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Research Reactor Operation and Maintenance and Ageing

RENEWAL OF THE OPERATING LICENCE AND UPGRADING THE REACTOR FOR CONTINUING AND IMPROVED SERVICE OF THE FIR 1 INTO THE 2020'S

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ABSTRACT

The FiR 1 reactor, a 250 kW TRIGA reactor, has been in operation since 1962. The current operating licence expires at the end of the year 2011. An application for a new operating licence will be submitted to the authorities by the end of the year 2010. The documents needed for the application will be prepared or revised during 2010.

The long term strategy for FiR 1 has been updated. The continuation of the development of Boron Neutron Capture Therapy (BNCT) cancer treatments is considered nationally and internationally important and even ethically necessary. Development of linear accelerators to substitute reactors as a neutron source for BNCT have started. Meanwhile development of BNCT to an established treatment for several cancers requires FiR 1 as a demonstration and reference facility at least until the end of year 2023.

Upgrades for continuing and improved service into the 2020's are under consideration. These include planning for renewal of the instrumentation in the midway of the new licensing period. Preparations for a power upgrade during the next licensing period will also be made. Doubling the power to 500 kW will allow increasing the capacity for BNCT treatments by more than a factor of three, meeting the prospects of increased demand for these treatments, while still keeping the current capacity for radioisotope production, education and training.

For the power upgrade two options are considered. The first option is to replace the old aluminium clad fuel still constituting close to half of the reactor core with new steel clad fuel suitable for higher power use. The other option is to use less new steel clad fuel by using old aluminium clad elements in low power density parts of the core. This option was already preliminarily studied in 2000 and now more detailed reactor physics and thermal hydraulics calculations using the most advanced codes in use at VTT will be performed to evaluate this option.

The modelling work will be performed by the young generation of reactor physicists at VTT and will thus educate a next generation of experts familiar with the TRIGA reactor. They are needed already on the ground that part of the current reactor personnel will retire during the next licensing period.

1. Introduction

FiR 1 -reactor is a TRIGA Mark II type research reactor manufactured by General Atomics (San Diego, CA, USA). The FiR 1 started operation in 1962, reactor power was increased in 1967 from 100 kW to 250 kW, and in 1982 the reactor instrumentation was renewed. In 1996-1997 the reactor building was completely renovated and the ventilation and reactor cooling systems were replaced. The main purpose to run the reactor is now the Boron Neutron Capture Therapy (BNCT). The BNCT work dominates the current utilization of the reactor. The weekly schedule allows still three days for other purposes such as isotope production, neutron activation analysis and training. [1]

FiR 1 has a special irradiation facility for BNCT [2]. The neutron moderator block of the BNCT station is used as an epithermal neutron source to treat brain as well as head and neck tumour patients [3,4]. BNCT control and patient preparation rooms were built during the renovation in the 1990's. The FiR 1 core loading was made asymmetric in order to increase the intensity in the BNCT neutron beam without changing the 250 kW maximum steady state power. Further improvement of the treatment quality and capacity by either shortening the treatment time or increasing the therapeutic effect of the BNCT neutron beam would require upgrading of the reactor power used in the treatments. In evaluations performed by VTT doubling of the power from 250 kW to 500 kW has been set as a target.

The current operating license of the reactor expires at the end of the year 2011. An application for a new operating licence has to be submitted to the authorities at least one year earlier, which means by the end of the year 2010.

A long term strategy has been worked out for FiR 1 by VTT and its proprietor the Ministry of Employment and the Economy supported by an independent survey of an outside consultant [5]. According to the survey most actors in state administration, education and research consider the continuation of the development of BNCT nationally important and even ethically necessary. Developments of linear accelerators to substitute reactors as a neutron source for BNCT have started. The prospect is that in the future such a facility will be constructed at the Helsinki University Central Hospital. Meanwhile development of BNCT to an established treatment for several cancers requires FiR 1 as a demonstration and reference facility at least till the end of year 2023.

Based on the strategic needs of the BNCT treatment service and development VTT aims to apply for a license to operate the reactor till the end of year 2026. Before end of that period the needs for the reactor will be re-evaluated. The goal in BNCT treatment development is to be able by that time to treat all the patients with an accelerator based neutron source. In that case the remaining needs of short lived radioisotopes as well as education and training will dictate the future of the reactor.

For the new 15 years operation license period improvements in the reactor will be required and are proposed. It is estimated that the reactor control instrumentation has to be renewed latest half way in the new period. The current instrumentation electronics is from 1982; the control rod drives have been refurbished in 2006 to 2009. It is also proposed that for increased production capacity and improved reliability the rest of the aluminium clad fuel from year 1962 will be replaced with steel clad fuel – if funding will be available for this. Both these plans are under discussion with the licensing review authority STUK.

The importance of reliable operation is emphasised by the needs of the tight, inflexible schedules of the BNCT treatments. Once the boron carrier infusion has been given the irradiation has to start within a half an hour time window three to three and a half hour after start of the infusion. If a treatment has to be postponed the expensive infusion is lost.

2. Renewal of the operating licence

The current operating license expires at the end of the year 2011. An application for a new operating licence has to be submitted to the authorities at least one year earlier, which means by the end of the year 2010. All the documents needed for the application will be prepared or revised during the year 2010. During the following year (2011) the documents will be checked by the authorities and at the end of the said year the new license should be granted by the Government.

A license to operate a nuclear facility is applied for with a written application addressed to the Government. Nuclear energy matters belong to the Ministry of Employment and the Economy, former the Ministry of Trade and Industry. Reactor operation in general is under the supervision of the Radiation and Nuclear Safety Authority (STUK). The nuclear waste management, however, is mainly under the supervision of the Ministry of Employment and the Economy.

