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IAEA-CN-120
<table>
<thead>
<tr>
<th>TOPICAL ISSUE 1: CHANGING ENVIRONMENTS – COPING WITH DIVERSITY AND GLOBALIZATION</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety culture of nuclear R&amp;D organizations: a case study (IAEA-CN-120/11)</td>
</tr>
<tr>
<td>I.J. Obadia, M.C.R. Vidal, P.F. Frutuoso e Melo</td>
</tr>
<tr>
<td>Management of a multinational project on Kozloduy NPP modernization (IAEA-CN-120/21)</td>
</tr>
<tr>
<td>M.G. Yankov</td>
</tr>
<tr>
<td>Current issues in spent nuclear fuel management after the nuclear power programme termination (IAEA-CN-120/24)</td>
</tr>
<tr>
<td>V.M. Paliukhovich, A.A. Tukhto</td>
</tr>
<tr>
<td>Problem of nuclear power (IAEA-CN-120/40)</td>
</tr>
<tr>
<td>E. Batjargal, S.Enkhbat, P.Zuzaan, T.Tuul</td>
</tr>
<tr>
<td>Japanese contributions to Asian nuclear safety network (ANSN) (IAEA-CN-120/51)</td>
</tr>
<tr>
<td>F. Kudough</td>
</tr>
<tr>
<td>Used nuclear fuel treatment in the 21st Century (IAEA-CN-120/63)</td>
</tr>
<tr>
<td>M.-F. Debreuille, J.-G. Devezeaux De Lavergne, P. Kaplan, S. NG, R. Vinoche</td>
</tr>
<tr>
<td>Range of protective actions for nuclear power plant incidents - a case study (IAEA-CN-120/75)</td>
</tr>
<tr>
<td>A.P. Nelson</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TOPICAL ISSUE 2: OPERATING EXPERIENCE – MANAGING CHANGES EFFECTIVELY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Improving industry event trending to avoid significant events (IAEA-CN-120/2)</td>
</tr>
<tr>
<td>P.J. Di Rito</td>
</tr>
<tr>
<td>Analysis of failure rates in an operating variable energy cyclotron (IAEA-CN-120/16)</td>
</tr>
<tr>
<td>L.M. Chowdhury, P.K. Sarkar</td>
</tr>
<tr>
<td>Topical questions of managing safety of BN-350 reactor facility while transitioning from an operating to a decommissioning status (IAEA-CN-120/18)</td>
</tr>
<tr>
<td>S. Shiganakov, T. Zhantikin, A. Kim</td>
</tr>
<tr>
<td>Detection of damaged fuel assembly in LVR-15 reactor with spectrometric water activity measurement (IAEA-CN-120/23)</td>
</tr>
<tr>
<td>L. Viererbl, M. Marek, A. Voljanski, V. Broz, J. Burian</td>
</tr>
<tr>
<td>Application of lessons learned from Qinshan Phase III to a future project (IAEA-CN-120/29)</td>
</tr>
<tr>
<td>R. DeGregorio, S. Pang, K.J. Petrunik</td>
</tr>
<tr>
<td>Lack of safety culture as a contributing factor in major radiation accidents reported in Latin and South America (IAEA-CN-120/31)</td>
</tr>
<tr>
<td>G. Jean-Pierre</td>
</tr>
<tr>
<td>Scientific supervision of NPP Temelin commissioning (IAEA-CN-120/33)</td>
</tr>
<tr>
<td>I. Váša, C. Svoboda</td>
</tr>
<tr>
<td>Upgrading of nuclear power plants in the Slovak Republic (IAEA-CN-120/43)</td>
</tr>
<tr>
<td>M. Ziaikova</td>
</tr>
<tr>
<td>Operating experience in support of managing changes in Indian nuclear plants (IAEA-CN-120/47)</td>
</tr>
<tr>
<td>R. Chowdhury, P.V. Varde</td>
</tr>
<tr>
<td>Nuclear energy and system dynamic model for energy policy makers in the Slovak Republic (IAEA-CN-120/48)</td>
</tr>
<tr>
<td>S. Chakraborty; L. Tomik; W. Hoffelner; A. Stoian</td>
</tr>
<tr>
<td>Lessons learned in configuration management of NPPs with WWER440 (IAEA-CN-120/50)</td>
</tr>
<tr>
<td>P. Hlavac, Z. Kovacs, R. Spenlinger</td>
</tr>
</tbody>
</table>
Development of data base system for the inspectors of nuclear installations (IAEA-CN-120/52)...........................90
K. Aizawa

Safety management in industrial reprocessing: Lessons learned from COGEMA’s experiences (IAEA-CN-120/62)........................................................................................................92
R. Vinoche, C. Baganz, P. Mathieu

TOPICAL ISSUE 3: REGULATORY MANAGEMENT SYSTEMS – ADAPTING TO CHANGES IN THE ENVIRONMENT

Safety assessment methodology for NORM disposal trench at Abu Redees Site (IAEA-CN-120/3) ...97
G. Haseeb; A. A. Tawfik; M. Abdel Geleel

Iranian nuclear regulatory authority’s challenges in licensing of Bushehr NPP (IAEA-CN-120/8) ...101
F. Dastjerdi

Communicating risk to the public (IAEA-CN-120/9) .............................................................................................103
A.S. Greenman

Risk communication – the key of the policy success (IAEA-CN-120/10) .........................................................107
V. Covalschi

Some issues of Armenian NPP safety evaluation (IAEA-CN-120/12).................................................................111
A.A. Gevorgyan, A.M. Martirosyan, G.R. Markosyan

Decontamination of 125I in medical laboratory (IAEA-CN-120/14) .................................................................115
M. Abdel Geleel; Amaal A.Tawfik

Probabilistic safety analysis in particle accelerators (IAEA-CN-120/15) ........................................................120
L.M. Chowdhury, P.K. Sarkar

Authorization of nuclear installation personnel in Kazakhstan (IAEA-CN-120/19) .............................................124
A. Kim, M. Idrissova, Sh. Zhetbaeva, B. Shaikhislamova

Applications of probabilistic safety analysis (Level-3) in nuclear installation safety
(IAEA-CN-120/20)........................................................................................................................................128
L. Sági

The management systems and the role of the regulatory body (IAEA-CN-120/26).............................................135
I.P. Salati de Almeida

International feedback and safety reassessment - the French FBFC nuclear fuel fabrication plant case
(IAEA-CN-120/28)........................................................................................................................................140
J.P. Carreton, J.M. Dormanta, A. Denys, J. Jaraudias

International licensing of the ACR-700 (IAEA-CN-120/30).........................................................................145
V.G. Snell, N. Popov, V. Langman

Improvement of safety in reactor conceptual design: a cost-effective approach
(IAEA-CN-120/38)........................................................................................................................................149
M. O. Giménez, M.A. Schlamp

Safety improvement program NPP Mochovce (IAEA-CN-120/41).................................................................154
J. Sádovský

The national infrastructure on radiation safety of Azerbaijan (IAEA-CN-120/49) .............................................166
T. Aliyev, A. Mammadov, R. Babayev

Pursuit of new methodology on risk communication - research assistance program by open
application (IAEA-CN-120/53)........................................................................................................................170
N. Konoa, K. Takeshima

Use of probabilistic safety assessment in JNES (IAEA-CN-120/54)..................................................................174
M. Fukuda, M. Yamashita, M. Hirose, T. Uchida, M. Hirano

Towards safety culture strengthening (IAEA-CN-120/55).................................................................................178
M. Makino

Study on inspection capability of ultrasonic testing for fatigue crack in piping
(IAEA-CN-120/57).............................................................................................................................................182
K. Aono, H. Miharada, J. Sanoh, S. Takeyama
TOPICAL ISSUE 4: LONG TERM OPERATIONS – MAINTAINING SAFETY MARGINS WHILE EXTENDING PLANT LIFETIMES

Periodic safety review of the Budapest research reactor (IAEA-CN-120/4) ..............................................186
S. Tőzsér, J. Gadó, K. Késmárky and I. Vidovszky

Experience with licensing of Russian NPP operation extension (IAEA-CN-120/13) .................................191
A.B. Malyshev, B.G. Gordon, M.V. Kuznetsov

Ageing management review for main components of PWR nuclear power plant
(IAEA-CN-120/17) ........................................................................................................................................196
DOU Yikang, HE Yinhao, XU Xuelian, ZHANG Minga, HE Yunsheng, CAO Jian

Study and analysis of degradation processes in the electronic equipment operating at Kozloduy NPP
(IAEA-CN-120/22) ......................................................................................................................................202
A. Popov, N. Naydenov

Life extension and safety upgradation in Indian NPPs – A Regulatory Perspective
(IAEA-CN-120/34) ........................................................................................................................................206
J. Koley, R. Venkata Raman, S.K. Chande

Regulatory approach to safety of nuclear power plants built to earlier safety standards
(IAEA-CN-120/35) ........................................................................................................................................215
S. Forsberg

Present status of Japanese national research projects relating to ageing of nuclear power plants
(IAEA-CN-120/56) ........................................................................................................................................219
J. Sanoh, T. Noda, K. Maeda

Environmental impact assessment in the frame of licensing lifetime extension of Paks NPP at Hungary
(IAEA-CN-120/60) ......................................................................................................................................223
G. Volent, G. Németh

Operator – regulator interface in safety assessment of Kanupp for life extension
(IAEA-CN-120/64) ........................................................................................................................................226
W.M. Butt, A. Zia
TOPICAL ISSUE 1:
CHANGING ENVIRONMENTS – COPING WITH DIVERSITY AND GLOBALIZATION
SAFETY CULTURE OF NUCLEAR R&D ORGANIZATIONS:
A CASE STUDY

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Abstract. After the Chernobyl nuclear accident in 1986, the International Atomic Energy Agency (IAEA) established the safety culture concept as a proactive mean to contribute to safety improvement, starting a worldwide safety culture enhancement program within nuclear organizations mainly focused on nuclear power plants. More recently, the safety culture concept has been extended to non-power applications such as nuclear research reactors and nuclear technological research and development organizations. In 1999, the Nuclear Engineering Institute (IEN), a research and technological development unit of the Brazilian Nuclear Energy Commission (CNEN), started a management changing program aiming at improving its performance level of excellence. IEN is classified as a nuclear and radioactive installation, where a nuclear research reactor and two cyclotron-type particle accelerators are operated, and several nuclear and radioactive processes related to industrial, medical and environmental applications are performed. This changing program has been developed based on the assumptions that safety is a fundamental condition for excellence, and safety culture is crucial to the improvement of operational safety. A management system comprised of a safety culture enhancement practice integrated to a quality management process, based on a systematic and adaptive approach, has been developed and implemented at IEN. This paper presents the safety culture enhancement management practice and presents the corresponding evolution of IEN’s safety culture and safety indicators results.

1. Introduction

Due to the globalized, fast changing and more complex scenario, industrial organizations are facing threats and opportunities that affect their success and survival. Therefore, the implementation of organizational changing programs in search for excellence has become crucial. The 1999 Special Report of the World Association of Industrial Technological Research Organizations [1] observes that within the scope of research and development (R&D) organizations the scenario is the same. In hazardous technology organizations, such as the nuclear ones, quality and safety are inextricably linked. Therefore quality is a necessary but not a sufficient condition to excellence, since the establishment of a safety management system is required as well. Deep analyses of several incidents in these organizations have demonstrated that, due to technological development, a major part of its root causes depend no more on technical factors, but mainly on human and organizational issues. This has given rise to the development of new organizational approaches and methods aiming at improving the effectiveness of safety management systems. In the nuclear field, IAEA’s analyses of the TMI and Chernobyl accidents have given rise to the safety culture approach [2], mainly focused on nuclear power plants and on the regulatory body. Lately, this approach has been extended to nuclear fuel cycle organizations and nuclear research reactors. Nevertheless, its application on nuclear non-power technological research and development organizations related to the health, agricultural, industrial and environmental sectors, practically has not been addressed by IAEA. Only recently, during the IAEA International Conference on Safety Culture in Nuclear Installations [3], this application has been included within the discussed topics. An adaptive safety-oriented management system in search for excellence has been developed and implemented at IEN, based on the assumptions that in nuclear organizations excellence means, overall, excellence in safety, and for that, an effective safety culture is
I.J. Obadia et al.

a crucial condition [4]. This paper highlights the adaptive safety culture enhancement management practice which is part of the system developed and presents the corresponding evolution of IEN’s safety culture in the period from 2001 to 2003, as well as the results of IEN’s safety indicators.

2. Organizational Management Change and Organizational Culture

According to Schein [5] the elements of the organizational culture must be analyzed in three levels, as shown in Fig. 1, and effective cultural changes require modifications on the basic underlying assumptions level. Hofstede [6] observes that the way organizational culture mainly affects staffs is through its shared routine practices. When the members of the organization get a shared perception of the good results achieved by the new practices, a process of cognitive transformation starts so that the new practices begin to be taken for granted, becoming unconsciously accepted, thus modifying the cultural underlying assumptions [5]. So, one can assume that the organizational culture may be effectively modified by the introduction of new systematic management practices within the organization. Therefore, as illustrated in Fig. 2, the organizational culture and the daily management practices, and their corresponding results – which comprise the management process – are deeply interrelated under a complex fashion, where the organizational culture is influenced by the management practices and their results, while simultaneously an effective implementation of the management process is influenced by the organizational culture.

![Fig.1. Levels of the organizational culture [5].](image1.png)

![Fig. 2. Complex inter-relationship between the organizational culture and the management process.](image2.png)

3. The Adaptive Safety Culture Enhancement Practice

Considering the complex causal inter-relationships that occur between the implementation of the management system and the organizational culture, the integrated adaptive management system of Fig. 3 has been developed, assuming that in hazardous technology organizations the organizational culture can be represented by the safety culture. In this context, IAEA observes that in a nuclear organization safety culture is the dominant aspect of the organizational culture [7].

![Fig.3. Integrated adaptive management system](image3.png)
The IAEA safety culture self-assessment approach [2] is integrated to a quality management process based on the Model of Excellence of the Brazilian Quality Award (BQA) [8], that comprises a sociotechnical, holistic and non-prescriptive result-oriented quality management process structured on the following eight criteria of excellence: Leadership; Strategies and Plans; Clients; Society; Information and Knowledge; Personnel; Processes; and Results. This model considers that the organization's evolution towards excellence is a function of its continuous learning process, obtained through the evaluation of the practical results achieved by the implemented management practices, and the consequent implementation of improvements and innovations on the management system. The safety culture self-assessment identifies the critical organizational factors that require improvement. These organizational factors are then correlated to the criteria of excellence of the BQA management process, enabling new practices to be implemented within the scope of the requirements of each of these criteria. The new systematic practices implemented within the quality management process, and their consequent results, are supposed to affect organizational and individual attitudes and behaviors and, therefore, modify the associated safety culture underlying assumptions.

During each operational cycle the system undergoes a simultaneous and continuous adaptation, enabling modifications on the safety culture underlying assumptions, and simultaneously facilitating an effective implementation of the management process, thus providing a degree of governability to the system.

4. The Case of the Nuclear Engineering Institute (IEN)

In 1999, IEN started a management changing program aiming at reaching its 2005 Vision: “to be a recognized research and technological development center of excellence in the country by its effective contributions to the improvement of society’s quality of life”. IEN is a Brazilian nuclear technological R&D organization which develops activities in the fields of reactors, materials, chemistry, radiopharmaceuticals, instrumentation, human reliability, radiation protection, and radioactive waste management, providing technology to contribute to the improvement of the industrial, health, and environmental sectors. Some activities are performed under hazardous conditions such as the operation of the Argonauta nuclear research reactor, and CV-28 and RDS-111 cyclotron-type particle accelerators, and the exposure and handling of radioactive material. This changing program has been conducted based on the integrated adaptive system developed and on a facilitated changing type of intervention.

4.1. The Improvement Action Plan

The system improvement action plan results from the organizational learning process which takes into consideration: a) the results of IEN's performance critical analysis; b) the safety culture self-assessments; and c) the level of excellence achieved by the management system – performed through the BQA assessment method. It has been assumed that complex practical aspects are crucial to the success of organizational changing programs. Hence, a facilitated changing type of intervention, based on the self-organizing properties of the complex adaptive systems has been developed and actually used, instead of the 'planned-change' type of intervention commonly used, which is framed under the mechanical paradigm that ‘sees’ the organization as a controllable machine and whose results have little lasting radical effects [9]. The intervention has been implemented under an internal advisory framework, and conducted on an open, transparent and participation basis, with hierarchical power free discussions and no prescriptions, where all implemented improvements and innovations to the system have emerged from the following working committees whose own creation have also emerged as a result of the intervention developed:

- Eight Committees related to each criteria of excellence of the BQA Model;
- Safety Culture Committee, composed of members from all sectors of IEN, whose general objective is to foster institutional discussions related to the safety issues of all IEN’s activities. As
specific objective, the committee coordinates all safety culture activities, such as the identification of new management practices considering self-assessment results, the performance of new self-assessments, internal communication, etc. An important result of this committee was the establishment, in 2003, of IEN's Safety, Environmental and Health Policy.

4.2. Results

The evolution on the mean index of the 22 organizational factors assessed in years 2001 and 2003 is presented in Fig. 4 and the mean index of IEN's safety culture is shown in Fig. 5. One can observe that except for the organizational factor 9, which was slightly reduced in 2003, all other 21 factors have been improved, and that IEN's mean safety culture index has increased from 'regular' to 'satisfactory'. Figs. 6 and 7 show the results of IEN's accident indicators from 1999 to 2003.

![Fig.4. Evolution of the 22 organizational factors.](image)

![Fig. 5. IEN's safety culture mean index.](image)

![Fig. 6. Personal accident rate.](image)

![Fig. 7. Number of accidents involving ionizing radiation.](image)

5. Conclusions

Only more recently, IAEA has started to discuss the application of its safety culture approach to nuclear R&D organizations. The case study developed at IEN has demonstrated that the integrated adaptive system developed has provided an effective safety culture enhancement, the establishment of IEN's Safety, Environmental and Health Policy and a significant reduction on personal accidents rates. Therefore, one can conclude that the safety culture approach can be equally applied and is equally crucial to nuclear non-power R&D organizations as it is to the nuclear power ones. IEN has been a pioneer nuclear technological R&D organization in the implementation of a safety culture enhancement program integrated to a quality management system in search for excellence.

REFERENCES


1. Introduction

Kozloduy NPP is a joint stock company with 100 % state share. There are six pressurized water reactors constructed at the site with total installed capacity of 3760 MW, as well as a spent fuel storage facility.

The first four units, VVER –440, model B-230, were constructed and commissioned within the period 1970-1982, units 5 and 6, VVER – 1000, model B-230 in 1987-1991. Following a Government decision units 1 and 2 were shut down and disconnected from the national electricity grid on 31st December, 2002 and they were set into “E” status according to the technological regulations for operation.

 Nuclear Energy in the Republic of Bulgaria is faced with an increasing number of challenges in relation with the country groundwork for European Union accession. This accession means complying with the European Union’s high requirements for safe electricity generation and meeting all requirements in terms of environment protection and energy efficiency principles.

2. Background

Kozloduy NPP response to these aggressive environmental conditions is its commitment:

- to maintain the plant safety level in line with the latest international standards and

- to implement the necessary improvements regarding the units enhanced reliability and availability, and thus the plant competitiveness.

The most extensive investment project at Kozloduy NPP is the modernization program which is now in process of implementation at units 5 and 6. The modernization was launched as a response to the reported non compliances between the performance of the 1000 MW units and the requirements, developed in the nineties, in terms of plant safety, scope, number and quality of analyses and accidents.

Units 5 and 6 Modernization Program was developed on the basis of the complete range of IAEA recommendations for VVER 1000 (model B-320) units, and it was arranged as a set of 212 specific measures, allocated according to their main objective. The expected result of these measures implementation is to achieve:

- Units 5 and 6 safety enhancement by way of introducing new design solutions;

- Substantiation of adequate safety level through performing different analyses and additional studies, in conformity with international adopted normative documents;
M.G. Yankov

- Reliability enhancement by means of replacing equipment which life time is running out as well as critical equipment;

- Improvement of operation efficiency and conditions of operation.

The investments, associated to the implementation of this project, are definitely impressive. The financial resources, envisaged for the Modernization Program realization, amount to the total sum of 491 M€. About 135 M€ are KNPP intended own resources, and about 356 M€ are provided through credit agreements with different loan institutions.

By eliminating the design deviations from the current international practice in terms of safety, the units’ availability is improved. The implementation of the planned analyses and replacement of obsolete components and systems with new more reliable ones will enable us to move to the concept of risk-oriented outage and to shorten units outage, hence to increase units productivity.

For all these large investment projects, we use as a background the best world experience in this field. Since the very beginning of the concept for extensive modernization of units 5 and 6, we have always relied on a number of Bulgarian and foreign specialists. The Modernization Program was developed by specialist from Kozloduy NPP, Energoproekt - JSC, Risk Engineering Ltd, and EDF, taking into account IAEA requirements. So developed, this program has been reviewed two times concerning its comprehensiveness and adequacy (in 1995 and 2000), whereas in 1997 it was the subject of independent review and of Risk Audit review with participating specialists from French and German regulatory authorities (IPSN and GRS International).

2.1. The contractors for the priority measures were selected via international tender held in 1996, and the winners were the American company Westinghouse and the European Consortium Kozloduy established by the three leading European companies in the field of nuclear energy – Framatome, Siemens and Atomenergoeksport. Since the beginning of the new engineering phase in 1998, along with the Bulgarian team, Parsons Energy & Chemical works as a corrective company and a technical consultant providing assistance both by its office in the US and by its team at the plant site, including specialists from the USA, Great Britain, Italy, Russia and Bulgaria. Methodological and organizational support was provided at the beginning of the project by the Consortium including the Spanish Empresarios Agrupados and British Energy.

2.2. Organization and management of units 5 and 6 Modernization Program

Units 5 and 6 Modernization Program is connected to specific organizational problems. In order to perform this task, Kozloduy NPP has established a separate structural division, which acts both as program manager and coordinator within KNPP and is liable for the efficiency of performance and program success. The organization of this Modernization and Investment Division has a multidiscipline structure of matrix type, comprising engineering, planning, programs research and analysis, quality assurance, contracts administrative management, financial resources planning, costs management and reports and documentation control. The liabilities of the project participants are described in the Project Management Guidelines developed for units 5 and 6 Modernization Program (MP). The adequacy of liabilities allocation has been tested for almost two years of intensive work until the present moment, and has been verified by a lot of design solutions, made within this period.

2.3. Quality Assurance

Units 5 and 6 Modernization Program is implemented according to the quality assurance requirements by international (IAEA, ISO) and national (Nuclear regulatory agency) standards. Because of the project complex character, specific procedures were developed in Kozloduy NPP on the program activities regulation, and in particular –those of the main contractor, who provides quality supplies for KNPP. Each contractor has its own quality assurance program, meeting international and national standards. Except for that, KNPP and one of the contractors have developed together joint procedures
on quality and organization, covering common functions and activities. Contractors’ supplies were implemented in compliance with their internal quality programs and the relevant quality plans were reviewed to guarantee that each quality step has been reviewed and supervised by the authorized personnel. Contractors’ deviations from quality requirements have been reflected in notes to supplies, before they were accepted, and in case there were recurring deviations signal notes for non performance were issued for corrective actions and termination of deviations. Up to the present, the program has demonstrated a small number of non compliances; however documentation supplies had to be revised by the Contractors in order to meet receipt criteria, based on Kozloduy NPP and Parsons notes on quality.

“Learned lessons” plans were developed together with the two main Contractors. These plans reviewed and analyzed the activities and results from the performed work during units 5 and 6 outages in 2002. Positive sides were emphasized, so that they could be used for preparation and planning of future activities, while deficiencies were referred to as area for improvement for further activities during unit 6 outage in 2003.

2.4. Licensing

The licensing process, implemented for units 5 and 6, meets the requirements for the three-step approach, defined by the Nuclear regulatory agency (NRA) of Bulgaria in Regulation No 5. Request for set up permits of contracted operations were submitted to NRA on behalf of the main Contractors, and the contractors organizations were presented, as well as their quality assurance programs, and the lists of potential suppliers and subcontractors, which have ISO 9000 standards qualification and certificates – first step.

After the main Contractors were authorized to start work, the second step in the regulatory process was made. It is with regard to the permit to start performance of the tasks - designing and supplies, followed by presentation of Technical specifications on measures and the related substantiation documents. The third step of the regulatory process was successfully made for the measures performed in 2002 and 2003, as engineering solution are presented and in some cases, the draft designs accepted by Kozloduy NPP.

Cooperation between the Nuclear regulatory agency and Kozloduy NPP is open and unobstructed, and when there is a request, the clarifications are submitted. NRA questions are answered promptly and NRA experts are satisfied.

An important activity is envisaged, in cases when parts of safety critical measures are to be included in the Safety Analysis Report (SAR). In such cases, intensive coordination and communication is expected.

2.5. Technical review of compliance

The technical review of compliance of Contractors’ supplies and documentation is extremely important for units 5 and 6 Modernization Program. The established technology is based on trilateral review by different organizations and subdivisions – project participants, i.e. Modernization and Investment Division, production subdivisions of the plant (EP 2 – Electricity Production -2), and the Consultant (Parsons E&C). The results from the review are summarized and presented at Technical council meetings, where balanced decisions are made by the Chairman of the meeting in terms of the supply acceptance, or the possible shipment back to the Contractor for further improvement.

This technology allowed Kozloduy NPP to establish a firm concept on reviewing, based on compliance of documentation with the contracted technical specifications, international and national standards and norms, quality, safety, earthquake engineering, and environment protection requirements, and the established positive practice in engineering and construction.
On behalf of Kozloduy NPP, an active approach was adopted for prompt notification of the Contractor on the reviews results, and acceleration of revised documents submitting. Internal terms for reviewing of Contractor’s supplies in all cases were in compliance with the acceptable terms according to the contracts, and even shorter, when people’s engagements allowed them to do it.

Contractors benefited from the solid engineering base, established by Kozloduy NPP, supported by Parsons, and the notes and opinions on submitted documents contributed significantly for achieving the desired compliance and improvement supplies quality. These notes and opinions were essential for providing satisfactory completion of Measures, or their good realizability.

2.6. Communication and correspondence

Units 5 and 6 Modernization program is characterized by intensive communication between the Project participants, and relevantly with an extremely large flow of correspondence. Kozloduy NPP plays a leading management role. The organization and performance of the project makes the plant the centre of sending and receiving correspondence related to the project. A procedure was established to regulate this flow. It is based on a standard system of enumeration, allowing easy classification, archiving and documents retrieval.

2.7. Training

Units 5 and 6 Modernization Program imposed the corresponding requirements to training and qualification enhancement, referring to Plant personnel training for operation and maintenance of installed equipment and systems. With regard to this, the technical specifications of the new highly technological equipment including personnel training and preparation are a mandatory requirement to achieve full compliance of the scope against the expected one. In these cases manufacturers of original equipment on their utilities’ sites provide training and specialization. The main contractors establish organization, while Kozloduy NPP agrees questionnaires and training programs before the training is held.

2.8. Decision making

The decision making technique, applied in units 5 and 6 Modernization Program, is based on multiparty sequence of components:

- Knowledge of Russian nuclear reactors VVER, of Western pressurized water reactors, Russian standards and norms and those of IAEA, NRC, Methods and technologies of engineering, designing, constructions and assembling operations, and commissioning;
- Information about the best technologies and solutions for modernization of VVER reactors plants, about the results of operation of other reconstructed/ modernized plants, about specific research on occurred events and accidents, about cost-effective modernization methods, used in energy (nuclear and conventional), for manufacturer’s possibilities;
- Strategy on the program development, on the complete strategic plans of Kozloduy NPP for electricity generation and interrelation with the national electricity grid, for contracts applying, for technical, economic efficiency and benefit and for personnel qualification enhancement;
- Expert knowledge of stage by stage implementation of the project in nuclear and conventional energy, as a conceptual project, draft design, specification and orders of equipment, production, construction and installation, testing, commissioning, operation, and maintenance. Expert knowledge of project and quality management, technical, economical and operational planning, administrative management of contracts, costs supervision and assessment.
3. Conclusion

Engaging the potential of the best specialists in European and American nuclear field, taking into account the latest requirements of the normative documents of the Main Designer and of IAEA, we are convinced, that upon completion of Modernization program, Kozloduy NPP units will definitely be some of the safest and most reliable in the world, which lifetime can be extended by 15-20 years beyond the design lifetime.
CURRENT ISSUES IN SPENT NUCLEAR FUEL MANAGEMENT AFTER THE NUCLEAR POWER PROGRAMMES TERMINATION

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Abstract. Nuclear power programmes have been abandoned in the Republic of Belarus after Chernobil accident. Test mobile nuclear power plant (MNPP) “Pamir-630D” with 630 kW was shut-down. The nuclear power plant consisted of a reactor cooled by gas, a gas turbine set and three additional units, all items to be put on transport platforms. When the decision was taken to shut down and decommission the MNPP it was foreseen that the spent fuel would be shipped to Russia. Unfortunately, due to the break-up of the Soviet Union, the spent fuel was left in the territory of the Republic of Belarus. The reactor and turbine parts, including their biological shielding, were disposed of without dismantling, as entire units, in appropriate concrete vaults. The non-radioactive equipment and parts of “Pamir-630D” were converted for new purposes. New hazardous facilities: a sub-critical assembly, an electron accelerator, a γ-ray facility are set in the building which was used for “Pamir-630D” operation. One of two operational fuel storage pools of MNPP, built near the reactor hall, has been used as temporary spent nuclear fuel storage. Since then the spent fuel has been stored in water the quality of which is maintained in accordance with appropriate chemistry requirements. Since Belarus does not have substantial nuclear programme, there is neither appropriate dry storage facility no reprocessing plant for spent fuel. Belarus also lacks financial resources for long-time storage or spent fuel reprocessing. Moreover, even the reprocessing technology for this kind of nuclear fuel has not been developed mostly due to the fact that this sort of fuel was designed only for “Pamir-630D” and was not used in any other type of reactor.

1. Introduction

This paper addresses current issues in safe management of spent nuclear fuel from the mobile nuclear power plant “Pamir-630D” with 630 kW after the nuclear power programmes were terminated in the Republic of Belarus. Two pilot prototypes of “Pamir-630D” were constructed. The first one was put into operation in 1985 and was in operation for 3668 hours. The second one was not put into operation at all. The mobile nuclear power plant “Pamir-630D” was shut down 26.11.1987 and up to 24.06.1988 the reactor was cooled by gas-liquid coolant circulation in the main loop. After that time the reactor was cooled by liquid coolant circulation through auxiliary loop of accident cooling system of the reactor. The coolant was removed from all circuits 22.05.1989 and the reactor was filled with nitrogen and cooled by it until the fuel was removed.

The nuclear power plant consisted of a gas cooled reactor, a gas turbine set, two control units, and an auxiliary unit. The reactor (weight of 76 500 kg) and turbine set (76 000 kg) were supposed to be put on transport platforms and carried by tractors. The control (20 000 kg) and auxiliary (20 000 kg) units were set on track beds. “Pamir-630D” was constructed and tested in appropriate building. Set-up time was no greater than six hours after all units of the MNPP had reached the site. “Pamir-630D” was ready to be moved to the other site in 30 hours after the shut down. Service lifetime of “Pamir-630D” was 10 years: 7 years of storage and 3 years of operation. Operational lifetime was no less than 10 000 hours (non-stop operational period was no longer than 2000 hours). Dose rate at the boundary of restrictive area was no more than 65 μSv/h at the time of reactor operation. The dose rate was no greater than 3 mSv/h on side surface of biological shielding and 10 mSv/h on end surface of biological shielding of the reactor, 24 hours after the shut down.
V.M. Paliukhovich and A.A. Tukhto

The reactor unit included: a vessel reactor cooled by gas “nitrin” based on N$_2$O$_4$, a biological shielding and pipelines with valves. The reactor vessel and other auxiliary equipment were made of stainless steel. The reactor core consisted of 106 fuel assemblies, each of them contained 7 fuel rods surrounded by stainless steel claddings of wall thickness $4\times10^{-3}$ m and diameter of $6.2\times10^{-3}$ m. Fuel spherical particles of UO$_2$ enriched to 45% $^{235}$U were embedded in Ni-Cr matrix. The share of nickel and chrome in fuel composition was 40%. Weight of $^{235}$U in reactor core was 18.7 kg; weight of the reactor core - 5700 kg.

The turbine set consisted of a turbine, an electrical generator and auxiliary equipment to maintain coolant quality in accordance with appropriate chemistry requirements. All parts of the turbine unit were made of stainless steel.

2. Decommissioning

The decommissioning plan included a short period of preparation for disposal followed by reactor and turbine units of “Pamir-630D” dismantling, a safe long-term keeping of spent fuel in wet storage facility and reprocessing. All procedures of decommissioning were thoroughly prepared to protect the personnel against radiation and to meet the nuclear safety requirements. Special equipment included a tank for temporary keeping of radioactive pieces of reactor, a turntable of biological shielding and a turntable with devices and a tank (height of 5.3 m, a diameter of 2 m) for the removing of the fuel assemblies, which were designed and constructed to unload the reactor core under water. All personnel involved in decommissioning activities were made familiar with “Pamir-630D” site and safety procedures for the safe and effective conduct of their duties.

The following measures contributed to the safety assurance programme during nuclear fuel unloading, transportation and storage: proper organization of all activities, observance of the safety requirements, use of protective system, use of transport containers and fuel storage of special design and radiation monitoring. Decontamination procedures were applied to detect contamination over limits.

The non-radioactive parts of the reactor unit were recycled or used for other purposes. Dismantled radioactive pipelines were blanked off and welded. The reactor vessel with biological shielding was disposed, as an entire unit, without dismantling in concrete vault 6m long by 3m wide by 3m high.

Others materials, equipment and parts of the MNPP, with significant activity despite of the decontamination, were removed for disposal.

3. Storage of spent fuel

The storage facility consists of two pools (volume of $2\times28$ m$^3$) with a water shielding (thickness of 3.1m) and a concrete shielding (thickness of 1.8m). One of two operational storage pools of MNPP, built near the reactor hall, has been used as temporary storage of spent nuclear fuel. Since then the spent fuel assemblies (average burn-up – 1.16 %) have been kept under water. Surveillance, monitoring and inspections are carried out to ensure that the spent fuel storage remains in good condition. The water quality is maintained in accordance with appropriate chemistry requirements (pH=5.9-6.0, conductivity – 2-2.4 cm/m, temperature – 4-16 C). Frequency of monitoring: temperature – daily, pH and conductivity – weekly. The flasks with the fuel assemblies are tested for leakage by the check weighting under water quarterly if its weight is stable and weekly in case of the weight changes. The flask is removed for testing and correcting the leakage in hot cell in accordance with appropriate procedure if the reference weight increasing is 0.1kg. Support plates on the bottom of the pools were presumably designed to space the fuel (in their storage cassettes) far enough apart to avoid criticality safety issues. The pool lids are currently sealed by the IAEA inspectors.
4. Spent nuclear fuel management after the nuclear power programmes termination

When a decision was taken to shut down and decommission the MNPP it was foreseen that the spent fuel would be shipped to Russia. Unfortunately, due to the break-up of the Soviet Union, the spent fuel was left in the territory of the Republic of Belarus.

Most countries with nuclear power programmes have active programmes under way to develop the technology for the management of spent nuclear fuel. At the moment there are three major options for classifying spent fuel management policies and practices: a closed fuel cycle with reprocessing of spent nuclear fuel, a ‘once through’ fuel cycle, which ends with the disposal of the spent nuclear fuel, and a ‘wait and see’ approach. Since Belarus does not have substantial nuclear programmes, there is neither appropriate dry storage facility, no reprocessing plant for spent fuel. Belarus also lacks financial resources for long-time storage or reprocessing spent fuel. Moreover, even the reprocessing technology for this kind of nuclear fuel has not been developed mostly due to the fact that this sort of fuel was designed only for “Pamir-630D” and was not used in any other type of reactor. At the same time, there is no reason to develop such technology, design and construct the reprocessing plant for 43 kg of spent fuel. For Belarus having only small nuclear programmes and spent fuel after Soviet Union nuclear activity to dispose of, the costs of sitting and developing a geological repository are enormous. Moreover, Belarus has not suitable geological formation to host a geological repository. So, there is one way - a ‘wait and see’. Negative aspects of the ‘wait and see’ are known. This option could be perceived as being indecisive, avoiding decisions, or passing an issue on to future generations.

In addition, it will be soon necessary to move the spent fuel from the temporary wet storage facility to a dry storage facility given the fact that the flasks with fuel assemblies which originally were not designed and constructed for the purposes of long-time storage can lose tightness any time.

Besides, under the Convention on the Physical Protection of Nuclear Material this fuel has become Category I material due to the decrease in the radiation level. It requires application of strengthened physical protection measures.

The best solution of this problem is to return the spent fuel from “Pamir-630D” to a reprocessing plant of the Russian Federation as it had been planned before decommissioning of MNPP in 1987. A positive decision on its return to the Russian Federation will contribute to strengthening the physical protection as well as reducing its maintenance expenditures.

The Fact-Finding Mission visited Belarus in March 2004. The mission was organized in the framework of the IAEA Technical Co-operation project RER/9/058 “Safety Review of Research Reactor Facilities” that was established in 1998 to assist mainly Member States operating Russian-origin research reactors in Central and eastern Europe and the NIS with the decommissioning plans and the return of the fuel to the country of origin. The mission task was to make a preliminary assessment of the physical, technical, administrative and financial resources that would be required to ship the spent fuel from Belarus to Russia.

5. Regulatory issues in the current environment

As a Contracting Party to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, Belarus has a national policy for the management of spent fuel, in order to ensure that acceptable levels of protection of human health and environment, now and in the future, can be adequately achieved without composing undue burdens on the future generations. After the nuclear power programmes termination the regulatory authorities are faced with the changing environment. The national legislative and regulatory system for safety of spent nuclear management in Belarus deals with follow types of challenge:

a. The scientific and technical challenge. To ensure that all reasonable events, features and processes relevant to the long term safety of storage facility for spent nuclear fuel are duly
identified, modeled and taken into account in the safety assessment. The pools were built for MNPP operation and are not suitable for long term keeping of spent fuel. So, the ageing management programmes have to be elaborated as soon as possible. We are also faced with using spent fuel facility that was not designed and constructed in accordance with the current knowledge and experience base and design criteria.

b. The staff ageing challenge. To ensure that persons involved in spent fuel management have appropriate level of qualification, knowledge and experience. Since decommission of MNPP the storage facility staff has consisted of operative personnel of nuclear reactor. Number of workers is being decreased and, moreover, now most of them have near pension age. Nonetheless, no one young scientist or engineer has joined spent nuclear fuel facility. May be in ten years we will have the situation when no one will be familiar with storage facility and spent nuclear fuel management. Countries with only small nuclear programmes or after the nuclear power termination may face this challenge in future.

c. The financial challenge. To ensure that no undue financial burdens are imposed on future generation. Unfortunately, decommission fund was not established and budget financial resources for long-time storage of spent nuclear fuel is not enough. Now regulatory body has no possibility to change this situation.

6. Summary

The nuclear power program in Belarus was abandoned due to the break-up of the Soviet Union. The spent fuel, as result of the Soviet Union nuclear activity, was left in the territory of Belarus. Surveillance, monitoring and inspections are carried out to ensure the spent fuel storage remains in good condition. The reprocessing technology for such kind of nuclear fuel has not been developed. Taking into account the fact that this particular sort of fuel has been designed only for “Pamir-630D” and has not been used in any other types of reactor, technology of its reprocessing perhaps will not be elaborated in the nearest future, so it is necessary to provide long-term safe keeping of spent fuel. However, the lack of financial resources for long-time storage or reprocessing spent fuel, ageing the equipment of storage facility will not allow assuring safe storage of spent fuel in the future. There is neither appropriate dry storage facility no reprocessing plant for spent fuel in Belarus, so the ‘wait and see’ option means passing an issue on to future generations. For Belarus, having only a small nuclear programme, the solution of this problem is to return the spent fuel after Soviet Union nuclear activity to the country of origin with the IAEA assistance.

The Republic of Belarus would highly appreciate IAEA assistance in facilitating the return of the spent nuclear fuel to the Russian Federation within the framework of the Trilateral Initiative of the Agency, the Russian Federation and the United States of America.
PROBLEM OF NUCLEAR POWER

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1. Introduction

For the Mongolian Central Electrical Supply (CES) system electricity demand projections are summarised in the table 1.

Table 1: Electricity Demand Projections for CES System

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Total Demand (GWh)</td>
<td>3311</td>
<td>4219</td>
<td>5505</td>
<td>7260</td>
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<tr>
<td>Max. Demand (MW)</td>
<td>581</td>
<td>732</td>
<td>942</td>
<td>1230</td>
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</table>

Average growth rates: total demand 5.3 % per year
peak demand 5.0 % per year

Electricity Demand Projections for Aimag centres show

average growth rates: for total generation

(5.3 %- 6 %) per year

Heat demand projections for major cities in the CES system (Table 2)

Average growth rates for heat energy and peak demand are about 4 % per year. For the aimag centres the average growth rate is slightly higher at around 5 % per year.
Table 2- Heat demand projections for major cities in the CES system

<table>
<thead>
<tr>
<th></th>
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<tr>
<td><strong>Ulaanbaatar</strong></td>
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<tr>
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<td></td>
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<td></td>
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<td>Total Demand (GWh)</td>
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<td>1598</td>
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<td>Max. Demand (MW)</td>
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<td>256</td>
<td>314</td>
<td>396</td>
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<td><strong>Erdenet</strong></td>
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<tr>
<td>Total Demand (GWh)</td>
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<td>Max. Demand (MW)</td>
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<td>437</td>
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<td>Total Demand(GWh)</td>
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<td>545</td>
<td>710</td>
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<td>Max. Demand (MW)</td>
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<td>107.9</td>
<td>136.2</td>
<td>175.5</td>
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</table>

2. **Power Demand in Mongolia**

1. Electricity demand in the CES system is expected to grow at an average rate of 4.5% per annum.

2. Heat demand in major cities covered by the CES system is expected to grow at an average rate of 4% per annum.

3. Electricity and Heat demands in Aimag centres is expected to grow at an average rate of 5.5% per annum.

In 2001 there were 440 Nuclear Power Plant (NPP)-s operating in 31 countries.

1. A significant part of power generation in the world is produced by nuclear power.

2. Nuclear power is well developed and proven.

3. It is economical and competitive to other forms of power generation in a range of situations.

4. Nuclear power generation is environmentally benign.
Table 3- The comparisons of different power plants

<table>
<thead>
<tr>
<th>Kind of energy sources</th>
<th>Power, MW</th>
<th>Capital investment US$/kW</th>
<th>Cost of electricity, US c/kWh</th>
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<td>20000-30000</td>
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<td>Wind</td>
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<td>300</td>
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<td></td>
<td>500</td>
<td>1800-3000</td>
<td>4.1-6.0</td>
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3. Private Nuclear Power in the World Today

- The first commercial nuclear power stations started operation in the 1950s.
- There are now some 440 commercial nuclear reactors in 31 countries, with over 350,000 MWe of total capacity.
- They supply 16% of the world's electricity, as base-load power, and their efficiency is increasing.
- 59 countries operate a total of 273 research reactors.
- Canada is the world's leading supplier of uranium.
Table 4

Private World Nuclear Power Reactors 1999-2001
and Uranium Requirements

<table>
<thead>
<tr>
<th>COUNTRY</th>
<th>NUCLEAR ELECTRICITY GENERATION 2000</th>
<th>REACTORS OPERATING June 2001</th>
<th>REACTORS CONSTRUCTION June 2001</th>
<th>ON ORDER or PLANNED June 2001</th>
<th>URANIUM REQUIRED 2000</th>
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<td>437</td>
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4. World Energy Needs and Nuclear Power

- Nuclear power provides over 16% of the world’s electricity, almost 24% of electricity in OECD countries, and 35% in the EU. Its use is increasing.

- Without nuclear power most of the world would have to rely almost entirely on fossil fuels for base-load electricity production.

- Nuclear power is the most environmentally benign way of producing electricity on a large scale. If nuclear were replaced by coal-fired generation, carbon dioxide emissions would increase by over two billion tonnes per year.

- Renewable energy sources other than hydro have high generating costs but are suitable for small intermittent-load electricity demand.
5. Plans for New Reactors Worldwide

- Nuclear power capacity worldwide is increasing steadily but not dramatically, with over 30 reactors under construction in 11 countries.
- Most reactors on order or planned are in the Asian region.
- Significant further capacity is being created by plant upgrading.
- Plant life extension programs are decreasing the need for new capacity.

6. Economics of Nuclear Power

- Nuclear power is cost competitive with other forms of electricity generation in the OECD, except in regions where there is direct access to low-cost fossil fuels.
- The decreasing cost of fossil fuels in the past decade has eroded nuclear power's previous cost advantage in many OECD countries.
- Fuel costs for nuclear plants are a minor proportion of total generating costs and often about one-third those for coal-fired plants.
- In assessing the cost competitiveness of nuclear energy, decommissioning and waste disposal costs are taken into account.
7. **Future Cost Competitiveness**

Some comparative electricity generating cost projections for year 2005-2010

<table>
<thead>
<tr>
<th></th>
<th>nuclear</th>
<th>Coal</th>
<th>gas</th>
</tr>
</thead>
<tbody>
<tr>
<td>France</td>
<td>3.22</td>
<td>4.64</td>
<td>4.74</td>
</tr>
<tr>
<td>Russia</td>
<td>2.69</td>
<td>4.63</td>
<td>3.54</td>
</tr>
<tr>
<td>Japan</td>
<td>5.75</td>
<td>5.58</td>
<td>7.91</td>
</tr>
<tr>
<td>Korea</td>
<td>3.07</td>
<td>3.44</td>
<td>4.25</td>
</tr>
<tr>
<td>Spain</td>
<td>4.10</td>
<td>4.22</td>
<td>4.79</td>
</tr>
<tr>
<td>USA</td>
<td>3.33</td>
<td>2.48</td>
<td>2.33-2.71</td>
</tr>
<tr>
<td>Canada</td>
<td>2.47-2.96</td>
<td>2.92</td>
<td>3.00</td>
</tr>
<tr>
<td>China</td>
<td>2.54-3.08</td>
<td>3.18</td>
<td>-</td>
</tr>
</tbody>
</table>

**REFERENCES**

[5] Internet materials
JAPANESE CONTRIBUTIONS TO ASIAN NUCLEAR SAFETY NETWORK (ANSN)

F. Kudough

International Affairs Group, Japan Nuclear Energy Safety Organization (JNES), Tokyo, Japan

Abstract. The ANSN operations in the IAEA’s Extrabudgetary Programme (EBP) Phase II are briefly overviewed and Japanese contributions to the ANSN are discussed. The Japanese role in the realization of autonomous, self-sustaining and user-friendly mechanism is profound.

1. Introduction

The IAEA’s Extrabudgetary Programme on the Safety of Nuclear Installations in the South East Asia, Pacific and Far East Countries started in 1997. The Recipient countries are China, Indonesia, Malaysia, Philippines, Thailand and Vietnam and Donor countries are Japan, France, Germany, the Republic of Korea, Spain and the United States. During the Phase I (1997-2003), a total of 155 national and regional activities were implemented including the Asian Nuclear Safety Network (ANSN)---an initiative to enhance regional nuclear safety awareness. The ANSN is a regional network to share the nuclear safety information and to provide a common database on nuclear safety education and training materials. The pilot project was prepared in 2002 and implemented in 2003.

This paper describes the current status and future directions of the ANSN and Japanese contributions to the ANSN.

2. Current Status and Future Directions of ANSN

2.1. Concept of ANSN

The ANSN is composed of three Hubs (China, Japan and Korea) and national centers (Indonesia, Malaysia, Philippines, Thailand and Vietnam; sometimes referred to as non-hub countries) with the Master Index Database integrated by the IAEA as shown in Fig. 1. The Hubs basically provide education and training materials on nuclear safety at each of the web sites and national centers provide indigenous training course textbooks in their own languages as well as IAEA’s and Hub's textbooks translated in their own languages.
2.2. Current Status

The full-scale implementation of ANSN started in 2004 in synchronizing with the start of the EBP Phase II. In the EBP Phase II, the transfer of IAEA management to regional autonomous, self-sustaining and user-friendly operations of the ANSN is strongly encouraged. As part of the movement, the ANSN is managed by the Steering Committee with Chairperson designated from Japan. Under the framework of the ANSN, four topical groups -- Education and Training, Safety Analysis, Operational Safety for NPPs and Safety Culture -- and one IT support group (ITSG) were established to discuss specialized issues. Japan coordinates the Education and Training Group, the Republic of Korea coordinates the Safety Analysis and Safety Culture Groups and China coordinates the Operational Safety for NPPs Group.

ANSN knowledge base and knowledge management activities and other EBP’s activities such as national and regional activities are continued for the time being. The current EBP structure is shown in Fig. 2.

2.3. Japanese Contributions to ANSN and Future Directions

In order to encourage and promote the widespread use of the ANSN, Japan is ready to assist non-hub countries in establishing national centers. As part of this effort, Japan will take part in Caravans in the non-hub countries. Objectives of the Caravan are to:

- demonstrate ANSN functions directly to each of the staff members of the nuclear regulatory organizations in non-hub countries;
Fig. 2. Overview of EBP Activities

- disseminate widely the ANSN in their countries; and
- know the usability of the ANSN and discuss further improvement and user-friendliness.

As mentioned earlier, Japan will coordinate the Education and Training Topical Group. In this group, the way to effectively utilize EBP education and training treasures gained by several workshops and seminars will be discussed and implemented. The group will also discuss training course construction in each of the non-hub countries and provide textbook catalogues to be used in the indigenous training courses.

3. Conclusion

IAEA’s ANSN activities and the Japanese contributions to the ANSN have been overviewed.

The realization of autonomous, self-sustaining and user-friendly operations of the ANSN is strongly dependent on the Hub countries’ eagerness and non-hub countries’ awareness and active participation in ensuring nuclear safety. The Japanese Hub is committed to promoting the ANSN with strong leadership.
USED NUCLEAR FUEL TREATMENT IN THE 21ST CENTURY

M.-F. Debreuille, J.-G. Devezeaux De Lavergne, P. Kaplan, S. NG, R. Vinoche

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Abstract. This paper addresses the role that used nuclear fuel treatment will play with the advent of the next generation (Generation IV) of nuclear reactors. Given the wide range and hence uncertainty regarding the exact composition and size of the future global nuclear fleets, used fuel treatment positions itself with a pivotal role in that it allows a large flexibility with regard to the main parameters to be optimised in the framework of sustainable development: economics, safety, and environmental protection. In addition, the great majority of the Generation IV reactors relies on the closed fuel cycle, with great benefits in the fields of resource conservation, waste conditioning performance, and waste toxicity management. Advanced versions of aqueous-based treatment plants (such as at COGEMA's La Hague site) are likely to occupy the majority of the fourth generation of these plants. Those based on pyroprocesses are not yet industrially developed, but could play a part as a niche technology, for instance, coupled with molten salt reactors. This paper presents the scope of future treatment technologies as envisioned today, together with their necessary evolutions linked to the requirements of Generation IV systems, and details the benefits they provide to the issue of geological repositories.

1. Introduction

The nuclear industry faces today a period of great opportunity. Long-awaited research projects on the next generation of nuclear reactors may finally be industrially realised with countries such as China, France, and the United States on the brink of building, maintaining, or growing their respective nuclear fleets. The challenge will be to nurture this now "global" growth of a vast range of nuclear reactors of different technologies, while respecting the constraints of sustainable development. Used fuel treatment technologies are particularly implicated in this challenge. The choice to treat used fuel allows a significant flexibility in an environment where decisions often follow a stop-and-go process, and where the range of possible future paths, including reactor technologies, is wide. COGEMA has over thirty years of experience in managing the internal and external safety aspects of its used fuel treatment plants at La Hague. Today, faced with the wide range of prospects for the future global nuclear fleet, COGEMA is actively researching – in collaboration with the French Atomic Energy Commission (CEA) – the treatment technologies that will be best adapted to the next generations of reactors and subject to our high safety and quality standards. This paper represents a snapshot of this research, and a panorama on used fuel treatment for tomorrow's reactors.

2. A brief history of used fuel treatment plants

Before discussing the evolution of the back-end of the fuel cycle for the Generation IV reactors, it will be useful to recall briefly the steps that have led to the treatment plants in operation today. These plants may be classified based on the successive generations of power plants they serve, and are summarised in Table 1. The basis of the PUREX process – that used today at COGEMA's La Hague reprocessing plant – was established in the 1950s with the first generation of nuclear plants designed for defence purposes. Industrial treatment, beginning with the second generation of plants, began in France soon after in the 1960s with the construction of the UP2 plant at La Hague. This was designed to process the used fuel discharged from commercial electricity-generating reactors, the majority of which were natural uranium gas-graphite plants. With the advent of light water reactors in many nuclear fleets around the world in the 1980s-1990s, the third generation of treatment plants were...
designed and built to treat used fuel solely from these reactors. The most important plants of that
generation are the UP2-800 and UP3 plants at La Hague with a total capacity of 1700 MTHM per
year. The plant at Rokkasho-Mura, which is likely to enter operation in 2006, also belongs to this
generation, being based essentially on the UP3 plant. However, it can be further qualified as
"Generation III+", due to the integration of technical progress accumulated over ten years of UP3
operation, specifically adapted to meet Japanese requirements.

<table>
<thead>
<tr>
<th>Used fuel treatment plants</th>
<th>Time period</th>
<th>Nuclear reactors</th>
</tr>
</thead>
<tbody>
<tr>
<td>Generation I</td>
<td>Hanford-Savannah</td>
<td>1950s</td>
</tr>
<tr>
<td>Generation II</td>
<td>UP1, UP2 (La Hague), Tokai (Japan), RT1 (Russia)</td>
<td>1960s – 1980s</td>
</tr>
<tr>
<td>Generation III</td>
<td>UP2-800 / UP3 (La Hague), Sellafield (UK), Rokkasho-Mura (Japan) – III+</td>
<td>1990s – today</td>
</tr>
</tbody>
</table>

Table 1: The three generations of used fuel treatment plants with their corresponding nuclear reactors

COGEMA has continuously worked on technical improvements at the La Hague site, while
maintaining its excellent safety record in terms of waste quality, exposure to workers, and radioactive
releases. It should be noted that the strong cooperation between researchers, designers, engineers, and
operators has played a key role in this successful track record. Significant progress has been made in
the optimisation of the process (e.g. extraction in two cycles instead of three) and in the optimisation
of waste management (e.g. volume reduction of high-level waste by a factor of six). In particular, two
new facilities at La Hague entered commercial service in 2002: a plutonium finishing facility (R4) and
a hulls and end fittings compaction facility (ACC). These represent a first step at COGEMA towards a
new generation of treatment plants.

3. Prospects for Generation IV treatment plants

This leads us to the question: what characteristics will the treatment plants of the future require? In a
deviation from previous decades, the notion of the Generation IV reactors integrates wholly the choice
of fuel cycle. Indeed, one refers to Generation IV nuclear energy systems rather than the separate
elements. The technology goals for such a system have already been identified in the Generation IV
International Forum as: sustainability (effective fuel utilisation, safe and cost-effective waste-
management, proliferation resistance), safety and reliability, and economic competitiveness.
Extrapolating from the above, we can infer for a future treatment plant the following technical factors
that should influence its design:

- Characteristics of the fuel to be treated;
- Choice of waste management and degree of separation of elements;
- Type of treatment technology;
- Type of fuel cycle, for example, if recycling or transmutation is envisaged.

As can be seen from Table 2, the Generation IV reactors that will in all likelihood form the future
nuclear fleet at the horizon 2030-2040 differ widely in the type of fuel used. Moreover, the current
fleet of light water reactors still has several decades of operation to come, and thus any considerations
for a future used fuel treatment plant must also take into account the fuel from these reactors.

Clearly, the type of treatment should be adapted to each type of fuel, and among these we can
distinguish two main treatment technologies: aqueous-based processes and pyroprocesses. The
technology used in today's used fuel treatment plants, and notably at La Hague, is an aqueous-based chemical extraction process (such as PUREX). As we have seen in the previous section, this technology has been well developed over several decades and further progress is envisaged in future plants. In contrast, pyroprocess – a non-aqueous, high-temperature, pyrochemical process – is to date a significantly less mature technology. However, it will have an important role to play as it is much better adapted than an aqueous-based process to treating fuels such as ceramics and liquid metals. In what follows, we will attempt to sketch, based on ongoing research, the prospects for both types of treatment technology in the framework of Generation IV systems.

