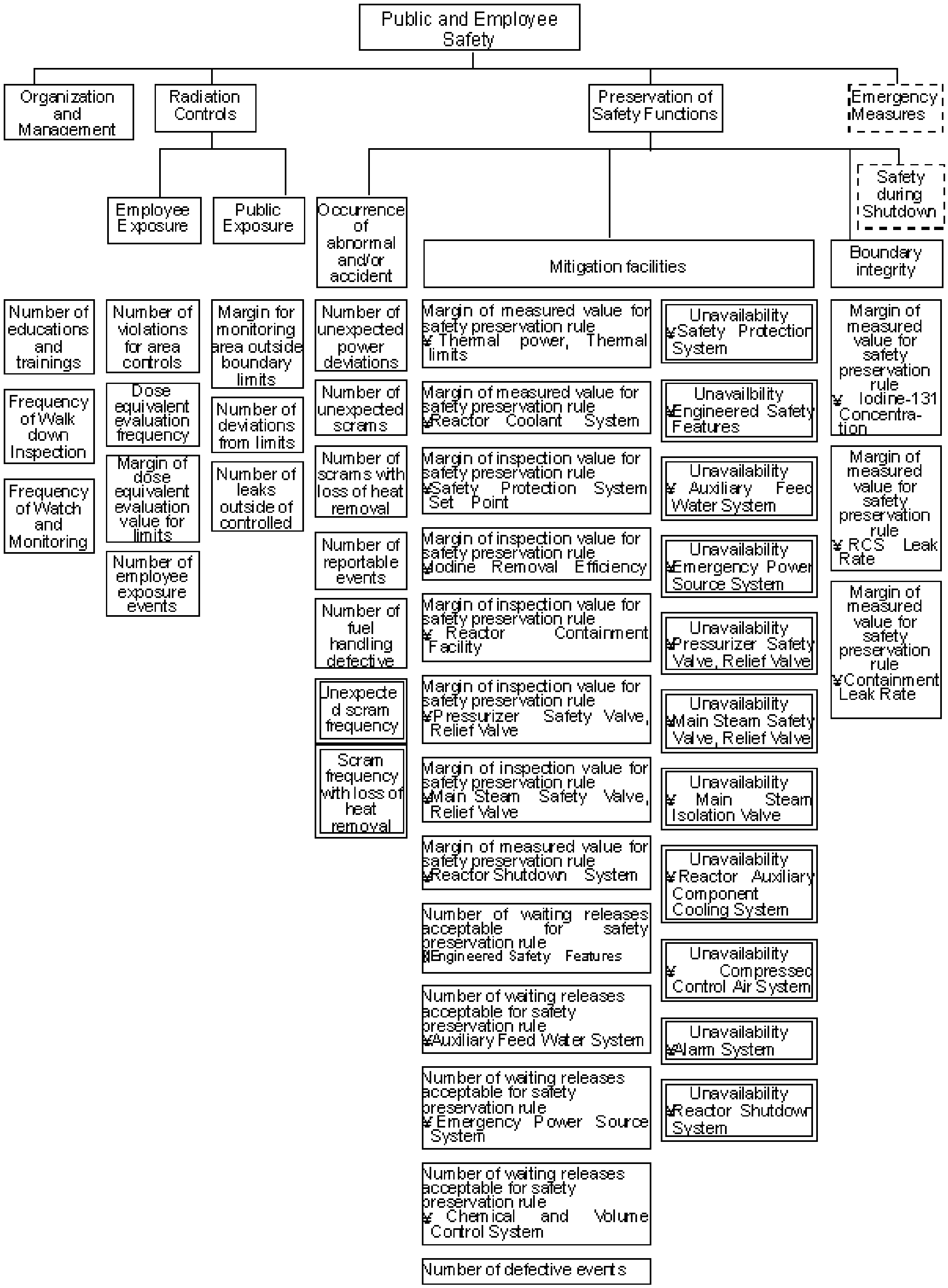


Fig.8. Example of Overall performance indicator framework filled with the

reasonably achievable (ALARA). The dose equivalent mean values in Japan reduced from 3.5 mSv/yr in 1980s to 1.5 mSv/yr in 1990s. In this performance examples having qualitative principle, it is preferable to set target value based on the mean value of interior plants in addition to the limitation value.

While, for evaluation using trend, evaluation value is generally concentrated at the time of the periodical inspection and affected by the procedures and terms of the inspection. That requires specific evaluation for the trend to take into account of these worsening causes. Therefore, it is necessary to set the target values depend on the characters of the performances to evaluate the performance indicators as in this examples.

(3) Performance indicator evaluation methods applying the risk information:

The core damage frequency is the first expected risk applying information as the results of Probabilistic Safety Assessment (PSA) for the inner events during power operation that is reported in the Periodic Safety Reviews (PSR). In addition, the containment failure probability is also reported in PSA

Even in case of applying the core damage frequency, it is not clear that the evaluation will be appropriate, because the safety control target and its base safety target values are not established in Japan, and on safety margin for the interior plant mean value indicate the different PSA results between plants.

Regarding the trend evaluation, it is considered that the core damage frequency will have dominant tendency to change temporarily due to the short term waiting release or cause events occurrence, and appearance of worse trend for long term are not considered practically.

As the applicable evaluation method in Japan, it can be considered the method to perceive the degree of degradation from the base core damage frequency. In this case, the base core damage frequencies are available from the Individual Plant Examination (IPE) performed every 10 years in PSR, and have different values per plant.

Based on the above discussions, the method that evaluates the performance indicators adopting risk information and applies to safety control regulations in Japan has the following problems:

1. Is it sufficient to adopt only the core damage frequency as a base indicator?
2. In case of adopting the core damage frequency as a base indicator, how to set the target value? And how to determine the evaluation term?
3. How to determine the unavailability evaluation extent for the facilities on the performance indicators?
4. How to deal with the facilities differences between plants?
5. Is it possible practically to evaluate periodically the unavailability, or to report? (It is not reportable at present.)

The feasible evaluation model at this stage will be evaluated hereafter considering the above problems.

(4) Investigation of the evaluation method of proposed performance indicators and judgment criteria

Based on the fundamental evaluation methods, extracted performance indicators are classified as follows at the standpoint of the performance evaluation methods. Fig.9.shows the results.

Type□A: Existing limitation values are available to refer;

Quantitative existing limitation values are available for evaluation. It is considered the evaluation will be feasible by the trend evaluation or comparing with the interior plant results. These indicators are evaluated by margin to the limitation values specified in safety preservation rules, and the way of the trend evaluation or the comparison with the interior plants will be evaluated individually hereafter. Trial evaluation for "Margin of dose equivalent evaluation value for limits" indicates that the comparing with the interior plants method is more appropriate than the trend evaluation.

Type□B: Quantitative reference indicator is not exist, but evaluation is feasible by using the trend or comparison methods;

There is no quantitative limitation value to refer, but evaluation will be feasible by using the trend or comparison methods.

Type□C: Quantitative reference indicator is not exist, and objectivity is not enough;

Regarding the performance indicators extracted from "Organization and Management Controls", it may be difficult to compare between plants or with the interior plant mean value from the standpoint of objectivity.

It is considerable to evaluate by using the trend method individually for the plant, but will be difficult to find the clear basis for the quantitative judgment basis.

Therefore, evaluation on appropriate judgment basis will be necessary base on the trend results.

Type□D: The limitation deviation items;

Occurrence itself will be violation of regulatory limitations.

These items are excluded from the proposed performance indicators, because these are apparently the caution items on safety controls and considered as out side of the performance indicator evaluation scope.

Type□E: Risk information applying items on which present PSA is applicable;

Risk information applying items on which the evaluation subjected facilities and functions are modeled by present PSA and feasible to get the performance risk effect quantitatively.

There are some problems to apply the risk information to regulations, but evaluation is feasible technically.

Type□F: Risk information applying items on which the evaluation is not feasible at present ;

Risk information applying " Unavailability□Alarm System" evaluation is not feasible at present, because they are not modeled except important alarms in present available PSA.

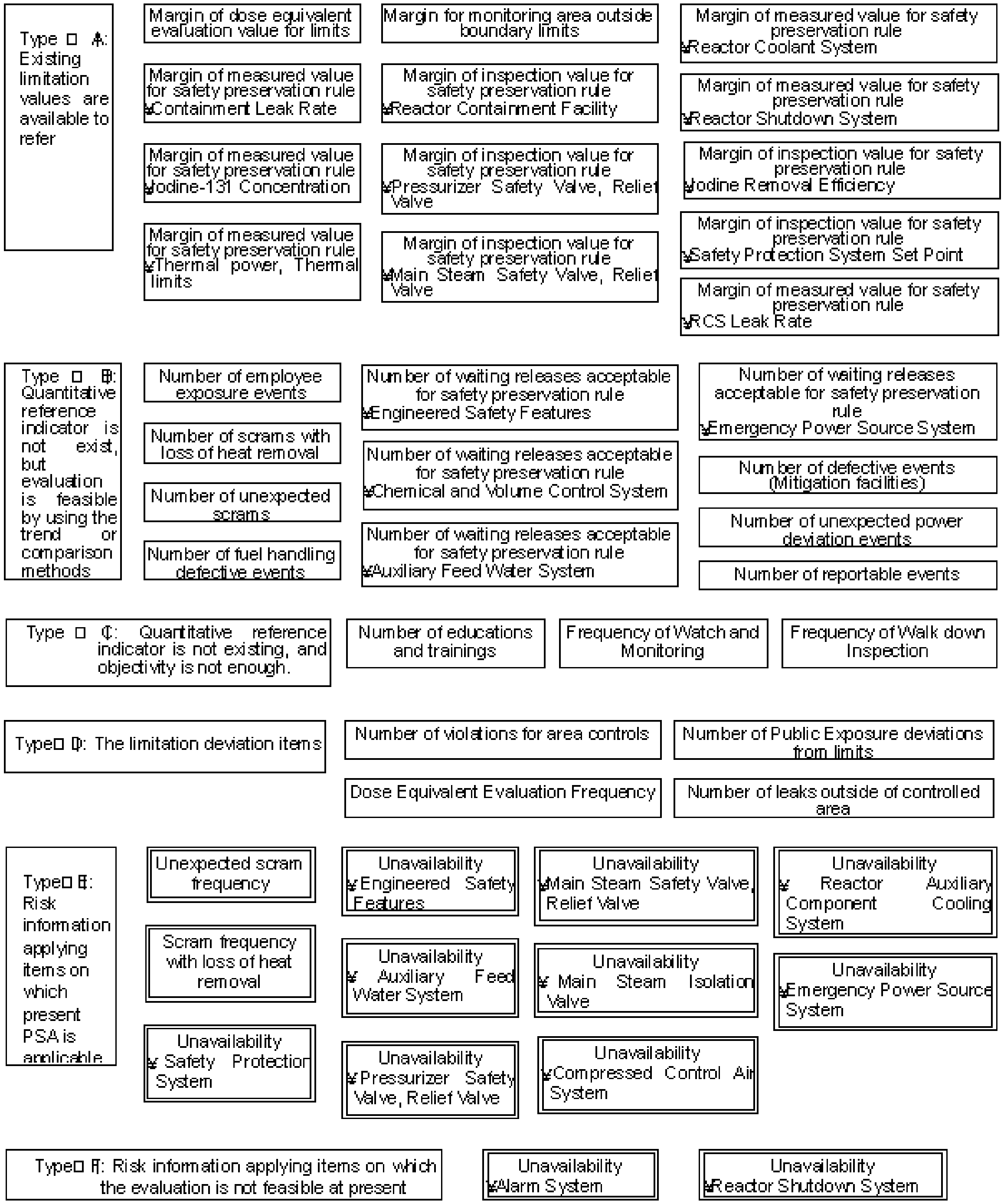


Fig.9. Classification of the proposed performance indicators based on the evaluation methods

In case of "Unavailability□Reactor Shutdown System", only " reactor auto trip failure" of control rod drive system is modeled, and a part of the function only is feasible to evaluate.

Although it will be possible to evaluate the safety significant alarms and functions (the other’s are not significant on core damage), and these items are excluded from the evaluation subjects, since the concept of applying the risk information to regulations is not clear at this stage. These are the subjects of hereafter.

7. Evaluation Example on Indicator "Margin of Dose Equivalent Evaluation Value for Limits"

(1) Evaluation on judgment basis

Considerations

: Regulatory quantitative limits (50 mSv) have been set.

: Qualitative target " ALARA" has been set.

: Employee dose equivalent are largely depend on the periodical inspection procedure.

Data: Interior plants mean value data, based on past results, are available. (For resent 10 years 1~2 mSv/year, limits 50 mSv)

In this evaluation, it is possible to set the results of the interior plants mean value as the base considering margin for limitation value.

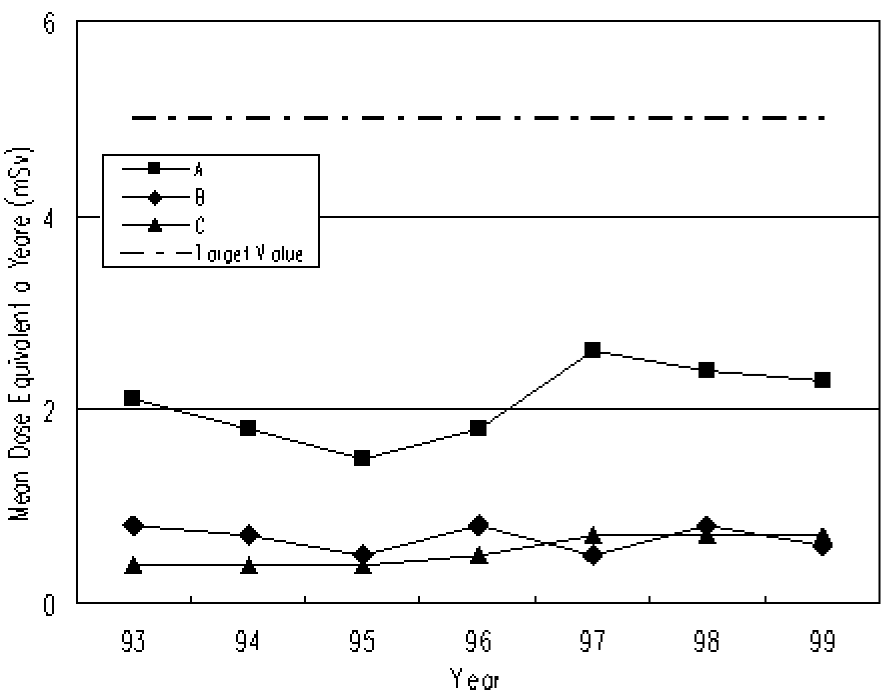


Fig.10 Trial Evaluation on Indicator "Margin of Dose Equipment Evaluation Value for Limits"
(mean dose equivalent for each site)

Regarding the subject measured values, plants mean or site mean value versus interior plants mean value, and employee maximum dose equivalent are considerable. The former is appropriate for the performance relative evaluation between plants and the latter is appropriate for evaluation on margin to regulatory limits. Both values are evaluated here.

(2) Trial evaluation using performance indicators

We performed trial evaluation for the deterministic approach based performance indicators of some model plants using the operation results to determine the adequacy of evaluation method and judgment criteria as practical safety control regulations.

Fig.10. shows the progress of mean dose mean values of each site and all of them are largely below the target value 5 mSv.

Plant A started operation in early times shows the larger values than the other two, and the other evaluation results for the past plants also indicate that the earlier plants have the tendency relatively higher values.

Although some specific plants may have the lower margins for target value, the results of this evaluation suggest the potential to over the target values is extremely low, and the regulations is not sufficiently enough.

Fig.11. shows the dose equivalent distribution for the individuals in most resent year (1999) results.

The results indicate that the maximum value for plant A is over 20 mSv, although no employee over target value 25mSv, and margin for target value is small. The mean dose equivalents for each site in that year are the same as those in other year and have no big difference.

This suggests that the performance indicators evaluation method and target value on the individuals dose equivalent are appropriate for the interior plant results.

It is concluded that the evaluation using "Margin of Dose Equivalent Evaluation Value for Limits" will produce the basic valuable results for the dose equivalent for the individuals with 25mSv target value and the evaluation for mean value with target value 5mSv show only supplemental reference results.

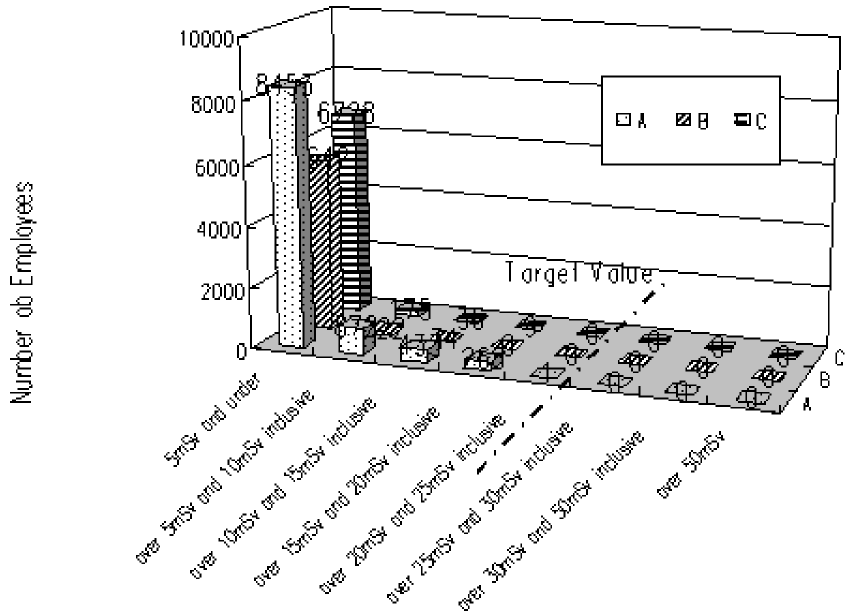


Fig.11 Trial Evaluation on Indicator "Margin of Dose Equivalent Evaluation Value for Limits"
(dose equivalent distribution for the individuals)

8. Conclusion

This report includes the investigation performed mainly last year and part of the investigation being performed this year mainly on the problems.

This investigation will be continued next year, and after the completion of fundamental evaluation , the regulatory authorities will decide whether if / how the new set of overall performance indicators be introduced to Japanese regulatory scheme.

Papers of Specialist Meeting

on Safety Performance Indicators

October 17-19, 2000, Madrid, Spain

Development of NPP safety indicator system at Ukrainian Nuclear Regulatory Authority

Oleksandr V. Pecherytsya

Nuclear Regulatory Department (NRD)

State Scientific and Technical Center on Nuclear and Radiation Safety (SSTC NRS)

Department of NPP operational safety analysis

35-37, Radgospna str., Kyiv, 03142, Ukraine

tel.: 380 44 450 4634 ext. 3 26

fax: 380 44 452 4269

e-mail: alexander@dep320.kiev.ua

Abstract

In the article materials the description of existed system being used for the assessment of Ukrainian NPP safety performance is presented. The main deficiencies of the system are pointed out. Detailed description of the new system of safety indicators being proposed to use by the Regulatory Authority is presented. Main attributes of this system are specified along with the nomenclature of indicators.

1 Existed system description

At the present time the "Temporary Provision on the annual reports on assessment of NPP Operational Safety Current State" ***[1]*** is utilized as a regulatory basis for the assessment of Ukrainian NPP operational safety. The document enforced in 1992 was developed by Allunion scientific research Institute of NPPs, Russia. The document requirements were disseminated on Ukrainian NPPs in accordance to NRD Letter 07/35 from January 21, 1994. Regulation "Requirements to content of the Reports on technical state of Chornobyl NPP units" ***[2]*** enforced in 1995 was developed in order to support the evaluation of the operational assessment of RBMK units.

The intention of regulation ***[1,2]*** development was to extend the "General Provisions on NPPs Safety" ***[3]*** requirements, which obligate the Utilities to provide to Regulatory Body the periodical reports about the level of NPP unit operational safety.

These regulations establishes as follows:

- list and methods of calculation of operational indicators, characterizing the current level of NPP unit safety;

- contents and scope of appropriate reporting documents (i.e. "Annual Report on assessment of the NPP operational safety current state" for WWER type of units and "Report on technical state of Chornobyl NPP units" - hereinafter referred to as Annual Reports).

According to mentioned regulations the Annual Reports are aimed at as follows:

- determination of trends of NPP operational safety;
- assessment of efficiency of the measures being undertaken to improve an operational safety level;
- determination of impact of particular operational aspects upon NPP operational safety as a whole;
- comparison of operational indicators of different NPPs;
- development of the corrective measures for improvement of operational safety;
- overall evaluation of the unit operational safety level being performed upon the information that collected and analyzed in the Annual report.

The specific list of indicators is used for mentioned task realization under the Annual report development. Besides, the complementary information, being used for analysis of the unit performance in particular area, is presented in the Annual report chapters.

For example, in the chapter deals with the safety system equipment availability assessment, the estimated indicators of front line and support system unavailability have to be presented along with description of all defects, leaded either to equipment unavailability or spurious actuation of safety system trains. The description includes defect causes, conditions of the defect discovery and duration of equipment recovery. Besides, the following aspects are to be evaluated thoroughly within this Annual report chapter:

- initiating events that were progressed by the safety system equipment failure;
- modifications in the safety system design, maintenance procedure and etc., with description of the modification causes;
- causes of the safety system testing schedule violations.

The conclusions about safety system state, analysis of the indicator dynamic and description of the measures on increasing of safety system availability shall be presented at the end of this Annual report chapter.

The Annual Reports are prepared yearly by the Licensee for each power unit separately and submitted to the NRD for approval. NRD sends the Annual Reports to SSTC NRS for expert assessment. The goal of such expert assessment is to evaluate the level of operational safety and the unit technical state upon the review of Annual Report materials. Main tasks being pursued under expert assessment implementation are as follows:

- assessment of the consistency of Annual Reports' materials with the ***[1,2]*** requirements;
- assessment of validity of the information being presented in the Annual reports.

The sources and documents being used for assessment of Annual report validity are as follows:

- SSTC NRS databases on operational events, component reliability and etc.;

- documents, which are used under expert review. (It is concerned about various documents, being requested by SSTC NRS from Licensee. For example, the "technical solution" (technical application for modernization, improvements and etc.) on extension of the time between tests of Reactor Protection System control rods was submitted by Licensee for NRD approval. NRD requested SSTC NRS expert review of this technical solution. The list of RPS control rod defects was requested by SSTC NRS from Licensee as a specific "ground" document. The list of defects was used later during analysis of "Safety system availability" and "Stability of the unit operation" chapters of appropriate Annual report).

The results of above mentioned SSTC NRS expert assessment are the basis for the decision of NRD regarding either approval or rejection of the particular Annual Report.

The Annual reports are the one of two "external" (i.e. available outside of the plants) data sources on Ukrainian NPP operational history. The operational event reports being developed by Licensee in accordance to "Provision on an order of the Ukrainian NPP operational events account and investigation" ***[4]*** are the second source of data. These two sources compose the informational basis for any activities related to assessment of Ukrainian NPP safety performance.

2 Background for the development of new system of safety indicators

System of indicators, described in the previous subchapter, is under operation since 1992. No revision of regulations that are the basis of this system was performed till present time. Experience of the system utilization demonstrates the necessity of it improvements. The intent of this subchapter is to present the insights of Annual report expert review being performed by SSTC NRS.

As it was mentioned above there are two directions of the Annual report expert review: review of the consistency with regulatory requirements and presented data validation.

Absence of the analytical processing being based on presented data is the main problematic area related to the first direction (contradiction of the Annual Reports to the documents ***[1, 2]***). Comparison of the indicator values with the similar ones on previous years is not implemented. The causes of the indicator value evolution are not analyzed properly. Such problem is very important especially in case of the negative tendentious in indicator evolution dynamic. As it was mentioned, the Annual reports are the relatively good and well-structured source of the statistical information on NPP operational history. However, the comprehensive analysis of the operational safety level within the certain period of time are not implemented or substantiated properly in the majority of the Annual reports reviewed by SSTC NRS till present time. Another important deficiency, being related to the direction, is the absence of certain information in the Annual report chapters. This is a direct violation of the regulatory requirements. Such violations are revealed mostly in the chapters describes the efficiency of personnel radiation protection and radwaste handling.

Comparison of the primary data used for indicator estimation and information that available in SSTC NRS reveal the incompleteness or improper classification of the Annual report data in half of cases. This fact leads directly to the incorrect estimation of the indicators (in most cases the indicator value looks more optimistic when real one). Such incompleteness of data is revealed mostly for both equipment defects and staff errors, which have to be accounted under indicator estimation.

According to SSTC NRS point of view the main reasons for mentioned deficiency existence are as follows:

the deficiencies (the absence in some cases) of quality assurance under the Annual reports development. This fact leads to improper data classification and absence or falsity of data as well;

the deficiencies of the regulatory basis for both assessment of indicators of Ukrainian NPP operation and reporting of Ukrainian NPP operational events

The deficiencies of regulations ***[1, 2]***, are as follows:

- Uncertainty of the Annual report status (i.e. the Annual report level within the hierarchy of NPP reporting documentation). This leads to uncertainty of interrelation between the Annual reports and other elements of operating feedback analysis system and the uncertainty of process of the Annual report development, approval, and future utilization as well. (As it was mentioned, the document ***[1]*** was enforced in 1992. The state structure of the nuclear regulation and nuclear power utilization is completely different from one being described in the document. The list of organizations which are involved in document approval process as well as the prescribed order of the Annual report development and approval are not consistent with reality. The feedback procedure on the Annual reports is not determined. Ukrainian NPPs develop a several similar reporting documents upon the Utility headquarters' requirements along with the Annual reports. In some cases the duplication of data has existed and the prioritization from quality assurance point of view has not been arranged properly.)

- Structure of the Annual reports is not consistent with the general requirement to such type of the documentation. The standard set of necessary information within the indicator presenting is uncompleted. (It is concerning about the necessity of presentation of unambiguous formulations on the purpose, definition and primary input data for every indicator. The illustrative examples of indicator estimation as well as useful explanation regarding the set of primary data being used are not demonstrated).

- The indicator definition is ideologically wrong in several cases. As an example, the one of primary circuit integrity indicators (depressurization rate) could be chosen. According to document ***[1]*** this indicator should characterize the integrity of primary circuit and should be calculated according to the formula:

$$Кн1 = \frac{n}{To}1000,$$

where

n – number of equipment failures being leaded to loss of integrity of primary circuit,

To – duration of unit operation for reportable period.

It will be assumed that where were three events with the primary circuit leakage occurred at the NPP unit "A". The rate of each leakage was close to minimal detectable ones. In the same time the large LOCA with two-side rupture of main circulating pipes occurred at NPP unit "B". So, mentioned indicator will be in three times worse for unit "A" then for unit "B").

Thus, it should be stated that the existed system of assessment of Ukrainian NPP indicators has a set of serious deficiencies.

3 Work objectives and scope

The overall task of this work implementation is creation of the safety indicators system that is to be used by NRD for assessment of the level of Ukrainian NPP operational safety. The created system should be able to provide the objective measures for monitoring of Licensee activities implementation process as regards to nuclear and radiation safety. The assessment being implemented through the evaluation of set of specific indicators will be a basis for making better-informed decisions on deviations in Licensee performance from expectations.

The task is being realizing through the several stages (subtasks) implementation. The matter of these stages is expressed below.

Stage 1. Review of indicator systems being used or being proposed to use by different organizations of different countries. These systems are as follows:

- performance indicator system being used by United States Nuclear Regulatory Commission (hereinafter referred to as US NRC) ***[5]***;
- indicator system being proposed to use by US NRC ***[6]***;
- safety indicator system being used by Radiation and Nuclear Safety Authority, the nuclear regulatory body of Finland (hereinafter referred to as STUK) ***[7]***;
- indicator system being proposed to use by Swedish Nuclear Power Inspectorate (hereinafter referred to as SKI) ***[8, 9]***;
- performance indicators being used by members of Word Association of Nuclear Operators (hereinafter referred to as WANO) ***[10]***;
- system of indicator for NPP operational safety performance assessment being proposed by International Atomic Energy Agency (hereinafter referred to as IAEA) ***[11]***.

The review is being performed in order to determine the directions, in which the existed non-Ukrainian approaches for NPP performance and safety assessment could be utilized under creation of NRD system.

Stage 2. Determination of the requirements to the system being created and it structure as well. The work is being implemented on the basis of above mentioned review and the available experience. The following issues are being accounted under the requirement determination:

- broadest utilization of the positive aspects of existed system on Ukrainian NPP safety and performance indicator assessment;
- hierarchical organization of the indicators' set for structural cover of the operational aspects relevant to the nuclear and radiation safety;
- determination of the indicators' nomenclature through the estimation of adequate balance between necessity of assessment of all key area where the interest of NRD is existed and the availability of primary data the validity and completeness of which will allow SSTC NRS to perform the indicator assessment;
- broadest utilization of state-of-the-art techniques for NPP safety performance assessment (i.e. the risk-oriented approach, utilization of the "retrospective" indicators (quantitative assessment of the operational aspects for the past period of time) along with the indicators that allow to access the evolution of certain matters in the future).

Stage 3. Case study is being implemented in order to test the applicability of the system. The study includes the steps as follows:

- determination of the indicators-representatives from each level of hierarchical structure;
- collecting of the real primary data on Khmelnitsky unit 1 (WWER-1000) operating history for the period from 01.01.96 till 01.01.2000;
- development of software for primary data storage and processing, indicator estimation and appropriate demonstration of indicator estimation results;

- input of the collected primary data and estimation of the indicators set for Khmelnitsky unit 1;

- interpretation of the obtained results and analysis of the system features as well.

Stage 4

The detailed propositions for the pilot project is being developed in order to provide the effective introduction and productive utilization of the created system. In the frames of the pilot project the real data for all Ukrainian NPP unit operational history will be collected. The scope of collected data will allow to implement the proper estimation of all indicators included into the created system. Besides that, the appropriate updating of the software will be implemented along with development of the writing guidelines for primary data collection, indicator estimation, and the result interpretation.