2.1 Documents for the new licence

Finland has a new modified nuclear energy act and decree. In practice, however, the documents needed in the operating license application are still nearly the same as ten years ago, when the licence was last time renewed.

The application shall include naturally the basic facts about the nuclear facility. The supplementary documents to the application are described in Table 1. They are mainly rather short and they give only a general description about the nuclear facility in question.

1. **DETAILS OF THE SITE**
2. **THE QUALITY AND MAXIMUM AMOUNTS OF THE NUCLEAR MATERIALS**
3. **AN OUTLINE OF THE TECHNICAL OPERATING PRINCIPLES AND OTHER ARRANGEMENTS WHEREBY SAFETY HAS BEEN ENSURED**
4. **A DESCRIPTION OF THE SAFETY PRINCIPLES THAT HAVE BEEN OBSERVED, AND AN EVALUATION OF THE FULFILMENT OF THE PRINCIPLES**
5. **A DESCRIPTION OF THE MEASURES TO RESTRICT THE BURDEN CAUSED BY THE NUCLEAR FACILITY ON THE ENVIRONMENT**
6. **THE EXPERTICE AVAILABLE TO THE APPLICANT AND THE OPERAING ORGANIZATION**
7. **PLANS FOR ARRANGING NUCLEAR WASTE MANAGEMENT**

Tab 1: Supplementary documents to the application for an operating license.

When applying for an operating license the applicant must also submit more detailed documents to the Radiation and Nuclear Safety Authority (STUK). These documents are listed in Table 2. During earlier application processes the demand for a probabilistic safety analysis was removed from the list by STUK.

1. **THE FINAL SAFETY ANALYSIS REPORT**
2. **A PROBABILISTIC SAFETY ANALYSIS**
3. **A QUALITY ASSURANCE PROGRAMME FOR THE OPERATION OF THE NUCLEAR FACILITY**
4. **TECHNICAL SPECIFICATIONS**
5. **A SUMMARY PROGRAMME FOR PERIODIC INSPECTIONS**
6. **A DESCRIPTION OF THE ARRANGEMENTS FOR PHYSICAL PROTECTION AND EMERGENCIES**
7. **A DESCRIPTION ON HOV TO ARRANGE THE SAFEGUARDS THAT ARE NECESSARY TO PREVENT THE PROLIFERATION OF NUCLEAR WEAPONS**
8. **ADMINISTRATIVE RULES FOR THE NUCLEAR FACILITY**
9. **A PROGRAMME FOR RADIATION MONITORING IN THE ENVIRONMENT**
10. **AGEING MANAGEMENT SYSTEM**

Tab 2: Documents to be submitted to the Radiation and Nuclear Safety Authority.

When the Radiation and Nuclear Safety Authority is going through the updated documents, in some cases it may return documents to the licensee for new revisions. Finally the documents will be satisfactory and the Radiation and Nuclear Safety Authority will grant its acceptance and will send its statement to the Ministry of Employment and the Economy. The Ministry presents then the application to the Government for decision.

3. Upgrading the reactor for continuing reliable and improved operation

Upgrades for continuing and improved service into the 2020's are under consideration. These include planning for renewal of the instrumentation in the midway of the new licensing period. Preparations for a power upgrade during the next licensing period will also be made. Doubling the power to 500 kW will allow increasing the capacity for BNCT treatments by more than a factor of three, meeting the prospects of increased demand for these treatments, while still keeping the current capacity for radioisotope production, education and training.

With increased power the patient irradiations would become shorter bringing increased patient comfort and positioning accuracy, as well as more optimised irradiation timing with rapid kinetics boron carriers. Increased power would on the other hand allow to use an attenuating lithium filter for hardening of the neutron spectrum in the beam for deeper treatment penetration.

3.1 Instrumentation renewal

The reactor power control instrumentation at FiR 1 is from 1982. The frequency of component failures is increasing all the time. The instrumentation is maintained by replacing the failed components in house or by sending whole subunits for maintenance at the manufacturer in Hungary. It is estimated that operation of the reactor beyond 2016 will require a total renewal of the current instrumentation. This would include the nuclear channels and the control electronics.

3.2 Renewal of part of the fuel with possibility for higher operating power

When FiR 1 started operation in 1962 its licensed power was 100 kW. In 1967 the power was increased to 250 kW but already then parts of the required modifications were designed for 1000 kW. In 1996-97 the reactor building and the reactor cooling system were completely renovated. The cooling system was rated for 400 kW with most of the parts capable for even higher power. So no major changes to the cooling system are foreseen. Required changes and improvements in instrumentation are relatively easy to realise. Current control rods will be sufficient for safe and reliable power control of the reactor.

There are several other TRIGA Mark II reactors in the world operating at 500 kW or higher power. A straight forward method for a power upgrade will be replacing those aluminium clad fuel elements which still remain in the reactor core since the start-up of the reactor in 1962 with new steel clad elements. The steel clad elements are used in many countries and can be licensed also in Finland for high power operation of a TRIGA. The need is for 40 to 50 pcs of new fuel elements which would be a major investment requiring special funding.

The other option is to use less new steel clad fuel by using old aluminium clad elements in low power density parts of the core. This option was already preliminarily studied in 2000 and now more detailed reactor physics and thermal hydraulics calculations using the most advanced codes in use at VTT will be performed to evaluate this option.

The power upgrade will require a licensing from the Radiation and Nuclear Safety Authority (STUK). This will be dealt with separately from the operation licence renewal.