<table>
<thead>
<tr>
<th>Reactor type</th>
<th>Fuel type</th>
<th>Treatment type</th>
<th>Treatment location</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gas-cooled fast reactor (GFR)</td>
<td>Compound ceramics, advanced particles, or elements containing actinides and ceramic cladding</td>
<td>Advanced aqueous or pyroprocess</td>
<td>Reactor site</td>
</tr>
<tr>
<td>Lead-cooled fast reactor (LFR)</td>
<td>Metal or nitride</td>
<td>Advanced aqueous or pyroprocess</td>
<td>To be chosen</td>
</tr>
<tr>
<td>Molten salt cooled reactor (MSR)</td>
<td>Mix of sodium zirconium and uranium fluorides</td>
<td>Process based on fluoride volatility</td>
<td>To be chosen</td>
</tr>
<tr>
<td>Sodium-cooled fast reactor (SFR)</td>
<td>Metallic alloy U-Pu-MA-Zr MOX fuel</td>
<td>Pyroprocess</td>
<td>Reactor site</td>
</tr>
<tr>
<td>Supercritical water-cooled reactor (SCWR)</td>
<td>Oxide fuel</td>
<td>Advanced aqueous (if fast spectrum)</td>
<td>To be chosen</td>
</tr>
<tr>
<td>Very high-temperature gas-cooled reactor (VHTR)</td>
<td>Prismatic fuel (open cycle) Nitride-graphite particles (closed cycle)</td>
<td>Advanced aqueous or pyroprocess (if second option)</td>
<td>To be chosen</td>
</tr>
</tbody>
</table>

Table 2: Generation IV reactors and their associated fuels and back-ends

As far as aqueous-based processes are concerned, we can already envision today the development and innovation necessary to meet the Generation IV demands:

- Different forms of nuclear fuel rods may necessitate the implementation of novel mechanical processes at the head-end of the plant, while different fuel compositions will require new chemical processes, for example, to deal with carbides and nitrides;

- Progressive integration of all aspects of the fuel cycle on the same site will continue in order to optimise material transfers, for example, from the treatment plant to a storage location. In addition, constraints linked to dismantling of nuclear installations will be integrated into plant designs;

- Optimisation of the treatment process itself will continue, with maximal recycling of reagents and effluents, prolongation of equipment lifetime, reduction of plant size, and minimisation of high radioactivity zones within the plant;

- Advanced processes such as the partitioning of minor actinides and long-lived fission products described in the Advanced Fuel Cycle Initiative (US) could be implemented industrially and evolve according to specific needs;
M.-F. Debreuille et al.

- Waste management will be adapted to the process chosen (for example, regarding which elements should be conditioned together), while improvements in conditioning techniques, and volume and toxicity reduction will continue, taking into account final repository requirements.

One important consequence of the predicted improvements above is the minimisation of waste impact on geological repositories. Treatment of used fuel can reduce by more than 50% the area required for a geological repository as compared to direct disposal, and the associated costs have been established to be reduced by the same factor. Optimisation and continual evolutions of treatment processes can only serve to further improve these already substantial figures.

As for pyroprocesses, given its comparatively immature level of development, much of the progress in this technology remains to be done at a laboratory stage. Consequently, it is likely to remain in the position of a niche technology in comparison to advanced aqueous processes. Nevertheless, it is essential that intensive research and development continues in this technology in conjunction with the Generation IV reactors to which it is adapted, as it is clear that it has a possible role to play in future nuclear systems.

4. Discussion and Conclusions

In this paper we have presented our vision – based on the current state of industry and research – of treatment plants that will accompany the next generation of nuclear reactors. The Generation IV programme will allow the nuclear industry to finally cover the whole of the fuel cycle – from reactor to exploitation to dismantlement, and from fuel fabrication to used fuel treatment to waste disposal. Although the Generation IV used fuel treatment plants are unlikely to enter into operation before 2030, it is vital that we begin to contemplate the possibilities today, particularly since the next generation of nuclear reactors integrate so completely in their design, aspects of the rest of the fuel cycle. COGEMA, with its extensive experience and together with the other subsidiaries of AREVA, is in the strong position of encompassing the totality of the fuel cycle in its portfolio. In addition, we have seen that used fuel treatment will be an influential ingredient in future nuclear systems, not only because it provides a strong capacity for flexibility and adaptation to different reactors and specific needs of countries, but equally importantly because of the benefits it imparts regarding the Generation IV technology goals, namely, sustainability, safety and reliability, and economic competitiveness.

Specifically, concerning sustainability, used fuel treatment in association with fast breeder reactors will allow the conservation of more than 90% of natural uranium resources, and enables a substantial reduction in toxicity and volume of waste to be disposed in a geological repository. Economically, COGEMA aims to achieve up to a 50% reduction in the back-end costs of the fuel cycle. This target may appear challenging, but it is attainable due to numerous optimisations forecasted in the design and operation costs, while of course maintaining COGEMA's proven high safety standards.

REFERENCES

RANGE OF PROTECTIVE ACTIONS FOR NUCLEAR POWER PLANT INCIDENTS - A CASE STUDY

A.P. Nelson
Nuclear Energy Institute (NEI), Washington, DC, United States of America

Abstract. The range of protective actions that would be used to protect the public during a nuclear power plant incident has been based on a strategy of evacuation and sheltering since emergency plan development nearly 25 years ago. This paper identifies a number of issues concerning evacuation, sheltering and heighten awareness and provides industry implementation recommendations.

NEI Protective Action Recommendations April 2004
The views in this paper represent NEI and not the US govt or USNRC

1. Introduction:
To detail the range of early phase protective actions that may be used for nuclear power plant incidents.

2. Discussion:
2.1. History
The range of protective actions that would be used to protect the public during a nuclear power plant incident has been based on a strategy of evacuation and sheltering since emergency plan development nearly 25 years ago. This paper will not attempt to recount past strategies or their associated bases, but will examine the protective actions detailed in current guidance.

2.2. Current Guidance
10 CFR 50.47(b)(10) Ref. [1] contains the requirement for a licensees emergency plan to contain a range of protective actions. Guidance to implement a range of protective actions was revised in the mid 1990’s in response to the issuance of NUREG 0654 Supplement 3 Ref. [2], EPA 400 Ref. [3], and in 2001 to accommodate a change to 10 CFR 50.47 Ref. [1].

Each of the subject guidance documents contains the same basics concepts of evacuation and sheltering as protective actions. However, sufficient ambiguity exists within the guidance to have resulted in divergent implementation of protective action schemes within the industry. Specifically, the indications for, and implementation of, each protective action differs among licensees. The remainder of this section examines the features of each guidance document.

2.2.1. Environmental Protection Agency 400
EPA 400 retained the concepts of evacuation and sheltering as protective actions from previous guidance. EPA 400 revised the Protective Action Guidelines (PAG) Ref. [3 Table 2-1] and provided a basis for those guidelines Ref. [3 Appendices B and C]. That document did not effectively link its
revised guidance to nuclear power plant conditions, such as emergency action levels or emergency classification levels, nor did it provide specific guidance on how to use the diverse implementation concepts it contained. In the absence of such guidance, many nuclear power plant licensees, in consultation with offsite officials, provided their own interpretation of when and how the PAG’s would be utilized. This resulted in multiple different implementation schemes being implemented by licensees. In addition, dose and dose rate terminology used in EPA 400 differed from that used in a companion revision to 10 CFR 20 (Standards for Protection Against Radiation).

2.2.2. UREG 0654 Supplement 3

This document was issued two years after EPA 400 and was intended to simplify and clarify previously issued guidance. This guidance references the dose-based protective action concepts in EPA 400, but relies primarily on plant conditions as an indication for protective actions. NUREG 0654 is aligned with EPA 400 with respect to sheltering, recommending it as an alternative to evacuation for short term releases or when impediments to evacuation exist.

2.2.3. 10 CFR 50.47(b)(10):

This citation was amended in 2001 to include the consideration for the use of thyroid prophylaxis. It required states to formally consider the inclusion of potassium iodine (KI) as a thyroid blocking agent and incorporate it into their emergency plans as appropriate. Given this, KI would only be included in the licensees range of protective actions if the affected State(s) decided to include it.

2.2.4. Summary of requirements and guidance

Table 1 provides a summary of the guidance, including indications and implementation.

<table>
<thead>
<tr>
<th></th>
<th>Evacuation</th>
<th>Sheltering</th>
<th>KI</th>
<th>Heightened Awareness *</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>10 CFR 50.47</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>When to implement</td>
<td>Not provided</td>
<td>Not provided</td>
<td>Dependent on state/local decision to use</td>
<td>Not mentioned</td>
</tr>
<tr>
<td>How to implement</td>
<td>Not provided</td>
<td>Not provided</td>
<td>Not provided</td>
<td>Not mentioned</td>
</tr>
<tr>
<td><strong>EPA 400</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>When to implement</td>
<td>• Table 2-1: dose based</td>
<td>• Preferred when it will provide protection equal to or greater than evacuation</td>
<td>Projected thyroid dose &gt; 25 rem</td>
<td>Not mentioned</td>
</tr>
<tr>
<td></td>
<td>• Evacuate general population at dose of 1 rem or &gt;</td>
<td>• Consider implementing at doses &lt; 1 rem</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>• Special populations may be evacuated at higher doses</td>
<td>• Consider when doses &gt; 1 rem but can’t evacuate due to impediments</td>
<td></td>
<td></td>
</tr>
<tr>
<td>NUREG 0654 Supp3</td>
<td>Evacuation</td>
<td>Sheltering</td>
<td>KI</td>
<td>Heightened Awareness *</td>
</tr>
<tr>
<td>------------------</td>
<td>------------</td>
<td>------------</td>
<td>----</td>
<td>----------------------</td>
</tr>
<tr>
<td><strong>How to implement</strong></td>
<td>Not provided</td>
<td>• Provides multiple actions to limit infiltration of outside air into structure</td>
<td>Not provided</td>
<td>Not mentioned</td>
</tr>
<tr>
<td><strong>When to implement</strong></td>
<td>• Actual or projected severe core damage or loss of control of facility</td>
<td>• When conditions exist that make evacuation dangerous</td>
<td>Not provided</td>
<td>Recommend to EPZ populations that have not been advised to evacuate</td>
</tr>
<tr>
<td></td>
<td>• Consider EPA PAG’s in modifying initial protective actions</td>
<td>• For short term (puff) releases for populations near the plant</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Transit dependent persons awaiting transportation</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>How to implement</strong></td>
<td>Not provided</td>
<td>Not provided</td>
<td>Not provided</td>
<td>Advise population to go indoors and listen to EAS</td>
</tr>
</tbody>
</table>

*Not considered a protective action, but included for completeness

2.3. **Industry issues**

2.3.1. **Evacuation**

**Issue 1: Evacuation triggers**

*Industry recommendation:*

Evacuate 2 miles around and 5 miles downwind upon declaration of a General Emergency. Subsequent evacuations will be based on the EPA PAG’s or changes in wind direction. Ref. [2, 3, and 5]

2.3.2. **Sheltering**

**Issue 2: Use of sheltering as an alternative to evacuation for short term releases**

*Industry recommendation:*

A licensee may choose to not integrate the use of sheltering for short term releases into their protective action scheme. Ref. [2 and 3]

**Issue 3: Use of sheltering for special populations and impediments**

*Industry recommendation:*

Licensees should incorporate sheltering into their emergency plans consistent with existing guidance, noting that the use of sheltering as an alternative to evacuation is a decision that will be made by
offsite officials. Implementation of the licensee emergency plan commitment should incorporate allowance for offsite officials to utilize sheltering as an alternative to evacuation at their discretion, in accordance with the guidance. These plans should be developed and maintained in collaboration with those offsite officials. Thus, licensees will typically recommend evacuation as dictated by the guidance, but will incorporate the proviso that the use of sheltering as an alternative is a local or state decision, and is acceptable. Ref. [2 and 3]

**Issue 4: Effectiveness of sheltering**

*Industry recommendation:*

Licensee or offsite officials may opt to utilize a range of sheltering implementation schemes, including:

- The use of qualitative methods for determining the effectiveness of sheltering. Example, if certain plant or radiological conditions exist, then shelter, OR
- The use of quantitative methods for determining the effectiveness of sheltering. Example, the comparison of sheltering versus evacuation doses.
- Utilization of simple public instructions. Example: stay indoors and limit outside sources of air, OR
- Utilization of more complex public instructions. Example: in addition to the above simple instructions, recommend going into a basement or more substantial building, use of respiratory protection. Ref. [3]

2.3.3. *Heightened Awareness*

**Issue 5: Use of Heightened Awareness**

*Industry recommendation:*

Licensees should incorporate the use of heightened awareness in their protective actions schemes consistent with the guidance. Ref. [2]

3. **Conclusion**

The requirement to have a range of protective actions is contained in 10 CFR 50.47(b)(10), EPA 400 and NUREG 0654 serve as guidance for implementation of the requirement. From this, the range of protective actions that shall be included in each licensees emergency plan are:

- Evacuation
- Sheltering
- KI

The protective action scheme should make use of heightened awareness in order to maximize the efficacy of evacuation.
REFERENCES

[1] 10 CFR 50.47(b)(10): A range of protective actions including sheltering, evacuation and Prophylactic use of iodine have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during and emergency, consistent with Federal guidance, are developed and in place and protective actions for ingestion pathway EPZ appropriate to the locale have been developed (66 FR 5440, Jan 19, 2001)


TOPICAL ISSUE 2:
OPERATING EXPERIENCE – MANAGING CHANGES EFFECTIVELY
IMPROVING INDUSTRY EVENT TRENDING TO AVOID SIGNIFICANT EVENTS

P.J. Di Rito

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Abstract. Recent significant nuclear power plant events have reinforced the need to more proactively identify adverse event trends that could put the industry at risk in the future. These recent events called into question the adequacy of decision-making processes, contractor control, and nuclear safety environments at the stations involved. While the industry is taking action to address these issues, lack of detailed minor or non-consequential event information made it difficult to predict these adverse trends. Low consequence event information needs to be promptly and thoroughly reported and analyzed by the industry to improve the identification of undesirable issues. The international coordination and consolidation of industry event information can be a daunting task. In this regard, WANO Atlanta Center activities over the last several years to improve the timely and accurate sharing of member event information may be of interest to the rest of the industry. The evolution of these activities is provided to encourage further collaboration and improvement in identifying adverse trends and avoiding significant industry events.

1. Introduction

The mission of WANO is to maximise the safety and reliability of nuclear power plants by exchanging information and encouraging communication, comparison and emulation amongst its members. In carrying out this mission, WANO strives to screen and analyze events that occur at nuclear power plants worldwide to identify possible precursors of more serious events and disseminate the lessons learned.

2. Background

On April 10, 2003, Paks Nuclear Power Plant experienced severe damage to 30 irradiated fuel assemblies undergoing a cleaning process outside the reactor vessel. The event was rated as an International Nuclear Event Scale (INES) level 3 event. Although there were many contributors to this event, one of the more significant aspects was the failure of management to create an adequate nuclear safety culture at the station. Adverse aspects of this culture included nuclear standards not being applied to the cleaning vessel process, decision-making being production focused, and accountability for the cleaning system being inappropriately transferred to contract personnel. While certain aspects of the Paks event appear to be isolated conditions that do not exist at other stations, WANO has identified other recent significant industry events with similar faulty operational decisions. Shortfalls in operational decision making were also identified by WANO in significant events that occurred at Brunsbuettel, Phillipsburg, and Sizewell B. In response to this event trend, WANO launched an initiative in 2004 to further help plant and utility managers improve operational decisions. WANO has also developed industry documents that analyze problems with contractor control and improving station safety culture based on industry performance concerns.

The industry should not have to experience significant events to identify adverse trends and take action. There are normally many related minor or non-consequential events that occur prior to a significant event that could be used to identify adverse trends and can be predictors of future significant events. To identify these trends, less important industry events must be universally shared.
in a timely manner to provide sufficient and relevant data to analyze. To identify industry organizational issues like operational decision-making, individual station investigations also need to analyze beyond the surface or direct causes of important events.

3. WANO-Atlanta Center Activities

WANO-Atlanta Center (WANO-AC) has focused over the last several years on both improving the level and timeliness of event reporting and improving the quality of event reports shared by our members. At a November 2000 WANO-AC Governing Board Meeting, the WANO-AC governors requested an action plan to address concerns with member timeliness and level of event reporting. In subsequent discussions with WANO-AC members, barriers were identified that prevented timely event reporting. These barriers included event reporting threshold uncertainty, difficulty in translating and formatting event reports, and lack of management support for event reporting. In addressing these concerns, issues were also identified in station root cause analysis and in analyzing industry data.

3.1. Reporting Threshold

WANO-AC members were requested to share with the industry those events that warranted the performance of a root cause investigation at their station. Although frequently members expressed the desire for more specific criteria than this generic guidance, it was found that continuous discussion with station personnel was the best way of clarifying what events should be reported. Often, the best guidance was asking, “If this event had happened at another station, would I have wanted to know about it?” In addition, protocols were established between WANO-AC and major WANO-AC members, such as the Institute of Nuclear Power Operations (INPO) and the CANDU Operating Group (COG), to exchange current event information and event trends. These protocols established weekly meetings to discuss event information and encourage event sharing with the industry when appropriate. Other WANO regional centers are also starting to establish similar protocols with their larger members.

3.2. Translating and Formatting Event Reports

Although several WANO-AC members indicated translating event documents into English or formatting station documents into existing WANO report templates was a barrier to reporting, WANO-AC assistance in translating and formatting did not dramatically increase the number of event reports that were shared. What frequently emerged from subsequent discussions with WANO-AC members was a lack of adequate station priority in performing these reporting activities or allowing an excessive amount of time to perform station event investigations. WANO-AC took specific action with the stations involved to address these issues.

3.3. Management Support for Event Reporting

While station management was asked to place a higher priority on the sharing of operating experience with other WANO members, this action by itself did not immediately improve event reporting performance. In October 2001, WANO-AC began to mail quarterly operating experience performance indicator information to member executive points of contacts. Graphs of all WANO-AC member performance in event reporting level and timeliness were provided as well as WANO-AC analysis of the associated member’s individual performance. WANO-AC also established an event report timeliness goal of 50 days from the date of an event until the event is reported to WANO. Through comparison, emulation, and on occasion targeted follow-up, WANO-AC was able to facilitate member event reporting improvement. The number of events reported to WANO has steadily increased, and the timeliness goal of 50 days is now being achieved. The attached graphs indicate the relative improvement in WANO-AC event reporting level and timeliness over the last three years.
3.4. Root Cause Investigations

In posting member event information, WANO-AC identified that the underlying organizational causes or contributors for important station events were frequently not adequately investigated or identified. WANO-AC subsequently began a series of root cause investigation courses at member stations to help station personnel use root cause investigation methodology to better determine how individuals, leaders, and the organization often contributed to important station events. WANO found that station event investigations were normally technically strong, but lacked depth in investigating human performance issues. On occasion, assistance was also requested and provided to help improve station corrective action program weaknesses. A direct correlation was often found between the timeliness of correction action program investigations and reporting to the industry. The station’s corrective action process was reviewed relative to industry best practices, and action plans were developed with the station to schedule improvement activities.

3.5. Activities in Development

With event reporting improving, WANO-AC is now looking at improved methods of analyzing existing data. Improved application of WANO event codes to all reported events is being pursued to provide a more consistent method of sorting existing data. Other tools such as data text mining and concept mapping are also being tested for their ability to provide unique perspectives for event information. Even with these tools, human analysis is still required to establish appropriate perspective and relevance for issues identified. WANO is also evaluating improved methods of communicating the results of the trends identified from minor or non-consequential events or trends identified by our larger members to the industry.

4. Conclusions

This year marks the 25th anniversary of the Three Mile Island accident, another significant event that should have been prevented by proper use of operating experience. Unfortunately, recent significant events continue to reinforce the importance of sharing and using industry event information. In particular, industry event trends that could result in future significant events need to be identified through effective sharing and analysis of industry event information. Experience has shown that station management needs to keep plant personnel focused on properly investigating, learning from, and sharing important event information. Only with improved sharing of event information and trends can industry groups such as WANO identify adverse industry issues that need to be addressed. In this regard, nuclear utilities are encouraged to continue sharing operating experience information with WANO and the industry to further promote nuclear power plant safety and reliability.
Level of Reporting Performance Indicator

![Graph showing the Level of Reporting Performance Indicator.]

Timeliness of Reporting Performance Indicator

![Graph showing the Timeliness of Reporting Performance Indicator.]

38
ANALYSIS OF FAILURE RATES IN AN OPERATING VARIABLE ENERGY CYCLOTRON

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Abstract. Failure rate data necessary for system reliability analysis of accelerators are scanty. The need for such analysis exists because of the increasing use of accelerators in various applications. In this paper we present the data for system failure in the Variable Energy Cyclotron at Kolkata, India, for the past twenty-five years. We identify the components responsible for frequent failures and the variation in failure rates over these years.

1. Introduction

The largest cause of radiation exposure to accelerator workers arises from maintenance of radioactivated components, handling and moving of activated items, Radiation surveys and radioactive waste handling. Accelerators that are capable of liberating hadrons or high energy photons will induce radioactivity in accelerator components and concrete walls. With higher intensity of the accelerated beam the amount of activation increases. Particularly a high beam loss situation leads to higher radioactivation of the components. This leads to higher dose to workers during maintenance.

The accelerator may be considered as a maintained system. Reliability of an accelerator depends both on its inherent reliability (design specific) and the operating plant reliability (dependent on operating conditions). Quantification of the probabilities involved in the reliability estimation of an accelerator is a difficult challenge. To have an idea of the operating plant reliability it is essential to carry out a detailed analysis of the available failure data in the accelerator. We have done this for the Variable Energy Cyclotron (VECC) at Kolkata, India, operating for the last twenty-five years. Our intention is to augment the scantily available failure rate data of accelerators presently available worldwide. A brief description of the accelerator is given below.

2. Description of the accelerator

The Variable Energy Cyclotron consists of the following components:

- Main magnet of pole diameter 224cm with an average field of 17.1 Kgauss main coil power of 450kW.
- Ion sources of two types
  a. Hot cathode PIG
  b. 6.4 GHz ECR
- RF system with frequency range 5.5 to 16.5 MHz with oscillator power 300kW.
- Dee system 180 degree with maximum Dee voltage of 70kV.
- High current and high voltage power supplies.
- Deflector electrostatic type with 120kV maximum voltage.
- Vacuum systems maintaining a working pressure of $5 \times 10^{-6}$ torr.
The cyclotron is capable of delivering alpha beams in the energy range of 25-130 MeV, proton beams in the range of 6-60 MeV and heavy ions of maximum energy $130 \frac{Q^2}{A}$ (Q: charge state, A: atomic mass). There are beam transport systems consisting of beam lines, two switching magnets, one analyzing magnet and five quadrupole magnets.

Failure of any of the first five components will lead to the failure of the machine. The deflector and the beam transport system deliver the accelerated particles on the targets. Maximum radioactivation takes place in the deflector followed by the targets, dee and the beam transport systems.

3. Maintenance related dose

Dose received by radiation workers depend on both planned and unplanned dose.

(1) Planned dose: Planned maintenance activities would come under this category. The field would depend on the system being maintained. Here the field is known and the time the worker spends in the field is planned in advance. The dose depends on the job required to be done.

(2) Unplanned dose:- This would depend on the unplanned events in the accelerator. Reducing the number of breakdown failures can also minimize unplanned exposure.

This paper gives the results of system failure and maintenance data analysis based on the permission issued by health physicist to the maintenance people.

4. Data analysis:

Radiation dose received by workers would depend on residual activity and/or contamination where the work is being carried out. There is a "Work Permit" form which is filled up before any maintenance activity is carried out. The health physics unit, VECC, monitors the actual radiation at the time and place of work which is recorded in the work permit. The number of people who actually enter the radiation area and the duration of their stay are also recorded.

The work permit data collected during operation of VECC are being analyzed as part of an effort to understand the operating accelerator reliability.

A plot of the number of failures per year in the accelerator, versus the year of operation is shown in figure 1. Its very clear that initially the number of failures was high and it gradually reduced till the fifth year of operation. In the fifth year there were some major problems. After repair and restarting of the accelerator, the failure frequency reduced and the accelerator operated successfully with minimum failures for seven years. Presently there appears to be an increase in failure again which must be due to the aging effect in some systems.

Figure 1. is similar to a bath tub curve where it appears that presently ageing has set in. Further analysis was carried out system wise. It was found that the RF system dominated the failures in initial years. The RF failure shows a decreasing trend over time. Figure 2. shows the trend.
Figure 3. shows an increasing failure rate for deflector system with time. VECC is now undergoing a modernization program which when completed should again result in reduced unplanned shutdowns.
5. Conclusion

Detailed Data collection of the operating accelerator and statistical inference and analysis of the data to evaluate operating accelerator reliability and identify major causes of failure. Information on the risk and what failures contribute most to the risk are of great value in helping to determine the acceptability of a facility's overall design and operation. A Reliability centred maintenance approach using preventive and predictive maintenance practices can help in minimizing unplanned failures, thus reducing the maintenance dose received. Additionally, reliability estimates of accelerators is becoming a subject of prime concern for proposed developments of Accelerator Driven Subcritical Systems (ADSS) and for industrial and medical applications. Referring mainly to ADSS, a high reliability is necessary to avoid thermal shocks on structures following the accelerator trip transient with sudden coolant temperature decrease and consequent thermal stress on structures and components. Further data analysis is being carried out to estimate quantitatively the radiation dose received by workers due to maintenance and system availability. Also, continued research and methodology development are necessary for future accelerator system design satisfying requirements of availability and safety.

REFERENCES

Abstract. In 1999 the Government of the Republic of Kazakhstan adopted the Decree on the decommissioning of the BN-350 reactor - a fast breeder reactor using liquid sodium as a coolant. KAEC (as a regulatory body) was authorized to provide the coordination of BN-350 decommissioning work. As the decision on the reactor decommissioning was adopted before the end of scheduled operation (2003) the Decommissioning Plan had not yet been developed. For implementation of the Government Decree and for determination of activities by the reactor safety provision and for preparation of its decommission for the period till the Decommissioning Project will be ready “The Plan of priority measures for BN-350 reactor decommissioning” was developed. By now a considerable amount of works concerning to the transitional period from an operating to a decommissioning status were performed in the frame of this Plan. This paper describes the current situation related to BN-350 decommissioning activities which are performed in the frame of the chosen decommissioning strategy and licensing requirements.

1. Introduction

Experimental-industrial BN-350 reactor facility – fast neutron sodium-cooled reactor [1] – is located near Aktau city in the part of the eastern Caspian Sea shore belonging to the Republic of Kazakhstan. It was commissioned in 1973 for electricity generation and seawater desalination for the Aktau region. Planned thermal power of the reactor is 1000 MW, electrical power is 350 MW.

In April 1999, taking into consideration financial and technical problems of further reactor BN-350 operation, in view of plant conclusion of an unsecuring acceptable safety level during reactor operation and IAEA OSART mission recommendations [2] the Government of the Republic of Kazakhstan adopted the Decree № 456 “On decommission of reactor BN-350 in Aktau city Mangistau region”. According to this Decree, Kazakhstan Government intends to place the BN-350 facility into a safe storage (SAFSTOR) condition for a period of up to 50 years with a view to complete decommissioning and disposal thereafter.

The SAFSTOR option means implementation of the following conditions (criteria):

— the fuel is unloaded from the reactor core, packaged, and placed in long-term storage.
— the liquid metal coolant is removed, processed, and the radioactive products of the processing are placed in long-term storage.
— radioactive waste is processed and placed in long-term storage.
— radiation monitoring in and around the SAFSTOR area and in on-site temporary waste storage areas is provided.
— equipment and structures are either dismantled, mothballed, reinforced, or modified in accordance with the decommissioning plan.
— a documentation package for the first stage is prepared, approved, and appropriately archived.
At the end of the SAFSTOR period, the nuclear facility shall be decommissioned to a condition allowing potential reuse of the site for industrial purposes.

As the decision on the reactor decommissioning was adopted before the end of scheduled operation (2003) the Decommissioning Plan (which in Kazakhstan is called “The Project of BN-350 Reactor Plant Decommissioning”) had not yet been developed. Furthermore, the situation was complicated by Kazakhstan not having been in possession of its own storage facilities for spent fuel and radioactive wastes of the BN-350 reactor, adapted technologies on treatment of radioactive wastes to the forms acceptable for its disposal. An especially urgent question was the management of the sodium of the primary circuit – its decontamination, draining from reactor and conditioning. In addition the appropriate regulations needed for efficient managing of the works related to decommissioning were not ready.

2. Management of first stage decommissioning works

2.1. General organizations - participants

According to the Decree of the Government of the Republic of Kazakhstan the Ministry of Energy and Mineral Resources of the Republic of Kazakhstan (MEMR RK) was charged with conducting organizational and controlling activities under the project of safe decommissioning of the reactor, BN-350.

Kazakhstan Atomic Energy Committee (KAEC) is a part of the Ministry of Energy and Mineral Resources and as the nuclear regulator in Kazakhstan is responsible for supervision of nuclear and radiation safety, and the nuclear weapon non-proliferation regime. According to the Ministry order, KAEC was authorized to provide coordination of BN-350 decommissioning work fulfilment with periodical reporting to the Ministry.

Pursuant to the Ministry order, Joint Stock Company “KATEP” (KATEP) is authorized to represent the state interests in the BN-350 Decommissioning Project. KATEP, being a general customer defines the responsible subcontractors for each stage of the work, as well as their functions and responsibilities and provides reports to the Ministry. Pursuant to the given order, KATEP has developed "The scheme of management the activity on BN-350 decommissioning", that is the basis defining both the organizational structure and procedure for interaction of BN-350 decommissioning organizations-participants. The block-scheme of management and the activity on BN-350 decommissioning is shown in Fig. 1.

According to this scheme the main participants of the project are as follows:

— General Contractor - Mangyshlak Atomic Energy Complex (MAEC, currently MAEC-Kazatomprom Ltd), BN-350 site, Aktau City, Republic of Kazakhstan;
— General Designer - the state unitary institution, Russian Design and Research Institute of Complex Power-Engineering Technology (VNIPIET), St-Petersburg, Russian Federation;
— Scientific manager - the state research center of Russian Federation Physics-Power-Engineering Institute (GNC RF FEI), Russian Federation, Obninsk City.

The other main contractors (heading institutions) of BN-350 decommissioning project are as follows:

— Nuclear Technology Safety Center (NTSC), Almaty, Republic of Kazakhstan;
— National Nuclear Center of the Republic of Kazakhstan (NNC RK), Kurchatov City, Republic of Kazakhstan;
— Argonne National Laboratory, USA
Fig. 1. The Scheme of management the activity on BN-350 decommissioning

2.2. Main functions of project participants

MEMR RK:

— organization, control over conduct of works and financing of activity under the project of safe decommissioning of the reactor BN-350;
— agreement and approval of main project decisions;
— coordination of activities of involved ministries and agencies, enterprises and organizations of the Republic of Kazakhstan;
— coordination of activities with foreign project participants.
KAEC:

— engagement of only state licensed institutions for Project development and implementation;
— supervision of Project implementation by licensing of activities related to atomic energy use in accordance with the legal procedure and control of license regulations observation within its competency;
— supervision of standard and legal acts in the field of atomic energy use by individuals and legal entities independently on their department subordination and type of ownership;
— consideration and approval of materials validating nuclear objects and facilities safety at all stages of decommissioning;
— state registration of nuclear materials and supervision of physical safety while their storing, transportation and use;
— consideration, approval and authorization of technical documentation of enterprises and institutions implementing the Project within its competency;
— inspection and control of any activities under the Project related to atomic energy use in accordance with legal procedures with the right of free access to atomic energy using objects and to documentation characterizing their activities, as well as measuring, sampling, probing and appropriate instruments and equipment installation for implementation of state supervision;
— annual reporting to the Government of the Republic of Kazakhstan about status of works under the Project;
— on-line control of safety situation on atomic energy using objects of the Project;
— control of emergency response readiness while implementation of Project works;
— organization of training, skills mastering and examination of the personnel engaged in Project implementation;
— preparation of on-line messages about malfunctions of supervised objects for specialized international institutions and supervising authorities of other countries.

KATEP:

— determines organizations, participating in BN-350 decommissioning process;
— distinguishes duties and responsibilities between all participants of BN-350 decommissioning process;
— concludes contracts (agreements) with contractors;
— develops draft budget for decommissioning works;
— organizes analysis and expertise (agreement) of the results of work.

MAEC:

— fulfils the whole scope of works concerned with decommissioning of BN-350 (in accordance with contractor's agreements);
— participates in development and agreement of project documentation concerned with individual projects and Decommissioning Project as a whole;
— operates existing and newly installed facilities and systems.

Contractors (subcontractors) - consultancy, technical expertise, engineering-design, design, building, erecting, setting organizations as well as the equipment manufacturers participating in BN-350 decommissioning activity are responsible for:
quality of work (supply, services) fulfilled;
— observance of work (supply, services) schedule in accordance with the agreement commitments;
— safety of the work under fulfillment.

### 2.3. Organization of first stage decommissioning works

The Decommissioning Project has not been produced in Kazakhstan before, and even the process of nuclear decommissioning (and its licensing) is new to both the customer organization and to the government as a whole. Taking into account the BN-350 uniqueness and high potential hazard as well as the technological complicity of the forthcoming work on reactor plant decommissioning, it was judged necessary to produce a specification for the technical specification itself. It was necessary to establish the responsibilities and inter-relations of the main parties, and to set out the subsequent stages in the regulatory and approval process. This process has resulted in two “Special technical requirements” (STR) documents:

— Special technical requirements “General Provisions on BN-350 Reactor Plant Decommissioning Project Development”;
— Special technical requirements on development the BN-350 Decommissioning Project.

Special technical requirements “General Provisions on BN-350 Reactor Plant Decommissioning Project Development” cover the development and implementation of the BN-350 reactor plant Decommissioning Project (hereinafter referred to as Project) and are its integral part. The STR’s define and establish the project stages and the main requirements for technical tasks, Decommissioning Project, as well as define the procedure for coordination and approval of project documentation and basic laws and standards, to which the project should correspond. Observing the main provisions and meeting the requirements of the STR is mandatory for all legal and physical entities, which take part in project development and implementation.

The STR “General provisions on BN-350 Reactor Plant Decommissioning Project Development” defines two main stages of project development:

— development of Decommissioning Project;
— development of project documentation.

Special technical requirements for project development define the BN-350 Decommissioning Project development procedure, establish the project stages, as well as the project documentation composition and contents and procedure for coordination and approval.

For activities connected with disposal of spent fuel of BN-350 for long term storage was developed the same document – Special Technical Specifications for Design of Cask Dry Storage Facilities and Transfer Facility for the BN-350 Reactor Spent Nuclear Fuel.

To determine the activities required for ensuring reactor safety and in preparation for decommissioning, the Ministry of Energy and Mineral Resources of the Republic of Kazakhstan developed and approved, a “Plan of Priority Measures on BN-350 Reactor Decommissioning”. This Plan has the status of managerial and ruling document and defines the activity on the provision for safety of BN-350 and preparation for decommissioning within the period until the “Project of BN-350 Decommissioning” is approved. Actions provided for the Plan (fourth edition) include the following:

— measures on BN-350 Decommissioning Project development;
— measures on provision the reactor safety within the transitional period;
— measures on sodium drainage and utilization;
— measures on spent fuel disposal for long term storage.
3. Principal activities of first stage decommissioning works

From 1999 until present, the activities have been intensively conducted according to the “Plan of Priority Measures on BN-350 Reactor Decommissioning”.

The main task of the stage of setting BN-350 reactor into safe long-term storage status in the field of spent nuclear fuel (SNF) management is removal of SNF from the reactor site.

By now the following activities have been completed:

— for the purpose of preparation of SNF for transportation and storage, and assurance of favorable conditions for temporary storage of fuel in the reactor cooling ponds before transportation, nuclear fuel was packed in sealed casings filled with inert gas;

— according to a decision made by the Government of RK, the complex “Baikal-1” located on the site of National Nuclear Center (Kurchatov city) in Eastern-Kazakhstan region was selected as the place for long-term storage of SNF of BN-350 reactor. SNF will be transported there in transport packages.

— Special Technical Specifications for Design of Cask Dry Storage Facilities have been developed as well as Transfer Facility for the BN-350 Reactor Spent Nuclear Fuel;

— Technical Task on Design Development “Transportation Package for BN-350 SNF” has been approved.

The second main task is removal of liquid metal coolant from heat-exchange circuits and equipment. This coolant will be brought into explosion-and fire-safe condition and placing of products of processing for long-term storage.

In order to decrease radiation activity of sodium in the primary circuit and, as a result, to decrease dose burden on personnel at processing of sodium and handling with products of processing, it was planned to carry out purification of sodium of the primary circuit from radionuclides of cesium.

By now the following activities have been completed:

— Purification of sodium using cesium traps on the basis RVC installed in the primary circuit have been completed;

— Drainage of bulk sodium of the primary circuit has been completed;

— KAEC approved the Technical Task on “BN-350 reactor sodium processing facility design”.

The next main task is the processing (conditioning) and storage of radioactive waste accumulated in the process of operation and formed during the decommissioning process.

Technical requirements to the project of the technological installation of the processing of the liquid radioactive waste has been approved. An ion-selective purification method will be used (with the purpose of reducing the volume of radioactive wastes) with further cementing of radioactive deposits and placing of cement compound in concrete containers.

To attract the international society to the fulfillment of activities on the decommissioning of the BN-350, reactor the decision was made to develop “The Decommissioning Plan of the Transfer the BN-350 Reactor into a status of Long-Term Safe Enclosure”. The development of this Plan is being conducted under financing from the US State Department of with the technical support from USA National Laboratories, EC and other international organizations (including the Russian ones).

REFERENCES


DETECTION OF DAMAGED FUEL ASSEMBLY IN LVR-15 REACTOR WITH SPECTROMETRIC WATER ACTIVITY MEASUREMENT

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Abstract. The LVR-15 reactor is a light water research type reactor, which is situated in Nuclear Research Institute, Rez near Prague. At present IRT-2M fuel of Russian production with enrichment of 36% is used. In the reactor core there are usually from 28 to 32 fuel elements with the total mass about 5 kg of $^{235}$U. The reactor is cooled by demineralized water. The maximum thermal power is 10 MW and the reactor is operated in 21-days irradiation cycles, with 8 to 10 cycles per year.

The reactor is used as a multipurpose facility and its main use is in the following areas: material research carried out at reactor loops and rigs, production of radiation doped silicon, production of radioisotopes for the radio pharmaceuticals and technical radiation sources, irradiation devices for special irradiation, pneumatic rabbit for activation analysis, development of boron neutron capture therapy at the epithermal neutron beam and neutron physics research (e.g. neutron diffraction for different purposes) at reactor horizontal channels.

One of the most important aspects for safe operation of the reactor is the early detection of a damaged fuel assembly in reactor core or in the storage pool. An indication of a damaged fuel assembly can be made e.g. from gas effluents. This paper deals with the methods used, based on the water activity measurement.

Four types of water samples are used for this purpose:

(1) Primary circuit water
(2) Storage pool water
(3) Water from sipping test of fuel assemblies from the core
(4) Water from sipping test of fuel assemblies from storage pools

From measured volume activities of selected radionuclides and stated limits of volume activities it can be determined if the tested fuel assemblies are undamaged. The measurement of primary circuit water and storage pool water is performed regularly (weekly and monthly, respectively). A sipping test of fuel assemblies from the core is made when there is suspicion of a damaged fuel assembly in the core and a sipping test of a fuel assembly from the storage pool is performed before the transport of the assembly from the reactor building.
2. Methods

2.1. Water sampling procedure

Originally the Marinelli beakers were used for the activity measurement. Since 1998 the same 0.5 l PET bottle has been used both for taking of the sample and measurement (no trapping of radioisotopes on the walls of the bottle used for taking of the sample and on the walls of the Marinelli beaker, minimum handling with the radioactive water).

Primary circuit water and storage pool water are taken in defined points with the valve.

The sipping tests are performed in the storage pool with the sipping test assembly (see 3 in the Fig. 1). The fuel assembly is put into the sipping test assembly. The sipping test assembly is closed with a cap. The sipping test assembly is filled up with demineralized water. These conditions (without input of the water) are kept for 3 hours, than the water sample (4) is taken.

FIG. 1. Schema of sipping test

2.2. Activity measurement

A spectrometric assembly (Canberra) with an HPGe detector with relative efficiency of 18 % and FWHM=1.8 keV for energy of 1332 keV is used for measurements made according to Error! Reference source not found.. The detector is placed in a shielding box with 5 cm thick lead walls. For calibration special radionuclide standards of 0.5 l bottle has been used. The minimum detectable volume activity is 10 Bq/l for $^{137}$Cs and measurement time of 3600 s, which is sufficient for the measurements. The analysis is made for the library of about 100 radionuclides but only subset is used for evaluation (26 radionuclides for primary circuit water and storage pool water measurement, 3 radionuclides for sipping tests).

2.3. Volume activity operational limits

Operational limits of volume activity $a_{\text{VLim}}$ are defined for each of type of water measurement.

1. Primary circuit water: $a_{\text{VLim1}}=10^5$ Bq/l for $^{131}$I and $a_{\text{VLim2}}=10^6$ Bq/l for sum of fission products.
2. Storage pool water: $a_{\text{VLim}}=300$ Bq/l for sum of $^{134}$Cs, $^{137}$Cs and $^{144}$Ce.
3. Sipping test of fuel assemblies from the core: no quantitative criterion, total activity for damaged fuel assembly should be greater by 2 or 3 orders compare with undamaged ones.
3. Results of the measurement

3.1. Primary circuit water measurement

Because of the great number of measurements made since 1996 only typical examples are presented. List of radionuclides used for evaluation of primary circuit water of LVR-15 reactor and example of volume activities are given in Table 1.

Table 1. Example of volume activities $a_V$ for primary circuit water sample taken on 2. 4. 2004

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Half life (days)</th>
<th>$a_V$ (Bq/l)</th>
<th>Radionuclide</th>
<th>Half life (days)</th>
<th>$a_V$ (Bq/l)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{95}$Zr</td>
<td>64.03</td>
<td>1.26E+03</td>
<td>$^{24}$Na</td>
<td>0.623</td>
<td>1.74E+06</td>
</tr>
<tr>
<td>$^{95}$Nb</td>
<td>35.15</td>
<td>1.52E+03</td>
<td>$^{46}$Sc</td>
<td>83.9</td>
<td>1.41E+04</td>
</tr>
<tr>
<td>$^{99}$Mo</td>
<td>2.78</td>
<td>2.55E+03</td>
<td>$^{51}$Cr</td>
<td>27.7</td>
<td>2.40E+05</td>
</tr>
<tr>
<td>$^{103}$Ru</td>
<td>39.5</td>
<td>5.56E+02</td>
<td>$^{54}$Mn</td>
<td>303</td>
<td>1.49E+03</td>
</tr>
<tr>
<td>$^{106}$Ru</td>
<td>368</td>
<td>&lt;2.00E+02</td>
<td>$^{60}$Co</td>
<td>1921</td>
<td>1.08E+04</td>
</tr>
<tr>
<td>$^{131}$I</td>
<td>8.05</td>
<td>4.70E+03</td>
<td>$^{65}$Zn</td>
<td>243.9</td>
<td>5.21E+03</td>
</tr>
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<td>$^{132}$Te</td>
<td>3.24</td>
<td>3.40E+02</td>
<td>$^{110m}$Ag</td>
<td>255</td>
<td>1.86E+02</td>
</tr>
<tr>
<td>$^{133}$Xe</td>
<td>5.3</td>
<td>6.00E+03</td>
<td>$^{105}$Rh</td>
<td>1.475</td>
<td>&lt;2.00E+02</td>
</tr>
<tr>
<td>$^{134}$Cs</td>
<td>752</td>
<td>&lt;2.00E+02</td>
<td>$^{122}$Sb</td>
<td>2.8</td>
<td>1.13E+04</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>11000</td>
<td>4.58E+02</td>
<td>$^{124}$Sb</td>
<td>60.2</td>
<td>8.72E+03</td>
</tr>
<tr>
<td>$^{140}$Ba</td>
<td>12.746</td>
<td>6.97E+03</td>
<td>$^{239}$Np</td>
<td>2.36</td>
<td>2.84E+03</td>
</tr>
<tr>
<td>$^{140}$La</td>
<td>1.678</td>
<td>9.36E+03</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{141}$Ce</td>
<td>32.5</td>
<td>7.66E+03</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{143}$Ce</td>
<td>1.375</td>
<td>2.37E+03</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{144}$Ce</td>
<td>284.3</td>
<td>3.70E+03</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

In Fig. 2 the time dependence of volume activity for $^{131}$I and $^{137}$Cs is presented. The increased values in 1996 correspond to presence of a damaged fuel assembly in the core (see clause 3.3). The missing values in 2002 are related to flood near the reactor building when values of activity were under detection limit because of longer interruption of the reactor operation.

Increased values of fission product activities can indicate damaged fuel assembly but fission products are produced also out of the fuel due to contamination of fuel cladding and presence of natural uranium in the demineralized water. Actinide radionuclides (e.g. $^{239}$Pu, $^{241}$Am) are more specific for damaged fuel assembly, their activity can be measured with alpha spectrometry of an evaporated water sample.
3.2. Storage pool water measurement

In the storage pool water usually only activity of $^{137}$Cs is above the detection limit and values are between 50 Bq/l and 200 Bq/l.

3.3. Sipping test of fuel assemblies from the core

In 1996 the sipping test was performed on all the fuel assemblies from the core. One assembly was identified as damaged assembly, the total volume activity was $5 \times 10^5$ Bq/l, and the main radionuclide was $^{137}$Cs. For undamaged fuel assemblies the total volume activity was usually less then $5 \times 10^2$ Bq/l.

3.4. Sipping test of fuel assemblies from storage pool

Usually only activity of $^{137}$Cs is above the detection limit; values are between 50 Bq/l and 200 Bq/l.

4. Conclusion

Gamma spectrometric measurements of water activity made from 1996 have verified sufficient ability for detection of damaged fuel assembly. Alpha spectrometry measurement is prepared, to increase the sensitivity of the detection.

ACKNOWLEDGEMENTS

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REFERENCES

APPLICATION OF LESSONS LEARNED FROM QINSHAN PHASE III TO A FUTURE PROJECT

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Abstract. The Qinshan III Project achieved, through the use of advanced construction and project management tools and techniques and harmonious and cooperative relationships between project participants, first ranking international levels of nuclear project performance and quality, which include the shortest nuclear construction schedule in China and the shortest nuclear construction schedule for the first plant of a specific design in any country. The Qinshan III plant also incorporated the most advanced features of any CANDU plant, and, where objective measurements have been made, the highest quality achieved to date.

Even though the Qinshan III Project achieved world-class performance, there is potential for further improvements to the plant design and project implementation, which could improve the cost, schedule, quality and national self-reliance.

This paper describes and quantifies such potential improvements to assess their significance. For example, it is estimated that rearranging the schedule to take full advantage of the short component installation times and procurement times actually achieved on Qinshan III could shorten the construction schedule by as much as six months. Restructuring the project organization and scope splits to place more responsibility on Chinese organizations and reduce the scope performed by the foreign partners would result in improved costs and increased Chinese content. By adopting these measures, AECL estimates that:

- The schedule could be reduced by six months,
- The cost could be reduced by approximately 25%, and
- Chinese content could be increased to over 50%.

1. Introduction

The major project participants on the Qinshan Phase III CANDU Nuclear Power Project conducted a Lessons Learned Review to benchmark “best practices” from the Project for application to their individual companies on future nuclear work. The review identified tools, skills, techniques and contractual and organizational arrangements that represent advances to Canadian and Chinese nuclear plant project practice and to recommended measures to encourage their application to future nuclear projects. The review consolidated the gains in capability made by the Chinese and foreign partners while working on the Project, and identified further gains that could ultimately be realized in a replication project.

2. Conduct of Review

The Review covered each key project discipline area to identify performance on the Qinshan III Project, including successes, achievements and areas that could be improved and developed further. It recommended changes, further development and investments required to secure the advances in areas such as documentation, training, development or purchase of tools, changes to organizational or contractual arrangements or changes which would facilitate enhanced Chinese participation (localization) in future nuclear plant projects. Particular attention was paid to identifying and
3. Findings by Discipline

3.1. Project Execution

The Project achieved several records for schedule performance, including the shortest construction schedule for any nuclear unit in China for Unit 1. Many individual construction and installation activities established new records, some of them dramatically superior to previous performance.

The project execution was characterized for the most part by smooth relations between the participants. Contractors quickly familiarized themselves with CANDU specific technology and adopted new project management systems, quality assurance programs and technologies.

Wherever an objective measurement of construction quality was possible, it appears that the plant as built has achieved the highest standards to date. For example, the reactor building leak rate tests for both Units 1 and 2 showed superior results to those of the eight CANDU 6 containment buildings previously constructed, with the Unit 2 leak rate measuring a little over half that of the best previous result.

3.2. Engineering

Many suggestions for specific design improvements were identified during implementation of the Qinshan III Project. These had several objectives, including improved compliance with Chinese codes and standards, conformance to international practices, cost reduction, reduced schedule risk, improved plant quality, enhanced supply localization from Chinese sources, reduced susceptibility to obsolescence, incorporation of operational and commissioning feedback and improved maintainability and operability.

The review also recommended that more design change authority be delegated to the field and that engineering support to construction be improved.

3.3. Procurement and Material Control

The review recommended that material control be extended so that all materials are controlled through the main material control system and that complete equipment lists should be made available right from the start of the project.

Many substitutions of materials, which had been originally planned to be imported, were made during the execution of the Project, with beneficial results. Such substitutions would be extended to a replication project and have the potential to reduce cost and schedule, and also facilitate spares procurement. A recommendation was made to review materials requirements to identify Chinese codes and standards equivalent to those foreign codes and standards specified in the design documents. This would broaden the base of materials and components, including spares, which could be procured from Chinese sources and simplify the procurement process.

Major Chinese nuclear equipment manufacturers have reviewed procurement documents and technical specifications for major Nuclear Steam Plant (NSP) equipment. They have concluded that almost all items could be fabricated in their facilities without major investment in new equipment. A similar review has been performed for the Balance Of Plant (BOP) equipment, which concluded that 35% localization of major equipment items could be achieved, and a higher proportion of bulk materials.

The review concluded that a localization level of at least 50% could be achieved for the entire replication project.
3.4. Project Management

A strong recommendation was made for having an integrated construction and installation management team, to minimize interfaces, with their related complexity and issues. Improved interface controls between construction, installation and commissioning would result in improved performance. Since the Chinese would lead the team, this approach would also result in reduced costs and increased localization.

The project management approach used by AECL along with its high technology tools (AIM/TRAK, CADDs, InTEC, CMMS, P3) were accepted by project participants, and the Owner is extending their use to the Qinshan III operating station. These systems and their capabilities can be applied to new projects. The Chinese Construction Contractors also developed and successfully implemented some of their own control software for various operations. The methodology and practices used on Qinshan III were key factors in achieving the short project schedule. Use of these tools would be expanded on a replication project, leading to greater schedule reductions.

A particularly successful technique adopted by the project participants was the program of continuous improvements. This resulted in clear improvement of performance and quality between work done on Unit 1 and Unit 2, and contributed to the final design and project documentation being far superior to that which existed at the commencement of the project. This has been aided by the use of fully electronic documentation for all aspects of the project. This improvement trend clearly shows the potential for better performance on future nuclear projects.

3.5. Construction

The participants agreed that mastery of the open top construction approach was key to the success of the project. The open top construction approach and associated schedule flexibility was decided only after the Qinshan III contract was signed, so that the project plan was not optimized in advance to take full advantage of the resulting reduced durations for key construction activities. Many improvements could be achieved simply by planning work and procurement ahead of project commitment. AECL and other project participants concluded that a six-month schedule reduction (from 72 to 66 months) could be achieved, and identified specific measures that would contribute to achieving such a result.

An outstanding improvement over past practice was modularization of the housing system and its supporting structures. Installation durations were reduced to a small fraction of those experienced on previous CANDU 6 projects, while quality was improved. Modularization of additional systems offers the prospect for further reduced schedule risk and improved quality. As part of the Lessons Learned Review, a comprehensive evaluation was carried out by AECL and Hitachi to identify additional NSP and BOP systems that could be modularized further on a replication.

Further integration of construction management information into project databases was also recommended, including extending the application of the CMMS material controls to cover all construction materials.

3.6. Commissioning

Commissioning on Qinshan III was carried out by an integrated Commissioning Team comprised of staff from AECL and the Owner. The Owner’s commissioning team of about 1,000 staff were supplemented by about 50 foreign staff, who combined supervisory, guidance, mentoring and training roles.

The Owner, assisted by AECL, was responsible for all operations-related activities to support commissioning, including Health Physics and Radiation Protection, Chemistry, Nuclear Safety and Training. The Owner prepared a complete set of operating policies and procedures, which are now in place.
R. DeGregorio et al.

It is envisioned that AECL’s support during the commissioning of a replication project would be reduced substantially from the Qinshan Phase III levels.

4. Conclusions

The Qinshan III Project has successfully exceeded its goals of placing the first unit in service by 2003 February 12, and the second unit by 2003 Nov 12. Operation of both Units to date has been smooth.

The combination of international and Chinese experience and expertise on the Qinshan III Project has allowed the achievement of performance beyond what any of the participants could have achieved separately. The project benefited from drawing together expertise from companies with experience in implementing all of the internationally successful nuclear technologies, including CANDU (AECL), PWR (Chinese and Bechtel) and ABWR (Hitachi). Substantial transfers of capability were achieved which will assist the Chinese nuclear program to move to even higher levels of performance in future.

Several avenues for improving future project performance were identified. The Qinshan III Project was structured to meet Chinese needs and capabilities in 1996. These are different today and present new possibilities for future cooperation between the project participants, first in supporting the successful operation of the Qinshan III Plant and secondly in assisting the success of other nuclear projects in China and elsewhere.

The participants conclude that, based on the Qinshan III Project experience, were they to perform the same project today, they could achieve substantial savings in time (6 months), cost (25% reduction) and import dependence (>50% localization) while achieving a superior product.
LACK OF SAFETY CULTURE AS A CONTRIBUTING FACTOR IN MAJOR RADIATION ACCIDENTS REPORTED IN LATIN AND SOUTH AMERICA

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Abstract. In the last decades, the issue of improving safety culture to reduce the upward trend of radiation accidents has received considerable attention from scientific organizations. After the Chernobyl accident, IAEA invited worldwide experts in nuclear safety and formed a working group called “International Nuclear Safety Advisory Group”. The Safety Culture concept has been developed by the Working Group and published in the Safety Series collection of IAEA in 1986 as No 75-INSAG-4 which provided a logical framework for establishing safety culture policy at individual and organisational level[1]. The aim of this paper is to review the role of insufficient safety culture in the occurrence of 5 major radiation accidents reported in Latin and South America and the remedial action taken by competent authority and oriented towards improvement of basic principle of safety culture.

1. Introduction

Safety Culture: The initial definition proposed by INSAG stated that “Safety Culture is that assembly of characteristics and attitudes in organizations and individuals that, as an overriding priority, nuclear plant safety issues received the attention warranted by their significance”.

Requirements of safety policy at Government level are: Statement of safety policy, management structure, resources, self-regulation, commitment

At managerial level the features are: definition of responsibility, definitions and control of working practices, qualifications and training, rewards and sanctions

2. Case Studies

# 1 Case study: Radiation accident in San Jose, Costa Rica [2] in August 1996. After the replacement of a Cobalt 60 source of a radiotherapy unit, a source was calibrated, an error was made in calculating the dose rate. The error was discovered in late 1996. This miscalculation resulted in the death of 42 patients receiving higher radiation doses than those prescribed. Although Costa Rica completed in 1995 an important step by enacting a national radiation decree, in an IAEA official report of the accident, recommendations to the Government of Costa Rica are consistent with the universal features established for a sound safety culture policy.

# 2 Case study: Radiation accident in San Salvador [3] in February 6th, 1989. The source rack of a cobalt-60 gamma industrial sterilizer got stuck into irradiation position. Bypassing rules, the operator entered the room with two co-workers to free the rack. The three were exposed to high radiation doses. Amputation was necessary for two of them and the most exposed died 6 months after the accident. Effecting safety culture improvement a regulation was approved by this contry authority in March 2002 and various safety improvement programmes have been incorporated into a national policy.
# 3 Case study: Radiation accident in Mexico [4]. In Juarez City, from December 6th 1983 to January 1984. The triggering factor was: a ruptured Cobalt 60 teletherapy source. Dose estimates of 762 mrem received were by 12 workers involved in a junkyard in Phoenix. No deaths were recorded. Government enacted in February 4th, 1985 the Regulatory Law in its Constitutional Article 27 dealing with Nuclear Matters, incrementing the Commission regulation power and legal capacity in line with safety culture as recommended by INSAG.

# 4 Case study: Radiation accident in Brazil [5] at the Institute Goiano of radiotherapy, in Goiânia, Brazil, end of 1985. The Institute left the premises, leaving behind a cesium - 137 source which stayed insecure for 2 years. Scavengers took the unit home and ruptured the source capsule. 814 buildings have to be demolished due to contamination. More than 100.000 persons had to be monitored for radiation. In light of a good safety culture the experts have recommended that managers should assume their responsibility and good communication is required between all concerned parties along with appropriate training.

# 5 Case Study : Radiation accident in Peru [6] in the hydroelectrical power plant in Yanango, February 20th, 1999. A source from an industrial Iridium-192, was left unattended by a radiographer and pick up by a welder who took it home. He was hospitalized for severe burns from radiation, his wife suffered ulcerative lesion at the lower back IPEN, the regulatory authority, ordered analysis for all involved persons, stopping the company's operation and requesting the camera for investigation, documenting all records in writing, informing the media those measures were consistent with safety culture practices.

# 6 Case Study: Radiation accident in Panama [7] at the Instituto Oncológico Nacional (ION) in August 2000. Changing the method of digitalization of shielding blocks of a computerized treatment planning system (TPS) used by ION to calculate resulting dose distribution and determine treatment time. As a consequence 5 patients died. For effecting safety culture, recommendations have been made to put in place a quality assurance system, quality control procedures for treatment planning systems and testing new procedures.