4 Attributes of system and approach being used under its development

By the definition from document ***[3],*** "Safety of NPP is its feature under normal operation and in a case of accident to restrict radiation impact upon staff, public and the environment by the established limits". Safety indicator system being proposed could be used as the tool for objective assessment of the named feature of NPP. The named system has the following attributes (characteristics):

- Spectrum of indicators being used within the system was identified depending on effect of the assessed by each indicator value upon the NPP safety.

- Hierarchical structure of indicators includes several levels described below.

- Indicators of upper level being themselves values describing safety of NPP.

- Indicators of lower level are the values monitoring of which allows to determine dependencies in changing of the upper level indicators.

- Indicators of intermediate level that were introduced to provide and to track the interdependency in the system frames.

- Information used to calculate the safety indicators is taken directly from a number of reporting documents. The procedure of their development and usage as well as requirements to these documents structure and contents are defined and valid throughout the quite long time. This allows to state that the reliability of the information they contain is sufficient.

- Mathematical grounds of the safety indicator calculation were identified proceeding from the existing experience as to processing and analysis of the data on history of Ukrainian NPP operation, carried out analysis of existing foreign systems of indicators and proceeding from the principle on provision for objective assessment of the particular safety aspect.

- Periodicity of the indicator assessment was selected in such a way to provide for efficiency and obviousness of the process of indicator evolution assessment, i.e. evaluation of trends.

- Comparatively small set of the indicators and relative simplicity of the data collecting and processing procedure allow to arrange the system operation with acceptable level of labor expenses.

Deterministic approach was used together with probabilistic one to develop the safety indicator system proposed to NRD. Overall framework of the safety indicator system is presented in Fig. 2.

	Assessment of NPP operational safety		
Deterministic approach			*Base for usage of probabilistic approach*
Upper level indicators Indicators of protecting barriers integrity			Upper level indicator Frequency of reactor scram
Intermediate level indicators Number of violation of safety limits/conditions Number of operational events per unit			Intermediate level indicators Safety system train unavailability
Lower level indicator Annual number of operational events with repeated root causes			Lower level indicator Ratio of corrective versus preventative maintenance work requests on Frontline safety system equipment

Figure 2. Structure of safety indicator system being proposed to NRD.

Usage of deterministic approach allowed to take experience into account as to using of existing system of efficiency and safety indicators at NPP in Ukraine which operation is regulated by the documents ***[1, 2]***. The basic positive point here is the list of input data needed for calculation of the indicators (each of which is deterministic by its nature) regulated by these documents. Since indicators based upon these data were calculated during about decade, then it is possible to state with sufficient extent of confidence that on-site and/or off-site the acceptable procedure exists for registering, collecting and processing of input data to calculate the indicators. The fact is obvious that when structured system for collecting of input information is available, then quantitative estimation of a indicator is performed much easier as compared with a case when it is necessary to identify the nomenclature of input data, procedure of their collecting, and the most important – the input data as such of scope sufficient enough for at least provisional assessment of this performance applicability.

The most analyzed systems used by the Regulatory Authorities in different countries and by international organizations also were developed on the basis of deterministic approach. Usage of this approach in the frames of this work allowed to use some indicators from the named systems as the basic ones for the safety indicators system proposed to NRD.

So, by the document ***[12],*** all the technology to provide for safety of NPP is based upon the defense in-depth principle. This principle "… is implemented, first of all, by creation of the series of barriers which, in principle, shall be by nothing and never jeopardized, and which, in turn, shall be violated before any harm can be made to human and the environment. These physical barriers provide for a possibility of successive confinement of radioactive substances". In accordance with ***[3],*** this system of barriers includes:

- Fuel matrix;
- Cladding of fuel element;

- Boundary of coolant circuit that cools core;
- Containment systems.

Thus, monitoring of the named physical barriers integrity was placed on the top of the list under determination of the concept to perform analysis of NPP safety based upon deterministic approach. Appropriate *upper level indicators* were identified in the frames of this system in the purpose of to arrange the named monitoring. The set of physical barriers integrity indicators presented in ***[6]*** was taken here as the basis.

The plant design envisages the limits and conditions of its safe operation. Violation of the named limits and/or conditions can lead to occurrence of accident with severe radioactive consequences as to both staff of plant and to the public and the environment as a whole. That is why, monitoring of events occurred at NPP and led to violation of NPP safe operation limits and/or conditions shall be performed in the frames of the safety indicator system.

The total number of events occurred during the certain period of time (for example - for one year) reports on which investigation were submitted to the Regulatory Authority (i.e. NPP operational events are dealt in the terms of document ***[4]***) could be chosen as another significant indicator of similar level. The threshold on reporting of such events is defined in such a way, that any event, report on which investigation was submitted to the Regulatory Authority, was of negative impact upon the NPP safety.

Using of two above mentioned *intermediate level indicators*, namely: "Number of NPP safe operation limits and/or conditions violation" and "Number of operational events" will allow to monitor overall tendency about how well the NPP staff maintain the plant within licensing requirements and comply with other procedures and rules. The indicators similar to the mentioned above are presented in the most of foreign systems analyzed.

Reoccurrence of the events significant for safety of NPP due to these causes were not properly analyzed and eliminated is the evident of deficiencies of operational experience feedback utilization. The document ***[12]*** repeatedly stressed importance of the efficient system arrangement to learn lessons from the history of operation and analysis of root causes of events to prevent their reoccurrence in future. To arrange monitoring as to efficiency of feedback from operational experience it is proposed to use the lower level indicator "number of NPP operational events with repeated root causes" (this indicator was taken from the list of D-type indicators described in the document ***[9]***). Usage of this indicator will allow to detect problematic areas in operation of NPP which require attention to be paid by the Regulatory Authority and, maybe, measures to be undertaken, to prevent from growing of negative tendencies which lead to reduction of NPP safety level.

The existing indicator system in various countries were devised based on engineering common sense and some knowledge of interplay of plant design and operation to risk; however, the relationship of the indicators to risk was never established prior to actual implementation of these systems in various countries. As an example of the advanced system of risk-oriented indicators that can be referred, proposed for usage by the Regulatory Authority in Sweden ***[10,11]***. Necessity and usefulness of the risk-oriented indicator usage is also stressed in the relevant IAEA recommendations ***[11]***.

All the referred sources indicate that availability of NPP probabilistic safety analysis results is the necessary element for arrangement of the systematic approach with the use of risk-oriented indicators. The basic result of PSA depending upon scope of the performed calculations is, respectively, core damage frequency (CDF), frequency of containment failure, and frequency of occurrence with impermissible radiation impact upon public. Each of the named values itself is indicator of NPP safety, while the rest of risk-oriented indicators are those of deliberately lower level. They are either derivatives from the named values, or characterize their separate elements.

Presently, PSA is performed for power units of NPP in Ukraine as part of Safety Analysis Report development. Stages of PSA completeness are different for various power units, and its results are inaccessible for official usage. Thus, by the moment, it is impossible to start with using of the recommended spectrum of risk-oriented indicators in the framework of NRD system. However, performed analysis of the foreign systems showed the availability of data being necessary for creation of the basis for subsequent monitoring of Ukrainian NPP safety level with the use of probabilistic approach is available. It would be reasonable to arrange the quantitative analysis "components" of the above mentioned indicators (mostly it is related to CDF). These components, in turn, are also being safety indicators that characterize values connected with risk caused by operation of NPP.

Therefore, it is proposed to use the "scram frequency" as the probabilistic upper level indicator within NRD system. Usage of this indicator will allow to assess the intensity of initiating events occurrence. This indicator reflects efficiency of activities on NPP safety improvement by reduction of transients' number, which lead to actuation of reactor protection system.

After adoption of the proposed concept for the NRD system, it is proposed to expand later on the spectrum of upper level indicators by a number of indicators, which characterize the frequencies for different groups of initiating events. Such approach to determination of nomenclature as to the risk-oriented indicators was recommended in the documents ***[8,9,11]***. These documents present as an example classification of initiating events according to the following four groups:

- Loss of primary coolant accident;
- Loss of off-site power;
- Steam generator tube rupture;
- Transient (operational event with scram actuation).

SSTC NRS used similar approach to classification of initiating events under performing of works ***[13,14,15]***. However, analysis of the historical data on Ukrainian NPP operation (1992-1997) had detected insufficient extent of statistic representation by the first three of the above mentioned categories. From another hand, flow of statistical data by the fourth one allows to talk about possibility to single out some groups of these events (for example, "Trip of MCP", "Loss of feedwater", etc.). It will allow to follow up in future frequencies of such initiating events and use these indicators during the process o risk-oriented monitoring of NPP safety.

The certain list of additional upper level indicators, which characterize intensity of appearing of different groups of initiating events can be identified as the result of studying of statistical data on history of Ukrainian NPP operation. It is expected to perform these studies later.

Indicator of the frontline and supporting safety system train unavailability could be used as intermediate level indicator under development of the basis for risk-oriented monitoring of NPP safety. Usage of this indicator will allow to follow up the level of plant availability with the use of appropriate technical means, adequately responding to occurrence of initiating events. It is proposed to calculate the indicator of train unavailability for the following systems:

- High pressure emergency injection;
- Low pressure emergency injection;
- Emergency feedwater;
- Emergency power supplying.

The list of safety systems which trains availability monitoring will be performed was selected proceeding from these systems significance under elimination of consequences caused by initiating events addressed under performing of probabilistic safety assessment. (The systems have been selected for this indicator based upon their importance in preventing core damage. The selected systems include the principal systems needed for maintaining reactor coolant inventory following a loss of coolant, for decay heat removal following reactor trip or loss of feedwater, and for providing emergency AC power following a loss of plant off-site power). Limitation of safety indicator number (i.e. the decision made to restrict the list of safety systems) was stipulated by necessity of overall restriction of NRD system indicator number in a purpose to provide its compactness and simplicity of its operational process. (This attribute of the system was already mentioned at the beginning of this sub-section). However, it does not mean that this list could not be expanded in future in case of necessity to arrange monitoring as to level of any another safety system availability.

Finally, as the lower level indicator concerning probabilistic aspects it is proposed to use in the NRD system " Ratio of corrective versus preventative maintenance work requests on Frontline safety system equipment". To calculate the indicators integral magnitudes of appropriate values will be used (number of recoveries after defects and number of maintenance cycles) for all the four above mentioned safety systems. The relevant D-type indicator (i.e. value which "correlates with a change in failure rate … or frequency of initiating events" ***[9]***) was used under identification of this indicator. Usage of the mentioned lower level indicator will allow to trace interrelation between level of the safety systems availability and quantitative, while the most important, qualitative, characterization of maintenance of the NPP safety significant equipment.

Therefore, the safety indicator system proposed to NRD includes above described indicators. The nomenclature of these indicators was identified proceeding from the tasks to be solved by the named system along with its characteristics. At the present time the case study is implemented in order to test the applicability of the created system. The six indicators-representatives were selected for this study. The collecting of primary data for the indicator estimation is in progress now. Expected date for the case study implementation is December 2000.

Nuclear Regulatory Department will use the safety indicator system being created under the work implementation for maintaining the effective oversight of Ukrainian NPP safety performance.

References

1. "Temporary Provision on the annual reports on assessment of NPP Operational Safety Current State", 1992.
2. "Requirements to content of the Reports on technical state of Chornobyl NPP units", 1995.
3. "General Provisions on NPPs Safety". NP 306.1.02/1.034-2000.
4. "Provision on an order of the Ukrainian NPP operational events account and investigation", RD 306-205.96.
5. Description of the performance indicator Report, INEL-95/0356, 1995.
6. Regulatory Assessment Performance Indicator Guideline, NEI, Draft, 1999.
7. The development and use of safety related indicators at STUK, STUK, 1999.
8. Guidance on the Implementation of risk-based safety performance monitoring system for NPP, SKI report 97:46, 1997;
9. Research project implementation of a risk-based performance monitoring system for NPP, Phase II – type-D indicators, SKI Report 99:19, 1999.
10. Detailed description of WANO NPP performance indicators, Annex 1 to WANO Implementing Guideline IG19.1, 1998.

11. IAEA-TECDOC-XXX, draft, Operational safety performance indicators for NPP, IAEA, 1999.

12. Basic safety principles for NPP, 75-INSAG-3, IAEA, Vienna, 1989.

13. Final report, "Initiating event and equipment reliability data collection for it usage in the accident sequence precursor models", task 6.1.15 of 1996 NRD Request, SSTC NRS, Kyiv, 1996.

14. Final report, "Development of the accident sequence precursor models for WWER-1000 type of units", task 6.2.7 of 1996-1997 NRD Request, SSTC NRS, Kyiv, 1998.

15. Final report, "Implementation of risk assessment of Ukrainian NPP operational events from 1992 to 1997", task 3.1.6.4 of 1997-1998 NRD Request, SSTC NRS, Kyiv, 1999.

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

SAFETY INDICATORS IN THE NUCLEAR REGULATORY PROCESS IN SOUTH AFRICA

Tim Hill

National Nuclear Regulator

Koeberg Project Department

Private Bag X10

Kernkrag 7440

Cape Town

South Africa

Tel (+27) 21-5539500

Fax (+27) 21-5532060

Email thill@nnr.co.za

Abstract

In support of the approach adopted by the nuclear regulatory authority (National Nuclear Regulator – NNR) in South Africa, which combines risk, deterministic and process based thinking, the use of safety indicators is considered an important means of providing objective evidence of the success or failure of safety related activities or processes, as well as direct evidence of the status of safety factors relating to the plant, licensed site, the workforce, the public and the environment.

The use of safety indicators helps to focus attention on weak areas and to provide information in a format which can be trended and which is readily reportable and comprehensible to the licensee management, public and different levels of the various regulatory and government organisations.

The primary objectives of the system of safety goals and indicators is as follows:

- To provide an objective measure of compliance of the licensed site with the safety case for the site, in terms of factors relating to plant health, workforce, public and environment, and trending thereof.

- To provide an objective measure of the status of supporting safety related activities and safety factors.

- To help focus resources on areas of concern in all licensing and safety assurance activities (inspection, assessment, pro-active work, guidelines etc), in a graded manner taking into account the status of relevant safety factors.

- To facilitate reporting of the safety status of licensed facilities to different levels of the licensee and regulatory organisations.

This paper describes the recent developments of safety goals and indicators as applied in the reactor licensing process in South Africa.

INTRODUCTION

Safety goals and indicators have been developed by the nuclear regulatory authority (National Nuclear Regulator - NNR) in South Africa in support of the compliance assurance programme.

In the development of such a system, safety goals were established first with a view to addressing all significant safety factors enveloping the overall safety case for the licensed facilities, including those aspects of the licensee organisation relating to safety, in a top-down approach designed to provide assurance of safety in broad perspective in terms of the safety requirements of the regulator.

The safety goals address the fundamental safety standards of the regulator covering risk to the public arising from normal operations and potential accidents, ALARA, quality management requirements, defence-in-depth, comparison with and assessment against acceptable international benchmarks, the ALARA principle, and emergency planning requirements.

The above safety requirements imply numerous provisions, undertakings and assumptions which underpin the safety assessment. These are to a large extent covered by the conditions of license in terms of the licensees safety assurance processes, the design, operating rules, specifications, and procedures themselves. In line with the objective to provide focus on all safety assessment and assurance activities, relevant safety goals were established to address these factors as far as practicable.

The safety indicators, which are in one-to-one correspondence with the safety goals, provide a measure of the extent to which the safety goals are being achieved (direct indication) or could be challenged (leading indication).

A system of ranking the level of safety concern and enunciating the status of each indicator is used. This is based on a qualitative or quantitative assessment in terms of the safety fundamentals referred to above with the aim of minimising subjectivity.

The safety goals and indicators have been linked to a compliance assurance programme.

1. DEVELOPMENT OF SAFETY GOALS AND INDICATORS

The approach taken was to use safety indicators as a measure of conformance with a set of safety goals which would in principle cover all aspects of the safety case and compliance assurance for a specific plant. The first priority was therefor to develop the safety goals based on the following considerations.

The licensee is required to develop and maintain a plant specific safety case in support of an application for a (nuclear) licence for a new installation or for any modification or change to an existing installation which impacts on nuclear safety.

The safety case comprises all documentation relevant to the demonstration of compliance with the fundamental safety requirements of the NNR which include the following:

- Risk criteria for public and plant personnel (normal operations and accident conditions)
- Requirements relating to conformance with international norms and practices
- Compliance with the ALARA principle
- Requirement to ensure an effective emergency plan
- Requirements relating to quality assurance.

The scope of the safety case is summarised in table 1.

Once the safety case has been approved by the regulator, licence conditions are established which bind the licensee to the provisions and undertakings identified in or implied by the safety case.

In the development of the safety goals, a top- down approach was used based on the hierarchy of the safety case and supporting processes (table 1) to ensure a close correspondence between the safety goals and the elements of the safety case, and to make it easier to assign levels of concern to findings through the context of the associated safety goals in terms of the safety case.

The safety goals were organised into a tree structure comprising three tiers, the first tier representing a breakdown of the safety case hierarchy (table 1) into broad disciplines as follows:

1. Assessment
2. Operations
3. Engineering - Mechanical
4. Engineering - Electrical
5. Engineering - Civil
6. Engineering - Instrumentation
7. Engineering - Systems
8. Engineering - Nuclear
9. In-service inspection
10. Maintenance - General
11. Maintenance – Mechanical
12. Maintenance – Civil

13. Maintenance – Instrumentation

14. Maintenance – Electrical

15. Radiation Protection - Environment

16. Radiation protection – Operations

17. Radiation protection – Radioactive waste

18. Radiation protection – Emergency plan

The second tier represents a breakdown of the first tier into attributes which are applicable to the above disciplines as follows:

1. Standards, codes, criteria
2. Safety case
3. Compliance
4. Processes
5. Management, organisation, conduct
6. Corporate oversight

The third tier represents essentially a list of attributes or questions specific to the first and second tier .

For example for 1.3 Assessment-Compliance:

1.3.1 Accident risk – public

1.3.2 Accident risk – plant personnel

1.3.3 Normal operations – public

1.3.4 Normal operations – plant personnel

1.3.5 Defence-in-depth

1.3.6 International Benchmark

1.3.7 Emergency Planning

1.3.8 ALARA

As another example for 2.5 Operating – Management:

2.5.1 Organisation

2.5.2 Leadership

2.5.3 Direction

2.5.4 Monitoring of activities

2.5.5 Follow-up/feedback

2.5.6 Planning

2.5.7 Human resources/competence

2.5.8 Safety culture

2.5.9 Facilities and equipment

2.5.10 Communications

Further explanation of the safety goals is given in tables 2 and 3 and in sections 1.1 to 1.5.

1.1 Assessment

Consideration is given firstly to safety assessment as the process which ultimately demonstrates compliance with the fundamental safety requirements. This is expanded upon in table 2 and under the subheadings below.

Codes/standards/criteria

With reference to table 2, under safety assessment, the first subcategory in this second tier of safety goals refer to the safety standards and codes adopted by the licensee to conform to those prescribed by the regulator. These typically refer to general codes and standards (eg US or French regulations), whereas those specific to various disciplines (operations, engineering, maintenance, RP) are addressed specifically under these headings.

Safety case

The second subcategory of safety goals refer to the safety case (table 1) itself which must be plant specific and maintained to demonstrate ongoing compliance with the safety standards and codes accordingly. The documentation must be satisfactory in terms of scope, coherence and comply with QA requirements. The safety case must be linked to the bases for the general operating rules and define a safety envelope for plant operations, engineering, maintenance and radiation protection.

Compliance

The third subcategory of safety goals refer to the extent to which the safety case demonstrates compliance with the safety standards referred to in section 1, and adequacy of safety margins.

Processes/supporting activities

The fourth subcategory of safety goals refer to supporting activities for safety analysis and licensing.

Management/organisation

The fifth subcategory of safety goals refer to organisational aspects, competence and quality assurance.

Corporate

The sixth subcategory refer to the corporate oversight activities of the licensee.

The safety goals were developed further to reflect the fact that the validity of the safety case hinges on the following:

- Operation of the plant in a manner consistent with the safety case.
- The plant hardware being designed, constructed, installed and tested in a manner commensurate with the safety case.
- The plant being maintained in a manner commensurate with the safety case.
- The plant environs do not change in a manner which would invalidate the safety case.

Sections 1.2 to 1.5 describe the above (supporting) safety goals, with radiation protection included as a distinct set in view of its direct impact on nuclear and radiation safety and to include additional factors such as off-site developments, emergency planning and environmental monitoring.

1.2 Operations

The safety goals for operations were based on the consideration of (mainly procedurised) human actions assumed or implied in the safety case to prevent (or reduce the probability of) transients or accidents and to mitigate their consequences. The human factors relating to engineering, maintenance and RP are included under their respective headings (1.3 to 1.5). The safety goals for operations are expanded upon in table 3 and under the subheadings below.

<u>Codes and standards</u>

The heading "codes and standards" here refers to all standards relevant to nuclear safety pertaining to operations, and include standards for training, examinations, medical, and psychometric analysis.

<u>Safety case</u>

The safety case must provide a well defined basis for operations, including technical bases for operating/accident/incident procedures, and operating technical specifications.

<u>Compliance</u>

Evidence on competence, human reliability, medical, psychometric and safety culture.

<u>Processes/supporting activities</u>

Evidence of adequate document control, training, procedure change control etc.

<u>Management</u>

Evidence of the adequacy of the organisation, quality assurance, planning, and safety culture pertaining to operations.

<u>Corporate</u>

Corporate oversight activities of the licenced facility (or site) on operations.

1.3 Engineering

The safety goals for engineering refer to plant engineering aspects (systems, structures, components, instrumentation, fuel, software etc) being designed, constructed, installed, tested in a manner consistent with the safety case.

<u>Safety case</u>

The safety case is required to demonstrate that the design, construction, manufacturing, installation and testing of the plant is acceptable in terms of specifications which are commensurate with the safety case. The associated supporting analyses must comply with QA requirements.

<u>Compliance</u>

Evidence of compliance with the specifications referred to above.

<u>Processes</u>

Adequacy, implementation, auditability of processes relating to design, supporting analysis, modification, testing, commissioning, document control, training, etc.

<u>Management</u>

Organisation, quality assurance, planning, conduct, safety culture pertaining to engineering.

<u>Corporate</u>

Corporate oversight activities of the licensee on engineering.

1.4 Maintenance and in-service inspection

The safety goals for maintenance and in-service inspection are analogous to those for engineering, with the focus of compliance being on reliability and integrity of all engineering aspects commensurate with the safety case.

1.5 Radiation Protection

The safety goals for radiation protection are analogous to those for maintenance but are split into the following (first tier) subgroups, with compliance referring to compliance with administrative controls:

- Environment
- Plant
- Radioactive Waste
- Emergency Planning

The safety goals described above cover the attributes necessary to demonstrate compliance with nuclear safety criteria. The safety indicators are a measure of the extent to which these safety goals are met. The safety indicators are in one-to-one correspondence with the safety goals. The indicators are not defined a-priori in terms of for example number of reactor scrams per annum, but in terms of the level of safety concern associated with each safety goal. The process of establishing the level of concern is described in section 2.

2. ASSESSMENT PROCESS

Guidelines are given to technical and inspection staff within the regulatory body on the classification of findings in terms of level of concern and linkage to the safety indicators. A qualitative process is used as a first level of screening in all cases. This may be followed up by a quantitative analysis.

If it is believed that a finding challenges the validity of assumptions or data used in the safety case, then a quantitative analysis should be performed. The Probabilistic Risk Assessment (PRA) provides a framework for assessing the risk significance of findings both individually and on a collective basis. This could be important when a number of concerns could have a greater significance when considered collectively than when they are considered individually.

2.1 Qualitative Classification of Findings

The qualitative approach is intended to reduce the level of subjectivity in the classification process without recourse to a quantitative analysis, but still enabling the assessor to relate the concern to basic principles of nuclear safety.

The qualitative safety assessment process addresses the following safety principles:

- Risk to the public and plant personnel due to potential accidents
- Risk to the public and plant personnel due to normal operations
- Supporting safety principles such as:
    - Defence-in-depth
    - ALARA
    - Capacity and redundancy
    - Protection against common cause failures
    - Diversity
    - Safety margin
    - Level of reliance on programmatic activities

2.2 Quantitative Classification of Findings

<u>Accident risk assessment</u>

A quantitative risk analysis is performed when it is perceived that the validity of the assumptions or data used in the approved PRA for the licensed facility is in question. The PRA may be used to assess the safety impact of individual concerns or a group of concerns collectively.

Although in principle the PRA addresses all aspects of safety, the level of detail may not be sufficient to directly model a specific concern. An attempt should however be made to reflect the concern in the input data used in the plant-specific model.

The safety impact be expressed in terms of the following:

1. Increase in system reliability
2. Increase in initiating event frequency
3. Increase in core damage frequency
4. Increase in source term magnitude
5. Increase in source term frequency
6. Increase in average population risk due to potential accidents
7. Increase in peak risk to plant personnel due to potential accidents

Any decision on the level of acceptability of the issues analysed would be taken on the basis of both the qualitative and quantitative assessments. It is considered virtually impossible to a priori specify precise criteria in this respect. As a guide however, the NNR would consider the following to be intolerable:

- A risk increase (either of the items 6 or 7 above) greater than the equivalent of about 5% of the corresponding risk criteria.
- A risk increase resulting in non-compliance with any risk criteria pertaining to accident risk.

In addition, the principles of defence-in-depth, ALARA and international norms are required to be taken into account.

<u>Deterministic assessments</u>

Depending on the nature of the concern, various analyses may be performed in support of follow-up actions. These may include:

Analysis of compliance with dose criteria for plant personnel for normal operations

Analysis of compliance with dose criteria for members of the public for normal operations

Analysis of compliance with plant design requirements.

Analysis of compliance with emergency planning bases.

3. APPLICATION

The development of the safety goals and indicators led to a reassessment of the compliance inspection programme. Additional tasks were identified to provide input to the safety indicators. A baseline inspection programme was developed and implemented on an electronic task management system and linked to the safety indicators (figs. 1 and 2). The inspection and assessment reports are linked to the inspection tasks and stored on the same system. This serves as a useful base for the findings which serve as input to the safety indicators. Each safety indicator may be traced back electronically to specific findings with reference to the reports on the task management system.

Input to the safety indicators is provided by the various monitoring processes implemented by the regulator which include inter alia the following:

1. Inspections and audits conducted in terms of the compliance inspection programme.
2. Technical assessments conducted on submissions by the licensee, mainly for modifications.
3. Reports submitted by the licensee in terms of licence compliance.
4. The licensee safety indicators (performance and safety indicators).
5. Periodic reviews or other proactive assessments conducted by the regulator (including international experience feedback).

The NNR technical specialist or inspector responsible for a finding arising from any of the above processes, performs a provisional classification of the finding using the qualitative risk assessment processes described in section 2.

The findings, along with their provisional classifications, are discussed by the regulator at meetings, attended by inspection and technical staff, generally held on a weekly basis, or on an ad hoc basis should the severity of the finding demand an earlier response. A final classification is established.

A specific individual in the project department of the regulatory authority is allocated the task of entering the data into the NNR Safety Indicators Database, for maintaining records generated associated with the classification of the findings and for generation of reports directly from the indicator software.

The safety indicators may be viewed at any time by regulatory staff via the network, but input may only be made at a single point.

The level of concern of any finding is established on the basis of the qualitative or quantitative assessment (if performed).

Depending on the level of concern, the follow-up actions are generally as follows:

Intolerable (red): Report to line management immediately for identification of appropriate follow-up action.

Tolerable – high (orange): Report to line management within several days for identification of appropriate follow-up action.

Tolerable – medium (yellow)

or

Low (blue): Report at next department meeting.

Acceptable (green): Report at next department meeting.

At the project department meetings the findings and corresponding levels of concern are discussed. The appropriate level of interaction with the licensee is decided upon. This may be any of the following:

- Nuclear Safety Assurance Group
- Licensing and Liaison Committee Meetings
- Ad hoc communication between NNR inspector/assessor/manager and licensee counterpart.

Single Point Contact meetings between regulator and licensee specialists may be called for arising from any of the above initial interactions.

If a satisfactory response is not obtained from the above interactions, the issue may be raised to a higher level of interaction with the licensee accordingly, ie executive or board level.

Figure 1 shows a schematic representation of the application of the safety indicator system.

Figure 2 shows schematically how the indicators may be viewed and analysed. It should be noted that although the top level indicators default to the highest level of concern down the tree, the expanded analysis provides an indication of the spread of levels thereby providing a more balanced picture overall, to some extent removing the purely negative bias of the top level indicators.

4. INSIGHTS FROM SAFETY INDICATORS

The safety goals and indicators described in this paper cover safety related activities and processes, as well as direct evidence of the status of safety factors relating to the plant, licensed site, the workforce, the public and the environment in line with the licensing approach in South Africa. Although this approach is intended to be as objective as possible, a measure of subjectivity is unavoidable. The approach is more often qualitative than quantitative.

The NNR view is that the quantitative safety and performance indicators currently in use internationally have the advantage of providing an objective and unambiguous means for comparison of different plants in certain respects and may be used to provide some level of confidence that a specific plant is comparable to similar plants elsewhere. These indicators are however essentially performance indicators and are not sufficient to provide a comprehensive indication of safety.

Typically, although the approach described in this paper makes provision for indication of plant health, the concerns arising from the inspection programme tend to relate more to organisational aspects, human resources, competencies and performance, compliance with and interpretation of standards, procedures and technical specifications, technical bases for general operating rules, and compliance with an acceptable international benchmark. Safety issues of this nature, some of which are extremely safety significant, are not always apparent in the performance indicators currently in use internationally.

5. CONCLUSIONS

The monitoring of a nuclear facility in terms of nuclear safety assurance involves a large volume of information from diverse sources and of different levels of significance. An indicator system linked to a compliance assurance programme and reporting and assessment system as described in this paper provides an efficient means of processing findings, prioritising regulatory activities and tracking the progress of follow-up and close-out actions.