4. Benefits of the renewals to the services of the reactor and to the education of new experts in nuclear energy

The current weekly operation schedule with two days allocated for BNCT treatments gives a maximum annual capacity of 92 patient irradiations (the reactor being available 46 weeks / year). Little over half of that is used today. The treatment organization [6] has set the goal to increase the number of patient irradiation within the next five years to over 200 irradiations annually. To meet this demand without a power increase nearly all operation days of the reactor should be allocated to BNCT. The isotope production as well as the education and training activities would suffer. In practise, to keep the isotope production viable, two shift operation would be required during two days a week.

The power increase will allow a new weekly schedule with a capacity for seven patient irradiations per week, three days with two patients and one morning for one. This would mean a yearly capacity of 322 patient irradiations. The capacity for isotope production, two half days, would be the same as today without need for two shift operation. The reactor would be available for education and training half day a week making together 23 days which is equal to the current utilization.

The renewal of the control electronics will ensure high availability and reliable operation of the reactor even in the future high capacity and high utilization situation.

The upgrading activities will create also new general knowledge on reactor physics, nuclear fuels and reactor instrumentation at VTT. Involving the young generation of researchers in these activities gives additional opportunities for their education and thus strengthens the position of VTT as one of the important European nuclear research organizations. The continuing operation of the FiR 1 research reactor is important also in general for the nuclear energy sector in Finland.

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KEEPING RESEARCH REACTORS RELEVANT: A PRO-ACTIVE APPROACH FOR SLOWPOKE-2

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ABSTRACT

The SLOWPOKE is a small, inherently safe, pool-type research reactor that was engineered and marketed by Atomic Energy of Canada Limited (AECL) in the 1970s and 80s. The original reactor, SLOWPOKE-1, was moved from Chalk River to the University of Toronto in 1970 and was operated until upgraded to the SLOWPOKE-2 reactor in 1973. In all, eight reactors in the two versions were produced and five are still in operation today, three having been decommissioned. All of the remaining reactors are designated as SLOWPOKE-2 reactors.

These research reactors are prone to two major issues: aging components and lack of relevance to a younger audience. In order to combat these problems, one SLOWPOKE-2 facility has embraced a strategy that involves modernizing their reactor in order to keep the reactor up to date and relevant.

In 2001, this facility replaced its aging analogue reactor control system with a digital control system. The system was successfully commissioned and has provided a renewed platform for student learning and research. The digital control system provides a better interface and allows flexibility in data storage and retrieval that was never possible with the analogue control system.

This facility has started work on another upgrade to the digital control and instrumentation system that will be installed in 2010. The upgrade includes new computer hardware, updated software and a web-based simulation and training system that will allow licensed operators, students and researchers to use an online simulation tool for training, education and research.

The tool consists of:

- A dynamic simulation for reactor kinetics (e.g., core flux, power, core temperatures, etc). This tool is useful for operator training and student education;
- Dynamic mapping of the reactor and pool container gamma and neutron fluxes as well as the vertical neutron beam tube flux. This research planning tool is used for various researchers who wish to do irradiations (e.g., neutron activation analysis, neutron radiography or in-pool mixed field irradiations); and
- On-line viewing of archived data (temperatures, neutron flux, rod position, etc).

This modernized digital control system, along with new tools for training, education and research will ensure a viable platform for teaching and research while at the same time reduce vulnerability due to an aging control system.

1. Introduction

SLOWPOKE (Safe LOW POwer c(K)ritical Experiment) is a small, inherently safe, pool-type research reactor that was engineered and marketed by Atomic Energy of Canada Limited (AECL) in the 1970s and 1980s. The SLOWPOKE-2 research reactor at the Royal Military College (RMC) in Kingston, Ontario was commissioned in 1985. A neutron beam tube was later installed to allow for neutron radiography (a non-destructive imaging technique). The reactor has been used for teaching and research for the past 25 years.

The reactor core sits inside a sealed reactor container in a pool of regular light water, 2.5 m diameter by 6 m deep, both of which provide cooling via natural convection. The core is an assembly of 198 low-enriched uranium fuel pins, 22 cm diameter and 23 cm high, surrounded by a fixed beryllium annulus and a bottom beryllium slab. Criticality is maintained by adding beryllium plates in a tray on top of the core.

The reactor produces neutrons and gamma rays that are used in many areas of research. The main uses for the research reactor are:

- Neutron activation analysis (e.g., material identification, composition, etc.);
- Radioisotope production (e.g., tracer elements and other radioisotopes);
- Neutron radiography (e.g., non-destructive testing of aircraft control surfaces);
- In-pool mixed field irradiations (e.g., gamma/neutron radiation for advanced material design and testing); and
- Teaching, training, and other research and custom applications.

2. A proactive approach for SLOWPOKE-2

An unplanned, forced, or otherwise inadvertent reactor shutdown or power reduction is a significant event for a nuclear power plant or research reactor. So significant is such an event that nuclear reactor organizations are willing to proactively invest resources to reduce these occurrences to a minimum [1]. One of the ways to reduce these events is to have a robust maintenance regime that will maximize system reliability, availability and maintainability.

At this facility, there is an ongoing effort that has seen the successful replacement of aging reactor components over the last decade and improvements in operational practices. Two recent areas of interest have been the reactor control/instrumentation system and training/simulation.

The next generation upgrade to its digital control and instrumentation system will be installed in 2010. The upgrade includes new computer hardware, updated software and a web-based system that will allow licensed operators, students and researchers to use an online tool for training, education and research.

2.1 Control and instrumentation system upgrade

In 2001, this facility replaced its aging analogue reactor control system with a digital control system called the SLOWPOKE Integrated Reactor Control and Instrumentation System (SIRCIS) [2]. The system was successfully commissioned and has provided a renewed platform for reactor operation, student learning and research work.