# 7 Case Study: Radiation accident in Bolivia [8] involve radiation exposure due to an Iridium -192 source when four industrial radiographers manipulated and transported the equipment with the source outside the container and were overexposed. During an eight-hour bus trip, 30 passengers were also exposed since the source was carried as a normal luggage without shielding.

3. Conclusion

This retrospective overview pinpoints the fact that in some countries with acceptable safety culture, improvement was needed in some areas as in the cases of Brasil, Mexico and Peru and reassessment of their safety culture is essential as in the case of Bolivia. For others like Panama, Costa Rica, continuous self-assessment and further enhancement of safety culture will be necessary to prevent future radiological accident. National competent authorities should increase control in key practices like industrial radiography, irradiators, radiotherapy and implement a national program for possible lost sources. Since human error seems to be always a pattern in most cases, human training and management commitment are essential in setting safety policies, check lists, closed supervision and written procedures.

REFERENCES


Abstract. In April 1999 the Czech Power Utility (ČEZ a.s.) with the objective to achieve maximum level of nuclear safety and quality within the NPP Temelín commissioning procedures has established a special body - Scientific Supervisory Group (SSG) and requested Nuclear Research Institute Řež plc to perform the required function. This paper summarises the main tasks of SSG and provides a review of its activities and achieved results. Mentioned are also the main characteristics of Temelín NPP commissioning.

Nuclear Research Institute Řež plc ensures scientific supervision of NPP Temelín commissioning by fulfilling function of Scientific Supervisory Group for NPP Temelín Commissioning established by the plant Operator – Czech Power Utility (ČEZ a.s.).

1. Introduction

The SSG proceeds in accordance with its Statute and provides an independent specialised professional and expert work focused on nuclear safety assurance, assessment of the selected documentation related to plant commissioning and operation, assessment of the plant preparedness for the individual commissioning stages, and, of course, on assessment of the commissioning tests results. While performing its function the SSG is guided by the Atomic Act and the relevant Directives of State Office for Nuclear Safety; its activities are in compliance with the applicable IAEA recommendations.

2. The SSG effort fulfils the following tasks:

- reviews the licensing and operation documentation as to its completeness and scientific-technical state-of-the-art – from the nuclear safety assurance standpoint
- assesses and recommends for approval the nuclear safety related documentation
- assesses and recommends for approval the low-power start-up and power ascension programmes
- within the Hot Functional Test programme, assesses preparation, performance and results of the equipment tests related to the plant unit nuclear safety. The SSG makes out its position as to the tests launching and then carries out supervision over the experimental part of the Integral Hydraulic Test
- within preparation and realisation of the “Active Testing” it examines the unit preparedness and compliance with the approved programmes, examines also how the partial programmes were fulfilled
- supplies recommendations (if needed) on repetition, completion and/or changes in the tests scope
- checks on the fulfilment of the nuclear safety principles and rules, on the observance of Technical Specifications for Safe Operation during activities within the start-up tests. Immediately notifies the Operator’s shift personnel on the cases when nuclear safety could have been or was violated
- evaluates and recommends for approval the low-power and power ascension tests results as to their completeness, quality of performance and observance of the tests acceptance criteria
- recommends for approval the summary reports on the completion of the individual commissioning stages and their parts
I. Váša and C. Svoboda

- monitors whether conclusions which have followed from the commissioning programme evaluation are included into the unit operational documentation
- if needed, provides expert opinion on proposals for equipment and documentation modifications following from the commissioning results
- if needed, provides expert opinion on recommendations for safety enhancement following from the commissioning results.

3. Evaluation

Important parts of the SSG activity are preparation and presentation of independent evaluations of the commissioning tests (on each prescribed power level) for the SUJB Chairman Advisory Committee on Nuclear Safety. These evaluations and expert opinions are transmitted to the SUJB as parts of the package of the required by law licensing documentation submitted by the NPP Operator. SSG activities are however not limited to “surveillance” over nuclear safety during individual commissioning procedures and to evaluation of the results, in the majority of cases the SSG specialists also perform their own parallel evaluations using the primary input data. SSG recommendations to the Operator to continue the commissioning process are conditional on the positive results of such evaluations.

Main characteristic of NPP Temelín commissioning:

Physics start-up tests represent relatively short but very important phase of any nuclear power plant commissioning.

3.1. Physics start-up tests

Physics start-up tests of both Temelín units were in compliance with the approved “Phase Programme of Physics Start-up Tests” divided into three time periods, while

1st period includes:
- fuel loading
- assembling of the reactor
- filling-in of the primary circuit by coolant
- leak-tightness pressure tests of the primary and secondary circuits
- repeated containment integrity test

2nd period includes:
- heating above the brittle fracture limit temperature
- heating to semi-hot (Mode 4) and hot (Mode 3) state
- performance of tests prescribed by the operational procedures for reactor start-up
- repeating some tests performed during Hot Functional Tests (to complement systems and equipment testing)

3rd period includes:
- reaching initial reactor criticality
- low – power tests

The low-power tests

The low – power tests are performed with the objective to verify characteristics of the reactor core important for nuclear safety as well as for more accurate determination of neutron-
physical characteristics, which are used (as an operational procedure) in the course of unit operation. These tests were performed without any problem on both units.

The power ascension tests on the Unit 1 were realized in the accordance with the programme. The reasons for changes and revisions were following:
- Some tests were supposed on inconvenient power level. Typical example are some tests (with working turbine) on 30% Nnom. The generator power was 200 MWe approximately; this value is the minimum turbo-generator power, recommended by producer.
- Some test requirements, or supposed test conditions were not in accordance with the unit technology or operation regulations
- Some tests had to be repeated (on high level or under more proper conditions) because of their failure
- The most number of changes were caused by two main problems, which represent the turbine vibration and incorrect function of ARGUS valves

The timetable of the energetic start-up is given in table below.

<table>
<thead>
<tr>
<th>Power level</th>
<th>Period</th>
<th>Interruption</th>
<th>Cause</th>
</tr>
</thead>
<tbody>
<tr>
<td>5%</td>
<td>Oct 30 – Nov 10 2000</td>
<td></td>
<td></td>
</tr>
<tr>
<td>12%</td>
<td>Nov 14 – Dec 9 2000</td>
<td></td>
<td></td>
</tr>
<tr>
<td>30%</td>
<td>Dec 15 2000 – Mar 8 2001</td>
<td></td>
<td></td>
</tr>
<tr>
<td>55%</td>
<td>Mar 19 – Sep 27 2001</td>
<td>Apr 3 – 16</td>
<td>Turbine vibrations</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Apr 26 – Aug 12</td>
<td></td>
</tr>
<tr>
<td>75%</td>
<td>Oct 19 – Dec 18 2001</td>
<td>Nov 1 – 26</td>
<td>MCP 2 leakage</td>
</tr>
<tr>
<td>90%</td>
<td>Dec 21 2001 – Jan 3 2002</td>
<td></td>
<td></td>
</tr>
<tr>
<td>100%</td>
<td>Jan 10 – May 24 2002</td>
<td>Feb 24 – Apr 22</td>
<td>ARGUS valves malfunction</td>
</tr>
</tbody>
</table>

4. Conclusion

The power ascension tests on the Unit 2 were realized in the accordance with the programme. There were made some changes in this programme in comparison with Unit 1, using the experience obtained during Unit 1 commissioning. The number of power levels was reduced and also the number of „heavy“ dynamic tests was reduced too. The tests were planned on more convenient power level.

Because of application the design changes, according the experiences from Unit 1, the problems of steam supply pipelines and turbine vibrations and problems of malfunction of feed water tank heat-up control valves did not occurred at Unit 2.
The experiences obtained during the dynamic tests on Unit 1 were applied also in Unit 2. Because of this fact, only one dynamic test was not successful and had to be repeated.

The quite unexpected problem was short-circuiting on the both generator rotors. The programme was revised as a consequence of this problem. The tests, needed the generator in operation, planned at power level 30% Nnom, were realized on the next power level.

The another problem, occurred at power level 55% Nnom, was the longitudinal crack at steam supply pipeline. After temporary reparation the tests were finished without troubles from the point of view of this pipeline. Nevertheless this problem must be analyzed and solved. In general it is possible to say that Unit 2 secondary side leaks represented bigger problem in comparison with Unit 1, but it is the only one exception. In general, the power ascension tests of the Unit 2 in comparison with Unit 1 met less troubles and difficulties.

The timetable of the energetic start-up is given in table below:

<table>
<thead>
<tr>
<th>Power level</th>
<th>Period</th>
<th>Interruption</th>
<th>Cause</th>
</tr>
</thead>
<tbody>
<tr>
<td>30%</td>
<td>24.6.02 – 3.9.02</td>
<td>6.7.02 – 16.8.02</td>
<td>TG – generator rotor replacement</td>
</tr>
<tr>
<td>55%</td>
<td>5.11.02* - 31.1.03</td>
<td>15.11.02 – 9.12.03</td>
<td>Generator rotor repair</td>
</tr>
<tr>
<td></td>
<td></td>
<td>5.1.03 – 18.1.03</td>
<td>I.O., II.O. leakage fix</td>
</tr>
<tr>
<td>75%</td>
<td>6.2.03 – 19.2.03</td>
<td></td>
<td>I.O., II.O. leakage fix</td>
</tr>
<tr>
<td>100%</td>
<td>25.2.03 – 7.4.03</td>
<td>6.3.03. – 27.3.03</td>
<td>I.O., II.O. leakage fix</td>
</tr>
</tbody>
</table>

- Between power levels 30% Nnom and 55% Nnom the second generator rotor replacement
Comparison of Unit 1,2

- Days HFT 1st loading low power tests power asc. Tests EFPD
- Unit 1: 38 26 113 102 570
- Unit 2: 26 8 112 312 122,7

Bars represent the duration of each phase for Units 1 and 2.
UPGRADING OF NUCLEAR POWER PLANTS IN THE SLOVAK REPUBLIC

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Abstract. Upgrading of technical status of all nuclear power plants represent an important feature for global upgrading of nuclear safety in the world. Thought the backfiting programmes were established for NPPs operated in Slovakia – Russian design NPPs with WWER 440 reactors – the comprehensive upgrading programmes started in nineties of the previous century. The most comprehensive upgrading project was accomplished at NPP Bohunice V-1, which represents the first generation of NPP with the reactors WWER 440 model V230. All upgrading programmes benefited from the IAEA technical co-operation projects oriented to upgrading of the safety of Russian designed NPP built to early standards.

1. Introduction

The Slovak Republic represents the country with limited primary energy sources. This fact is not evident from the electric installed capacity distribution shown on Fig. 1. Installed electrical capacities in Slovakia but fully visible from the electricity generation in Slovakia shown on Fig. 2. From the above mentioned figures it is clear that major portion of the electricity – approximately 55% - is produced by nuclear power plants and that there is the only one owner and operator of these NPPs – Slovenske elektrarne, a.s. (SE, a. s.).

Fig. 1 Installed electrical capacities in Slovakia

Slovenské elektrárne, a.s. operates two sites with nuclear power plants. At Bohunice site there are two nuclear power plants. The first one called as V-1 is the nuclear power plant with two reactors WWER 440 model V 230 they were commissioned in 1978 and 1980 respectively, the second one is a nuclear power plant with two reactors WWER 440 model V 213 which were commissioned in 1984 and 1985 respectively. At Mochovce site there is a nuclear power plant with two reactors WWER 440 model V 213 commissioned in 1998 and 1999 respectively. Two additional units with the reactors WWER 440 model V 213 are under construction which has been frozen.
To assure safe and reliable operation of all units a huge upgrading programmes has been carried out in Slovakia.

2. Nuclear Power Plants Upgrades and Backfitings

As a rule, safety upgrading measures for WWER 440 reactors have generally been oriented towards improving reliability, redundancy, single failure concept (in particular with respect to V-230 reactors), and physical separation of safety systems.

Lists of safety-related imperfections the treatment of which is included in the safety upgrading program for specific reactor types have been a result of recent developments in the field of primary circuit integrity, assessment of events at nuclear installations, results of beyond-design basis accident analyses etc.

Nuclear Regulatory Authority of the Slovak republic (ÚJD) uses deterministic approach to effectively manage the safety upgrading process, in particular to improve the safety of emergency systems (independence, redundancy). PSA for specific reactor is used to prioritise the individual safety improvement measures, in particular those which may contribute most to the core damage.

Safety upgrade related requirements are partially linked to the probability of accidents. Acceptance criteria set forth by ÚJD for emergency analyses are generally expressed in terms of acceptable radiological consequences which differ according to the probability of the initiation event. In addition, conservative or so-called best estimate procedures were described for emergency analyses. Best estimate procedures are only accepted for accidents with the least probability of a specific initiation event (below 1E-6).

2.1. Upgrading of NPP Bohunice V-1 (WWER 440/V-230)

The original Russian design of units VVER-440/230 series originates from the end of sixties and the beginning of seventies. Rupture of primary coolant circuit with equivalent diameter of 32 mm was stated as maximal design basic accident, for managing of which the emergency core cooling systems were dimensioned in compliance with those days valid legislation.

The design of V-1 units, which are the youngest ones of V-230 series, took into account experience and knowledge from previous units, what was reflected in higher safety level and operational reliability in comparison with other units of VVER 440/V-230 series. Among the most important improvements of V-1 units design there are:

- supplementing of steam generator super-emergency feed-water system,
higher capacity of emergency core cooling system,
• supplementing of automatic links between primary and secondary circuit systems,
• higher level of secondary system automation.

Process of safety improvement and operational reliability of V-1 units began immediately after commissioning. There were performed more than 1200 design modifications of bigger or smaller scope, which resulted from operation status evaluation, operational experience and from various international recommendations and regulations. The modifications were performed by own workers and also in contractual way. Among the most important arrangements belong:

• supplementing of shielding assemblies into the core to decrease neutron flux on reactor vessel and reactor pressure vessel brittle fracture risk,
• supplementing the emergency power supply of the most important consumers of emergency systems,
• reconstruction and supplementing of operation computer systems and supplementing the monitoring of parameters after failures,
• supplementing the diagnostic systems for monitoring the status of power plant main components,
• reconstruction and replacement of lower reliable I&C components and supplementing of more automatic functions for improvement of safety and operational reliability.

Despite these facts the strict requirement on safety upgrading was formulated at the beginning of nineties.

Based on a review of the nuclear safety of reactors V-230, Czechoslovak Atomic Energy Commission (CSKAЕ), the predecessor of ÚJD, by CSKAЕ Resolution No. 5/1991, defined measures to upgrade the safety (known as „Small Reconstruction“), in response to the safety shortcomings identified. The „Small Reconstruction“ was completed in 1993. It focused on the most urgent safety aspects, in particular on the tightness of the confinement, seismic upgrading, reactor vessel integrity, emergency sources of electric supply, reliability of steam generators feed water systems, the application of „leak before break“ principle, on the reduction of fire risk and the capacity of the fire extinguishing equipment, etc. The implementation of the mentioned safety-related measures was a condition for the licensing of further operation.

Being established in 1993, ÚJD took additional measures to upgrade V-230 reactor safety. Based on the inception safety report for so-called „Gradual Reconstruction of V-1 Units“, ÚJD issued Resolutions No 1/94 and 110/94 in 1994, which regulated the conditions for further operation of V-1 units until the end of the designed life cycle. ÚJD Resolution No.1/94 contained additional 59 safety measures to be gradually implemented within 1996-2000.

ÚJD Resolution 1/1994 had five topics-based parts. In each of them, ÚJD prescribed measures to upgrade safety level; any further operation of the power plant would be conditional on the implementation of those measures.

Part one took measures to maintain primary circuit integrity including analyses of the integrity of reactor pressure vessel. Part two contained measures focusing on a safety improvement of the reactor core cooling systems during the operation and accidents. Part three, reactor core cooling during LOCA, set forth conditions for an improved core cooling to manage a new level of design basic accident, i.e. when a primary circuit pipe of the diameter of 200 mm ruptures (the original design basic accident only considered the rupture of a primary circuit pipe of a diameter equivalent to 32 mm), as well as 500 mm diameter main primary pipe ruptures using best estimate approach. Part four contained measures to improve the safety level of confinement, in particular by improving its tightness, efficiency and demonstrating integrity upon LOCA. Part five focused on an improvement of the safety
M. Ziakova

level of auxiliary systems of the nuclear power plant such as service water, electric supply, and instrumentation & control system.

Resolution No. 1/1994 was also based on recommendations for safety level upgrading contained in IAEA documents developed as a part of the TECDOC 640 document: Ranking of Safety Issues for WWER 440/230 Reactors. Because of the extent of the project, the deadlines for the measures to be implemented have been spread over the period of five years. ÚJD Resolution No.110/1994 complemented CSKAЕ Resolution No. 5/1991 and ÚJD Resolution No.1/1994, by defining conditions to be met by operator to obtain license for further operation of the respective unit. The license validity has been restricted to one year.

In assessing the safety level of nuclear power plants, results of probabilistic safety analyses (PSA) have been used since 1995; they are also used to assess the benefits of suggested safety improvements. Preliminary safety analysis report for gradual reconstruction and IAEA experts meeting for its evaluation were the basis for ÚJD resolutions.

2.1.1. Small reconstruction of NPP V-1

Implementation of measures during SMALL RECONSTRUCTION V-1 units became (from the core damage probability point of view) a power plant, which further operation is acceptable from safety point of view, but further safety improvement are necessary.

<table>
<thead>
<tr>
<th>Area of Improvement</th>
<th>Probability of malfunction per year</th>
<th>Improvement index</th>
</tr>
</thead>
<tbody>
<tr>
<td>Emergency reactor shutdown caused by low pressure</td>
<td>$1.92 \times 10^{-2}$</td>
<td>$3.1 \times 10^{-4}$</td>
</tr>
<tr>
<td>Emergency reactor shutdown caused by loss of feed water</td>
<td>$9.1 \times 10^{-3}$</td>
<td>$1.8 \times 10^{-4}$</td>
</tr>
<tr>
<td>Emergency reactor shutdown caused by opening pressurizer safety valves</td>
<td>$1 \times 10^{-3}$</td>
<td>$2.2 \times 10^{-5}$</td>
</tr>
<tr>
<td>Loss of emergency power supply of the 2nd category</td>
<td>$2.4 \times 10^{-2}$</td>
<td>$1.24 \times 10^{-3}$</td>
</tr>
<tr>
<td>Malfunction of emergency feed-water system</td>
<td>$3.2 \times 10^{-1}$</td>
<td>$7.44 \times 10^{-2}$</td>
</tr>
</tbody>
</table>

Tab. 1 Comparison of CDF for V-1 before and after small reconstruction

2.1.2. Gradual Reconstruction of NPP V-1

Regarding decision No. 5/91 of Czecho-Slovak Nuclear Regulatory Commission on possibility to operate V-1 units after 1995 under the condition, that nuclear safety will be increased to European standards level, the preparation of "Principal reconstruction" started with working teams of EBO, VÚJE, Škoda and EGP since 1991. Result of mentioned above is issuing the "Safety concept for V-1 principal reconstruction" dated June 1992, which was re-elaborated in February 1993 to the form of "Safety concept for gradual V-1 units reconstruction" and which was reviewed by IAEA mission on July 1993 and approved by ÚJD decision No. 1/94 dated February 24, 1994. ÚJD decision No. 110/94 dated August 25, 1994 elaborated the original Czecho-Slovak Nuclear Regulatory Commission decision No. 5/91 in the field of V-1 units operation conditions after 1995 and has changed the original license for V-1 operation till the end of designed lifetime to approval for operation only for the next year under the condition of positive evaluation of works performed last year and refueling and approved scope of reconstruction works for next year and refueling. ÚJD decision No. 1/94 supplemented and made more precise the safety concept for V-1 units gradual reconstruction in 59 measures (37 measures for analyses and evaluations and 22 measures for realization of reconstruction works). This decision of ÚJD concerns following safety related functions:
M. Ziakova

- Primary circuit integrity
- Core cooling during operation and in case of accident
- Core cooling in case of coolant leakage from primary circuit
- Confinement
- Auxiliary systems
- Service water
- Power supply
- Instrumentation & Control
- Fire protection
- Seismic measures

Further organizational measures were stated by ÚJD decision No. 110/94.

Implementation of this V-1 units gradual reconstruction is focused on fulfilling following probabilistic and deterministic targets:

- managing the newly defined DBA (coolant leakage through $\phi 200$ mm) by conservative approach and BDBA (coolant leakage through $\phi 500$ mm) by best estimate methods,
- confinement tightness and localising systems must ensure, that in case of coolant leakage through $\phi 200$ mm the dose equivalents will not be exceeded (50 mSv for whole body and 500 mSv for thyroid) in monitored power plant area and in case of coolant leakage through $\phi 500$ mm by best estimate methods 250 mSv on whole body and 1,500 mSv on thyroid,
- separation of redundant trains of safety system and supporting systems
- completion of seismic resistance of all safety related systems and equipment of the unit and corresponding buildings and systems to 8° MSK-64 (250 cm/s² horizontally and 130 cm/s² vertically).
- safety systems must fulfils the requirement on reliability with malfunction probability $10^{-3}$ per demand or less,
- safety systems will provide, that probability of core damage probability is better than $10^{-4}$ for reactor per year,
- reliability of reactor protection system is $10^{-5}$ per demand, at least

![Core damage frequency development for NPP V-1](image.png)

**Fig. 3 Core damage frequency development for NPP V-1**
M. Ziakova

Elaborating the BASIC ENGINEERING for V-1 units gradual reconstruction was provided mainly by the company Siemens KWU in co-operation with Slovak companies like VÚJE Trnava, VÚEZ Tlmače, PPA Bratislava, EZ Bratislava and other.

Basic Engineering was finished in November 1996. The results assumed were assessed and commented by EBO and independent organizations and finally they were submitted to Slovak Nuclear Regulatory Authority for approval.

Based on approved results from BASIC ENGINEERING and PSAR the documentation for each individual system was completed up to the level of realization documentation and consequently according to this provided realization of proposed and approved reconstruction works.

Implementation of modifications and reconstruction of individual systems are done gradually in scheduled unit refueling outages. Unit refueling outages are extended according to the necessity. Assumed deadlines for realization of V-1 units gradual reconstruction are from the end of 1996 and last modifications are supposed during year 2000.

After finishing of the gradual reconstruction of the V-1 units and realization of measures to increase operational safety, among which also belong the measures to increase the fire-safety, the fire-protection of the blocks of the nuclear power plant Bohunice reached an acceptable international standard applied to the unites of the same vintage at west European countries.

Carrying out the Gradual reconstruction, which is considered as pilot complex safety upgrading program of WWER 440/V-230 reactor design we assume to reach the internationally acceptable level of nuclear safety and operational reliability of the Bohunice V-1 units. This will establish technical assumption for reliable, safe and economic operation of V-1 units at least up to the end of design lifetime.

2.2. Upgrading of NPPs with reactors WWER 440 model V-213

The original design of NPP Bohunice V-2 and NPP Mochovce both with reactors WWER 440 model V-213 had safety improvements as compared to their predecessors NPPs with the reactor WWER 440 model V-230. These relate to a higher capacity of safety systems the majority of which have been designed based on 3 x 100% back-up, additional of the containment with a pressure reduction system to cope with any accident involving a coolant loss. Even despite this measures are being progressively implemented to maintain high safety standards also at V-2.

2.2.1. NPP Bohunice V-2

Over 300 technical modifications have been made at the nuclear power plant V-2 since it was commissioned in 1984 and 1985 respectively. Between 1993 and 1996 the first reassessment of V-2 safety had been carried out, under which an updated Safety Report After 10 Years in Operation was prepared as one part. In addition to the domestic reassessment also IAEA international safety assessments were made in 1994 and 1996.

ÚJD responded to the situation following the Safety Report After 10 Years of Operation by issuing the Resolution No. 4/1996 which contained measures to upgrade safety in three parts.

Part one contained measures concerning the completion of the safety report after 10 years of operation. The safety report, as a basic document providing evidence for the safety level of the nuclear power plant, was updated according to the actual state-of-the-art condition of the plant, in particular with respect to better safety analyses. Measures to complete the safety report were taken by ÚJD based on the IAEA 1994 Mission, PSA analyses conducted by Nuclear Power Plants Research Institute, Trnava, and analyses developed by ÚJD staff. Requirements to complete the safety report in addition to general issues, also concerned the integrity of components at the primary circuit pressure limit, seismic review, safety analyses, additional safety analyses, limits and conditions.
Part two identified technical measures for safety upgrading of safety systems and auxiliary systems of safety relevance. This part further specifies requirements with respect to the development of a modernisation concept for the instrumentation and control systems. The resolution also contains organisation-related measures to improve operation documentation, regulations for tests of equipment and emergency planning. This procedure prescribed by ÚJD shall maintain the required safety level of the nuclear power plant.

Part three prescribed the operator to periodically review the nuclear power plant safety following the schedule developed by ÚJD. From the aspects of their deadlines, two types of measures were distinguished: short-term measures including organisation-related measures, and completion of the safety report. Long-term measures represent requirements concerning the specification of objectives, development of concept and safety upgrade program.

Annex to Resolution No. 4/1996 contained ÚJD comments on the individual chapters of the Pre-Operation Safety Analysis Report. Program for safety upgrading has been developed by utility based on ÚJD resolution No. 4/1996.

Based on the requirements of the ÚJD, stated in the decision No. 4/96, of the operator and of the recommendation of the IAEA (EPB-WWER-03) SE,a.s. prepared in 1998 a material concerning the present state of the V-2 units called ”Updated Safety Improvements of the V-2 Units and a Proposal for Their Solution”. This material is the essential part of the Modernization Program, which will be detailed up in the ”Safety Concept”.

The safety concept for V-2 modernisation and safety enhancement has been developed by the Slovak engineering, design and research company VÚJE Trnava, a.s. which has also become the general designer of upgrading of the two V-2 units. The V-2 safety enhancement programme has been divided into two parts - the safety concept and the very implementation of V-2 modernisation scheduled for three stages: by 2004, 2006 and 2008. The modernisation includes 50 main tasks.

The adopted V-2 units upgrading and safety enhancement programme includes the results of the activities have been carried out over the past years with the aim of defining all important actions leading to increased nuclear safety, operation reliability and economy during the period of their life and creating conditions for operation extension.

The goal of the Modernization Program is to reach the safety of the units in the following points:

- to increase nuclear safety to the level of the proposals and recommendations of the IAEA (according to IAEA-EBP-WWER-03)
- to reach probability goals of the reactor:
  - probability of an active zone damage (CDF) <10E-4/year
  - probability of leak of RA elements exceeding the permitted doses for human population <10E-5/year
  - probability of failure of the safety systems <10E-3/demand
  - probability of failure of the damage shields <10E-5/demand

The time frame for implementation of Safety Improvement Programme Bohunice V-2 in all significant issues within this programme is setting-up in UJD SR decision No 214/2000.

ÚJD SR required from Bohunice V-2 to develop specifications of individual measures in compliance with the document ”Safety concept for V2 NPP upgrading and safety improvement”, as well as their implementation time schedule in order to: ”Increase the safety level of V-2 NPP in compliance with IAEA recommendations for NPPs of WWER 440/213”, which are defined in the IAEA project EBP-WWER-03, published in the document ”Safety Issues and their ranking for WWER - 440/213 NPPs, EBP-WWER-03, April 1996”, as follows:
• Implement the measures of category III by 2004
• Implement the measures of category II by 2006
• Implement other measures, specified in technical specifications and time schedule by 2008
The programme is implemented in accordance with the schedule.

2.2.2. **NPP Mochovce**

Due to the fact that the construction of NPP Mochovce was interrupted at the beginning of nineties there was possibility to implement upgrading programme before the finishing of the construction and commissioning of the plant.

The aim of the safety improvements through the safety measures (SM) is to achieve a safety standard for NPP Mochovce to meet the requirements of „in-depth safety“ concept according to IAEA - INSAG 3. Cooperation of SE-EMO with other countries having interest in improving the safety standard may be expressed by the following activities:

• Replacement of the original instrumentation and control system ASRTP by SIEMENS system,
• Supply of a security and protection system by the company LANDIS & GYR,
• Development of an operation quality program in cooperation with the company IVO International from Finland,
• Construction of a full-scope simulator for the training of the operating staff, supplied by companies S3 Technologies USA and SIEMENS,
• Development of a study: „Assessment of NPP Mochovce’s Environmental Impact“ by the English company AEA TECHNOLOGY in 1994 as a part of the „Project Documentation for Public Participation Program„, developed by EDF and SE a.s. This documentation was developed according to EBRD requirements, and was at that time a precondition for foreign capital input to the NPP Mochovce project.
• Participation of experts from Electricité de France in the construction and start-up.

NPP safety improvement is an ongoing concern of the SE-EMO operator. A logical continuation of the activities of EdF in the area of safety reviews as summarized in „Safety Improvement Report„, (SIR) which was opened to public commenting, is the „NPP Mochovce Safety Improvement Program“ developed in 1995. This program has been conceived as a long-term one. It however is aimed at reaching a safety standard upon the NPP commissioning at the level corresponding to the internationally recognized requirements and standards included in the IAEA safety guides accepted by ÚJD, and at setting up good conditions for continuous safety improvements in the future. This approach is in compliance with the actual world trend of safety improvements, extension of life cycle, upgrading, and raising the outputs of already constructed NPP; it is based on a continuous monitoring of world progress in the field of nuclear safety.

The NPP Mochovce safety improvement program is based on the document entitled „Safety Issues and their Ranking for NPP WWER 440/213“; other basic documents include the outcomes of the safety review conducted by RISKAUDIT in 1994, as well as conclusions at the IAEA Safety Improvement of Mochovce NPP Project Review Mission - SIRM taking place at Mochovce in June, 1994.

The classification of the safety issues is based on the IAEA document „Safety Issues and their Ranking for NPP WWER 440/213“, the only exception being the safety measures added based on the specific situation of NPP Mochovce which are not classified according to IAEA. The following Table shows the description of the various categories. As will be shown below, there was no safety issue concerning the power plant Mochovce classified among the most serious category IV.
The NPP Mochovce safety improvement program is based on the defence-in-depth concept, and its aim is to verify and demonstrate the fulfillment of „general principles of nuclear safety“ as defined in the IAEA document INSAG-3. This approach is deterministic in nature, it however also combines with the probabilistic approach in accordance with the technical safety objective as defined by INSAG-3:

- The probability of serious reactor core damage should be less than 10^{-4} per year of reactor operation. The application of safety principles should bring this value down to 10^{-5} per reactor/year operation,
- The probability of significant leak of radioactive substances to outside of the NPP site should be below 10^{-6} per reactor/year operation.

### Definitions of safety issue categories according to IAEA

<table>
<thead>
<tr>
<th>Category</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Category I</td>
<td>Deviation from recognized international practices. Remedial measures are advisable.</td>
</tr>
<tr>
<td>Category II</td>
<td>Safety significant problems. In-depth protection weakened. Remedial measures are needed.</td>
</tr>
<tr>
<td>Category III</td>
<td>Problems of high safety significance. In-depth protection insufficient. Immediate remedial measures are to be taken. Temporary measures may be necessary.</td>
</tr>
<tr>
<td>Category IV</td>
<td>Problems of highest safety-related importance. In-depth protection unacceptable. The issue requires immediate intervention and substitution measures have to be immediately taken until complete resolution of the problem.</td>
</tr>
</tbody>
</table>

Probabilistic safety review (Level 1 PSA study and Shutdown PSA) is a part of the NPP Mochovce´s safety improvement program. Based on the results of the analyses already conducted within the PSA studies for WWER-440/V213 type NPP, mainly PSA for Bohunice V-2 units, as well as based on the interim results of the PSA study for NPP Mochovce, it could be stated that:

- the achievable probability of a damage to reactor core for WWER-440/213 NPP projects, including NPP Mochovce, is below 1E-4 per reactor/year operation without any extensive hardware adjustments,
- Implementation of safety measures focusing on hardware adjustments may bring the probability of core damage to the level of 1E-5 per reactor/year operation.

Based on the above facts, the safety-related objective of the NPP Mochovce safety improvement program from the aspect of active core damage is to achieve a value lower than 10^{-5} per reactor. year operation.

The operator of the plant in cooperation with VÚJE, a.s. developed a set of technical specifications for 87 safety measures (TSSM) to be implemented under the „NPP Mochovce Nuclear Safety Improvement Program“, with taking into account specific measures as identified by the RISKAUDIT and SIRNM Reports and experience with Bohunice V-2 and NPP Dukovany units (see Table 2.3.2). This has introduced certain differences between the „NPP Mochovce Safety Improvement Program“ and the IAEA document „Safety Issues and their Ranking for NPP WJE 440/213“ (certain measures have been added characterized as no-category measures). The differences are as follows:

- Issues concerning ergonomy of control rooms have been omitted from I&C issues. Since the entire I&C system has been replaced, the issue is not relevant to NPP Mochovce. A safety measure was added concerning the replacement of the computer equipment of the in-core monitoring system,
Two safety measures have been added with respect to electrical systems; both are specific to NPP Mochovce,

A safety measure has been added with respect to containment; the objective is the application of the full-range experiment planned to be conducted under PHARE projects,

Separate accident analyses on: „Full Loss of Electric Supply“ and „Full Loss of Heat Removal“ have been added to the final version of the IAEA document „Safety Issues and their Ranking for NPP WWER 440/213„, which was issued after the „NPP Mochovce Safety Improvement Program“, 

The „NPP Mochovce Safety Improvement Program“ does not include „surveillance programs“ (based on a contract with VÚJE, a.s.) and „operating staff training programs“ (in connection with the implementation of a full-scope simulator) since those programs were developed before the „NPP Mochovce Safety Improvement Program“ was defined.

<table>
<thead>
<tr>
<th>AREA</th>
<th>Category III</th>
<th>Category II</th>
<th>Category I</th>
<th>No Category</th>
</tr>
</thead>
<tbody>
<tr>
<td>BGeneral</td>
<td>1</td>
<td>2</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Active zone</td>
<td></td>
<td>1</td>
<td></td>
<td></td>
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<tr>
<td>Component integrity</td>
<td>1</td>
<td>4</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>Technological systems</td>
<td>2</td>
<td>12</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>Control and management systems</td>
<td>8</td>
<td>2</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>Electric systems</td>
<td></td>
<td>2</td>
<td>3</td>
<td>2</td>
</tr>
<tr>
<td>Containment</td>
<td>1</td>
<td>3</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>Internal risks</td>
<td>2</td>
<td>4</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>External risks</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>Accident analyses</td>
<td>5</td>
<td>8</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Operation</td>
<td></td>
<td></td>
<td></td>
<td>11</td>
</tr>
<tr>
<td><strong>Total 87</strong></td>
<td><strong>8</strong></td>
<td><strong>42</strong></td>
<td><strong>22</strong></td>
<td><strong>15</strong></td>
</tr>
</tbody>
</table>

TSSM became the basis for contract negotiations with the contractors responsible for individual safety-related issues. The following organizations have been selected to implement the various parts of the TSSM: EUCOM (a consortium of FRAMATOME, France and SIEMENS, Germany), ŠKODA Praha a.s., ENERGOPROJEKT Praha a.s., Russian organizations representing the original designer and constructor under the umbrella of ATOMENERGOEXPORT (Atomenergoproekt, Kurchatov Institute, Moscow, Zarubezhatomenergostroj, VNIIAES Moscow, SNIIP Systematom, Tenzor Dubna), and VÚJE, a.s.

SE, a.s. has kept the responsibility for 7 safety-related measures which are dealt with by own staff or in cooperation with other organizations. As an example, a joint team of SE-EMO experts and those of the Brussels branch of WESTINGHOUSE developed new guides for the management of emergency situations. The safety measures focus on the organization of operations (area „Operation„, according to „Safety Issues and their Ranking for NPP WWER 440/213„).

An implementation model has been developed for the various safety measures; the model respects the original guarantees for the NPP project and its realization. From the nuclear safety viewpoint, this means keeping and upgrading of the original safety concept. The CM implementation was split into the following stages:
M. Ziakova

The individual safety-related issues were analyzed by the investigators and concepts of technical solutions (so-called „basic design“) have been suggested if there was a need to modify the NPP design at this stage. The results of the analyses and the suggested basic designs were reviewed by all organizations involved, including VÚJE, and were discussed with ÚJD on a continuous basis.

As soon as the basic solution had been approved, the change in NPP design was accepted based on the corresponding QA programs and the Decree No. 105/81. The „amendment procedure“ was instituted, and the general designer developed addendums to the initial project. The task of the general designer (EGP) was to coordinate the technical solution with the original concept of NPP Mochovce, including links to other technological systems (CMS, electrical systems) and implications for the construction part. Addendums to the initial design were submitted to ÚJD for approval. Construction plans were then prepared based on the approved addendums, to the initial project serving the implementation of adjustments to the existing technological systems and buildings.

Safety measures resulting in adjustments of technological systems or buildings (i.e. with „hardware“ implications) were implemented by the general contractor of the construction and/or technological part and final suppliers. Results of safety measures without „hardware“ implications have been documented in the safety report and/or additional supporting documentation.

Of the total amount of safety measures, 70% were implemented prior to the first unit being put into operation, category III and II safety measures (which all were implemented) being taken as top priority. The remaining safety measures will expectedly be implemented before the second unit is put into operation in 1999.

The original project of NPP Mochovce was developed based on the knowledge of seismic risks of the site acquired during the preparation and design of the construction of NPP Mochovce in the 80s, accounting for MSK scale grade VI at which a safe shut-down of the reactor has to be secured upon an earthquake with a horizontal acceleration value of PGA = 0.06 g. The legislative development represented by the IAEA Safety Guide 50-SG-D15 has recommended for nuclear power plants a minimum acceleration value in horizontal direction of 0.1 g.

Owing to the above, „Selected Buildings and Technological Systems“ have bee reassessed from seismic aspects, and improvements of buildings are gradually implemented by setting supports to the beams of existing walls, reinforcing light walls with steel profiles and wire mesh; new pillars to were erected reinforce floors in some buildings. The upgrading of the seismic behavior of the technological equipment mainly consists in reinforcing their anchoring and in reinforcing the tanks.

The operation of NPP Mochovce was licensed based on the pre-operation safety report prepared in accordance with internationally recognized standards. The overall concept has been based on US NRC RG 1.70; for accident analysis the IAEA document „Guidelines for Accident Analysis for WWER Nuclear Power Plants„, has been used in this report, taking into account the applicable Slovak legislation. ÚJD approved, based on positive position on POSR and the checks performed by ÚJD’s inspection teams, and in accordance with the ČSKAE Decree No. 6/1980 on nuclear safety upon start-up and operation of nuclear energy facilities, the starting of the physical and power start-up. At the same time, POSR represents the basic background document for the issuance of ÚJD’s approval of the commissioning protocol (commercial operation licence).

It was stated by one of international review team that “NPP Mochove is the first power plant of the former eastern block constructed according to a Soviet project that meets the requirements put on safety of units operated in western countries”.

3. Conclusion

Modernizing an NPP is at least as challenging as building a new plant. The renewed and the remaining plant systems have to match properly in order to achieve the goal of increased nuclear safety and plant availability and finally to guarantee the integral function of the modernised plant.
M. Ziakova

Modernisation or upgrading of NPPs is all over the world a continuous process which concerns all types of plants. It has to be stated that the huge experience has been gathered in the field of technical upgrading of NPPs.

REFERENCES

OPERATING EXPERIENCE IN SUPPORT OF MANAGING CHANGES IN INDIAN NUCLEAR PLANTS

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Abstract. Feedback of operating experience is a key factor for implementation of changes in Indian nuclear plants for enhancement of plant performance and safety, improvements in plant design and implementation of changes to existing practices etc. For this, information on plant performance is collected, analyzed and reviewed in very systematic manner. This information is then utilized for evolving a change management plan. Implementation of change management activities are carried out in a professional manner. Subsequently effect of these changes are evaluated by performance monitoring and trend analysis. PSA methodology is being put to use in arriving at change management decisions. This paper focuses on the practices followed in India for use of operating experience for managing changes effectively and highlights issues of importance.

1. Introduction

In India, over a few hundred reactor-years of experience in safe operation of nuclear power plants and research reactors has been accumulated so far. The knowledge base of operating experience has been further growing with the presently 14 nuclear power plant (NPPs), 3 research reactors and a fast breeder test reactor in operation. In addition Kamini, a 30 kW research reactor, possibly the only reactor in the world having U-233 fuel has also been built with the ultimate aim of large scale utilization of thorium in power production.

As for the research reactors, while Apsara - the first research reactor built in 1956 marked the beginning of nuclear research in India, commissioning of 40 MWth Cirus in 1960 and 100 MWth high flux research reactor Dhruva in 1985, provided the thrust for Indian nuclear energy programme. Of the total 14 operating NPPs, 12 are Pressurized Heavy Water Reactors (PHWR) and 2 are BWRs. 6 more power plants based on PHWR design and 2 Russian VVERs are under construction. Construction activities are also in progress for building a 500 MWe prototype fast breeder reactor.

It can be seen that India has a very ambitious nuclear energy program. It is needless to mention that all these requires a good support from general public and the government. This support has been forthcoming because Indian nuclear industry has demonstrated that nuclear power is safe, reliable, cost-effective and environmentally benign.

A system for utilization of operating experience for managing changes has been one of the major contributors to ensure and enhance safety and availability of nuclear plants. The lessons learned based on the operating experience whether in power reactors or in research reactors provides invaluable information in respect of managing changes. Changes could be related to a) design optimization b) plant modification or back-fitting c) operating policy, d) refurbishment following ageing studies to meet current safety standards following ageing studies e) regulatory directives regarding safety significant issues etc. Major areas of work to realize the implementation of the system which effectively cater to the management of change are a) data and information collection, b) review of data and information, c) identification of changes required

and evolving a plan for change, d) implementation of corrective action programme and e) follow-up steps to track the effectiveness of the change and f) re-orienting the existing R&D programme. Change
The management process followed for Indian nuclear plants is similar to that indicated in IAEA TECDOC 1226 [1].

The governing considerations for utilization of operating experience and implementation of safety management programme at nuclear power plants and research reactors are similar. Hence this paper deals with the subject in a general manner as applicable to both. Nuclear Power Corporation of India, an independent utility organization under the Department of Atomic Energy, manages the activities related to nuclear power plants. However, most of the state-of-the-art developmental work in the area of nuclear engineering and safety research is done at BARC. Certain safety studies are being carried out at Safety Research Institute of Atomic Energy Regulatory Board. Activities related to fast reactor development program are conducted at Indira Gandhi Centre for Atomic Research.

The section 2 in this paper discusses the existing methodology for the data and information collection in support of identification / assessment of areas for change management. Multi-tier review process has been discussed in section 3. Section 4 brings out the implementation aspects. The section 5 discusses the role of PSA in change management and section 6 presents final remarks.

2. Data & information for identification and assessment of areas for change management

The plant log-books, incident reports, maintenance work permits, test & surveillance reports, reports on radiation exposure and radioactive waste, quality audit and review reports, condition monitoring reports, etc form the root source of plant data. Apart from the sources of information covered above, insight into performance of nuclear plants are obtained through specific exercises carried out from time to time for assessment of their performance. These include periodic regulatory inspection, assessment of plant performance over a given period as per guidelines given in safety guides for renewal of authorization for operation, special investigations as per specific directive of regulatory body, or review following any major developments in nuclear field internationally. For older plants ageing studies, safety analysis and fresh assessment plant design for conformance to applicable current international standards, etc. provide additional input for the change. The peer review reports from WANO or similar international organizations also form input for evolution of change decisions.

There are three aspects which are important with respect to plant data and information collection and reporting. First, the data collection activity should be able to address the objective of data utilization, and second interpretations of data & information should be such that chances of mis-interpretation are reduced to the extent possible. And third there should be provision to refine the data into the plant performance indicators, like level of safety, system reliability / availability, number of trips / shutdowns, cost estimates, profitability, etc. These requirements are implemented in an integrated manner. Based on the lessons learned over the years on the data collection and assessment, a system called Reliability Data and Information Management System (REDIMS) has been developed at BARC as part of a IAEA CRP on ‘Update and Expand Reliability Database for the Research Reactor PSA’. This is a computerized data collection and analysis system where the information / data about a) regular plant operation (daily summary reports), b) maintenance data on components (work permit records), and c) information about incidents are compiled for generating reliability estimates of components. The data analysis module in this system offers the facility for i) failure rate estimates of the components, ii) Bayesian updating of plant data, and iii) reliability prediction of electronics components employing MIL-217 methodology. The design of this system in general and the user-interface in particular reduces the chances of error during the data collection and analysis. The results produced from this system could also be directly utilized for various safety studies like Probabilistic Safety Assessment (PSA), reactor system performance analysis, trip / shutdown analysis, etc.

3. Review of plant performance and evaluation of options for change

Assuring high safety standards has been of prime importance since the inception of nuclear programme in India. Recognizing the fact that research reactors and NPPs require plant specific safety considerations multi-tier safety regimes are instituted at an early stage of design and construction of
the plant. The first review of plant performance, incidents or other related safety matter is performed at
the plant level by a committee consisting of members of site management and further at utility level by
a safety review committee. This forms the first level of review in the safety review process. The
observations and the details including the recommendations of these committees on safety significant
matters are then forwarded for review by regulatory body (RB). In the RB safety issues related to
operating plants are subject to a three-tier-review process. First stage of the review is done by a unit
safety committee appointed by RB and consisting of experienced personnel most of whom are from
organizations other than the utility. Where required issues may also be referred to Committees of
Specialists’ in specific disciplines such as Standing Committee on Control and Instrumentation,
Standing Committee on Reactor Physics, etc. In the second stage, Operating Plant Safety Review
Committee of RB, a body consisting of senior experts in the field reviews issues brought before it for
consideration by the utility or others together with the recommendations of unit safety committee
and/or specialists’ committee and makes stipulation for implementation by the utility. This body is
common for all NPPs. The final stage of review is by RB Board. The RB is independent and its Board
has the final say in safety matters and has the responsibility for enforcing safety by issue of directives.
RB is empowered to ask for any information, demand changes and to order curtailment or suspension
of operation and impose other restrictions as felt necessary in the interest of safety.

The above process of review has inherent benefits and very effective in evolution of scope for
management of change as a) review by O&M staff of the plant and other utility staff brings out the
consolidated finding with respect to plant performance and recommendations thereof, b) the next level
of review process, which involves specialists/experts outside the operating organization, allows an
independent review, and c) the apex operating plant safety review committee brings in maturity,
consistency and transparency in decision making. Information on changes carried out is widely
circulated. For the plants under design and construction, similar review committees including those
constituted by the RB ensure that changes implemented in operating plants are covered in the design
of new plants.

4. Implementation of change management

For systematic implementation of change management, activities are performed in three steps, viz.,
one, the development of the detailed implementation plan, second actual implementation of the plan
and third performance monitoring and trend analysis. Following section discusses these steps in brief.

4.1. Development of change plan

The major steps involved in development of change plan includes preparation of, a) a comprehensive
plan document, b) safety management plan, c) directives on, line of communications and
responsibilities, d) procedure for regulatory and management oversight, e) safety review strategy, f)
quality management and assurance plan, g) proposed changes in training / qualification manuals, h)
schedule for execution and monitoring the progress of work, i) contingency plan, in case there is
deviation j) and performance assessment plan. Preparation of change plan is done keeping the ultimate
objective that plant safety criteria are met during the execution of the plan. While the plant
management or the utility may plan and implement changes which do not compromise the plant safety
in any way, all safety significant changes need prior approval of regulatory body. Desired goals for the
change are decided by utility or regulatory or both. Options available for the change are normally
identified and evaluated by utility or experts and final approval of the chosen option is generally given
by the regulatory body. Availability of a change plan duly approved by plant management and
subsequently the regulatory authority is the pre-requisite for entering into implementation /
incorporation of change.

4.2. Implementation of change

The implementation of change requires a project management approach for, a) coordination of various
jobs / agencies, b) monitoring the progress of the work, c) ensuring that the risk criteria are
adhered to on continued basis d) management of financial aspects, e) compliance with the set quality
R. Chowdhury and P.V. Varde

management practices and quality control measures, f) facilitating the regulatory and management oversight, g) identifying the areas where the change requires a special provisions for transfer of responsibilities and accordingly incorporate the organizational changes and h) affecting the changes in the O&M documents in case change options deviate from the initial design, significantly. After the completion of the project, it is ensured that project objective has been met. In case of a major design modification, the commissioning tests are conducted to demonstrate that the project objective has been achieved. The transfer to operation regime is done in such a manner that the plant operating staff gets adequately trained and the regular plant O&M procedures reflect the change operating practices. In specific cases, regulatory clearance for handing over the project to plant management may also be required.

4.3. Performance monitoring and trend analysis

Following the design changes or back-fitting, the plant management carries out the tasks related to performance monitoring and trend analysis. It is ensured that all the activities are performed in accordance with the regulatory requirements. Based on the feedback generated after performance review and analysis, long term monitoring plans are drawn. Performance analysis may include safety and reliability analysis. Based on the insights obtained from performance analysis bench marking is done for various parameters, operating conditions, provision of man-power, etc.

5. PSA in change management

The Probabilistic Safety Assessment (PSA) studies are being used for taking risk informed decisions related to changes in design and operations of the plant [2]. During the design stage the PSA is performed using the generic data, however, as the plant is put into operation these studies are updated based on the plant operating experience and data. In India RB desires the submission of a Level 1 PSA study for giving clearance for construction and operations of a new plant. Along with deterministic studies, PSA results are used to check that desired level of safety is achievable. The use of PSA for assessing the component and system performance is increasingly being followed. Results of PSA are being applied in review of changes in technical specification requirements, where required. Work is in progress to draw up risk-based in-service-inspection programme for nuclear plants and other facilities. Since, PSA enables integration of human factor with the hardware model of the plant, further, research work is in progress towards the development of PSA based operator support system with an objective to enhance operational safety of the plant.

6. Final remarks

This paper brings out the implementation of change management strategy in Indian nuclear plants. Both utility and regulatory body recognizes the primacy of the safety in nuclear activities and that utilization of the feedback information on the operating plant performance is the time-proven method for implementing changes for improved plant performance and enhanced plant safety. Approach followed for plant performance data collection, analysis and review in nuclear plants is systematic and comprehensive and it results in implementation of change management process in a transparent and rigorous manner. The experience gained over the year has been vital to make this programme effective. A programme to realize change management by taking risk-informed decisions utilising PSA methodology has been initiated.

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NUCLEAR ENERGY AND SYSTEM DYNAMIC MODEL FOR ENERGY POLICY MAKERS IN THE SLOVAK REPUBLIC

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Abstract. By the new statute as member of European Union, Slovakia faces the new challenges characteristic of a transitional period for the European community, both as regards the issue of combating climate change and for the completion of the internal energy market. In this respect, based on the order received from Ministry of Economy of the Slovak Republic, CENS performed feasibility study for NPP Mochovce Units 3 and 4 in order to analyze the use of nuclear power as a viable alternative for electricity generation to meet the emission reduction targets that are set forth in the Kyoto Protocol „commitment period 2008-2012“. The particular competitive advantage of maintaining nuclear competence is important from the standpoint of long-term political, social and technological development in the enlarged EU.

1. Introduction

The commitment with Kyoto Protocol is a great opportunity for Slovakia to play a leading role in achieving CO\textsubscript{2} emission reductions in the enlarged EU, and to set an example for social and economic development, that is in turn, dependent on the energy sector, where one of the key assets is the availability of competence in nuclear technology.

By the “2008-2012 Kyoto Protocol commitment period” the EU must actually reduce emissions to at least 8% below the level of the year 1990. The Protocol contains legally binding emissions targets for Annex I countries (Slovakia included) requiring them to reduce their collective emissions of six key greenhouse gases (GHG) by at least 5.2% by the commitment period of 2008-2012. The GHG are carbon dioxide, methane, nitrous oxide, hydro fluorocarbons, per fluorocarbons and sulfur hexafluoride.

Nuclear energy makes a significant contribution to the reduction of carbon emissions from the energy sector. According to an OECD study, the expansion of nuclear powers contribution to energy supply has made an important contribution to the avoidance of CO\textsubscript{2} emissions worldwide. The study also shows regional and global CO\textsubscript{2} emissions from fossil fuel combustion in 1990 and in 1999. The Slovak Republic has reduced the CO\textsubscript{2} emissions by 28.9%, particularly due to use of nuclear power in the energy mix, even though the economic recession during the same period also has contributed to this reduction.

2. Energy Market in EU Context

- EU is in the process of creating the largest competitive, deregulated market for electricity and gas in the world.
- EU is consuming more and more energy and as a result external dependence for energy is constantly increasing. It has been estimated that an additional 300GW of generation capacity will need to be constructed over the next 15 years in the EU. This is equivalent to 750 new large power stations.
As already established in the Commission’s Green Paper on Security of Supply\(^1\), generation adequacy is also an ongoing concern for the EU. The EU 25 baseline forecast\(^2\) indicates a 44% growth in electricity consumption between 2000 and 2020.

### 3. General Overview on Situation in Slovakia

In accordance with the government decision 5/2000, the energy policy is defined by three pillars:

1. Preparing Slovakia for integration into the European Union (EU) energy market;
2. Security and quality of energy supply in the Slovak Republic;
3. Sustainable development of energy sector and of all society.

The Slovak energy sector is characterized by high level of energy intensity in comparison to the EU. Over 90% of primary energy sources are imported: coal, oil, gas, nuclear fuel [Statistical Office of the Slovak Republic].

The optimum portfolio set up of the important commodities is directly related to the security of energy supply.

As an instrumental step to the Slovak Republic’s admission to the European Union, it was approved by Decree No. 801/1999 as a realistic date for NPP Bohunice Units 1 and 2 closure the years 2006 and 2008, respectively.

The NPP Mochovce ranks among the latest nuclear units VVER 440 type V-213 and benefit from all the improvements of the project that have been progressively implemented. The current project for NPP Mochovce Units 1 and 2 combines Western safety and operating improvements with the original Soviet-era technical design, applying among other recommendation, the conclusions of the IAEA Design Review.

Units 3 and 4 of the Mochovce nuclear power complex follow up on the operated Units 1 and 2 and may utilise the already erected and commissioned auxiliary systems common to all the four units. The construction virtually represents the second half of making use of the design generation capacity of the NPP Mochovce project.

### 4. Energy Profile in 2001 in Slovakia

- **Primary energy sources** generated 813208 TJ, the gaseous fuels (mostly natural gas) have the the most significant share of primary energy sources (32%), followed by nuclear energy (25%), solid fuels (23%) and liquid fuels (15%).
- **Energy production** accounted for 552.032 TJ, the main contributor are liquid fuels with 37%, followed by heat (28%), electricity (18%), solid fuels (9%) and gaseous fuels (8%).
- **Gross electricity production** was 115.363 TJ, the major contributor being nuclear energy with 54%, followed by coal 19%, hydro 16%, gas 8%, oil 2%, renewable, small hydro and others sources with 1%.

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\(^1\) COM(2000) 769 final 29 November 2000

82
5. System Dynamic Model

5.1. General Description

In order to study the development of the Slovakian energy supply and demand a system dynamics approach was used. The behaviour of a system is modeled over a certain period of time. Different dependencies can be changed and be adjusted allowing the investigation of the impact of different scenarios on the system. It must however be stressed that such a model can only give the general trends and it can therefore be used primarily to support management decisions.

The energy system was modeled with a commercially available software (STELLA 8). Figure 1 shows a scheme of the model. The time-frame considered is 12 years.

![Flow Sheet of the system dynamic model](image)

5.2. Model Assumptions

Generally, for all variable values which can be operated from the user interface a two component approach was used. One component is a function concerning the expected behaviour of a variable over the time. This function is multiplied by an adjustable constant taking into consideration the strength of the influence of a variable on the system.
Gross national product and efficiency improvement: a linear relationship between changes in gross national product and expected energy consumption is assumed.

Environmental awareness acts in two directions: it reduces the energy demand and it changes the energy mix.

For price development of fossil, nuclear and renewables relative changes to the current situation can be made, influencing the expected energy mix.

Nuclear Power Plants Bohunice 1,2 and Mochovce 3,4 can be activated or deactivated by switches. The consequences can be seen from the energy required and (if replaced by gas/coal) by an increase in the CO2-amount.

5.3. Outputs of the model

- Energy needed, as heat and electricity.
- Excess energy, as electricity for export and heat as potential for retirement of old plants.
- CO2.
- Total Energy Demand

With respect to all assumptions for the above outputs, it must be clearly stated that these are approximations which shall only be used for the investigation of general trends. The model was primarily designed to support strategic decisions and not for detailed technical analyses.

6. Evaluation

6.1. Baseline Investigations

For baseline investigations only changes in the gross national product (GNP) and changes in the expected energy efficiency were considered over a period of 12 years. These changes are discussed with respect to different scenarios of nuclear power.

For GNP and efficiency expectations a zero scenario (no changes over the next 12 years), a moderate scenario (intensity level 3 from 10) and a maximum scenario (intensity level 10) were considered. For nuclear power the baseline assumption was that Bohunice 1,2 will be switched off in 4 years and Mochovce 3,4 will be switched on at the same time. As alternatives the options Bohunice 1,2 off and Mochovce 3,4 not on as well as both on were considered. In order to take steps as a result of switching plants on-off properly into consideration for some scenarios also the values after 3 years were reported.