The use of such a system provides the regulator with a means of assessing overall compliance objectively and to link the conclusions to technical and inspection reports efficiently.

The use of a transparent indicator system provides an incentive for the licensee to address safety concerns timeously so as to avoid unfavourable indication.

The use of a completely independent system of indicators by the regulator tends not to discourage the licensee staff from reporting problems and deviations, which may be the case if both regulator and licensee were to use a common system.

The (three tier) tree structure of indicators described in this paper provides a convenient means of communication between technical and inspection staff with management and decision makers, and between different levels within the regulatory authority (management, executive, board etc), both on an ongoing basis (ie network system) and periodic reports (bimonthly, annual etc).

The use of such a transparent system provides an incentive to inspection and technical staff to report findings clearly regardless of whether the findings are positive or negative and to take ownership of their findings and follow-up actions as the indicator system is traceable back to these reports.

TABLE 1. SAFETY CASE – DOCUMENT HIERARCHY

NNR Standards and Requirements
Nuclear/radiation safety standards Policies
Organisational aspects
Safety Analysis Report (SAR) Probabilistic Risk Assessment (PRA) Safety envelope Technical bases for General Operating Rules

Supporting documentation (eg design documentation)
General Operating Rules (GOR): Operating Instructions Operating Technical Specifications Accident/incident procedures Severe Accident Management Emergency Plan Physical Security Examination, Inspection, Maintenance and Testing Decommissioning Radiological Protection Programme Effluent and Waste Management Programme
General processes, including: Quality Assurance Programme Modification, change control and review processes Processes for compliance with international standards/benchmark, ALARA Risk Management Incident and occurrence reporting Problem notification and follow-up Reporting to NNR

TABLE 2. SAFETY GOALS - ASSESSMENT

ATTRIBUTE	**INDICATORS**

APPLICABLE CODES/STANDARDS/ CRITERIA	SAFETY CRITERIA, STANDARDS, RULES General, eg US or Fr regulations, NQA-1
RELEVANT ASPECTS OF SAFETY CASE	SAFETY CASE (SAR and supporting documentation) Documentation, coherence, safety envelope
COMPLIANCE	COMPLIANCE - ACCIDENT RISK/PUBLIC
	COMPLIANCE - ACCIDENT RISK/WORKER
	COMPLIANCE - NORMAL OPS/PUBLIC
	COMPLIANCE - NORMAL OPS/WORKERS
	COMPLIANCE - DEF.-IN-DEPTH
	COMPLIANCE - INTERNATIONAL BENCHMARK
	COMPLIANCE - EMERGENCY PLAN
	ALARA
PROCESSES AND SUPPORTING ACTIVITIES	PROCESSES AND TECHNICAL SUPPORT FOR SAFETY CASE Screening/evaluation process, analysis support, review process, international practice, experience feedback, QA
MANAGEMENT AND CONDUCT	MANAGEMENT OF SAFETY AND LICENSING Organisation, human resources
CORPORATE	CORPORATE Organisation, implementation

TABLE 3. SAFETY GOALS – OPERATIONS, ENGINEERING, MAINTENANCE, RADIATION PROTECTION

DISCIPLINE AREA ATTRIBUTE	**2. OPERATING**	**3-8. ENGINEERING** **MECH|ELEC|CIV|INSTR|SYS |NUC**	**9-14. MAINTENANCE/ISI** **MAINT|ISI|MECH|ELEC|** **CIVIL|INSTR**
CODES/STANDARDS	CODES/STANDARDS Specific to operations	CODES/STANDARDS Specific to engineering disciplines	CODES/STANDARDS Specific to maintenance/ISI disciplines
SAFETY CASE	SAFETY CASE Technical bases for relevant GOR (operating and accident procedures, OTS etc)	SAFETY CASE Safety analysis/design/manuf documentation, QA/QC, supporting analyses, definition of safety envelope	SAFETY CASE Technical bases for maintenance and ISI, supporting analyses, definitic of safety envelope, QA/QC.
COMPLIANCE [Direct evidence of compliance with conditions/assumptions of safety case]	HUMAN FACTORS Competence, psychometric medical, safety culture	PLANT/ENVIRONS Engineering aspects are demonstrated to be within specified limits commensurate with the safety case	PLANT/ENVIRONS Maintenance aspects are demonstrated to be within specified limits commensurat with the safety case
PROCESSES [Activities underpinning the safety case and technical bases for GOR]	PROCESSES RELEVANT TO THE CONDUCT OF OPERATIONS (eg Document control, change control, work control, operability, training, facilities)	PROCESSES RELEVANT TO THE CONDUCT OF ENGINEERING (eg Design, installation, commissioning, modification control, analysis, review, QA/QC, work control)	PROCESSES RELEVANT T THE CONDUCT OF MAINTENANCE (eg Testing, materials management, repair and replacement)
MANAGEMENT Organisation, Direction, Follow-up/feedback, Planning, Resources, Competence	MANAGEMENT OF OPERATIONS Organisation, Direction, Monitoring of activities, Follow-up/feedback, Planning, Resources, Competence, Safety culture, Facilities and equipment, Communications	MANAGEMENT OF ENGINEERING Organisation, Direction, Monitoring of activities, Follow-up/feedback, Planning, Resources, Competence, Safety culture	MANAGEMENT OF MAINTENANCE Organisation, Direction, Monitoring of activities, Follow-up/feedback, Plannin Resources, Competence, Safety culture, Facilities and equipment, Communications
CORPORATE	CORPORATE OVERSIGHT ON OPERATIONS	CORPORATE OVERSIGHT ON ENGINEERING	CORPORATE OVERSIGHT ON MAINTENANCE/ISI

Figure 1. Safety Indicators - Application

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

Regulatory Body Experience with the Safety Indicator Use

Radomir Rehacek

State Office for Nuclear Safety

Department of Inspections

Senovazne nam. 9

110 00 Praha 1

Czech Republic

phone: +420 2 21624 729

fax: +420 2 21624 202

e-mail: radomir.rehacek@sujb.cz

Abstract

The Czech regulatory body (SÚJB) has been using a set of Safety Indicator for nuclear safety assessment of NPP operation since the late 80´s. As the bases for this set WANO, IAEA and other regulatory body documents have been used. The regulatory body is intenting its attention on nuclear safety, that is why Safety Indicators were mostly selected to the collection describing five different areas of the NPPs operation – Significant Events, Human Factor, Safety Systems Performance, Barrier Integrity and Radiation Protection.

The set is still under further development. Study in the field of Risk Based Indicators (RBI) was performed during the years 1998 and 1999. A small set of RBIs is now in trial use.

The results of the annual Safety Indicators evaluation are presented in the SÚJB annual report and used during the SÚJB and licensee top management meeting as bases for the nuclear safety assessment.

introduction

At the late 80´s, there was a need for the Czech Regulatory Authority (correctly Czechoslovak Atomic Energy Commission the State Regulatory Authority of that time and SÚJB´s predecessor) to create a tool, by which would give the possibility accurately and objectively quantify nuclear power plant safety performance. The other purposes were to use the tool for Regulatory Body inspection recourse allocation, as a base for the regulatory actions, as a communication tool to the industry and the public.

WANO and IAEA documents were used as a starting point to develop the first set of indicators, which was more or less a collection of performance indicators. That set contained nine indicators and seven of them was identical with WANO indicators. As SÚJB is mainly focused on nuclear safety, the areas of NPP operation the most influencing the level of nuclear safety performance, were identified and some old and new indicators were attached to appropriate area. The first set of Safety Indicators was used for the test evaluation of the NPP Dukovany operation in the year 1990.

Since that time the selected areas have been several times more precisely specified and also new indicators have been included into the set. Nowadays SÚJB set of Safety Indicators covers five different areas of the NPP operation and it is created by 21 indicators [1].

Safety indicators set used by SÚJB

There was identified five areas of the NPP operation. The areas are as follows:

1. Significant Events

2. Human Factor

3. Safety System Performance

4. Barrier Integrity

5. Radiation Protection

The number of operational events ranking as INES 1 and higher, number of reactor scrams, violations of the Limits and Conditions for operation are counted in the Significant Events area.

Human factor is the root cause or significant contributor in many operational events. That is why the significant operating events are monitored from the human factor point of view in the second area.

The preparedness of the safety systems to cope with emergency situation is very important part of the NPP`s operation. In the area of the Safety System Performance the unavailability and reliability of the specific safety systems is assessed. Following systems were selected to be monitored: diesel-electric generators, high and low pressure injection system, spray system, accumulators, auxiliary and emergency feedwater systems. If the unavailability is evaluated for all those systems, the reliability is evaluated just for the first four of them.

The status of so called first and third barrier is by the means of the primary coolant radioactivity and the size of the confinement leakage monitored in the Barrier Integrity area.

SÚJB is carrying out also its regulatory activities in the field of the Radiation Protection and the same named area follows collective dose and radioactive discharges to the environment.

Setting up the indicators set raw data collection rules are to be established. There was a SÚJB activity to make collecting of the data as much independent on the utility as possible. SÚJB is using the capacity of its resident inspectors, who are monthly reporting to headquarters the needed information. The ratio between independently gained data and data provided by the utility is forty to sixty percent.

Safety Indicators process and some results of the annual evaluation

As there is still only one NPP in operation in the Czech Republic, the set of Safety Indicators is evaluated just for it. The evaluation of the Safety Indicators set is made on a quarterly/yearly basis and the basic element of the evaluation is a unit or system for the area. The plant values are the next step and they are done by the average of the unit values. In some cases when the value is only plant dependent there are no unit values. It is mostly used for the evaluation in the Radiation Protection area.

The complete results of the Safety Indicators set together with their analysis are published as an Appendix to the SÚJB Annual Report for Government [2]. It serves also as a basis for the top management of SÚJB and NPP Dukovany annual meeting assessing the previous year results. The trends of values in different area are of major interest. There was made and common agreement to present the results for six years period.

The figures in Appendix are pointing out a few results of the last year evaluation. The year 1999 was very successful operating year for NPP Dukovany, which was confirmed with that evaluation as well.

Figures 1 and 2 show the number of the reactor scrams for plant and units. The unit one has been operating for more than nine years without reactor scram.

Figure 3 presents the number of operational events ranked as INES 1 and the percentage of human factor failures taking part in those events on Fig. 4. One hundred of the operational reports ranking as INES 1 were due to human factor failures during the years 1997 and 1998. This was strongly criticized by SÚJB and licensee was demanded to pay more attention to human factor problem.

The diesel-electric generator system was only one safety system, which reached worse unavailability values than in previous years (Fig. 5). Another point of view at safety system unavailability is given by Type Specific Unavailability (Fig. 6). It is clear, that the biggest contribution to the total unavailability is done by tests.

At the end the Figures 7 and 8 give the evidence about very good results in the last two areas Barrier Integrity and Radiation Protection.

Risk based indicator study

SÚJB financed a Risk based indicators (RBI) study [2] in the years 1998 and 1999. The goal of the study was to find the way, how to use the PSA tools for evaluation of the data collected for Safety Indicators Set. There was no intention to develop a tool that should substitute the Risk Monitor.

The review of worldwide used techniques was conducted at the beginning. Coupling of the system unavailability time or initiating events occurrence with the risk-increased factors or the fractional contributions was basis of the three RBIs at the end. Those indicators are:

ECCS unavailability risk indicator,

Emergency electric systems risk indicator,

Initial Events criterion.

Figures 9 and 10 present the results of the trial evaluation of the two out of three RBIs. The trends of Indicator values are of the highest interest in that assessment similarly to the other Safety Indicators.

Conclusion

SÚJB has ten years experience with the using of the Safety Indicators set. The set is monitoring five areas of the NPP operation and it is very useful tool for safety assessment. Similar set of Safety Indicators is roughly prepared for NPP Temelin, which is now in the start-up period.

Three RBIs are still under trial use and further development will be discussed after evaluation of the year 2000.

References

[1] Safety Indicators, Radomir Rehacek, ISBN 80-7073-044-7, 1992, (in Czech)

[2] SÚJB Annual Safety Indicator Set Evaluation (year 1999), (in Czech)

[2] Risk Based Indicators, Jiri Sedlak, Nuclear Research Institute Rez, 1999, (in Czech)

APPENDIX

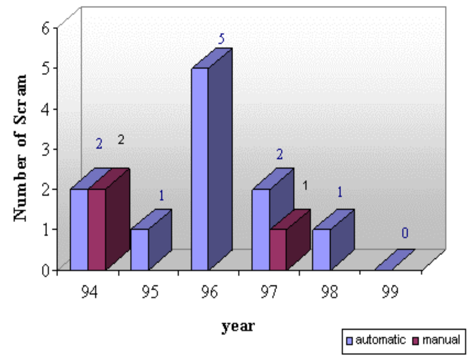


Fig. 1 Number of Scrams

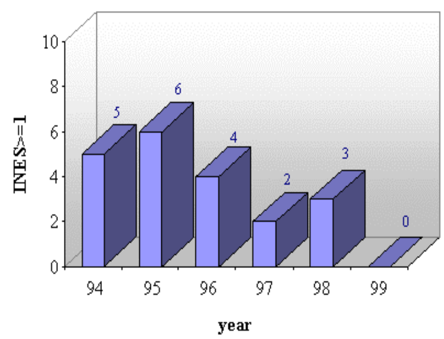


Fig. 3 Number of Operational Events

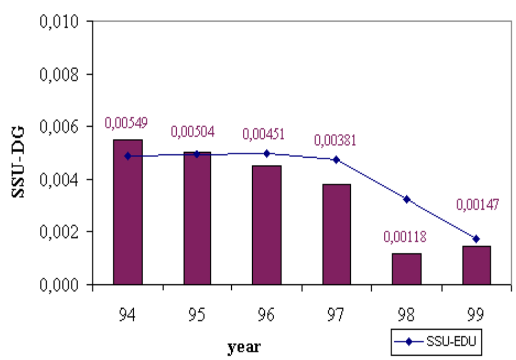


Fig. 5 Diesel-electric generator system Unavailability

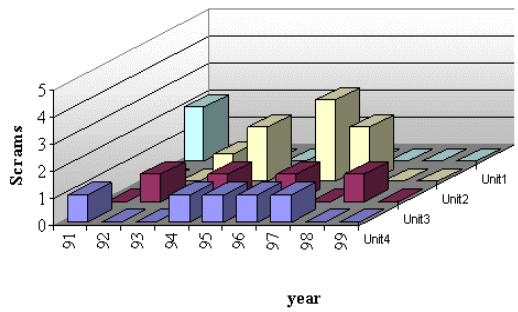


Fig. 2 Number of Scrams (Unit values)

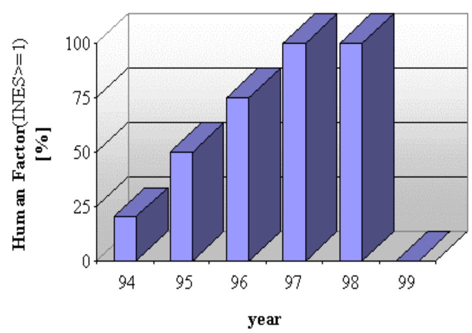


Fig. 4 Human Factor Failures

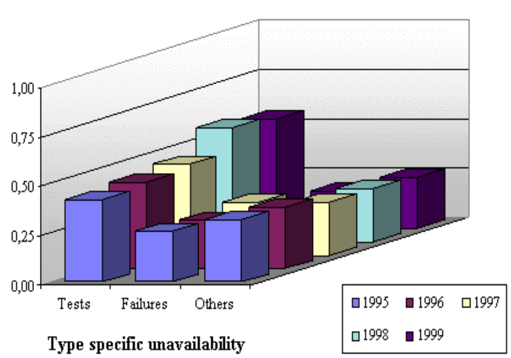


Fig. 6 Type Specific Unavailability of Safety Systems

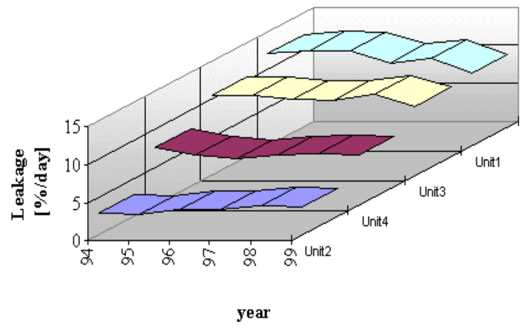


Fig. 7 Confinement Leakage

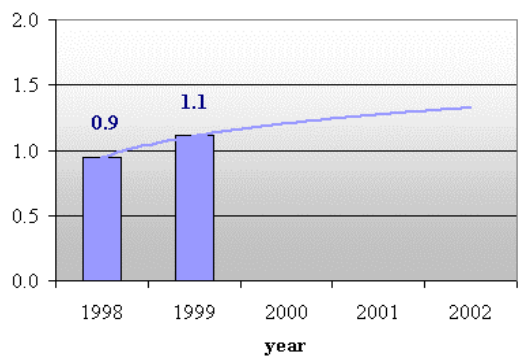


Fig. 9 Emergency electric systems Risk Indicator

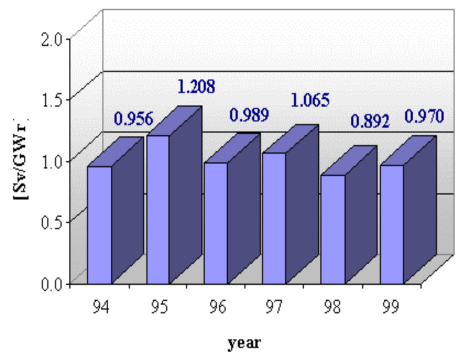


Fig. 8 Specific Collective Dose

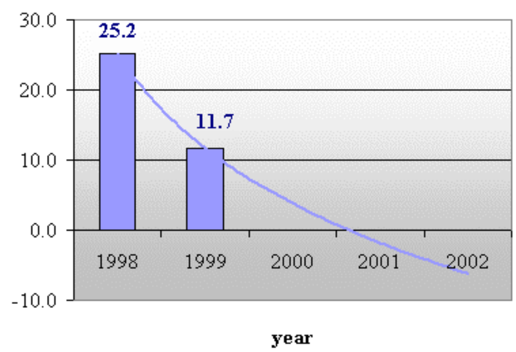


Fig. 10 ECCS unavailability Risk Indicator

-

-

-

SPECIALIST MEETING ON SAFETY PERFORMANCE INDICATORS

OECD Nuclear Energy Agency (NEA)

International Atomic Energy Agency (IAEA)

Consejo de Seguridad Nuclear (CSN)

Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT)

Madrid, Spain

October 17-19, 2000

TITLE OF THE PAPER: "COFRENTES NPP INDICATORS TO MONITOR OPERATIONAL SAFETY PERFORMANCE"

AUTHORS: Francisco Calduch. Iberdrola - Cofrentes NPP

Tef.: +34-961894300 - Fax: +34-962196477

e-mail: f.calduch@iberdrola.es

José Palomo. Iberdrola - Cofrentes NPP

Tef.: +34-915776500 - Fax: +34-915776228

e-mail: jose.palomo@iberdrola.es

CONTENTS Page

EXECUTIVE SUMMARY

The Cofrentes NPP management believed that there was a clear need to develope a comprehensive set of indicators to monitor plant operational safety performance beyond those WANO's indicators already deployed throughout the nuclear community.

IBERDROLA, Owner of the Cofrentes NPP, knew that the IAEA-Division of Nuclear Installation Safety was undergoing a significant effort in trying to develop such a more complete set of indicators since 1995. IBERDROLA joined the group of participating Organizations in this endeavour since 1996 and Cofrentes management offered the possibility of being the plant considered in the pilot programme to demonstrate the validity and completeness of the indicators framework as worked out in the consultant meetings held in Vienna during the past three years.

Since that it is recognized that other indicators may be in used or may be developed by each plant management specifically, it was agreed to give flexibility to each selected plant for the pilot exercise so as to obtain in each case a set of specific plant indicators following the guidelines established in the framework document.

Cofrentes began its pilot study in January, 1998.

Selection of specific indicators, clear and detailed definition of each one, organizational support (responsible people/owner for each indicator), development of the appropriate software for user-friendly information, and data sources and its collection were activities carried out during the first half of 1998.

It was also agreed with the Agency, to prepare three four-monthly and an annual report which will be supplied to the Scientific Secretary along the pilot exercise period.

The annual report was issued in March 99 to complete our goal and commitment and it contained the Pilot Exercise Programme results as carried out in Cofrentes NPP during 1998. The report also presented the detailed explanation of the development of the Programme, including the process of selecting and defining the Indicators, the Software Application used to calculate and depict such Indicators, as well as the selection of technical specialist responsible/owner for each indicator, data collection, formats filled-up, trend analysis, the channels to be used for data gathering and quality checking, and a general description of the four-monthly reports prepared during the exercise.

Based on the experience gained from the Pilot Execise itself along the year, and on the outcome of diferents workshops and meetings held in 1998, a final selection of Indicators was deployed in March 1999.

This paper summarizes the whole process as depicted above and the Pilot Exercise outcome and assessment, together with the conclusions obtained and a summary of the 1999 programme results evaluation.

Finally, in the Annexes of the paper, the complete set of specific indicators (in the way of list and chart), together with some specific examples of Indicators, as depicted in the application, are included.

The results obtained from the programme throughout the year, wich were shown in the four-monthly reports, were discussed with the corresponding responsible/owners of the Indicators and presented to the Cofrentes Executive Safety Oversight Committee, in its periodic meetings, for a decision making process in order to implement any corrective action, as necessary, as a consequence of observed deviations and/or proposed changes.

1. INTRODUCTION

IBERDROLA and its Cofrentes NPP have been participating in the IAEA's Programme: "Indicators to Monitor NPP Operational Safety Performance" since 1996. The purpose of it was to establish a very complete set of useful indicators so as to get a key valuable tool, at all tiers of plant management, for decision making based upon indicators trends and targets accomplishment.

The Cofrentes management offered to the IAEA Nuclear Safety Division the possibility for including the plant in its pilot study.

Cofrentes NPP was selected by the IAEA to be one of the three plants of different design and belonging to different continents, in order to develop a plant specific set of indicators and thus make it possible for the validation of the programme.

The necessary effort to implement the programme has been worthwhile since that the man-power cost associated with its development is more than compensated by the benefits of, first, having a significant tool for plant management feedback about areas needing attention, and, second, sharing experiences and information exchange with other nuclear power plant management throughout the nuclear community.

The plan and schedule for the Cofrentes Pilot Exercise was an agreement with the Agency's framework document, and it took fifteen (15) months. The pilot study began in January 1998 and finished in March 1999, and this paper presents a description of the whole programme, as well as a summary of the 1999 programme performance results assessment.

2. PILOT EXERCISE PROGRAMME

-

TIME	1998												1999		
ACTIVITY	JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP	OCT	NOV	DEC	JAN	FEB	MAR
Selection of indicators															
Review and definition of indicators															
Organizational support															
Data collection															
Development of application software															
First four-monthly report (Jan-Apr)															
Second four-monthly report (May-Aug)															
Third four-monthly report (Sep-Dec)															
Annual report															

Schedule for the Pilot Study at Cofrentes NPP

3. DEVELOPMENT OF THE PROGRAMME

Being aware of the importance to obtain an appropriate set of indicators that when properly used can be a very valuable tool to help the plant management in the safe operation of the Cofrentes NPP, IBERDROLA decided to participate in the IAEA's endeavor of developing and drafting the material to present a framework document of indicators to monitor NPP operational safety performance. Cofrentes was also chosen among others by the Division of Nuclear Installation Safety of the IAEA to make a pilot exercise so as to provide a feedback on the feasibility of the programme for being used to meet the needs of others plants in the future.

The diferent steps taken and activities performed during 1998 to carry out the development and implementation of the Programme are described in the following paragraphs.

3.1. SELECTION OF INDICATORS

The Cofrentes management began its programme to develop a plant specific set of indicators in January, 1998. A selection of indicators was made for Cofrentes NPP after a complete and thoughtful revision of the IAEA's framework programme, in which Cofrentes was a participant since 1996, proposing a reasonable set of performance indicators, and following the hierarchical structure presented in the main body of the working material,. This initial set of specific indicators was set up and sent to the IAEA's Scientific Secretary for this programme, in February 1998.

All the Cofrentes Units and Sections managers participated in both the selection and definition of these indicators, except for those already included in the Cofrentes Indicators Programme and WANO's Indicators.

3.2. DEFINITION OF EACH INDICATOR

A clear and simple definition of each indicator was considered a key part of the programme. As a matter of fact, the section leaders, who were going to be the responsible owners for data collection, tracking and trend analysis of each indicator, were directly participating in its definition as well as in the selection of targets and goals later on.

A presentation of this specific set of indicators and corresponding definitions was made to the Cofrentes Steering Committee for its approval before being sent to IAEA's Scientific Secretary in March 1998.

3.3. THE SOFTWARE APPLICATION

To allow and make easier the pursuit and data collection of selected Indicators, an application software was developed at Cofrentes NPP.

Each Indicator consist of two pages: The first one being for monthly pursuit and the second one for yearly pursuit.

The first page contains the following fields:

- Operational Safety Attribute.
- Overall Indicator.
- Strategic Indicator.
- Specific Indicator, which in its turn includes:
- Definition.
- Goal.
- Graphic display.
- Reference.
- Comments and Actions.
- Responsible/Coordinator.
- Monthly numerical Anticipated and Actual values.

3.4. ORGANIZATIONAL SUPPORT

3.4.1. Responsible Sections/Owners

A Responsible and/or Coordinator for each of the initial set of Indicators was appointed depending on the Area which the Indicator belongs to. Production, Operation, Engineering, Maintenance, Chemistry, Health Physics, Training, Q.A. and Licensing were the Sections involved in the process.

Licensing and Technical Support Section-leaders actuated as general Supervisors of the Programme.

3.4.2. Data collection

The Indicators were updated by the corresponding Responsible/ Coordinator, through the adequate collection of data carried out by its personnel, in a monthly basis.

Licensing and Technical Support, were in charge of attending the IAEA's workshops, preparing the four-monthly, annual and any other specific reports, as responsible of the whole Programme.

3.5. PLANT-SPECIFIC INDICATORS AND TARGETS

Once responsible owners were chosen for each plant-specific indicator, a meeting was held with anyone of them in order to clearly establish the sources of data to be compiled for every specific indicator.

On the other hand, the amount of data-years to be analyzed, depending upon its useful meaning to provide a goal measure of each indicator, was discussed so as to be able to set up 1998 targets which be at the same time challenging but achievable.

It was clear that figures showing indicators trend over a period of time, usually four-monthly basis, would provide an early warning to plant management for decision making, avoiding complacency and detecting incipient weaknesses.

Therefore, both numerical values and trends using the corresponding targets and diagrams would supply a good package of information for a complete indicator analysis by the responsible owner and plant management, as well as it would be presented in four-monthly reports for further distribution as appropriate. Each indicator is depicted in one sheet including all information needed for management analysis.

Data are presented for each month and plant-specific targets are shown based upon thoughtful consideration of the indicators results during the last three (3) to five (5) years where applicable. Comments are also introduced in those cases in which any detailed information can be useful.

3.6. FOUR-MONTHLY AND ANNUAL REPORTS

A total of three four-monthly reports were prepared in the year which corresponded to the months: 1st: January through April; 2nd: May through August and 3rd: September through December. Besides, the annual report, was issued in March 1999, summarizing the evolution of the Programme throughout the year, general comments and results and decisions taken as well.

The information gathered in these reports has been disseminated within the plant, for all tiers of Cofrentes management, in IBERDROLA and the Spanish nuclear units owned by IBERDROLA, and it was also sent to the Scientific Secretary of the IAEA-Division of Nuclear Installation Safety for this Programme, for which Cofrentes NPP was committed to make this pilot exercise.

3.6.1. Executive Safety Oversight Committee (ESOC)

The ESOC is the higher rank Committee to deal with operational safety issues and make recommendations to the Cofrentes General Manager for final decision making. Functions of the ESOC are contained in Cofrentes Technical Specifications Section 6.4.2.

The four-monthly and annual reports of Cofrentes indicators to monitor operational safety performance were distributed to the ESOC's members, and the Cofrentes Safety and Licensing Manager made a presentation at every meeting hereinafter to emphasize the highlights and results against objectives, trends and targets, in order to investigate the reasons behind any observed deviation and/or change.

Corrective actions, if any, were portrayed in the Committee's minutes of meeting and presented to the Plant General Manager for his final approval.

4. FINAL SELECTION OF INDICATORS

During the first quarter of 1999, according to the plan scheduled by the IAEA for the development of this Indicators Programme, the results of the Cofrentes NPP Pilot Exercise carried out during 1998, were assessed.

The Responsible/Coordinators of each Indicator were interviewed to request their opinion on how their corresponding indicators went on during the year, whether they had any proposal to delete or improve indicators, or modifications to the software application, targets, etc...

Based on that and on the outcome of some workshops and experiences exchanged with other Plants, the final set of Cofrentes NPP specific indicators was agreed and is presented in Annexes 1 and 2.

5. PILOT EXERCISE OUTCOME AND ASSESSMENT

The main objective of this pilot exercise was to validate the usefulness of the set of indicators to monitor operational safety performance, as depicted in the framework document of the IAEA, and provide the corresponding feedback to improve this tool, knowing that it was a meaningful information to our plant management for assessment of Cofrentes operational safety, as well as one more tool available for self-assessment.

We have learned through the programme implementation that, first of all, it was necessary to convey the message on the importance of this endeavour, convincing plant staff that these indicators were not "more of the same", or just additional work. There was a general impression that the plant was going to have too many indicators. Secondly, it was clear to anyone that the programme had the complete support of the plant director and the upper staff, as well as the awareness of the resources needed and process development by the Plant Nuclear Safety Commitee.

It was a good structure for the programme to involve Sections head and supervisors so as to have an active participation in the selection, definition, and identification of goals in the areas covered by the indicators, as well as the appointment for each indicator of a responsible "owner" in addition to Licensing and Technical Support designees as general supervisors of the programme.