As part of the SIRCIS-2010 upgrade, the control rod portion of the reactor head has been redesigned to remove redundant mechanical parts and provide a simplified mount for the control rod motor and optical encoder that provides rod position.

The upgrade to digital control offers many advantages over the outdated analogue system such as:

- Fewer mechanical parts meaning less downtime for maintenance
- Improved features such as
  - Gravity based control rod insertion
  - Core high temperature shutdown
  - More accurate flux control;
- Automated features such as calculate excess reactivity and flux hours;
- A modern dynamic user interface;
- Improved digital data logging (flux, rod position and core temperatures); and
- Fault tolerance and graceful degradation:
  - Redundant remote display;
  - Redundant digital storage;
  - Hot swappable input device;
  - Triple redundant main power supply.

Figure 1 shows the three generations of control system used at the facility.

Figure 1: Analogue control system (1984), SIRCIS digital control (2001), SIRCIS digital control (2010)

2.2 Advanced simulation, research and training tools for SLOWPOKE-2

A new software-based tool has been developed for the SLOWPOKE-2 that consists of:

- Dynamic reactor simulation;
- Dynamic flux mapping; and
- Data analysis

This tool is web based and allows operators, students and researchers access over a secured network.

2.2.1 Dynamic reactor simulator for training

As part of the 2001 upgrade work, a software based SLOWPOKE simulator was created for training and demonstration. The simulator is normally run in a stand-alone mode that closely emulates the real operating environment. There is a hardware component to the simulator (e.g., control rod drive, key switches, etc) that connects to a dynamic simulation of reactor kinetics (e.g., core flux, power, core temperatures, etc). This simulator has been upgraded to integrate the latest changes to the digital control system and will be used for system validation/verification efforts as well as operator training.

In addition to the stand-alone simulator update, a completely software-based simulation has been created and is available to researchers and students over the World Wide Web while using a secure connection to the simulation server. Once the SIRCIS 2010 update is complete, this software will be released for the use of facility personnel.

Figure 2: SIRCIS simulator showing instrumentation tab

Figure 2 shows the instrumentation tab of the web-enabled simulator.

2.2.2 Dynamic flux mapping

When researchers use the SLOWPOKE-2 reactor, they are generally interested in the neutron and gamma ray fluxes. The flux fields in the reactor core, outside the reactor container and at the neutron beam tube image plane, are complex and dependent on many factors such as geometry, materials, shielding, shimming and distance from the core (both radial and axial).

Most irradiations in the core use inner and outer irradiation sites with reasonably well-established neutron fluxes. However, for advanced material research and other projects, the gamma/neutron flux outside the core is not well characterized such that a researcher can easily select the best site for experimentation.

Previous research projects and computer models have characterized fluxes at these other sites; however, until now there has been no single repository of data that could be used to visualize the fluxes with any degree of clarity or certainty.

This portion of the software provides researchers with a real-time computer tool that will allow the prediction of the neutron and gamma fluxes at a specific location within the reactor. This will facilitate and expedite research activities.

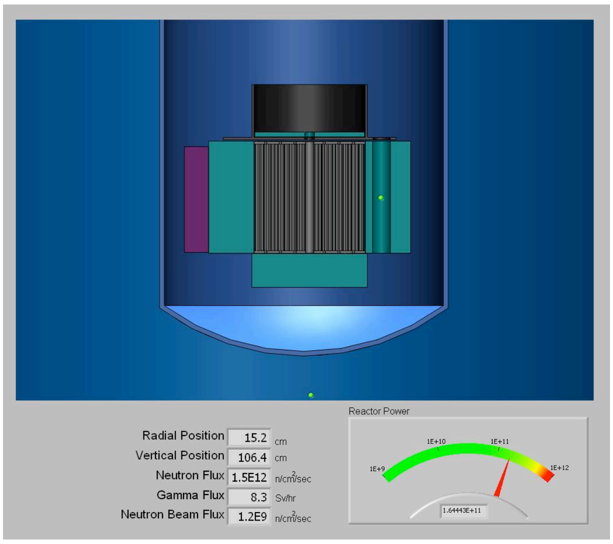

Figure 3: Flux map interface showing core profile

The primary interface for the flux map is shown in **Figure 3** where a profile of the core is shown. The user can move the cursor to any position and predict the flux that is also dependent on the reactor power level. The user can also create and explore a 3D contour model of the fluxes.

2.2.3 Data Analysis

Data from SIRCIS is archived digitally and available for review by facility operators and researchers. This tool allows for the electronic review of:

- Event logs
- Flux, rod position and temperature data
- Excess reactivity logs
- Operating hours
- Error logs

The ability to search and review data from operations and research remotely is a significant advantage to facility users.

3. Conclusion

This modernized digital control system, along with new simulation and training tools, will ensure a viable platform for teaching and research while at the same time reduce vulnerability due to an aging control system.

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RELIABILITY DATABASE OF IEA-R1 BRAZILIAN RESEARCH REACTOR: APPLICATIONS TO THE IMPROVEMENT OF INSTALLATION SAFETY

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ABSTRACT

In this paper the main features of the reliability database being developed at Ipen-Cnen/SP for IEA-R1 reactor are briefly described. Besides that, the process for collection and updating of data regarding operation, failure and maintenance of IEA-R1 reactor components is presented. These activities have been conducted by the reactor personnel under the supervision of specialists in Probabilistic Safety Analysis (PSA). The compilation of data and subsequent calculation are based on the procedures defined during an IAEA Coordinated Research Project which Brazil took part in the period from 2001 to 2004. In addition to component reliability data, the database stores data on accident initiating events and human errors. Furthermore, this work discusses the experience acquired through the development of the reliability database covering aspects like improvements in the reactor records as well as the application of the results to the optimization of operation and maintenance procedures and to the PSA carried out for IEA-R1 reactor.