6.1. General Interpretations and Conclusions

- Switching off Bohunice 1,2 and not switching on Mochovce 3,4 leads to an energy need which must be counterbalanced either by investments in other power plants or by import of energy. As far as domestic production is concerned even assuming 50 % CHP production leads to partly remarkable additional CO2-production which might infringe on a long term basis with the Kyoto goals particularly when not the pure CO2 but the CO2-equivalent level is considered. Assuming additional CO2-emissions from increasing traffic real problems can be expected in such a case.
- If both nuclear power options are maintained then it becomes obvious that electric excess energy is produced which could be exported. This export option remains still for the situation that a strong growth of the gross national product is assumed without any increase in energy efficiency.
- Between these extremes is the scenario that Bohunice 1,2 off is balanced by Mochovce 3,4 on which has a more moderate but still not negligible potential of CO2 increase and which does not provide significant export possibilities if a certain economic growth is assumed.
Increasing energy efficiency can compensate partly or in extreme completely the economic growth but it is not very likely to assume that such an extreme efficiency improvement will happen within the envisaged short period of time (12 years).

The systematic approach presented in this study by modelling the behaviour of Slovak market on the basis of system dynamics (Forrester), enables real energy mix in the extent of 12 years and helps understanding of the market limits. In addition it enables to prepare for the possible trends in the economic development of the Slovak Republic.

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LESSONS LEARNED IN CONFIGURATION MANAGEMENT OF NPPS WITH WWER440

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Abstract. The paper presents the EOOS (Equipment Out Of Service) risk monitor developed for the J.Bohunice nuclear power plant. The EOOS software is part of the Risk and Reliability (R&R) Workstation, developed by EPRI. The R&R Workstation is a suite of tools created to support risk and reliability engineers. The monitor is based on the level 1 full power and shutdown PSA (Probabilistic safety Assessments) model, developed for the plant. The risk is presented in the form of core damage frequency (CDF).

1. Introduction

PSAs are developed at nuclear power plants for a variety of reasons, but generally they are used to evaluate the level of safety at a plant. Additional applications beside detection of weaknesses, design improvement and backfitting are support of plant operation, incident analysis, accident management, emergency preparedness and support of regulatory activities. The plant configuration management in form of risk monitor is one of the most important PSA application.

Four units with WWER440 are operating in the J.Bohunice NPP (2 units with V230 type reactors and 2 units with V213 type reactors). Level 1 and 2 PSAs are available for full power and shutdown operating modes for these units. In addition the EOOS (Equipment Out Of Service) risk monitor is developed for the units based on the level 1 PSA. A risk monitor, also known as a safety monitor, is an on-line, plant-specific, real-time analysis tool used to determine the instantaneous plant risk based on actual status of systems and components. It is based on the plant’s current as-built, operated, and maintained configuration and reflects component statuses (e.g., any components out-of-service for maintenance or test). The risk monitor model is consistent with the plant’s living PSA, and it is updated along with the PSA.

The risk monitor provides the ability to easily examine past operating practices, current plant conditions, and future planned activities (e.g., maintenance schedules). This is accomplished by providing an easy-to-use interface that is consistent with terminology and nomenclature familiar to plant staff. The risk monitor makes the PSA model and resulting insights available to non-PSA analysts (i.e., it “main-streams” the PSA to those actually operating and maintaining the facility). The result is a plant staff who is risk aware and more thoroughly considers impacts to safety from its past, current, and planned activities.

The technology and application of risk monitor is not new. Development of the technology began in the early 1980s, and the first risk monitor was installed at Heysham Nuclear Power Plant in 1987 in USA. Further technology development and refinement eventually led to two commercially available risk monitors (EOOS from EPRI and Safety Monitor from Scientech) that began to be installed in U.S. plants in the mid-1990s. At present, risk monitors are installed at over 100 plant sites in at least 8 countries. In this presentation application of EOOS and lessons learned from it in the J. Bohunice plant is described.
2. Implementation of EOOS in the J. Bohunice plant

The Bohunice plant uses its primary support organization, RELKO Ltd, to maintain and update its PSA models. To accomplish this task, RELKO uses Risk Spectrum PSA Professional, a Windows-based PSA software. The plant also hired RELKO to convert and maintain the PSA model that the EOOS risk monitor is based on. The current PSA models constitute a level 1 and 2 PSA for all plant operating modes (full power to full shutdown and defueled). The PSA and EOOS models were recently updated to account for the latest plant upgrades and implementation of symptom-based emergency operating procedures (EOPs).

A dedicated PSA team at the Bohunice plant is tasked with responsibility for PSA activities at the plant. This team is responsible for using the EOOS to examine the plant’s activities on a monthly basis and with coordinating a more extensive implementation of the EOOS in other departments. Activities are concentrated on the control room (where a version has been implemented for operator) and with the maintenance team. The EOOS has a screen for operators and has a scheduler mode where maintenance planners can examine the impact of their planned activities during operations and shutdown conditions on plant risk.

The PSA team also is tasked with keeping RELKO informed of changes in plant configuration and operations activities that constitute needs to change the PSA and EOOS models. Staff member from RELKO, the PSA team, and the regulatory agency are examining several changes to the plant’s technical specifications regarding AOTs (Allowed Outage Time).

3. Description of EOOS

An integrated level 1 full power and shutdown PSA model is the basis of the EOOS. The risk monitor has two screens: screen for the operator and screen for the scheduler.

3.1. The operator’s job

Operators make decisions about when to perform tests and maintenance tasks over a period of hours or days. These activities affect plant safety. For example, operators may disable a safety system for a short time so that workers can safely gain access to the equipment. Several administrative constraints prevent operators from performing too many tasks at the same time. Faced with these constraints, operators use their detailed knowledge of plant systems to decide which combinations of work activities to avoid.

An operator’s job is complicated by the need to accommodate unscheduled events. Equipment sometimes breaks down. Support system alignments change. The working environment sometimes changes (e.g., bad weather). Scheduled activities finish early or late. The combinations of scheduled and unscheduled events require operators to constantly reevaluate plant safety. Another factor influencing operators is the utility’s desire to minimize plant downtime. This operating objective leads to two types of decisions which sometimes conflict with one another. On one hand, utilities want to maximize system availability during power operation, in order to minimize the chance of a plant transient or forced outage. On the other hand, utilities also want to minimize the duration of scheduled plant outages to reduce expenditures on replacement power. Operators are being asked with increasing frequency to shift test and maintenance tasks from scheduled outages to power operation. As a result, operators must find the balance between plant safety and economics. EOOS helps operators focus on safety. The combined effect of many, simultaneous work activities occasionally has an unexpected impact on front-line safety systems. With each new task, operators make a complex decision to act based on their perception of how it affects plant safety. The EOOS plant risk monitor screen helps operators make these decisions by:

- Showing the core damage frequency
- Showing the status of plant systems affected by various test and maintenance activities (providing measures of “defense-in-depth”).
- Showing a list of current activities that affect plant equipment.
- Showing ranked lists of the most important in-service and out-of-service items.
- Quickly recalculating these safety measures for a variety of “what-if” tests.

Example of the screen is presented in figure 3.1. The monitor shows the system status in varying colours, for example green when all trains of a system or safety function are available, orange when one or more trains unavailable and red when all trains are unavailable.

![Figure 3.1 The screen for the operator](image)

3.2. The scheduler’s screen

The EOOS scheduler’s screen shows how plant operations affect safety over a period of time. A typical user would be a plant scheduler who makes decisions about when to perform testing and maintenance on plant equipment over periods of several weeks or months.

The work involved in scheduling of maintenance activities is complicated. Schedulers often use a computer program to harmonize high number of work orders. The computer program helps schedulers identify critical path activities and monitor the demands for critical resources. These are the standard analytic tools for project management. EOOS helps schedulers focus on safety. The combined effect of many simultaneous maintenance activities occasionally has an unexpected impact on front-line safety systems. To avoid this, schedulers spend a good deal of time performing safety reviews. EOOS helps schedulers perform these safety reviews by:

- Generating timelines showing the changing status of plant systems and safety functions.
- Generating a timeline for a plant risk measure.
- Identifying the specific equipment and activities that have the strongest influence on safety.
The screen for scheduler is divided into three parts: schedule, status and risk profile card. The schedule chart shows activity names and timeline bars representing the start and finish dates for each task. The status chart shows availability of the components. The risk profile shows the changes in the risk level. In the VVER440 type plants the preventive maintenance of the safety systems is performed only during reactor shutdown. An example is provided for planning the maintenance activities. The risk profile is given in Figure 3.2. The planning is performed in such a way that the risk level is only in the green area and the yellow area with increased risk is not achieved.

### 4. Conclusions

The risk monitor in the J.Bohunice NPP is used to determine the instantaneous plant risk, calculate AOT, plot the operational risk versus time, etc. The monitor is a good tool also for the plant maintenance schedulers. The plant has programs to maintain the currency of their PSA model and risk monitor. The PSA and risk monitor models were recently updated to account for the latest plant upgrades and implementation of symptom-based emergency operating procedures.

The J.Bohunice plant has made significant efforts to ensure appropriate implementation of EOOS at the facilities. Two important efforts involved defining the proper scopes and updating the PSA modeling approaches to support the monitor at the plant. The plant has taken steps to ensure its models reflect current plant conditions. The close relationships among the support organization, the plant staff, and the regulators ensure both regular updates to the PSA models and a thorough understanding of model limitations when time lags occur between feature (design, operational, maintenance, etc.) changes and model updates.

While excellent tools for mainstreaming PSA information into operations, maintenance, and training activities, the risk monitor at a nuclear power plant should not be used to drive a PSA program into a proper scope or living PSA. Rather, the risk monitor at a plant should be the culmination of PSA efforts after these criteria are already met. PSA activities should concentrate on efforts to expand the PSA scopes, establish the PSA pedigree, and on improving the documentation and models needed to develop and implement living PSAs. These efforts will provide the foundation for mature and constructive use of PSA applications, including eventual implementation of risk monitors at plants.

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DEVELOPMENT OF DATA BASE SYSTEM FOR THE INSPECTORS OF NUCLEAR INSTALLATIONS

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Abstract. In accordance with the Japanese law, Nuclear and Industrial Safety Agency (NISA) verifies the safety of nuclear installations. NISA’s inspectors for safety management of nuclear installations are stationed at nuclear installations throughout the country to survey installations such as the main control room and turbine generator on a daily basis and, verify the operating status of each facility. From October 1st, 2003, Japan Nuclear Energy Safety organization (JNES) also started to perform part of inspections shown above basing on share with NISA. Now, under the above-shown inspections, a lot of significant data continue to be produced, but those data exist individually and have not been efficiently utilized in the nuclear regulation. NISA and JNES are, therefore, developing the data base system for the inspectors of nuclear installations. The first data base system started the development to assist the inspectors in 2001, and now, under servicing and improving. The second data base system is being developed, and will store most of sources including inspection reports, government’s evaluations. The data will be able to be transmitted to the first system through network.

1. Background

In accordance with the Japanese law, Nuclear and Industrial Safety Agency (NISA) verifies the safety of nuclear installations by conducting a 'pre-service inspection' to confirm that construction is carried out in line with approved design and construction methods and meets the required technological standards. Besides, NISA conducts 'periodical facility inspections' periodically every year to verify that the performance of a nuclear installation is meeting these standards.

Additionally, NISA’s inspectors for safety management of nuclear installations are stationed at nuclear installations throughout the country to survey installations such as the main control room and turbine generator on a daily basis and verify the operating status of each facility.

They also conduct 'nuclear safety inspections' four times a year to ensure that plant operators are observing the safety preservation and operating in accordance with them.

Since October 1st, 2003, Japan Nuclear Energy Safety organization (JNES) started to perform part of above-mentioned inspections basing on share with NISA.

Now, under the performance of above-mentioned inspections, a lot of significant data is continuing to be produced, but those data exist individually and have not been efficiently utilized in the nuclear regulation. NISA and JNES are, therefore, developing the data base system for the inspectors of nuclear installations, in order to assist for their job activity utilizing the above-mentioned data.

2. Explanation

The first data base system was started the development to assist the inspectors of nuclear safety inspection in 2001, and now, under servicing and improving.
Inspectors access the database system through LAN and Internet at the local office or nuclear facility as shown below figure. In that case, high security level is established using Firewall and identification gate with a password that is given to each inspector.

For Inspectors, the safety management rule is basis for their job activities. And then, they can see the rule that established for each nuclear facility, on the database system. The system can also supply interpretations of rule, standard technical specification, safety inspection manual / reports and other significant informations for Inspectors.

If Inspectors find some notice that is nonconformance, trouble or good-practice during their job activities, they can register those notices and be mutually informed all the notices on the database system.

Nonconformance information that found by Inspector (if happened) is more important when the inspection procedures are planned by Inspectors for next nuclear safety inspection term. That information can also be supplied on the database system.

The database system can contribute for the regulatory activity assisting Inspectors on nuclear safety inspection, surveying installations on a daily basis and verifying the operating status of each facility.

The second database system is being developed, and is going to store most of documents which issued for the above-mentioned inspections, including inspection reports, government’s evaluations, as electronic data. Those data are input as text type data producing documents on the system, and as PDF data.

The data are, of course, able to be searched with any preferred condition such as time, facility and / or category of document, and obtained on personal computer by Inspectors and the other approved personnel.

The database system also is planned to be added the data processing function basing on the risk informed regulation. The more effective method of data processing is now surveying. After completion of the surveying, the development of data processing system will be proceeded.

This second database system can also contribute to assist for Inspectors and others on the nuclear regulatory activities.

Two above-mentioned database systems will be linked through network, therefore, the data enable to be mutually transmitted between both systems.

3. Conclusion

The above-mentioned database systems have started to contribute the assistance of Inspectors efficiently utilizing the significant data that produced in the above-mentioned inspections under the nuclear regulation.
SAFETY MANAGEMENT IN INDUSTRIAL REPROCESSING: LESSONS LEARNED FROM COGEMA'S EXPERIENCES

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Abstract. Today every responsible industry strives to carry out its activity following the lines of sustainable development. Consequently, protection of workers and the natural environment, along with other determining factors such as economic and social impacts, must be taken into consideration within an overall industrial development plan. This applies in particular to the nuclear industry. Specifically, the different operations implied by spent fuel management must ensure, in all circumstances, a minimisation of radiological impact both on humans and surrounding environment.

COGEMA has over several decades developed and continuously aimed to optimise the technical and organisational features necessary to ensure protection of its personnel, the public and the environment.

1. Introduction

The objective of this paper is to describe the way safety is respected and risks are managed in COGEMA's fuel cycle installations, in particular in its reprocessing La Hague plant.

2. Risk management and "defence-in-depth" concept

The nuclear safety covers all the measures taken to protect human and their environment, during all the phases of the life of a facility: R&D, design, construction, operation, cleaning and dismantling. In a reprocessing plant, the safety purpose is to limit in all cases at the lowest level the spreading of radioactive materials in the environment and the external exposure of workers.

Series of physical and functional barriers are placed between radioactive materials and workers and the environment. The choice and the implementation of these barriers are based upon a concept of "defence-in-depth".

This consists of:
- aiming to avoid incidents and accidents by considering all possible weaknesses,
- closely monitoring the installations in order to detect and subsequently correct potential changes on the functional main domain,
- limiting the consequences for potential incidents which may occur despite precautions taken.

With the objective of reaching the excellence in nuclear safety, we have taken advantage of the past experience.

The implementation of these principles (experience, continuous improvement) are illustrated below, in Section 2.1, for some significant risks.

In addition, the implementation of specific organisations (dependent on reliable and trained people), as well as technical requirements (notably for the design of a facility), participates in the risk management. COGEMA's organisation for the operation of the La Hague plant is covered in Section 2.2.
To manage the risk of dissemination of radioactive materials the reprocessing facilities are organized in containment systems. The first containment system is composed with process equipments (tanks, pipes), cells and enclosures and is completed with a forced ventilation which maintains a depression between the equipments, cells, enclosures and the adjacent rooms. The second containment system adjacent to the first one is usually made of the rooms surrounding the active cells. It is completed with a ventilation which maintains an airstream from the less to the more risky areas. Numerous filters are placed along the ventilation networks; the number of filter stages depends on the activity in the equipments and cells.

The feedback experience of the operation of the La Hague facilities allowed the optimisation of the ventilation of the first containment system:

- the ventilation of equipments and cells is realised with separated networks,
- the reliability of the air-extracting networks (fans, filters) has been increased.

Important progress was also made regarding the choice of material for the welded process equipments: the choice depends on characteristics of the fluid and on the operating conditions such as temperature, acidity, deposit risk. For example, COGEMA has manufactured welded equipments in different materials such as zirconium, titanium, used in the UP3 plant (in operation since the 1990s).

In order to minimise the workers’ risk of external exposure, the working stations are located far from the radioactive materials. In addition, radiological protection is added or the exposure time is limited.

The automation of working stations, the remote control of process, the use of manipulators, etc, limit the level and the time of exposure. Taking into account the maintenance constraints as early as the design of facilities also aids to reduce the exposure.

The design of the glove boxes in new facilities has been improved with the feedback experience from UP3 facilities: materials such as motors, electrical cabinets, valves, necessary to run the boxes were not put in the same cell as the box but in an adjacent room, whenever possible. And in all cases, motors were located outside the glove boxes. These measures allowed the reduction of the external exposure during the maintenance of the glove boxes.

In reprocessing facilities the criticality is prevented by:

- a limited mass of fissile material,
- or a geometry which favours the escape of neutrons,
- or the absence or the limitation of a moderator such as water or hydrogenated products,
- or neutron poisoning,
- or a limited concentration of fissile material.

The previous control items may be used alone or combined.

The increase of initial U235 enrichment in spent fuel beyong the initial values used for designing the La Hague plant, allowed the optimisation of criticality constraints, by taking into account the realistic wear of the fuel. A significant R&D program - "burn-up credit" - was initiated in the 1990s in France. Its final aim is to take into account the fission products in the criticality calculation codes. Hence it will be possible to optimise both design and operating conditions for equipment such as rotary dissolver, for example, by eliminating or decreasing the gadolinium poisoning.

Overheating of nuclear materials may lead to a dissemination of these materials or to damages caused to the building or equipment. The implementation of cooling systems allows the prevention of this risk.

For example, solutions of concentrated plutonium nitrate are put in specific tanks. Their design permits the prevention of both criticality risk and overheating risk. This kind of concentrated solutions are thus usually (in UP3 plant in operation since 1990) put in annular storage tanks whose cooling is made with an internal cooling coil and external air circulation.
In order to reduce the tank size and improve the cooling system, COGEMA innovated and designed a new concept of tanks for the concentrated solutions of plutonium nitrate: tube bundle tanks with a subcritical geometry, twice as compact as conventional annular tanks, enabling easy homogenization of batch preparation and accountability, which are failproof in case of cooling system shutdown. These tanks were implemented in a new facility (R4), in operation since 2002.

- To prevent the risk of fire, the quantities of flammable materials and lighting sources are limited. The limitation of consequences in case of fire depends on:
  - the division into sectors of the high-risk rooms and of the rooms to protect, completed with specific ventilation,
  - the setting up of detection devices,
  - the setting up of movable or fixed extinction devices,
  - the implementation of access routes for fire fighters.

In newer facilities in operation for few years, the design of the corridors has been improved, in order to mitigate the consequences of a potential fire (which may occur despite precautions taken). The corridors are divided into two compartments: the first one for personnel and the second one for cables routing. To avoid smoke being transferred to the first corridor in case of fire, the two compartments have their own ventilation and the cables compartment has low pressure.

2.2. COGEMA’s organization for the operation of the La Hague plant

The safety requirements result from the COGEMA's safety analyses, which are reviewed then validated by the French Safety Authorities. These safety requirements are translated in the unit operating instructions, in the maintenance instructions and in the modification procedures.

An operating range is defined which respects the safety requirements, called the "prescribed range". Inside this range, security margins are taken, whenever possible. This more restrictive range is called the "authorized range".

Risk prevention and risk management are an integral part of our company culture. A high level of vigilance and thoroughness are maintained at all times with employees deeply involved in the effort, at every operational and functional level.

In the La Hague plant, the safety culture is the result of a safety policy which consists of:

- Maintaining a high level of safety:
  - by an efficient organization,
    - that complies with safety rules thus making controls unnecessary, as well as complying with objectives concerning staff radiation protection and environmental impact,
    - where every intervener is aware of his responsibility and the implication of his activity on the plant safety,
    - based on prevention by aiming for a zero serious incident rate,
  - by the automatic analysis of any malfunctions likely to have an impact on plant safety and carrying out improvements,
  - by strengthening the safety culture of all participants,

- Communicating and encouraging a better understanding of the hazards linked to the activities performed and their control, notably by informing the staff and the general public of any event classified in accordance with the international nuclear event scale.

One of the main criteria to define the organisation of the plant is the safety culture. The head of the La Hague plant is responsible of the nuclear safety. He is assisted by the supervisors of each facility to whom he delegates the responsibility to maintain safety conditions, and by safety assistants and by criticality specialists.
In a functional level, the Department of Quality Security, Safety and Environment provides control, support and assessment for operators, as well as interface with the external organisations intervening as regards safety.

Operationally, there is a double scrutiny of the TSQ, Chief Shift Leader, and the ISE, Safety and Operating Engineer which ensures that the continuous operation shift team activity is monitored.

3. Results and perspectives

These features allow COGEMA to run La Hague facilities under satisfactory safety conditions:

- Since the implementation in France in 1994 of the INES scale of classification of nuclear events, no level 2 \(^1\) event has been noted.
- The radiological impact of the release of COGEMA La Hague plant on the reference groups of inhabitants was less than 0.01 mSv in 2003, much below the European Union limit for members of public of 1 mSv per person per year.
- The average dose from radiation exposure for workers was less than 0.071 mSv in 2003, well below the European Union limit of 20 mSv per person per year.

Numerous improvements have been implemented through a continuous analysis of operation experience.

During the same time, operating range limits were broadened, and quantities of effluents and waste were reduced.

Two examples which illustrate in particular the progress which has been achieved are the evolution of environmental impact and the annual dose on workers.

The radiological impact on local population demonstrates the efforts realised by COGEMA to reduce the environment impact of its activities. The following figure illustrates the significant decrease of impact of liquid release.

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\(^1\) The International Nuclear Event Scale (INES) has eight levels, beginning with level 0, which has no safety incidence, through level 7, which corresponds to a major accident with substantial health and environmental impacts. The level 1 corresponds to an event which lead to operate outside the "authorized range" (defined in section 2.2)
Taking radiation protection into account at the design stage along with a strong programme of optimisation of maintenance translates into continuous reductions in personnel exposure over the years. This is illustrated on the following figure.

Moreover, other improvements are planned for the next few years mainly relating to reduction of waste and release.

In that field, in partnership with the CEA, COGEMA continues to improve the knowledge regarding some radioelements such as C14, I129 and Ru106. The purposes of the R&D actions in progress are to better understand the behaviour of these elements in the reprocessing process and to improve analytical methods.

Other actions concern cleaning and dismantling:
- development of decontamination processes with two targets: reducing the waste quantities and limiting effluents;
- development of dismantling tools to downgrade the waste and to reduce the personnel exposure to radioactive materials.

In conclusion, we can summarise the key factors required to assure the implementation of future fuel cycle facilities in safe and secure conditions: appropriate technical and organisational dispositions are fundamental. After thirty years experience in the design and operation of reprocessing facilities, COGEMA has acquired significant experience and a genuine safety culture which are essential to the success of new industrial projects with cost effectiveness and risk reduction.
TOPICAL ISSUE 3:
REGULATORY MANAGEMENT SYSTEMS – ADAPTING TO CHANGES IN THE ENVIRONMENT
SAFETY ASSESSMENT METHODOLOGY FOR NORM DISPOSAL TRENCH AT ABU REDEES SITE

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Abstract. There are a number of industries generating NORM wastes in Egypt. These include oil and gas and minerals/ores processing industries. A safe management of radioactive wastes is required. The petroleum industry adopted methods for managing and disposing of NORM-contaminated wastes that are more restrictive than past practices and are likely to provide greater isolation of the radioactivity. A trench was used as a disposal facility for NORM waste at Abo Redeess site. The aim of this work is to prepare safety assessment for this site after direct closure and after 500 years. RESRAD computer code with two different scenario was used for this purpose. Total effective dose received after direct closure and after 500 years was 1.7 E-6 and 4 E-5 mSv/y respectively, while the health risk was 3E-8 and 6E-8. This study provides evidence that trench disposal of NORM waste poses a very low human health risk.

1. Introduction

Oil and gas production and processing operations sometimes accumulate NORM at elevated concentrations in by-product waste streams. The sources of most of the radioactivity are isotopes of uranium-238 (U-238) and thorium-232 (Th-232), which are naturally present in subsurface formations from which oil and gas are produced. The primary radionuclides of concern in NORM wastes are radium-226 (Ra-226) of the U-238 decay series and radium-228 (Ra-228) of the Th-232 decay series. Other radionuclides of concern include radionuclides that form from the decay of Ra-226 and Ra-228, such as radon-222 (Rn-222). The production waste streams most likely to be contaminated by elevated radium concentrations include produced water, scale, and sludge\(^{[1]}\). Radium, which is slightly soluble, can be mobilized in the liquid phases of a formation and transported to the surface in the produced water stream. Dissolved radium either remains in solution in the produced water or precipitates out in scales or sludge. Conditions that appear to affect radium solubility and precipitation include water chemistry (primarily salinity), temperature, and pressure.

2. Background

NORM contamination of scale and sludge can occur when dissolved radium Co-precipitates with other alkaline earth elements such as barium, strontium, or calcium. In the case of scale, the radium co-precipitates, primarily with barium, to form hard, insoluble sulfate deposits. Scale typically forms on the inside of piping, filters, injection wellhead equipment, and other water handling equipment, but also can form as a coating on produced sand grains. In the case of sludge, radium can be present in several forms. It can co-precipitate with silicates and carbonates that form in the sludge, or it can be present in pieces of barium sulfate scale that become incorporated into the sludge. NORM-contaminated sludges can accumulate inside piping, separators, heater/treaters, storage tanks, and any other equipment where produced water is handled. The EPA estimates that approximately 25,000 tons of NORM-contaminated scale and 225,000 tons of NORM-contaminated sludge are generated annually by the petroleum industry\(^{[2]}\).

NORM waste is physically and chemically similar to nonhazardous oil field waste (NOW). Its primary difference from NOW is the presence of radionuclides in NORM waste. The presence of radionuclides...
may require additional safety precautions when handling the NORM waste, but the actual disposal process would be no different from that for NOW\textsuperscript{(3)}.

The safe handling of these NORM in Egypt implies identification of the responsibilities of both the producers of the NORM and the Central Radioactive Management Authority. In Egypt, this authority is the Hot Laboratory and Waste Management Centre (HLWMC)\textsuperscript{(4)}. Dealing with these wastes requires developing both the required technologies and the relevant regulations to determine the responsibilities and identify the safety requirements for the handling of such wastes. The responsibilities of the producer include waste collection, packaging of category 1 and interim storage of category 2. The responsibilities of the HLWMC include transportation and long term storage of category 1.

3. NORM Management Practices

NORM was not recognized as a waste management issue, however, until the mid-1980s, when the industry and regulators realized that NORM occurrence was more widespread than originally thought and that activity levels could be high. The petroleum industry adopted methods for managing and disposing of NORM-contaminated wastes that are more restrictive than past practices and are likely to provide greater isolation of the radioactivity. The largest volume oil and gas waste stream that contains NORM is produced water.

3.1. Exposure Assessment

In this study, exposed individuals are expected to be those drinking groundwater contaminated by releases of NORM constituents from disposal facilities containing NORM wastes. The exposure pathway would consist of release from the facility, transport through groundwater, and human exposure through ingestion of the contaminated groundwater. This section describes the scenarios and mechanisms that could lead to human exposure to NORM constituents and estimates radiological doses and human health risk to a potential receptor. Once the facility was full of waste, it would be covered and abandoned. At the time of sealing, the facility would be mostly filled with solids and semi solids that were not fully compacted. When risks to the public from disposing of NORM waste in trench are being assessed, potential release modes must be determined.

3.2. RESRAD Computer Code

RESRAD computer code version 6.21 was developed under the joint sponsorship of the U.S. Department of Energy and the U.S. Nuclear Regulatory Commission for site-specific dose assessment of residual radioactivity.

Assuming

A 70 Kg weight man drinks 2 liters of ground water from a well located 5000 m distance a way from the disposal site.

The assumptions and the parameters used in this analysis depend on:-

1- disposal site characterization
2- types of NORM waste arising
3- design of the disposal trench
4- the hydrology and geohydrology of the site
5- the worst case; there is no unsaturated zone
6- the cover and barrier is completely failed (no barrier exist)
The input data and the assumptions are:

- Porosity : 0.4
- Precipitation : 1 m/y
- Irrigation : 0.2 m/y
- Density : 1.5 g/cm$^3$
- Hydraulic conductivity : 100 m/y
- Hydraulic gradient : 0.02

### 3.3. Results and Discussions

Fig. 1 shows a relation between the total effective dose (mSv) with the time (years) that received as a result of direct exposure. From this figure it's clear that, in the case of $^{232}$Th, the dose increases with increasing time till it reached to 50 years it levels off, while for $^{238}$U it's directly proportional with time.

The dose received from drinking two liters of ground water from a well located 5 km distance away from the trench disposal site are presented in Fig. 2. This figure shows that $^{238}$U is more effective than $^{232}$Th in the contamination of water well.

Fig. 3 shows a relation between a human risk against time and this fig. indicated that trench disposal of NORM waste poses a very low human health risk.

It is clear from the radioactive dose assessment results that the total annual exposure dose to the whole body is less than the limit; 0.25 mSv (10 CFR 61). This means that trench disposal is safe to dispose the NORM waste.
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IRANIAN NUCLEAR REGULATORY AUTHORITY’S CHALLENGES IN LICENSING OF BUSHEHR NPP

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Abstract. Bushehr is a VVER-1000 nuclear power plant under construction after 15 years of interruption. It was initially designed and partly constructed by German KWU in the mid 70s. The contract for completion of the plant was signed with Russians in 1995. This change of design faced the Iranian Nuclear Regulatory Authority (INRA) with challenges in Licensing and supervision of this hybrid project.

1. Introduction

Bushehr NPP is by far the most complex delayed NPP project. The problems facing the completion of a hybrid NPP are unique and complex. Nevertheless they can be categorized under the following headings:

- Contractual Matters
- Licensing issues
- Integration of old systems with new ones
- Technical issues related to modifying civil structures to accommodate new systems.
- Requalification of useable components and systems.
- Seismic calculations related to substantially altered dynamic loading of buildings and structures
- etc.

The focus of this paper will be on the licensing issues of Bushehr NPP and the approach adopted by the Iranian Regulatory Body to overcome the unique licensing and supervisory difficulties of this project.

2. INRA’S Challenges in Licensing of Bushehr NPP

2.1 Legal and organizational change

The issue of licensing delayed and hybrid nuclear power projects is a topic of great interest and complexity to many regulators. First of all, INRA realized that its legal status should be upgraded and its organization modified in order to be able to cope with the issue. The Iranian Atomic Energy Act of 1974 was not able to provide the necessary legal framework for licensing and supervision of the project. Hence it was upgraded by the assistance of IAEA. At the same time the organization of INRA was modified by recruiting experts familiar with various aspects of nuclear power projects, independent budget was allocated to INRA by the Government.

Secondly, INRA’s outdated nuclear codes and regulations were upgraded based on the IAEA codes and Guides.
F. Dastjerdi

At the stage of contract negotiations with Russians, INRA played a major role in developing the Appendix M to the Contract which creates the basis for nuclear codes and standards governing the NPP project.

2.2. Technical Challenges

The technical challenges INRA faced towards licensing of Bushehr project cover a wide spectrum. Some of which are outlined here:

- **Mix of design and technology**
  To cope with licensing problems arising from mixing of design, INRA used a comprehensive analysis format and a systematic guidance for the review (RG-1.70 and NUREG-800 SRP). It required the designer to perform a detailed PSA that will be reviewed by INRA prior to issuance of the operating license. INRA also avoided as much as possible mixing of standards and requirements, and adopted a single set of General Design Criteria and safety guides.

- **Modified layout of Russian Design**
  The impact of the new layout, specially in reactor building, on safe operation of the plant in normal operation and accident condition has been evaluated.

- **Ageing and/or deterioration of installed old components**
  A systematic review and inspection of components and an increased testing program is envisaged for prenuclear phase.

- **Interface between old and new structures and equipment.**
  A detailed evaluation is being performed for:
  - Seismic design of the containment.
  - Analysis (seismic, high energy line breaks) of connected mechanical components.
  - Analysis of electrical systems loading.

3. Conclusion:

Reactivating a dormant NPP project requires much efforts by the owner and Regulator of the plant. INRA has enhanced its structural and technical capabilities to cope with ever increasing challenges of licensing such a plant. In fulfilling this job, the assistance received from the IAEA is highly appreciated.
COMMUNICATING RISK TO THE PUBLIC

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Abstract. The risks of nuclear power, from the public’s perspective, are frightening for two reasons: its technology is unknown to them and this energy source can negatively destroy what is most valuable in their lives – their home, family, health and security – what they protect the most. Thus, the public’s support for commercial nuclear power cannot be achieved unless nuclear communicators are able to explain the technological events in an understandable manner and allay the public’s fears. There are principles and techniques for nuclear communications which can assist in addressing these goals and in building communication with the public.

1. Introduction

Risk. WHAT IS RISK? As in any communication situation, unless you understand or know how the audience defines the subject, there will not and cannot be successful communication.

2. Public’s definition of risk

In the commercial nuclear arena, the public defines risk in terms of what will impact on their family, their home, their health, their security. Security in the sense of their ability to live their daily lives free of fear and without the loss of control.\footnote{International Atomic Energy Agency, \textit{Communications on nuclear, radiation, transport and waste safety: a practical handbook}, IAEA-TECDOC-1076, (Vienna, Austria: IAEA, 1999) 29.} \footnote{United States Nuclear Regulatory Commission, \textit{Communicating Risk Issues to the Public}, (1992) 1} Think of “public” as any one not in the nuclear field – politicians, activists, news media and residents near a facility.

3. Why communicate risks and events to the public?

Do they need to know the risks? Do they need to know about events at nuclear facilities? Nuclear communicators need to build a link over which they can constantly communicate with the public.

“…In order to develop trust and understanding... [with]...audiences, communication obviously needs to be open and honest, but the development of such a relationship also depends on regular and consistent communication...but the communication in ...‘crisis’ situations is likely to be much more effective if a relationship has already been established through regular and consistent communications.”\footnote{IAEA 3}

The Hungarian Atomic Energy Authority certainly had the challenge of communicating risk issues to the public as a result of a recent event at the Paks Unit 2 facility.\footnote{On April 10, 2003, there was an incident in the cleaning tank used for chemical cleaning of the fuel assemblies placed temporarily in a pit in the reactor hall. Incident was classified as 3 on the INES scale.} Here are some comments regarding lessons learned from the event: “From the public opinion poll and from speaking with colleagues, friends, neighbors and journalists, we learned that it could have been better from the very beginning if
we – both the plant and the authority – communicate every tiny event independently of its relevance to the incident otherwise [we] could be blamed for hiding important facts.”

In the U.S. nuclear industry, there is a long history of reporting minor events by the regulator and utility and this has helped significantly in developing communicative channels.

4. Communications with the public

Communicators must break through the concrete and steel walls of nuclear technology and reach into the core of the human heart: family, home, health and daily survival, in other words, satisfying basic human needs. The public’s definition of risk must be directly addressed.

The public does not want to become immersed in the details of probabilistic safety criteria, risk-informed decision making or configuration management. Not only are these technical and scientific models confusing to the non-technical person, using such methods and models as explanations are perceived by the public as a way to avoid addressing their real issues. All too often, the technical communicator falls into the trap of not communicating in terms the public can understand.

However, whether people live near a commercial nuclear facility or not, they want to know that the risks from such facilities are low. An event from these facilities can affect all of the population of the world.

5. Nuclear communications principles and techniques

How can the risks from commercial nuclear power be communicated without using scientific or technical jargon? By directly addressing the public’s concerns non-technically with clear specifics and action statements either verbally at public meetings or in supporting written documents. Communications and technical experts working together on inspection and activity reports or other technical documents can ensure the important balance of technical accuracy and being open to the public. The event or risk can be presented non-technically in its proper context. The cover page, summary section and various parts of reports can be made less technical or have explanatory notations added to facilitate the communications bridge.

Why is having inspection reports or reports of activities especially important? Because it demonstrates action. Something was done. Inspectors were there. Things were checked. Procedures were followed. Findings were documented. These are tangible actions. This answers the basic question that the politicians and the public always ask: What are you doing about it?

One technique that can often be used by nuclear communicators is to use examples that can be understood from daily activities. To explain something like the reactor water cleanup system one can say: “works in much the same manner as a home water softener system – it filters and removes undesirable materials and results in purified water.”

Making the technical understandable to the public is an important principle of nuclear communications. Nuclear communicators need to make sure the public is not afraid of nuclear power and to put any fears at rest with action statements. How can the public support commercial nuclear power if they can’t understand it? Understanding can be achieved with the teamwork of the communications and technical experts of an organization. It takes time and skill as any important activity does. Another example of this principle is how to explain a main feedwater pump malfunction: “pump that provides cooling water to the reactor did not operate correctly.”

As discussed earlier in this paper, a very important technique that is very effective and often neglected in nuclear communications is the use of action statements. Action statements make a technical issue or an event much more real and specific. For example:

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6 IAEA 5, 28
A. S. Greenman

— “Two inspections were conducted in the last year regarding radiation protection and engineering.”

— “systems were tested and test results proved the system was operable”

— “Four inspectors with special radiation detection equipment re-traced every step of the workers.”

— “Inspectors observed the repair of the pump”

The important thing is to state that something was done (action was taken) and in terms that the public can understand. To use technical names of equipment and procedures and systems often blocks (inhibits) communication. Time is spent trying to understand what a “Troxler 110” is when it would have been much clearer to say “nuclear gauge”. The technical terminology confuses most members of the public.

At every opportunity, explaining and pointing out the safety defenses and barriers at a nuclear facility is another principle in nuclear communications with the public. If a nuclear communicator were to receive the question “Can you guarantee that this plant is safe?” there are numerous opportunities within the answer to specifically detail safety aspects, for the public. Of course while no one can absolutely guarantee the safety of any technology, one can point out the various safeguards designed to ensure that nuclear power can be operated safely:

Some of the answer possibilities: “I can guarantee that we have done everything possible to ensure its safety (another possible way to answer this question is to use the phrase ‘designed safety into the plant’). Either way, one can then go onto specific examples of safety:

— There are 24 hour radiation monitors surrounding (or around the perimeter of) the nuclear power plant. If there was a radioactive release it would be detected by these monitors.

— The regulatory authority has conducted 35 inspections this year at our plant looking at security, radiation protection, etc.

— We have done our own audits which include bringing in outside nuclear experts to inspect our internal security and maintenance programs.

— Plant employees undergo ongoing training in…

6. Building the public’s confidence

Be the first one to get your information out and keep getting it out. That way you control it in the news media. That way you maintain public confidence in the safety of your nuclear facility. That way you enhance your professional credibility. Your organization needs to be professionally credible in order to be believed or listened to. Targeted distribution lists are an effective practice to provide regular and ongoing communication to politicians, news media and activists and any other interested public. Every time an inspection or activity report or other documents on a particular facility of interest is done, those on a distribution list automatically get sent this document.

If one can demonstrate to the public through various actions that there are no “hidden dangers” – such as making documents available which put the issue or event in perspective – then a large step has been made in building successful communication with the public. Making documents available for the public and news media and sending it to local politicians, indicates that the utility or regulatory authority is not hiding or slanting events or problems.

The public does not have to agree with you BUT if the information is available, then they have to at least admit that you are technically accurate about the operation of the facility. If the public, journalists and politicians become aware that the utility published numerous reports on its activities and that the regulatory authority conducts confirmatory inspections in which the inspection results and conclusions are also available to them, then the public, politicians and news reporters do not have much, other than emotion, to debate with you. The accessibility of the documents and the facts contained within the reports not only build or enhance credibility but also defuse the anxiety, the fear, the anger by demonstrating to the public that their foundations of security have not been breeched.
A. S. Greenman

For the public, knowledge is security. Fear or anger most often is a result of the lack of knowledge. They “fear” no one is taking action to protect them. They become “angry” that they are denied information that they feel is important because it involves their family, home and their health.

7. Reaching the public

“Risk perception has its own logic, therefore we cannot translate technical models into a communication tool (user instruction...). Nuclear industry needs to break the scientist’s stereotype in order to develop the “neighborhood attitude”, through the trust as a cornerstone for a better understanding about perceived and assessed risks. Nuclear industry has a global (not merely technical or economic) responsibility towards the public: being totally law-abiding is not enough. We need to go further…to maintain a close monitoring of the local concerns in order to take up the ground, to have a proactive attitude in order to pursue our business in best conditions.”

The U.S. Nuclear Regulatory Commission (NRC) in response to the Davis-Besse incident, developed a newsletter that was put on the internet (www.nrc.gov) and detailed planned inspections, upcoming public meetings with the utility, a checklist of what was still needed to be done to start the plant up, etc. While this was in response to an event, there were a few other noteworthy aspects of this document. A “non-technical” write up with drawings of what happened at Davis-Besse and a section on the barriers that are built into nuclear plants to protect the public are part of this newsletter.

Another excellent outreach tool is a “Speakers Bureau”. Individuals who are perceived as neutral and respected can assist in communicating a message to the public that is understandable and credible. These individuals can go on television, be at public meetings, on radio talks shows or used in advertisements.

A proven helpful technique is to also use respected public figures to moderate public meetings. An example of an individual would be a local official or an individual who is from the academic arena.

Ongoing open public meetings with the regulatory authority or utility, with local authorities, with the news media and with the general public is one of the most effective tools available to the nuclear communicator.

REFERENCES


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8 In March 2002 plant workers discovered a cavity in the head or top of the reactor vessel. Corrosion, caused by boric acid, damaged the vessel head creating an irregular cavity about 4 inches by 5 inches and approximately 6 inches deep.
RISK COMMUNICATION – THE KEY OF THE POLICY SUCCESS

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Abstract. Today, in a democratic society, nuclear power development is subject to public acceptance. The acceptance of the nuclear activities development implies both the approval by the government’s proper authorities and also the standpoints of the civil society, expressed in forms more or less institutionalized.

The public has an important role to play in addressing issues of health, safety and environment. Therefore, all activities of a nuclear organization need to be both transparent and should provide for the public’s appropriate involvement, with input not only from the nuclear community, but also from members of the public, interested groups, media, as well as public representatives in local and national councils and groups. How to communicate clearly with the public is a very challenging job that requires special attention.

Risk communication is the art or practice of talking about scientific information and principles to a non-expert audience. Its goal is to convey accurate and trustworthy information about safety to decision-makers, the public, or anyone else with an interest in the safety of the public or themselves. The science of communication, public education for a proper perception of risks are the key for attaining social acceptance of any technology that is about to become part of the sustainable development process and hence, of nuclear energy.

The paper describes the way our nuclear organization is acting and the results in risk communication activity it achieves.

1. Introduction

The Cernavoda Nuclear Power Plant Unit 1, the first nuclear reactor of 706 MWe CANDU 6 in Romania, has been operating since December 1996 as a result of the works carried out for the implementation and upgrading of the national nuclear programme.

Having as its main mission the generation of electricity and heat as well as CANDU 6 nuclear fuel, our state-own company “Nuclearelectrica” has an active participation in the power development programme in Romania.

Today, in a democratic society, nuclear power development is subject to public acceptance. The acceptance of the nuclear activities development implies both the approval by the government’s proper authorities and also the standpoints of the civil society, expressed in forms more or less institutionalized.

2. Background

The public has an important role to play in addressing issues of health, safety and environment. Therefore, all activities of a nuclear organization need to be both transparent and should provide for the public’s appropriate involvement, with input not only from the nuclear community, but also from members of the public, interest groups, media, as well as public representatives in local and national councils and groups. To be sure the public is informed, information needs to be technically coherent, clear, accurate, reliable and comprehensible to the public. If a communication gap between the nuclear
organization and the public will manifest, this gap could be filled up with speculations, rumors or misinformation, leading to undesirable impact on the programme. How to communicate clearly with the public is a very challenging job that requires special attention.

3. Communication

Risk communication is the art or practice of talking about scientific information and principles to a non-expert audience. Its goal is to convey accurate and trustworthy information about safety to decision-makers, the public, or anyone else with an interest in the safety of the public or themselves.

Risk communication can be considered as a subset of the overall field of risk analysis. As NRC stated, there are three stages of risk analysis: risk assessment, risk management and risk communication.

We have set up a complex information programme on safety radiation, radioactive wastes and other issues on nuclear energy. The programme comprises a set of structured and systematic activities aiming at establishing or improving communications between the nuclear organization and target audiences.

Within the communication strategy, we have provided the opportunity for the members of the public to express their opinions and to provide information and comments to nuclear organization for major nuclear implementations. Various formats are used to invite public to make comments, such as informational public meetings, roundtable discussions and formal public hearings. The education and training integration will mitigate the barriers, which may inhibit the interaction and communication process. An effective way to avoid negative reactions consists of the extensive consultation to identify the public’s concerns and needs, the access to suggestive and attractive programmes for education and training.

We have received a significant support in the communication activity from the worldwide scientific development. One of the most dramatic changes in the dissemination of scientific information has been the prolific use of the World Wide Web. Widespread distribution of scientific information has opened up unprecedented direct avenues of communication between the scientific community and the interested public, using this the most dynamic medium.

The Romanian experts in nuclear power, organized in associations like AREN (Romanian Association for Nuclear Energy) and ROMATOM (Romanian Atomic Forum – active FORATOM’s member) pay permanent attention to the communication with the public. Thus, they add their competence, objectivity and credibility to the actions undertaken by the economic agents, institutes and organizations acting in the nuclear sector.

“The Days of Nuclear Energy” event is already traditional. This event is organized by AREN and it addresses to everybody taking interest in the domain, but mostly to the young people. The event has a great section dedicated to drawings and pictures on nuclear items made by children between 6 and 17 years old, on themes like: “What do we know about energy?”, “The atom – our friend”, “The energy of the new millenium”. It is very interesting to see their perception on the atom and the nuclear energy as life generating. We consider this event a very good way to educate the young people to understand the benefit of nuclear energy and application to mankind. They are encouraged by the awards offered by the “Alexandru Ene” Foundation, a foundation created to support the actions in favour of nuclear energy.

The “Ionel Purica” Foundation offers annually the award “Ionel Purica” to a personality with special contribution in promoting the Romanian nuclear energy.

On an annual basis, these associations organize nuclear events as SIEN (International Symposium on Nuclear Energy), gathering Romanian and foreign specialists in the nuclear field, who present papers on items like: the functioning NPP in safety conditions, nuclear engineering, young generation – nuclear knowledge management, public acceptance.
The magazines “Energetica” and “Nuclear Energy” are other steps towards the dialog with the Romanian civil society. The papers on different items of nuclear energy can be considered an initiative of an effective campaign for the correct information of mass media and public opinion regarding the nuclear energy.

The magazines together with the Politechnica University – Bucharest, the Romanian National Committee of the World Energy Council are boosting round tables dedicated to nuclear power issues debates. These events are attended, besides specialists pertaining to kindred domains, by various representatives of civil society: people in the education area, members of non-governmental entities, persons working in central and local administration, ecologists etc.

Daily central magazines like “Economical True” or “The Economist” objectively present multiple aspects related to nuclear energy. The article “Nuclear Energy: Pros and Cons” published in “The Economist” magazine brings to the chief editor the prize for the most active journalist in the objective promotion and support of the nuclear energy, offered by ROMATOM. Starting with 1994, the Romanian national television broadcasted talk shows on nuclear energy.

Exhibitions are organized on a regular basis on nuclear energy addressing both to the specialists and to the general public, where informative materials, video records, computer modelling of nuclear themes, as well as panels showing Romanian and foreign companies involved in the nuclear energy are displayed. The public debates carried out last year at Constantza, Medgidia, Cernavoda and Bucharest concerning the results of the impact study of Cernavoda NPP’s Unit 2 on the environment are typical for our transparency and openness towards civil society and for accepting democracy rules. In our country, this kind of debates becomes a mandatory stage, provided in the present Romanian legislation and applied in the completion of the works performed on the project.

Some actions are developing towards mass media and population to promote the use of nuclear energy by explaining the economic and environmental benefits, the positive contribution in avoiding the climate change, in the context of the sustainable development concept. Coming to better understand the benefits of nuclear science and technology may occur through more awareness of how nuclear activities contribute to our everyday life: how it helps us improve our activities, how it provides safe electricity, heating or potable water to our houses.

For Cernavoda community there are direct and indirect support programmes, including income increase activities, public works, educational activities such as scholarships for local students and local information program. Indirect support activities include employment of local population to work at the nuclear organization. The local community support provided through improving the living conditions in the town of Cernavoda started in 1991, including a number of 21 objectives related to the project of the Cernavoda NPP: urbanistic, social, cultural buildings, as well as dwellings for the operation and executive staff of the nuclear power plant. Two important objectives were completed and inaugurated in 2002: Cernavoda Town Hospital and “Saint Maria” bridge over the Danube – Black Sea Channel. There are other social and economic benefits for the regional public that should be underlined: provides over 1,300 jobs, provides activities for 15 contractor companies having over 350 jobs, provides accommodation for over 500 plant employees, provides heating for more than 60% of Cernavoda inhabitants at the lowest price in the country. The perception of the contribution of nuclear energy to regional prosperity is a very important indicator of a good communication activity.

The Cernavoda NPP has in its organizational chart an Information Centre having the supporting role of bringing to the public’s attention the factual information and more awareness of how nuclear activities contribute to everyday life. A couple of thousands of visitors have visited the Unit 1 within the “Open doors” programme initiated by the plant. Students visits are often organized in Cernavoda for attracting the younger generation to join and keep the nuclear option alive. Special information seminars and workshops have been organized for the representatives of mass-media, followed by the plant tour.
A considerable amount of printed materials such as brochures, leaflets, information documents have been elaborated and distributed to public, policymakers, opinion leaders, media and non-governmental organizations. Information about events in the company are provided to the media within press releases, press conferences, interviews and by internet on the company’s website www.nuclearelectrica.ro.

4. Conclusion

The science of communication, public education for a proper perception of risks are the key for attaining social acceptance of any technology that is about to become part of the sustainable development process and hence, of nuclear energy. At present, this science is an remarkable stage of development and progress. We have to assimilate it, including all new achievements. We must understand public points of view and address it using its own language. What we need is not only a unidirectional flow of information towards the public, but also an effective and steady dialog with representatives of the general public. It shows the change of culture from information to communication.

Public confidence, built on open, credible communication, patience and perseverance, will make a safe ground for social acceptance of nuclear energy.

REFERENCES

SOME ISSUES OF ARMENIAN NPP SAFETY EVALUATION

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Abstract. There have been analyzed the measures that were already implemented since the ANPP restart. It was described the course of realization of the following analyses: the Deterministic Safety Analysis, Probabilistic Safety Analysis and Safety Analysis review for the ANPP. Some quantitative results of Core Damage Frequency are presented that were estimated due to the PSA Level 1

1. Introduction

After gaining independence, Armenia faced the energy crisis. To overcome it, the Government of RA made the decision to restart the Armenian NPP (ANPP).

When preparing the restart of the ANPP, the “ANPP Restart Concept” and the “List for Safety Upgrading Activities of Unit 2 of the ANPP” were developed taking into consideration the measures needed for the ANPP safety upgrading. The Concept provisions were based both on materials of “TECDOC-640” and the corresponding safety upgrading measures implemented at the similar nuclear units in other countries. The technical and financial help of the Russian Federation made it possible to restart Unit 2 of the ANPP in 1995.

2. Safety Upgrades

Since the ANPP restart, more than 120 safety upgrading measures from the List and more than 1000 modernizations have been implemented at the ANPP. Some important issues of implementation from the List, which have been already realized, are grouped in the following way:

Reactor Core – improvement of fuel design, improved analysis of core-wide margins, axial monitoring of some elements; improvement of water chemistry control, source range monitor.

Systems - decay heat removal: nuclear service water system, emergency feedwater system, emergency condensers, confinement; reduced leakage,

pressure Relief: pressurizer relief valves, steam generator relief valves,

ECCS: Leak detection system,

others: turbine generator seals; upper head gas release.
Component integrity - ultrasonic test equipment and training, pump shaft alignment equipt, thermographic, transformer oil sampling and analysis equipt, valve repair, valve condition monitoring equipt, SG heat-exchange tubes replacement.

Instrumentation & Control – safety parameter display system, plant computer, radiation monitoring system, replacement of SG level control regulators by the regulators of firm FISHER.

Electric - diesel generator upgrades, reversible motor generator sets, circuit breakers – 220V, 6kV, battery room cooling, replacement of accumulator batteries by the accumulators of firm VARTA.

Accident analysis - design basis accidents, seismic response spectra, seismic analysis, seismic walkdowns.

Fire protection - fire resistant floor coating, fire doors, fire detection and alarm system, coating structural steel in turbine hall.

Equipment qualification - equipment qualification

Operation - safety culture, multifunction simulator, training modules, training room upgrade, rewrite of operational procedures, updated design drawings.

The implementation of safety upgrading measures was realizing under the financial and technical support of Governments of the USA, France, UK and Italy, as well as of the European Community, IAEA, WANO, etc.

3. Safety Analysis

So, the most important issues of safety upgrading presented by the “TECDOC-640” relevant to the WWER-440 model 230 have been, in general, implemented. Now the level of the ANPP safety is within such a rate that makes it possible to evaluate the real ANPP safety grade by carrying out the Deterministic Safety Analysis (DSA), and the Probabilistic Safety Analysis (PSA).

The Armenian Institute “Armatom” has been developing for the ANPP the Deterministic Safety Analysis in the frame of the US DOE Technical Assistance program. The part of this work, which includes the mathematic models for the primary and secondary circuits has been already completed. It was modeled 32 of technological systems of the primary and secondary circuits which are important for the ANPP safety. Approximately 70 design basis accidents have been estimated. In the near future, the Armatom will initiate the development of the non-design basis accidents estimation.

Two Armenian institutes - ARMATOM and ATOMSERVICE, together with the SOGIN (Italy) have completed the PSA Level 1 for the ANPP. The objectives for this work implementation were:

- to provide an integrated view of the ANPP behavior in response to the transients and accidents thus to develop a decision-making tool for the ANPP safety assessments,

- to provide input to identify further plan modification needed to optimize the ANPP safety.

The technical task for the ANPP PSA are defined in the following categories:

- Level 1 Analysis for full power Internal Events,

- Sensitivity, Importance, and Uncertainty Analysis for Internal Events.

The Level 1 analysis includes:

- Internal initiating events evaluation,
A.A. Gevorgyan et al.

- Event tree and success criteria analyses,
- Plant systems analysis using fault tree models
- Common cause failure and human reliability analyses,
- Data analysis
- Fault tree and event tree quantification using a fault tree linking approach to calculate the core damage frequency.

The major activities performed during this study include the following:

- Initiating event and event tree analysis—evaluations are performed to identify a comprehensive set of initiating events. This evaluation includes review of WWER-440 operating experience, past PSAs, and consideration of ANPP-specific features. For each initiating event category, an event tree is constructed to model the accidents sequences that may result.

- Sources criteria—analyses are performed with the RELAP-5 computer code to determine the success criteria for system mitigation following initiating events.

- Analysis of individual system—qualitative analysis and fault tree construction are performed for all safety related and supporting systems that contribute to prevention or mitigation of severe accident events.

- Human reliability analysis—a detailed human reliability analysis is performed, with emphasis on the evaluation of the effect of single operator decision on more than one system. Operator actions are mainly modeled in the fault trees.

- Common cause failure analysis—an analysis is performed to identify and model the dependencies (common cause failure), both internal to individual systems and among systems, that use similar components exposed to similar environments. For the most dominant common cause failures, the Multiply Greek Letter method is used for components with greater than two–fold redundancy.

The results of the work performed are as followed:

The core damage frequency (CDF) is composed by three contributors:

a) Sequences derived from failure of confinement safety valves,

b) Sequences derived from failure to close steam lines to the turbine after the reactor scram (value is 6.01E-6 events per reactor year),

c) All other remaining sequences (value is 5.33E-5 events per reactor year).

The contribution from point b) represents the most important contributor to the CDF derived from the operation a single system (components such as confinement safety valve to open and subsequently to close after large LOCAs, medium LOCAs and EMS injection line break events).

The contribution from point b) has not been further accounted in the Analysis because their preliminary assess was equalized to the lost of secondary side heat removal function but a more detailed thermal and hydraulic analyses are needed to correctly assess the plant behavior following the occurrence of such event.
Therefore, the CDF analyzed in detail in the following subsections will be that derived from the sequences belonging to points a) and c).

The ANPP core damage frequency for full power internal events from at-power conditions is $9.28 \times 10^{-5}$ events per reactor year.

Operator actions are important for the unit behavior after an accident. In fact, if no credit is taken for operation actions, the plant CDF is 1.0, that means without operator at any occurrence of an accident the core damage positively occurs. It is therefore important to add some automatisms not only to further reduce the CDF, but mainly to have a unit less sensitive to the operator behavior.

This work has been done as a previous development. It is necessary to continue and enlarge the process.

The Scientific Engineering Centre of ANRA is carrying out the PSA for the ANPP Unit 2 relevant external events such as seismic events, fire.