The additional effort required to develop and run the programme was less significant than thought at the beginning because Cofrentes already had implemented an indicators programme covering many aspects of plant performance and due to a good distribution of work as mentioned before.

Some dificulties were encountered in adapting the proposed set of indicators to the plant characteristics and in defining them clearly and simply. However, the direct participation of the Section heads and/or supervisors who are the responsible owners for each indicator, facilitated quite a bit the process.

The final set of selected indicators was made based upon the suggestions received by the responsible owners in operating the programme and, of course, through the analysis of information feedback from the meetings held in Vienna and Lubljana.

The assessment of results and comments was made considering our plant-specific set of indicators evolution throughout the year, as presented in the four-monthly reports, emphasising highlights and programme outcomes compared to the established targets and taking into consideration the indicators trend. Any significant deviation and/or change was investigated to identify any reason behind it, and take the corresponding corrective actions in the Executive Safety

Oversight Commitee.

A cluster of additional indicators were proposed and approved for the completeness of the whole set.

These added indicators were as follows:

- Emergency Preparedness.
- Material condition (Ageing).
- State of Barriers (RCS and Containment Leakages).
- Number of Findings in Configuration Control.

To summarize about the programme results, we may say that:

- Improvements are clearly recognized in the old set of plant performnace indicators as result of this pilot exercise, not only by having a real complete set of indicators to monitor operational safety performance but also because of the team-work environment created in the people who participated in this programme development.
- The additional effort devoted to this study was more than compensated by the benefits obtained from it as a useful tool for our Sections head and supervisors as well as for the plant management.
- Plant-specific goals were chosen based on past plant operational experience and data by the indicator responsible owner. The goals were established as real challenges but achievable.
- Some weak points were identified as a feedback of this pilot exercise, pinpointing the areas needing further attention by the Cofrentes management.

6. 1999 PROGRAMME RESULTS EVALUATION

A meeting was held at the begining of 2000 between the two general supervisors of the IAEA Indicators Programme, i.e. Licensing and Technical Support, in order to proceed to do a global evaluation of the Programme performance during 1999, after the end of the Pilot Exercise, in March of the same year.

The main goal of this evaluation being to inform to the Executive Safety Oversight Committee (ESOC) about the outcomes obtained in the year for the Programme as a whole and for those specific Indicators whose figures and/or trends during the year were seen as worthy mentioning, so as to have the possibility by this Committee of requesting possible corrective actions.

The overall result of the Programme can be considered as highly positive including both aspects: The Indicator values achieved in relation to the assigned goals, and their implementation process along the year by the respective responsible/owners.

Some of the comments, conclusions and/or suggestions presented to the ESOC, as consequence of this assessment of the Programme, were as follows:

- Number of Reportable Events: 5 (The goal for 1999 was : 10).
- Number of Scrams: 0 (Goal: 1).
- Safety Index during the Outage: 147 (Goal: 670).
- The possibility of removing from the list the Indicator "Forced Power Reductions and Outages due to External Causes" was considered. The reason being that the Plant has not any influence on this Indicator.
- The Indicator "Number of hours devoted to Training in the Nuclear Organization" is going to be complemented by adding an specific indicator for the Training of each Section.

Some other minor changes to several Indicators were commented with the corresponding owners.

To sum up, it can be said that the IAEA Safety Performance Indicators Programme is, as of today, completely implemented at Cofrentes NPP, and the results of its performance during the last two years were extremely encouraging.

7. CONCLUSIONS

The Cofrentes Pilot Study to validate the applicability and viability of the approach for implementation of a programme to monitor its operational safety performance has been accomplished with successful results.

The set of indicators finally established, after including plant insights as well as new thoughts and comments from the IAEA´s meetings held throughout 1998, is considered as a very valuable tool at all tiers of plant management for sound decision making based on indicators data and trends. It is a good complementary programme to others developed at Cofrentes NPP such as, self-assessment deployment, continuous quality improvement, safety culture strengthening, etc., to help the plant management in handling nuclear safety avoiding complacency and detecting incipient weaknesses.

General and plant specific objectives were fulfilled about the feasibility of the programme, its usefulness and management feedback on plant performance and weak points to pay attention to. Improve understanding of plant section leaders regarding this set of indicators has been achieved in spite of the large number of indicators depicted and because of its comprehensiveness.

We believe that this programme can be considered as the basis for development of an indicators subset to which the Spanish Regulatory Body and Plants management, through the Spanish Nuclear Energy Committee of UNESA, could reach an agreement on a new assessment framework that builds upon specific indicators to monitor operational safety performance.

The new set of indicators and its clear thresholds will constitute the key material for making performance based and risk informed inspection and it will ensure adequate Regulators oversight and assessment of licensee performance.

This future assessment process will drive into results which can be communicated to the public based upon objective conclusions. It will finally help to focus Regulators and Operators attention to measure NPP´s safety programmes outcomes to:

- maintain safety.
- reduce unnecessary burden.
- increase public confidence.
- increase efficiency / effectiveness of key processes.

8. ANNEXES

ANNEX-1

<u>SET OF INDICATORS (LIST)</u>

PLANT OPERATES SMOOTHLY

- ■ OPERATING PERFORMANCE

➢ FORCED POWER REDUCTION AND OUTAGES

1. Forced Power Reduction and Outages due to Internal causes.

2. Idem due to External Causes.

3. Plant Capability Factor (WANO).

4. Unplanned Capability Loss Factor (WANO).

- STATE OF SSC (STRUCTURES, SYSTEMS AND COMPONENTS)

➢ CORRECTIVE WORK ORDERS ISSUED

5. Number of Corrective Work Orders (CWOs) for Safety Systems.

6. Number of Total CWOs Issued.

7. Ratio of Preventive Orders vs. Preventive + Corrective Work Orders (P.O./P.O. + CWO).

8. Lag time between CWOs Issuance and Closing.

Dates: Urgent < 2 days

Important < 4 days

Normal < 30 days

➢ MATERIAL CONDITION

9. Chemistry Index (WANO)

10. Aging

➢ STATE OF BARRIERS

11. Fuel reliability (WANO)

12. RCS Leakage

13. Containment Leakage

- EVENTS

➢ REPORTABLE EVENTS

14. Number of Significant Reportable Events.

15. Number of Licensing Event Reports.

PLANT OPERATES WITH LOW RISK

- CHALLENGES TO SAFETY SYSTEMS
  - INITIATING EVENTS

16. Unplanned Scrams/7000 hrs. (WANO).
17. Number of Demands RPS/ECCS/RHR and EPS.
18. Number of Demands of Other Safety Systems.

  - POTENTIAL CHALLENGES

19. Number of RPS/ESFAS Failures.
20. Number of Other Safety Systems Failures.

- PLANT ABILITY TO RESPOND TO A CHALLENGE
  - SAFETY SYSTEMS PERFORMANCE

21. Number of hours a Safety System is Unavailable.
22. Number of times a Safety System is Unavailable.
23. Safety Systems Performance (WANO).

  - OPERATOR PREPAREDNESS

24. Number of hours devoted to Training.
25. Number of hours in Simulator Training for Licensed Operators

  - EMERGENCY PREPAREDNESS

26. Findings during Emergency Plan (EP) Audits.
27. Number of hours devoted to training on the EP.
28. Number of people receiving training on the EP.

- LANT CONFIGURATION RISK
  - RISK DURING SHUTDOWN

29. Severity Index.

➢ RISK DURING PLANT OPERATION

30. Instantaneous Risk (CDF1): Short term.

31. Average Risk (CDFA): Long term.

32. Conditional Probability of Core Damage per Initiating Event Frequency vs. Total Core Damage.

$$(CPCDF)_{IE} = \frac{\sum CDF_{IE}}{CDF_{TOTAL}}$$

PLANT OPERATES WITH A POSITIVE SAFETY ATTITUDE

- ATTITUDE TOWARDS SAFETY

➢ COMPLIANCE WITH PROCEDURES, RULES AND LICENSING REQUIRE

33. Number of Violations of Licensing Requirements.

34. Number of Technical Specifications Violations.

35. Number of Technical Specifications Exemptions.

36. Number of Deviations found in QA-Audits Related to Pro-cedures.

➢ ATTITUDE TOWARDS PROCEDURES, POLICES AND RULES

37. Number of Lit Control Room Annuntiators.

38. Number of Temporary Modifications.

39. Ratio of Downtime to Allowed Outage Time.

40. Number of findings in Configuration Control.

41. Number of total Deviations found in QA-Audits.

➢ RADIATION PROTECTION PROGRAMME EFFECTI-VENESS

42. Number of workers receiving Dose above Limits.

43. Collective Radiation Exposure (WANO).

44. Effluent Activity vs. Allowed Limits.

- HUMAN PERFORMANCE

45. Percent of Licensing Event Reports (LERs) due to Human Error.

46. Percent of LERs due to Training Deficiencies.

47. Percent of LERs due to Procedures Deficiencies.

- BACKLOG OF SAFETY RELATED ISSUES

48. Number of Pending Licensing Commitments-Analysis Phase.

49. Number of Pending Licensing Commitments-Imple-mentation Phase.

- SAFETY AWARENESS

50. Percent of Plant Staff Trained in Safety Management & Safety Culture.

51. Number of Seminars on Safety Related Matters.

52. Percent of Plant Staff Attendants to Safety Related Matters Seminars.

53. Number of Plant and Executive Safety Committee Meetings.

54. Frequency of Self-Evaluation in Safety Culture.

■ STRIVING FOR IMPROVEMENT

- SELF-ASSESSMENT

55. Number of Independent Internal Safety Systems Inspection and Evaluation (ISEG).

56. Number of QA-Inspections and Audits.

57. Number of Findings from ISEG.

58. Pending Findings with Overdue Date

(Time greater than 3 mo/6 mo/1 yr).

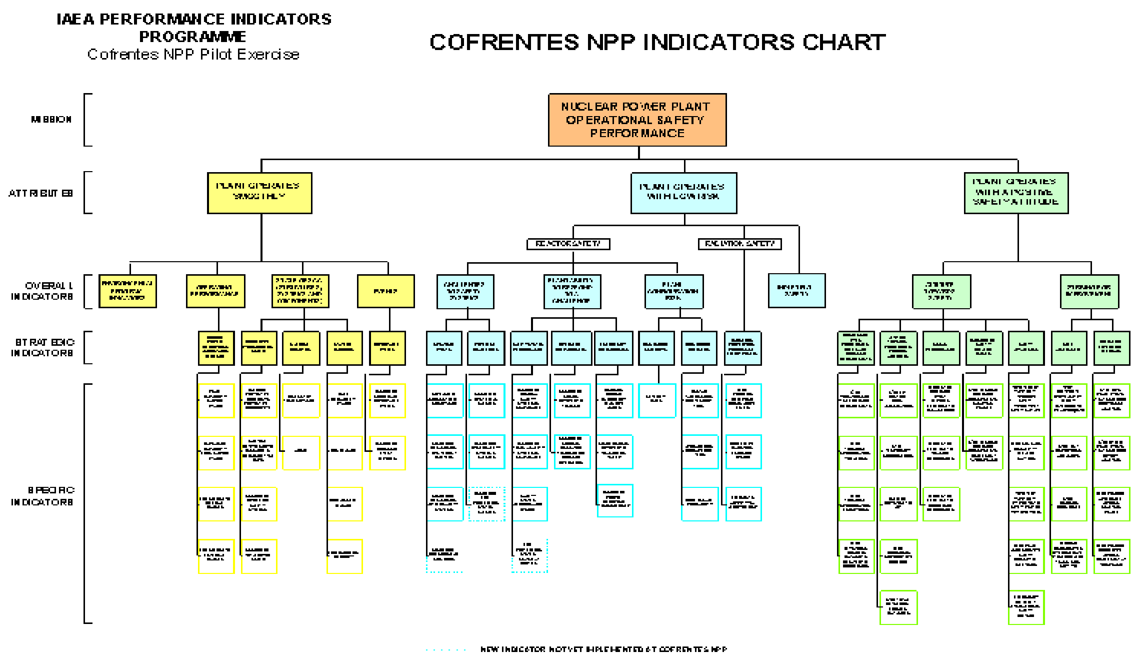
- OPERATING EXPERIENCE FEEDBACK

59. Number of Own Plant Events that Undergo Root-Cause Analysis.

60. Number of Other Plants Events that Undergo Review Analysis.

61. Number of Pending Applicable Actions-Analysis Phase.

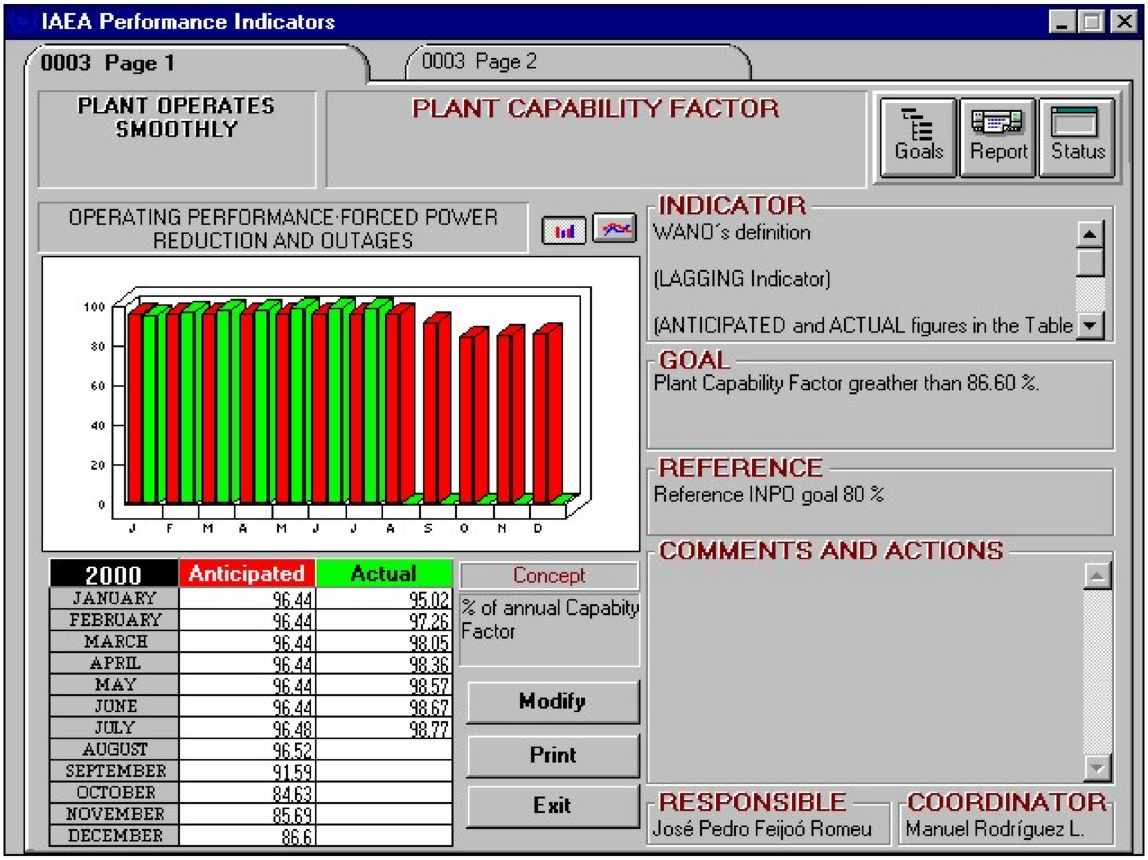
62. Number of Pending Corrective Actions-Implementation Phase.



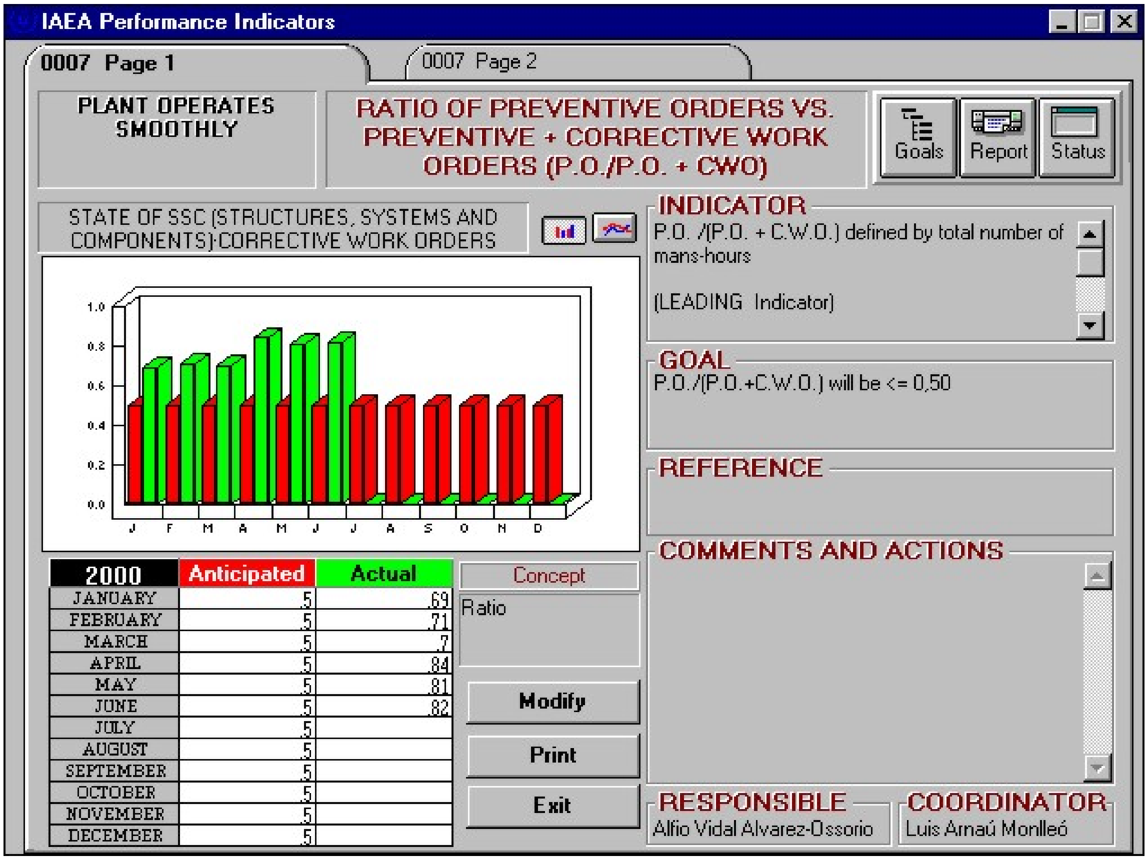
ANNEX-2

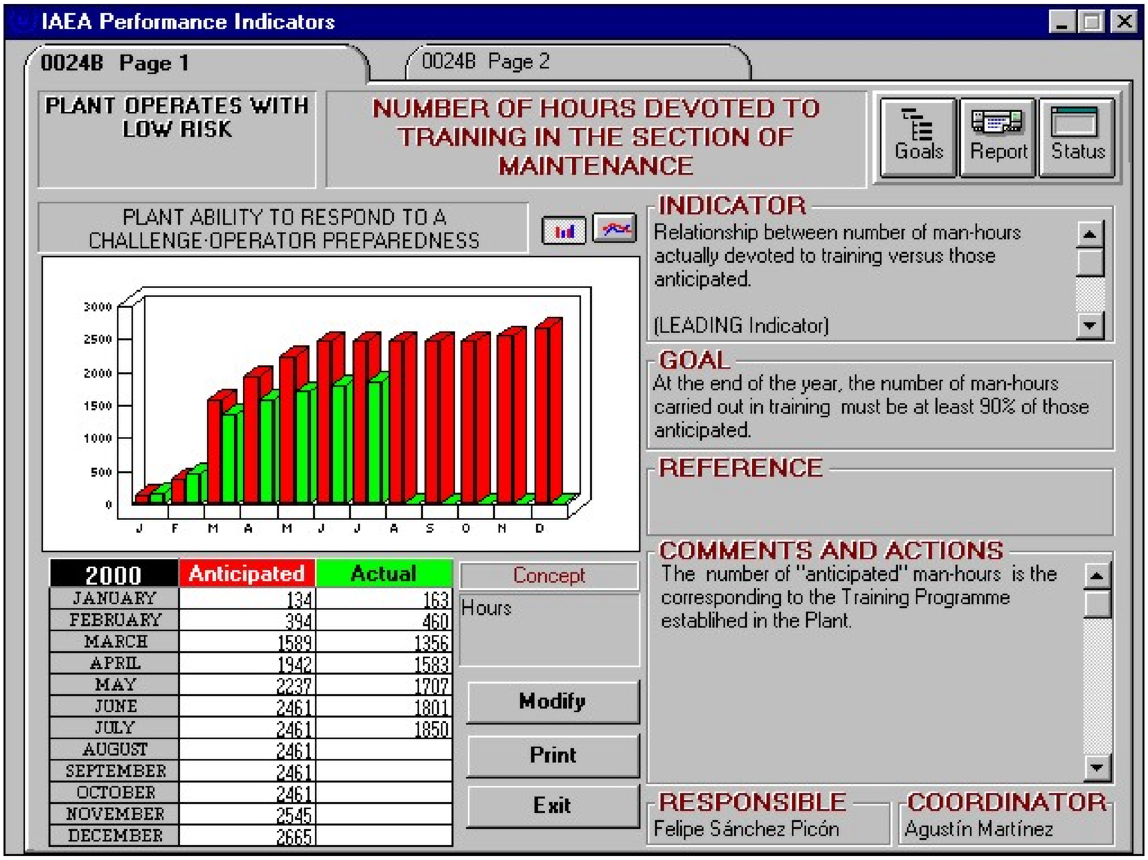
ANNEX-3

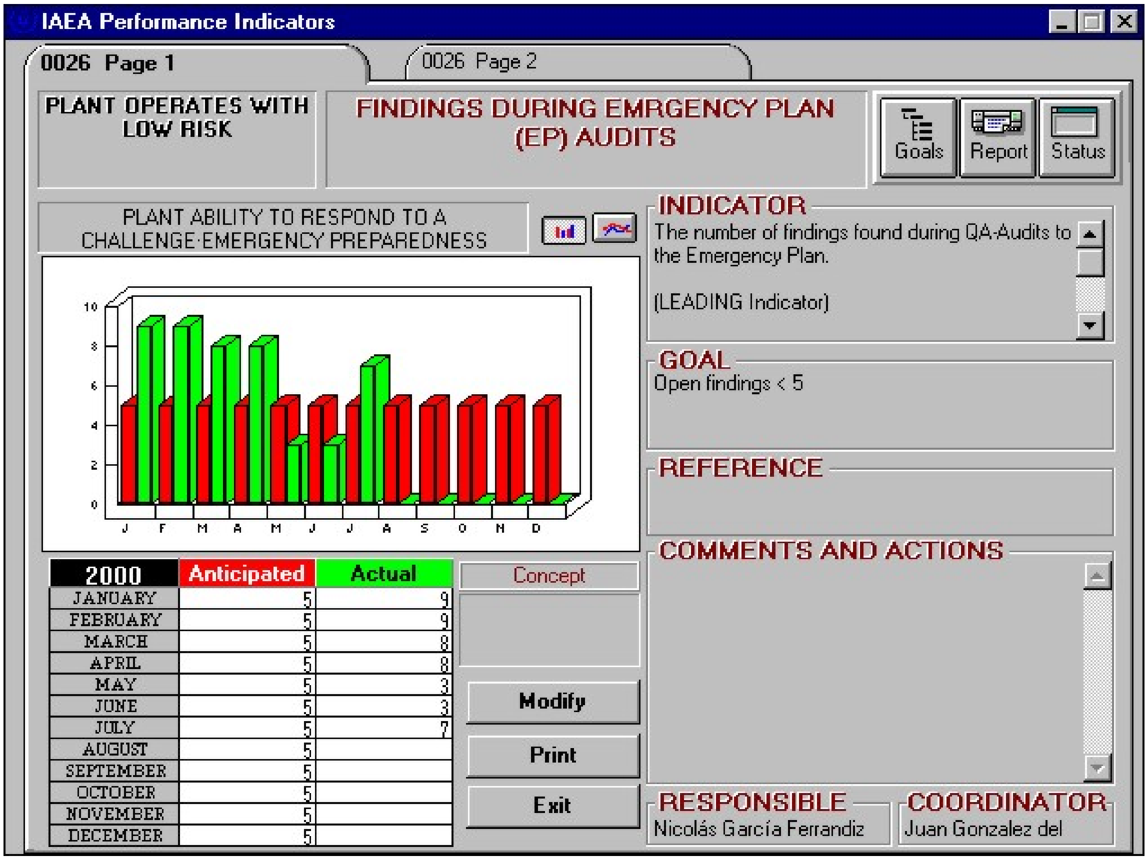
Some examples of Specific Indicator



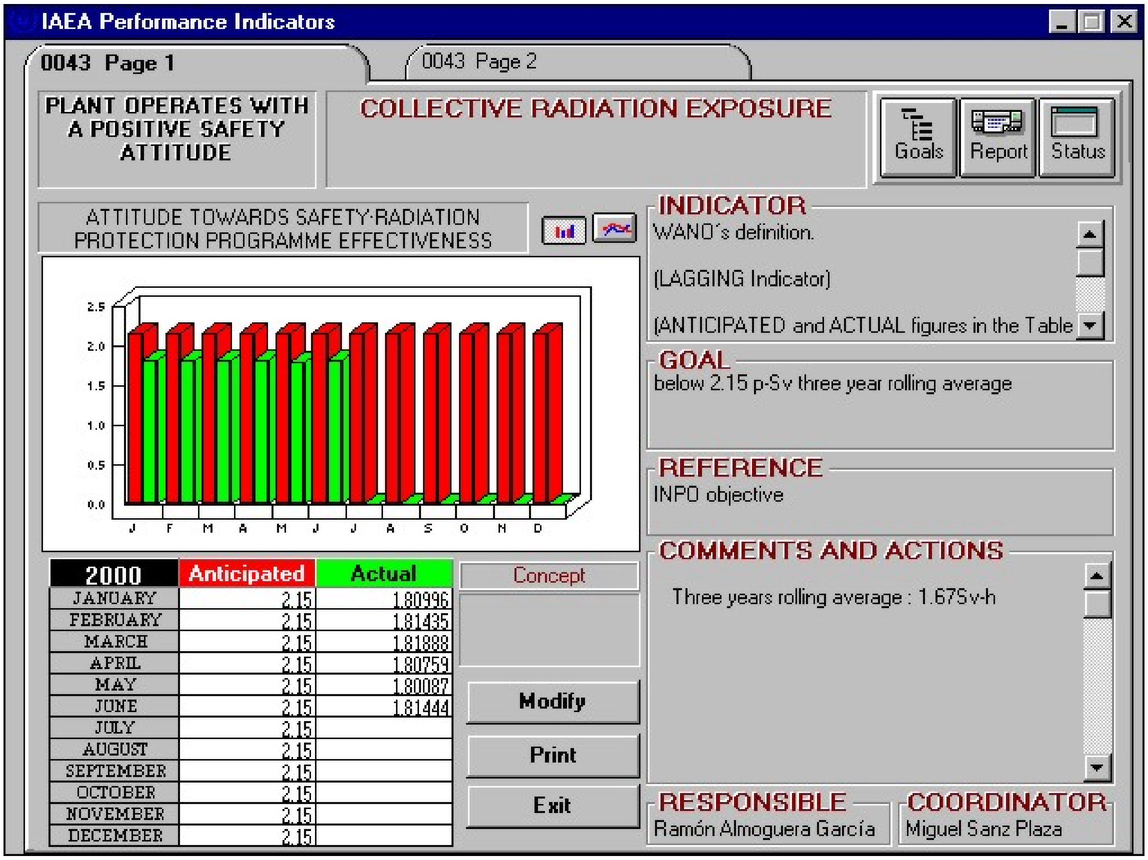
IAEA Performance Indicators
0003 Page 1
0003 Page 2
PLANT OPERATES SMOOTHLY
PLANT CAPABILITY FACTOR
Goals
Report
Status
OPERATING PERFORMANCE·FORCED POWER REDUCTION AND OUTAGES
INDICATOR
WANO´s definition
(LAGGING Indicator)
(ANTICIPATED and ACTUAL figures in the Table
GOAL
Plant Capability Factor greather than 86.60 %.
REFERENCE
Reference INPO goal 80 %
COMMENTS AND ACTIONS
2000
Anticipated
Actual
JANUARY 96.44 95.02
FEBRUARY 96.44 97.26
MARCH 96.44 98.05
APRIL 96.44 98.36
MAY 96.44 98.57
JUNE 96.44 98.67
JULY 96.48 98.77
AUGUST 96.52
SEPTEMBER 91.59
OCTOBER 84.63
NOVEMBER 85.69
DECEMBER 86.6
Concept
% of annual Capabity Factor
Modify
Print
Exit
RESPONSIBLE
José Pedro Feijoó Romeu
COORDINATOR
Manuel Rodríguez L.