1. Introduction

The development of a reliability database for the research reactors located at Ipen-Cnen/SP started in 2001 when Brazil took part in an IAEA Co-ordinated Research Project (CRP). The IAEA CRP was entitled “CRP to Upgrade and Expand the IAEA Reliability Database for Research Reactor PSAs” and had participants from eleven Member States: Argentina, Australia, Austria, Brazil, Canada, Czech Republic, India, Indonesia, Republic of Korea, Romania and Vietnam. The main objective of this CRP was to generate a new version of IAEA reliability database to supersede IAEA-TECDOC-930 [1], since it had a broader scope providing guidance on a wider range of issues pertaining to reliability data for the Probabilistic Safety Analysis (PSA) of research reactors. However, this new version, which was concluded in 2005 and sent to the IAEA for a final review, has not been published yet.

In the case of Brazil, a specific reliability database for IEA-R1 reactor continued being updated and improved. IEA-R1 is a 5 MW pool type reactor, cooled and moderated by light water, and it uses graphite and beryllium as reflectors. First criticality was achieved on September 16, 1957 and the reactor has been operating regularly and safely for over 50 years. For this period, it has been intensively used for basic and applied research, training and production of radioisotopes.

In this work the current stage of IEA-R1 reliability database is presented, and the main results regarding the compilation of component failure data, accident initiating events and human errors are discussed. These results are based on the facility records from January 1999 to December 2007, which means a nine-year-observation period.

The main features of the reliability database are described in section 2. A brief explanation of the process for collection and updating of data regarding operation, failure and maintenance of IEA-R1 reactor is presented in section 3. The methods used for the compilation of data are explained in section 4. This work also covers the main applications of the reliability data in the safety analysis of IEA-R1 reactor as well as in the improvement of operating and maintenance procedures of the facility. These aspects will be discussed in section 5 followed by the conclusions in section 6.

2. Main Features of the Reliability Database of IEA-R1 Reactor

The reliability database of IEA-R1 reactor consists of a set of connected Microsoft Excel spreadsheets (input data and output/final data) with necessary information:

- to generate estimates of component failure rates/probabilities of failure on demand and accident initiating events frequencies; and
- to compile human error evidences related to reactor operation and maintenance.

The generation of these data aims to give support to several technical areas of Ipen-Cnen/SP for the development of reliability and safety analyses of the local research reactors or other similar facilities.

The information gathered in this database mainly covers:

Component Technical/Engineering Data. Technical characteristics of IEA-R1 reactor components are stored in the database (type, size, rating, fluid, manufacturer, model, location, etc.). For the definition of component boundaries, the criteria defined by the reactor personnel are followed. Also, the guidance given in IAEA-TECDOC-636 [2] and some definitions in OREDA Handbook [3] should be considered.
Component Operational Data. Records of continuous operating times between consecutive interruptions (either planned shutdowns or not) and the number of demands of the components per reactor operation are stored in the database. In addition, cumulative operating times and number of demands are also computed. In some special cases, the operating time is recorded using one of the installed "timers" and the number of demands is recorded by "counters".
Component Maintenance Data. Every maintenance activity (preventive, corrective or predictive), concerning each reactor component, is recorded in the database. Some details such as: description of the work done, execution time, and so on, are also registered. For IEA-R1, these data may be extracted directly from the maintenance database already developed by the Reactor Maintenance Division personnel.
Component Failure Data. All component failures are reported and verified in order to identify their causes, effects on system / subsystem, actions taken and recovery time. The database stores, whenever possible, the exact times of failure occurrence and detection as well as the time of component restoration to service. Component failure modes are identified and coded according to Table III of IAEA-TECDOC-930 [1].
Data Analysis. Part of the data stored in the database can be processed in order to generate estimates of component reliability parameters. The approach implemented in this database is based on the assumption that failure times are exponentially distributed. It generates an estimate of the constant failure rate (that is the inverse of the “mean time to failure”) associated to each time-related component failure mode. The analysis includes the calculation of a 90% confidence interval estimate (uncertainty limits) for each component failure rate or probability of failure on demand. The MLE (maximum likelihood estimator) for the failure rate / probability of failure on demand is calculated. The Chi-Square distribution is used to derive the confidence bounds for the mean of these parameters. The algorithms to calculate these estimates are implemented in a Microsoft Excel spreadsheet which was developed during the IAEA CRP.

Accident Initiating Events and Human Errors Data. Occurrences identified as accident initiating events precursors and/or human errors are stored in the database in order to be investigated and properly grouped.

3. Process for collection and updating of data related to operation, failure and maintenance of IEA-R1 Reactor

Although IEA-R1 reactor has been operating for over fifty years, it has not been possible to completely restore its operational experience since many records of the past history of this facility had information which was incomplete or dubious. Besides that, some important modifications occurred in IEA-R1 in the past. In 1976 the reactor cooling system was duplicated whereas in 1997 a few changes were accomplished in order to increase its power level from 2 MW to 5 MW and to extend its operating cycle to continuous 64 hours per week. Therefore, it was agreed with the reactor personnel to consider the restoration of the records from the year of 1999.

The process for collection and updating of data regarding IEA-R1 reactor will be described below. These tasks have been conducted by the reactor personnel under the supervision of specialists in reliability engineering and PSA. Data on abnormal / unusual occurrences are extracted from the following sources: logbooks, reactor start-up checklist, reactor shutdown checklist, reactor operation datasheets, corrective maintenance datasheets, etc.