4. Conclusion

The ANRA has already developed the requirements for the contents and structure of the Safety Analysis Report (SAR) which were approved by the Government decision of 21.11.2002. Now, the SAR is under the process of realization, which will be submitted to the Armenian Nuclear Regulatory Authority (ANRA) late in 2005 for receiving the operation license. The work on the projects mentioned above has been performing by experts from the ANPP, ARMATOM, ATOMSERVICE, as well as by the experts from US INEEL, Italy SOGIN, UK SERCO Co, Slovakia VUJE, etc.

After all those activities have been completed, it will be possible to do an eventual assessment of the ANPP safety and to judge of it with more accurate data.
DECONTAMINATION OF $^{125}\text{I}$ IN MEDICAL LABORATORY

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Abstract. A radiological laboratory for diagnoses was contaminated by $^{125}\text{I}$. A large-scale survey of gamma-radiation has been made in different locations of the floors and walls of the lab to determine the contaminated area and its activity. The activity level before decontamination for the wall and floor was 1400 and 2000 Bq/cm$^2$ respectively. Decontamination was carried out by using ethyl alcohol, potassium permanganate, ethylene diamine tetracetic acid and tissue papers. Decontamination factor has been calculated and it was 175 and 200 for the wall and floor respectively. D&D computer code has been used to calculate Total Effective Dose Equivalent (TEDE). TEDE from the wall and floor before decontamination were 3.05 and 4.35 (mSv/yr) while after decontamination were 18 and 23 µSv/yr respectively. These results are lower than the Egyptian and the international regulations (10 mSv/y for the public) according to International Atomic Energy agency, IAEA, Safety Series, SS, no. 115 (1994).

1. Introduction

There are many thousands of small nuclear facilities including radiochemical laboratories, hospital installations, hot cells, waste treatment and decontamination facilities, which will eventually need to be decommissioned. Before starting dismantling activities, the characteristics and distribution of the radioactive and toxic materials (e.g. asbestos) and radiation fields in the facility must be defined. Removal of contaminants from surfaces before decommissioning process is generally based on cleaning techniques which have been well established for several decades.

2. Background

Decontamination involves the removal of bulk contaminants, and decommissioning involves the removal and cleanup of contaminated equipment. Almost all the procedures and reagents used in decontaminating equipment and facilities in the nuclear industry were first developed and used for cleaning conventional non-nuclear plants. Decontamination is understood as a process that should meet several criteria, such as: adequate decontamination factors; acceptable corrosion rates; easy availability of the equipment; simple operation and control; acceptable time requirements; satisfactory safety aspects (both radiological and non-radiological); acceptable costs of application and minimal amounts of wastes. Many decontamination methods and techniques are available that would facilitate operation, inspection, maintenance, modification or decommissioning of nuclear installations. The selection of an appropriate procedure is an important task for policy makers of national nuclear energy programmes and especially for operators of nuclear installations. Removal of contaminants from metallic surfaces depends on many physico-chemical variables that govern the process of contamination. Among them the nature of the surface is the most important factor (i.e permeability, film formation, age and chemical composition of the oxide layer). Other factors should be considered such as the base material and contamination history. Chemical decontamination processes are generally necessary where metal or oxidized surfaces have become contaminated. In general, decontamination of buildings and structures, except for decommissioning purposes, should be accomplished so as to avoid airborne contamination and thus protect personnel from hazards due to dust inhalation and external contamination during operating and maintenance work. Most of these methods are usually mechanical methods such as water jetting, abrasive techniques, etc.
3. Materials and Methods

The activity of radionuclides before and after decontamination was measured by using calibrated surface contamination monitor type Berthold LB 1210 B and the contaminated area was determined.

Swabbing, washing, scrubbing and brushing were used for decontamination of floors and walls of the contaminated area. The decontamination is achieved by the following process:

— The painted surface of the contaminated wall swabs from out to in by towel wetted with water.
— Mechanical action of brushes and swabs, in combination with an adequate solvent (Ethyl Alcohol C₂H₅OH).
— Dilute chemical decontamination 1% KMnO₄ followed by 1% Ethylene-Diamine-Tetra-Acetic acid (EDTA) was used⁸.

Solid and liquid of radioactive wastes produced from the decontamination process were segregated and collected in a plastic containers and the activity level on the surface of these container was measured.

Total Effective Dose Equivalent (TEDE) calculated by using Decontamination and Decommissioning (D&D) computer code version 2.1.0. This code⁹ implements the methodology and information contained in NUREG/CR-5512, Volume 1 as well as the parameter analysis in Volume 3 that established the probability distribution function (pdfs) for all of the parameters associated with the scenarios, exposure pathways and models embodied in D&D. Two scenarios are implemented in D&D: building occupancy and residential. In our case building occupancy scenario relates volume and surface contamination levels used to estimates the total effective dose equivalent (TEDE) received during a year of exposure with the conditions defined in the scenario. This scenario accounts for exposure to fixed and removable to thin layer or surface contamination sources within a structure. The building occupant is defined as a person who works in a commercial building following license termination. The pathways that apply to the building occupant include:

— External exposure to penetrating radiation from surface sources,
— Inhalation of resuspended surface contamination, and
— Inadvertent ingestion of surface contamination.

The Building Occupancy Scenario Model

The building occupancy scenario model includes eight parameters:

— External dose rate factor for exposure from contamination uniformly distributed on surfaces, DFES \(_j\) (mrem/h per dpm/100 cm\(^2\))
— Inhalation committed effective dose equivalent (CEDE) factor, DFH \(_j\) (mrem/pCi inhaled)
— Ingestion CEDE factor, DFG \(_j\) (mrem/pCi ingested)
— Length of the occupancy period, \(t_o\) (d)
— Time that exposure occurs during the occupancy period, \(t_e\) (d)
— Resuspension factor for surface contamination, RF \(_o\) (m\(^2\)/h)
— Volumetric breathing rate, \(V_o\) (m\(^3\)/h)
— Effective transfer rate for ingestion of removable surface contamination from surfaces to hands, from hands to mouth, GO (m\(^2\)/h)

The length of the occupancy period (\(t_o\)), the time that exposure occurs (\(t_e\)), and the effective transfer rate for ingestion (GO) are behavioral parameters. The volumetric breathing rate (\(V_o\)) is a metabolic
M. Abdel Geleel and Amaal A.Tawfik

The committed effective dose equivalent factors and the resuspension factor are physical parameters. As discussed below, the committed effective dose equivalent factors are classified as physical parameters because their values depend on the source geometry and contaminant solubility class.

The annual TEDE for a parent radionuclide in the building occupancy scenario $\text{TEDE}_0^i$ is calculated as a sum of:

- external dose resulting from external exposure to penetrating radiation from the surface sources represented by the parent and daughter (if any) radionuclides, $\text{DEXO}_i$;
- CEDE for inhalation resulting from inhalation of resuspended surface contamination represented by the parent and daughter (if any) radionuclides, $\text{DHO}_i$; and
- CEDE for ingestion resulting from inadvertent ingestion of surface contamination represented by the parent and daughter (if any) radionuclides, $\text{DGO}_i$.

The mathematical formulation of the above is (NUREG/CR-5512, Vol. 1, p. 3.14\(^{(10)}\)):

$$\text{TEDE}_0^i = \text{DEXO}_i + \text{DHO}_i + \text{DGO}_i$$

$\text{DEXO}_i$, $\text{DHO}_i$, and $\text{DGO}_i$ are calculated using the average annual surface activity per unit area of the parent, $C_i$, and daughter radionuclides, $C_j$, during the first year of the building occupancy scenario. Although ingrowth of daughter nuclides may, in some cases, cause TEDE to increase with time, in the default scenario model the maximum TEDE is assumed to occur during the first year of the scenario to simplify the analysis.

Calibration of The Contamination Monitor (Berthold LB 1210 B)

Calibration was performed at Secondary Standard Calibration Laboratory (SSDL) at the National Institute for Standards, Cairo, Egypt. It was compared by a calibrated instrument (Automess-6150) which was used for relative calibration. In the calibration process a standard point source Cs-137 with activity $2.2 \times 10^9$ Bq placed at 9 cm. The correction factor for the instrument used is 1.088 \(^{(11)}\).

4. Results And Discussion

ACTIVITY LEVEL MEASUREMENT

The activity of radionuclide was measured before and after decontamination at a distance about 10 cm from the contaminated area. The decontamination factor was calculated and the data are represented in table 1. From this table it is clear that the decontamination factor is very high value for both wall and floor. This means that high efficiency decontamination process has been carried out and this is may be due to the presence of active group in the polar solvent (–OH, –COOH, –NH\(_2\) groups) that makes chelation and adsorption of radioisotopes. Also, the volume of radioactive liquid and solid wastes that produced was very low.

Table –1- The activity level before and after decontamination

<table>
<thead>
<tr>
<th>Contaminated Place</th>
<th>activity level $\text{Bq/cm}^2$</th>
<th>decontamination factor (DF)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Before decont.</td>
<td>After decont.</td>
</tr>
<tr>
<td>Wall</td>
<td>1400</td>
<td>8</td>
</tr>
<tr>
<td>Floor</td>
<td>2000</td>
<td>10</td>
</tr>
</tbody>
</table>

Total Effective Dose Equivalent (TEDE) before and after decontamination
M. Abdel Geleel and Amaal A.Tawfik

Total Effective Dose Equivalent (TEDE values (mSv/yr) for all pathways, external, inhalation and ingestion before and after decontamination was calculated using D&D computer code version 2.1.0 and the data are present in table 2. This table shows that, the doses received from external exposure - before and after decontamination- is greater than that received from inhalation or ingestion. Also, the doses received from the floor is higher than that from the wall, this may be due to the concentration of radionuclide –before and after decontamination - on the floor greater than that on the wall.

Table-2- Total effective dose equivalent values before and after decontamination

<table>
<thead>
<tr>
<th>Contaminated place</th>
<th>Dose (mSv/yr)</th>
<th>Before decontamination</th>
<th>After decontamination</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>All pathway</td>
<td>External inhalation</td>
</tr>
<tr>
<td>Wall</td>
<td>3.05E-0</td>
<td>1.18E-0</td>
<td>1.1E-0</td>
</tr>
<tr>
<td>Floor</td>
<td>4.35E-0</td>
<td>1.68E-0</td>
<td>1.6E-0</td>
</tr>
</tbody>
</table>

Table-3- Reduction factor due to decontamination (ratio of exposure values before and after decontamination)

<table>
<thead>
<tr>
<th>Contaminated place</th>
<th>All pathways</th>
<th>External</th>
<th>Inhalation</th>
<th>Ingestion</th>
</tr>
</thead>
<tbody>
<tr>
<td>Wall</td>
<td>168</td>
<td>175</td>
<td>171</td>
<td>175</td>
</tr>
<tr>
<td>Floor</td>
<td>192</td>
<td>200</td>
<td>199</td>
<td>200</td>
</tr>
</tbody>
</table>

Radioactive wastes produced from the decontamination process (disposal gloves, tissue paper, some pieces of cotton) were collected in a plastic container and the dose rate on the surface was measured and it was 0.02 µSv/h. This value is accepted according to Egyptian and international regulations\(^\text{(12)}\). The total weight of the solid radioactive wastes was about 2 kg, while the volume of the liquid wastes was about 0.25 L.

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PROBABILITY SAFETY ANALYSIS IN PARTICLE ACCELERATORS

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Abstract. An attempt has been made to carry out a Probabilistic Safety Analysis (PSA) of particle accelerators. Possible hazards in the accelerator have been identified and a qualitative analysis related to radiological risk has been undertaken. Hazards like beam loss, target rupture et. have been envisaged from the viewpoint of PSA. The present analysis leads to the conclusion that inadequate knowledge, complacency and overconfidence are major contributing factors of human error, responsible for accelerator hazards.

1. Introduction

Particle accelerators have grown past the stage of infancy and are now widely installed with diverse projectile types and increased beam current and energy. For medical and industrial applications such accelerators often need to be located in populated areas. Consequently, there is significant increase in the radiological risks posed by them. Accelerators have been in operation for several years, yet there is a deficiency in reliability estimates of accelerator systems. There is a need for evaluating the reliability of the new accelerator systems from the design stage itself. This requires a systematic approach including both qualitative and quantitative evaluation. In the present paper a radiological safety analysis using the probabilistic safety analysis (PSA) methods are being carried out for the accelerator. Since an accelerator does not contain any fissile material inventory like in a nuclear reactor the PSA approach followed here is also different from that in a nuclear reactor [1]. It is proposed to identify all possible hazards in the accelerator. Potential deviant conditions and their associated risks are discussed. This paper briefly describes and analyzes the adequacy of the measures taken to eliminate, control or mitigate identified hazards.

2. Modeling the Accelerator

There are numerous variations and designs of accelerators existing. However, in general an accelerator consists of an ion source, the main accelerator and the target. There are magnet systems, with high voltage and high current power supplies, cooling arrangements, vacuum systems, ventilation systems and safety interlock systems. The accelerator beam is transported to the target using the beam transport system. The material, design and dimensions of the target vary depending on the end use and user requirements, experiments planned and beam to be extracted. Failure of any of the major accelerator systems or essential support systems is expected to result in shutdown of the accelerator.

3. Proposed methodology for accelerator safety evaluation

For safety analysis, it has been decided to take the approach of hazard identification. Their possible causes, severity of impact and adequacy of mitigating measures are studied. Possible hazards in the accelerator are identified as:

1. Ionizing radiation inside accelerator
2. Ionizing radiation exposure outside accelerator
3. Exposure to hazardous materials
Lekha M. Chowdhury and P.K. Sarkar

(4) Electrical hazard
(5) Non-Ionizing radiation exposure – RF
(6) Environment pollution
(7) Fire Inside accelerator Building
(8) Fire in equipment and control areas
(9) Seismic hazards

Each hazard identified can be caused by various Initiating events. Maximum impact is for the hazard due to Ionizing radiation inside the accelerator, which can be due to both prompt and residual radiation or any contamination. Presently, we have restricted our analysis only to hazards due to ionizing radiation inside accelerators.

4. Listing of undesirable Initiating Events.

The following initiating events, which are undesirable and can lead to severe radiological consequences at site are identified as follows:

(1) Beam losses
(2) Target rupture
(3) Faulty components causing radiation leak
(4) Trapping of persons inside high radiation areas
(5) Failure or bypassing interlock facility

4.1. Beam losses

An errant beam, if it were to strike an internal component of the machine, could produce a large dose confined to a small angle. This is possible due to human error, malfunctioning of precision instruments or some equipment failure. Improper tuning of the beam can be possible due to inadequate skill and/or knowledge of the operator. Beam tuning is a multiparameter optimization problem where several optimal combinations are possible. The combination chosen depends on the operator. If the beam is switched to the wrong beamline it can lead to high radiation at unexpected locations. For complex facilities having many different operating modes and several rooms, tunnels or caves, the beam loss scenarios are difficult to assess and may be different for different operators. In case of any equipment failure, if the failure is not monitored, the operator may not be aware of the equipment failure leading to larger beam loss. The effect of any beam loss is the activation of the accelerator parts and shield materials. There will be streaming of large amount of radiation through ducts (S-bends) and labyrinths. Workers during maintenance will receive a high dose. There will also be generation of a large quantity of radioactive waste during decommissioning.

4.2. Target rupture

The consequence of irradiation of a target by the accelerator beam can result in different toxic radioactive products. It is important to have an idea of the target, experiment being performed and possible radionuclide production after irradiation. This can help in estimating residual dose and monitoring different kinds of radioactivity during and after the experiment. When high beam current is used for experiments, the target rupture probability is also high. This probability would also depend on the target material, its form, design, cooling facility required and available. The effect of target rupture is extensive beamline contamination. Radionuclides produced will contaminate the pump oil and some of it may be released to the air.

As an illustrative example we give in Figure 1 the fault tree diagram for target rupture. Several event sequences for the target rupture scenario are possible and the event tree would depend on whether the target is in air or vacuum, does a beam window exist or not, byproducts of target irradiation are in a closed loop or not etc.
4.3. Faulty components causing leak in radiation

A simple example is a shield wall plug not closed properly, beamline leak etc. A sudden vacuum leak occurring would take sometime before it is detected. This will produce increasing radiation dose till the chamber filled with air. In a complex accelerator facility, different components of the accelerator, may not work efficiently at the same time. Accelerator components may be damaged or destroyed when power deposition (from radiation) is too high. Inadvertent placing of high Z materials (or Be) in electron/gamma flight path may cause neutrons to be emitted where they are not expected.

4.4. Persons trapped in a radiation area

The possible situations are:

— Maintenance person remaining inside the cyclotron and the accelerator is started.
— For experiments where low beam current is used, target rupture consequences are low but generally more people are experimenting here. So the possibility of a person getting trapped is higher.
4.5. Failure or bypassing of Interlock facility

This can lead to the accelerator operating when it should be shutdown. Authorities sometimes tolerate bypassing of some interlock facility. At other times it is due to the overconfidence of the operator in their own capability or the capability of the machine.

5. Recommendations for Risk reduction

The risk importance of the initiating event depends on the consequences the events. All the events can lead to an increased radiation dose to one or more persons. The following recommendations are made for reducing the probabilities of occurrence of the hazardous events.

5.1. Beam losses

Appointing skilled operators and and arranging for their regular training should be done.

5.2. Target rupture

Details of target design and experimental procedure is to be scrunized by competent persons. This will help in planning any special safety measures required. It should be ensured that all users are trained in radiation protection procedures.

5.3. Faulty components causing leak in radiation

Continuous monitoring is essential to have an idea of the change in field in any area of the cyclotron. From Control room the data recorded from monitors should be connected to the local LAN for viewing from any LAN terminal.

5.4. Person being trapped in a radiation area

Administrative controls should be used and announcements should be made so that any maintenance person inside the cyclotron can leave the place before it is started. Access control system should be implemented, which will give an online indication of any personnel working, the location and duration of work. Starting of the accelerator should be interlocked with access control.

5.5. Failure or bypassing of Interlock facility

There a large number of interlocks, which need to be tested at regular intervals to ensure that they operate on demand. Bypassing of any interlock can be done only after proper authentication.

6. Conclusion

The present qualitative analysis provides us with just a starting point for future detailed PSA of accelerators. Further study and work is required for an in-depth analysis of the events and their consequences. Event trees have to be drawn and quantified to have a quantitative estimation of the risk involved. This will help in risk based decision making in accelerators. However lack of data hampers any meaningful quantitative estimation in this respect. A detailed study of the "human error" in an accelerator is also planned since it has been identified as one of the major contributors to plant risk in accelerator, partially because of the complexity of the accelerator operation. An online Radiation monitoring system and an access control system recording all entries to restricted areas are also being installed.

REFERENCES

AUTHORIZATION OF NUCLEAR INSTALLATION PERSONNEL IN KAZAKHSTAN

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Abstract. In this paper is described the present situation in the field of authorization of nuclear installation personnel in the Republic of Kazakhstan, results achieved during last period. Current matters and needs for improving of the system of personnel authorization in the country and some difficulties, which Kazakhstan encountered during the transition to an independent state are discussed.

1. Introduction

The Republic of Kazakhstan one of the new independent countries of the former Soviet Union is located immediately to the south of Russian Federation and west of China. It encompasses over 2.7 million sq. km of land area and has a population of over 14 million. Before the break up of the Soviet Union, Kazakhstan had no legislative base in the field of atomic energy use and had no state bodies, called to execute the control at observance of security measures by atomic energy use. Since 1991, after getting sovereignty, Kazakhstan started developing its own legislative and regulatory system in this area. In accordance with the Decrees of the President, appropriate structures in Kazakhstan were created. They are: Atomic Energy Agency(since 1999 Atomic Energy Committee of the Republic of Kazakhstan), as the main Supervising governmental body, National Nuclear Centre combining all nuclear related scientific institutes, and National Corporation of Atomic Energy and Industry Enterprises KATEP. On 14 February 1994, Kazakhstan joined the International Atomic Energy Agency.

The general purpose of activity in the field of atomic energy use in Kazakhstan is to safely and effectively ensure the safety of the present and future generations and to protect the environment from radioactive contamination both by normal and extraordinary situation.

One of the main encountered needs in the field of the nuclear installation safety is the necessity of authorization of personnel in nuclear installations. It is very topical for the Republic of Kazakhstan as there are following nuclear installations [1]:

— nuclear power plant BN-350 in Aktau-city (since 22 April 1999 is under decommissioning procedure)
— four research nuclear reactors of the National Nuclear Centre (one of them is located in Almaty, and three of them are located in Kurchatov-city on the former Semipalatinsk Test Site).
— Ulba metallurgical plant for nuclear fuel pellets production in Ust-Kamenogorsk.
— Storage facilities for nuclear materials, radiation substances and radioactive waste.

According to the legislation of the Republic of Kazakhstan, the Atomic Energy Committee of the Republic of Kazakhstan (KAEC) as a regulatory body authorizes personnel who conduct a number of activities in the field of atomic energy [2]. This authorization is carried out by means of examinations in accordance with regulatory requirements [3, 4].
2. Activity and Present Status of Personnel Authorization

For the purposes of authorization, a new set of regulatory documents were developed. Assistance of international organizations in this area is very important and useful. During the participation in the IAEA TC Project KAZ/9/006 “NPP Sitting” only few regulatory documents on qualification, selection, training and authorization of personnel involved in activities connected with the use of atomic energy has been developed. However, from the support of Division of Nuclear Power (Department of Nuclear Energy of the IAEA) were developed and now are used in Kazakhstan. This work is very important for the present and future activity of all entities involved in the field of atomic energy use. Requirements of those documents are mandatory for all entities, involved in this activity.

Support in personnel-related field provided since 1999 had a multi-aspect nature and was derived from actual needs of the Republic of Kazakhstan in the establishment of national systems for reliable training, qualification and authorization of the personnel involved in nuclear activities in Kazakhstan. Due to the large variety of Kazakhstan nuclear facilities and the considerable amount of personnel involved in nuclear activities in combination with limited resources, there is an outstanding need for implementation of thought-out, balanced and systematic approach to the development and upgrade of Kazakhstan nuclear training and qualification infrastructure. These considerations were the basis for planning and implementation of the IAEA and Kazakhstan joint activities. It may be stated with confidence that the year 1999 gave the results serving as the cornerstones for stage-by-stage and systematic further developments. The 2000-2004 activities included the development of personnel-related sample regulation for Kazakhstan taking into account the needs identified in 1999.

During the considered period the following specific activities have been conducted in the field of Training, Qualification and Licensing of the personnel involved in nuclear activities and nuclear facilities:

- A number of Workshops on NPP and Nuclear Facility Personnel Training System Development was conducted for the representatives of the KAEC and other national organizations involved in nuclear activities (e.g. Kazakhstan National Nuclear Center, Joint-Stock Companies “KATEP” and “KAZATOMPROM”, Mangyshlak Atomic Energy Complex).

- The current status of the Kazakhstan nuclear facility personnel training system was evaluated, major needs identified and discussed in proactive and realistic manner. Kazakhstan legal texts, regulations, normative documents and sample training programmes were analyzed in regard to the issues of personnel qualification, training and authorization/licensing. Urgent needs in the upgrade and development of above mentioned documents were identified and a corresponding Action Plan was developed and implemented.

— The KAEC’s personnel responsible for the document development were identified, coached and facilitated by IAEA representatives. Good practices and sample documents on NPP personnel qualification, training and licensing were presented by the IAEA expert mission teams.

— The documents covering outstanding and urgent needs were developed under assistance of the IAEA experts and staff members:

- Comprehensive structure of documents concerning the requirements for Qualification, Selection, Training and Licensing of Kazakhstan nuclear facilities’ personnel was developed and being logically implemented. Action Plan to develop and implement the whole structure of necessary documents was elaborated, discussed, agreed, and serves as thought-out guidance for the Counterpart.
• Requirements for Kazakhstan Regulatory Body Personnel were developed and adopted.

• Relative Position and Structure of Regulatory Body was prepared and serves for further development of the KAEC.

• Common Regulations on the Authorization of Personnel for the Right to Conduct Activities in the Field of Atomic Energy Use has been developed and adopted. The document serves as one of the most important national regulations in the field of personnel certification and authorization.

• Basic Requirements for Qualification, Selection, Training and Authorization of Personnel Involved in Activities Connected with the Use of Atomic Energy has been developed and provided to the KAEC.

• Regulation for the Authorization of Nuclear Power Plant Personnel by the Committee for Atomic Energy of Kazakhstan has been developed and provided to the KAEC.

• Common requirements for organization providing training for personnel involved in the field of atomic energy use in the republic of Kazakhstan.

• Regulatory guide on the development and conduct of authorization examinations of the personnel subject to the authorization by the atomic energy committee of the Republic of Kazakhstan,

• Quality management system description for KAEC.

• Glossary of terms in the field of nuclear facility personnel qualification, recruitment, selection, training and authorization.

All staff of enterprises and companies conducting activity in nuclear area, who are responsible for radiation and nuclear safety, must have examinations by the Atomic Energy Committee of Kazakhstan [3]. There are five categories of personnel:

— Management staff of nuclear entities and nuclear facilities, whose job description provides for direct responsibility for ensuring safety during the use of atomic energy. The authorization is granted for the right to safely manage the nuclear facilities.

— Personnel of nuclear entities and nuclear facilities engaged in the supervision and control of nuclear and radiation safety. The authorization is granted for the right to conduct activities on supervision and control of ensuring nuclear and radiation safety.

— Management staff of nuclear facilities, whose job description provides for both direct control of the nuclear facilities and responsibility for ensuring safety of the nuclear facilities. The authorization is granted for the right to manage the operation of the nuclear facility and to ensure its safety.

— Individual categories of operating personnel. The authorization is granted for the right to conduct routine operations at the nuclear facilities.

— Management staff of nuclear entities and nuclear facilities, whose job description provides for both inventory and control of nuclear materials, ionizing radiation sources, radioactive substances and wastes, and physical protection of nuclear facilities and nuclear materials. The authorization is granted to manage and conduct the activities to ensure both inventory and control of nuclear materials, ionizing radiation sources, radioactive substances and wastes, and physical protection of nuclear facilities and nuclear materials.
A. Kim et al.

A computerized system «KAEC Examiner» has also been designed during this time, developed and implemented for conducting authorization examination in the field of atomic energy use. This system is intended for examination of the personnel of the Kazakhstan facilities using atomic energy. Twelve sets of questions for testing competence on regulatory documents in the field of nuclear and radiation safety have been developed and installed into the computerized system. Now the system «KAEC Examiner» is used in KAEC for conducting examinations on personnel of entities including nuclear installations.

3. Conclusion

Currently, in the Republic of Kazakhstan, the state system of authorization of personnel involved in the nuclear activity has been established. Under IAEA TC Projects, a number regulatory documents in this field and computerized system «KAEC Examiner» for exams for personnel of entities including nuclear installations have been developed.

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[3] Common regulation for the authorization of personnel for the right of conduct of activities in the field of atomic energy use. KAEC, Almaty, 2000
APPLICATIONS OF PROBABILISTIC SAFETY ANALYSIS (LEVEL-3) IN NUCLEAR INSTALLATION SAFETY

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Abstract. The actual methods used in probabilistic safety analysis (Level-3) are demonstrated. The results of analyses for environmental consequences of hypothetical severe accidents at Paks Nuclear Power Plant (Hungary) are presented. The effects of countermeasures implemented according to Hungarian regulations are also investigated.

1. Introduction

The nuclear safety of Soviet designed reactors has become a major factor of world-wide. In Hungary the Paks Nuclear Power Plant consist of four VVER-440/-213 units, which were put into operation between 1982 and 1987. As the safety of these second generation VVER (water cooled water moderated power reactor type 213) reactor can hardly be assessed based solely on original documentation, these calculations were supplemented with the investigation of severe accidents in the form of project „Assessment of the safety of Paks Nuclear Power Plant from the aspect of high radioactive releases”. Using the results of probabilistic analysis (Level-2) [1] the environmental consequence calculations (PSA, Level-3) were performed applying the PC COSYMA code [2] recommended by EC. After adapting the code to the Hungarian circumstances (using site specific meteorological and population data) detailed probabilistic analyses were performed focused on radiological consequences. The impact of countermeasures (evacuation, sheltering, iodine profilaxis) introduced in accordance with Hungarian regulations and EC recommendations are also presented. As a result of these calculations it was verified that the international requirements are satisfied.

2. The Codes and data used

2.1. Level 2 PSA as a Starting Point of Level 3 PSA

The Level 2 PSA study has been basically performed [1] in accordance with the guidelines formulated in IAEA Safety Series No. 50-P-8 (1995). As a result of this project the release parameters calculated by MAAP code (nuclide specific activities, release duration, release point etc.) were the main input parameters of our environmental consequence calculations.

2.2. Environmental Models

To assess the radiological consequences the PC COSYMA program was used. The code previously adapted to Hungarian circumstances with site specific data on:

(a) meteorological parameters

The data (wind direction, Pasquill stability category, rainfall, wind speed) are based on the actual measurements at the Paks station of Hungarian Meteorological Service. From these one year’s data 144 sequences were used in our calculations.
The Hungarian population is about 10.5 million, with a quite uneven: slightly more than 2 million people live in Budapest. The data library of PC COSYMA is filled with actual population (using a grid of 10 km x 10 km) taken from the national census.

In the analyses of severe accidents, we have assumed that the countermeasures featuring in international recommendations and domestic (Hungarian) specifications are implemented in time and executed successfully. These countermeasures (and the condition of their implementation) are:

- **sheltering** (10 mSv advertable dose, during a period not longer than 2 days),
- **evacuation** (advertable dose of 50 mSv, for a period of not more than one week),
- **iodine prophylaxis** (100 mGy advertable collected dose resulting from radioactive iodines in the thyroid).

3. Consequence calculations

3.1. Accident scenarios

Using the results of PSA (Level 2) the environmental consequences were calculated for 13 hypothetical scenarios at full power and 2 scenerios for shutdown accidents.

Accidents at full power

- High pressure reactor vessel rupture – accident scenario 1
- Bypass – accident scenario 2
- Early containment rupture – accident scenario 3
- Early containment leakage growth – accident scenario 4
- Late containment rupture - accident scenario 5
- Late containment leakage growth – accident scenario 6
- Early containment rupture with sprinkler – accident scenario 7
- Early containment leakage growth with sprinkler – accident scenario 8
- Late containment rupture, sprinkler is operational – accident scenario 9
- Late containment leakage growth, sprinkler is op. – accident sc. 10
- Intact containment, sprinkler is not operational – accident scenario 11
- Intact containment, sprinkler is operational – accident scenario 12
- Partial core damage – accident scenario 13

Shutdown accidents

- Core melting accident after refuelling - accident scenario 14
- Core melting accident prior to refuelling - accident scenario 15
3.2. **Dosimetric consequences**

At the boundary of the exclusion zone – after taking into consideration the effect of protective measures – the early doses were calculated (see Table 1). Due to the effectiveness of the early protective measure (evacuation, sheltering and iodine prophylaxis) the average doses (averaged on 144 whether sequences) were below 0.4 mSv, but even the maximum doses were below 10 mSv.

The breakdown of the effective doses on nuclide basis was also calculated and summarised in Table 2. The early doses are dominated by the contribution from I-131. The next larger contributor is I-133 nuclide.

The percentage breakdown of early effective doses on an impact route basis is summarised in Table 3. The early doses comes mostly from the inhalation in the vicinity of the exclusion zone boundary. This is followed by the exposure from the groundshine and finally from the cloudshine. In the case of accident scenarios where the noble gas release is significant as against the iodine release (e.g. in the case of accident scenarios 9 and 13), the exposure resulting from the cloudshine becomes dominant and its contribution may even reach 85-89%.

The percentage breakdown of collected effective doses on an impact route basis is summarised in the following in Table 4. In the case of all assumed severe accidents at a distance of 3 km from the power plant, mostly the groundshine and the ingestion pathway are responsible for the late doses. Of the radionuclides released, mostly the cesium isotopes are responsible for the exposure.

The average and maximum risk rates (using the conditional probabilities) applying to each accident are shown by the Table 5. The risk factor is a value associated with a dose estimated at the boundary of the 3 km exclusion zone.

4. **Conclusions**

Without introducing protective measures, on the basis of the doses estimated at the boundary of the exclusion zone, the severe accident scenarios (on the basis of more conservative estimates) have been classified into 'consequence categories'. The estimated dose and the cesium release show a relatively good correlation, and hence in the future – unless there is a significant change in the release composition and in the release conditions (release altitude, duration), the new accident scenarios can be classified into these categories.

<table>
<thead>
<tr>
<th>Consequence category</th>
<th>Cs release ((A:Bq))</th>
<th>Early effective dose at the boundary of the exclusion zone ((ED: mSv))</th>
<th>Accident scenario</th>
</tr>
</thead>
<tbody>
<tr>
<td>I.</td>
<td>&gt;1E+17</td>
<td>ED &gt;1000</td>
<td>1</td>
</tr>
<tr>
<td>II.</td>
<td>1E+17&gt;A&gt;1E+16</td>
<td>1000&gt;ED&gt;100</td>
<td>2,3,14,15</td>
</tr>
<tr>
<td>III.</td>
<td>1E+16&gt;A&gt;1E+15</td>
<td>100&gt;ED&gt;10</td>
<td>7</td>
</tr>
<tr>
<td>IV.</td>
<td>1E+15&gt;A&gt;1E+14</td>
<td>10&gt;ED&gt;1</td>
<td>4,5,6,10,11</td>
</tr>
<tr>
<td>V.</td>
<td>1E+14&gt;A&gt;1E+12</td>
<td>ED&lt;1</td>
<td>13,12,9,8</td>
</tr>
</tbody>
</table>
L. Sági

The finding is that as a result of successful early protective measures, the risk of so-called acute effects is zero in every category. The so-called stochastic (late individual) risk – because the cancer diseases do not have an exposure threshold – involves some part of the domestic population in spite of the protective measures. As a results of these calculations it was verified that the international requirements are satisfied.

Table 1. The calculated early doses at the boundary of the exclusion zone
(taking into account the effect of countermeasures)

<table>
<thead>
<tr>
<th>Accident scenario</th>
<th>Early doses [mSv ] (7 days)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>average</td>
</tr>
<tr>
<td>1</td>
<td>0.40</td>
</tr>
<tr>
<td>2</td>
<td>0.30</td>
</tr>
<tr>
<td>3</td>
<td>0.30</td>
</tr>
<tr>
<td>4</td>
<td>0.30</td>
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<tr>
<td>5</td>
<td>0.30</td>
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<tr>
<td>6</td>
<td>0.20</td>
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<tr>
<td>7</td>
<td>0.30</td>
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<tr>
<td>8</td>
<td>0.16</td>
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<td>9</td>
<td>0.10</td>
</tr>
<tr>
<td>10</td>
<td>0.07</td>
</tr>
<tr>
<td>11</td>
<td>0.20</td>
</tr>
<tr>
<td>12</td>
<td>0.02</td>
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<tr>
<td>13</td>
<td>0.02</td>
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<tr>
<td>14</td>
<td>0.20</td>
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<tr>
<td>15</td>
<td>0.20</td>
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</table>
Table 2. The breakdown of early effective doses on a nuclide basis:

<table>
<thead>
<tr>
<th>Accident scenario</th>
<th>Nuclides</th>
<th>NG</th>
<th>Mo-99</th>
<th>Te-132</th>
<th>I-131</th>
<th>I-132</th>
<th>I-133</th>
<th>Cs-134</th>
<th>Cs-136</th>
<th>Cs-137</th>
<th>Ba-140</th>
<th>Other</th>
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<tbody>
<tr>
<td>1</td>
<td></td>
<td>16</td>
<td>30</td>
<td>20</td>
<td>4</td>
<td>8</td>
<td>8</td>
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<td>2</td>
<td></td>
<td>20</td>
<td>-</td>
<td>38</td>
<td>6</td>
<td>12</td>
<td>11</td>
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<td>6</td>
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<td>4</td>
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<td>9</td>
<td>-</td>
<td>33</td>
<td>2</td>
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Table 3. The percentage breakdown of early effective doses on an impact route basis

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Table 4. The percentage breakdown of collected effective doses on an impact route basis
Table 5. The average and maximum risk rates applying to each accident

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REFERENCES

THE MANAGEMENT SYSTEMS AND THE ROLE OF THE REGULATORY BODY

I.P. Salati de Almeida

Directorate of Institutional Management, Nuclear Energy National Commission – CNEN, Rio de Janeiro, Brazil

Abstract. Management System is a quite new managerial structure that deals with the organization as whole, integrating processes in the direction of the shared goals and objectives. Nuclear organizations have been able to adapt themselves to the new managerial and administrative trends and improvements driven by experience along the time. The role of the Regulatory Body has been well established, taking care about avoiding undue interference in the activities of the Operating Organizations. The organizational integration proposed by the Management System brings new risks about how to define the field of the Regulatory Body actions.

1. Introduction

The current trend in management theories consider the organization as a complete entity, where each compound interacts with the others and everyone contributes for a common aim, set by the mission and objectives of the organization. The so-called Management Systems include the analytical approach to identify, in a systematic way, the processes that contribute for the success of the enterprise. The strategic planning, the search for customer satisfaction and the respect to the society are characteristics that are present in these methodologies. It is also paid attention for the several external and internal partners of the organization called stakeholders. Stakeholders are customers, suppliers, the population and other individuals and organizations affected by activities and attitudes of the organization, which also includes the Regulatory Bodies. These features are shared in the distinguished methodologies that deal with the Management Systems. Management System can be defined as the set of philosophies, procedures and functioning rules expressed by documents that describe and establish the general control and the measures to be taken by the organization to succeed in their goals and objectives. These rules and procedures are to be applied to the whole organization. The Management System also establishes the responsibilities throughout the functional structure. Additionally, it establishes the mechanisms that are able to bring satisfaction to the various stakeholders of the organization. The Management System methodology is considered an important tool to provide the necessary means for adequate planning, accomplishment, evaluation and measurement of results.

Several specific methodologies are developed based on this kind of approach or others similar. Among them, some are very well known as the European Foundation Quality Management (EFQM), in Europe, the Malcom Balridge National Quality Award, in USA, and in Brazil, the Model of Excellence of the Brazilian Quality Award (BQA). All are systems which can be classified as Quality Management Systems, as defined by the documents ISO 9001:2000 [1][2].

Texts about the evolution of managerial systems, as Chiavenato [3], show that, since Adam Smith, new aspects have been introduced and changes have been added to the managerial systems as improvements. The managerial approaches have always been mutable and organizations have developed using distinctive management styles in different times. Therefore, it is difficult to say that there is a definitive way of doing organization administration. Nowadays, there is a reasonable agreement about some good characteristics for managerial systems, but, in fact, there are few
organizations that can be classified as absolutely similar when talking about managerial systems and styles. Some management styles, out of fashion, have been succeeding, in spite of not keeping the recommended formulas. Examples are styles centred in manager autonomy or intuition, some of which have shown themselves surprisingly effective, mainly in situations of very unstable and uncertain environment. The culture and some conditions of the places where the organization is installed can also play a major role to indicate the more adequate style to be adopted.

The nuclear area, fortunately, has obtained an extensive standardization in the main characteristics of its managerial systems. The main reasons are that it is strongly regulated worldwide and a great consensus was reached for a set of important subjects related to the area. This consensus was obtained by a high degree of co-operation and participation in international and national bodies of discussion, coordination and technical assistance, as IAEA, NEA/OECD, WANO, INPO and several others. Even though it is not obvious that a total standardization can be expected in every such aspects of the managerial systems, and each organization will keep some of its own characteristics in different countries and in different environments.

2. The nuclear area and the safety

The commercial nuclear area was constituted in the expectation of the use of a clean and compact energy source for the near future. At the beginning of the development process, as shown by Thompson and Beckerley [4], the accidents of research reactors pointed out the need of an outstanding and rigorous treatment for safety of installations. In the early 50’s the first Regulatory Bodies were created, inside other organizations, and gained strength and independence mainly in the 70’s. These bodies assumed the role of establishing the rules and requirements needed for the construction and operation of the nuclear facilities. Okrent [5] describes the way this process was fundamental in the USA, where the interest of private sector demands that rules were established. This was important to allow that the entrepreneurial activities could be carried on in a competitive capitalist environment. The evolution of nuclear area is marked by the fact that it has been open to incorporate the experience gained from the events and accidents. One example of this attitude was shown by the arrangements taken by NRC just after Brown Ferry’s fire as told by Hanauer [6]. Other efforts were made in the direction of the prevention of accidents. New instrument for evaluation of the accidents probabilities and consequences were developed, as WASH-1400 [7] in 1975, improving the experience gained from other fields of activities with the collaboration of private companies and various governmental bodies. The need for establishing strict systematic control made the nuclear area one of the precursors in stating the principles of the quality assurance, with the aeronautical area (Wells, [8]). The Three Mile Island accident showed up that the human factor and its relationship with the systems of control were an important compound in the occurrence of faults. The Bhopal accident created the term “culture of safety”, which appeared at the first time in the paper presented by Kharbanda [9], and introduced a new approach to the behavioural and managerial matters: the safety culture. The nuclear area, immediately after the Chernobyl accident, adopted the term and this new philosophy from the INSAG-4 document [10]. This continuous and steady evolution has shown that the nuclear area is able to accept, to incorporate and even lead changes in organizational and administrative systems. The needed for cooperation and for the standardization of concepts related to safety was showed after Three Mile Island (March 1979) and Chernobyl (April 1986) accidents. The negative repercussion that they caused in public opinion indicated that nuclear accidents, in spite of the reactor model or the country where they took place, would cause consequences for the future of the nuclear energy in the whole planet. The IAEA along these years established channels for discussion and technical cooperation that promoted the improvement of safety conditions. Its INSAG and Safety Series documents became references and guidance for the establishment of safety requirements by the Regulatory Bodies.

3. The role of the Regulatory Body

The uprising of the Regulator State in the past century was a needed step for the development of the modern industrialized democracy. The growing complexity and diversity of the societies and economies requested new models of government in the way that the concurrent interests could be balanced, new values attended and the externalities of the fast economic progress be managed (ENAP
Every agency or regulatory body is created in a specific area of activities with the aim of execute an external role allowing that some definite characteristics, desirable in the point of view of the society or government, can be reached and maintained. In some cases, these characteristics are related to the protection of the competitiveness and of the market, in others, to the safety and to the health of the public and the workers and to the environment. As states Kelman in [12]: “Many of the benefits of social regulation have no ready dollar value because they are not traded on markets.”...

“Most reasonable people agree that there is a place for markets in society, but most reasonable people also agree that market relationships have their costs as well”. Then, in some way, the Regulatory Bodies always interferes in the Operating Organizations acting. In the nuclear area, however, one of the main concerns of the Regulatory Bodies is to preserve the concept that the Operating Organization has the prime responsibility for safety. There are several diverse models for the acting strategy for the Regulatory Bodies, as discussed by Durbin et all [13]. These can vary from a lower to a greater prescriptive way of regulating activities, but even though the “operator prime responsibility” concept has an almost total acceptance and is also a fundamental statement in IAEA documents. The IAEA Safety Code GS-R-1 [14] states: "The prime responsibility for safety shall be assigned to the operator". The activities accomplished by the Regulatory Body has the objective of assuring that the Operating Organization is fulfilling the requirements to reach a good safety condition. The GS-R-1 lists the activities, all safety related, which the Regulatory Body has to do to accomplish its statutory duties. The code also determines: "When such functions are undertaken, care shall be taken by the regulatory body to ensure that any conflict with its main regulatory functions is avoided and that the prime responsibility of the operator for safety is not diminished". These principles have the objective of preventing that the action of the Regulatory Body can diminish the Operating Organization responsibility under any reason.

4. The Management Systems and the nuclear regulatory requirements

Many of the principles of the Management Systems are described in the documents ISO 9001:2000 Quality Management Systems- Fundamentals and Vocabulary [1] e ISO 9001:2000 Quality Management Systems – Requirements [2]. These principles are: Customer Focus; Leadership, Involvement of People, Process Approach, System Approach to Management, Continual Improvement, Factual Approach to Decision Making, Mutually Beneficial Supplier Relationship. Two of the main characteristics of the Management System are Customer Focus and Customer satisfaction. The organization is stimulated to find out what the customer needs are and to provide what the customer wants. The customers are not only traditional customer, but also can be thought as everyone who receives the various outcomes of the organization. Management Systems are also considered an instrument to enhance an organizational culture.

The document Safety Reports Series No. 22 [15] compares the differences between the IAEA Safety Standards related to quality assurance [16] and the documents ISO 9001:2000 [1][2]. The main differences are listed as: customer focus, internal communication, customer propriety and customer satisfaction. As the document describes: “...the objective of the IAEA Code is to establish basic requirements for quality assurance in order to enhance nuclear safety by continually improving the methods employed to achieve quality. ISO 9001:2000 specifies requirements for a quality management system that can be used for internal application by organizations, or for certification or contractual purposes. It focuses on the effectiveness of the quality system in meeting customer requirements”; or as it concludes: “ In summary, the IAEA Code 50-C-Q is focused on meeting the overall safety requirements for the plant, personnel and society in general, whilst ISO 9001:2000 is focused on satisfying the requirements of the customer”.

Interests as safety and profits, customer satisfaction and Regulatory Body satisfaction can be conflicting. In the licensing and control process, the Regulatory Body establishes minimal requirements for the action of the Operating Organization. The Regulatory Body acts only over the aspects that can affect the safety of the facilities. Then, what is the role of Regulatory Body if a nuclear organization is now using the Management System structure for its administration? What are the aspects of the Management Systems that will be considered requirements for regulation and control? What are the limits where the Regulatory Body will establish the requirements for the
Operating Organization Management System as done now for the Quality Assurance System? Has the Regulatory Body the right to interfere in the managerial style of the Operating Organization? How can the Regulatory Body assess if the Management System is really enhancing the organization safety culture?

Sorensen [17] analyses in an extensive way the role of the safety culture inside the evolution of the nuclear regulation and its importance for the additional assurance in prevention of accidents. He points out to the remaining difficulties for the assessment of the cause and effect factors and the regulatory arrangements that can be adopted. The safety culture inside the Operating Organization is still an open discussion, but as culture, its development and improvement rely mainly in the internal environment promoted by the direction of the organization. The role of the Regulatory Body is to assess the adopted system in order to assure that it fulfils the defined safety requirements. Camargo et all [18] suggest ways for the action of the Regulatory Body about the safety culture and also stress the care the Regulatory Body should take to avoid the interference in issues that are not safety related.

Lacoste [19] states it in a very explicit way: “The task of the regulator is to control, even in the field of safety culture: monitor what happens, from top to bottom of nuclear operations, and respond to deficiencies (...). There should be limits to regulatory involvement however: the regulator should not try to command operations or prescribe model organizations”. Therefore, the best way the Regulatory Body can do this job is assuring that the Management System establishes, in an unequivocal way, a priority for safety over all the other factors. Factors as profit, customer focus, customer satisfaction and other similar are not factors in which an interference of Regulatory Body must happen. The Management System must also establish that the Regulatory Body is a preferential stakeholder among all the stakeholders. One of the ways of assuring that the Management System has in fact a safety focus is to establish safety as the start point for the Management System. This could be similar to that presented by Obadia et all [20] for a nuclear research centre. However, even if the use of Management Systems methodologies or designs can be suggested as good practices examples, factors not safety related must not be object to a regulatory definition or requirement.

5. Conclusions

The adoption of the Management System by the Operating Organizations demands an adequate planning for the Regulatory Body action. The Regulatory Body must assess and check if the Management System provides the maximum priority for safety over all the other matters. It is not the role of the Regulatory Body to establish the way the Operating Organization has to run in subjects like customer satisfaction or profits.

The interference of the Regulatory Body in aspects of regulation other than those safety related will not contribute for the improvement of safety and will bring an additional undue effect of disturbance to the process of licensing and control of nuclear facilities. This interference also takes the risk of surpassing the legal statutory mandate and the technical competence of most of the Regulatory Bodies.

REFERENCES

I.P. Salati de Almeida


INTERNATIONAL FEEDBACK AND SAFETY REASSESSMENT -
THE FRENCH FBFC NUCLEAR FUEL FABRICATION PLANT
CASE

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\textsuperscript{b}Direction générale de la sûreté nucléaire et de la radioprotection, PARIS Cedex 12,
France

Abstract. With a ten year periodicity the French nuclear safety authority (DGSNR) and its technical support
(IRSN) reassess the safety of the plants under survey. A special part is devoted to internal experience feedback
and the events that occurred during the past working period are carefully analysed. In the case of the French
FBFC uranium fuel fabrication plant, it was decided to make a more systematic use of international feedback in
the usual reassessment work for non-reactor facilities. A program has been set up on the basis of peer-to-peer
level bi-lateral meetings organized with the European safety authorities having similar plant under their
responsibility. A special safety oriented standard questionnaire was previously sent to each and returned.
Each bilateral contact consists of a two-day visit, which began with discussions about particular topics in the
regulation area and methods for an efficient survey. In order to illustrate this, the two authorities visited the
nuclear site the second day. In this way, European safety authorities of five countries (Belgium, Germany,
United-Kingdom, Spain and Sweden) have met and plants belonging to FRAMATOME-ANP,
WESTINGHOUSE and ENUSA groups visited. All the contacted authorities participated and when, this
program was completed, it was possible for DGSNR and IRSN to get a more accurate measure of particular
reassessment work.
The main results of these actions were to "standardize" the consultancy's judgement, and taking advantage of
encountered good safety practices and good safety designs it was possible to reinforce specific requests during
the reassessment work. Moreover, it was an opportunity for the operators to participate in deepening information
exchanges about safety topics.

1. Introduction

For a long time the safety reassessment of NPPs has been the subject of a large exchange of
experience between European regulators. In addition the 58 French power reactors are characterized
by an overall standardization which offers the opportunity of a very efficient national operating
experience feedback. On the other hand, fuel cycle installations, as fuel fabrication plant, are in most
of the cases unique and the related domestic database comparatively reduced. So, it appeared to be
interesting to improve this with external sources, by mean of consulting other national nuclear safety
authorities. This has been recently done in France, for the safety reassessment of the FBFC uranium
oxide fuel fabrication plant.

During the last months of year 2001, a program has been set up; it may be presented as follow:
— general approach and principle,
— preparation and organization of the program,
2. General approach and principle

— bi-lateral meeting preferred to multilateral,
— peer-to-peer level (senior staff non directly involved in order to enhance free discussion),
— the host nuclear authority is the only interface with the French team,
— be aware of the visit to the nuclear site is not regarded as an inspection by operator,
— the visit of the plant and associated options are organised by the host authority.

A final meeting has been organized and took place in Lyon (France) in January 2004. All the involved European nuclear safety authorities attended this meeting and a synthesis of the information coming from the different visits has been presented, followed the day after by a visit of the FBFC manufacturing plant in Romans-sur-Isère.

3. Preparation and organisation of the program

3.1. The program

The initial choice has been to restrict the program to western European nuclear authorities, the American Japanese and Russian one remaining optional.

Six European countries have an established nuclear fuel fabrication plant: Belgium, Germany, United Kingdom, Spain, Sweden and France. Most of these facilities belong to Westinghouse or Areva/Siemens Group except the Juzbado case in Spain which is a subsidiary of the Spanish industrial group ENUSA but works using a Westinghouse licence. See in Appendix general information on the different visited plants.

3.2. Standard questionnaire

To make easier to undertake the final analysis and get a common outline, a “Standard questionnaire” has been established and previously sent (early months of 2002) to each nuclear authority. The main structure of this document roughly consists in two parts:

— a description including plant production functions (UF6 conversion, pelletization…) and main authorizations (production level, onsite nuclear material storage capacity, radioactive releases..),
— a set of more detailed questions related to different hazards usually reviewed during a reassessment operation:
  — nuclear material containment (ventilation principle, contamination level, workstation specially contained…),
  — criticality ( design principle, codes..),
  — radiation exposure (regulation, limit, improvement approach, target figure, ALARA..),
  — fire and explosion (prevention, intervention and training program, use of pure hydrogen or gas mixture in the process ..).

Special attention was paid to prevention, measurements and quality insurance program. Except in the case of criticality, the questions related to human factor were considered as more difficult to approach because its cultural sensibility and were not accounted for.
3.3. Organisation of the visits

Two days work program has been finalized for a five members French visiting group composed of: two from DGSNR, one from DSNR (regional bureau of the French authority) and two IRSN experts.

The following agenda has been proposed to the visited nuclear safety authorities and accepted:

The first day is essentially devoted to discussions with host authority representatives and the involved topics that have been generally discussed being:

— regulation principle and recent evolutions,
— nuclear Safety authority organisation and relation with technical support (if relevant),
— control and survey practices, inspection (site inspectors or not),
— practice of delegation of safety related decision to plant operator, safety contract,
— quality assurance controls,
— radiation protection.

By way of illustration, the second day was devoted to the visit to the facility, the French group being accompanied by at least one member of the host safety authority. For evidence, specialized technical discussions were spontaneous with met technical staff and safety managers.

4. Results

The visiting program started in July 2002 with the German safety authorities (Federal and Low Saxony) and arrived to completion in mid November 2002, with the Belgium authority.

4.1. Authority action

Considering specifically the authority point of view, the main interest was to compare organisation and practices to ours and highlight the topics that may be held of use in future improvement work. In this way, it’s possible to mention the following item:

— efficiency of safety control by authority and responsibility delegation to operator,
— the balance between technical arrangements and operator response,
— practice of site inspections, and safety reviews.

Concerning the public information, practices generally appeared to be very different from the French’s and it was estimated to be not possible to efficiently use, directly, these external references.

4.2. To the FBFC reassessment benefit

Once the program was over and taking into account the results of the received standard questionnaires analysis it was possible to identify some, purely technical or relevant human factor points to reinforce the reassessment work.

At first, the most important external feedback concerns nuclear material containment. This point is strongly related to radiation protection (internal dosimetry) and the following points has been particularly highlighted and finally taken into account within the FBFC reassessment framework:

— general workstation containment,
J.P. Carreton et al.

— safety arrangements of containment losses due to process design.

The autoclave used for the conversion process in each plant having this process stage, is the best solution to limit the risk linked to the hexafluoride uranium in liquid phase and will replace the ovens used in the FBFC plant.

This program equally shows that:

— it is possible to manufacture uranium oxide fuel using a mixture of gas in the sintering furnace, like that for MOX fuel manufacturing.

— assurance quality validation procedures when applied to codes (criticality) remains a difficult question

— considering the human factor question, the importance of
  — keeping and improving the competence level,
  — preventing workers turnover and ageing.

These latter points have been mentioned by some of the visited operators, the university-level student having less interest in nuclear industry in comparison to previous generation.

4.3. To the European safety benefit

This exercise showed that the approach is sometimes different in the European countries, especially in the case of:

— the gaseous and liquid discharge authorisations,
— the worker internal dose monitoring.

Technical exchanges on these topics would pave the way for a better understanding and would foster harmonization of practices.

5. Conclusions

The initial targeted objectives have been reached. Our initial judgement was often confirmed and some of the recommendations we formulated have been reinforced or justified by the foreigner experience feedback. These original objectives have been exceeded, plant visits and information exchanges among the industry’s representatives related to safety lead to define and share good practices.

The obtained results, at the end of this program, demonstrated the possibility of international contribution to installation nuclear safety improvement. However, we believe this could be further improved by:

— European harmonization of practices,
— recognition of these practices at the international level.
These latter points with the possibility of joint international inspections have been discussed during the final meeting we organized and, looking forward, they may be regarded as starting points for further international discussions or actions in the fuel cycle nuclear plant safety area.

Appendix

<table>
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(1) : mothballed

(2) : storage limit: 80 tons of powder, 50 tons of pellets and 400 assemblies
INTERNATIONAL LICENSING OF THE ACR-700

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Abstract. The ACR-700\textsuperscript{TM} is an evolutionary design suitable for deployment to meet generation needs in North America and Asia for in-service in 2012-2013. To reduce the project risk due to licensing, ACR-700 is currently undergoing licensing reviews, in parallel, in both Canada and the US. Successful completion of these reviews will result in a product that is licensable both in the country of origin and in the USA, and therefore comes as close to internationally licensable as practical. Canadian and US regulators are co-operating in their reviews, while respecting the different legal processes and requirements of each. As other countries become interested in ACR-700, this co-operation model can be expanded to facilitate their own review of ACR.

1. Introduction

The ACR-700\textsuperscript{TM} is an evolutionary adaptation of the CANDU 6 pressure-tube reactor, but using slightly enriched uranium fuel, and introducing a number of innovations to enhance reactor safety and economics.

Figure 1, from the plant CADD model, shows the reactor assembly.

In terms of economics, the use of slightly enriched fuel enables the use of a light-water coolant as well as a reduction in the size of the core and therefore in the volume of heavy-water moderator – the net effect being a 75\% reduction in heavy water volume. It also enables an increase in coolant pressure and temperature, which in turn increases thermal efficiency.

In terms of safety, the inherent safety characteristics of CANDU reactors have been combined with a passive large-capacity elevated water source to enhance both the prevention and mitigation of severe accidents. For example, in current CANDU reactors, the moderator can act as an emergency heat sink, and prevent fuel melting, for the combined failure of a LOCA with Loss of Emergency Core Cooling. This capability has been made passive in ACR-700. A large elevated Reserve Water Tank inside containment can add makeup water, at a rate sufficient to remove decay heat, to the reactor coolant system, the steam generators, the moderator, and the reactor vault surrounding the core, providing flexible tools to stop or manage a beyond-design-basis accident.
The immediate markets for ACR-700 are in Canada and the United States of America, since both countries forecast a need for new nuclear build coming in to service around 2012-2013; and in Asia, which is currently building new nuclear plants and could deploy advanced evolutionary designs in the same period.