IAEA Performance Indicators
0007 Page 1
0007 Page 2
PLANT OPERATES SMOOTHLY
RATIO OF PREVENTIVE ORDERS VS. PREVENTIVE + CORRECTIVE WORK ORDERS (P.O./P.O. + CWO)
Goals
Report
Status
STATE OF SSC (STRUCTURES, SYSTEMS AND COMPONENTS)-CORRECTIVE WORK ORDERS
INDICATOR
P.O. /(P.O. + C.W.O.) defined by total number of mans-hours
(LEADING Indicator)
GOAL
P.O./(P.O.+C.W.O.) will be <= 0,50
REFERENCE
COMMENTS AND ACTIONS
1.0
0.8
0.6
0.4
0.2
0.0
J F M A M J J A S O N D
2000
Anticipated
Actual
JANUARY .5 .69
FEBRUARY .5 .71
MARCH .5 .7
APRIL .5 .84
MAY .5 .81
JUNE .5 .82
JULY .5
AUGUST .5
SEPTEMBER .5
OCTOBER .5
NOVEMBER .5
DECEMBER .5
Concept
Ratio
Modify
Print
Exit
RESPONSIBLE
Alfio Vidal Alvarez-Ossorio
COORDINATOR
Luis Arnaú Monlleó


IAEA Performance Indicators
0024B Page 1
0024B Page 2
PLANT OPERATES WITH LOW RISK
NUMBER OF HOURS DEVOTED TO TRAINING IN THE SECTION OF MAINTENANCE
Goals
Report
Status
PLANT ABILITY TO RESPOND TO A CHALLENGE-OPERATOR PREPAREDNESS
3000
2500
2000
1500
1000
500
0
J F M A M J J A S O N D
INDICATOR
Relationship between number of man-hours actually devoted to training versus those anticipated.
(LEADING Indicator)
GOAL
At the end of the year, the number of man-hours carried out in training must be at least 90% of those anticipated.
REFERENCE
2000 Anticipated Actual
JANUARY 134 163
FEBRUARY 394 460
MARCH 1589 1356
APRIL 1942 1583
MAY 2237 1707
JUNE 2461 1801
JULY 2461 1850
AUGUST 2461
SEPTEMBER 2461
OCTOBER 2461
NOVEMBER 2545
DECEMBER 2665
Concept
Hours
Modify
Print
Exit
COMMENTS AND ACTIONS
The number of "anticipated" man-hours is the corresponding to the Training Programme establihed in the Plant.
RESPONSIBLE
Felipe Sánchez Picón
COORDINATOR
Agustín Martínez



IAEA Performance Indicators
0026 Page 1
0026 Page 2
PLANT OPERATES WITH LOW RISK
FINDINGS DURING EMRGENCY PLAN (EP) AUDITS
Goals
Report
Status
PLANT ABILITY TO RESPOND TO A CHALLENGE-EMERGENCY PREPAREDNESS
10
8
6
4
2
0
J F M A M J J A S O N D
2000 Anticipated Actual
JANUARY 5 9
FEBRUARY 5 9
MARCH 5 8
APRIL 5 8
MAY 5 3
JUNE 5 3
JULY 5 7
AUGUST 5
SEPTEMBER 5
OCTOBER 5
NOVEMBER 5
DECEMBER 5
Concept
Modify
Print
Exit
INDICATOR
The number of findings found during QA-Audits to the Emergency Plan.
(LEADING Indicator)
GOAL
Open findings < 5
REFERENCE
COMMENTS AND ACTIONS
RESPONSIBLE
Nicolás García Ferrandiz
COORDINATOR
Juan Gonzalez del


IAEA Performance Indicators
0043 Page 1
0043 Page 2
PLANT OPERATES WITH A POSITIVE SAFETY ATTITUDE
COLLECTIVE RADIATION EXPOSURE
Goals
Report
Status
ATTITUDE TOWARDS SAFETY-RADIATION PROTECTION PROGRAMME EFFECTIVENESS
2.5
2.0
1.5
1.0
0.5
0.0
J F M A M J J A S O N D
INDICATOR
WANO´s definition.
(LAGGING Indicator)
(ANTICIPATED and ACTUAL figures in the Table
GOAL
below 2.15 p-Sv three year rolling average
REFERENCE
INPO objective
COMMENTS AND ACTIONS
Three years rolling average : 1.67Sv-h
2000
Anticipated
Actual
JANUARY 2.15 1.80996
FEBRUARY 2.15 1.81435
MARCH 2.15 1.81888
APRIL 2.15 1.80759
MAY 2.15 1.80087
JUNE 2.15 1.81444
JULY 2.15
AUGUST 2.15
SEPTEMBER 2.15
OCTOBER 2.15
NOVEMBER 2.15
DECEMBER 2.15
Concept
Modify
Print
Exit
RESPONSIBLE
Ramón Almoguera García
COORDINATOR
Miguel Sanz Plaza

IAEA Performance Indicators

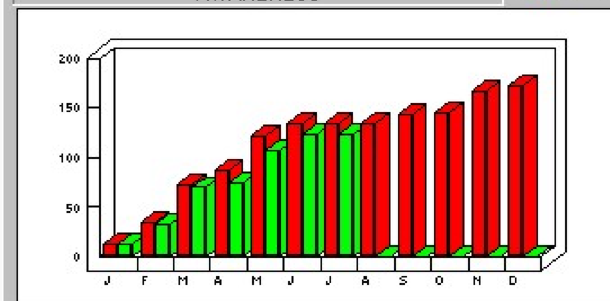
0051 Page 1 | 0051 Page 2

PLANT OPERATES WITH A POSITIVE SAFETY ATTITUDE

NUMBER OF SEMINARS ON SAFETY RELATED MATTERS

Goals | Report | Status

ATTITUDE TOWARDS SAFETY·SAFETY AWARENESS



2000	Anticipated	Actual
JANUARY	12	12
FEBRUARY	35	33
MARCH	73	70
APRIL	87	74
MAY	122	107
JUNE	134	124
JULY	134	124
AUGUST	134	
SEPTEMBER	144	
OCTOBER	146	
NOVEMBER	167	
DECEMBER	173	

Concept

Number

Modify

Print

Exit

INDICATOR

Self-definition (number/year).

(LEADING Indicator)

(ANTICIPATED and ACTUAL figures in the Table

GOAL

At the end of the year, the number of seminars carried out on safety related matters will be at least 90% of those anticipated.

REFERENCE

COMMENTS AND ACTIONS

The number of "anticipated seminars" is the corresponding to the Training Programme established in the Plant.

RESPONSIBLE

Felipe Sánchez Picón

COORDINATOR

Agustín Martínez

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

Indicators of plants performance during events identified by recuperare method

Geneviève Baumont IPSN/DES/SEREP B.P 6
92265 Fontenay-aux-roses cedex France
tel:01.46.54.82.44/fax: 01.46.54.35.60
genevieve.baumont@ipsn.fr

Sophie Bardou IPSN/DES/SEREP B.P 6 92265
Fontenay-aux-roses cedex France
tel:01.46.54.88.34/fax: 01.46.54.35.60
sophie.bardou@ipsn.fr

0 0 0 0 0 0

Naoëlle Matahri IPSN/DES/SEREP B.P 6
92265 Fontenay-aux-roses cedex France
tel:01.46.54.79.01/fax: 01.46.54.35.60
naoelle.matahri@ipsn.fr

0 0 0 0 0 0

Abstract

The authors present a model developed for operating experience feedback analysis, built on the French HRA principles. This model puts emphasis on the recovery process during events and makes a classification of events based on the ways by which errors and recovery are linked. 900 events reported by EDF had been reviewed through this model. Some results are presented and some safety indicators of plants performance are proposed.

introduction

As part of its mission as expert adviser to the Safety Authority, the Institut de Protection et de Sûreté Nucléaire (Nuclear Protection and Safety Institute - IPSN) examines operating events that could affect safety in the nuclear facilities of which it is in charge. Within this framework, the Safety Authority has requested the French electricity board (EDF) to systematically report the most important events, known as "significant incidents for safety", in accordance with a required and homogeneous declaration procedure used by all French nuclear power plants. The IPSN thus has homogeneous incident reports which it analyses on an ongoing basis. By way of indication, it should be noted that EDF operates 58 nuclear reactors and declares, on average, 400 "significant incidents for safety" per year.

IPSN has developed various methods for analysis of operating feedback in order to optimize extraction of lessons from significant incidents and has created several databases serving as memory aids for incident analysis. Until now these databases focused preferentially either on technical factors or on human ones, and on the description of the causes and the consequences for facility safety. Analysis of operating feedback generally aims to reduce the frequency of occurrence of causes or the severity of consequences.

But reducing consequences of deficiencies by a quick and correct recovery is also a way to improve safety.

So, IPSN developed the RECUPERARE method aiming at the following objectives:

1. identify the main mechanisms and parameters which characterize events, occurring on French PWRs during one year,
2. provide a way of classifying deficiencies and associated recoveries,
3. provide a way of classifying events according to previous parameters,
4. record these data in a base to make trend analyses.

Finally some safety indicators of performance derived from this method are presented in this paper.

PAPER CONTENTS

1. A new model for events description

1. Static model

The concept underlying the RECUPERARE model (see the figure below) presented here is the development of an "incident model" based on the "accident model" from "Probabilistic Safety Analysis" (PSA), structuring the data normally studied in operating experience feedback.

This model describes:

Ú the emergence of the fault (human errors, technical failure or organizational failing);

Ú the characteristics of the incident development;

Ú the recovery of the situation which is based on the human system and/or safeguarding automatic systems.

The detailed model structure and the fields studied in each event are presented in the appendix.

2. Dynamic model

The dynamic aspect of the model relies on the analysis of the fault-recovery connection by introducing a temporal reference which is the actuation of the system concerned by the fault.

The notion of the fault-recovery pair leads to the proposed classification of incidents into six families:

Type A: latent fault discovered and corrected before activation of the system in question.

Type B: latent fault recovered after activation of the system in question.

Type C: fault appearing when the system is already in service.

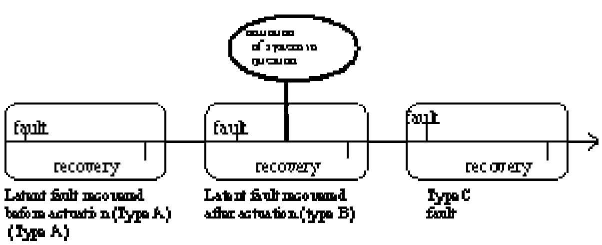
Type D: cumulative faults (e.g. repetition of same error, fault during correction of an initial fault, strategic fault generating various inappropriate actions).

Type O: organizational fault without direct impact on facility or process.

Type R: event related to radioprotection.

These six families make it possible to characterize the various types of histories outlined in the significant incident reports for a year and to monitor developments in the nature of incidents.

The figure below illustrates the principle of the classification of the first three families.



2. Safety indicators

Given the abundant data studied, the results are multiple and make it possible to respond to different types of questions which IPSN analysts may ask. The base makes it possible to obtain breakdowns of the incidents according to several criteria; graphs of detection and recovery times according to various factors (for example the dependency between contingent error factors and contingent recovery factors) or families of incidents.

Some of these results can be used as safety indicators of plants performance:

· the annual frequency of human errors and technical failures as well as their context of occurrence;

· the "response times" of operators in relation to detection of problems and to their recovery.

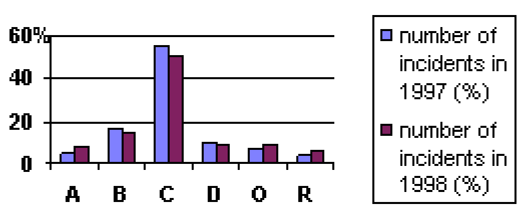
The period of latency, diagnosis and recovery of situations, considered as performance aspects, are studied. This point is particularly important as the consequences of an incident can be more or less serious, depending on recovery-of-incident-situation performance.

Some results and indicators are illustrated below, from 1997-1998 analyses.

NB: The results are presented in a relative manner. The graphs were made using samples containing at least 20 elements.

2.1 General breakdown of incident families

From year to year the breakdown of incidents into the different families varies only slightly (e.g. there are around 25 per cent of latent faults). This seems to be a consequence of reactor design or mode of operation, and organization of the operator's activities – elements that vary little from one year to the next. This stability in breakdown validates the initial choice of the different families. For example, it can be noted that, each year, cumulative incidents represent around 10 per cent of the total. This stability makes it possible to measure the effect of major modifications to organization or in facilities by comparison of results. The breakdown of incidents into six families is presented below.

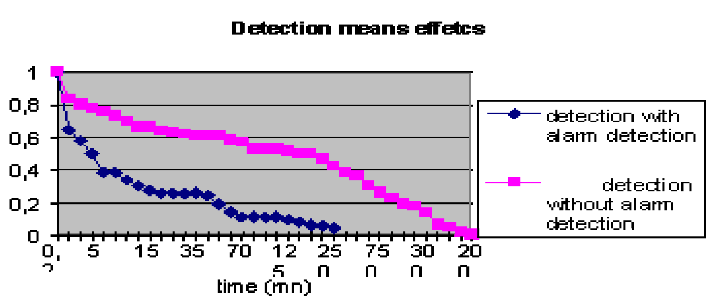


2.2 Authors of errors, fault detection and fault recovery

"Recovery" actors are generally members of the operating crew and especially operators, who correct more errors than they initiate. It has also been noted that instrumentation and control personnel are actors in correction of 20 per cent of incidents.

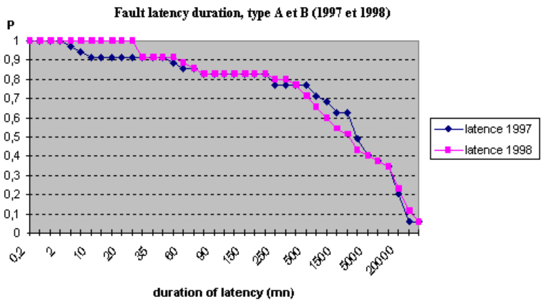
Author (%)	Error	Detection	Recovery
0 0 Operation	54.4	62.1	52.4
0 0 Multi services/authors	15.3	11.9	17.7
0 0 Maintenance	13.6	5.9	3.4
0 0 Safety engineer	0	5.9	0
0 0 I and C technician	11.4	5.5	19.7
0 0 Chemist	1.7	1.7	2.0
0 0 Other	1.7	3.0	1.4
0 0 Tester	1.3	3.0	1.4
0 0 Electrician	0.4	0	2.0

0 0



2.3 Fault latency periods

It should be remembered that a fault is considered "latent" if it is present in a system before activation of the system without being detected. This type of fault is especially important as the safety of facilities can only be ensured if all of the systems important for safety (especially the safeguard systems) are actually available, i.e. they do not have latent faults. The figure below shows that the 1997 and 1998 graphs for probability of failure to detect a latent fault over time correspond when overlaid, although the faults in question are, no doubt, very different in nature. This would lead us to the hypothesis whereby it is not technical or fault-specific factors that enable detection, but rather factors due to the organization of activities and modes of operation and monitoring.

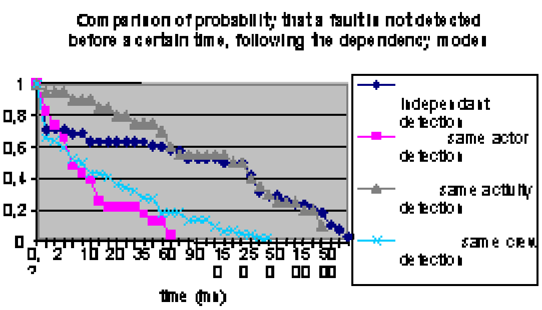


2.4 Factors influencing detection and recovery times

For 1997 and 1998, the graphs of probability of failure to detect faults "during action" (type C faults) are very similar. The factors most influencing detection times were identified and their effect quantified (means of detection, dependency mode, etc.).

- dependency (type C)

For 17 per cent of incidents for which information was given in the report, the actor "recovered" (i.e. corrected) the error in less than 20 minutes in 80 per cent of cases. When it was the actor's crew or an independent actor who corrected the error, correction was slower.



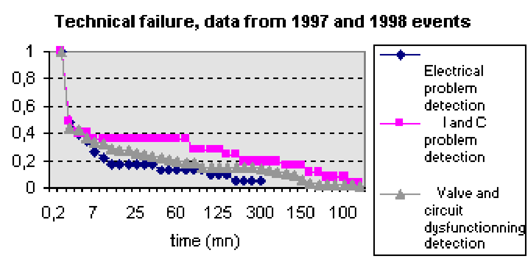
- means of detection (type C)

- fault in procedure

In one-fifth of incidents, the procedure used showed a fault and the impact on recovery time could be measured.

- technical failure

Concerning the technical failures, we can see that the nature of the material concerned influences the detection time. For example, an Instrument and Control (I and C) failure is detected within more time than an electrical problem.



3. Link between impact on safety and detection and recovery times

Analysis showed that incidents regarded as the most significant for safety by the IPSN analysts – due to their human context and potential consequences on safety (without concertation with the authors of the study) – are incidents for which detection or recovery is particularly lengthy. This result shows that detection and recovery times are probably pertinent indicators of safety performance.

4. Different uses of indicators

4.1 French nuclear power plants "safety log"

We have seen in the previous parts that results of RECUPERARE method can provide a panel of different indicators for the whole population of incidents within a year. The comparison between different years displays different types of indicators:

- Time-stable indicators draw global characteristics of nuclear invents in France. For example, each year, 10% of events are type D events.

- Variations of other indicators show downgrading and ameliorations in the global safety of the French power plants. For example, the number of organizational failures shows a real increase from 1997 to 1999.

4.2 Safety level assessment on a particular power plant

Experience feedback analyses may focus on a particular power plant. In this type of analyses, RECUPERARE results can be used in two different ways:

- The evolution of indicators over the years, for one particular power plant, can outline its weak points and progress concerning causes, gravity and recovery of events. Moreover, as the results are based on the event reports, this type of analyses can also show the plant motivation for understanding, explaining and correcting a specific issue.

- The comparison, for each indicator, between a particular plant and the rest of the French plants can give its position among the others. In this way, bad and good practices can be outlined and specific problems may appear.

5. Conclusion

This study opens up new lines of analysis for operating feedback and provides new safety indicators of plants performance. Although nuclear power plants do not all give the same type of information and some data are missing, the large amount of available data makes it possible to discern trends and profiles for the nature of incidents over a year.

Thus, this method provides tools to establish an overview of incidents each year and to characterize equipment performance during correction of incident situations.

Furthermore, the results can be used to draw up a log from one year to the next if certain relevant parameters are chosen, making allowing to form an overall opinion based on objective data as to the safety of facilities over the years.

These data were analyzed using a very recent statistical method, i.e. PLS method, which overcomes the problem of missing data. The purpose of this new study was to attempt to "predict" the detection times from factors that are contingent to the situation. The initial results are very encouraging and should be the subject of future articles.

For the probabilistic studies, although use of these results in PSS is still a question, it should be noted that this approach appears promising for better:

1. identification of relevant data for analysis of dependency between faults and actors (cumulative situations and analysis of actor dependence allow study of the failure and recovery mechanisms);
2. specification of the pre-accident context;
3. analysis of pre-accident fault latency;
4. obtaining of a realistic idea of the times required for detection and action.

References

1. Amalberti, R., La conduite de système à risques, Le Travail Humain, Paris, PUF., 1996.

2. Baumont G. Rapport DES 371 " Modèle et méthode Recuperare" (French and English version)

3. Baumont G, Matahri N " Résultats de l'application de la méthode Recuperare aux incidents de 1997 et 1998 (in discussion)

4. Faverge, J.M., "l'homme, agent de fiabilité et d'infiabilité" Ergonomics, Vol 13, N°3, p. 301-327, 1970.

5. Faverge, J.M., Le travail en tant qu'activité de récupération". Bulletin de Psychologie, Vol 33, N°344, p 203-206, 1980.

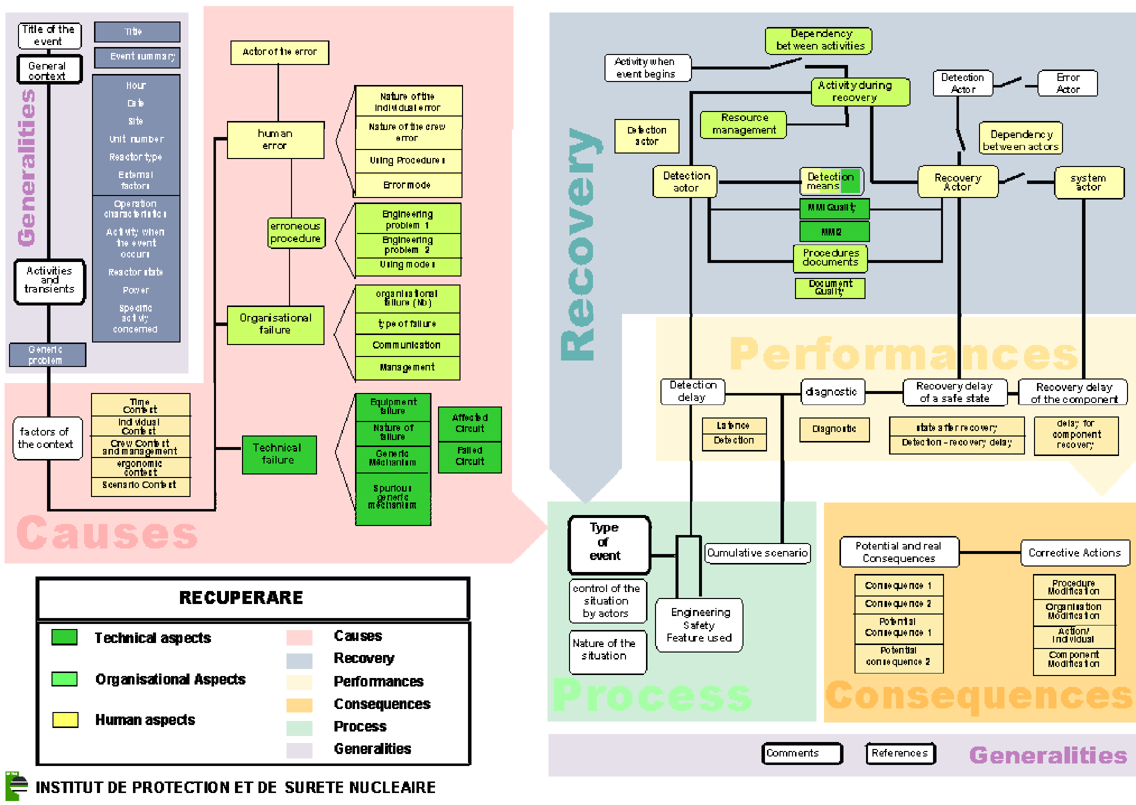
6. Mosneron D.F., Saliou G. & Lars R.. Probabilistic Human Reliability Analysis, the lessons derived for plant operation at Electricité de France. In Proceedings of the PRA'91 International Symposium on P.S.A. for operational Safety, IAEA, Vienna, (pp. 625-638). 1992.

7. NUREG/CR 6093, An analysis of operational experience during low power and shutdown and a plan for adressing human reliability assessment issues. 1994.

8. NUREG-1624 Draft for Comment, Technical Basis and Implementation Guidelines for a Technique for Human Event Analysis (ATHEANA), May 1998.

9. Reason; J., l'erreur humaine, le travail humain, P.U.F., Paris, 1993.

Appendix : detailed model


Title of the event
Title
General context
Event summary
Hour
Date
Site
Unit number
Reactor type
External factors
Operation characteristics
Activity when the event occurs
Reactor state
Power
Specific activity concerned
Activities and transients
Generic problem
Generalities
factors of the context
Time Context
Individual Context
Crew Context and management
ergonomic context
Scenario Context
Actor of the error
human error
Nature of the individual error
Nature of the crew error
Using Procedures
Error mode
erroneous procedure
Engineering problem 1
Engineering problem 2
Using modes
Organisational failure
organisational failure (Nb)
type of failure
Communication
Management
Technical failure
Equipment failure
Nature of failure
Generic Mechanism
Spurious generic mechanism
Affected Circuit
Failed Circuit
Causes
Recovery
Dependency between activities
Activity when event begins
Activity during recovery
Detection Actor
Error Actor
Resource management
Detection actor
Dependency between actors
Detection actor
Detection means
MMI Quality
MMI
Procedures documents
Document Quality
Recovery Actor
system actor
Performances
Detection delay
diagnostic
Recovery delay of a safe state
Recovery delay of the component
Latence Detection
Diagnostic
state after recovery
Detection - recovery delay
delay for component recovery
Type of event
Cumulative scenario
control of the situation by actors
Engineering Safety Feature used
Nature of the situation
Process
Potential and real Consequences
Corrective Actions
Consequence 1
Consequence 2
Potential Consequence 1
Potential consequence 2
Procedure Modification
Organisation Modification
Action/ Individual
Component Modification
Consequences
Comments
References
Generalities
RECUPERARE
Technical aspects
Organisational Aspects
Human aspects
Causes
Recovery
Performances
Consequences
Process
Generalities
INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, Spain

Assessment of Human Performance and Safety Culture at the Paks Nuclear Power Plant

János TÓTH, Paks NPP

Department of Nuclear Safety,

H-7031, Paks, POB 71, Hungary

phone: +36 75 50 79 21

fax:+36 75 50 82 97

e-mail: tothj@npp.hu

Lajos HADNAGY, Paks NPP

Department of Nuclear Safety,

H-7031, Paks POB 71, Hungary

phone: +36 75 50 80 12

fax:+36 75 50 82 97

e-mail: hadnagyl@npp.hu

Abstract

Evaluation of human performance and safety culture of the personnel at a Nuclear Power Plant is a very important element of the self assessment process. At the Paks NPP a systematic approach to this problem started in the early 90's. The first comprehensive analysis of the human performance of the personnel was performed by the Hungarian Research Institute for Electric Power (VEIKI). The analysis of human failures is also a part of the investigation and analysis of safety related reported events. This human performance analysis of events is carried out by the Laboratory of Psychology of the plant and a supporting organisation namely the Department of Ergonomics and Psychology of the Budapest University of Technical and Economical Sciences.

The analysis of safety culture at the Paks NPP has been in the focus of attention since the implementation of the INSAG-4 document started world-wide. In 1993 an IAEA model project namely "Strengthening Training for Operational Safety" was initiated with a sub-project called "Enhancement of Safety Culture". Within this project the first step was the initial assessment of the safety culture level at the Paks NPP. It was followed by some corrective actions and safety culture improvement programme. In 1999 the second assessment was performed in order to evaluate the progress as a result of the improvement programme. A few indicators reflecting the elements of safety culture were defined and compared. The assessment of the safety culture with a survey among the managers was performed in September 2000 and the results are being evaluated at the moment.

The intention of the plant management is to repeat the assessment every 2-3 years and evaluate the trend of the indicator.

introduction

Even modern technical activities require good performance of human during task performance. In-spite of high level safety systems and devices, the role of personnel cannot be ignored in nuclear facilities, and even in other industrial plants, transports etc. Importance of evaluation of operator performance, investigation of human errors was recognised early at Paks Nuclear Power Plant. Efforts were also made in order to improve the level of operator performance. Implementation of recommendations made by safety analyses and project for improvement of safety culture are the most important of those.

The definition of direct indicators for the assessment of human performance and safety culture is not simple. The human and the organisation is a very complex "system". Any correction taken in order to improve the "system" performance cannot promptly change the behaviour of the human. This is not like the performance of a technological component which, if corrected, will improve the performance immediately, and the indicator reflecting the performance of the concerned component will change immediately too. To realise the change in the human performance of individuals or the safety culture of the organisation needs time. If one would like to define indicators for assessing human performance or safety culture he should be very careful and the indicator should be monitored in a longer time scale since these indicators respond to the corrections more slowly.

Human performance evaluation

Evaluation of operator performance

Projects for evaluation of operator performance apart from serving as input for safety analyses had other important results, as well. They helped the plant operational and safety management to specify the safety level of operator activities and gave some suggestions how to improve it or how to decrease the contribution of human performance to the overall plant safety. By this the results provided a kind of indicator that have a direct relationship with the safety.

First assessment of human performance was required by the AGNES (Advanced General and New Evaluation of Safety) Project of the Paks NPP.

The work related to the operator performance evaluation was carried out by VEIKI in autumn of 1992. Its general objectives were establishment of a comprehensive human failure (HF) data base and specification of performance shaping factors (PSF).

After preparation of the necessary procedure the data collection was performed on the full scale simulator of the Paks NPP with different shift crews. During this work 120 exercises were demonstrated for operators in order to have information about their behaviour during emergency conditions.

The Periodic Safety Review (PSR) of units 1 and 2 was required by the regulatory body in order to have further operating licence of these units and it included the operator performance evaluation.

The work carried out was a kind of extension of which was performed during the AGNES Project and had the objective of more exact specification of level of the human reliability and processing of possible safety enhancement solutions.

During the practical data collection three additional transients were examined:

- medium LOCA with partial ECCS actuation (LOCA);
- main steam line break outside the confinement (MSLB);
- load drop to the self consumption level (LDSC).

During performing this task high performance quality of the personnel was proved. (General evaluation see in Figure 1.) However, the results identified weaknesses in the next areas: transient diagnosis, use of procedures, communication during complex events.

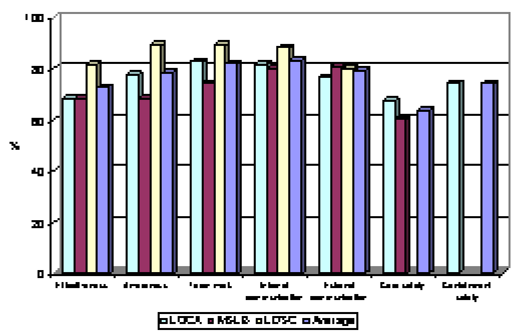


Figure 1: Evaluation of operator performance during simulator training at Paks NPP in 1996

The subsequent recommendations stressed the importance of simulator training and proper use of procedures. In order to eliminate contribution of failed operator diagnosis the Paks NPP prepared and in 2001 will implement a new symptom based emergency operational procedure package. The I&C refurbishment of the reactor protection system (unit 1 & 2) enhanced the man-machine interface, as well.

HRA for the shut-down and low power safety assessments can be considered as a new important task. This complex activity was carried out in a framework of a DOE-VEIKI project during 1997-2000. Its objectives were identification of HE modes and performance shaping factors (PSF) and also identification of operator actions and responses. The source of information were the results of event investigations, data sheets of personal errors. A lot of information was collected about maintenance performance during outages based on expert judgement.

Among the results of this project factors with the highest contribution to safe human performance during outages were defined. Those are: working conditions and workload. Therefore in 1999 the new plant management announced that safe, organised and calm maintenance work performance has an overall importance during outages.

Event investigation

Analysis of human errors during different operational events has been performed for a long time. Currently this task is performed in the framework of the regular event investigation process according to the related written procedure.