Considering the proposed content of IEA-R1 reliability database, two different forms have been introduced to collect input data: (1) Form-1 used for the identification and analysis of abnormal / unusual occurrences during reactor operation and maintenance; (2) Form-2 used to record operating times and number of demands of the main reactor components.

3.1. Identification and analysis of abnormal/unusual occurrences during IEA-R1 reactor operation and maintenance

To fill in the spreadsheet used for the identification and analysis of abnormal / unusual occurrences, the following procedures and criteria are adopted:

- Data on abnormal / unusual occurrences extracted from logbooks or from other documents are first assessed and then classified according to the event type: component failure, accident initiating event, human error, common cause failure and maintenance activity.
- Occurrences related to inadvertent reactor shutdown (SCRAM) are taken into consideration in order to record the number of demands of the Reactor Protection System.
- Occurrences related to inadvertent reactor shutdown (SCRAM) due to loss of offsite power (LOSP) are recorded so as to estimate the number of demands of the emergency power supply systems.
- Failures detected upon preventive maintenance or in a shutdown state of the reactor may be compiled if there is evidence that these failures will be revealed in the next operational phase of the component.
- Further screening can be done to exclude component failures that have no significant impact on reactor safety, and to include component failures related to safety systems and their support systems.
- Failures in which the direct cause is a human error are not taken into account for component failures rates, but are modelled separately.
- Occurrences classified as potential accident initiating events (or initating events precursors) are reproduced in a separate datasheet to verify their association with the events listed in IAEA Safety Standards Series No. NS-R-4 [4].
- Each item failed is assigned a component type code according to standardized codification set by IAEA [1]. Examples: fission counter - ACF; ionisation chamber - ACI; sensor pool water level - ALR; sensor pressure difference - APD; sensor conductivity - AQC; radiation monitoring alarm unit - ARU; sensor temperature - ATA; diesel generator,

emergency ac - DGA; etc. This task requires special attention to assure that the identified failure corresponds to an item within the component boundaries.

- Component boundaries are defined according to the guidelines provided in IAEA-TECDOC-636 [2]. Whenever the boundaries presented in reactor maintenance reports differ from the boundaries recommended in [2], specific definitions of reactor maintenance personnel are preferably adopted.
- Each failure record is assigned a failure mode code according to standardized codification set by IAEA [1]. Examples: failure to function - F; degraded failure - B; spurious function - K; failure to run - R; open circuit - I; erroneous signal - N; etc.
- All the results should be carefully checked before being incorporated into the database.

3.2. Recording the operating times and number of demands of the main IEA-R1 reactor components

The operating times of IEA-R1 components are mainly derived from the records found in the datasheets generated during each reactor operation. Timers and counters are installed in a few important components, which are: control console, primary and secondary circuits cooling pumps, cooling tower fans, Ventilation and Air Conditioning (VAC) system fans, diesel generator groups and instrument air compressor. As a whole seventeen timers and three counters are installed in IEA-R1 in order to record exact operating hours and number of demands of these main components. In the case of the other components the operating times and number of demands are estimated based on either the values recorded in the reactor operation datasheets or the timers and counters of the main components.

In Figure 1 some examples of tables summarizing the annual totals calculated from the collection of operating times and number of demands of the main IEA-R1 components.

Year	Reactor Operating Time (hours)	Reactor Operating Time (hours)	Number of Reactor Operations	Reactor Operations - High Power	Reactor Operations - Low Power	Number of SCRAMs	Loss of Off-Site Power (LOSP)
1999	2445:57:30	2445,96	65	49	16	26	9
2000	2705:06:00	2705,10	69	57	12	21	17
2001	1923:42:00	1923,70	55	49	6	23	7
2002	1767:15:30	1767,26	73	50	23	10	8
2003	1606:11:00	1606,18	54	44	10	35	4
2004	2491:48:00	2491,80	77	46	31	17	8
2005	2873:56:00	2873,93	57	49	8	22	14
2006	2735:07:00	2735,12	70	50	20	38	19
2007	1440:27:00	1440,45	42	30	12	20	3
Total	**19989:30:00**	**19989,50**	**562**	**424**	**138**	**212**	**89**

Mean Values			
Operating hours/ year	Operations/ year	SCRAM/ year	LOSP/ year
2.445,96	65,00	26,00	9,00
2.575,53	67,00	23,50	13,00
2.358,25	63,00	23,33	11,00
2.210,50	65,50	20,00	10,25
2.089,64	63,20	23,00	9,00
2.156,67	65,50	22,00	8,83
2.259,13	64,29	22,00	9,57
2.318,63	65,00	24,00	10,75
2.221,06	62,44	23,56	9,89

Year	Control Console			Primary Circuit Pump B101-A (CP-BOM-01)			Primary Circuit Pump B101-B (CP-BOM-02)			Heat Exchanger A		Heat Exchanger B	
	No. of Demands	Operation Data Sheet (hours)	Timer (hours)	No. of Demands	Operation Data Sheet (hours)	Timer (hours)	No. of Demands	Operation Data Sheet (hours)	Timer (hours)	No. of Demands	Operating Time (hours)	No. of Demands	Operating Time (hours)
1999	71	2544:03:00	2544,05	27	1101:49:00	1101,82	34	1327:00:00	1327,00	17	665,65	33	1739,79
2000	83	2834:37:00	2834,62	36	1284:35:00	1284,58	39	1477:03:00	1477,05	27	1219,55	29	1540,23
2001	71	2004:26:00	2004,43	59	1540:08:00	1540,13	16	378:15:00	378,25	21	834,65	28	1071,38
2002/b.t.	16	407:58:00	407,97	16	383:20:00	383,33	0	0:00:00	0,00	5	183,05	6	200,28
2002/t.	52	1419:32:00	1595,50	15	462:04:00	466,48	26	885:41:00	894,44	9	406,24	29	950,11
2003	58	1646:32:30	1930,58	8	282:17:00	286,65	38	1161:40:00	1275,69	9	557,94	35	1558,04
2004	94	2602:52:00	2725,50	58	1722:03:00	1746,71	23	711:23:00	727,80	8	485,49	39	2139,27
2005	63	2962:45:00	3058,67	19	1109:08:00	1057,34	114	1581:34:00	1872,78	6	356,05	43	2545,12
2006	85	2814:52:00	2939,04	42	1267:46:00	1314,91	83	1355:51:00	1307,99	0	0,00	49	2678,40
2007	45	1481:00:00	1567,49	19	272:44:00	282,69	44	1149:48:00	1177,41	2	0,15	33	1451,63
Total	**637**	**20718:37:30**	**21607,85**	**299**	**9425:54:00**	**9464,65**	**417**	**10028:15:00**	**10438,41**	**104**	**4708,77**	**324**	**15874,26**