To assure utilities that the project risk due to licensing activities is small, a pre-licensing programme has been undertaken in parallel, in both Canada and the USA. This paper describes the ACR-700 pre-licensing programmes in both countries.

2. Pre-Licensing in Canada

The Canadian Nuclear Safety Commission (CNSC), the regulatory body in Canada, is performing a pre-licensing review of the ACR-700 to assess the licensability of the design in Canada, in particular to “determine if the ACR-700 design meets the Nuclear Safety and Control Act licensing requirements in Canada for a nuclear power plant design; and if there are fundamental barriers that would prevent licensing of the ACR-700”.

2.1. Precedents

The CNSC Staff has previously performed (at AECL’s request) such a licensability review in Canada on the CANDU 9 design (an adaptation of the 900MW class of CANDU reactors to a single unit containment design). This review was successfully completed in 1997 January. After reviewing over 200 AECL reports, CNSC stated:

“[CNSC] staff conclude that there are no fundamental barriers to CANDU 9 licensability in Canada.”

3. ACR-700 Process

A similar process is being followed for the ACR-700. AECL has requested, and CNSC Staff has agreed, to perform an in-depth review of ACR-700 with respect to its licensability in Canada. The scope of documents submitted will be greater than that for CANDU 9, recognizing that in the interim regulatory standards in Canada have changed, and that the ACR-700 design is more advanced than CANDU 9. In particular, the scope includes, among other items, proposed safety design and safety analysis requirements, quality assurance programme (and CNSC Staff audits thereof), safety analysis methods and assumptions, descriptions of safety-related systems, a generic environmental performance basis, a safety analysis, a probabilistic safety assessment, supporting R&D programme, and an assessment of beyond-design-basis severe core damage accidents.

The result of the review will be issuance of a statement by CNSC staff on whether there are fundamental barriers that would prevent licensing of the ACR-700 design under the Nuclear Safety and Control Act. This statement will not constitute a licence, but will provide sufficient assurance to potential customers that should a licensing be initiated for particular site, there will be no fundamental barriers to licensing from the reactor design perspective. As with CANDU 9, a detailed report by CNSC will provide justification for the statement.

3.1. Status

The review is divided into two phases. The Phase I effort started in 2003 and concentrated on three areas: safety methodology, design requirements and research and development. It will be completed in September 2004. It has involved intensive familiarization of the CNSC with the design, focusing on 19 topics. The actual design information, flowsheets, etc, will be reviewed in Phase II. The CNSC pre-licensing review will be documented in a final report scheduled for completion March 31, 2006, with several key milestones and interim reports in each phase.
In parallel CNSC Staff have been reviewing the licensing basis of plants in Canada, with a view to bringing it closer to international practice, using the IAEA report “Safety of Nuclear Power Plants: Design” (IAEA Safety Standards report NSR-1) as a basis.

3.2. Pre-Licensing in the USA

The process of pre-licensing a design in the US involves a formal review and acceptance by the USNRC of the standard design (Standard Design Certification (SDC)). SDC may be preceded by an optional, less formal Pre-Application Review, which is used to identify and resolve major issues prior to an submitting an application for SDC. The ACR-700 is currently in pre-application review. This pre-application review (sponsored by AECL Technologies (AECLT), a wholly-owned US subsidiary of AECL) will be completed in September 2004. The pre-application review will address the scope, schedule, and cost of SDC; and licensing issues associated with the CANDU reactor technologies in ACR that depart from the light water reactor, pressure-vessel based regulatory framework in the USA. Therefore, during the course of the ACR pre-application review, major USNRC issues with the ACR design will be identified early and the scope of the work required to address these concerns, along with associated completion schedules, will be formulated and ultimately agreed upon with the USNRC.

AECLT has been informed by the NRC staff that the results of their pre-application review will be documented in a Safety Assessment Report (SAR), which will state whether there are any major impediments to licensing the ACR in the United States. In particular, the SAR should provide confirmation of the licensing criteria applicable to the ACR, provide an assessment of the completeness of AECL’s Research and Development (R&D) programs that exist or are planned in support of the ACR, provide an assessment of the suitability for purpose of the computer codes used in the safety analysis of the ACR, and provide estimates of the cost and schedule for the remaining scope of the USNRC’s efforts on the identified focus topics in preparation for a Design Certification review.

Successful completion of this pre-application review by the NRC will help to facilitate AECL Technologies to obtain a Standard Design Certification (SDC) for the ACR, and ultimately enable a utility to obtain a Combined License (COL).

The pre-application review of the ACR has been structured in two phases. During Phase 1 of the pre-application review (mid-2002 to August, 2003), technical overview meetings were extensively used to familiarize the NRC staff with the concepts underlying the ACR design, the existing technology base, and the scope of the available and planned analysis, testing and operational experience in support of the design. During Phase 2 (September 2003 to September 2004), the NRC staff are performing more in-depth assessments of the potential issues associated with the focus topics identified by AECLT for pre-application review. These focus topics generally cover the unique CANDU reactor genealogy aspects of the ACR design, as related to the NRC regulatory framework. NRC assessment work performed during pre-application will be carried seamlessly into the design certification review process.

3.3. Regulatory Co-operation

The ACR-700 is being licensed at the same time in two neighbouring countries on the same continent by two different, mature, and capable regulatory agencies. There is thus an opportunity for regulatory synergy in licensing the same product. Cooperation and coordination between the two regulators has the potential (while respecting the legal and jurisdictional requirements of each regulator) to ensure an effective and efficient regulatory review, make optimal use of regulatory resources, accelerate the NRC's familiarization with CANDU safety technology, share best regulatory practices, and assure the public on each side of the border that any plant in the neighbouring country meets acceptable safety requirements. The two regulatory agencies have therefore agreed to initiate a collaborative review program subject to some initial conditions and to the bounds of current international arrangements. To date, cooperation has consisted of:

- Attendance at public meetings in each other’s jurisdiction;
- Review of common or mostly-common submissions from AECL on technical topics;
• Exchange of views on ACR-700 issues;
• Joint representation at the meetings of the Expert Panel hired by the USNRC to develop a Phenomena Identification and Ranking Table (PIRT) for selected ACR postulated accidents;
• Discussions at executive levels in each organization; and
• Identification of technical champions for issues for which a common technical understanding is desired.

As other countries become interested in ACR-700, this co-operation model can be expanded to facilitate their own review of ACR. Both CNSC and the USNRC have had long histories of co-operation with fellow-regulators. Indeed the CNSC has been co-operating for many years with regulators of countries which operate CANDU, most recently and most extensively with NNSA on the Third Qinshan Nuclear Power Plant project. This cooperation is being extended with NNSA/NSC staff to participate in the ACR pre-application review.

4. Conclusion

Successful completion of pre-licensing will result in a product that is licensable both in the country of origin and in the USA, and therefore comes as close to internationally licensable as practical. The result will be a product that meets regulatory requirements both in countries that already use, and have licensed, current CANDU® reactors as well as those that use, and have licensed, LWRs.
IMPROVEMENT OF SAFETY IN REACTOR CONCEPTUAL DESIGN: A COST-EFFECTIVE APPROACH

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Abstract. Future nuclear power plants must be easier to build, simpler on their operation and maintenance, safer and more economic than the present ones, to be ordered by the utilities and accepted by the public. To achieve these objectives, safety aspects related with accident prevention and mitigation should be balanced with economy. The proposal presented in this work is that, in the conceptual design stage, economy and safety will be evaluated together through a feedback between the design parameters set and the criteria established for reactor safety through accident analysis, while the economical optimization process is running in order to improve a figure of merit related with cost. The methodology to perform this feedback is by means of Safety Design Maps, where safety performance indicators are correlated with design parameters, building an enclosed safety design space by means of proper restrictions, which are determined by design criteria or regulations. This methodology also allows the Regulatory Body to obtain a better insight of how safety aspects are covered through the safety performance indicators comparison with regulatory limits, which in turns gives a greater flexibility to the designers in contraposition when restrictions are imposed to the design itself.

1. Introduction

The electricity demand in the world will grow significantly in the near future. The competition among its supply sources is becoming fiercer and fiercer. Future nuclear power plants must be easier to build, simpler in their operation and maintenance, safer and more economical than the present ones, in order to be part of this new power supply demand—not to mention the replacement of the existing reactors reaching their lifetime. This is accomplished by being ordered by utilities and accepted by society. To achieve this objective the classical design methodology must be reviewed and new ones developed to perform nuclear reactor design. It is important to carry out this process with a global approach, contemplating the design feedback between all the systems and involved areas.

Reactor design is an intrinsically complex task, due to the quantity of parameters whose dimensions have to be determined and the existing relations between them. Nowadays the increasing computer capacity has made possible the development of computational tools as support to the design team to solve the complex calculations involved in reactor design, solving the relations between the design parameters and determining their dimensions in order to optimize the design to minimize the plant costs.

Although the current methodologies, classical or more advanced ones, like a steady state optimization, fulfil the requirements of design relative to safety, the lack of balance between economy and safety is evident. It is important to perform this balance with a global approach, contemplating the design feedback between all the systems and involved areas. Safety aspects are part of the most important contributors to costs, hence they must be considered in an efficient way. As other authors have already noticed, the new approach must consider new methods for cost-benefit and ALARA analyses, employing modern PSA techniques and fulfilling basic safety requirements instead of overly detailed prescriptions, with realistic models and assumptions [1].
1.1. Safety Analysis and Safety Costs

Though safety principles do never guarantee that nuclear power plants will be absolutely free of risk, when adequately applied, the plants should be effective in meeting society needs for useful energy. The most general nuclear safety objective is “to protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazard.”

In accordance with the safety objectives, some fundamental principles are applied, such as employing the safety culture in every stage of the design and plant cycle, or making use of the strategy of defence in depth. The regulatory bodies, which check and verify that the designs are safe enough, set quantitative safety targets with this goal. The design teams usually have their own design restrictions, including those for guaranteeing that the reactors will be safe and others related to economics and to the feasibility of being built. Both the design teams and the regulatory bodies apply the safety principles — and they must do so. However, there are often some unnecessary restrictions imposed, that prevent the improvement of the designs and their performance, without improving the safety response. A flow chart showing these and other contributions to the design restrictions is shown in Fig. 1. Formulating risk policy is a political issue, whereas implementing it is a technical one. Official bodies should (and must) set the rules to control and verify that things have been well done, by comparing the reactor safety response with the established safety targets. These targets should be as global as possible and with as few specific details as possible, so new designs find no unnecessary obstacles to be licensed. This will have two main effects: faster licensing processes and flexibility for enhanced designs. Both of them will assist the nuclear industry to satisfy society's main demands: safety and cost-effectiveness.

![Figure 1: Paths that contribute to the design restrictions.](image)

2. Proposal to internalize cost-effectively safety requirements

Considering the above-mentioned concepts, the following proposal arises for the conceptual global design process is [2], [3]:

1) **Preliminary conceptual design and qualitative optimisation based on designers’ judgement.** Stage based on designers’ expertise and research results, recognising alternatives that aim to simplify the design and to reduce initiating events and diminish their incidence, among other design goals for taking the soundest decisions. Different alternatives for safety and process systems are proposed at this stage, for being evaluated in the next one. Thus, the design basis is now obtained.
2) Integrated conceptual design and quantitative optimisation. This second stage consists of an integral design optimisation process in order to improve a figure of merit. To perform this, neutronic, thermal-hydraulic, mechanical, safety and economical dimensioning modules are required. Safety ones are used to simulate the plant performance in steady state and in transients or accidents and to characterise it by means of safety performance indicators. This evaluation is performed for each set of parameters that defines a possible reactor design that may be found during the optimisation process. Safety goals determined by regulators and designers are embodied in practical quantitative safety targets. They are applied as limits to the selected safety performance indicators and therefore considered as restrictions on the design parameters. Then the economic figure of merit is calculated given the main design parameter values. Finally, the optimisation gives a new set of parameters improving the value of the figure of merit. This stage is repeated until the design converges.

3) Final conceptual design stage based on experts’ judgement. Evaluating the alternatives results, the best design options are chosen [4]. Eventually, feedback to previous steps will be necessary.

The most innovative concepts of the present work are introduced in the second stage of the proposal. In order to face the posed design optimization problem, an objective is selected, a feature that is being analyzed and should be reached with the design. It is a result of the design parameters, which witnesses how good or bad a design is, in relation to the proposed goal. It is called figure of merit. Aiming at designing competitive nuclear power plants, adopted strategies may include the reduction of capital costs or other economic figures of merit. Several results of the design process can be selected as figure of merit for economical optimization. They are typically electricity generation cost, cost of investment by power unit ($/kw), total investment cost (releasing power as a parameter to optimize) and net present value of the project (assuming a known price of sale of the energy unit).

To verify reactor safety criteria fulfilment, the concept of safety performance indicators is introduced, also known as response functional. Each one of these variables is chosen in order to characterize and represent reactor safety levels or reactor degree of exigency during an accidental sequence. The idea is that for each accidental sequence, one or more indicators or observable variables can be defined. It is important to identify all the observable variables, which can be critical for assuring the reactor safety in every transient and postulated initiating events, because the success of the design will depend on the restrictions applied to them. Probabilistic limits are also supported by the methodology, included as further safety indicators. Probabilistic safety indicators, such as the core damage probability, can also be considered. Operational performance indicators are studied in reference.

There are also restrictions, which are limits that a particular design must fulfil and are applied to the design parameters as well as to the safety indicators. It is evident that the value of each safety indicator will be function of the design parameters. During the optimization process developed, while looking for an appropriate set of design parameters that optimizes a given figure of merit related with cost, safety indicators are compared with imposed limits. Should any of these limits be violated, the direction of the design parameters movement is changed in order to keep the reactor safe enough. Therefore, the safety indicators will be used to evaluate the safety degree and to determine the direction the design parameters must move towards, within the general scheme of optimization, as explained below.

Besides verifying safety criteria, the safety indicators can be also considered as a figure of merit to be improved instead of a cost related one. Cost-related or other design restrictions can either be considered or not, depending on the designers’ choice. For instance, this could be used to find a feasible design (one that does not violate any restriction) when some safety restrictions are being violated, for a posterior economic optimization inside the feasible design region. Other uses would be to search the safest design alternative for a given generation cost or the “safest limited-budget design”. The safety criteria fulfilment could be verified after these ALARA-like optimizations take place.

Considering then that the parameters dimensioning influences both the figure of merit and the safety indicators limited by restrictions, the concept of Design Map is reached [3]. A Design Map is a
representation of the safety indicator dependence in order to translate to the design parameters the restriction applied to the safety indicator.

3. OPTIMIZATION

This section describes how the methodology developed works. Having created the design maps (either a priori or online), the process goes on with the optimisation of the parameters that influence the selected figure of merit. Based on the design parameters, the value of the figure of merit, $M(\hat{x})$, is calculated in each step of the optimisation. Afterwards the partial derivative of $M$ with respect to each one of the design parameters –selected with sensitivity analysis– is evaluated. Thus, the optimal jump of the parameters vector is determined, parallel to the gradient of the function of merit. During the optimisation process successive $\delta x$ vectors are found approaching towards the point $\hat{x}$, in the $n$-dimensional space of the design parameters, whose function of merit is the best in the neighbourhood of the departure point. Throughout this process, the imposed restrictions to these indicators will be eventually reached. If in any step of the optimisation, for the new $\hat{x} = x_{\text{old}} + \delta x$, at least one safety performance indicator crosses its limit, it is necessary to reduce the parameter vector jump, in order to verify that $O(\hat{x}_{\text{old}} + \delta x) = \text{limit}$. In the next step, that restriction must be respected and, as the figure of merit improvement pointed to cross it, it is not convenient to go backwards. Therefore, the solution is to keep the observable variable constant and equal to its limit. This means that the parameters vector jump must be perpendicular to each one of the observable gradients that should be kept constant. In other words, $\delta x$ must have null components in the directions determined by these gradients. However, it is also required that $\delta x$ goes on being as similar to the vector determined by the search of a better figure of merit as possible. To do this, a subspace generated by the gradients of the observable variables whose limits were reached is built. Hence, for this subspace an orthogonal basis is obtained with the Gram-Schmidt orthogonalisation process. Next, all of its projections on each one of the components of this basis are subtracted from the parameters vector jump $\delta x$. Then the new parameters vector jump remains projected on the $(n-K)$ dimensional subspace, which is orthogonal to the space constituted by the gradients of the limited observable variables.

A diagram of the whole calculation paths in the global design process is shown in Fig. 2. In the bottom loop, an unrestricted optimisation path can be observed. In the upper part, the verification path for restriction fulfilment by means of the design maps is shown.

![Diagram](image-url)

Figure 2: Calculation diagrams. Figure of merit: reactor cost

Another advantage is that in case of need, one of the safety performance indicators can be used as figure of merit to be improved. This situation could occur if a safety related variable violates a restriction in a given reactor design. Cost could, or could not, be considered as a new observable subject to restrictions in the same way as the rest of the safety indicators are.
4. CONCLUSIONS

The present work presents a methodology to balance safety and economy of a nuclear power plant, aiming at achieving an efficient internalisation of the external costs. One of the main outcomes is that it is possible to optimise a reactor design internalising its safety costs efficiently. This process tends to cost reduction, a greater simplicity and a better strategy for prevention and mitigation. All of this is performed by integrating safety evaluation with neutronic, thermal-hydraulic and mechanical calculations in the design optimisation. This methodology provides the instruments necessary to be able to guarantee that the adopted criteria for reactor safety (restrictions applied to safety-related performance indicators) are verified in each one of the optimisation steps toward optimal cost.

Furthermore, a relevant issue is that the present methodology allows to incorporate reactor dynamic response during transients or accidents in an early engineering stage for design parameter integral optimisation, by using safety design maps. This is done through new rules for neutronic, thermal-hydraulic and mechanical calculations additional to those necessary for steady state dimensioning. This is a promising methodology for equalising and optimise reactor and safety systems design in an early engineering stage. Therefore, a balance between reactor inherent capability and safety systems to cope with the postulated initiating events can be achieved. This equilibrium prevents that the search for economic performance should cause less safe reactors and, likewise, guarantees the design competitiveness in spite of the unavoidable safety costs. Furthermore, by means of this methodology a simplified design can be obtained, compared to the resultant complexity when these concepts are introduced in a later engineering stage.

A further application could be evaluating the additional costs of higher safety levels, or those due to the uncertainties in the limits applied to safety performance indicators. If necessary, this methodology also allows to use one of the safety performance indicators as figure of merit to be improved. Cost may be then considered as a new observable subject to restrictions, in the same way as the rest of the safety indicators.

In addition, the developed methodology offers the possibility of handling probabilistic limits to avoid the occurrence of non-wished events, such as core melt probability. Moreover, the uncertainty treatment can also be handled, considering both the uncertainties in the design parameters (and their effects on the costs and on the safety performance indicators) and those due to the models used by the code. It is important to mention that this methodology does not replace the judgement of experts and detailed accident simulations must still be done in order to verify reactor safety. Finally, a great deal of work remains to be done in order to explore and to make concrete the potential benefits of the methodology. This is why there are some aspects that are described as a general concept or idea without giving explicit examples or specific detailed guidelines.

REFERENCES

SAFETY IMPROVEMENT PROGRAM NPP MOCHOVCE

J. Sádovský
Nuclear Safety Supervision Department Mochovce NPP, Mochovce, Slovakia

Abstract.

- Introduction
- External Review Missions
- Technical Specifications of Safety Measures
- Main project modification: general, component integrity, systems, instrumentation & control, containment, electrical power supply, external risks, internal hazard
- Conclusion

1. Introduction

- The Mochovce NPP is located in the south-west part of Slovak Republic between the towns Nitra and Levice, approximately 120 km from Bratislava.

- The Nuclear Power Plant consists of four reactor units of WWER 440/V-213 type.

- At present two units are in operation, first one since July 1998, second one since December 1999.

- The third and fourth units construction finishing should be decided by the government of Slovakia.

- As these units represent second generation of WWER reactor design, the additional requirement to enhance operational and nuclear safety according to recommendation of performed international audits and operational experience based on exploitation of the similar units (as Dukovany and V-2 Jaslovske Bohunice NPPs).

- These requirements result into a number of safety measures grouped according their purposes to reach recent international requirements on nuclear safety and reliability.
2. External Review Missions

Pre-OSART Mission (IAEA) date: 9 - 29 January 1993. focus: Review of Pre-Operational Safety at Mochovce Nuclear Power Plant

- Mochovce NPP Safety Improvement Review Mission (IAEA) date: 5 - 13 May 1994, focus: Review of safety improvements proposed for Mochovce NPP
- IAEA Mission - Review of Seismic Safety of Bohunice and Mochovce NPPs. date: 31 October - 4 November 1994. focus: Seismic safety of NPP
- RISKAUDIT - Review of Mochovce NPP Safety Improvements. date: 20 December 1994. focus: Safety assessment of the project

Technical Specifications of Safety Measures (TSSMs)

The following documents were used as a background for the elaboration of the Mochovce NPP Safety Improvement Programme:

- SAFETY ISSUES AND THEIR RANKING FOR WWER 440/213 NPPs
- IAEA WWER-SC-108

as the generic document for all WWER 440/213 NPPs type.

Other two documents, which concern of the Mochovce NPP:

- Mochovce NPP Safety Improvements Review Mission
- IAEA WWER-SC-102
- Mochovce NPP Safety Improvement Evaluation
- RISKAUDIT Report No. 16

These documents were used for the performance of the TSSMs in 1995 as a basis for the Mochovce NPP Units 1&2 Safety Upgrading Programme and these were used for contractual negotiations.

The list of Safety Measures (SMs) was prepared by Nuclear Research Institute VUJE Trnava, Inc., together with Mochovce NPP staff.
Different items were grouped into the areas of technical interests with the precise definition of SMs.

Individual works consist of analytical and design parts and these works were performed before the plant construction finishing. List of contractors was created - fulfillment of the works in the frame.

8 SMs - Rank III, significant impact to the safety
41 SMs - Rank II, the safety is affected
25 SMs - Rank I, deviation from recognized international practises
13 SMs - Not ranked

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TOTAL: 88 42 41 22 25 15 13

87 SMs - Total number of SMs for the Mochovce NPP Units 1 & 2 Contractual negotiations started in October 1995 with participants involved in the completion project:

SKODA Prague, (Electrical & Mechanical part)

Energopunkt Prague, (Main Designer)

Hydrostav, (civil part)

EUCOM - Consortium Siemens/Framatome

(Implementation of SMs)

Atomenergoexport - Russian Organisations

EdF (Assistance to Project Management & Quality Control)
a. Nuclear safety improvements

Main project modification

- to make a comparison between the Russian safety codes OPB, which were the basis for construction of VVER-440/213 NPP with NUSS requirements
- to document an accordance of the Decree 436/90 Coll. with international standards

G 02: Equipment qualification

- As per NUSS 50-C-D, chapter 12, the qualification of safety-important equipment shall prove equipment ability to fulfil required functions. The qualification requirements are applied for normal operational states, emergency conditions, and for internal and external risk factors.

G 03: Reliability analysis of safety class 1&2 systems

- The analyses was performed in order to confirm system reliability according to the Designer’s assumptions. The reliability analysis of the safety class 1&2 systems and components was crucial as the first step of a global PSA analysis.

b. Nuclear safety improvements

Main project modification

COMPONENT INTEGRITY

- updated programme of verification samples for RPV embrittlement addition of NDT (non destructive testing) in accordance with EU practices
- requirements on NDT resulting from LBB application
- introduction of the „anticipated failure“ approach
- qualification of inspections, equipment, and personnel
- use of up-to-date methods and equipment
- application of LBB procedure to the NSSS piping
- implementation of the analytical part
- three diagnostic systems with required sensitivity supplied as a part of I&C 06
- both operating units have statute of LBB at NSSS piping as well as connecting piping to the pressurizer
- safety improving HW modifications of SG
  - replacement of lids of primary as well as secondary SG headers
  - replacement of Ni seals with graphite ones
  - replacement of SG feedwater distributions with stainless steel
c. Nuclear safety improvements

Main project modification

SYSTEMS

S 01 - NSSS protection against cold over-pressurisation
S 04 - Qualification of PRZ relief and safety valves for water

GOAL:

reduction of probability of initiating events resulting in cold over-pressurisation transients

Modifications implemented:

- implementation of an automatic monitoring system able to prevent pressure thermal shocks and to guarantee the reactor pressure vessel integrity
- modification of pump control (make-up pumps, HP ECCS pumps, HP boron concentrate pumps)
- S 04: modification of PRZ SV - valve qualification for water and steam-water mixture

d. Nuclear safety improvements

Main project modification

S 05: Risk of clogging the water collectors on SG room floor

Scope of modifications:

- reconstruction of all collectors on the floor with enlarged flow area of sieves and their protection against impact of water flow in case of surrounding piping rupture
- addition of a collector sieve clogging monitoring system on the floor of the SG room in LOCA mode

The hardware modifications consist of:

- installation of a new anchorage of the filters,
- installation of two independent water level measurement systems for each ECCS system

S 13: Emergency feedwater system vulnerability

GOAL: re-routing of emergency feedwater lines from the risky area of the electrical building at floor + 14.7 m

- take into account all the three EFWS systems cannot be impacted by flooding and confirmation of seismic resistance of all EFWS components and their resistance to impacts of other non-seismic components

Modifications implemented:

- construction of a new seismic gallery housing the new lines out of the electrical bldg.
Installation of a connection to a mobile equipment for back-up water feeding to SG in the room below tanks with a special mouth piece to the mobile equipment pressure hose

- installation of Sempell valves in EFWS room (bldg. 810/1- 01) from pumps
- new solution of a common seismic-resistant channel for EFWS and heating water and chill piping at the lines upstream of unit 1&2 nuclear island

f. Nuclear safety improvements

Main project modification

S 15 : Hydrogen removal system

GOAL:
- risk analysis of hydrogen accumulation in some containment areas
- installation of the hydrogen elimination system - 16 hydrogen recombiners inside the containment
- installation of 8 hydrogen concentration measurements inside the containment

Hydrogen recombiner

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g. Nuclear safety improvements Main project modification

INSTRUMENTATION & CONTROL

IC 01 : I&C reliability
IC 02 : Safety system actuation project
- reactor AO (automatic protection) system structure did not ensure execution of the „Trip“ command in all modes
- the system was resistant neither to the „single-criterion failure“ nor „common-mode failure“
- organisation of emergency power supply system could be the source of false actuation
- New system implemented - Russian Fed. delivery
- New structure proposed
  - new equipment hasn’t got the shortcomings and meets the requirement „safe during failure“

**IC 01: I&C reliability**

**IC 02: Safety system actuation**

- Project
The following diagnostic systems have been installed at Mochovce NPP as a part of the safety improvement with the aim of meeting the LBB (leak before break) requirements:

ALUS 3.00 - the system works on the principle of acoustic signal sensing, which propagates via the NSSS construction due to the leak. The sensors are placed on the main circulation pipes, PRZ, SG, and RPV.

AKFLUS - the system works on the principle of activity measurement in immediate environment of the components monitored. Humid air from the crack area is sucked through suction openings of a special tube rolled around the monitoring piping and ended in a measurement equipment evaluating presence of N-16, F-18 and humidity. Sampling points are determined by the suction tube route which covers the reactor coolant piping, PRZ, and connecting piping.

i. Nuclear safety improvements

Main project modification

HUMON - diagnostic system of humidity monitoring. It consists of a set of suction tubes, through which the air is running from the SG room area, RCP platform, and reactor shaft to the measurement probes. The system contains 19 sampling points, one of which provides monitoring of RPV lid tightness.

All the three systems are independent, and based on on various physical principles. Sensitivity and reaction time of the systems meet the LBB requirement for NSSS leak monitoring.

- sensitivity of LBB systems - 3.8 l per hr.
- CS LBB - Centralised system has been designed as a computer extension that gains data required for its activity through a communication with the unit information system (MADAM-S).
- Other diagnostic systems, which are not a part of LBB: KUS95, SUS95, FAMOS, DAKEL.
- RCP monitoring system, which was a part of the original design, has also been installed. Nuclear safety improvements

Main project modification

IC 09 Instrumentation for accident monitoring (PAMS)

Scope

Selection of parameters with a focus on qualification requirements of measurement chains and their scope was done in accordance with the US standard RG1.97 .

- Implementation level measurement in the SG room (two levels)
- temperature measurement at fuel assembly outlet with extended measurement scope up to 1200°C.
- redundant measurement of reactor level

k. Nuclear safety improvements
Main project modification

IC 10 / OP 13 Technical Support Centre, Emergency Centre

- **Safety measure scope**
  - Technical Support Centre (TSC) under the AB (administrative building) bldg.
  - Emergency Control Centre (ECC) is also located under the AB bldg. in a civil defense shelter.

Off-site back-up ECC has been established in the building of the Environmental radiation monitoring laboratories in Levice.

IC 10 / OP 13 Technical Support Centre, Emergency Centre
1. **Nuclear safety improvements**

*Main project modification CONTAINMENT*

- calculations of the bubble-condenser (BC) strength, based on guidelines prepared by IAEA
- analysis of BC construction resistance - resulting BC loading defined in the scope of the SM CONT 02 - 04 (conservative loads for most of the design-basis serious accidents - large LOCA, damage of MSL, feedwater lines)
- implementation of BC strengthening - anchorage of the 1st and 12th floor beams, ensuring the 1st floor against lifting, strutting of troughs
- development of a special testing programme - simulation of LOCA 500 mm
- analysis of LOCA consequences
- test of the containment integrity

*The PHARE Programme (No. PH 2.13/95) confirmed that the Mochovce NPP bubble-condenser system assessment procedure was correct and results of experiments are identical. Strutting of troughs*

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**m. Nuclear safety improvements**

*Main project modification*

CONTAINMENT - Hermetic Zone
n. Nuclear safety improvements

Main project modification

EXTERNAL RISKS: EH 01 - 03

- Seismic resistance of Mochovce NPP (EH 01)
- Extreme meteorological conditions (EH 02)
- Explosions (EH 03a)
- Aircraft crash (EH 03b)

Nuclear safety improvements

Main project modification Seismic reinforcement

- Seismic characteristics (seismic spectrum) of NPP Mochove site - were specified before startup (Original HPGA=0,06g, new HPGA = 0.1g)
- Based on the seismic characteristics- realized seismic reinforcement within the scope of safety measure EH 01 (external hazard). Buildings and equipment reinforced for 0.1 g;
- Buildings: reinforcement of light-weight walls, DGS buildings, emergency feedwater system, steel structure in the auxiliary building etc.
- Equipments: reinforcement of footing and anchorage, axial piping compensation of emergency systems, modifications of suspensions and anchorage, replacement of SIAZ (seismic monitor. system)

Project „Probabilistic seismic assessment of NPP Mochove site“

- In scope of the project has been realized amount geologic a seismologic work (particular results in 2001) .Co-operation with the Slovak Academy of Sciences: probabilistic assessment of the site endangering
- The seismic monitoring network continuously monitors seismic activity of Mochovce surroundings

Main project modification

ELEKTRICAL POWER SUPPLY:

- EL01 - Start up logic for the emergency diesels
- EL02 - Diesel generator reliability
- Protection signals for emergency diesel generators
- EL04 - On-site autoconsumption power supply for incident and accident management
- EL05 - Batteries discharge time
- EL06 - Enhancement of autoconsumption off-site power reliability
- EL07 -Enhancement of autoconsumption power reliability

Main project modification

INTERNAL HAZARD

IH 02: Fire prevention
Safety goal: Establishment of an adequate fire protection level in accordance with NUSS 50-SG-D2, meeting the requirement of the fire protection concept stage 1 - prevent fire.

Measures implemented:

- Redundancy and fire protection of cable lines was improved by replacement of Dexaflam with a new type fire spray Flammplast KS-1 in the reactor and auxiliary bldg., connecting bridges, venting stack, electrical bldg., DG station, turbine hall.
- The fire barriers were implemented in accordance with the existing design; repairs of installed barriers were done.
- Smoke-tightness of existing fire doors was improved by a metallic batten with a foaming tape.
- Fire flaps qualified for 90 minutes were replaced.

Fire protection of a system cabling with FLAMOPLAST KS-1 cover. Nuclear safety improvements

Main project modification

IH 07: Internal risks caused by high-energy piping rupture

Safety measure scope

The analysis of the high energy pipe break effects covered the following steps:

- identification of weak and possible break points and analysis of the impact on the safety, identification of safety components to be protected against mechanical and hydraulic effects resulting from analyses
- pipe break (mechanical and electrical components, ESFAS instrumentation),
- protective measures wherever needed.

4. Conclusion Nuclear safety improvement programme

re-assessed by IAEA Mission (1998)

and RISKAUDIT (1998-99)

"Mochovce Nuclear Power Plant is the first Soviet-design nuclear power plant completed in an East European country that achieved a safety level comparable with western standards."
THE NATIONAL INFRASTRUCTURE ON RADIATION SAFETY OF AZERBAIJAN

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Abstract. In this article the questions of the national infrastructure on radiation safety of the Azerbaijan Republic are considered. Government bodies, which are implementing the control of nuclear safety in the country, are showed. There are given the offers for strengthening and maintenance of normal activity of Regulatory Authority.

1. Introduction

The Azerbaijan Republic actively participates in efforts of the World Community on maintenance of observance within the framework of corresponding obligations, the international mode of nuclear safety and takes steps, directed on its strengthening. The politics of our state in this area is based on principles of the international cooperation and openness. Azerbaijan is going to develop widespread cooperation with IAEA which member we became in May, 2001. In 1998 Azerbaijan has signed Agreements on application of guarantees and the report enclosed to it. In 2000 our state had been signed the Additional report to the Agreement on application of guarantees. From its part Azerbaijan has accepted the Law «About radiation safety of the population» and a number of other statutory acts. Now there are held works above «Basic Norms and Rules of Radiation Safety in the Republic of Azerbaijan», which are the main part of a national infrastructure.

1.1. IAEA Projects

A major landmark in development of attitudes of our state with IAEA became the visit of the Director General of the International Agency, Mr. Mohamed ElBaradei, to Azerbaijan on May, 2002. During a meeting with the Director General the President of the Azerbaijan Republic has noted great value of cooperation with such authoritative international organization as IAEA and has put a number of the important questions, which could be solved with the help of the Agency. With a view of the further expansion and a deepening of communications of our Republic with the Agency, the President of Azerbaijan with the Decree from September 5 of 2002 year has created the state commission on cooperation between the Azerbaijan Republic and IAEA. The head of this commission is one of the First Vice-premiers of the Azerbaijan Republic.

Today the priority directions of our cooperation with the Agency are improvement of radiation system’s protection and safety of the radiations sources used in the industry, agriculture and medicine. We have submitted IAEA project on improving the ways of radiological treatment in the field of oncology, struggle against illegal use of radioactive materials and other projects, the majority of these projects have been realized within the framework of the program of technical cooperation between

166
Azerbaijan and IAEA, in 2002 and the first quarter of 2003 the only project of technical cooperation carried out now has the name « Perfection of an infrastructure of radiation protection and safety of radioactive sources». The Agency helps Azerbaijan in strengthening its infrastructure radiation protection and creation of the adequate system of control over the radiations’ sources with the basic norms and safety rules of the Agency.

The reforms, which are spent by the government, are carried out within the framework of the wide program of the technical help. In this context we consider the future program of Technical cooperation as a unique source of knowledge in the field of peace use of nuclear technology and it will promote the decision of important problems facing to Azerbaijan according to national priorities and the coordinated document in which should take into account potential and an opportunity of a national nuclear infrastructure.

The list of the important questions, which we wanted to be solved, includes:
1. Creation in republics the only regulatory authority and strengthening of a legal infrastructure;
2. Modernizations of services in the field of nuclear diagnostics and radiotherapy;
3. The manipulation with radioactive waste products;

2. Strengthening Regulatory Body

In our opinion strengthening and maintenance of normal activity of the independent effective regulating body allocated necessary personnel resources and an infrastructure, is the precondition for safe and appropriate applications of nuclear technology and radiations in the country and a condition for technical cooperation with IAEA in this area. In this respect the Azerbaijan Republic will continue the work above the finishing and acceptance of all corresponding legal infrastructure with a view of strengthening structure and potential effective and functional regulatory authority.

For the maintenance of strengthening the effective national regulating infrastructures, which can carry out functions of independent regulating authority, will need an establishment of the intermediate purposes. It will allow us to carry out completely the Basic norms and safety rules and to execute the international requirements, in particular, concerning supervision and the control over radiation safety, inspections and licensing, concerning the boundary control over illegal circulation of radioactive and nuclear materials in conformity with is international recognized as practice and norms.

According to the general attitude of a society to factors of an environment and technical development, the purpose of the radiation protection is maintenance of a sufficient level of the people’s health protection will derive benefit thus, brought by sources of radiation and a nuclear energy.

The concept of radiation protection always was the basic effective radiation protection, should is in the consent:
- With modern knowledge of biological influence ionizing radiations;
- With modern general approaches of a society to health protection before factors of technical development and an environment;
- With versatile needs of modern and future practice, i.e. should take into account all situations of an irradiation of people, which arise or can arise and find for them the decision.

The nuclear infrastructure of the Azerbaijan Republic is based on componental bodies including:
• State Commission on cooperation between Azerbaijan and IAEA, created according to the Decree of the President of the Azerbaijan Republic from September, 5, 2002;

• State Committee for Industrial and Mining Safety (Gosgortehnadzor) - according to the law is the regulating authority responsible for safe transportations, storages and uses of radioactive materials and sources in the country and independent state control over all kinds of activity in the field of radiation safety;

• The Azerbaijan Republican center of Hygiene and Epidemiology within the framework of Ministry of Health, according to the law is the regulating body responsible for sanitary inspection and the regulating control over the using of the ionizing radiations sources including the problem of sanitary-and-hygienic passports.

• The ministry of National Safety bears the responsibility for the lost radioactive sources, inventory and the control of all sources of radiations, counteraction to illegal circulation, maintenance of physical protection and transportation of waste products;

• The ministry of Ecology and Natural resources, which submits directly to the President of Republic, bears the responsibility for development of programs on preservation of the environment and management of them.

At figure 1 is showed control implementing government bodies.

In Azerbaijan, there are currently about 350 installations and establishments of radiation sources. There are maintained more than 900 x-ray devices, point for a burial place of radioactive waste products, 30 laboratories with open radioactive sources, used mainly for medical diagnostics, and also 22 sources scale of the radiations used for medical, agricultural, industrial and exploratory purposes.
State Committee on Safety Work Control in Industry

- Using ionizing radiation sources with the purpose of scientific-investigational and practical-constructive works
- Designing, preparation, disposition, building, production and removal of a production the plants considered the sources of ionizing radiation, storing radioactive substances, warehouses of radioactive waste
- Preparation and designing of radioactive safety means, technological equipments for these plants, stations and warehouses
- Mining, production, processing, transportation of radioactive substances and works with for using these substances

Issues special permission in Radiation Practice Area

- Publishes instructions on safety technique
- Determine rule of control and implement measures on ensurance of radiation safety in enterprises and entities

Ministry of Health

- Sanitary passport
- Issues special permission for storage and burial of radioactive waste
- Implements daily control (3 times a day) on radiation background of the environment monitoring of pollution of soil and reservoir

Ministry of Ecology and Natural Resourses

- Publishes sanitary-hygienic norms, implements fulfillment check-up of these norms

FIGURE 1
CONTROL IMPLEMENTING GOVERNMENT BODIES
Abstract. In the latter half of 1990s a series of incidents occurred in Japan such as MOX fuel inspection data falsification, Monju fast breeder reactor sodium leakage accident, Tokai nuclear fuel plant (JCO) criticality accident and so on. It is thought that existing measures based on nuclear technology are not well cope with those incidents and another countermeasure utilizing new methodology of cultural and social sciences was keenly felt by both administration agencies and nuclear industries. Above all, the technique such as risk communication to inform the influence of trouble correctly and convincingly to the residents and mass media and to prevent the harm due to rumor is obviously inevitable. Based on these circumstances, Japanese NISA (The Nuclear and Industrial Safety Agency) initiated in 2002FY new project by open application in the field of cultural and social sciences, and risk communication was one of the principal subject of study. Up to now, 6 risk communication studies are currently in progress. The project was taken over from NISA to JNES (Incorporated Administrative Agency Japan Nuclear Energy Safety Organization) since 2004FY. This paper shows the overall structure of the project and the outline of the running studies.

1. Background

Following a series of incidents occurred in the latter half of 1990s as is described in “Abstract”, the scheme of study by open application was initiated in 2002FY for introducing newest knowledge of cultural and social sciences.

Japan has been establishing and managing safety regulations in order to deal with the hardware troubles arisen since the incipient stage of nuclear power plant introduction, yet above mentioned troubles, which gave serious influence on the recognition and acceptance of the general public in Japan, were not well coped with by current measures concentrated on hardware improvement, and hence the need of introducing new methodology of cultural and social sciences was keenly felt by both administration agencies and nuclear industries.

It is the first and unique trial in Japan that the regulator (NISA) invites various research institutes to participate in the study contest on nuclear safety in the field of cultural and social sciences.

2. Structure Of The System

2.1. Objectives

The aim of this scheme is to strengthening administration’s comprehensive evaluation and judgment and, as a consequence, to improve the safety of nuclear power plant by enhancing the knowledge basis in the field of cultural and social sciences including risk communication in addition to existing nuclear technology.
2.2. Contents of system

In order to cope with the nearly happened safety issues insoluble by nuclear technology, the necessity of amplification of knowledge basis is widely pointed out in the field of cultural and social sciences, natural science such as geoinformatics and seismology.

Colleges, research laboratories and institutes are invited to annual open application as a three-year project in the field of cultural and social sciences, natural science and so on.

In selection, the committee consists of well-knowledged specialists is organized and examines the proposed theme fairly and impartially from the standpoint of the contribution to safety and reliability improvement of nuclear facilities. The committee also evaluates the results of the selected studies, and judges the executability in next year.

In the field of cultural and social sciences, JNES performs the integrated study incorporating the result of the individual selected study and proposes the plan to contribute the improvement of nuclear energy safety basis.

2.3. Project Scheme

2.4. Work Schedule

<table>
<thead>
<tr>
<th>Research Assistance Program by Open Application</th>
<th>Fiscal Year</th>
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<tr>
<td></td>
<td>'02</td>
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<tr>
<td>1. Study by Open Application for Enhancement of Knowledge Basis</td>
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<tr>
<td>2. Integrated Evaluation (Proposal to NISA on Nuclear Safety Administration Work and so on)</td>
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\[\text{(N): Sequence of Work}\]
3. Outline of Study on Risk Communication

3.1. Status of Open Application

In the area of risk communication, 2 and 4 applications were selected out of 13 and 15 candidates in 2002FY and 2003FY respectively. Among 6 adopted applicants, 4 are proposals from universities and the rest are a research institute and a corporation.

3.2. Outline of Study

Followings are the outlines of studies on risk communication.

a) Development of Risk Communication Technique under the Uncertain Situation

Applicant: Koshien University

First aim of the study is to establish the method of risk communication from a standpoint of 'fairness' through clarifying the standard of risk communication process felt fair by recipient who informed on the subject of nuclear power and other issues which make the general public feel uneasy. Second aim is to develop a training program for the sender (administration and/or company) to promote the skill of communication. The outcome is expected to improve the risk information method by administration and company.

b) Pilot Research Project of Risk Communication on Nuclear Technology and its Utilization toward Communication and Collaboration with Community

Applicant: Central Research Institute of Electric Power Industry

This study is to prepare a practical guide of risk communication indispensable to the safety regulatory utilizing risk information. In this study, the staff conducts risk communication experiments with administration, residents and company on some risk issues associated with nuclear technology and its utilization in Tokai village of Ibaragi Prefecture. They make up practical guidelines for process design, operation and its implementation utilizing the experience and knowledge obtained through experiments, and then they clarify social effects of risk communication activities.

c) Study on Communication System of Social Risk Information on Nuclear Energy

Applicant: Kyoto University

This study is to enhance the safety activities by company and to improve the effective releasing method of safety information promoting the risk acceptance of residents. The effectiveness of this new risk communication method is verified through the internet web virtual networks named 1) “community network” for fostering safety culture and 2) “socially opened communication network” discussing the risk problem of high-level waste disposal.

d) Study on Public Relations and Residents Participation toward Improvement of Risk Literacy

Applicant: Center for Environmental Information Science, Incorporated

This study aims to improve the safety education, safety regulations and public relations. From the standpoint of improvement of risk literacy, company’s public relations are investigated. Through the construction activities of education program to promote industry risk communicators and risk communication practice with residents, NPO and risk communicators, risk information improvement method is proposed and verified.
e) From Safety to Peace : Study on Risk Communication Support Method

Applicant: Chiba Institute of Technology

This study aims to raise right safety consciousness of local residents through the realization of bilateral communication and the buildup of mutual trust. Residents near nuclear power plant are invited to join in policy exercise (simulation activities in the virtual space of personal computer to exchange views and build consensus). Through this experience, residents become responsive to the need of evacuation training and raise sense of security as consequence.

f) Investigation and Study on Publication of Safety Evaluation Information and Risk Communication in Various Stages of Nuclear Power Plant

Applicant: Chiba University of Commerce

The staff investigates the process of effective mutual consent among stakeholders (residents, industries, administrations, mass media and opposition faction) who have different behavior and common purpose to achieve safety. The guide for creating mutual consent is prepared through this investigation and then verified by the simulation utilizing the game theory.

4. Conclusion

Each study by open application is currently in progress as a three-year project, and JNES integrates and evaluates the results of the selected studies from the point of the application to the nuclear safety administrative activities. In 2005, workshop on risk communication is planned to open for making the results known to everyone concerned.

ACKNOWLEDGEMENTS

The project has been mainly carried out by NISA and excellent co-operation with NISA in preparing this paper is kindly acknowledged.

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USE OF PROBABILISTIC SAFETY ASSESSMENT IN JNES

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Abstract. In Japan, Nuclear Safety Commission (NSC) proposed Safety Goals [1] and stated a policy to examine the usefulness of Risk Informed Regulation (RIR) [2]. Nuclear and Industry Safety Agency (NISA) started discussion to adopt RIR reflecting the NSC policy statement [3]. On the other hand, Nuclear Power Engineering Corporation (NUPEC) has developed the Probabilistic Safety Assessment (PSA) methodology. The NUPEC’s activities were succeeded by Japan Nuclear Energy Safety Organization (JNES). The risk information derived from PSA has already been utilized to some kinds of regulatory activities for utility’s voluntary works such as the implementation of Accident Management (AM). Under these circumstances, JNES will keep feasibility studies to support NISA to systematically adopt risk informed regulation.

1. Introduction

Importance and usefulness of PSA was recognized at the onset of TMI-II accident. NUPEC has developed the PSA methodology and studying the use of PSA to the regulatory activities since 1989. Since then the risk information derived from PSA has already been utilized to some kinds of regulatory activities of NISA for utility’s voluntary works such as the implementation of AM [4,5] and PSA in Periodic Safety Review (PSR) [6]. PSAs conducted by NUPEC have also been utilized to discuss the adequacy of allowed outage times [7], test intervals and inspection intervals, and to make decision on the appropriateness of the countermeasures for the rupture event of the steam-condensing mode piping at Hamaoka Unit-1 [8]. Thus, the usefulness of risk information derived from PSA has been gradually recognized in Japan.

In 2003, NSC proposed Safety Goals and stated a policy to examine the usefulness of RIR. NISA started discussion to adopt RIR reflecting the NSC policy statement. The NUPEC’s activities were succeeded by JNES when it was established in 2003. Under these circumstances, JNES will keep feasibility studies to support NISA to systematically adopt risk informed regulation.

This paper summarizes the status of use of PSA, which JNES conducts to support NISA.

2. JNES Activities to Evaluate the Utility-Submitted PSA

JNES, conducting PSAs independently of utilities, contributed to the review of utilities’ voluntary work on the implementation of AM, PSA at PSR, the revision of technical specifications mainly focusing on the appropriateness of the changes in AOT of the safety systems. In addition, NUPEC, using the PSAs, supported NISA at the regulatory decision-making on the appropriateness of the countermeasures for the rupture event of the Steam-Condensing Mode (STCM) piping, which occurred on Nov. 2001 in Hamaoka-1.

2.1. Accident Management

In July 1992, NSC recommended the regulatory body and utilities to conduct PSA and implement AM. The utilities conducted level 1 & 2- PSAs on all NPPs, and submitted their results to MITI (Ministry of International, Trade and Industry, currently Ministry of Economy, Trade and Industry(METI)) at the end of March 1994. MITI and the technical advisory committee in support of
NUPEC (currently JNES) reviewed the results submitted by utilities. The review report was submitted to NSC in October 1994. NSC reviewed and admitted it to be approvable in November 1995. Utilities implemented AM for operating and constructing NPPs. The implementation reports on AM were submitted to NISA in May 2002. The effectiveness of the AM for severe accidents was evaluated by level 1 & 2-PSAs. NISA in support of NUPEC reviewed these AM implementation reports including the effectiveness of the AM measures on Core Damage Frequency (CDF) and containment Conditional Failure Frequency (CFF) using the PSAs conducted by NUPEC independently of utilities. In the review process of the effectiveness of AM, both PSAs were compared and the differences were analyzed between utilities and JNES. The review reports by NISA were submitted to NSC in October 2002. This report concludes that AM is effective to reduce CDF and CFF of domestic LWRs.

2.2. **PSA at Periodic Safety Review** MITI requested utilities to carry out PSR in 1992. Utilities voluntarily introduced PSR for the safety enhancement activities under close deliberation with MITI. Since the third PSR, utilities have conducted PSA at power operation. Furthermore, since the seventh PSR, low power and shutdown (LP & SD) PSAs were included in PSR to assess the safety feature of their NPPs and to confirm the appropriate safety margin during the annual maintenance. Utilities deliberately examined the maintenance procedures and countermeasures of events that might occur during shutdown operation. The information gained from these activities was appropriately reflected to utilities’ LP&SD PSAs. NISA reviewed the utilities’ LP & SD PSAs in support of NUPEC, where the procedure guide on LP& SD PSAs, prepared by Atomic Energy Society of Japan (AESJ), was used. In the same way as the AM review, in the review process, both PSAs were compared and the differences were analyzed between utilities and NUPEC. The review results were submitted to NSC in August 2002. The report concluded that PSAs for LP& SD operation by utilities sufficiently conform to the procedure guide for LP&SD PSA and these PSAs can show the characteristics of plants on the viewpoint of LP& SD operation.

2.3. **PSA based technical specification change; AOT**

The quantitative PSA perspectives were utilized to the technical specifications, especially to determined AOT of the safety equipment. NUPEC developed PSA based approach using allowable risk criteria for AOT evaluation and trial evaluations of AOT were conducted using this approach. NUPEC tentatively set Incremental Conditional Core Damage Probability (ICCDP) value and calculated ICCDP during safety related system outage period as follows;

<table>
<thead>
<tr>
<th>Safety criteria for risk increment for one outage</th>
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<tr>
<td>Allowed incremental conditional core damage probability (ICCDP)</td>
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<tr>
<td>Allowed incremental conditional large early release probability (ICLERP)</td>
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<th>Safety criteria for annual risk increment</th>
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<tr>
<td>Allowed annual increment of core damage frequency (CDF)</td>
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<tr>
<td>Allowed annual increment of large early release frequency (LERF)</td>
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</tbody>
</table>

2.4. **PSA application for the countermeasures for the rupture event of the steam condensing mode (STCM) piping**

On November 7, 2001, a pipe rupture occurred in the steam condensation line of the residual heat removal system at the Hamaoka Unit-1 while operating at rated power, resulting in steam release with radioactivity into the reactor building, when the high pressure coolant injection was forced to be unavailable state.

The cause of pipe rupture was found to be the inadvertent ignition of accumulated hydrogen in the pipe high point. Although this incident was of no safety significance, the type of rupture was unprecedented in Japan. Thus, the regulatory body investigated the cause of this event and corresponding corrective actions, the short term one and long term ones, by using PSA. The short-term action is the removal of non-condensable gases accumulated in the steam condensation line at the monthly surveillance test. Long-term action is the installation of normally closed valves, which prevent the accumulation of non-condensable gases, or the dismantlement of the steam condensation
lines. Then NISA examined the effectiveness of corrective actions for preventing recurrence and issued the report describing the investigation results.

At the NISA’s request, NUPEC evaluated the safety significance of the incident and the effectiveness of corrective actions from the risk point of view by carrying out PSA. The results by NUPEC showed that the incident has no risk significance and that the devised individual corrective actions would not lead to any significant increase in CDFs (Figure 1). This study is epoch-making because this is the first PSA application case to the regulatory decision-making process in Japan.

3. **Application of PSA Information to Inspection**

According to the RIR policy of NISA, the most urgent issue to be studied is improvement of inspection of nuclear power plants. JNES has already made feasibility studies on the categorization of structure, system and component (SSC), risk-informed in-service inspection, performance indicators and significance determination of inspection findings (SDP), using perspectives gained from PSA. Based on these studies the followings are the candidates of the risk informed improvement.

1. Enhance the effectiveness and efficiency of the inspection system by;
   - Identifying the SSCs, which should be inspected under the annual regulatory inspection, by referring to the risk importance of SSCs.
   - Prioritizing the piping segments to improve the effectiveness of the in-service inspection.
   - Optimizing the inspection intervals considering the risk perspectives.

2. Monitor & assess the plant safety level and the effectiveness of the inspection system through;
   - Observing the performance indicators (PI).
   - Significance determination of inspection findings (SDP).
These types of indicators and their threshold values may be determined utilizing the risk information.

(3) Enhance the accountability through the quantitative risk criteria.

(For example; Criteria for FV and/or RAW, ΔCDF, ΔLERF, etc.)

4. Concluding Remarks

Based on the above PSA-related experiences of JNES, JNES will support NISA to adopt the RIR concept in the nuclear regulation in accordance with NISA’s approach [9]. In order to support NISA, JNES will firstly investigate the areas where the RIR concept can be introduced, and, after that, will draw perspectives of the nuclear regulation with the RIR concept and the roadmap. The most urgent issue to be studied now is improvement of inspection of nuclear power facilities. In parallel, JNES will contribute to (1) establish safety performance objectives, which are compatible to the safety goals, evaluating seismic risks of existing nuclear power plants through a seismic PSA methodology, and (2) standardize PSA methodologies in academic society.

REFERENCES

TOWARDS SAFETY CULTURE STRENGTHENING

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Abstract. Through the amendment of the Reactor Regulation Law in 1999, following review of the JCO criticality accident, the Nuclear Safety Inspection System was established and the Safety Preservation Rules were refined in an attempt to further improve nuclear safety maintenance. According to the Safety Preservation Rules, the safety preservation activities at nuclear power plants must be built upon safety culture. Under such circumstance, the TEPCO issue was revealed in 2002. Triggered by this issue, NISA has been implementing a variety of improvements, one of which involves the establishment of a study group in 2003, to discuss on how to implement safety culture sufficiently and possible recommendations. Subjects such as the followings piled in the report will indicate leading keys in case it is going to realize such efforts: “Foundation of safety culture is a quality management” and “Realistic and scientific technique is necessary for the evaluation of safety culture”. In order to respond to these requests, JNES has been developing an Integrated System for Monitoring & Evaluation of Safety Culture, of which concept is structured by 3 elements: (1)“System for safety (formal structure, method, and activity)”, (2)“Safety attitude and behavior of organizational members” and (3)“Sharing of safety attitude and behavior of organizational members”. This paper delivers the background of the development, illustrates a representative early sign of declining plant performance extracted by a root cause analysis of human and organizational factors, and introduces the Integrated System examined and verified from various viewpoints.

1. Early sign of declining plant performance

Fig.1 shows the trend of plant performance, and recently among the reported incidents, the frequency of incidents caused by human errors has been kept less than 0.1/reactor year. However, an operator error caused reactor scram at a nuclear power plant of TEPCO in 2001. The reactor was tripped by the protective signal of high neutron flux, because the operator continued manipulation of control rods without carefully monitoring the reactivity under the process approaching to the reactor critical. This kind of incident was the first experience in Japan after we began to accumulate the data of root causes. JNES analyzed the root cause of this incident from viewpoint of human and organizational factors and confirmed latent factors in the safety culture, which proceeded from administrative factors, organization factors and individual factors.
It is a prevailing concept that signs of declining safety performance will appear when a week safety culture exists for a period of time. Among them, an operator error in reactivity manipulation typically designates one of the early signs of declining safety performance at a nuclear power plant\(^1\). Based on this concept, this incident indicated the sign of a week safety culture of the enterpriser. In 2002, just after this incident, the weak safety culture of the concerned enterpriser was revealed by the scandal of falsification and concealment of voluntary inspection data.

2. **Outline of the study group report discussing on how to implement safety culture**

Triggered by the TEPCO issue, NISA has been implementing a variety of improvements. One of them was the establishment of a study group in 2003, to study on how to implement safety culture sufficiently and possible recommendations. Since the issue of falsification and concealment is covering various domains, this study group invited experts from other fields as well as from nuclear-related industries.

The study group report\(^2\) is compiled with three subjects, the subject to be considered by the nuclear industry, the subject to be considered by the government and regulatory bodies, and the subject to be considered by the general society containing the mass media.

In addition, this report put a special note about the role of the regulatory bodies related to safety culture. The primary party directly responsible for the realization of safety performance shall be a nuclear enterpriser. The regulatory bodies' function is to offer suitable support, in order to have a good harvest for an effort of the enterpriser to such safety culture strengthening, and it is necessary for regulatory bodies to warn themselves against being trapped in a mind-set which makes safety culture stiffen by regulation. It is important for regulatory bodies not to be negligent on the status of safety culture, including its framework, of nuclear enterpriser and to improve communication with them aiming at improvement in safety culture. Here again, safety culture should be cultivated by every person, from the top to the bottom of the related organizations. All of them should concentrate daily constant efforts and there is no end in the process for lasting safety culture.