Based on international practice a plant specific Root Cause Analysis (RCA) method was prepared and implemented at the Paks NPP. The RCA is very useful for definition of direct and main cause(s) of events. Concerning human failures the RCA method contains three different personnel type cause groups (personal, man-machine interface, training). Based on statistics, no significant and unambiguous trend can be recognised in annual number of events with human error. (See Figure 2.)

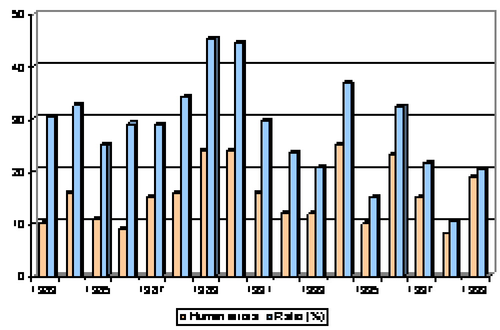


Figure 2: Number and ratio of events with HE

at the Paks NPP during 1983-99

Within the regular process of event investigation and specification of causes of events more detailed assessment of human failures is being carried out. The information about HE is collected using a Personnel Error Sheet (PES) filled in by the operator just after the event anonymously. Information of these sheets are to be processed by the plant Laboratory of Psychology (LOP). As an example, the next statement can be made for the 1999 year according to the main results of this work:

- working under stress, lack of practical experience, bad systematisation of information have low contribution to operator failures

- too much self-confidence, undisciplined behaviour have higher contribution to operator failures

In order to have independent analysis of HE events and well established recommendations for enhancement of operator performance, events with personnel failure will be assessed by an outside contractor in the future. The planned work includes reassessment of the existing investigation process, the PES content, events with human failure for the last few years and will give suggestions how to improve the HE evaluation process in order to find weaknesses and to decrease possibility of personnel failure.

Assessment of safety culture

Definition of Safety Culture

The first kind of definition of safety culture is given in the INSAG-3 document (Basic safety principles for Nuclear Power Plants) published by the IAEA in 1988. According to this document the safety culture is "something" that governs the actions and interactions of all individuals engaged in activities related to safety. It was understood that both the individuals and the organisation have a major role in the safety culture of the nuclear power plant.

The INSAG-4, document (Safety Culture) addressed the safety culture in more details and responded to the international interest in the expansion of the concept of safety culture. The members of the INSAG tried to come to a common consensus on the meaning of safety culture and the result was the INSAG-4 report. The definition of safety culture given in that document is:

> ***"Safety culture is that assembly of characteristics and attitudes in organisations and individuals which established that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance."***

According to INSAG-4 the safety culture has two general components. The first is the necessary framework within the organisation and is the responsibility of the management. The second is the attitude of staff at all levels in responding to and benefiting from the framework.

When assessing the safety culture at Paks apart from the IAEA definition some other approaches were taken into consideration as well. Such approaches and terms are: the "organisational culture" used by Deal and Kennedy as well as Peters and Waterman in their books published in 1992, Reason in 1997 defined the terms "reporting culture", "just culture", "flexible culture" and learning culture" as a subcategory of the safety culture in terms of interpretation of safety related events.

Enhancement of Safety Culture at the Paks NPP

The level of safety culture has been a primary concern at the Paks NPP since the term "safety culture" was first used in the nuclear industry and the international nuclear community started focusing on it. It actually came into the light of attention after the first ASSET and WANO missions carried out at Paks. In 1993 the plant management initiated a programme for the improvement of safety culture. Eventually an IAEA Model Project titled „Strengthening Training for Operational Safety" at Paks NPP was launched in 1994. This project comprised of three major elements:

- Systematic Approach to Training
- Maintenance Training Centre
- Enhancement of Safety Culture

The project of Safety Culture Enhancement was completed in a close collaboration with the IAEA and the experts of the Paks NPP. A special taskforce was established at the plant with the representation of the different plant disciplines and a systematic work started.

As a result of the project the first main actions to improve Safety Culture were the following:

- Plant Safety Policy and Safety Culture Guiding Principles were issued.
- Operational and safety indicators were summarised and explained in a brochure, which was distributed among plant employees.
- A Workshop on Safety Culture was organised by the IAEA. The participants represented the plant management and staff as well as the Regulatory Body and the involved institutes.

- Training material including examples of good safety culture practices was developed for selected jobs such as maintenance shift foreman and shift supervisor.

- A safety culture survey was carried out for employees of the operations, radiation protection, maintenance and engineering staff.

The last item is the one which established the basis for the regular assessment and monitoring of the safety culture level at Paks.

Safety culture indicator

To define one single indicator that can precisely characterise the level of safety culture of an individual, an organisation or the whole industry is not easy. The safety culture indicator (if we can define one) cannot be fully separated from the other indicators included into the usual commonly used indicator systems. Many of those in some extent give indication of the safety culture (human failure probability, procedure adherence, technical specification violations, repeated events, backlogs etc.) Probably the safety culture of one individual person has no meaning. Individuals have to be considered as part of the organisation and be evaluated in complex. Safety culture indicator is not a simple indicator, it is a function of many variables. Important contributing factors are the clear safety policies and management expectations with priority of nuclear safety, sound procedures and adherence to procedures, implementation of self-assessment and reviews and staff training and education. All these and many more factors should be considered when safety culture is evaluated. The INSAG-4 document provides a tentative list of questions to be asked in order to get some indication about the safety culture level of the organisation. The provided list of questions is only example, it can be expanded and tailored to the specifics of the concerned country or organisation. The areas/organisations to be evaluated span from the governmental organisation through the operators, regulators down to the research and design institutes.

Safety culture assessment at Paks

As part of the above described safety culture improvement project an initial evaluation of the safety culture level was performed in 1994. The objectives of the assessment were the following:

- Define the basic level of the safety culture at the Paks NPP using the self-assessment method outlined in the INSAG-4 document. Use the results as a reference for future analysis.

- Assess the attitude of the plant personnel towards nuclear safety

- Identify main features of good practices related to safety

- Identify corrective actions which can be included into the improvement programme.

In 1999 a repeated assessment was performed using practically the same approach. The same questions were used for surveying the staff of the same plant disciplines. Contrary to the 1994 survey - which was carried out by the experts of the plant - in 1999 a professional organisation the Department of Ergonomics and Psychology of the Budapest University of Technical and Economical Sciences was contracted for performing the survey and analysing the results. During the project a close collaboration was established between the contractor and the concerned plant section.

Method of the assessment

For the selection of individuals to be questioned standard statistical methods were used that means:

- random selection of the personnel was assured from each area (plant discipline)
- equal representation of staff from the different disciplines so that the result can give real reflection of safety culture in the concerned area
- at least 30 people were selected from each area.
- individuals from contractors were selected as well

The survey covered employees of the Operations, Maintenance, Radiation and Engineering disciplines.

The assessed components

As it is stated in the definition the safety culture has two basic components (the framework within the organisation and the responsibility of the management and the attitude of staff at all levels). Of course there is an interaction between these components and the technology. Taking into account the ASCOT guidelines and the SOL (Safety through Organisational Learning) method the target of the assessment within the two major components were the:

- organisation
- group/team
- individual
- technology and
- environment

The assessment was carried out using questionnaires and interviews. (A simplified questionnaire is provided in the appendix).The questionnaires were distributed among all the selected individuals but only a limited number of people were interviewed. The answers to the questions were the major source of the information for the assessment whereas the interviews provided additional information.

Based on the above the questions covered the following topics:

Regarding the framework in which the individual is located:

- Safety policy of the plant (operating organisation)
- Safety practices
- Responsibilities
- Training

- Safety related events and training
- Selection of managers
- Safety attitude of the managers
- Consequences of reported events
- Collaboration of the different organisational units
- Balance of the safety end production (economics)
- Documentation
- Decision support
- Working environment

Regarding the individuals' safety approach and reactions:

- Information load
- Physical load
- Personal "safety strengths"
- Personal safety weaknesses"
- Personal opinion about the possibilities of safety improvement
- Near-miss events

Analysis of the results

The information provided in the questionnaires were analysed by the *SPSS for Windows* computer code with a systematic input data checking by statistical hypothesis analysis. All the answers to the more than 80 questions were statistically analysed the feasibility of the data was assessed by cross checking the answers to different questions and eventually a value was calculated for each individual variable (question). After this a numerical value was calculated for the groups of questions and eventually a single indicator was processed for the different plant disciplines and an overall indicator for the whole plant.

The main indicator values and the comparison of the results of the 1994 and results are given in the following table:

	1994 %	1999 %
Safety culture level	61	77

The safety policy is clear	69	67
Safety practices are good	64	64
Training activities are appropriate	70	68
Attention to safety	55	78
Workload	58	60

The main findings of the 1994 assessment

After the first assessment in 1994 the deficiencies in the plant activities having influence on the safety culture were identified in order to define corrections into the then ongoing safety culture improvement programme

Communications deficiencies were identified in the technical area as well as in the area of communicating company's strategic questions, personal performance results and safety related actions to the plant employees.

Training deficiencies were identified such as

- inadequate managers' involvement in the training
- training procedures' quality
- deficiencies in emergency drills, industrial, fire and radiation safety practices
- inadequate focus on human factor's importance
- quality of training materials
- inadequate focus on the importance of safety policy

The following actions responding to the findings were taken in 1995-96:

The safety culture working group started to work out new development strategies taking into account international experience.

- the content of safety culture training package for managers was defined
- team sessions were organised to develop questioning attitude
- subject proposals were prepared for IAEA working group meetings
- example training materials were prepared in order to show how safety culture elements could be reflected in training materials

- direct support was given to the SAT working group in job competency analysis
- training of operational and maintenance staff in order to introduce the STOP method.
- Development of training objectives and examination questions regarding safety related elements of the general employee competency;
- Co-operation with psychological laboratory in defining the elements effecting human errors and the elements hindering communication in the organisation;
- Preparation work of the company level safety culture development programme;
- Continuing teamwork supporting the development of questioning attitude for selected professional areas;
- Determination of requirements and preparation of training materials on human factor;
- Developing a model for the instructors conducting the analysis.

The results of the 1999 assessment

In course of the analysis of the results of the 1999 survey a detailed correlation analysis was carried out on all levels of the assessment. The conclusions of the correlation analysis on the top level of the assessment are the following:

If

- the relationship between the managers and subordinates is good
- the atmosphere at the work places is good
- the safety supervisory activity is good
- the workers are involved in the solution of safety related questions
- the training responds to the workers' demands

then the workers

- recognise their responsibility
- adhere to safety rules (even with missing supervisor)
- apply self-checking method
- implement in practice the principles of safety policy.

General conclusions of the 1999 survey

The first conclusion is that the safety culture level has virtually increased since the first assessment performed in 1994. This improvement can be attributed to the positive effect of the Model Project's activity. However such an improvement is much higher than can be expected for such an indicator which has a very huge inertia as mentioned in the beginning of this paper. Since the survey was carried out not too long after the appointment of a new management, part of the positive effect can be explained by the fact that the changes of the management caused positive expectations (this is the subjective nature of this indicator).

Another general conclusion that can be drawn is that the safety culture is highly influenced by

management attitude, by good communication between the employees of different organisations and by training and good practices.

The interviews highlighted some additional interesting facts which should be taken into account when defining corrective actions. Such facts are the following:

- The documentation system of the plant is complicated
- The organisational structure of the plant changes too frequently
- Differences between the plant employees and the contractors

The results of the assessment were presented to the managers and employees not participating in the survey.

Current and future activities

In December 2000 a similar survey was carried out with the participation of the whole management. The approach is identical, the analysis is performed by the same professional organisation. The questions are slightly modified in order to take into account the specifics of the management. The analysis of the results is going on at the moment.

In order to continue the monitoring of the safety culture similar surveys will be performed regularly in the future.

References

1. Bareith, Z. Karsa et al.: Evaluation of Reliability of Operators of NPP. VEIKI Research Reports, December 1994, October 1996, Budapest
2. Bareith et al.: Paks Shut-down Data Collection and Analysis for HRA, Reports. VEIKI Institute for Electric Power Research, Tasks 2-5, 1998-2000, Budapest
3. International Atomic Energy Agency, International Nuclear Safety Advisory Group: 75-INSAG-3, IAEA, Vienna, 1988
4. International Atomic Energy Agency, International Nuclear Safety Advisory Group: 75-INSAG-4, IAEA, Vienna, 1991
5. Report on Comprehensive Assessment of Safety Culture Level Performed at Paks NPP. Technical University of Budapest, Faculty of Ergonomics and Psychology, 1999, Budapest.
6. J. Tóth: Safety Culture Review at the Paks NPP. Presented on IAEA TCM, November 1999, Karachi

Appendix 1

Assessment of safety culture

Questionnaire

Explanation

How the questionnaires were filled in?

Below the question tables with rating scale are given. The rating considered to be appropriate had to be marked with "X". When answering some questions in addition to the rating the participants were asked to give written answers as well (text or precise numbers).

Examples:

1. If your answer to the question is "satisfactorily" fill in like this:

1. Not at all	2.	3. Satisfactorily	4.	5. Fully
		X		

2. If your answer to the question is "85%" fill in like this:

1. Never	2.	3. 50% of the cases	4.	5. Every time
			X	

Please give a quantitative assessment to your answer: 85%

Basic data

- What is your professional area?
- Who is your employer? (plant employee or contractor)
- How long have you been working in this position?
- How many years have you worked at nuclear power plant?
- What is your qualification?

General part/Safety policy

Before the questions on safety policy the following two definitions were given:

- Safety Policy – reflects the commitment to act always with objective to improve safety, to reveal factors affecting safety, includes the efforts to continuously improve safety culture and plant safety level.

- Safety Culture –assembly of characteristics and attitudes in organisations and individuals, considered to be significant from safety point of view

General part/Safety policy

- Can it be declared, that the company has a safety policy?
- How are you familiar with the plant safety policy?
- Is the plant safety policy understandable and clear?
- To what degree are you able to apply the principles defined in plant safety policy in your daily work?
- Do you and persons working with you adhere to safety procedures even if it cannot be checked by your supervisor?
- Do people working with you try to encourage others to work safely?
- Is it typical that your supervisors from time to time take efforts to make you understand the plant safety policy ?
- Considering the definition and questions 6-12 in your opinion can it be declared that at Paks NPP the level of safety culture is high?

General part/Safety practice

- Is it usual that you become involved into resolution of safety related questions?
- Do you think it is worth reporting those minor events, which could have led to more significant consequences?
- Do you think it is true that honest admission of committed errors would not result in calling people to account for mistakes, provided they were not caused by gross carelessness or repeated deficiency?
- How has the safety responsibility for your position been defined?

General part/Attitude of managers to safety

- How often do the plant top managers hold safety briefings or forums?
- Is there a process by which the non-supervisory level staff can report safety related concerns directly to the plant managers?
- How correct and friendly is the relationship between the staff and the line managers?
- How correct and friendly can be considered the relationship between the line managers and the top managers?

- Does the motivation system include safety related issues?
- In your opinion can the surveillance activity related to your work be considered adequate?
- To what degree in your opinion the quality control related to your work can be considered adequate?
- To what degree in your opinion the quality assurance activity related to your work can be considered adequate?
- How often do your supervisors take part personally in presentation of training courses?

General part/Work-load

- Do you consider the overtime and on-duty time ratio adequate?
- How often are you informed about the reasons of over-time assignments?
- What fraction of your working time is spent on administration and clarification of work conditions and tasks?
- Do you consider your work physically overloading?
- Do you consider your work mentally overloading?
- Briefly describe how you usually perform your work, have rest during a typical workday! (Please indicate the time and duration of short breaks and breaks for meals!)

General part/Strengths and weaknesses

- Briefly describe your personal "safety strengths"! (What are those activities in your work, which are important from safety point of view and in your opinion you perform them well?)
- Shortly describe what are your relative personal "safety related weaknesses"! (What are those activities in your work, which are important from safety point of view and in your opinion you perform them relatively poorly?)
- Briefly describe the possibilities by which in your opinion the safety can be improved in your area!
- Briefly describe some cases, which in your opinion could have led to more or less significant events, but due to some reasons finally they did not have consequences!

Professional part/Training

- How often are you not able to participate in periodic training due to high work load?
- How often are you not able to participate in periodic training due to private reasons?
- How often are you able to make up the missed training during work time?

- Do you think the current simulator training practice is effective and improves safety?
- How much do the malfunctions and their effect on safety become subject of training?
- Have you received the necessary organised training to perform your job?
- How many cases do you know when the training program was modified following committed real errors? Please write here the estimated number:
- Do the quality and efficiency of regular training practice meet your expectations?
- How deeply have you been trained in the characteristics of organisations and individuals important from safety point of view?
- Do the documents important for your work receive the necessary emphasise during training?
- What is the ratio of the answered questions you asked during the lectures?

Professional part/Self-checking

- Is the continuos and conscious self-checking typical for you?
- Is it typical that you think of whether you fully understand the given task?
- Is it typical that you think about what your personal responsibility is in the task?
- Is it typical that you think about the safety relevance of the task?
- Is it typical that you think about whether you have the necessary knowledge and skills to perform the task?
- Is it typical that you think about the responsibilities of other people in the task?
- Is it typical that you consider whether unusual conditions related to the job exist?
- Is it typical that you try to find out if you need assistance in performance of the task?
- Is it typical that you try to find out the potential consequences of making an error during the task?
- Is it typical that you think about how to prevent errors in performance?
- Is it typical that you think about what you should do when you committed an error and you noticed it?

Professional part/Job definition and support

- Are the boundaries of authority and responsibility of job positions in your area well defined?
- Do you feel that the expectations regarding your position are clear from all points of view?

- Do you consider important that during event evaluation your colleagues use a common approach considering the actions to be taken and the risk?
- Do you consider important that when evaluating events your colleagues know the standpoints and attitude of each other?
- Do the valid procedures determine your tasks in case of unusual events?

Professional part/Co-operation

- Is the co-operation of personnel belonging to different organisations co-ordinated during performance of common tasks?
- Can the solution of unexpected tasks, which require co-operation of personnel belonging to different organisations be evaluated as fast?
- Can it be stated, that personnel of organisations responsible for operation communicate with each other well, they use the same professional terminology and accept the same priorities?
- Can be stated that the operational procedures and other documentation are adequate and up-to-date?
- In your opinion how the I& C reconstruction of the control rooms will effect safety? Please give your short explanation!
- What are those organisational problems or technical problems significantly related to organisation, which you do not have influence on, but they challenge safety?

Professional part/Weaknesses

- Please list here those three potential accidents you are most afraid of:
- Please list here those three operational areas, where in your opinion the safety upgrading is most justified:
- Considering the whole plant which of the following general areas is the most critical one in your opinion, which is the second and which is the least critical? Please give some explanation!

  Operations ………………….

  Maintenance ……………….

  Engineering Support ……….

Global opinion

- In your opinion to what degree the external factors (social, economical, political factors, media, public, authorities) do influence the safety at Paks NPP?
- In your opinion to what degree the plant’s technical level does influence the safety at Paks NPP?

- In your opinion does the level of development of organisation and management influence the safety at Paks NPP?

- In your opinion do the characteristics of personal qualification and behaviour influence the safety at Paks NPP?

- Please fill in the following table putting the appropriate grade

Global opinion

- Please evaluate on a 0-100 scale the level of safety of those nuclear power plants, where everything has been done to upgrade safety and to improve technical, organisational and human factors

- Please evaluate on a 0-100 scale the level of safety of Paks NPP.

- Please evaluate on a 0-100 scale in your opinion how the Hungarian public assess the level of safety of Paks NPP

- Questions, remarks, suggestions regarding the above issues.

Papers of Specialist Meeting on

Safety Performance Indicators

October 17-19, 2000, Madrid, SPAIN

PERFORMANCE INDICATORS AT DAYA BAY NPP

Chunfa FANG / Operations Department of Guangdong Nuclear Power Joint Venture Company, Zip. 518124, Shenzhen, China, Tel. +86 755 4476349, Fax. +86 755 4476070, e-mail fangcf@gnpjvc.com.cn

Weixiong ZHONG / Operations Department of Guangdong Nuclear Power Joint Venture Company, Zip. 518124, Shenzhen, China, Tel. +86 755 4476082, Fax. +86 755 4476070, e-mail zhongwx@gnpjvc.com.cn

ABSTRACT

Ever since the commercial operation of Daya Bay Nuclear Power Station (GNPS), dynamic objective management concept that features modern enterprises has been adopted by the station to manage all operational activities with the guidance of business plan. And some quantitative indicators have been employed in order to measure effectively the progress of these operational activities. After several years of evolvement, a hierarchical and standard performance indicators system has been developed and is playing an active role in the plant's efforts towards top quartile of the world nuclear power industry.

Structured hierarchically with higher levels resolving into lower levels and lower levels committing to higher levels, the indicator system not only reflects performance-based management concept, but also shows the process-oriented control concept. Indicators of a certain level serve as both early warnings to superior indicators (lagging indicators in this case) and effects to inferior indicators (leading indicators in this case). The dynamic status of these indicators, numbered more than 230, will eventually be fed back to the business plan and realized through daily work of every branch, and even every member of the workforce.

With the indicator system as a quantitative management tool and an effective tracking system, GNPS has achieved good results in self-assessment, objective definition, improvement follow up, resource allocation, and management-staff communication. Periodic plant performance assessment is performed through spider chart and other patterns of graphics. Indicators are displayed at the plant entrance, offices, Main Control Room and CIS (Corporate Information System) network, where every worker gets access to and care for the performance of the plant. Root cause analysis is carried out and improvement measures are made when certain indicator is at unfavorable trend.

The indicator system, together with its tracking system, has been applied allover the station and is contributive to the realization of corporate objectives. Its effective implementation is well supported, recognised and involved by management of all levels.

CONTENTS

Selection of Indicators

GNPS started commercial operation in 1994 and developed its first business plan at that time. A set of performance indicators was selected, with reference to the practices of other utilities that operate the same type of plants, and incorporated in the business plan to provide a statistical measure of how plant performance changed over time. The safety indicators were mainly lagging in nature. Amongst them, some of the ten WANO indicators were used, but only those with clear definitions and standard formulas for calculation. The plant utilized the same set of performance indicators with very few changes until 1997.

After the first OSART mission in the fall of 1996, the station management put additional emphasis on improving management. In a world of diminishing resources, improvement of the management program and services was considered critical.

In late 1997, the first five-year corporate business plan was drawn up and put into effect in 1998. In order to align it with the corporate business plan, the station management plan for 1998 was revised to include a new set of performance indicators. Indicator selection was accomplished using the IAEA-J4-CT-2883 draft working document for selecting safety performance indicators. All ten WANO indicators were included. In November 1998, a workshop on "operational safety performance indicators" organized by IAEA was held in the station. The station management adopted the indicator framework developed at the IAEA, including the three attributes and the strategic groupings. The current management plan was subsequently expanded to include a total of 230 specific indicators (or first level), most of which represented plant specific measures.

The implementation of the program required the modification of computer programs and plant procedures, training and communication to the staff. During the process of selecting the performance indicators, some concerns were raised that there were "already too many indicators" and that "some indicators identify problems that need fixing and activities that have positive impacts". The latter could create unnecessary personal conflicts amongst concerned departments, especially with regard to aggregated measures for reporting to senior management. For this reason, the communication and discussion process to reach final agreement on the indicator definitions took much longer than expected.

The station is still evaluating the application of risk based indicators. Currently, resource priority has been given to finalizing the level 2 PSA study. This group of indicators (e.g. risk index during shutdown and normal operation) has been incorporated in the management plan for the year 2000. The framework adopted by GNPS is sampled in Figs 1 to 3.

Establishing Indicator Definitions

The station believes that the key to success in the implementation of the operational safety performance indicator program is to adopt a "disciplined approach". All too often performance measurement programs were established with good intentions, but some failed because they were shortsighted, ill conceived and unfocused. Most of these shortcomings can be traced to one source: the lack of a viable approach to performance measurement from the start.

The development of a precise description for each indicator definition is an important step for data collection and calculation. All the indicators selected were studied by a cross-functional group of experienced staff who had attended the IAEA's operational safety performance indicator workshop. The program facilitator reviewed all the proposed plant specific definitions before they were included in the standardized computer input card. The computer-input card was formatted to provide the necessary information, such as the names of the responsible branches for data collection and verification, calculation formulas, etc. A responsible manager was assigned to ensure the quality of data collection, data processing and to co-ordinate data trending and follow-up of relevant corrective actions derived from the performance variances. The criteria for evaluating the performance variances were formulated on each input card and a color coding system was used to identify the variances.

Identification of Goals

The station management believes and widely communicates the slogan "what gets measured gets done".

For example, once an indicator was developed to track an item (e.g. "lit anunciators"), the operating staff became much more aggressive in reducing this number. However, it should be noted that the development of certain indicators could produce unexpected results. An indicator to reduce the number of outstanding alarms in the control room may lead to an unintended outcome of increasing the number of jumpers. Additional measures would then be needed to ensure jumper control.

In setting goals, the station management observed the following five points:

- The cause and effect of outcomes are not easily detected.
- Poor results do not necessarily reflect poor execution.
- Numerical quotas do not fix defective processes.
- Measurements only approximate the actual system.
- Performance measures do not ensure compliance with laws and regulations.

The station management plan requires the plant to achieve the WANO top quartile performance by the year 2002. As a result, goal setting for the ten WANO indicators was clear. For the remaining indicators, goals were established with the consideration of:

- Industry benchmarks
- Corporate business plan expectations
- Previous plant performance
- Achievability of the goal, with some measure of flexibility

Setting goals required extensive discussions and negotiations with responsible managers and staff concerned to convince them to use the model developed at the IAEA and the indicators proposed.

Data Display and Interpretation

The introduction of the computerized Station Information System (SIS) in 1998 has helped improve communication of performance measurement internally amongst employees, as well as externally between the organization and its customers. The emphasis on measuring and improving performance (result-oriented management) has created a new climate, affecting all departments within the company. Station staff believe that a result-oriented organization requires timely and accurate information depending on the effective communication of mission-critical activities. Additionally, the performance-monitoring program is helpful in justifying plant improvement programs and their costs.

Prior to starting the operational safety performance indicator program, an implementation procedure was written and incorporated in the station procedure manual. Training was provided to all concerned staff prior to the use of new or modified procedures.

Standard data collection forms and computer data formats were also designed for use by authorized persons in each individual area. While the computer format governs the method of data processing, each performance indicator coordinator is responsible for data verification. Any missing information can easily be seen and picked up by the responsible manager. The most important step is the verification by the performance indicator co-ordinator of the effectiveness and validity of the data. The performance indicator co-ordinator also carries out calculation and trend analysis. Performance variances are reported to the responsible manager for strategic actions.

Information obtained by the performance indicator program is available at all 1000 computer terminals. In addition to providing a graphic display of information and trends, the station utilizes a color rating system to assess indicator performance relative to established goals. Color ratings for each indicator are aggregated to produce ratings for higher level indicators or "windows". If any specific indicator in a given area is rated "red" or unsatisfactory", the higher level window is also assigned a "red" rating to flag the area for management attention and action. These color "windows" provide an effective management tool for review of performance in critical areas.

To allow multiple sets of indicators to be compiled into an overall measure, the station has developed a performance index system though in the initial stage. The indexes to date include WANO performance index, safety culture index and corrective action index. This system is expected to enhance management review of station performance for the purpose of decision-making, and will be further improved in the years to come.

Key indicator results are also displayed graphically on 2 large LED display boards at the plant and office entrances. The responsible manager for each performance indicator must review performance results with his branch heads on a monthly basis and establish improvement strategies for those indicators rated other than "green" or "significant strength". Indicator reported as "unsatisfactory" or those persistently rated as "needs improvement" are reviewed in the plant nuclear safety committee. A performance indicator program report is distributed to all branches and departments monthly and discussed in the senior management meeting.

The Station Information System was upgraded to Corporate Information System (CIS) in 1999, using improved information technology, for faster and more convenient dissemination of station information to a wider scope of users in-house. The performance indicators program resides in this Web page. The Web page is menu-driven and user-friendly (see Figs. 4 to 11). A click on the "station performance indicators program" menu button on the Corporate Information System guides the user, on any computer terminal within the station, to input source data or to obtain indicator information from the system. Evaluation, verification of data and trend analysis can also by performed at authorized computer terminals. The inclusion of performance indicators on the LAN computers has made a great contribution toward the success of the performance indicator information dissemination, and has also contributed to spreading a clear message that "what gets measured gets done".

Management Involvement

Plant management reviews the performance indicators on a monthly basis in the management direction team meeting. Responsible managers analyze all performance variances and set strategic actions for improvement. The plant nuclear safety committee reviews all variances concerning nuclear safety. Plant performance results are reviewed and discussed with corporate management monthly. Plant and corporate management are strongly supporting the development of risk based indicators, with the aim of generating forward looking risk profiles for performance assessment. Figs. 12 to 14 are samples of management review tools.

The station has received excellent support and encouragement from the corporate management for upgrading all LAN computers and for developing the Intranet for effectively enhancing plant performance measurement, analysis and results dissemination. Drive from the corporate management to look into the different approaches that can be taken to develop a performance index system is another example of keen support from the senior management.

Insights and Lessons Learned

The station's operational safety performance indicator program is on the charted path of achieving its goals. However, it is difficult to obtain agreement from all departments on the unified indicator system, given the complexity of the four levels and with the expectation that the lower level indicators should serve its next upper level as leading indicators.

The assumption of achieving the set goals at the lowest level indicators should warrant the zero performance variances at the first level indicators.

Senior management's enthusiasm for arranging a performance indicator seminar requiring all department managers and facilitators to attend during the third quarter of 1999 has been a very promising support to the indicator program based on the one developed at the IAEA.