Figure 1 - Operating times and Number of Demands of IEA-R1 Reactor Components

4. Compilation of collected data to estimate component reliability parameters and other failure event frequencies

The compilation of collected data is developed in a spreadsheet that is a calculation report to generate final data concerning component failure rates / probabilities of failure on demand. This calculation report (or intermediate spreadsheet) uses both data on abnormal / unusual occurrences mentioned in subsection 3.1 and data on operating times / number of demands described in subsection 3.2.

In this calculation report the components are codified according to their types based on the three letter coding system standardized in IAEA-TECDOC-930 [1]. Besides that, this calculation report contains detailed technical descriptions of the analyzed component types and can be cross-referenced with the reliability parameters data table.

The data collected and reported in this study cover major components of the following IEA-R1 reactor systems: Reactor Core; Reactor Cooling System – Primary and Secondary Circuits; Instrumentation and Control System; Electrical Power Supply System; Ventilation and Air Conditioning System; Instrument Air System; etc.

Considering the observation period from January 1999 to December 2007, 557 failures of 108 different component types were compiled. The total operating time of IEA-R1 reactor during that period was 19989,5 hours. Mean values of component failure rates / probabilities of failure on demand and respective confidence intervals are calculated using the algorithms developed during the IAEA CRP and are compiled in a spreadsheet which format is presented in Table 1. Examples of the technical descriptions of IEA-R1 components are shown in Table 2.

Data stored in the IEA-R1 database can also be used to estimate the frequencies of accident initiating events and to assess occurrences related to human errors during the operational and maintenance procedures. During the nine-year-observation period from 1999 to 2007, over 350 events were identified as initiating events precursors. Among these events, 82 occurrences might have evolved to the initiating event "fuel cooling channel blockage". Based on these evidences, the estimated initiating event frequency is $4{,}1 \times 10^{-3}$/hr, since the total operating time of the reactor during the observation period is 19989,5 hours. In addition, 38 human errors were identified and grouped according to event types: failure to follow procedures or maintenance error (26); error of commission (9); and design error (3). Among these 38 events related to human errors, at least 25 could also be classified as precursors of accident initiating events. The scope of this database does not include quantitative derivation of human error data.

5. Application of the results to safety aspects of IEA-R1 Reactor

It is important to mention that enhancement of accuracy and quality of reliability data has already been observed along these years at IEA-R1 reactor. In this sense, some factors presented a major contribution: more detailed descriptions of the abnormal/unusual occurrences or failures; more precise data on component failure times or failure detection times; records of exact operating times and number of demands obtained from timers and counters installed in the main reactor components; and more detailed descriptions of corrective maintenance activities including information on their durations. Besides that, there were changes in format and content of the records in logbooks and other operation datasheets aiming to include the information necessary to compose the reliability database.

Furthermore, component failure data resulting from this work was applied in the Probabilistic Safety Analysis (PSA) of IEA-R1 reactor. A master dissertation covering a partial PSA of IEA-R1 reactor was developed at Ipen-Cnen/SP and concluded in 2009 [5]. This work addressed the detailed analysis of the two main accident initiating events identified in the Safety Analysis Report of this facility, which are flow channel blockage and loss of coolant due to rupture of primary circuit boundary.

Code	Component type description	Reactor Code	Component Population #	Cummulative calendar time hs	Cummulative operating time hs	Demands #	Failure mode	Failures #	Failure rate 1/h	Failure probability 1/demand	90% Confidence bounds 5%	95%
ACF01	Fission Counter	BR01	2		2,16E+04		F	12	5,55E-04	-	3,20E-04	9,00E-04
ACF01	Fission Counter	BR01	2		2,16E+04		B	7	3,24E-04	-	1,52E-04	6,08E-04
ACF01	Fission Counter	BR01	2		2,16E+04		I	1	4,63E-05	-	2,37E-06	2,20E-04
ACI01	Ionisation Chamber - compensated	BR01	3		2,16E+04		F	3	1,39E-04	-	3,78E-05	3,59E-04
ACI01	Ionisation Chamber - compensated	BR01	3		2,16E+04		B	1	4,63E-05	-	2,37E-06	2,20E-04
ACI01	Ionisation Chamber - compensated	BR01	3		2,16E+04		N	1	4,63E-05	-	2,37E-06	2,20E-04
ACI02	Ionisation Chamber - non compensated	BR01	5		6,00E+04		F	2	3,34E-05	-	5,93E-06	1,05E-04
ALR01	Sensor - pool water level - float type	BR01	1	7,88E+04			K	4	5,07E-05	-	1,73E-05	1,16E-04
ALR02	Sensor - pool water level - ultrasonic type	BR01	3	2,37E+05			F	1	4,23E-06	-	2,17E-07	2,01E-05
ALR02	Sensor - pool water level - ultrasonic type	BR01	3	2,37E+05			N	3	1,27E-05	-	3,46E-06	3,28E-05
APD01	Sensor - pressure difference	BR01	4	3,15E+05			F	1	3,17E-06	-	1,63E-07	1,50E-05
AQC01	Sensor - conductivity (water re-treatment)	BR01	2	1,58E+05			N	1	6,34E-06	-	3,25E-07	3,01E-05
AQC02	Sensor - conductivity (water treatment)	BR01	4		1,58E+05		B	1	6,33E-06	-	3,25E-07	3,00E-05