2.1. **Issues to be considered by nuclear industry**

(1) On Stage of Policy Level Commitment
- The clear message of giving top priority to safety is made to permeate to the end of an organization.
- It is important for the strengthening of safety culture to be addressed as a part of organization management activities (corporate governance) by the top management.

(2) On Stage of Implementation
- Improvement of an organizational climate and the safety activity by the top of organization
- The enhancement of motivation, and the confirmation of the importance of safety section
- Quality management as foundation of safety culture
- Accountability for the general public
- Improvement in a decision-making person's knowledge and Cautions against group-thinking.

(3) On Stage of Verification and Evaluation
- Enhancement of a self-evaluation function and a periodical self-evaluated report is essential for the senior management of an organization to strengthen safety culture.
- The necessity for realistic and scientific methods for monitoring and evaluating safety culture.

2.2. **Issues to be considered by regulatory body**

(1) On Stage of Policy Level Commitment
- The reform to rational and effective regulation, structure, and implementation

(2) On Stage of Implementation
- The implementation side of inspection and audit needs to establish the structure which evaluates and rationalizes the quality and quantity.
- Practical use of the knowledge about safety culture
- Accountability for the general public
- Regulatory bodies’ rethinking, sense-of-values conversion

(3) On Stage of Verification and Evaluation
- Training, education and self-evaluation of regulatory bodies
M. Makino

2.3. Issues to be considered by mass media, government and public

- All the sides to generate, transfer, and receive information should recognize that the peculiarity of radiation, not being visible, may give the general public a very big uneasy feeling compared with an actual risk.
- It is important for the media persons concerned to strive for mutual understanding with the everyday nuclear persons concerned.
- The educational administration persons concerned etc., are expected to make more opportunities for the general public to learn safety culture so that they can have sufficient informations that enables judgment about nuclear power by themselves.
- Substantial researches from social science-viewpoints, such as risk communication need to be achieved. Moreover, it is important that special consideration is paid concerning development of safety culture also from the field of educational administration or research administration (for example, support for the development of improved tools for monitoring and/or self-evaluation of safety culture, research on the role of the media in the process of risk communication, such as distortion induced in the society).

The activities, which aim at cultivation and enhancement of safety culture, have just stood on a fresh restart point rather. Each subject is believed that it will indicate a leading key in case it is going to realize such efforts.

3. An integrated system concept for monitoring & evaluating safety culture

The above mentioned study group reports the necessity of realistic and scientific methods for the evaluation of safety culture as one of leading keys to strengthening safety culture. In order to respond to these requests, JNES has been advancing to develop an Integrated System for Monitoring & Evaluation of Safety Culture\(^2\) of which concept is structured by three elements to monitor and evaluate safety culture as Fig.2 shows. For monitoring and evaluating safety culture from the viewpoint of the element I, a subsystem, OR model (ORM) system was developed, and from the viewpoint of the element II and the element III, another subsystem, SCEST system was developed.

In order to demonstrate the appropriateness and practicality of the SCEST, it was applied to 9 industrial organizations and the data were collected from a total of 1246 persons, including 134 managers, 196 supervisors and 916 workers\(^3\).

All scores of 36 basic items evaluated for managers group of an organization for example, can be easily and clearly unfolded on the visualized SCEST-MAP\(^3\) as shown in Fig.3. We can identify the score of a basic item circled on the map is lower than the others from the viewpoint of both scores of the gap deviation score \(Z_{\text{gap}}\)\(^3\) and the item deviation score \(Z_{\text{item}}\)\(^3\). We can focus on this basic item and analyze the score in more detail. Consequently we can identify a factor of the low score results from the basic item concerning to “recognition of improvement”.

The results of those examinations verified that the SCEST system is powerful system in order to evaluate and monitor the cultivation level of safety culture of an organization.

The ORM is constructed of the risk generating process model and the risk incubation process model. This model was applied to organizational accident and issues of JCO, Ontario Hydro NPP and TEPCOissue and factors specified in risk generating process and risk incubation process were verified to comprehend all factors of organizational accident and issues applied. The ORM contains 79 evaluation items. These items evaluate the system for safety (formal structure, method, and activity) which means the element-I of the three elements concept.

FIG.2. The Integrated System Concept by 3 elements
By using the items of the SCEST and ORM, it is possible to evaluate organizational safety culture systematically. Then the 79 items of ORM and 36 items of SCEST construct the baseline for evaluation of safety culture and in other words they should be specified as the inspection items. We selected these 115 items carefully and aimed at making them more substantial: 1) the overlap items were specifically unified, and careful selection of inspection items was performed; and 2) the correspondence relation between these inspection items and demand matters of ISO 9001 (2000 version) were clarified, and the important evaluation items which are not included in the present inspection items were positively adopted as new inspection items. As the result, the provisional safety culture inspection items were extracted. In order to evaluate the validity of the safety culture inspection items extracted by the above-mentioned examinations, verifications were performed by analysis of and comparison with the followings.

- The safety culture (and safety control) evaluation criteria in INSAG-13⁴ and INSAG-15⁵ of IAEA
- The factors of the accident pointed out by the root cause analysis report about the organization accident phenomenon of Davis-Besse nuclear power plant in the U.S.⁶
- The results of analysis for the space shuttle Colombia accident⁸ and the Milestone nuclear power plant⁹

4. Conclusion
The results of the above examination specified the evaluation criteria of nuclear safety culture which consists of 131 items, 12 evaluation categories, and 91 sub categories.

REFERENCES

STUDY ON INSPECTION CAPABILITY OF ULTRASONIC TESTING FOR FATIGUE CRACK IN PIPING

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\textsuperscript{a}Japan Nuclear Energy Safety Organization, Tokyo, Japan
\textsuperscript{b}Ministry of Economy, Trade and Industry, Tokyo, Japan

Abstract. Ultrasonic testing (UT) technique is widely used as an in-service inspection (ISI) method, to assure the integrity of nuclear safety power plant components. It is understood that detection and sizing are the key roles of inspections, because evaluation of the components necessitates such data as flaw size and distribution of flaws. Therefore, reliability or accuracy of the inspection is most important. However, data on inspection capability in terms of detection and sizing had not been necessarily sufficient. In Japan, comprehensive research program has been carried out, in order to verify UT performance on both detection and sizing since 1995. In this program, various types of specimens with realistic flaws were manufactured to be used in UT trial for detection and sizing with multiple inspection teams. They were eventually destructed for flaws to be investigated, and detectability and sizing accuracy of UT were evaluated. In this paper, results of UT verification test regarding to carbon steel pipes specimens with fatigue cracks are presented. The results of the tests, it was verified that UT had relatively high performance on flaw detection and sizing for fatigue cracks.

1. Introduction

It is necessary to detect flaws, to measure the flaw sizes accurately and to conduct the quantitative evaluation of the crack propagation, by the method of the fracture mechanics considering the plant operating conditions; to assure the integrity of the structure through the plant life.

The new regulation, the JSME Codes on fitness-for-service for nuclear power plants, on nuclear power plants was instituted in Japan in October, 2003. The concept of an allowable flaw size was introduced in the Codes. Therefore, it is important to detect and to size the flaws, especially flaws that are larger than the allowable flaw size. In Japan, a comprehensive research program has been carried out, in order to verify UT performance on both detection and sizing for the specimens simulating components (Piping, Pressure Vessel, a nozzle corner, etc.) to be inspected by ISI. The specimens have fatigue cracks and SCC. The objective of this program is to give guidelines that may enable the Japanese regulatory body, MITI, to make a proper judgment on inspection results.

In this paper, the results about the detectability and the accuracy of the sizing (the flaw length, the flaw height) by UT for the fatigue cracks located in the vicinity of welds in carbon steel piping, are reported.
2. Verification Test

2.1. Preparation of specimens

The configuration of specimens is shown in Figure 1 and Table 1. All the specimens were fabricated by applying TIG and SMAW welding. The flaws were the mechanically induced fatigue cracks located at the inner surface of specimen. The real sizes of fatigue cracks were investigated in a destructive test after UT measurements had been completed (Table 1).

![FIG. 1. Carbon steel pipe mock-up with fatigue cracks](image)

<table>
<thead>
<tr>
<th>Specimen size</th>
<th>The number of flaws</th>
<th>Flaw size</th>
</tr>
</thead>
<tbody>
<tr>
<td>OD (mm)</td>
<td>T (mm)</td>
<td>&gt;allowable flaw size</td>
</tr>
<tr>
<td>165.2</td>
<td>10</td>
<td>7</td>
</tr>
<tr>
<td>355.6</td>
<td>25</td>
<td>5</td>
</tr>
<tr>
<td>508.0</td>
<td>35</td>
<td>4</td>
</tr>
<tr>
<td>609.6</td>
<td>50</td>
<td>8</td>
</tr>
</tbody>
</table>

2.2. UT measurement

The UT test for detection and sizing were carried out by multiple inspection teams with blind test condition. UT was performed with direct contact technique using 45 degree transducer. Flaw length was measured based on echo height, with cut-off level of 20, 50 and 100% DAC (distance amplitude compensation). Tip echo techniques and time of flight diffraction (TOFD) techniques were used for flaw height measurement.

3. RESULTS AND DISCUSSION

3.1. The detectability of flaw by UT

We investigated the detectability of UT for the fatigue cracks. The detection rate for the cut-off level (20%DAC, 50%DAC & 100%DAC) by UT is shown in Table 2. It was found that all the flaws larger than the allowable flaw size are detected by the cut-off level of DAC20%.

<table>
<thead>
<tr>
<th>Specimen size</th>
<th>Cut-off level</th>
</tr>
</thead>
<tbody>
<tr>
<td>OD (mm)</td>
<td>T (mm)</td>
</tr>
<tr>
<td>165.2</td>
<td>10</td>
</tr>
<tr>
<td>355.6</td>
<td>25</td>
</tr>
<tr>
<td>508.0</td>
<td>35</td>
</tr>
<tr>
<td>609.6</td>
<td>50</td>
</tr>
</tbody>
</table>
3.2. **Sizing Accuracy by UT**

3.2.1. **The relation between the flaw length by UT and the actual flaw length**

The relationship between the flaw length by UT and the actual flaw length is shown in Figure 2. By comparison of three data, 20%DAC method showed best performance in length sizing, of which RMS error was 5.85 mm.

![Diagram showing flaw length by UT for DAC20%, DAC50%, and DAC100%, and the actual flaw length.](image)

- **Cut-off level: DAC20%**
  - Mean error: 2.67 mm
  - RMS error: 5.85 mm
  - Coefficient of correlation: 0.93

- **Cut-off level: DAC50%**
  - Mean error: -0.77 mm
  - RMS error: 6.08 mm
  - Coefficient of correlation: 0.89

- **Cut-off level: DAC100%**
  - Mean error: -3.73 mm
  - RMS error: 7.70 mm
  - Coefficient of correlation: 0.87

*FIG. 2. The flaw length by UT for, cut-off level DAC20%, DAC50% and DAC100%, and the actual flaw length*

3.2.2. **The relation between the flaw height by UT and the actual flaw height**

Sizing accuracy of flaw height by UT for the four kinds of specimens is shown in Table 3 and Figure 3. Sizing accuracy for all sizes of specimens is good (RMS error < 2 mm). The accuracy of measuring the flaw height by tip echo technique was similar to that of TOFD technique. Moreover, both techniques show good correlation between actual and measured flaw height. The height of all the flaws larger than allowable flaw size was measured.

**Table 3. Measurement accuracy of flaw height by UT**

<table>
<thead>
<tr>
<th>Specimen size</th>
<th>Tip echo technique</th>
<th>TOFD</th>
</tr>
</thead>
<tbody>
<tr>
<td>OD (mm)</td>
<td>T (mm)</td>
<td>Mean error</td>
</tr>
<tr>
<td>165.2</td>
<td>10</td>
<td>-0.35</td>
</tr>
<tr>
<td>355.6</td>
<td>25</td>
<td>-1.74</td>
</tr>
<tr>
<td>508.0</td>
<td>35</td>
<td>-0.17</td>
</tr>
<tr>
<td>609.6</td>
<td>50</td>
<td>-0.22</td>
</tr>
</tbody>
</table>
4. Conclusion

The results of this study examining the detectability and the sizing accuracy by UT for the fatigue cracks located in the carbon steel pipes are summarized as follows.

(1) The flaws larger than allowable flaw size could be detected by the cut-off level of DAC20%.
(2) The accuracy of measuring the flaw length by UT is fairly good by cut-off revel of 20%DAC (mean error: 2.67 mm, RMS error: 5.85mm).
(3) The accuracy of measuring the flaw height by echo technique and TOFD technique is relatively good (tip echo technique: mean error -0.34mm, RMS error 1.15mm) (TOFD technique: mean error -0.28, RMS error 1.10mm).

We have accumulated the key data on the inspection performance, that can be basis of regulatory decision making. We have already prepared an interim inspection guideline based on data obtained to date. In addition, to make the final guideline, additional evaluation is being carried out.
TOPICAL ISSUE 4:
LONG TERM OPERATIONS –
MAINTAINING SAFETY MARGINS
WHILE EXTENDING PLANT
LIFETIMES
PERIODIC SAFETY REVIEW OF THE BUDAPEST RESEARCH REACTOR

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Abstract. The Hungarian legal system obliges operators of a nuclear facility to prepare a safety review every decade (periodic safety review – PSR), the main goal being that the PSR should serve as one of the basic items for renewing the operation license of the facility. In 2002 the regulatory body issued us with a guidance document to carry out a PSR. This document very clearly defined the requirements. The general philosophy of the PSR was that the 10-years’ operational record should reflect the present condition of the reactor, and that the adequacy of the reactor systems and subsystems should be verified with special consideration to the next 10 years of operation (lifetime estimation). All operational experiences including event records were required to be reviewed; moreover, all aspects of human resources were required to be checked and evaluated with particular regard to safety. On the basis of these requirements we carried out a safety review of the Budapest Research Reactor (BRR), and prepared a report. The regulatory body accepted our PSR Report and in November 2003 renewed the operation license of the BRR, that is now valid until further notice.

The PSR procedure conducted at the BRR ensures a complex overall review of the research reactor by handling the reactor as a complex organization together with its service life. This PSR procedure could be one of the approaches for substantiating the renewal of the operation license by designating the conditions for safely extending the reactor’s lifetime; moreover, it may be useful as a model for conducting periodic safety reviews at other facilities.

1. Introduction

The Budapest Research Reactor (BRR) is a tank-type reactor, moderated and cooled by light water. The reactor, which went critical in 1959, is of Soviet origin. The initial thermal power was 2 MW. The first upgrading took place in 1967 when the power was increased from 2 MW to 5 MW, using a new type of fuel and a beryllium reflector. A full-scale reactor reconstruction and upgrading project began in 1986, following 27 years of operation since initial criticality. The upgraded 10 MW reactor received the operation license in November 1993. In line with Hungarian safety regulations a periodic safety review (PSR) was conducted in 2002-2003, as a result of which the operation license was renewed.

2. Guidance Document of PSR

The principal aim of the PSR is to serve as the basis for renewing the operation license of the given nuclear facility [1]. In 2002 the Regulatory Body issued the PSR requirements in the form of a guidance document [2]. This document divided the review items into 13 subjects (technical aspects, system audit, safety analyses, quality assurance, ageing, human factors, environment factors, etc.) and comprehensively defined the tasks to be carried out. The general philosophy of the prescribed safety review was to reflect the 10-years’ operation record (“horizontal screening”):

- the actual condition of the reactor had to be presented and the adequacy of the reactor systems and subsystems had to be verified including the drawing up of a lifetime estimation with special consideration to the forthcoming 10 years’ operation period (audit of technical correspondence);
the rules of procedures with special consideration to their periodic auditing and upgrading based on our own experience and that from other research reactors had to be reviewed and evaluated (audit of regulations);

- the activities, preparedness and all aspects of the organization of the staff had to be reviewed and evaluated with regard to human factors and safety culture (audit of human factor).

Based on the summarized evaluation results a PSR Report had to be formulated. In this Report, a detailed comparison (“vertical screening”) had to be made between the current nuclear safety regulations and the existing technical status of the reactor and/or applied procedures, practices, etc. having demonstrated and verified the conformity with the requirements (audit of conformity to nuclear safety regulations).

With the results of the above-listed audits: renewals, modifications and measures for eliminating any deficiencies and/or increasing safety and – in case of need – temporary or final restrictive and/or limiting measures had to be drawn up.

3. Periodic Safety Review

In accordance with the quality assurance (QA) system of AEKI the PSR was performed as a technical-and scientific project whose subject is the BRR and its 10-year operation history. The project included an overall review of the BRR’s systems and regulations as well as details of its operation- and events records. Including the formulation of the report, the PSR took about a year with the active participation of 5-12 reactor experts – depending on the phase of the project. Acknowledgement is made to the staff and to all the other experts involved for their unfailing support.

3.1. Audit of technical correspondence

Technical audit meant reviewing six issues, viz. technical status of the facility, system qualification, ageing, features of safe operation (safety margins), environmental influences, and safety analysis. Although utilization issues were not in the focus of the PSR, their overview and influences on safety were also taken into account.

From the time of start-up the upgraded reactor has been operating on average ≈3400 hours/year without any significant problem. The operation time record (scheduled and performed) is displayed in FIG. 1, the diagram of the integrated power in MW-days can be seen in FIG. 2.

![FIG. 1 Operation time record](image1)

![FIG. 2 Integrated MW-days](image2)

On comparing the yearly operation data (see FIG. 1) it can be seen that actual operation was close to the scheduled plan (coincidence>98%). From the restarting of 1993 the BRR fulfilled 19 refuelling cycles (campaigns), as indicated on the curve of MW-days in FIG. 2. The Performance Indicators (Table 1) display the most important reactor parameters (availability, unscheduled shut-downs, radiation doses and radioactivity released) in a summarized form from 1994 to 2003. It can be seen
that the availability of the reactor was over 98 % (as also shown above by FIG. 1) and the annual number of unscheduled shut-downs was around 4-6 (average 4) in the last ten years of operation.

During the safety review the most important parts of the validation test procedures of the reactor systems carried out during system installation and commissioning in the period of the reactor upgrading were repeated and the latest results were compared with the nominal database values recorded 10-12 years ago during the same validation test procedures. Based on the results of these comparisons and the operation data and event-audit it can be declared that no significant ageing problems, no unexpected degradation, and no singular phenomena on any safety-critical system or component were found. Any degradation is in accordance with the service life.

As part of the technical audit we reassessed the final safety analysis report (FSAR). Taking into consideration the results of the validation procedures and the 10-year event records we concluded that the assumptions of the PSAR (preliminary safety analyses report) and its extreme conservatism were confirmed and verified both by the operational experience and by the FSAR reassessments.

<table>
<thead>
<tr>
<th>Year</th>
<th>Availability [%]</th>
<th>Unscheduled shut-downs</th>
<th>Radiation doses [mSv]</th>
<th>Radioactivity released [TBq]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>A1</td>
<td>A2</td>
<td>B1</td>
<td>B2</td>
</tr>
<tr>
<td>1994</td>
<td>98.7</td>
<td>42.5</td>
<td>7</td>
<td>2</td>
</tr>
<tr>
<td>1995</td>
<td>99.9</td>
<td>51.1</td>
<td>3</td>
<td>0</td>
</tr>
<tr>
<td>1996</td>
<td>99.7</td>
<td>44.2</td>
<td>5</td>
<td>0</td>
</tr>
<tr>
<td>1997</td>
<td>99.8</td>
<td>25.6</td>
<td>3</td>
<td>0</td>
</tr>
<tr>
<td>1998</td>
<td>99.9</td>
<td>37.4</td>
<td>1</td>
<td>0</td>
</tr>
<tr>
<td>1999</td>
<td>99.9</td>
<td>35.7</td>
<td>4</td>
<td>0</td>
</tr>
<tr>
<td>2000</td>
<td>99.9</td>
<td>26.7</td>
<td>3</td>
<td>0</td>
</tr>
<tr>
<td>2001</td>
<td>98.1</td>
<td>39.3</td>
<td>6</td>
<td>0</td>
</tr>
<tr>
<td>2002</td>
<td>99.8</td>
<td>42.3</td>
<td>3</td>
<td>0</td>
</tr>
<tr>
<td>2003</td>
<td>98.7</td>
<td>42.5</td>
<td>2</td>
<td>0</td>
</tr>
</tbody>
</table>

Symbols: \( A1 = \frac{\text{No. of days at power} \times 100\%}{\text{No. of days at power} + \text{No. of days unscheduled shutdown}} \)
\( A2 = \frac{\text{No. of days at power} \times 100\%}{365} \)

B1: Number of unplanned reactor shut-downs initiated by reactor protection system / manual intervention  
B2: Number of unplanned reactor shutdowns initiated by experiments under irradiation  
D1(a): Collective radiation dose to Reactor Operations staff [mSv]  
D1(b): Number of Reactor Operations staff  
D1(c): D1(a)/D1(b) [mSv/man]  
D2(a): Collective radiation dose to all staff [mSv]  
D2(b): Total number of staff involved  
D2(c): D2(a)/D2(b) [mSv/man]  
E1: Rare gas released to atmosphere [TBq]  
E2: Tritium released to atmosphere [TBq]  
E3: Tritiated water discharge [TBq]  
E4: Iodine released to atmosphere [TBq]

Table 1. Performance indicators

3.2. Regulation audit

In accordance with the Guidance Document, under the regulation audit heading four items, viz. the rules of procedures, utilization of experiences, research and development (R&D) activities, and the QA program were reviewed. The regulations, from the operation limits and conditions (OLC) to the
safety regulations, as well as maintenance programs, education, training and examination rules were also reviewed as was their implementation in practice. As an important factor the utilization of experience as a feedback method for improving the procedures was also investigated. With regard to regulation audit, data log files on the 10-year operation period were processed and summarized from the viewpoint of OLC violation, ageing, campaign pointers, environment influences, fuel cycle handling, waste management, etc.

In terms of the results of the regulation audit it can be stated that most of the written procedures exist and they are generally up-to-date and have been improved by experience. However, some of the obligatory written procedures, viz. system qualification and ageing handling programs, are missing even though in everyday practice they are carried out. As future tasks, updating of the emergency plan and QA program are listed and the realization of the pre-decommissioning plan is also envisaged.

3.3. Audit of human factors

Thorough and painstaking reviews were performed during the PSR involving all safety issues, including all organization and administrative factors and human factors relating to safety culture. Also considered among the human factors were the PR activities and the links with the various authorities.

On the basis of the review it can be certified that the 10-year service life of the reactor was safe, there were no violations of the OLC. The operation and maintenance practices met with both the Regulatory Body’s regulations and those of the local regulations.

From the viewpoint of human factors and safety culture appearing in everyday practice, the operational environments promote safe and reliable reactor operation. Even so, it should be mentioned that the ageing of the staff is an ongoing problem that we have been facing for a couple of years. However, there are some good signs (in the Mechanical and Radiation Protection Groups we have employed some new colleagues in the last few years and there is no fluctuation in the staff). Inevitably, there is the need to effect a systematic rejuvenating program, and even a temporary increase in staff (employing young experts, carrying out on-the-job training). Ongoing consideration of this staffing problem was mentioned in the operation license conditions prescribed by the authority.

3.4. Audit of conformity to nuclear safety regulations

As vertical screening, a qualifying comparison was made during which we collated the points of the current nuclear safety regulations with the existing status of the reactor or practices and the regulations followed. In this itemized comparison we displayed and verified the adequacy or deficiencies of the BRR and its operation practices with the requirements. By virtue of the practices determined by the regulations and the long-term nuclear experience gained on research reactor operation, no serious difficulties were found that would have indicated any need for temporary or permanent restrictive and/or limiting measures. Conformity was nearly 100%. Any non-conformities found by the comparisons were the same as those revealed by the horizontal audits.

3.5. PSR Report

The PSR Report (1 summarizing volume, plus 14 attachment volumes) was compiled in which the main statements and conclusions of the audits listed in Paragraphs 3.1–3.4 are summarized, and renewals, modifications and measures for eliminating any deficiencies and/or increasing safety are drawn up. The report together with the request for renewal of the operation license was submitted to the Regulatory Body in April 2003.

4. Acceptance of the PSR and renewal of the operation licence

Based on the PSR, in November 2003 the Regulatory Body renewed the operation license until further notice, and accepted the technical renewals that had been initiated and the measures for increasing
safety (these tasks are prescribed in the resolution of the operation license). Their types and numbers are listed in Table 2.

<table>
<thead>
<tr>
<th>Tasks</th>
<th>Elimination of the deficiencies found</th>
<th>Preventive measures*</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>serious*</td>
<td>medium*</td>
</tr>
<tr>
<td>Technical tasks</td>
<td>−</td>
<td>1</td>
</tr>
<tr>
<td>Administrative tasks</td>
<td>−</td>
<td>3</td>
</tr>
<tr>
<td>Continuous administrative tasks</td>
<td>−</td>
<td>1</td>
</tr>
</tbody>
</table>

* Classification according to the Authority Guide

Table 2. Technical renewals and administrative measures prescribed in the license of BRR

Although there are a few technical tasks listed as deficiencies of medium importance, the majority of the discovered shortcomings are administrative tasks such as elaborating written programs for equipment qualification, management of ageing, rules of procedure for handling the operational documentation, as well as upgrading the quality assurance system, the decommissioning plan, and operation- and maintenance work procedures. However the most significant of the above tasks, which takes about 2-3 years from planning to the commissioning phase, is the modernization of the technological air ventilation system – including the inlet and outlet filters.

5. Conclusions

The PSR procedure conducted at the BRR ensures a complex overall review of the research reactor by handling the reactor as a complex organization together with its service life. A part of the audit items, in the form of a “horizontal screening”, reflects the 10-year operation record by reviewing the reactor systems, regulations and human factors, while another part, as a vertical screening, displays and verifies the conformity to the nuclear safety regulations. This PSR procedure could be one of the approaches for substantiating the renewal of the operation license by designating the conditions for safely extending the reactor’s lifetime; moreover, it may be useful as a model for conducting periodic safety reviews at other facilities.

REFERENCES

EXPERIENCE WITH LICENSING OF RUSSIAN NPP OPERATION EXTENSION

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Abstract. Since 2000, the works on operation of first generation NPP’s units extension are conducting in Russia. The Gosatomnadzor of Russia developed regulatory documents with requirements in what a way the possibility of those units operation extension have to be substantiated. The operating organization carried out the unit’s modification and performed the complex of measures on their preparation for operation extension, including survey works and in-depth safety assessment. A lot of deterministic as well as probabilistic safety analyses had been carried out. The independent expertise of the relevant substantiations was organized and conducted. The expertise results were considered by regulatory body in the licensing of the operation extension.

1. Introduction

The terms of operation extension of NPP’s units was determined, as one of the measures in the strategy of nuclear power development in Russia in the first half of XXI century \cite{1}, which was approved by the Government of Russian Federation. It is rather natural, that in the first place this is relating to the first generation of NPPs.

\begin{table}[h]
\begin{tabular}{|c|c|c|c|c|c|c|}
\hline
NPP (unit) & Reactor & Commissioning & NPP (unit) & Reactor & Commissioning \\
\hline
Novovoronezh\textsuperscript{3, 4} & WWER-440/V-179 & 1971, 1972 & Leningrad \textsuperscript{1, 2} & RBMK-1000 & 1973, 1975 \\
\hline
Kola \textsuperscript{1, 2} & WWER-440/V-230 & 1973, 1974 & Kursk \textsuperscript{1, 2} & RBMK-1000 & 1976, 1979 \\
\hline
\end{tabular}
\end{table}

It is evident from the Table that the first generation NPP’s units exhausted (or practically exhausted) the designed term of operation (30 years). Since 2000, the works on these units operation extension are being carried out in Russia (see for example \cite{2}).

2. Normative requirements and criteria for operation extension

In the strength of it’s authorities on regulation of nuclear power usage safety the Gosatomnadzor of Russia developed the system of requirements in what a way the possibility of first generation NPP’s units operation extension have to be substantiated and reflected it in the following regulatory documents:

\begin{itemize}
\item “Basic requirements for nuclear power plant unit lifetime extension”, (NP-017-2000); \\
\item “Requirements for justification of a possibility to extend design service life of facilities and installations of nuclear energy use”, (NP-024-2000); \\
\end{itemize}
A.B. Malyshev et al.

“Requirements to the structure and content of the set of documents justifying nuclear power plant safety during extended period of operation”, (RD-04-31 2001).

The criteria are given in these documents on the possibility of the NPP’s unit operation during extended term. In particularly it is necessary, that the remaining life time of expendability components important for safety must be justified and be sufficient during the extended term of the unit’s operation and that the undertaken technical and organizational measures are confirming it’s correspondence with the requirements of modern norms and rules or providing compensation of the remaining digressions.

As early as in 1997 Gosatomnadzor of Russia developed the document (RB G-12-42-97) “Guidelines for the In-Depth Safety Assessment of Operational NPP Units with VVER and RBMK Type Reactors” (OUOB AS), based on the regulatory documents and with consideration of the international expert’s group recommendations.

This Report is one of the main documents in which operating organization should justify the possibility of first generation units operation extension. To get the long term license for the first generation unit operation (on the period 3 years or more), the operating organization - Federal state unitary enterprise concern “Rosenergoatom” have to prepare OUOB AS and submit it to Gosatomnadzor of Russia together with other documentation justifying nuclear and radiation safety of the unit.

3. First generation units safety assessment before their operation extension beyond designed lifetime

In spite of the absence of special normative documents regulating periodical safety review performance in Russia, Gosanomnadzor constantly conducted such assessment, with respect to the first generation units on the regular basis, realizing continuous monitoring of their safety; which includes three types of assessments:

— The most complete safety assessment was carried out while granting annual licence on the unit operation.
— Partial safety evaluation is carrying out during giving permission on possibility for one or other change insertion in Conditions of Licence Validity (e.g. on equipment replacement or modernisation which causes updating of operational documents and parts of safety justification documents).
— Safety assessment limited to the problems of safety ensuring during operation, which is based on the results presented by operating organization in “Annual report dealing with assessment of safety state while NPP’s unit operation”.

4. Units preparation to the extension of operation

4.1. Prerequisites for the long term operation

The system of physical protective barriers consecutively located on the path of release of ionizing radiation and radioactive substances into the environment, is more or less realized in the design of all Russian NPP’s units. The operating organization is permanently proving the state of protective barriers, submitting the data got on the protective barriers and safety systems state to Gosanomnadzor of Russia in “Annual report dealing with assessment of safety state while NPP’s unit operation”.

The absence of full value containment on first generation units is compensating by implementation of measures realizing “leak before break” concept as well by the system of operational measures directed on prevention of large diameter pipe damage. Moreover, the low specific activity of the coolant circulating through reactor is maintaining be technical-organizational measures.
The water chemistry regime of the units is continuously perfecting, providing reduction of coolant and other medias impact on the corrosion stability of the NPP systems equipment and pipelines structural materials.

The conservative approach had been implement while Russian NPPs designing, which provides reliability reserve of systems important for safety equipment and sluggishness of transients. This approach advantages affirmation in the design of first generation NPPs was demonstrated during their safety assessment using modern safety analyses of such accidents as, for example ATWS, which never been done previously. In depth safety assessment of the installations with light water reactors revealed, that design basis reserves provide operating personnel with sufficient response time to manage even such kind of accidents.

To prevent operating personnel possible mistakes operating organization is deeply analyzing:

— Interface "personnel - machine" on specific workplaces, influence of environment and ergonomic conditions on the operator;

— The quality of operators training, including usage of full-scale and analytical simulators (all first generation units are equipped with such simulators).

4.2. Modernization and main measures on units preparation for operation term extension

In a compliance with Gosanomnadzor of Russia requirements, operating organization carried out such main measures on units preparation for operation term extension, as in-depth safety assessment and all-inclusive surveying, the last aimed on determination of the actual state of the unit and remaining life of the unit’s components, (equipment, buildings, construction, and building structures).

The program of surveillance, evaluation, prognosis and managing of the equipment remaining life is implemented at each Russian NPP’s unit for assessment of the unit’s equipment remaining life state. This program fulfillment is one of the criteria, established in document NP-017-2000 determining the possibility of unit’s operation extension.

In the frames of in-depth safety assessment, the analysis of digressions from regulatory documents in force was carried out using the methodology recommended by IAEA for safety assessment of the units built according early standards [3]. Besides, the results of IAEA extra budgetary program on NPPs with WWER and RBMK reactors over a period of time from 1990 to 1998 [4] was taken into consideration.

The first level probabilistic safety assessment (PSA) was done for each unit planned for operation extension. Results of digressions analysis as well as PSA results to a great extent facilitate the justification of units modernization programs. During units preparation for operation term extension those programs had been mainly realized before design life time completion, which provides improving of the units safety assurance.

The most significant among measures realized at the units are the following ones:

— Technical means realizing “leak before break” concept are installed, which significantly reducing probability of the large diameter pipes break;

— The guidance are developed and technical means are installed for beyond design accident management, as well the elaboration and implementation of symptom oriented accident instructions is carrying out.

On units with RBMK-1000

— Substitution of technological canals (TC) and recovery of the “canal- TC” design gap are done;
Reactors conversion on uranium-erbium fuel is under process, what improves thermo-physical and nuclear-physical characteristics of the core, including enabling to reduce steam reactivity coefficient;

— Reactor’s control and protection system is modified through the second reactor shut-down system implementation, that significantly extend the system functions and increase it’s reliability;

— Emergency core cooling system based on the passive principle of water injection in the multifold forced circulation circuit from hydro- balloons is implemented;

— Reactor protection system from technological parameters is improved;

— Safety of reactor’s core in enhancing by implementing of the cluster regulating rods.

On units with WWER-440

— Two independent channels of safety systems with internal reservation of active elements are installed;

— Reactors vessel annealing was done;

— Replacement of old protective valves by new valves, which are independent from aggregative state of media and providing the possibility of heat abstraction in “feed and bleed” regime for the first and secondary circuits;

— For feed water supply into steam generators the NPPs are equipped with transportable pump installation with diesel drive gear and autonomous source of power;

— Reactor protection systems and control safety systems are modified on the up-to-date software and hardware basis.

The modernization of units results in the following comparative core damage frequencies (CDF) got by PSA:

<table>
<thead>
<tr>
<th>Unit</th>
<th>CDF</th>
<th>Unit</th>
<th>CDF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Novovoronezh –3</td>
<td>$2,54 \times 10^{-5}$ 1/year</td>
<td>Leningrad -1</td>
<td>$9,5 \times 10^{-6}$ 1/year</td>
</tr>
<tr>
<td>Novovoronezh –4</td>
<td>$5,12 \times 10^{-5}$ 1/year</td>
<td>Leningrad -2</td>
<td>not submitted *</td>
</tr>
<tr>
<td>Kola –1</td>
<td>$2,9 \times 10^{-5}$ 1/year</td>
<td>Kursk -1</td>
<td>$6,2 \times 10^{-5}$ 1/year</td>
</tr>
<tr>
<td>Kola –2</td>
<td>$3,04 \times 10^{-5}$ 1/year</td>
<td>Kursk -2</td>
<td>not submitted *</td>
</tr>
</tbody>
</table>

* - PSA is not submitted to Gosatomnadzor of Russia as a part of Application on operation extension.

It is evident from the Table that the CDF value for all units is lower than the value equal to $10^{-4}$, which is recommended by IAEA in the document INSAG-8 (1996) for the first generation units.

5. Licensing

One of the main principles of the nuclear and radiation safety assurance is the provision of a permissive character of activity in the field of atomic energy using the licensing procedure mechanism [5].

Gosatomnadzor of Russia implementing this mechanism with the help of “ Licensing Regulations Governing the Use of Nuclear Energy” (Approved by RF Government at 14 July 1997), which stipulates the review of documentation, justifying nuclear and radiation safety assurance of the unit. By order of Gosatomnadzor of Russia SEC NRS, in accordance with established procedure, organized
and carried out expertise of documentation justifying units safety during the term of operation extension.

Thematic questions of the expertise were grouped in such subsections as for example:

- Site characteristics, including questions of the site location changes with respect to administrative districts, population distribution and sanitary protection zone evaluation;
- Remaining life of the reactor elements and main expendability equipment;
- Modernization and upgrading of safety systems;
- Results of all-inclusive surveying of the unit;
- Safety analyses;
- Fuel and radwastes treatment (at that, life time of spent fuel storage is assessing as well as life time of liquid radwastes storage);
- Radiation protection and unit’s influence on the environment.

The review of in-depth safety assessment and of all-inclusive surveying results have not revealed any impedimental factors for the possibility of operation extension for the units under consideration.

Besides, inspections on the units with independent experts participation were organized to check the actual state of modernization measures on safety enhancement fulfillment.

The review results served as a basis for assignment of term on what the licence for operation was granted. Periods of validity of the already issued licences are given lower. The Applications on licence for another units have not been submitted or relevant documentation is still under review.

<table>
<thead>
<tr>
<th>Unit</th>
<th>Periods of validity</th>
<th>Unit</th>
<th>Periods of validity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Novovoronezh -3</td>
<td>5 years</td>
<td>Leningrad -1</td>
<td>3 years *</td>
</tr>
<tr>
<td>Novovoronezh -4</td>
<td>5 years</td>
<td>Kursk -1</td>
<td>3 years</td>
</tr>
<tr>
<td>Kola -1</td>
<td>5 years</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* - With condition of operation at full power permission after all works on the unit modernization completion.

REFERENCES


AGEING MANAGEMENT REVIEW FOR MAIN COMPONENTS OF PWR NUCLEAR POWER PLANT

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Abstract. Implementing effective ageing management for main components of operational NPP is a necessity for plant life management. The idea of systematic ageing management should be reflected in daily activities of NPP including operation, inspection, maintenance and management. Taken 300 MWe PWR NPP as a background, this paper analyses major ageing mechanisms for main components of primary system, including RPV, SG and RI and summaries methodology and main steps for ageing management review. The paper also addresses the importance of establishment of systematic ageing management program to improve safety and economy of NPP operation.

Keywords: ageing management, review, ageing mechanism, main components

1. Introduction

The history of nuclear power in the mainland of China started in 1991. Until now China has a total of 8 reactors in operation with a total power capacity of nearly 6000 MWe. According to a new national development framework, to satisfy the increasing need of energy, quite a number of new NPPs will be built in the next 20 years. The safety of operational NPPs draws considerable attention to the nuclear safety regulatory organizations, owners and technical supporters. From the viewpoint of sustainable development, improving economic benefit of nuclear power while assuring its safety by implementing effective ageing and life management, is of importance to showing the advantage of nuclear energy over other energy sources. According to the requirements of NNSA\textsuperscript{1}, to every operational NPP, a periodic safety review (PSR) should be carried out every 10 years. Ageing management is one of the safety factors for PSR. Main components such as reactor pressure vessel (RPV), steam generator (SG) and reactor internals (RI) are the most important components for ageing management review (AMR).

Taken RPV, SG and RI as examples, this paper introduces elements and main steps for AMR, analyses main ageing mechanisms to be focused on and introduces methodology for the review. Finally, this paper discusses the importance of establishing a systematic ageing management system in a NPP.

2. Main elements and steps of AMR

As a part of PSR, AMR should generally follow the requirements of PSR while focusing on specific ageing management problems. NNSA published a PSR guideline\textsuperscript{1}, which mainly refers to relevant
IAEA guidelines and combined with Chinese practice. The PSR requires the owner of a NPP to perform a self-review, which usually takes about 2–3 years, and then, to hand over a whole set of self-review reports to NNSA. Then, NNSA will perform an independent review to the reports. In this paper, the AMR is a self-review carried out by owner or by a technical support organization entrusted by owner.

According to the NNSA’s guideline, the following elements should be focused on during AMR: 1) ageing management program; 2) documented method and criteria for identifying Safety System Components (SSCs) covered by the ageing management program; 3) list of SSCs covered by the ageing management and records which provide information to support management of ageing; 4) evaluation and documentation of potential ageing degradation that may affect the safety functions of SSCs; 5) the content of understanding the dominant ageing mechanisms of SSCs; 6) the program for timely detection and mitigation of ageing processes and/or ageing effects; 7) acceptance criteria and required safety margins for SSCs; 8) awareness of physical condition of SSCs, including actual safety margins.

The review can be divided into two parallel parts: component-oriented review and program-oriented review. This paper mainly describes with the component-oriented review, which includes the following 5 steps shown in the follows.

**Step 1 Screening** First, some typical structures or components will be selected as AMR components from large number of SSCs according to their safety importance, feasibility and cost for replacement. Although screening process and results may vary from plants to plants, main components of the primary loop, such as the RPV, SG and RI will usually be selected as AMR components.

**Step 2 Ageing mechanism analysis** For selected AMR object components, their ageing mechanisms, which may affect lifetime, should be described carefully according to the environment and working conditions.

**Step 3 Investigation and information collection** AMR is a process of information collection and evaluation. Information on design, manufacture, installation, commissioning, operation, inspection and maintenance is helpful to determine whether suitable countermeasures against ageing identified in Step 2 are taken during different stages. This is especially important in operation, inspection and maintenance process and how the countermeasures effectively mitigate the degradations. The investigation should also identify in what degree management documents and their implementation consider mitigation of ageing degradation. It should also identify whether plant resource configuration, on daily ageing management, is sufficient and whether new technical development and concept renovation are reflected in programs for maintenance, inspection and chemical control, etc. An AMR table will be described later in detail.

**Step 4 Assessment** Ageing management status for specific components should be assessed in detail and relevant improvement suggestions should also be given. To some important and quantifiable ageing mechanisms, such as fatigue caused by transient fluctuation, a structural integrity analysis may be performed according to actual operating conditions so that the actual safety margin can be measured.

**Step 5 AMR report** AMR reports for specific components will be compiled individually and finally a summary report will be made. All the reports will be sent to NNSA for approval. During the approval process, good communication between owner and NNSA is necessary. The approved AMR reports will then be taken as a basis for continuing improvement in the next 10 years.
3. Ageing mechanism analysis for components

For PWR NPPs, the degradation due to various ageing mechanisms of RPV, SG and RI can be observed in different stages of their lifetime. According to the ageing mechanisms, the degradation can mainly be classified into four categories, i.e. corrosion, fatigue, irradiation embrittlement and wear. Each ageing category may include a variety of failure modes. For example, stress corrosion cracking (SSC) can occur as transgranular stress corrosion cracking, intergranular stress corrosion cracking and irradiation assisted stress corrosion cracking etc.. The location of SCC occurred can be in primary water components and in secondary water components. Failures are often induced by compositive effects of multi-ageing mechanisms. For instance, the RPV core beltline suffers from a long-term strong irradiation and will become non-ductile. Flaws may initiate due to fatigue induced by fluctuation of temperature and pressure of the coolant and in the case of a PTS event, non-ductile failure of RPV under the combined action of irradiation embrittlement and fatigue may happen. So enough attention should be paid to each leading ageing mechanism during the ageing management and the combined action of multi-ageing mechanism should also be considered so that ageing or degradation of components can be effectively controlled. Therefore, AMR should start with an understanding of leading ageing mechanisms and investigating if management countermeasures aiming at the ageing mechanisms have been well prepared and effectively implemented. The summary of the leading ageing mechanism for RPV, SG and RI are listed in Table 1. The table identifies, SCC: Stress Corrosion Cracking; PWSCC: Primary Water Stress Corrosion Cracking ODSCC: Outside Diameter Stress Corrosion Cracking; IGSCC: Intergranular Stress Corrosion Cracking IASCC: Irradiation Assisted Stress Corrosion cracking.

4. Review methods

The AMR for components will finally finish a series of AMR reports. These reports shall include a comprehensive description for structure feature of the components, design requirements, manufacture process, information noticed during installation and test, operation environment, main inspection results, main maintenance records, etc. The reports shall also include a description for some important ageing mechanisms. The core contents of the reports have tabulated assessment tables for the specific ageing mechanisms on different locations. Each ageing mechanisms at different locations will have an independent table. As ageing mechanisms for different components vary from each other, the numbers of review tables for different components are quite different.

Table 2 shows the format of the review table and relevant remarks. From the above, it can be seen that when performing a review for the specific ageing mechanism at a specific location, the reviewer should, first of all, well understand the information on material, environment, ageing mechanism, ageing indicator, degradation effects and consequence, relevant codes, standards and specifications, existing ageing management program and procedures etc. Afterwards, the reviewer can work out an assessment for ageing management and ageing status and, finally, give out review conclusions and countermeasure suggestions.
<table>
<thead>
<tr>
<th>Component/Location</th>
<th>Corrosion</th>
<th>Fatigue</th>
<th>Irradiation</th>
<th>Wear</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core beltline</td>
<td>Environment fatigue</td>
<td>Neutral irradiation</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Stud bolts</td>
<td>Boric acid corrosion</td>
<td>Mechanical and thermal stress</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Corner of nozzle</td>
<td>Mechanical and thermal stress</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CRDM penetrations</td>
<td>PWSCC</td>
<td>Mechanical and thermal stress</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Outer surface of dome head</td>
<td>Boric acid corrosion</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Bottom head</td>
<td>Worn by loosing parts</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tube to tube-sheet services</td>
<td>ODSCC</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tube support plate</td>
<td>ODSCC</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sludge pile</td>
<td>ODSCC</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Free span</td>
<td>ODSCC</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Contacts between tubes and anti-vibration bars</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Contacts between tubes and loose parts</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Feedwater nozzle and nozzle-to-piping weld</td>
<td>Erosion-corrosion</td>
<td>High and low cycle fatigue</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Shell girth welds</td>
<td>SCC</td>
<td>Corrosion fatigue</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Feedwater nozzle bore, blend radius, shell inside surface beneath nozzle</td>
<td>High-cycle thermal fatigue</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>J-tubes and feedwater nozzle</td>
<td>Erosion-corrosion</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Barrel</td>
<td>Thermal stress, FIV</td>
<td>Neutral irradiation</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Baffle and former</td>
<td>Thermal stress</td>
<td>Irradiation swelling</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Secondary support assembly</td>
<td>FIV</td>
<td>Fretting, wear</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Guide tube assembly</td>
<td>FIV</td>
<td>Fretting, wear</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Radial locating key</td>
<td>FIV</td>
<td>Fretting, wear</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Instrumentation sleeve, control rod guide tube</td>
<td>FIV</td>
<td>Fretting, wear</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Irradiation surveillance capsule</td>
<td>FIV</td>
<td>Neutral irradiation</td>
<td>Fretting, wear</td>
<td></td>
</tr>
<tr>
<td>Connect parts such as bolts and pins</td>
<td>IGSCC, IASCC</td>
<td>FIV</td>
<td>Irradiation, swelling and creeping</td>
<td></td>
</tr>
<tr>
<td>Hold-down spring</td>
<td>Mechanical axial force</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
4.1. Summary

This paper presents the main steps and methodology of AMR during PSR. Carrying out AMR for main components of a NPP is helpful to make an overall and comprehensive assessment on ageing management and on actual ageing status of the components. It will form a basis for continued improvement on ageing management of NPP in next 10 years.

Although AMR is effective and constructive, it cannot substitute the daily ageing management activities of the NPP. According to IAEA’s point of view\(^3\), only after a systematic ageing management program is established, can an operational NPP continuously improve its safety features, component reliability and economic benefit through effective ageing management. A systematic ageing management program is not only aiming at concrete ageing problems for specific components, but also reflecting effective control to the ageing process from various aspects, including the management concept, program and procedure system, management operation inside the NPP, resource configuration, communication and correlation with outside supporters and suppliers. Through a Plan-Do-Check-Act cycle, a systematic ageing management process can become more and more effective and adaptive.

Table 2 AMR form and relevant remarks

<table>
<thead>
<tr>
<th>Name of Component</th>
<th>List name of reviewed component, for example, RPV.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Serial number of the review</td>
<td>List serial number of the review table. From serial number, the reviewed component can be distinguished.</td>
</tr>
<tr>
<td>Concerned location</td>
<td>List specific location of the reviewed component, for example, top head of RPV and concerned area (such as dome, flange, bolt and so on) corresponding to this table.</td>
</tr>
<tr>
<td>Material</td>
<td>List material brand corresponding to the specific concerned location.</td>
</tr>
<tr>
<td>Environment</td>
<td>Describe in detail the environment of reviewed part with measurable parameters, such as working temperature, pressure, humidity, etc.</td>
</tr>
<tr>
<td>Ageing mechanism and its indicator</td>
<td>List ageing mechanism concerned and its corresponding indicator.</td>
</tr>
<tr>
<td>Degradation consequence</td>
<td>Describe effects and consequences of the degradation caused by the ageing, especially ageing effect on safety.</td>
</tr>
<tr>
<td>Codes, standards and technical information</td>
<td>List codes, standards and relevant technical information on detecting, mitigating and solving the specific ageing problem.</td>
</tr>
<tr>
<td>Existing programs and procedures</td>
<td>List management programs and/or procedures made by NPP for effectively managing the specific ageing problem. If no corresponding programs or procedures in the NPP, programs, procedures and requirement from outside should also be listed.</td>
</tr>
</tbody>
</table>
DOU Yikang et al.

List comprehensive assessment on ageing management and ageing degradation status. For example, the following information can be listed:

- Whether there are preventive measures to the specific ageing mechanism;
- Whether there are surveillance, inspection requirements and measures;
- Whether there are acceptance criteria for the ageing;
- Have any events caused by ageing degradation ever happened after the component started operation;
- Once the degradation is not acceptable, are there any counter plans on maintenance, repair and replacement;
- Whether experience on ageing problems from inside and outside can timely and effectively reach to persons responsible for ageing management;
- Actual status of the component; etc.

Review conclusions

List review conclusions. Strong points and weaknesses should be emphasized in the conclusion.

Suggestions on countermeasures and corrective actions

Put forward suggestions on countermeasures and corrective actions aiming at weaknesses.

REFERENCES

STUDY AND ANALYSIS OF DEGRADATION PROCESSES IN THE ELECTRONIC EQUIPMENT OPERATING AT KOZLODUY NPP

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Abstract. The electronic equipment for control and monitoring of the different systems on Units 5 and 6 of Kozloduy NPP has been studied for identification of degradation processes, occurring during its long use under the conditions of monitored radiation background in accordance with the radiation safety norms at the plant. For this purpose operating electronic units have been selected, from which active components, such as diodes, transistors and integrated circuits, have been disassembled and their main parameters have been measured. It has been found out by comparison with the parameters of the same components from non-operated equipment that some changes occur which may impact the functionality of the electronic equipment. The most sensitive factor for the diodes is the ideality factor $\beta$ and for the transistors these are the current transfer constant $h_{21}$ and the collector current $I_c$. A theoretical model has been developed by means of which the equipment lifetime forecast can be made depending on the duration and intensity of the operation.

Key words: Nuclear Power Plant (NPP), diodes, transistors, electronics, degradation

The electronic equipment for control and monitoring of the different systems on Units 5 and 6 of Kozloduy NPP has been studied for identification of degradation processes, occurring during its long use under the conditions of monitored radiation background in accordance with the radiation safety norms at the plant. For this purpose operating electronic units have been selected, from which active components, such as diodes, transistors and integrated circuits, have been disassembled and their main parameters have been measured. It has been found out by comparison with the parameters of the same components from non-operated equipment that some changes occur which may impact the functionality of the electronic equipment. The most sensitive factor for the diodes is the ideality factor $\beta$ and for the transistors these are the current transfer constant $h_{21}$ and the collector current $I_c$. A theoretical model has been developed by means of which the equipment lifetime forecast can be made depending on the duration and intensity of the operation.

The electric parameters of the three groups of samples was measured and analyzed using the theoretical model, developed by professor A. Popov from the department of Semiconductors physics in the Faculty of Physics at Sofia University “St. Kliment Ohridski” and published in the international magazines Physica Status Solidi (1984)[1], as well as Crystal Properties and Preparation (1987). The model is based on the formation of shunt areas in the p-n junctions of diodes and transistors being in long operation, which change their volt-ampere characteristics and the remaining current characteristics accordingly. The main reasons for shunt areas formation are considered to be the diffusion and accumulation of residual impurities on the dislocation defects, arising in the process of
technological composition of the active semi-conductor structure of the diodes and transistors. Two types of dislocations are assumed in the model: linear and non-conformity dislocations. The former cross the p-n junction and have their origin from the semiconductor substrate or from non-conformity dislocations, arising in the interface area of the basic semiconductor substrate and the inverse epitaxial layer. Both types of dislocations accumulate foreign (impurity) atoms but the linear dislocations contribute to development of shunt channels, while the non-conformity dislocations result in development of clusters or compound complexes. The model with linear dislocations is demonstrated in Fig. 1. Part of these appear on the surface of the active semiconductor structure, composed of a substrate and an epitaxial or diffusion layer over it, while the other part of the dislocations remain closed in the interface area and are regarded as space defects, appearing as micro-pores. As a whole both types of defects form a unified dislocation grid in the semiconductor structure, whose parameters are determined by the crystal growth conditions. Under the impact of the electric field, applied on the p-n junction and its internal temperature in the course of operation accelerated diffusion of foreign atoms takes place (most frequently copper atoms due to their small covalent radius) in the so formed dislocation grid. Three typical cases are presented in Fig. 1.

![FIG. 1. Possible models of structural defects in the p-n junction area, resulting from degradation processes](image)

The first case (a) results from accumulation of foreign atoms on the dislocation defects having appeared on the surface of the layer thus forming directly shunt areas.

The second case (b) results from accumulation of relatively small quantities of residual atoms on closed defects, thus forming cluster, which does not cause directly a shunt effect. The third case (c) occurs, when large amounts of impurities accumulate on such a closed defect causing high mechanical stress on the grid and as a result the closed defect enlarges up to the point of egression on the free surface of the epitaxial layer. The occurrence of shunt channels and clusters, creating deep energy levels in the forbidden zone affects most strongly the volt-ampere characteristic of the p-n junction and more specifically the $\beta$ factor (ideality factor) in the equation:

$$I = I_0 e^{\left(\frac{qU}{\beta kT}\right)}$$

(1)

The change of the ideality factor is illustrative of the change of slope of the volt-ampere characteristic of the p-n junction and respectively of the current parameters of the ordinary diode, formed on its basis. As regards a bipolar transistor, based on two p-n junctions (emitter and collector ones) the most strongly affected are the current of the collector junction $I_{co}$, transfer constant by current $h_{21}$, as well as the steepness of volt-ampere family characteristics. The explanation of this degradation phenomenon can be followed on the model, based on the modification of the space charge of the p-n junction due to generation of shunt areas after accumulation of foreign atoms on dislocation defects. The space charge is shown in Fig. 2 in a mono-dimensional version together with the shunt area. The ideality factor of such a p-n junction is presented by the expression:

$$\beta L = \left[2\left(b + ch\left(\sigma / L\right)\right)\right]^{-1}$$

(2)

where

$\omega=x_n+x_p$ is the width of the space charge,
\( L \) is the diffusion length of the carriers, 
\( B \) is the ratio between the mobility of the electrons and holes.

This formula allows a quantitative evaluation of the current flowing through the p-n junction (respectively the diode current, shifted in forward direction or the collector current). If the ratio between the width of the space charge of the p-n junction and the diffusion length of the carriers tends to zero, then the ideality factor assumes values below 2. This case correspond to the normal condition of the p-n junction, when there is no generation of shunt areas in the space charge and deep impurity levels in the forbidden zone or their concentration is very low.

The generation of shunt areas and deep levels in sensitive concentrations, affecting the current parameters of the diode and transistor increases the ideality factor above 2. In this case the ration between the width of the space charge and diffusion length becomes higher than zero due to increase length of the space charge.

On the basis of this theoretical model lifetime forecast is made for the diodes and transistors from a given board, as part of the electronic equipment, and thereupon a decision is made regarding its operation extension or replacement.

Translated into practice this means that:

1. Boards of electronic equipment are selected (three sets of the same electronic equipment, out of which one set has not been in operation at all, while the other two have operated for a different period of time).
2. Using the sampling method several identical diodes and transistors are disassembled from each set.
3. For the diodes the ideality factor is measured, and for the transistor transfer constant by current is measured. If the ideality factor is less than 2 and the transfer constant by current has changed to not more than 25%, then a decision can be made to extend the lifetime, providing that both diodes and transistors meet these conditions.
4. In case these conditions are not met a straight line is constructed on three points and its continuation is extrapolated, keeping the resulting slope.
5. A graphics is constructed with the following coordinates:
   a) the abscise being the time in operation
   b) the ordinate being the measured parameter (ideality factor and transfer constant by current).
6. The approximation will show the time period whereupon the subject parameter will exceed the defined limit and based on this a decision could be made regarding the extension of lifetime of given electronic equipment or its replacement with a new one from the spare parts reserve.

Fig. 3.(a). presents the volt-ampere characteristics of one operated and one non-operated diodes of КД 105В type from the electronic unit БЛП 1-У, wherefrom the ideality factor for the former is determined to be \( \beta = 2.685 \) and for the latter \( \beta = 1.487 \). This means that in the sample that has been in operation degradation processes have occurred, related to the changed ratio between the diffusion and recombination components of the diode current, with a slight domination of the latter. The overall current significantly decreases thus changing the functionality of the unit.
A. Popov and N. Naydenov

Fig. 3(b). represents the volt-ampere characteristics of two types of diodes, disassembled from two electronic units: ПБР-2М, having been in operation for 6 years (diode Д223 В) and БЛП-1У, having been in operation for 10 years (diode КД105 В). According to the model for the $\beta$ factor, the lifetime of the first diode has not expired.