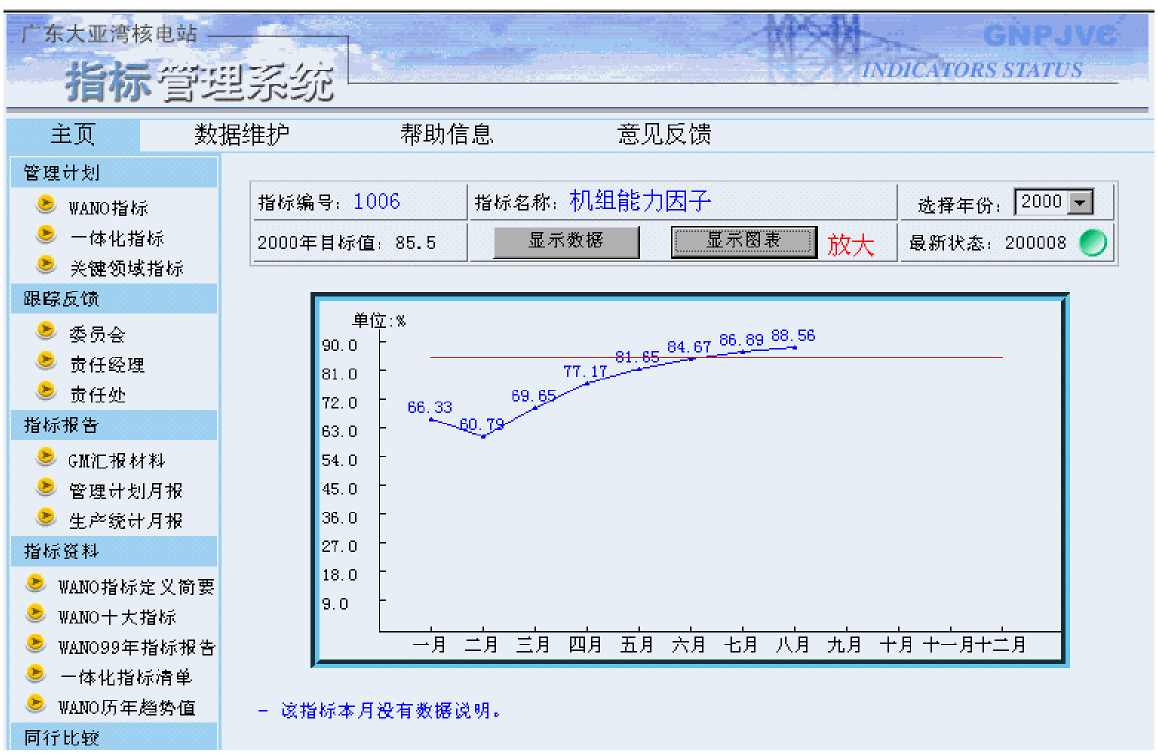
To support the indicator program, it is necessary for all managers to understand Dr. Deming's statement:

- Management's job is prediction and there is no prediction without theory;
- There are no data on the future, data from the past must be used to form a base for prediction;
- 94% of the changes required for improvement will require action by management.

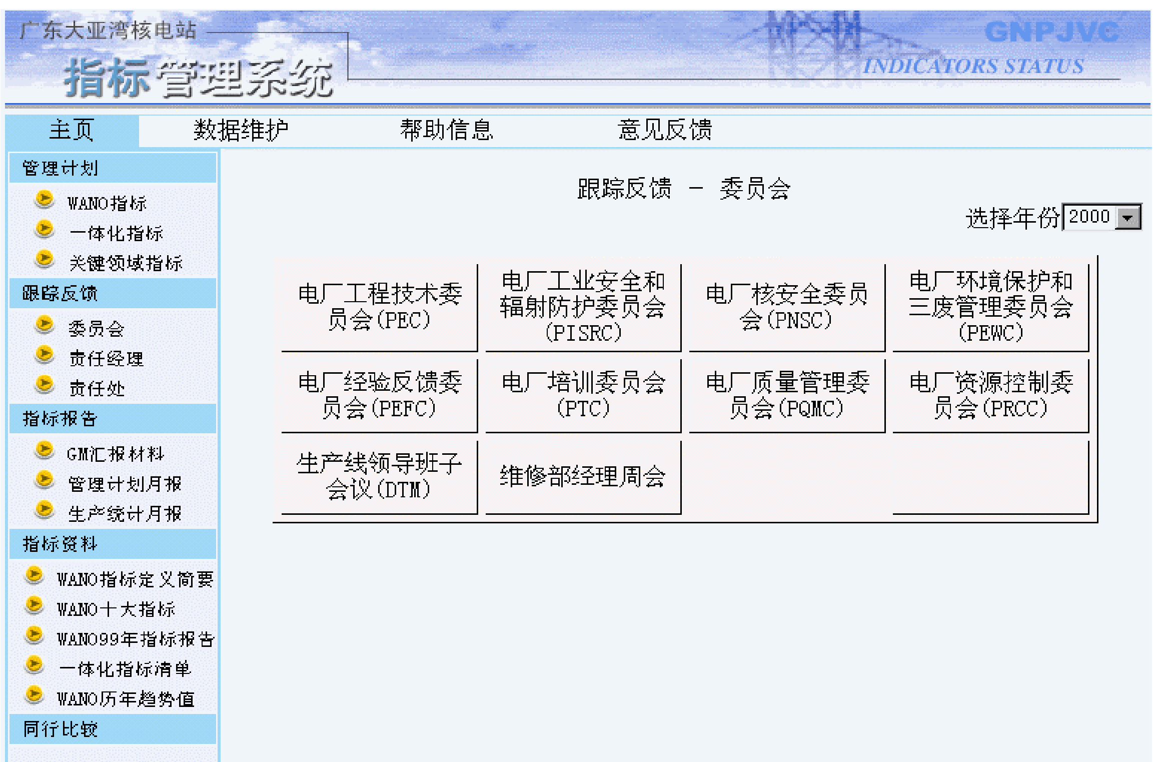
REFERENCES

- WANO-PC Indicator Program
- IAEA-TECDOC-1141 Operational Safety Performance Indicators for Nuclear Power Plants


广东大亚湾核电站
指标管理系统
GNPJVC
INDICATORS STATUS
主页
数据维护
帮助信息
意见反馈
管理计划
WANO指标
一体化指标
关键领域指标
跟踪反馈
委员会
责任经理
责任处
指标报告
GM汇报材料
管理计划月报
生产统计月报
指标资料
WANO指标定义简要
WANO十大指标
WANO99年指标报告
一体化指标清单
WANO历年趋势值
WANO 指标盘
选择月份 200008
固体放射性废物产生量
非计划能力损失因子
燃料可靠性
辅助给水性能
机组能力因子
应急柴油机性能
集体剂量
热性能
高压安注性能
工业事故率
化学指标
7000临界小时非计划自动停堆次数
本月业绩总体趋势正常。三个安全系统可靠性由于7月份1ASG003PO不可用273小时导致其可用率降低。
自2000年8月8日，已有 1054 人访问了[指标管理系统]
如果您有任何意见或建议，请发表


广东大亚湾核电站
指标管理系统
GNPJVC
INDICATORS STATUS
主页
数据维护
帮助信息
意见反馈
管理计划
WANO指标
一体化指标
关键领域指标
跟踪反馈
委员会
责任经理
责任处
指标报告
GM汇报材料
管理计划月报
生产统计月报
指标资料
WANO指标定义简要
WANO十大指标
WANO99年指标报告
一体化指标清单
WANO历年趋势值
同行比较
指标编号：1006
指标名称：机组能力因子
选择年份：2000
2000年目标值：85.5
显示数据
显示图表
放大
最新状态：200008
单位：%
90.0
81.0
72.0
63.0
54.0
45.0
36.0
27.0
18.0
9.0
66.33
60.79
69.65
77.17
81.65
84.67
86.89
88.56
一月 二月 三月 四月 五月 六月 七月 八月 九月 十月 十一月 十二月
- 该指标本月没有数据说明。


广东大亚湾核电站
指标管理系统
GNPJVC
INDICATORS STATUS
主页
数据维护
帮助信息
意见反馈
管理计划
WANO指标
一体化指标
关键领域指标
跟踪反馈
委员会
责任经理
责任处
指标报告
GM汇报材料
管理计划月报
生产统计月报
指标资料
WANO指标定义简要
WANO十大指标
WANO99年指标报告
一体化指标清单
WANO历年趋势值
指标编号：1006
指标名称：机组能力因子
选择年份：2000
2000年目标值：85.5
显示数据
显示图表
放大
最新状态：200008
日期 原始数据 打分 状态
200008 88.56 100
200007 86.89 100
200006 84.67 100
200005 81.65 100
200004 77.17 100
200003 69.65 100
200002 60.79 100
200001 66.33 100
- 该指标本月没有数据说明。


广东大亚湾核电站
指标管理系统
GNPJVC
INDICATORS STATUS
主页
数据维护
帮助信息
意见反馈
管理计划
WANO指标
一体化指标
关键领域指标
跟踪反馈
委员会
责任经理
责任处
指标报告
GM汇报材料
管理计划月报
生产统计月报
指标资料
WANO指标定义简要
WANO十大指标
WANO99年指标报告
一体化指标清单
WANO历年趋势值
同行比较
跟踪反馈 － 委员会
选择年份 2000
电厂工程技术委员会(PEC)
电厂工业安全和辐射防护委员会(PISRC)
电厂核安全委员会(PNSC)
电厂环境保护和三废管理委员会(PEWC)
电厂经验反馈委员会(PEFC)
电厂培训委员会(PTC)
电厂质量管理委员会(PQMC)
电厂资源控制委员会(PRCC)
生产线领导班子会议(DTM)
维修部经理周会


广东大亚湾核电站
指标管理系统
GNPJVC
INDICATORS STATUS
主页
数据维护
帮助信息
意见反馈
管理计划
WANO指标
一体化指标
关键领域指标
跟踪反馈
委员会
责任经理
责任处
指标报告
GM汇报材料
管理计划月报
生产统计月报
指标资料
WANO指标定义简要
WANO十大指标
WANO99年指标报告
一体化指标清单
WANO历年趋势值
跟踪反馈 – 委员会 – 电厂工业安全和辐射防护委员会(PISRC)
选择年份 2000
集体剂量
工业事故率
年剂量＞20mSv.的人数
火灾未遂次数
消防系统不可用次数与天数比
工业未遂次数
控制区污染次数
体表沾污人次
辐射防护设备异常比例（KZC）
放射性物品失控次数
通过C2门检测发现沾污人次
辐射防护整改项数
工业事故严重度
防火门完好率
起重设备完好率
消防演习不成功次数
消防系统可用性
工业安全/消防整改数


广东大亚湾核电站
指标管理系统
GNPJVC
INDICATORS STATUS
主页
数据维护
帮助信息
意见反馈
管理计划
WANO指标
一体化指标
关键领域指标
跟踪反馈
委员会
责任经理
责任处
指标报告
GM汇报材料
管理计划月报
生产统计月报
指标资料
WANO指标定义简要
WANO十大指标
WANO99年指标报告
一体化指标清单
WANO历年趋势值
跟踪反馈 – 责任经理
选择年份 2000
RICHARD
蔡康元
陈德淦
刘德强
卢长申
张善明

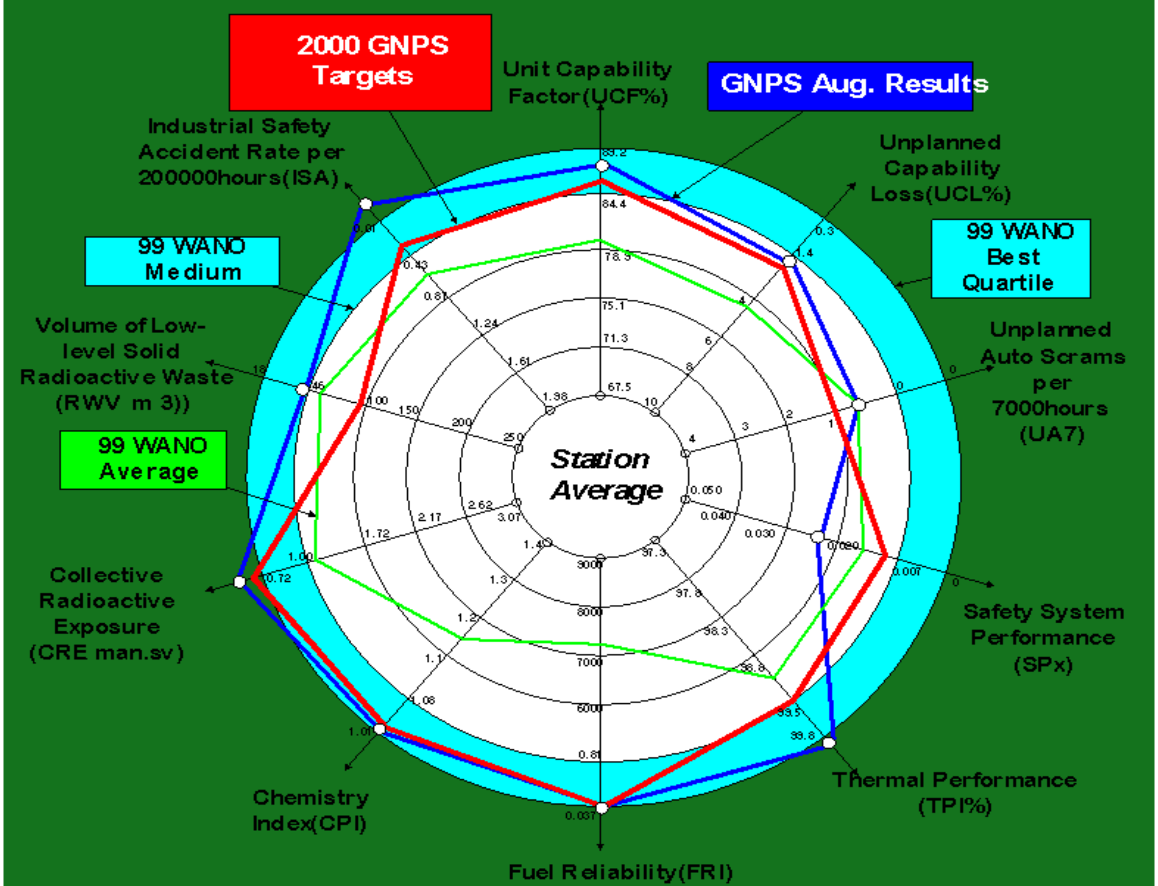
指标定义： 一定时间内机组的可用毛能量与额定毛能量之比。可用毛能量是额定环境条件下，电厂管理能力控制范围内（设备性能、人员状况、工作控制等）机组可以生产出来的电能。额定毛能量是参考环境条件下机组连续满功率运行可以生产出来的电能。
统计说明：应分机组/年月进行统计
计算公式：UCF=（额定毛能量-计划能量损失-非计划能量损失）/额定毛能量
参考程序：WANO第一章第四节
数据单位：%

指标领域：生产	责任部门：MTS
责任委员会：生产线领导班子会议(DTM)	改进计划：IP4
责任经理：张善明	年度目标值0：85.5
年度目标值1：85.5	年度目标值2：85.5

指示灯说明：

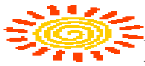
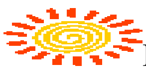
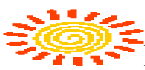
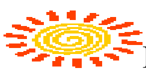
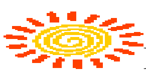
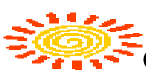
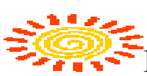
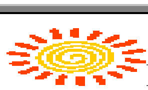
- 100%完成计划
- 完成计划达到或超过95%以上
- 完成计划80%以上但不到95%
- 完成计划80%以下
- 指标原始数据未录入或者未确认

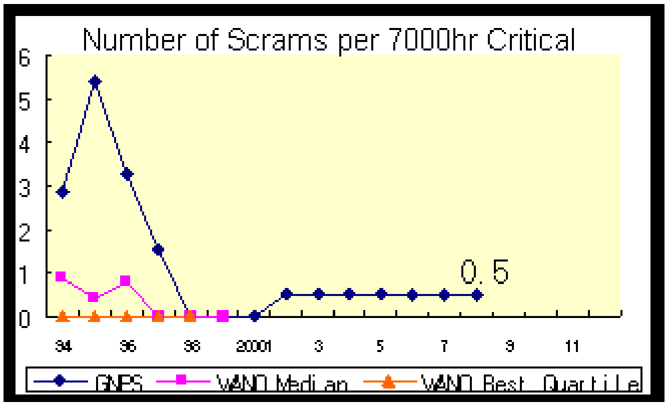
GNPS AVERAGE PERFORMANCE RESULTS £¨COMPARISON WITH WANO INDICATORS

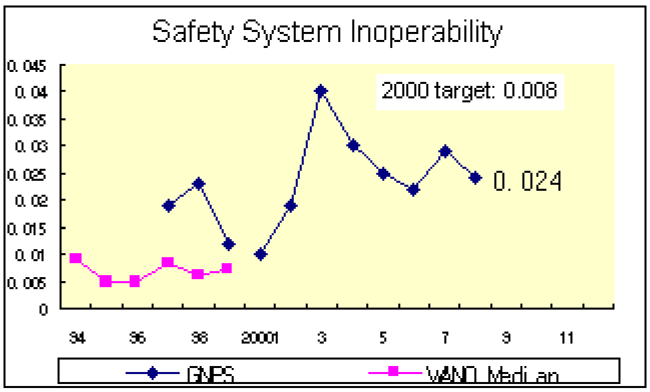


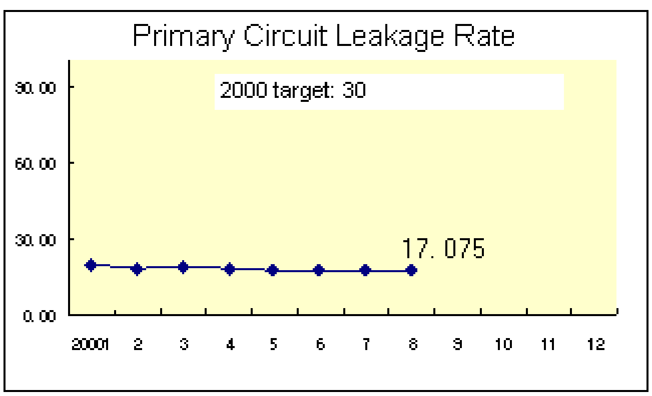
	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	YTD	2000 Target
UCF	66.33	54.97	86.78	99.97	99.15	100	99.9	100					88.56	85.5
UCL	0.19	2.20	10.76	0	0.82	0	0.10	0					1.78	2
UA7	0	8.1	0	0	0	0	0	0					0.7	1
SP1	0.01	0.007	0.002	0	0.002	0.00	0.001	0					0.002	0.002
SP2	0	0.022	0	0.001	0	0.002	0.061	0					0.011	0.001
SP3	0	0	0.08	0.001	0	0.006	0	0					0.011	0.005
TPI	100	100	100	100	100	100	100	100					100	99.5
FRI	0.037	0.037	0.037	0.037	0.037	0.037	0.037	0.037					0.037	0.037
CPI	1.11	1.02	1.00	1.00	1.35	1.02	1.0	1.00					1.02	1.03
CRE	0.1755	0.0811	0.0108	0.00786	0.00503	0.0067	0.0068	0.0044					0.30	0.7
RWV	6.675	7.735	3.57	13.47	2.6	6.015	0	5.46					45.525	95
ISA	0.0666	0.0671	0	0	0	0	0	0					0	0.2

Performance area	Year target	Monthly value	trends	situation

 **Safety system performance**	100	60		
 **Barrier integrity**	100	100		
 **Industrial safety**	100	70		
 **Radiation protection**	100	100		
 **Human performance**	100	100		
 **Forced power reductions**	100	100		
 **Equipment maintenance**	100	100		
 **Chimestry**	100	100		
 **Human resources**	100	100		
 **Producton cost**	100	100		


Number of Scrams per 7000hr Critical
6
5
4
3
2
1
0
0.5
94
96
98
20001
3
5
7
9
11
GNPS
WANO Median
WANO Best Quartile


Safety System Inoperability
2000 target: 0.008
0.024
0.045
0.04
0.035
0.03
0.025
0.02
0.015
0.01
0.005
0
94
96
98
20001
3
5
7
9
11
GNPS
WANO Median


Primary Circuit Leakage Rate
2000 target: 30
17.075
90.00
60.00
30.00
0.00
20001
2
3
4
5
6
7
8
9
10
11
12

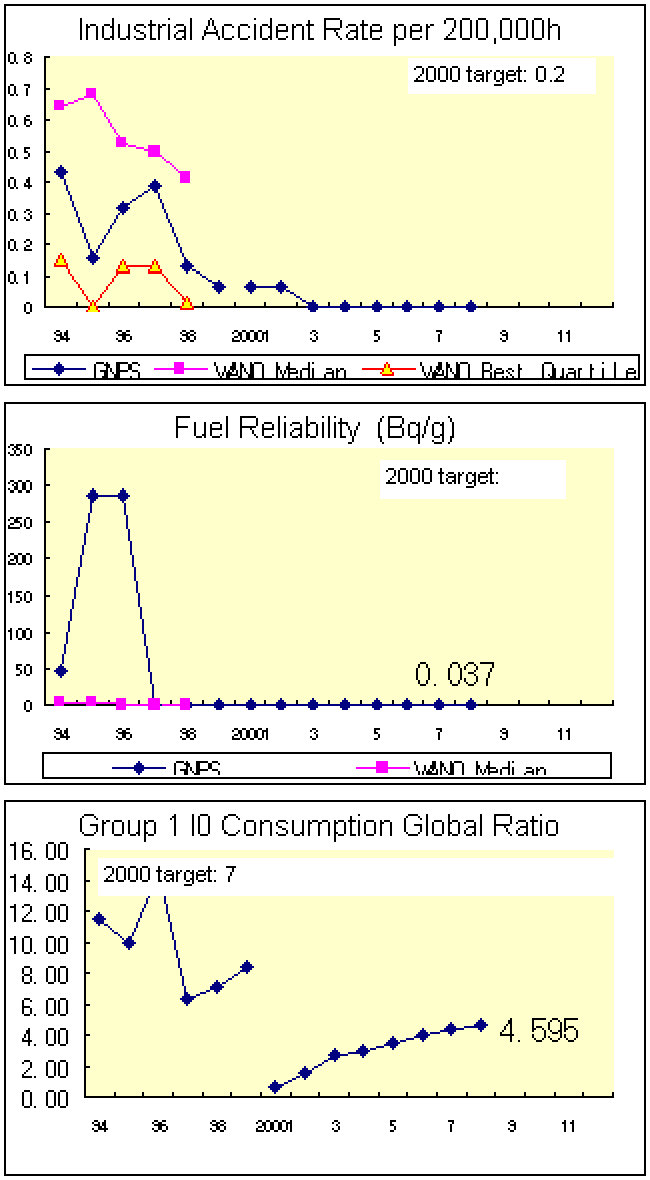


Fig.14 Management Review Tools – sample of indicator's monthly trending

Safety Performance Indicators Workshop

LIST OF PARTICIPANTS

Name, First name	Company	Country
ALONSO, Agustín	CSN	Spain
ANTOLIN, Leopoldo	WANO	France
ATKINSON, Carl M.	British Energy	United Kingdom
BALMISA, José	CSN	Spain
BARDOU, Sophie	IPSN/DES	France
BARLETT, John	HSE-NII	United Kingdom
CALDUCH, Francisco	Central Nuclear de Cofrentes	Spain
CARLSSON, Lennart	OECD/NEA	
CARNINO, Annick	IAEA	
CRIADO, Soledad	CSN	Spain
DE LOS REYES, Alfredo	CSN	Spain
DE SOUZA, Vaner	Electronuclear	Brazil
DJURSING, Dag	Vattenfall AB	Sweden
DUMONT, Laurent	EDF	France
ELVIRO, José L.	Central Nuclear de Almaraz	Spain
ENGBLOM, Ola	Ringhals AB	Sweden
ERIKSON, Hams	Swedish Nuclear Power Inspectorate	Sweden
FANG, Chumpa	Guangdong Nuclear P. Joint Venture Co	China
FLOYD, Stephen	Nuclear Energy Institute	USA
FORSBERG, Staffan	Swedish Nuclear Power Inspectorate	Sweden
FUCHS, Michael	DKVG	Germany
FULK, Siegfried	TUEV ET BW	Germany
GARCÍA SÄNCHEZ, José	ANA/CNV	Spain
GEA, Antonio	CSN	Spain
GLORIAN, Daniel	Electricité de France	France
HAMLIN, Kent	WANO	United Kingdom
HERMANA, Ignacio	Iberdrola Ingeniería y Consultoría	Spain
HICKMAN, Donald	US Nuclear Regulatory Commission	USA
HILL, TIM	National Nuclear Regulator	South Africa
HIMANEN, Risto	Teollisuuden Voima Oy	Finland
ISASIA, Rodolfo	CSN	Spain
IZQUIERDO, Carlos	UNESA	Spain
JOHNSON, Jon	US Nuclear Regulatory Commission	USA
KEUSENHOFF, Juergen	GRS	Germany

KHATIB-RAHBAR, Mohsen	Energy Research, Inc.	USA
LAFRENIERE, Ken	Canadian Nuclear Safety Commission	Canada
LEDERMAN, Luis	IAEA	
LEE, Saeyul	KINS	Korea
LEHTINEN, Esko	VFF Automation	Finland
MACSUGA, Geza	Hungarian Atomic Energy Authority	Hungary
MAMANI, Yuri	CNSNS	Mexico
MAROÑO, Marta	Ciemat	Spain
MARTÍN, Aníbal	CSN	Spain
MATAHRI, Naoelle	IPSN/DES	France
MORALES, Lola	Central Nuclear de Almaraz	Spain
MUHEIM, Josef A.	KKL, NPP Leibstadt	Switzerland
PRESTON, John	WS Atkins	United Kingdom
PROHASKA, Guenter	Swiss Fed. Nuclear Safety Inspector	Switzerland
RAYMOND, Patrick	DSIN	France
REHACEK, Radomir	State Office For Nuclear Safety	Czech Republic
REVUELTA-LAPIQUE, Ramón	Iberdrola	Spain
SCHLEGEL, Jurgen	WANO	France
SCHMOCKER, Ulrich	Swiss Federal Nuclear Safety Insp.	Switzerland
SEIDEL, Ernst R.	BAYER STMLU	Germany
SEIJA, Suksi	STUK	Finland
SHIMADA, Yuichi	World Association Nuclear Operators	
SHIMOMURA, Kazuo	OECD/NEA	
SOLÁ, María Rosario	Ciemat	Spain
STRAUSS, Gunter	TUR Suddeutschland	Germany
SUAREZ, Joaquín	Iberdrola Ingeniería y Consultoría	Spain
TANAKA, Jun	Nuclear Power Engineering Corp.	Japan
TIPPANA, Petteri	STUK	Finland
TOTH, Janos	PAKS NPP	Hungary
VAN EINNEBEEK, Jean-Jacques	AVN	Belgium
VANDEWALLE, André	AVN	Belgium
WILHELM, Hans	Swiss Federal Nuclear Safety Commission	Switzerland
YNDUÁIN, Félix	Ciemat	Spain
ZARZUELA, Javier	CSN	Spain
ZHONG, Weixiong	Guangdong Nuclear P. Joint Venture Co	China

Unclassified **NEA/CSNI/R(2001)11**

Organisation de Coopération et de Développement Economiques
Organisation for Economic Co-operation and Development

English text only

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2001)11
Unclassified

SUMMARY REPORT ON THE USE OF PLANT SAFETY PERFORMANCE INDICATORS

English text only

Document complet disponible sur OLIS dans son format d'origine
Complete document available on OLIS in its original format

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14th December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concerned with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

* * * * * * * * * * * *

Requests for additional copies of this report should be addressed to:

Nuclear Safety Division
OECD Nuclear Energy Agency
Le Seine St-Germain
12 blvd. des Iles
92130 Issy-les-Moulineaux
France

SUMMARY REPORT ON THE USE OF PLANT SAFETY PERFORMANCE INDICATORS

Index

SUMMARY REPORT ON THE USE OF PLANT SAFETY PERFORMANCE INDICATORS

1. Introduction on the use of performance indicators in a regulatory perspective [1]

2. Objective

In 1998, the OECD/NEA committee on Nuclear Regulatory Activities (CNRA) initiated an activity with the objective of advancing the discussion on how to enhance and measure regulatory effectiveness in relation to nuclear installations. One of the outcome of this activity was to establish a Task group to develop internal (direct) performance indicators which would be used to monitor regulatory efficiency ("do the work right").

In parallel, a joint CNRA/CSNI group was launched in December 2000 to exchange information and develop *external (indirect) indicators* to measure regulatory effectiveness, i.e. impact on licensee's safety performance ("do the right work"). These external indicators are, in other words, the traditional plant performance indicators and these are the ones that this report deals with.

According with CNRA and CSNI mandate [2,3], the objective of this joint activity is:

- to compile a summary report on plant performance indicators currently being used or being tested by regulatory bodies.
- to prepare sets of common performance indicators that could be used by each regulatory body
- to prepare a summary paper to be presented as input to the IAEA topical meeting in September 2001 on this subject

3. Background

On the initiative of the NEA/CSNI Working Group on Operating Experience the Spanish CSN hosted a workshop (Madrid, 2000) to review the state of the art on Safety Performance Indicators. This workshop, which was cosponsored by the IAEA and WANO was attended by 73 participants from 19 countries, representing the industry, regulators, service companies as well as international organisations. [4]

The conclusions were:

1. there is considerable development effort on performance indicators in many countries.
2. utilities continue to rely on the WANO Performance indicators system which consists of indicators in 8 key areas and receives data from virtually all commercial NPP's in the world.
3. Regulators do not have a common set off performance indicators.

This report presents the work performed by the joint CNRA/CSNI task group mentioned above. It provides a summary of the sets of PI's being used by different regulatory bodies and WANO, it describes the national practices on the use of PI's and proposes a set of PI's that could be used nationally describing regulatory effectiveness and also as a basis for an international system.

4. Task Force Method

According to CNRA/CSNI directive, the task force consisted of regulators, organisations which have a performance indicators system in operation or under testing. Members of the group are listed in Appendix 2.