![Fig. 3a. Volt-ampere characteristics of diode КД105: non-operated (upper curve) and long operated about 10 years (lower curve). (b) Same for diode Д223Б, operated for about 6 years](image)

**FIG. 3a.**  
**FIG. 3b.**

Fig. 4(a),(b). presents the modification of $h_{21}$ for transistor KT310 2Б and the base current for transistor KT315B from unit БУ3-У, being in operation for 10 years. This modification exceeds the admissible limits, therefore replacement is required.

![Fig. 4a. Static transfer constant by current $h_{21}$. (a) of a non-operated (upper curve) and of an operated for 10 years (lower curve) transistor KT310 2Б and base current $I_b$ (b) of a non-operated (upper curve) and operated for 10 years (lower curve) KT315B](image)

**FIG. 4a.**  
**FIG. 4b.**

**REFERENCES**

LIFE EXTENSION AND SAFETY UPGRADATION IN INDIAN NPPS – A REGULATORY PERSPECTIVE

J. Koley, R. Venkata Raman, S.K. Chande

Atomic Energy Regulatory Board, Mumbai, India

Abstract. Indian nuclear power generation programme started with commissioning of twin unit BWRs at Tarapur way back in 1969. Today 14 units, mostly PHWRs, with total installed capacity of 2720 MW are in operation and eight units with installed capacity of 3880 MW are under various stages of construction. The new plants are built to current standards and employ the present day technology and hence easily meet the present day safety requirements. The old plants obviously cannot meet these requirements to the same extent since they were built to the standards that existed at the time of their construction. AERB therefore has instituted certain mechanisms/procedures to address the issue of re-licensing vintage plants. License Renewal, Periodic Safety Review (PSR) and Life Extension programmes are used as regulatory tools for authorizing continued operation of NPPs with high level of safety. The regulatory criteria evolve continuously based on operating experience, identified generic safety issues and new developments in technology. The licensing as well as re-licensing procedure in India is designed to respond to these evolving safety criteria.

Technical assessment of components with respect to ageing, Review of the original design basis along with Final Safety Analysis Reports, Life assessment of irreplaceable equipment, structures and components and Plant specific PSAs and their relationship to the traditional deterministic methods are identified as key issues in the re-licensing or safety upgradation process. This paper deals with the present approach and regulatory mechanisms being followed for life extension and safety upgradation in Indian NPPs. Also, the safety upgradation and license renewal of older Pressurized Heavy Water Reactors (PHWRs) and Life extension studies carried out in vintage Boiling Water Reactor (BWR) are described.

1. Introduction

While the first twin unit plant in India at Tarapur was commissioned way back in 1969, the last four units at Kaiga and Rajasthan have started commercial operation in the year 2000. Thus we have today 14 units of different vintages operating in the country. The new plants are built to current standards and employ the present day technology and hence easily meet the present day safety requirements. The old plants obviously cannot meet these requirements to the same extent since they were built to the standards that existed at the time of their construction. Though modifications and repair are carried from time to time, ageing of equipment is another challenge that can impair safety. The regulator and the utility, therefore, have an additional responsibility of ensuring the validity of the safety case of such old plants.

In India both Life extension and Safety upgradation programmes are embedded in License Renewal procedures. Through safety upgradation and backfitting, measures are taken to enhance the safety level of the plant at any point in the life of the plant if it is necessary and convenient. Whereas the life extension programme allows plant for continuing operation with an acceptable safety level beyond design life, which was initially established by a safety evaluation. The opportunity of life extension is used to review the entire design basis of the plant and decide what safety improvements the user might reasonably be expected to make. The license renewal process focuses its review on detrimental effects of ageing and re-reviews a plant’s current licensing basis to comply with its regulatory regime, including generic safety issues. In India the license renewal by application for renewal of authorization
(ARA) is done once in three years as the current regulatory regime allows that. Therefore a brief but comprehensive safety review is done every 3 years. During a Periodic Safety Review (PSR), done once in 10 years, an integral safety assessment is made and the fitness for continuing operation of the plant is assessed.

2. Regulatory Approach

It is a basic requirement of Atomic Energy Act, India, 1962 and Rules framed under it that the licensees should carry out a continuous review of the safety of their plants and make whatever safety upgrades are necessary. It is normally their responsibility to propose the safety upgrades that they deem to be necessary and reasonably practicable, while it is the responsibility of the regulatory authority to assess and approve such proposals before the upgrades are carried out.

Atomic Energy Regulatory Board (AERB) issue fixed-term (for 3 years) licences for Nuclear Power Plants (NPPs). Continuous monitoring of operational and Safety performance is done to check the conformity with licensing conditions. Plus comprehensive periodic safety reviews (PSR), every ten years is also carried out. The objectives of these PSR are summarized as follows:

- To show that the plant is as safe as originally designed
- To show that it will still be safe for the next ten years
- To compare it against the most recent safety standards and determine which safety improvements are reasonably practicable.

Regulators apply various codes and guides in judging the acceptability of a licence renewal application (ARA) or a PSR. However the main guide for the above purpose is “AERB guide on ‘Renewal of Authorization for Operation of Nuclear Power Plants’ (AERB/SG/0-12), which was published in the year 2000.

New regulatory rules are not generally expected to apply retrospectively, in their totality, to existing plants. However during the PSR or Life extension process, licensees are required to assess the impact of new rules on the existing plants and determine the safety significance of any deviations. They then have to justify any such deviations to the regulator in terms of the risk involved or propose modifications to achieve the level of safety required by the current rules. Such modifications have to be shown to be reasonably practicable in terms of the safety gains to be achieved.

In recent years, AERB has instituted certain mechanisms/procedures to address this and these are described below:

2.1. Periodic Safety Review

In addition to the three yearly review of the Application for Renewal of Authorisation (ARA), AERB has prescribed a more comprehensive Periodic Safety Review (PSR) as mentioned earlier. In addition to the normal review of safety performance and operation experience feedback, PSR requires review of plant safety analysis in the light of current standards and actual condition of the plant. Due to modifications carried out from time to time and the effect of ageing, the present actual condition of the plant could be significantly different from the time it was constructed or since the last review. In addition, factors such as human performance and organisational changes are also considered. An integrated review of all these factors is carried out to provide assurance that till the next PSR, the plant can continue to operate with adequate safety margins.

While PSR needs to bring out the differences/shortcomings of the plant in its present condition in comparison to requirements of the current standards, it is not expected that the old plant should be
upgraded to the same extent as a new plant. The plant has to study the safety impact of all these differences and propose modifications wherever necessary. For deficiencies proposed to be left unaddressed, adequate justification needs to be provided.

Several such reviews have been conducted by AERB, over a period of time especially for older plants that include RAPS and MAPS. It was decided that all upgradation work identified as a result of this reviews, will be carried out during long outages connected with enmasse coolant channel replacement.

2.2. License Renewal

For plants nearing the end of the original licensed life, such as the two BWR based units at TAPS which are over thirty years old, a much more exhaustive review has to be carried out for considering continued long term operation. While formal guidelines for such process are under preparation, AERB had prescribed a procedure under which TAPS was required to prepare and submit reports covering the following:

- Review of Operational Performance
- Ageing management studies & residual life assessment
- Level 1 PSA
- Review of design basis and safety analysis

Continued long-term operation of TAPS, will depend upon the outcome of these reviews, consequent modifications/upgrades proposed and their implementation schedule.

3. PSA

The older plants were designed based on deterministic assessments only as PSA techniques were still in development stage. AERB has now recommended that PSA study should be carried out at the time of submission of first PSR or License Renewal application. It is expected that such a study will provide insights into important contributors to the core damage frequency based on which appropriate upgradations/modifications can be carried out to achieve a more balanced design. The effect of modifications already carried out or proposed to be carried out can also be evaluated by PSA studies.

Accordingly, a full scope Level – 1 PSA is a requirement for license renewal for TAPS.

4. Safety Upgradations in Raps and Maps

All the NPPs are designed and operated to meet the prescribed level of safety required by the standards and practices that existed at the time of their design. However, safety standards get revised from time to time based on operating experience, new developments in technology and improved understanding. Hence, it is necessary that all operating plants be periodically assessed to demonstrate that required level of safety is maintained. Towards this end, AERB conducts comprehensive Periodic Safety Reviews (PSR) for all operating NPPs. Based on several such reviews conducted by AERB, safety upgradation jobs had been undertaken.

4.1. Upgradation in RAPS Unit-2

RAPS Unit-2 is one of our old generations Pressurized Heavy Water Reactor (PHWR) based NPP of 200 MWe capacity. It was commissioned in 1981 and was meeting the specified safety requirements applicable at that time. The design of PHWRs has changed substantially over a period of time and the present day Indian PHWRs are built after taking account of all current safety requirements. In
accordance with AERB requirements, a detailed review of RAPS Unit-2 was conducted and a plan for required upgrades and modifications was finalized.

RAPS-2 was shutdown in 1996 for en masse replacement of all of its 306 coolant channels. The old coolant channels made of Zircaloy-2 pressure tubes were judged unsuitable for continued operation and were replaced by pressure tubes made of Zirconium Niobium alloy. This was done in a long shutdown of the reactor when major upgradations to improve safety were also implemented. All proposals concerning modifications to safety related systems were reviewed and approved by AERB before execution of the jobs. Some of the major modifications to safety related systems carried out during this shutdown are described below:

(a) Retrofitting of High Pressure Heavy Water Injection System into Emergency Core Cooling System.

Pursuant to the recommendations of AERB, a high-pressure heavy water injection provision was retrofitted in the Emergency Core Cooling System (ECCS). The retrofitted ECCS in RAPS Unit-2 provides for high-pressure heavy water injection during the initial short-term following a postulated break in the reactor coolant piping. This is in addition to the already existing long-term core cooling from low-pressure moderator system.

(b) Supplementary Control Room

A supplementary control room (SCR) was provided in a separate building from where important safety functions can be carried out in case the main Control Room becomes uninhabitable due to postulated initiating events like a localized fire or damage caused by turbine missiles. Functions that can be carried out from SCR include (i) tripping of the reactor, (ii) opening of steam discharge valves for assured core cooling and (iii) monitoring of essential system parameters. Independent sensors with separate power supply have been provided for instrumentation in the SCR to ensure their operability under emergency conditions.

(c) Segregation of Power and Control Cables

For the purpose of minimising the impact of fire and other common mode failures to acceptable level, segregation of routes of safety related power and control cables was carried out. With this, control cables of triplicated channel instrumentation signals run through three separate cable tray routes from reactor building to control equipment room.

In addition, a minimum physical separation has been maintained between the power cables and the high-energy steam lines.

(d) Additional Diesel Generator of 600 kVA

An additional Diesel Generator of 600 kVA capacity has been provided at a high elevation to ensure availability of essential power supply during a postulated scenario of total loss of power due to flooding, following a postulated failure of Gandhi Sagar dam which is located upstream of the Rana Pratap Sagar Lake that provides condenser cooling water to RAPS.

After completion of the above and various other upgradation jobs RAPS-2 was re-commissioned in 1998 as per the commissioning procedures approved by AERB.

4.2. Upgradation in MAPS Unit-1&2

MAPS-1 and MAPS-2 has been in operation since 1984 &1986 respectively. In same line as done for RAPS-2, the job of en masse replacement of all coolant channels were taken up in MAPS-2 after 8.5 EFPY and MAPS-1 after 10.1 EF PY of operation. Also a number of steam generator tube leaks had
occurred in these units in the past few years. All upgradations similar to RAPS-2 have been carried out in MAPS-2 and being done for MAPS-1. In addition to replacement of steam generator, the shutdown has also been used to install spargers in the calandria for improving moderator flow. This will enable the unit to be operated at 220 MW power.

5. Licence Renewal for Taps

Both Units of Tarapur Atomic Power Station (TAPS) were designed and constructed in late 60s with the assistance of GE of USA and were made operational in 1969. The design and analysis report specifies the design life of reactor vessel and major safety related equipments as 40 years. The continued operation of TAPS beyond the design life has been under consideration in AERB for quite some time. Based on this discussion, AERB approach for Renewal of Licence for TAPS beyond its design life of 40 years was formulated. AERB did not have guidelines for renewal of licence beyond design life. It was felt that the review of design basis and safety analysis in the light of current standards would be required and also of the other safety factors mentioned in the AERB guide (AERB/SG/0-12). This was an exhaustive work comparable to the safety review of the new plant.

After about 30 years of service in May 2000, AERB directed NPCIL, the utility to conduct a thorough review covering the following areas:

(a) Review of performance of equipment from operational experience.
(b) Review of Design Basis and Safety Analysis Report.
(c) Ageing assessment of systems, structures and components.
(d) Seismic re-evaluation.

All the studies have been completed and their reports have been reviewed by AERB. Based on these reviews, retrofits and upgradation requirements have been identified. Their implementation will be now progressively carried out over next 2½ years as per an agreed schedule between AERB and NPCIL.

5.1. Review of Operational Performance:

Operational performance of TAPS for the past 10 years was reviewed in accordance with the guidelines given in AERB/SG/0-12. The following safety factors relevant to operational performance were reviewed.

(a) Safety performance.
(b) Actual Physical Condition of the Plant
(c) Operating experience from other plants and research findings.
(d) Procedures
(e) Organisation and administration
(f) Human factors
(g) Emergency Preparedness
(h) Environmental Impact
5.2. Aging Management and Residual Life Assessment:

For reviewing aging management and residual life assessment guidelines given in AERB guide AERB/SG/0-12 were used. For the purpose of this review the components were categorized as

(a) **Major Critical Components:**

The major critical components, systems and structures are those that must remain functional during normal power operation and during those conditions and events for which the plant was designed including various anticipated operational occurrences, design basis accidents and external events. In general these components were:

(i) Part of reactor coolant pressure boundary.
(ii) Necessary to shutdown the reactor and maintain it in a safe shutdown condition.
(iii) Necessary to mitigate the consequences of serious accidents and prevent off site exposure.

Based on the above, the major critical components identified were:

(i) Components of Reactor coolant pressure boundary.
(ii) Reactor Containment.
(iii) Control Rod Drive mechanism.
(iv) Reactor recirculation pumps.

(b) **Important Systems:**

The important systems are the Engineered Safety Systems and other supporting systems necessary to operate the reactor in safe condition. These include:

(ii) Emergency Coolant Systems.
(iii) Decay Heat Removal Systems.
(v) Ultimate Heat Sink.

(c) **Other Critical Components:**

These are some important components necessary for safe operation of the plant and are not covered under (i) and (ii) above. These include:

(i) Reactor relief valves.
(ii) Primary steam isolation valves.
(iii) Primary feed pumps.
(iv) Reactor building ventilation fans.
(v) Civil structures other than containment.
5.3. **Review of Design Basis and Safety Analysis**

Objectives of this review is to assess the continued validity of (i) The original design basis for the plant systems and (ii) the original safety analysis.

Design basis of the plant systems is being reviewed taking into account changes in application codes/standards, better understanding of degradation mechanisms and availability of database on loads etc. The safety analysis is being reviewed to identify areas requiring revision in view of inputs from the review of design basis, changes in plant configuration, availability of better analytical tools etc. The review of the design basis covers the following systems:

i. Reactor Coolant System and Components.

ii. Reactor Protection system.

iii. Residual Heat Removal System.

iv. Engineered Safety features:
   - Primary Containment System.
   - Secondary Containment System.
   - Emergency Core Cooling System.

v. Waste Management System.

vi. Instrument Air System.

vii. Station Power Supply System.

viii. Ultimate Heat Sink.

ix. Fine Protection.

x. Control & Instrumentation Systems.

Apart from the above, the following issues are also considered for review:

a) ISI of important systems.

b) Plant diagnostic systems.

c) Results of Level-1 PSA.

d) Safety classification of Systems.

The system wise review is being carried out based on the guidelines of US NRC Standards Review Plan (SRP) for review of Safety Analysis Reports for Nuclear Power Plants (NUREG - 800).

5.4. **Irreplaceable Components:**

In addition to the above reviews, the key components, which cannot be replaced, are also being assessed carefully from the point of view of safety. These include Reactor Pressure Vessel, Containment, Reactor Pedestal, Reactor Vessel Support Skirt, Reactor Vessel Brackets and Fuel Storage pools.
(a) Reactor Pressure Vessel (RPV)

The reactor vessel is made of Carbon Steel (ASME SA-302 Grade B) and is 4.87 inches thick. The inside of the vessel is clad with stainless steel (5.6 mm thick). The state of the pressure vessel is monitored with the surveillance specimen programme. These include specimen drawn from the vessel base material, weld region and heat affected zone. So far, it is found that RPV material has adequate fracture toughness to assume safety of the pressure vessel till end of its service life of 40 EFPYs. In addition fatigue analysis of the RPV is being done considering thermal and pressure cycles. Certain welds and nozzles in the RPV are not accessible and have not been inspected. Program for inspection of some of these welds and nozzles is prepared and the inspection will be taken after development of remote tooling. The findings of these inspections will be extrapolated to assess the status of other welds and nozzles in RPV.

(b) Containment

Periodic inspection/monitoring of the status of the containment is carried out through Integrated Leak Rate Test and the thickness measurement of the drywell. No apparent reduction in thickness of drywell has been observed. Weld inspection of the drywell are also planned to be carried out. Visual inspection and thickness measurement of common chamber liner and suppression pool liners is also being carried out.

(c) Other non-replaceable SSCs

The remaining non-replaceable SSCs like Reactor Pedestal, Reactor Vessel Support Skirt, Reactor Vessel Bracket and Fuel Storage Pools are inspectable. Visual inspection of all these components indicated that their condition is satisfactory. In the case of reactor pedestal and fuel storage pools evaluation of concrete samples will be carried out. With respect to civil structures monitoring by visual and non-destruction techniques and necessary repairs based on assessment of condition of the structures will be carried out.

6. Conclusion

Since licence renewal review for BWRs at Tarapur plant (commissioned in 1969) was the first exercise of its kind, the AERB had to formulate the principles to be adopted and guidelines to be followed for such review. Safety upgradations identified after the regulatory review are being implemented to achieve reasonable safety margin.

First generation PHWRs at RAPS-2, MAPS-1&2 were commissioned in 1980, 1983 & 1985 respectively. Standardized Indian PHWRs were constructed in 1990 onwards and the current design confirms to the present regulatory requirements. Regulatory approach for safety upgradation and license review of these three earlier PHWRs was therefore different. The most recent version of standard PHWR was taken as reference for reviewing design basis. Safety upgradations were carried out in these plants during available long shutdowns.

The regulatory issues identified by AERB for the purpose of life extension and license renewals are,

- Review of the original design basis along with FSAR
- Systematic evaluation of plant and all safety-related structures and components
- Review of past plant performance and evaluation of safety improvement measures
J. Koley et al.

- Review of all relevant unresolved or generic safety issues.
- The implications of modern research and technology advancements
- Plant specific PSAs and their relationship to the traditional deterministic methods
- Review of the effects of ageing on safety-related structures and components
- Safety management and safety culture issues

REFERENCES


REGULATORY APPROACH TO SAFETY OF NUCLEAR POWER PLANTS BUILT TO EARLIER SAFETY STANDARDS

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Abstract. Swedish nuclear power plants have been built to earlier standards. To maintain safety for the plants a continuous process of modernisations has been in place since the commission stage. SKI’s regulatory approach for long term operation are described. New general regulations on design and construction for nuclear plants will be in effect next year in Sweden. The regulatory requirements will reflect an increased ambition level as regards safety.

1. Background

All Swedish nuclear power plants were designed and built in the late 60’s to the early 80’s to somewhat different safety standards and requirements and after commissioning continuously developed safety through several upgrading and modernisation activities.

The law in Sweden regulating nuclear safety is the Act on Nuclear Activities and it contains basic provisions on safety in connection with nuclear activities and applies both to the operation of nuclear plants and to the handling of all nuclear material and nuclear waste. The Act states that safety shall be maintained by taking all those measures required to prevent errors in or malfunction of equipment, incorrect handling or anything else that may result in a radiological accident. The law also says that the licensee of nuclear activities has to ensure that all measures are taken that are needed for maintaining safety. The licence to operate a nuclear facility is in principle unlimited in time.

2. Safety is a continuous process of modernisation

As a regulatory authority SKI has a duty to initiate safety improvements whenever justified by operating experience or research and development. Therefore SKI requires the licensees to conduct an active safety work including performing safety analyses using modern analytical tools. Deviations discovered have to be assessed and a yearly program for safety upgrading established. In SKI regulations it is stated that the licensee has to maintain and develop safety. In the word develop SKI includes a continuous work to search for safety deficiencies in the reactor design and the quality of the safety work and to take actions if deficiencies are discovered. The licensee is solely responsible for safety.

Also to maintain safety is a continuous process of modernisation. Modifications and replacements have to be made due to: operational experience, spare parts are difficult to find, I&C equipment becomes obsolete, ageing of components and materials, technical renewal and backfitting, regulatory requirements, utilities policies and so on.

3. Regulatory approach

The main focus in application of SKI’s regulatory strategy is on how the licensee fulfil his obligations by:
- Defining safety goals
- Further develop and implement effective organisations and efficient processes for maintaining safety as circumstances change
- Carry out all necessary self-assessments

The overall SKI regulatory approach to long term operation.

- Regulations with safety goals based on state-of-the-art in all important areas and that includes both technical, organisational and administrative requirements
- Inspection and supervision which focus on the licensees competence and resources to identify important ageing issues and to perform necessary measures
- Investigation and research to support SKI’s inspection and supervision activities and to keep the regulations up-to-date

4. Modernisation due to ageing and long term operation

For long term operation of older reactors it is important to have an ageing program to take care of physical ageing of systems, structures and passive and active components. In management programs the following aspects must also be properly addressed.

- Technology ageing e.g. instrumentation and control systems
- Requirement ageing, different requirements for different plant generations
- Safety analysis and documentation aging, safety analysis not always updated and based on new knowledge
- Personnel and management ageing by e.g. generation shifts and changes of attitudes

The approach for physical ageing

Requirements in SKI’s regulations on in-service inspections (ISI), in-service testing (IST) and maintenance. Systematic evaluation of failures and indications of generic ageing problems, in Swedish plants as well as in other similar plants, reporting of failures and indications of generic ageing problems. Inspection and supervision to monitor the licensees ISI, IST and maintenance activities.

To support SKI’s regulation and supervision data bases have been developed which include failures and incidents and all reported in-service induced degradation in all Swedish NPPs. SKI finance degradation and ISI related research

The approach for requirement ageing

SKI’s general safety regulation will be complemented with requirements for upgrading as far as possible to modern safety standard according to IAEA Design Standards. These requirements is expected to become effective 2005 with a reasonable transition period and will lead to extensive modernisation programs over the coming years.

The approach for technology ageing

The technical development in the I&C area is very fast and fundamental and equipment are becoming obsolet. Several programs for modernisation of I&C systems and control rooms have been carried out
in most plants and are envisaged to continue. SKI has followed the programs and related research activities.

The approach for safety analysis and documentation ageing.

In order to fulfil SKI’s requirements of up-to-date Safety Analysis Reports, the first and second generation BWR’s have re-assessed the plants safety analysis, produced more complete design specifications and performed many verifications (a design basis reconstitutions). The third and fourth generation BWR’s and the PWR’s will finalize this work in a couple of years. SKI follow the licensees work and assesses the results on a sample basis. The identified weaknesses have not been serious enough to interfere with the safe operation of the plants, but they have been corrected.

The approach for personnel and management ageing.

According to SKI’s regulation the licensee has to secure that enough personnel with necessary competence are available, responsibilities have been defined and are documented and the personnel have working conditions needed to perform their tasks in a safe way. During the last years SKI has systematically assessed the licensees activities to fulfil these requirements.

5. New regulation on design and construction

Based on operating experience, safety analysis, research and development and updated international standards, SKI has decided to issue general regulations on design and construction that SKI believes are justified, in the light of operation of Swedish nuclear reactors beyond 2010. All Swedish nuclear power plants shall comply with that regulation.

The new draft regulations include requirements on design principles, resistance against failures and events, environmental qualification, requirements on the central- and emergency control rooms, safety classification, Postulated Initiated Events classification, management following severe accidents as well as design requirements on the core and fuel. SKI considers the proposed regulations will correspond to a modern safety standard for existing reactors. SKI will allow time for implementation, as extensive plant modifications need to be done in same cases, particularly for the older reactors.

The proposed regulatory requirements will reflect an increased ambition level as regards safety, but in many cases they reflect a sort of “codification of modern practise” already used in Sweden but not yet formulated in general regulations. The proposed regulations will be put to the SKI Board for final decision later during this year.

SKI has used the IAEA documents as one important source to develop its new regulation. IAEA represents a development in reactor safety area that has a good general picture including safety goals and guidelines based on experience. Human factors and safety culture are given necessary focus.

The strategy for regulation

Deterministic rules are the basis for design and operation. Probabilistic safety analysis (PSA) is one of the necessary analysis to verify a sufficient safe plant but PSA itself is not enough. As a minimum, all problems and weaknesses that are known so far should be addressed. SKI:s regulations will not cover everything but focus on some important areas.

Experience from PSA should be taken care of. Other systems and functions than the safety systems have to be included in the defence-in-dept concept to achieve an acceptable level of safety. Also the safety graded system has to be used in an extended way beyond its original purpose and emergency procedures. The importance of auxiliary power and support systems was underestimated, the importance of external events was underestimated, the probability of common cause failures was underestimated. The importance of risk during shutdown period was underestimated.
Conditions for operators must be better. Grace time or consideration time before actions from control room and emergency control room must be taken. Clarity and transparency in control room. Human factors taken into account in a systematic way.

Robust to failures. Simplicity, robustness and transparency in design are necessary to maintain safety. Single failure criterion fulfilled also during repair and preventative maintenance. Diversified safety functions as a defence against common cause failure. Redundancy and diversification of the reactor protection system. Separation and independence between redundant components and different safety functions. Robustness is equally important to power system and auxiliary system as the systems they support. Fail-safe also in case of external events.

6. **Safety cost**

Swedens oldest reactor finished an extensive modernisation program in year 2002. It was upgraded to a level comparable to modern standards. That plant have been operating for almost 32 years and has been shut down in total for about 6 years during that time to correct detected safety deficiencies and to make safety upgrades. That is a substantial loss of production. Cost for engineering, assessment and review can also be large because the safety upgrading taken benefit from the original design can give rather unique solutions and than there are of course cost for hardware new buildings, new components and safety systems.

7. **“A modern reactor”**

A modern reactor from a technical safety view is a reactor that fulfil all original and additional demands given on design and operation, has evaluated operating experience and new knowledge and made reasonable improvements with the object to achieve a safety level comparable to new reactors. A reactor built to early standards with a continuous process of modernisation can in that sense be a modern reactor.

**REFERENCES**

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PRESENT STATUS OF JAPANESE NATIONAL RESEARCH PROJECTS RELATING TO AGEING OF NUCLEAR POWER PLANTS

J. Sanoh\(^a\), T. Noda\(^b\), K. Maeda\(^b\)

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\(^b\) Ministry of Economy, Trade and Industry, Tokyo, Japan

Abstract. At the present time, 11 national research projects are being conducted to provide the safety regulatory administration to the aged nuclear power plants. Some interim results have been acquired on development of a prediction equation on Charpy upper shelf energy reduction by neutron irradiation, reconstitution technology for surveillance test specimen, and surface modification technology for irradiated stainless steels. This report summarizes the present status of Japanese national research projects relating to ageing of nuclear power plants.

1. Introduction

In Japan, 52 commercial light water nuclear power plants are operating. Among them, 4 units have been operating over 30 years and 13 units are over 25 years. The Japanese countermeasures against the ageing of nuclear power plants (NPP’s) were already reported at the IAEA symposium on nuclear plant life management (PLM) held in Budapest [1]. The Nuclear and Industrial Safety Agency (NISA) of the Ministry of Economy, Trade and Industry (METI) had required the electric utilities to carry out the technical evaluation of their plants before 30 years operation and to establish the long-term maintenance programmes for the following 10 years operation as the PLM. The NISA has been evaluating the propriety of the technical evaluation and the long-term maintenance programmes carried out by the utilities.

In October 2003, the technical evaluation and the long-term maintenance programme as the PLM were enacted the Japanese domestic laws. Thus, the NISA strengthened the safety regulatory administration on the PLM for the aged NPP’s.

On the other hand, it was pointed out that latest technologies and knowledges should be applied on the PLM to provide the safety regulatory administration. Therefore many national research and development relating to the ageing have been conducted as national research projects.

By the end of March 2004, the Japanese regulatory authorities and electric utilities had evaluated totally 9 units using the latest technical knowledges by reflecting up-to-date results of research activities and operational experiences of NPP’s relating to ageing.

2. Current National Research Projects Relating to Ageing of Nuclear Power Plants

In 1996, the Japanese regulatory authorities issued the basic policy on aged NPP’s [2]. The basic policy pointed out the needs for advancing the technologies on inspection and monitoring, evaluation method for ageing, and protective maintenance and refurbishment. In order to enforce the safety regulatory administration precisely to aged NPP’s, the NISA has been promoting the national research projects which will contribute to enhance the knowledge on material degradation such as neutron embrittlement, and the strategies for the PLM of aged NPP’s. The NISA has entrusted enforcement of the national research projects to the Japan Power Engineering and Inspection Corporation (JAPEIC) and, since October 2003, to the Japan Nuclear Energy Safety Organization (JNES).
The national research projects had been started in mid 1980. The Pressurized Thermal Shock (PTS) project and the Plant Life Extension (PLEX) project had been conducted from 1983 to 1991 and from 1985 to 1996 respectively. The former established the Japanese own PTS criterion based on a deterministic fracture mechanics approach and the latter provided the basic data and knowledges on the PLM for the NPP ageing. Table 1 shows the current Japanese national research projects relating to ageing of NPP’s. The outline of some of these projects were reported at the IAEA Symposium in Budapest [3]. As of April 2004, 11 projects are carried out by the JNES. Among these projects, 7 projects are relating to evaluation of material degradation such as irradiation embrittlement, fatigue and stress corrosion cracking, 3 projects are relating to inspection and monitoring and 1 project is relating to preventive maintenance and refurbishment. The NISA requires that these research projects shall contribute to the safety regulatory administration on NPP ageing issue through standardization of their results.

Table 1. Current National Research Projects Relating to Ageing of Nuclear Power Plants

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<tr>
<th>Project Name</th>
<th>Category</th>
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<tr>
<td>NPP Integrated Safety Management Technology (PLIM)</td>
<td>Material Degradation</td>
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<td>Environmental Fatigue Test (EFT)</td>
<td>Material Degradation</td>
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<td>Integrity Assessment of Flaws in Structural Discontinuities (IAF)</td>
<td>Material Degradation</td>
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<td>Assessment of Cable Aging for NPP (ACA)</td>
<td>Material Degradation</td>
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<td>Evaluation Technology of SCC Growth for Nickel Base Alloy (NiSCC)</td>
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<td>Evaluation Technology of Irradiation Assisted SCC (IASC)</td>
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<td>Intergranular SCC of Low Carbon Stainless Steels for NPP (IGSCC)</td>
<td>Material Degradation</td>
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<td>Repair Welding Technology of Irradiated Materials (WIM)</td>
<td>Maintenance Technology</td>
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<td>NPP Maintenance Technology (PMT)</td>
<td>Maintenance Technology</td>
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<tr>
<td>Standards and Guides for Formation of Up-graded Inspection System</td>
<td>Inspection Technology</td>
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<tr>
<td>Non-Destructive Inspection Technology on Ni-Alloy Welded Joint (NW)</td>
<td>Inspection technology</td>
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<tr>
<td>Non-Destructive Inspection Technology for Shroud Integrity Assessment (NSA)</td>
<td>Inspection technology</td>
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3. Some Interim Results of Research Projects and Standardization

3.1. Neutron Irradiation Embrittlement

3.1.1. Prediction of Charpy Upper Shelf Energy Reduction

Japanese prediction equation for transition temperature shift of RPV steels as well as evaluation procedure and criteria for the integrity of RPV at the pressurized thermal shock events were already standardized as the Japan Electric Association Codes, JEAC 4201 and JEAC 4206 in1991 based on the results of the PTS project. As for the regulation of the upper shelf region, neither authorized method to predict the toughness for RPV nor the method to evaluate the integrity for RPV steels of which Charpy upper shelf energy (USE) is less than 68J have been prepared in Japan. Development of predicting equation for USE reduction and correlation equation between USE and fracture mechanics parameter were completed by the PLIM project at the end of 2002FY. Combined with Japanese surveillance data, the prediction equation for USE reduction was established from the results of PLIM project where 13 kinds of test steels were irradiated at the HALDEN test reactor in Norway and the effects of chemical compositions, neutron flux, etc. were investigated. Both Cu and Ni contents were identified as important variables. Neutron flux was excluded from the form of the models, because the result of reference monitoring material used in the PLIM test programme implied that the neutron flux effect on USE is not so large. As a consequence of evaluation of several candidate model forms by statistical analysis, USE prediction equations found are as follows [4].
(Base Metal)
\[ \Delta \text{USE} \% = -0.95 + (5.23 + 9.36(0.5 + 0.5 \tanh((\text{Cu} - 0.087)/0.034))(1 + 0.59\text{Ni})) \times f^{0.349 - 0.068 \log f} \]

(Weld Metal)
\[ \Delta \text{USE} \% = -2.78 + (9.78 + 3.96(0.5 + 0.5 \tanh((\text{Cu} - 0.086)/0.045))(1 + 3.63\text{Ni})) \times f^{0.234 + 0.015 \log f} \]

\( f \): neutron fluence \( (10^{19} \text{n/cm}^2, E>1\text{MeV}) \)

These equations have standard deviations of 6.9% for the base metal and 7.5% for the weld metal respectively. Standardization of the prediction method for USE reduction and evaluation procedure for integrity of RPV with low USE are proceeding to add to the JEAC 4201 and 4206. These guidelines are scheduled to be published within 2004.

3.1.2. Reconstitution of Surveillance Test Specimens

The PLIM project is also developing a reconstitution technique of neutron irradiated test specimens for surveillance programme. The reconstitution technique is, as shown in Figure 1, joining an insert which is machined from the broken specimen with tabs by the surface activated joining (SAJ) method. Although the length of the insert is required not less than 18mm in ASTM E1253-99, in the PLIM project, the minimum length of the insert is aimed for 10mm assuming that L-T direction Charpy surveillance specimens which have been applied to an early domestic NPP’s are reconstituted into T-L direction specimens. The effect of heating during joining process on USE with shortening the length of insert is investigated now. Figure 2 shows an example of the USE of reconstituted irradiated specimen with 16mm length insert [5]. The USE of reconstituted specimens coincides well with the original one. This length of insert is enough to prevent the interaction with heat affected zone of joining. Further tests with short length insert and reconstitution of the compact tension specimen are under succession. The reconstitution of surveillance test specimens is scheduled to be completed and standardized in 2005.

3.2. Surface Modification and Repair Welding of Neutron Irradiated Materials

Surface modification and repair welding of neutron irradiated materials are very important to maintain the integrity of aged reactor pressure vessels and core internals. The PMT project and the WIM project develop the surface modification and the repair welding process for neutron irradiated stainless steels. The problems to be solved are the deterioration of weldability caused by helium (He) produced through \( (n, \alpha) \) reaction during operation. The helium bubbles at grain boundaries of weld heat affected zone (HAZ) are one of the causes of the cracking. The countermeasure against the cracking is to limit the heat input during welding process. Yttrium-Aluminum-Garnet (YAG) Laser technique is known as one of the candidate methods for surface modification and repair welding of neutron irradiated materials. The results of the PMT project which has completed at the end of March 2004 reveals that both irradiated type 304 and type 316L stainless steels have sensitivity to He induced cracking under high heat input condition. Figure 3 shows an example of He induced crack observed in HAZ of Laser
J. Sanoh et al.

surface modified specimen. Figure 4 shows the He content (neutron fluence) and heat input condition which is free from the He induced cracking by Laser cladding of irradiated type 316L stainless steel [6]. This diagram is planned to be provided as one of the guidelines to evaluate the integrity of surface modification process for the neutron irradiated stainless steels. The WIM project is scheduled to be completed by the end of March 2005. The guidelines for repair welding of irradiated stainless steels will also be established.

FIG. 3. Appearance of He Induced Crack (Irradiated Type 316L)

FIG. 4. Crack free Range for Laser Cladding of Irradiated Type 316L Stainless Steel

4. Conclusion

Since mid 1980, the national research projects have been conducted to provide the safety regulatory administration on the aged NPP’s.

On the other hand, since 1996 the Japanese regulatory authorities have grappled with the PLM issue and the national research projects have been carried out to enhance the technical evaluation and the safety regulations for the PLM issues. At the present time, the standards and the guidelines for the PLM are establishing gradually based on the results of these research and development.

The results of the research projects shown in this paper shall be used to establishing the safety criteria and evaluation of aged NPP’s.

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[6] KANNO, M., et al., to be published
ENVIRONMENTAL IMPACT ASSESSMENT IN THE FRAME OF LICENSING LIFETIME EXTENSION OF PAKS NPP AT HUNGARY

G. Volent, G. Németh
Paks Nuclear Power Plant, Paks, Hungary

Abstract The lifetime extension of a nuclear power plant requires permit decisions in accordance with several acts. In the permit procedures, the environmental permit is the first stage that is based environmental impact assessment. In Hungary, the overall EIA process there is 2 main stages. The process of EIA is usually involves a number of steps. The order of steps in the process may vary.

The preparation of EIA procedure for life time extension of Paks NPP started in 2002, however the program for site evaluation including data collection, additional releases and environmental monitoring, performing impacts studies was completed in 1999 and was started in 2001. The environmental permit procedure of lifetime extension of Paks NPP has been started, the preliminary EIA Study was submitted to authority in April 2004. Paks NPP is planning to complete the detailed EIA Study till end of 2005.

1. Introduction

The four units of the Paks NPP were put into operation between 1982 and 1987. Taking into account the designed lifetime (30 years), they should be shut down between 2012 and 2017. In possession of our present technical knowledge it can be considered as a real long-term goal to extend the designed lifetime of the units with 20 years.

The lifetime extension of a nuclear power plant requires permit decisions in accordance with several acts. In Hungary the necessary permits are an environmental permit in accordance with Environmental Protection Act and operating license in accordance with the Nuclear Energy Act. The operating license is including multilevel permit procedure.

The first step of licensing process of lifetime extension is the environmental permitting, which is based on environmental impact assessment.

2. Environmental impact assessment

Environmental Impact Assessment (EIA) is a process, which identifies the environmental effects (both negative and positive) of development proposals. It aims to prevent, reduce and offset any adverse impacts.

In Hungary, the overall EIA process there is 2 main stages. In the first stage the applicant undertakes an assessment so that environmental issues can be taken into account during the design of the project. This involves consultations, data collection and environmental studies to identify the effects and propose mitigation measures to prevent, reduce and offset them. This is reported in a preliminary EIA Study, which is submitted to environmental authority. If the environmental authority agree to EIA Study the authority undertakes the second main stage by critically evaluating the statement, seeking further information from the applicant if necessary and taking into account additional consultations and public representations.
The process of EIA is usually involves a number of steps. The order of steps in the process may vary:

Scoping seeks to identify at an early stage the project’s possible impacts and from all the alternatives that could be addressed, those that are key significant issues.

Consideration of alternatives seeks to ensure that the proponent has considered other feasible approaches.

Description of the project/development action.

Description of the environmental baseline includes the establishment of both present and future state of the environment.

Identification if key impacts.

The prediction of impacts.

Evaluation of assessment of significance seeks to assess the relative significance of predicted impacts.

Mitigation involves the introduction of measures to avoid, reduce, remedy or compensate for any significant adverse impacts.

Public consultation and participating, including Environmental Impact Study presentation, which is a vital step in the process.

Post-decision monitoring involves the recording of outcomes associated with development impacts, after decision to proceed.

Auditing follows from monitoring.

When the environmental impact assessment is performed for an operating facility the content of the above-described steps can modify. Description of the environmental baseline, for example, does not mean the same for a projected power plant than after 20 years real operation. Also, the data on environmental conditions are differ in availability, in quantity and in quality in case of projected or operating facility.

3. Environmental impact assessment at Paks

The preparation of EIA procedure for life time extension of Paks NPP started in 2002, however the program for site evaluation including data collection, additional releases and environmental monitoring, performing impacts studies was completed in 1999 and was started in 2001.

There is only one nuclear power plant in Hungary, this is way the procedure of EIA of lifetime extension of nuclear power plant was not elaborated in Hungary. The steps and the content of the steps of EIA were developed in close cooperation between authorities, independent experts and experts of Paks NPP. In the frame of these consultations among others the followings were clarified:

- Interpretation of the environmental baseline
- Interpretation of normal and not normal operation of nuclear power plant
- Extension (time, distance and quantity) of environmental monitoring program around the plant
- Appreciation of the significance of predicted impacts
G. Volent and G. Németh

The well-defined and based system for appreciation of the significance of predicted impacts is particularly important because the involvement local authorities and citizens depending on that. The competent authorities approved the system for appreciation of the significance of predicted impacts. The appreciation of the significance of predicted radiological impact, as an example, is summarized in Table 1.

Table 1. Assessment of the significance of predicted radiological impact

<table>
<thead>
<tr>
<th>Change of state</th>
<th>Excess radiation exposure of public (effective dose /E/ microSv/y or microSv/event)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Neutral</td>
<td>( \leq 90 )</td>
</tr>
<tr>
<td>Tolerable</td>
<td>( 90 &lt; E \leq 1000 )</td>
</tr>
<tr>
<td>Disturbing</td>
<td>( 1000 &lt; E \leq 10000 )</td>
</tr>
<tr>
<td>Damaging</td>
<td>( 10000 &lt; E )</td>
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</table>

4. Summary

The environmental permit procedure of lifetime extension of Paks NPP has been started, the preliminary EIA Study was submitted to authority in April 2004. Paks NPP is planning to complete the detailed EIA Study till end of 2005.
OPERATOR – REGULATOR INTERFACE IN SAFETY ASSESSMENT OF KANUPP FOR LIFE EXTENSION

W.M. Butt, A. Zia
Karachi Nuclear Power Plant

Abstract. Karachi Nuclear Power Plant (KANUPP) a CANDU plant, started commercial operation in 1972 with nominal design life of 30 years. Throughout the operating life, plant safety has been continuously assessed by KANUPP staff, as well as by international experts for the safe operation of plant. The process of review of safety assessment was started in early 90’s to incorporate improvements in phases i.e. well before the completion of 30 years design life of the plant. This led to Life Extension of KANUPP.

As a part of Life Extension of KANUPP and to meet the regulatory requirements for re-licensing, safety assessments and engineering studies were carried out to ensure safe operation of plant beyond design life. Corrective action plan was developed for implementation in two outages.

The plant was shutdown in early December 2002 for the first phase of work related to re-licensing. In January 2004 special License was given by the regulatory body to re-start the plant. The remaining activities for re-licensing will be completed in the next outage planned in January 2005.

Since the life extension of a nuclear power plant was a totally new concept both for KANUPP and PNRA (Pakistan Nuclear Regulatory Authority), close collaboration and understanding was essential between operator and regulator. This paper describes the operator-regulator interface in the Safety Assessment of KANUPP for Life Extension, and the application of IAEA Standards / Guides for Life Extension.

1. Introduction

Karachi Nuclear Power Plant (KANUPP) is a 137 MWe CANDU Pressurized Heavy Water Reactor (PHWR). Built in the late sixties, it is among the first generation CANDU nuclear generating stations and is certainly the pioneer Canadian off-shore plant. The plant, commissioned in 1972 completed its nominal design life of 30 years in 2002.
During the design life of plant, safety has been assessed/reviewed by IAEA missions (OSART, ASSET), WANO Peers, regulatory body and expert missions in different areas. One of the main objectives of inviting the expert missions was to assess and improve the safety of the plant to meet the current safety standards as far as practically possible. Based on the assessments made by the missions and the results of periodic inspections, the critical equipment of the plant such as reactor fuel channels, steam generators, major nuclear island equipment and turbine-generator have been generally found to be in good health. Degradation analysis and life management programs were developed for the critical equipment from the point of view of cost benefits of operating the plant beyond design life. This led to our plan for Plant Life Extension.

In October 2001, Pakistan Nuclear Regulatory Authority (PNRA) was requested for the permission to operate beyond 30 years of design life. KANUPP is in the process of extending its life by 15 years. The challenges associated with safety assessment of plant for extended life operation are to develop a common basis with the regulator for the acceptable level of safety, to identify the safety issues, determine their safety significance and decide with the regulator about the acceptability of corrective measures to maintain the margin of safety by improvements and addition of the safety systems.

2. Systematic Approach to Life Extension of Older Plants – KANUPP

The plants built to earlier standards do not conform to all the current criteria and standards for design. This does not necessarily make them unsafe. As a part of life extension of older NPPs, neither it is necessary nor feasible for the existing plants to comply with all the new/current standards.

The extent to which plants built to earlier standards are upgraded to current standards vary from country to country. Even if such improvements fail to conform to current standards, this does not necessarily mean that plants built to earlier standard are unsafe; however, there is a need for a periodic and systematic review (SRS # 12).

Retrospective application of current applicable standard and practices need to be done carefully through a logical and systematic process of evaluation of safety significant issues. The aim is to do all that is reasonably and practically possible. In some cases, interim solutions and compensatory measures would meet the objectives of the applicable standards.

The older plants can maintain and demonstrate acceptable level of safety owing to conservative margin of safety in their design, regular in-service inspection results, ageing management and safety backfits.

The systematic approach to safety assessment for extending operating life of NPP includes:

- To determine whether the plant conforms to the standards for design and operation that were applicable when it was first licensed.
- Deterministic evaluation to assess the design and operation of the plant against current standards and practices.
- Identification of safety issues through comparison between current plant state and operating practices and the applicable current safety standards.
- Use of PSA to evaluate design weaknesses and to estimate core damage frequency.
- Potential corrective actions are screened to determine which are important enough for implementation.
- Results from the deterministic evaluation and PSA are used in prioritizing the corrective actions.
- Prioritization so that the time and resources spent on improvements are dedicated to the most urgent matters.
- To develop an integrated and prioritized action plan.
3. Why Plant Life Extension (PLEX) of ANUPP

Karachi Nuclear Power Plant during its 30 years operation, has been operating mostly below 100 MWe, being equivalent to nearly only 11 Effective Full Power Years (EFPYs). This is the first indicator for the remaining useful life of plant.

The results of periodic inspections and condition monitoring have revealed that major equipments of the plant including Fuel Channels, Steam Generator, PHT Piping, Feeders, etc. are in good condition when compared with the condition of these equipment / components in other operating CANDU plants.

KANUPP has operated safely with no incident that would cause any concern to the safety to the plant as well as general public in overall perspective. Plant has at all times met the radiation release limits while still working on the principle of ALARA.

Expert missions reviews covering almost all the areas and safety systems of plant, follow-up actions to enhance safety of plant and measures taken to overcome ageing and obsolescence since nineties for safe operation of plant during design life.

A joint PAEC / WANO seminar on “Nuclear Power Plant Ageing, Refurbishing and Plant Life Extension (PARPLEX)” organized in April 1992 at Karachi. This was the starting point to Life Extension.

4. Safety Assessment & Enhancements During Design Life

Like all other old plants, KANUPP has also encountered usual age-related problems, equipment obsolescence and safety issues arising from earlier safety standards as compared to the current standards. This is primarily due to the fact that the safety has continuously evolved as a result of research, technological advances and operating experience.

Based on the performance of the plant, reviews by various expert missions, results of periodic inspections and the operating experience feedback from other NPPs (through WANO, COG, IAEA etc), different programs were initiated in early nineties to enhance safety of plant in line with current international standards (as far as practically & reasonably possible) to operate plant safely during 30 years design life and with the ultimate objective to extend plant life. The areas covered under the safety enhancement program were:

- Operational Safety
- Design Safety
- Ageing
- Obsolescence

Being a nuclear power plant of sixties, KANUPP has always given highest priority to improve and maintain the safety of plant. All possible efforts have been made to share and exchange operating experiences with nuclear community. Under various IAEA Technical Cooperation projects, many expert missions were invited to KANUPP for the review of different plant systems with the objective to improve the overall safety of plant. Some of the areas covered by expert mission are:

- EQ Review
- Seismic Walkdown / Review
- Review/ improvement of Safety Systems of Plant
- Fire Protection Review
- Ageing Management Program
- ISI Program for SGs and other equipment
- Review of PSA results
- Waste Management Program
W.M. Butt and A. Zia

The expert mission reviews / walk downs and implementation of their recommendations has resulted in safety enhancement of KANUPP.

5. Regulatory Requirements for Re-Licensing

The regulatory requirements for re-licensing (plant operation beyond design life) in Pakistan are:

i) Periodic Safety Review (PSR)
ii) Revised Final Safety Analysis Report (KFSAR)
iii) PSA Level-1.
iv) Decommissioning Program
v) Environment Assessment
vi) Implementation plan for safety upgrades.

To meet the above mentioned regulatory conditions for re-licensing, comprehensive implementation plans were developed in consultation with regulatory body (PNRA). The progress of implementation work was reviewed by the regulators on regular basis throughout the re-licensing outage till the re-start of plant under special license.

6. Periodic Safety Review (PSR)

Periodic Safety Reviews (PSR) of plant was one of the requirement of re-licensing from PNRA in line with IAEA new Safety Standard NSR-1, IAEA Safety Guide No. 50-SG-012, IAEA Safety Reports Series No. 12 and PNRA Regulation PAK/909, PAK/912. PSR is considered best way for assessment of existing condition of plant to determine necessary or worthwhile changes that should be made in order to maintain a high level of safety.

To assess the existing condition of plant and propose corrective actions to improve the overall safety of the plant, PSR carried out by KANUPP covered the internationally acceptable elements or safety factors which are as under:

1- Actual Condition of SSCs
2- Safety Analysis (FSAR)
3- Equipment Qualification
4- Ageing Management
5- Safety Performance
6- Use of Operating Experience
7- Procedures
8- Organization and Administration
9- Human Factors
10- Emergency Planning
11- Radiological Impact on Environment
12- Plant Design
13- Hazard Analysis

Dedicated teams were constituted for assessment of above factors of PSR with the objective to complete the safety review within 6-8 months.

Comprehensive action plan was prepared in consultation with the regulator to resolve the weaknesses identified as a result of PSR. This action plan became part of the overall implementation plan for re-licensing. The corrective actions and time frame for completion were agreed with the regulator for follow-up.
7. Implementation Plan For Safety Upgrades:

Implementation Plan was prepared by KANUPP to address 24 Safety Issues pointed out by PNRA. The Safety Issues include the following:

**Category – I**

1- Emergency Injection System Inadequacies (ECI)
2- Control Room Habitability (CRH)
3- Fire Prevention and Control
4- Environmental Qualification (EQ)
5- Physical Protection Measures
6- Fuel Channel Integrity Assessment (FCIA)
7- Feeder Pipe Thinning (FPT)
8- Auto Boiler Crash Cooldown (ABCC)
9- Steam Generator Tube Integrity
10- Single Shutdown System
11- AD Sumps protection from LOCA debris
12- Reliability of D.C. Power Supplies
13- Post Accident Monitoring System (PAMs)
14- CGB connectors leak tightness
15- AH system tubing replacement
16- Seismic design
17- Ageing Management Program (AMP)
18- 

**Category – II**

1- Removal of booster rods
2- Reliability of off-site power
3- High radiation fields in PHT
4- KFSAR Update
5- Asbestos hazards
6- Unplanned outages
7- Waste Management

Most of the above mentioned safety upgrades have been completed/implemented except the redundancy of Emergency injection System which is planned in the next outage alongwith many other safety retrofits.

8. Operator – Regulator Interface:

Since the life extension of a nuclear power plant was totally new concept both for KANUPP and PNRA, all possible efforts were made to develop understanding and consensus (as far as possible) from the early phases of safety assessment for life extension of KANUPP. This requires high level of technical competence and trust to understand each others commitments towards the national program. This would develop over the years. The process of safety assessment covered the following main areas:

a) **Applicable Codes & Standards:**

After finalization of Regulatory requirements for Life Extension (refer Section-5), meetings were held with PNRA, to decide the applicable codes and standards for each activity of Re-licensing. A three-tier system was introduced by PNRA, which consisted of Regulations followed by Guides which in turn were followed by Industrial Standards. The specified regulations were deemed to be mandatory while the identified guides and industrial standards (if any) were not mandatory.
PNRA gave KANUPP the choice of following PNRA formulated PAK regulations or the equivalent USNRC regulations. KANUPP decided to opt for the PAK regulations for the re-licensing work. Generally the applicable PNRA regulations are equivalent to IAEA-NSRI and IAEA-NSR2 albeit more stringent. This resulted in more challenges to be faced.

In general PNRA and KANUPP agreed that it would be mandatory to use IAEA safety report series # 12 (Evaluation of the Safety of Operating NPP’s Built to Earlier Standards) for evaluating safety issues. The regulatory requirements and guides for the 13 factors for Periodic Safety Reviews and PSA-1 were mutually agreed. PNRA had identified 24 issues out of which 17 were considered of higher importance. Regulatory requirements and guides for all the 24 issues were also mutually agreed upon.

b) Safety Issue Categorization:

The weakness and safety issues identified as a result of Periodic Safety Review and safety assessment of plant systems can best be managed by organizing them into categories. The categorization of a safety issue depends on its impact on plant safety which in turn determines the return of actions required to resolve the safety issue. Categorization of safety issues was done according to their safety significance (High, Medium, Low). IAEA Safety Report Series 12 was accepted by the regulator for Safety Issue Categorization.

c) Corrective Measures:

Once the significance of each safety issue on safety of plant has been evaluated, corrective measures have to be developed to resolve the issue. The corrective measures could be installation of a new equipment or qualification of equipment or development /modification of procedures or operational restriction etc. or a combination of these. Wherever possible PSA was used for assessing the feasibility of alternative corrective measures. Cost benefit analysis was also used in the judgment of corrective measures. For all corrective actions, agreement with regulator was developed based on the studies carried out by plant.

d) Prioritization of Corrective Measures:

The corrective measures agreed with the regulator have to be prioritized to ensure that the available recourses are best utilized for effective upgrading of plant and net safety benefit. Highest priority is given to those measures which will achieve or restore the most important defense in depth feature and lowest priority is given to those corrective measures for enhancement of the mitigating plant features. Mandatory requirements need to be implemented separately from the process of prioritizing corrective measures. Based on this prioritization process, re-licensing activities of KANUPP were agreed and planned in two separate outages.

e) Implementation Plan:

Based on the process of safety assessment, categorization and prioritization, comprehensive plan was prepared in consultation with the regulator for implementation in two phases. The safety issues of high safety significance were addressed in the first phase or interim/compensatory measures were taken with supporting analysis to resolve all such issues to ensure that necessary condition for safe plant operation have been fulfilled.

The implementation plan to reflect all the details including work layout, procurement of materials, control of modifications, inspection requirements, personal training, health physics control, quality assurance etc.
9. Site Work for Life Extension

As a part of life extension, a comprehensive implementation plan for safety upgrades was finalized with the regulatory authority. Most of these safety issues were known to KANUPP and necessary studies had already been conducted under different safety assessments & enhancement projects (see section 4.0). To address and resolve these safety concerns, a long outage was necessary. Implementation plans were submitted to the regulatory authority for the execution of work in two phases i.e.

a) Short-term activities to be completed in the first outage starting from December 2002.

b) Long term activities requiring qualified equipment or major engineering / modification.

To review the progress of work for license renewal, utility – regulator meetings were held regularly with the objective to discuss and resolve all the technical matters and develop a consensus (as far as possible).

10. Quality Assurance Programs for Re-Licensing

Effective Operation Quality Assurance Program (OQAP) to cover the re-licensing activities and plant operation beyond design life was one of the regulatory requirements. The adequacy of the in-place OQAP was assessed and a full scope Quality Assurance Program was developed in line with national regulations (PAK/912) on the Safety of Nuclear Power Plant Quality Assurance. The following activities related to re-licensing particularly to safety and safety related system were incorporated in the OQAP:

- Design Control and Analysis for Re-licensing.
- Ageing Management
- Nuclear Safety & Licensing.
- Environmental Assessment
- Self Assessment
- Decommissioning
- Plant Physical Protection
- Radiation Protection
- Procurement, Receiving, Handling & Storage
- Non-conformance Control

11. Application of IAEA Standards / Guides

In absence of vendor support and non availability of applicable standards for life extension of a CANDU plant, IAEA Standards and Guides have been extensively useful in all stages of safety assessment program for life extension starting from Periodic Safety Review to prioritization and implementation. Some of the IAEA guides used are:

- IAEA SRS # 3 (50-SG-D11)
- INSAG-8
- INSAG-3
- INSAG-6
- INSAG-5
- IAEA TECDOC 832
The application of IAEA guides along with the national regulatory standards have been very challenging task to fulfill the regulatory requirements for re-licensing considering the constrains of non-availability of qualified equipment, practical problems of retrofits in the existing system, interpretation of IAEA standards/guides and sometime high expectations of the regulator.

12. Conclusion

After completion of design life, Karachi Nuclear Power Plant (KANUPP) has carried out many safety upgrades and retrofits in addition to the implementation of regulatory requirements for re-licensing. Now plant is operating at reduced power under special license. Additional safety retrofits are planned in the next outage to complete re-licensing requirements for life extension.