The task force met in Paris on February 19-20, 2001. Each participant provided a brief description of the PI System at his organisation and its usage.

The group identified a list of PI’s that are recommended to be used nationally by regulators.

This paper has been elaborated based on the information exchanged and discussions held in the February meeting. A first draft was prepared and group members comments and concurrence have been managed through e-mail communications.

5. Experience by the Users of Performance Indicators

The participating countries and WANO were asked to provide an overview of systems in use. The systems for Spain, Finland, US, Sweden (proposal) and WANO are attached in Appendix 3. Notice that, except for WANO system, this part deals with performance indicators used by regulators to monitor plant performance.

Finland

Indicators used at STUK are measures related to the safety of nuclear installations and regulatory activities. Indicators are numbers, ratios, percentages and amounts of matters that are found suitable for regulatory purposes that is assessment and trending of the safety of nuclear installations and regulatory activities. STUK's indicator system is divided into two main areas; safety of nuclear facilities and regulatory activities. Safety of nuclear facilities is divided into 3 areas based on the concept of defence in depth; safety and quality culture, operational events and physical barriers. Regulatory activities are also divided into 3 areas; working processes, resource management and regeneration and ability to work.

Data needed for the calculation of indicators related to the safety of nuclear installations is gathered mainly from the reports sent regularly and according to the reporting requirements to the regulator. However, there are some indicators which calculation requires data that is not regularly reported to the regulator. These are mostly related to the failure data. Every nuclear safety indicator has a responsible person who is responsible for the data collection, calculation, assessment and reporting of his or her indicator on annual basis.

Some of the indicators (bolded in the table in Appendix 3) are included in the management system of the department of Nuclear Reactor Regulation. This means in practise that these indicators have internal goal values and these are systematically (annually) calculated, assessed and reported within the regulator. Some of these indicators are also included in annual report of STUK, but no formal decision has been made so far to include all indicators to the annual report systematically. All indicators are available for the personnel of STUK in the Intranet. So far, there is no formal decision to open these indicators to the public at the STUK Internet site.

Indicators describing the safety of nuclear installations are used as a background material for the discussions between regulatory and licensee management, safety assessment and inspections and also focussing of regulatory investigations. How indicators are used to assess performance safety is mainly to identify changes in the trends of safety and then to find out the causes for the changes.

France

In France a committee has started the development of indicators for the confidence in the transparency of information about French nuclear facilities. The committee works are devoted to Government and Public.

The RECUPERARE method, a model developed by IPSN for operating experience feedback analysis presented at the October 2000 PI meeting in Madrid, is used to analyse incident reports. For the time being, IPSN emphasises the difficulty in connecting performance indicators to safety.

Spain

CSN has operated a PI system since the mid 1990's. The system is based on the one used by the USNRC and therefore can be used to compare the performance of Spanish and US plants of similar technology and vintage. When USNRC changed its system in 2000, CSN designed a new system that build on the experience of the existing one and gathers data on the following parameters: performance stability, reliability of mitigating systems, barrier integrity, and radiological impact.

The average values for the Spanish NPPs are currently made available to the public, in the future the values for each individual plant will also be released to the public.

Spain has found it useful to compare the performance indicators for its plants with those of US plants. As an example, differences have been identified in the reliability of external power supply due to the lower stability of the Spanish electrical grid, and in operating practices such as faster start ups after non scheduled shutdowns in Spanish NPPs. The root causes and impact of these differences on safety have been evaluated by the CSN.

Based on this experience the CSN supports international efforts to develop a common set of PIs that allow exchange of data among interested parties.

In Appendix 3 are included the list of the running Spanish PI System and the draft list of the new ones under development.

Sweden

SKI is developing a PI System. The indicators to be collected are given in Appendix 3. The set has been developed based on a research project presented at the Madrid meeting. In order to get a quick start and not so time consuming work SKI decided to test an indicator system without the part dealing with the probabilistic approach as shown in the Madrid meeting. The probabilistic part is planned to be used at a later stage. So far SKI has for the last half year used indicators for safety evaluation during the plant safety review meetings. It should be noted that this set represents SKI, Swedish Nuclear Power Inspectorates area of responsibility. In Sweden the Radiation Protection Institute provides information on radiological doses and releases.

United Kingdom

The NII is examining the requirements for safety performance indicators for the regulator and the industry in the UK. External and internal indicators are being considered, together with direct and indirect measures in order to develop the two sets.

The industry in the UK has seven major types of installation e.g. Large power reactors, waste treatment plants, submarine refuelling facilities. Therefore, longer term intention is to develop a set of indicators for the industry which can be applied to all types. This is likely to involve a common set of indicators plus surrogates for the different types. Earlier internal methods of NII for measuring the trends in the safety performance at large power plants and at reprocessing plants are being re-examined.

The seven or eight indicators *from this meeting* will be tried during 2001 in a pilot study of their practicality.

United States of America

No NRC representative attended the February meeting of the task force. However, NRC provided input for this activity. The input consisted of and excerpt of the US NRC document SECY 99-007 “Recommendations for Reactor Oversight Process Improvement”.

The NRC has revamped its PI System and the new one running since 2000 consists of 3 safety strategic areas: reactor safety, radiation safety and safeguards. Each area is split into one to four cornerstones: initiating events, mitigating systems, barrier integrity and emergency preparedness, for the area of reactor safety; public radiation safety and occupational radiation safety, for the area of radiation safety, and physical protection for safeguards.

For monitoring each cornerstone specific indicators have been developed. For each indicator there are numerical thresholds of acceptable performance and those thresholds are established making use of performance experience, PSA insights.

The NRC Performance Indicators System, as part of its Reactor Oversight Process is described at the internet address www.nrc.gov/NRR/OVERSIGHT/ROP/documents.html

WANO

An overview of WANO indicators is given in Appendix 3.

WANO emphasises that the definitions of their indicators are very specific, and closely connected to WANO's mission. The system is continuously under a review process.

WANO is developing a tool to enable their members to evaluate routinely the collected set of indicators according to their desires. Presently, about 200,000 data points are available. Some statistical indicator values are presented in a yearly trifold annual report publicly available.

6. Commonalities and Differencies in Used Performance Indicators

The presently used Performance Indicators were reviewed in a three steps process.

1. First indicators used in at least two agencies were identified.
2. The second step was to identify the most used indicators.
3. The third step was to assess if the indicators were universally understood, objective and obtainable from available data.

6.1 PI’s used in more than one agency

The task force participants reviewed the list of PIs in use at their agencies and withdrew the list of indicators used at least at two of them. The list obtained is presented in table 1.

TABLE 1.- LIST OF PERFORMANCE INDICATORS USED AT MORE THAN ONE AGENCY

	Running Programs				Programs in development			
	STUK	CSN	NRC	WANO	SKI	NII	IPSN	CSN
Unit Capability Factor				X			X	X
Unplanned capability loss factor				X	X			
Power reductions	X	X	X			X		X
Scrams	X	X	X	X	X	X	X	X
Availability of safety systems	X		X	X	X		X	X
Unplanned safety system actuations		X			X		X	X
Safety system failures in actual events		X	X		X			X
Fuel integrity	X		X	X	X			X
Chemistry PI	X			X				
Integrity of reactor coolant system	X		X		X		X	X
Integrity of containment	X				X		X	
Radioactive releases	X						X	
Radioactive dose to public	X		X					X
Collective radiation exposure	X	X		X		X	X	X
Significance of events	X	X		X	X		X	
Violations of technical specifications	X				X		X	
Delays in documentation of plant modifications	X					X		
Maintenance poor interventions	X				X		X	
Industrial safety				X		X	X	
Causes of events	X	X					X	

Table notes:

Running programs.- Performance Indicators programs that are presently run.

Programs in development.- Performance Indicators programs that are in a development phase. Indicators contained in these programs need still to be formally approved.

Significance of events.- this indicator has several meanings. It is, for instance, the conditional core damage frequency estimated for a given event after making use of Probabilistic Safety Analysis technics (STUK, IPSN). In the case of CSN, it is referred to significant operating events based on national reporting requirements? and in a case of WANO, it is referred to the events by separately issued confidential documents dealing with operating experiences that are considered specially relevant for their root causes, lessons learnt and/or risk to the plant.

<u>Number of events</u>.- some agencies take this figure as an indicator including safety significant events based on national reporting requirements. However, taking into account that national reporting requirements vary very much from country to country and that more explicit indicators related to number of events such as number of scrams and power reductions are already included in the list, the number of events was not included in the list.

6.2 PI's used in at least four agencies

There are seven indicators that are present at most agencies (at least at four out of the seven represented at the Task Force February 2001 meeting). These indicators are:

- Power reductions
- Scrams
- Availability of safety systems
- Fuel Integrity
- Reactor coolant system integrity
- Collective radiation exposure
- Significance of events s

6.3 Characteristics of the seven most used indicators

The <u>seven most used indicators</u> were assessed as to whether they meet the following characteristics considered recommendable for any use:

- *Universally understood*

  In spite of differences in definitions, it was found out that there is a rather broad consensus on the meaning of each of these indicators. Probably the main difference is for scrams, as WANO and NII are counting only automatic scrams, while all others are counting both automatic and manual scrams.

- *Objective*

  It was agreed that these indicators are not susceptible to manipulation, subjective approaches.

- *Easily obtainable from available data*

  It was verified that the data needed to obtain these indicators are already available at all participant regulators, the exception being "Safety System availability", which is not directly available at some regulators, but is obtainable by computing some data. Many NPPs have such data, anyway, as they are reporting this indicator to WANO .

- *Applicable to international exchange among interested regulators*

  As long as the definitions are the same, or close, they allow exchange among interested parties.

All of the above seven indicators meet these characteristics Significance of events excluded. The reason is that the definition of "significance" varies very much; even when PSA is used as the main "significance" measure. PSA models vary very much from place to place in terms of scope, depth of the model, etc., plus Accident Sequence Precursor techniques are also different. Therefore, the task group decided that this is not an appropriate indicator to be used at an international level for the time being.

The task group also checked out availability and publicity of these indicators, in terms of:

Availability.- Data needed to calculate the indicator is easily available for the regulator (regularly reported by licensees through licensee event reports, periodical reports (monthly, quarterly, annual).

Publicity.- the regulator is presently publishing these indicators on the annual report to the parliament, website etc.

The results of this survey are presented in Tables 2 and 3.

TABLE 2. INFORMATION NEEDED TO OBTAIN THE INDICATOR IS EASILY AVAILABLE FOR THE REGULATORS FOR THE INDICATORS MARKED WITH X ON THE TABLE

	STUK	CSN	NRC	SKI	NII	IPSN
Power reductions	X	X	X	X	X	X
Scrams	X	X	X	X	X	X
Availability of safety systems	X	X	X	X		X
Safety system failures in actual events	X*	X	X	X	X	X
Fuel integrity	X	X	X	X	X	X
Integrity of reactor coolant system	X	X	X	X	X	X
Collective radiation exposure	X	X		X	X	

* Data is available and reported to the regulator but the indicator is not calculated as a separate indicator in STUK because of the rarity of this kind of events.

TABLE 3. REGULATORS ARE PRESENTLY PUBLISHING FOLLOWING INDICATORS MARKED WITH X ON THE TABLE

	STUK	CSN	NRC	IPSN	SKI	NII
Power reductions	X	X	X	X	X	X
Scrams	X	X	X	X	X	X
Availability of safety systems			X			
Safety system failures in actual events			X			
Fuel integrity			X			
Integrity of reactor coolant system			X			
Collective radiation exposure	X	X	X	X	X	

Note to Table 3: WANO data are not generally available to public. Only the annual global results are available.

Performance Indicators definitions

The task group has noticed that even counting a specific indicator, there are some slight differences in definitions. E.g.:

- *Power reductions*: some agencies are counting any power reduction of rated power (STUK), over 20% power reductions (NRC), or turbine generator disconnection from external electrical grid (CSN, IPSN).

- *Scrams:* Some agencies are counting only automatic scrams (WANO and NII), while all others are counting both automatic and manual scrams. Some are defining a scram as a "an actuation of the reactor protection system that takes the reactor from critical to subcritical", while others are counting scrams only if the reactor power is above a given value, e.g. 5% of reactor power.

- *Fuel integrity*: some are assessing it as a % of the Technical Specification limit of reactor coolant concentration of equivalent I-131 equivalent (CSN, NRC), while others are considering the absolute reactor coolant activity.

- *Availability of safety systems*: most agencies are taking into account availability of the Emergency Core Cooling Systems (either high or low pressure), emergency feedwater and emergency AC power sources.

It was concluded that some harmonisation in definitions is needed to make sure that each one is counting precisely the same matter if these indicators are going to be used and presented at international level.

7. Conclusions and Recommendations

Experience

1/ Many regulatory bodies of the OECD countries have considerable experience in developing and using performance indicators. In several cases, performance indicators data have been collected for a significant period of time and the use of indicators has been gradually improved based on experience.

2/ Basic criteria for selecting and using a performance indicator system should be established. The Task Group recommends that the following criteria be used:

a) the indicators should provide an objective indication of safety performance;
b) the indicators should be easily understandable; and
c) the data needed should be easily obtainable from existing data collection systems

3/ The review of the Task Group indicates that there is a set of indicators which fit the above criteria and is already commonly used by a number of Regulatory Bodies.

These indicators are:

- power reductions
- number of scrams
- availability of safety systems
- fuel integrity
- reactor coolant system integrity
- collective radiation exposure

Limitations

1/ Performance indicators by themselves provide an indication but not a complete measure of the safety of a nuclear power plant. Furthermore, some indicators are an aggregation of several parameters. This must be carefully considered when trends in indicators are evaluated.

2/ Although the indicators shown above are common to several Regulatory Bodies, there are slight differences in the definitions used by various organisations. Caution must therefore be used when exchanging information even on indicators that have identical names.

3/ Performance indicators should be used preferentially to compare performance over time; caution must be used in comparing different plants and/or plants in different countries.

Task Force proposals and recommendations

1/ The minimum set of performance indicators listed above may be used by all CNRA/CSNI member countries. Experience with the use of these indicators should be reported to the CNRA/CSNI.

2/ Further work to harmonise the definitions of the various indicators is needed; this can be carried out in a follow-up meeting of the Task Force.

3/ There is development work being performed by various organisations, CSNI/CNRA should authorise the Task Group to meet periodically to assess these new developments and provide recommendation for a common set of performance indicators.

4/ CNRA/CSNI should also promote exchange of other nationally collected PI’s among interested regulators. However, new indicators need to be evaluated before added to list. For this task CNRA/CSNI should assign a group to harmonise definitions and evaluate experience with new indicators.

5/ Development of standard tools to display, interpret and analyse trends would be useful; this work could be carried out by the current Task Force supported by the NEA Secretariat.

6/ Given the vast experience acquired by WANO in operating a universal system of performance indicators, the Task Force recommends that co-operation with WANO be intensified.

7/ Co-operation with the IAEA should also be intensified to support them in their effort to promote the use of performance indicators world-wide.

Referencies

1. Improving Regulatory Effectiveness (NEA/CNRA/R(2001)3)

2. Summary Record of the Twelfth meeting of the Committee on Nuclear Regulatory Activities (CNRA), NEA/SEN/NRA (2001)1

3. Summary Record of the Twenty Eighth Meeting of the Committee on the Safety of Nuclear Installations (CSNI), NEA/SEN/SIN (2001)1

4. Proceedings Safety Performance Indicators, Madrid, Spain, October 17 – 19, 2000.

Appendix 1.- Programme Overview

Specialist Meeting on Safety Performance Indicators **Madrid, Spain – October 17-19, 2000**

	TUESDAY 17	WEDNESDAY 18	THURSDAY 19
8.00-9.00	*Registration*	*Registration*	*Registration*
9.00-9.30		**REGULATOR INDICATORS (Part 1 -21)** *Chair : A. Gea (CSN)* ▪ E. Seidel (Germany): PIs at Bavarian NPP ▪ P. Tiippana (STUK – Finland): Development and Use of Safety Indicators at STUK • Sae-Yul Lee, KINS (*Korea*): Experience in the Use of PIs in Korea	
9.30-10.00	***OPENING (10)*** *A. Martín (CSN) A. Carnino (IAEA),) , K. Shimumora (NEA) F. Ynduráin (CIEMAT*		ROUND THE TABLE DISCUSSION ON OPORTUNITIES FOR INTERNATIONAL CO-OPERATION ON PERFORMANCE INDICATORS (31) *Chair: J. Johnson (NRC)* *Table: M. Raymond DSIN L. Carlsson (NEA) L. Lederman (IAEA), P. Baranowsky (NRC)* ▪ J. Zarzuela (CSN – Spain): Presentation of a proposal for discussion
10.00-10.15	***UTILITY INDICATORS (11)*** *Chair: S. Floyd (NEI - USA) / Jürgen Schlegel (WANO)* ▪ S. Floyd (NEI – USA): U.S. Industry Perspectives on Role of Indicators in the Regulatory Process - ▪ D. Hickman (NRC-USA): Performance Indicators in the USNRC's Revised Reactor Oversight Process." ▪ L. Dumont (EDF -France): Can Safety be measured? ▪ C. Atkinson (B. Energy – UK): International PIs and the UK Nuclear Energy Generators.		
10.15-10.30		**10.15-10.30 BREAK**	
10.30-11.00		**REGULATOR INDICATORS (Part 2 - 22)** Chair: JJ V. Binnebeck (AVN - Belgium) ▪ M. Maroño (CIEMAT – Spain): New PIs System in Spain ▪ Khatoon (Pakistan): Development of Safety PI of Regulatory Interest (SAFFER) in Pakistan ▪ M. Raymond (DSIN – France):The development of safety indicators for NPP at the French Safety Authority	
11.00-11.30			
11.30-12.00	*COFFEE BREAK*		
12.00-13.00	***RISK INDICATORS (12)*** *Chair: M. Khatib-Rahbar (ERI - USA)/ U. Schmocker (HSK)* ▪ M. Khatib- Rahbar (ERI – USA): An Approach to Development of a Risk-Based Safety Performance Monitoring System for Nuclear Power Plants ▪ R. Himanen (TVO – Finland): Use of WANO PI and living PSA in Okiluoto NPP ▪ U. Schmocker (HSK– Switzerland): PIs: Relationship to Safety and Regulatory and Inspection Programs ▪ J. Suárez (IBERINCO – Spain): Risk Indicators at Cofrentes NPP	***REGULATOR INDICATORS (Part 3 -23)*** *Chair: P. Tiippana* ▪ J. Tanaka, NUPEC (Japan): Development of Safety PI in Japan ▪ O.V. Pecherytsya (SSTC –Ukraine): Development of NPP safety PI system at Ukrainian Regulator ▪ T. Hill (CNS- South Africa): Safety Indicators in the Nuclear Regulatory Process ▪ R. Rehacek (Czech Republic): Regulatory Body Experience with the Safety Indicator Use	**CLOSURE** **ROUND TABLE OF SESSION CHAIRMEN BRIEFING CONCLUSIONS (32)** *Chair : A. Alonso CSN / J. Zarzuela (CSN)* All session chairmen
13.00-13.30			
13.30-15.00	**LUNCH**		

	TUESDAY 17	WEDNESDAY 18
15.00-16.30	INTERNATIONAL PERFORMANCE INDICATOR SETS (13) *Chairman: L. Lederman (IAEA)/:L. Carlsson (NEA)* ▪ L. Lederman (IAEA): Indicators to Monitor NPP Operational Safety Performance ▪ J.J. Van Binnebeck (AVN- Belgium): Results of PWGIP Baltimore meeting related with PIs ▪ K. Hamlin, Y. Shimada (WANO): WANO PIs	*ORGANISATION &* ***SAFETY CULTURE INDICATORS (24)*** *Chair: Janos Toth (Paks, NPP) / F. Calduch (Cofrentes NPP)* ▪ F.Calduch (Cofrentes NPP-*Spain):* Cofrentes NPP Indicators to Monitor Operational Safety Performance ▪ S. Bardou (IPSN– France): Indicators of Plant Performance During Events Identified by Recuperare Method ▪ J. Toth (Paks NPP- Hungary): Assessment of Human Performance and Safety Culture at the Paks NPP ▪ C. Fang (Daya Bay NPP –China): PI at Daya Bay NPP

Appendix 2: PERFORMANCE INDICATORS TASK FORCE

LIST OF PARTICIPANTS – MEETING 19-20 FEBRUARY 2001

Issy-les-Moulineaux

Mr Petteri TIIPPANA
STUK-Radiation & Nuclear Safety Authority
P.O. Box 14
FIN-00881 Helsinki
E-mail: petteri.tiippana@stuk.fi

Mr. Didier WATTRELOS
IPSN
CEA-FAR
BP 6
F-92265 Fontenay-aux-Roses
E-mail: didier.wattrelos@ipsn.fr

Mr. Jose M. BALMISA
Consejo de Seguridad Nuclear
C/Justo Dorado, 11
28040 Madrid
E-mail: jmbg@csn.es

Mr. Javier ZARZUELA
Consejo de Seguridad Nuclear
C/ Justo Dorado, 11
28040 Madrid
E-mail: jzj@csn.es

Dr. John W. BARTLETT
NII-Nuclear Safety Directorate
St Peter's House
Stanley Precinct
Bootle, Merseyside L20 3LZ
E-mail: john.bartlett@hse.gsi.gov.uk

Mr. Jürgen SCHLEGEL
WANO-PC
43 rue Vineuse
75116 Paris
FRANCE
E-mail: schlegelj@wanopc.org

Mr. Lennart CARLSSON
OECD/Nuclear Energy Agency
Bât. B - Le Seine Saint Germain
12, Boulevard des Iles
92130 Issy-les-Moulineaux
FRANCE
E-mail: lennart.carlsson@oecd.org

APPENDIX 3

Spain running set of performance indicators:

- **Automatic scrams while reactor critical**
- Safety systems actuations
- Significant events
- Safety system failures
- Forced outage rate
- Forced outage rate for 1000 critical hours of critical commercial operation
- Radiation exposure to workers

Future Performance Indicators of Spanish NPP

<table>
<tr><th colspan="2">PERFORMANCE INDICATOR</th><th>AREA</th></tr>
<tr><td colspan="2">Unit Capability Factor (%)</td><td rowspan="4">PERFORMANCE STABILITY</td></tr>
<tr><td colspan="2">Non scheduled shutdowns/year (excluding SCRAMS)</td></tr>
<tr><td colspan="2">SCRAMS/7000 critical hours (auto + manual)</td></tr>
<tr><td colspan="2">Non scheduled Safety System Actuation [(1)] /year</td></tr>
<tr><td colspan="2">Safety System Failures[(2)] / year</td><td rowspan="2">RELIABILITY OF MITIGATING SYSTEMS</td></tr>
<tr><td colspan="2">Safety System Unavailability / year</td></tr>
<tr><td colspan="2">RCS Activity (% TS limit)</td><td rowspan="2">BARRIERS INTEGRITY</td></tr>
<tr><td colspan="2">RCS Identified Leakage (% TS limit)</td></tr>
<tr><td colspan="2">Collective Radiation Exposure to workers (Sv-year)</td><td rowspan="4">RADIOLOGICAL IMPACT</td></tr>
<tr><td>Volume of Low and Medium Level Solid Radioactive Waste</td><td rowspan="3">Under Consideration whether to substitute these three indicators by a new one:
radiological dose to critical individual of public</td></tr>
<tr><td>Activity of Gas Radioactive Release</td></tr>
<tr><td>Activity of Liquid Radioactive Release</td></tr>
</table>

[(1)] Safety System Actuation: It is counted as long as the challenged System fulfils its function: to inject water, to supply power. Auxiliary Feedwater actuations, when properly actuated, e.g., following a scram, are excluded.

[(2)] The Safety Systems considered vary at different reactor design:

PWR.	BWR
Emerg. AC Power system	Emerg. AC Power system
HPSI	HPCI/HPCS
RHR	IC/RCIC
AFW	RHR

Running set of performance indicators in Finland:

SAFETY OF NUCLEAR FACILITIES	
A1 Safety and quality culture	
A1.1 Failures and their repairs	number of failures of TS equipment
	ratio between corrective and preventive maintenance tasks
	repair time of failures of TS equipment
	number of single and multiple maintenance erros (CCF)
	number of technical common cause failures
A1.2 Number of TS deviations	**number of non compliancies with TS**
	number of exemptions from TS
A1.3 Availability of safety systems	unit specific WANO indicators
A1.4 Radiation doses	**annual collective dose**
	annual average of ten highest doses
A1.5 Radioactive releases	radioactive releases to the atmosphere
	radioactive releases to the water system
	calculated dose of the most exposed person living near the NPP
A1.6 Documentation	number of unupdated documents on the outage related to the plant modifications implemented during previous outage *(planned)*
A1.7 Investments	annual investment rate to plant modernisation
A2 Operational events	
A2.1 Number of events	number of reported operational events according to Guide YVL 1.5
A2.2 Significance of events	calculated risk significance of
	TS exemptions
	failures of TS equipment
	preventive maintenance of TS equipment
	operational events
A2.3 Causes of events	number of events caused by organisational factors
	number of events caused by technical factors
A2.4 Number of fire alarms	number of malfunctions
	number of real fire alarms
	number of fires
	number of other rescue missions
A3 Structural integrity	
A3.1 Integrity of nuclear fuel	a maximum activity of the primary circuit equivalent to I-131
A3.2 Integrity of primary circuit	WANO chemistry index
	volume of identified and unidentified leakage *(planned)*
A3.3 Integrity of containment	overall leakage of isolation valves compared with the highest allowed overall leakage of the isolation valves
	percentage of isolation valves that passed the leakage test at the first attempt
	an overall leakage of containment's entrance and other holes in relation to the highest allowed overall leakage of these holes

Introductory Indicator System for SKI

DEF. IN DEPTH	BARRIERS	
1.Prevention of abnormal oper. and failures	***1.1 Robust design and constr.:***	
	1.1.1 Fuel	1.1.1 No. of fuel failures
	1.1.2 Primary press.boundary	1.1.2. Later
	1.2 High qual. in maint.and oper.	1.2.1 Rate of violations of Tech.Spec by plant pers. contr.and others. 1.2.2 Rate of maint. problems (repeated maint. or overdue maint).
Initiating Events	***1.3 Initiating events***	1.3.1 No.of scrams 1.3.2 No. of safety system initiations
2. Control of abnorm. operation and detection of Failures	***2.1 Robust superv. systems***	2.1.1Unavail. for supervision and protect. systems 2.1.2 No. of incid.w.failing syst. at scram
	2.2 High quality in maint. and oper.	2.2.1and 2.2.2. See 1.2.1 o 1.2.2
3.Control of accid. within the design Basis	***3.1 Effective safety systems***	3.1.1Unavail of safety systems 3.1.2 Unavail.of separat. barriers 3.1.3 No. of leaking cont isolat. valves
	3.2 High quality in maint. and oper.	3.2.1 and 3.2.2. See 1.2.1 o 1.2.2
	3.3Effective emerg. oper. proceed.	
4. Control of severe plant conditions, incl. prevent. of accident progress. and mitigating of the consequences of severe accidents	***4.1 Conseq. mitigating measures***	4.1.1 Unavail.of conseq. mitig systems 4.1.2 Unavail. of supervision systems
	4.2 High quality of oper. and maint.	4.2.1and 4.2.2. See 1.2.1 o 1.2.2
	4.3 Physical protection	Later
	4.4 Effective accid.management	Later
5. Mitigation of radiological conseq. of significant releases of radioact. materials	***5.1 Prep. measur.for eff. Info to and protect. of the population***	Later
6. Global safety		6.1.1 Unplanned loss of production

WANO Performance Indicator Programme

The WANO Performance Indicator Programme supports the exchange of operating experience information by collecting, trending and disseminating nuclear plant performance data in eight key areas. The data is quarterly gathered for a set of quantitative indicators of plant performance in the areas of plant safety and reliability, plant efficiency and personal safety.

These indicators are intended principally for use as a management tool by nuclear operating organisations to monitor their own performance and progress, to set their own challenging goals for improvement and to gain additional perspective on performance relative to that of other plants.

The internationally agreed programme is continuously reviewed and further developed to allow individual plants to compare their performance more easily with industry average values. For many years the level of reporting data has grown to nearly 100% of the operating nuclear power plants reporting at least seven indicators.

Overall results are annually issued for public information. Confidential unit results are quarterly updated and continuously available for all WANO members.

The WANO Performance Indicator set comprises since year 2001:

Unit Capability Factor
The unit capability factor is the percentage of maximum energy generation that a plant is capable of supplying to the electrical grid, limited only by factors within control of plant management.

Unplanned Capability Loss Factor
The unplanned capability loss factor is the percentage of maximum energy generation that a plant is not capable of supplying to the electrical grid because of unplanned energy losses, such as unplanned shutdowns or outage extensions, limited only by factors within control of plant management.

Unplanned Automatic Scrams per 7000 Hours Critical
The unplanned automatic scrams per 7000 hours critical indicator tracks the mean scram (automatic reactor shutdown) rate for approximately one year (7000 hours) of operation.

Collective Radiation Exposure
The collective radiation exposure indicator monitors the effectiveness of total personnel radiation exposure controls.

Industrial Safety Accident Rate
The industrial safety accident rate tracks the number of accidents that result in lost work, restricted work or fatalities per 200 000 work-hours.

Safety System Performance
The safety system performance indicator monitors the availability of three important standby safety systems at each plant.

Fuel Reliability
The fuel reliability indicator monitors progress in preventing defects in the metal cladding that surrounds fuel.

Chemistry Performance
The chemistry performance indicator provides an indication of progress in controlling chemical parameters to retard deterioration of key plant materials and components.

The latter three indicators are defined in a manner that reflects differences in plant-specific design, configurations or operational practices. As a result, data cannot simply be summarised across all reactor types for comparison purposes.

The volume of solid radioactive waste indicator and thermal performance indicator as well were cancelled as WANO performance indicators, because the WANO members felt that these indicators were internationally not very well comparable and not very well connected to the WANO mission respectively.