Table 1 - Example of the IEA-R1 Component Reliability Database

Component Code		Description
ACF 01	Component Type / Name:	**Fission chamber**
	System:	Instrumentation and Control System
	Component "Tag":	IC-CAF-01; IC-CAF-02; IC-CAF-03; IC-CAF-04
	Population:	4
	Location:	reactor core
	Manufacturer:	IPEN / IST Imaging & Sensity Tec
	Component characteristics:	model WL-6376A; 93% enriched U
	Operating duty:	1 fission chamber functioning (reactor core), 3 standby
	Component boundary:	sensor and cabling (excluding components installed in the control console)
	Linked components:	ICC02 (safety channels)
ACI 01	Component Type / Name:	**Ionisation chamber - compensated (linear channel)**
	System:	Instrumentation and Control System
	Component "Tag":	IC-CIC-01; IC-CIC-02; IC-CIC-03
	Population:	3
	Location:	reactor core
	Manufacturer:	Westinghouse / IST Imaging & Sensity Tec
	Component characteristics:	WL-7741 (CIC-01); WL-23084 (CIC-02/03)
	Operating duty:	1 ionisation chamber functioning (reactor core), 2 standby
	Component boundary:	sensor and cabling (excluding components installed in the control console)
	Linked components:	ICC01 (linear channel)
ACI02	Component Type / Name:	**Ionisation chamber - non compensated (safety channel)**
	System:	Instrumentation and Control System
	Component "Tag":	IC-CNC-01; IC-CNC-02; IC-CNC-03; IC-CNC-04; IC-CNC-05
	Population:	5
	Location:	reactor core (CNC-01/02/04/05); basement (CNC-03)
	Manufacturer:	Westinghouse (CNC-01/02/04); Ipen (CNC-03); IST Imaging & Sensity Tec (CNC-05)
	Component characteristics:	model WL-6937 (CNC-01/02/04/05); HOM-CARREI-N16 (CNC-03)
	Operating duty:	3 ionisation chamber functioning, 2 standby
	Component boundary:	sensor and cabling (excluding components installed in the control console)
	Linked components:	ICC02 (safety channels); N-16 channel ; EPH01 (high power voltage)
ALR 01	Component Type / Name:	**Sensor - pool water level - float type**
	System:	Instrumentation and Control System
	Component "Tag":	IC-MDS-01
	Population:	1
	Location:	reactor pool
	Manufacturer:	NIVETEC
	Component characteristics:	NPR-EXD; series 600-010-304
	Operating duty:	continuous functioning
	Component boundary:	sensor, transmitter and local power supply
	Linked components:	control room indicating instrument; URS01 (Scram circuit)
ALR 02	Component Type / Name:	**Sensor - pool water level - ultrasonic type**
	System:	Emergency Core Cooling System
	Component "Tag":	RE-MDS-01; RE-MDS-02; RE-MDS-03
	Population:	3
	Location:	reactor pool; retention tank
	Manufacturer:	CONTROL LEVEL / INCONTROL
	Component characteristics:	model ES-9063A4048AZ
	Operating duty:	2 sensors in continuous functioning; 1 stand-by
	Component boundary:	sensor, transmitter and local power supply - excluding indicators RE-INN-01 (emergency room), RE-INN-02 (control room) and RE-INN-03 (laboratory) installed in the control room
	Linked components:	UIE02 (indicating instrument); URS01 (Scram circuit)

Table 2 - Example of the IEA-R1 Component Technical Description Database

6. Conclusions

The reliability database being developed at Ipen-Cnen/SP has brought some benefits to IEA-R1 reactor besides arising the interest of the reactor management staff in subjects related to safety assessment and reliability analysis.

Prior to this database project, the operational and maintenance records found in the facility had not contained the necessary information to estimate failure rates or other reliability / availability parameters of the reactor components. Then, some effort was required to introduce new forms to be fulfilled by the reactor staff so as to serve to the purpose of generating plant specific reliability data. In fact, the involvement and commitment of supervisors and senior operators for the past nine years have been essential to obtain more reliable records to update the database. Actually, the research work being carried out at Ipen-Cnen/SP has shown that it is fundamental to promote an integration of data collection activities with the existing organizational features and administrative routines of the facility in order to minimize the efforts and to provide good quality data.

It is important to mention that the safety aspects concerning the observed component failures or human errors have been discussed more often in the periodical meetings held in the facility. In these debates, it has been possible to notice that few events could compromise the reactor safety. Even so, the results obtained may contribute significantly to the improvement of operational procedures as well as to the optimization of the maintenance programme of the reactor.

In this work the following important aspects were not addressed although they are being considered by the specialists involved in the reliability database:

- promotion of a practice in the facility to proceed a root cause analysis to investigate failures or to assess the effectiveness of the corrective actions. Some corrective maintenance records indicate that the symptoms of the failures rather than the causes have been addressed; and
- elaboration of a software to manage the database. In this case, a relational database is being planned by a specialist who will be in charge of implementing its future updates and making it available to local computer network users.

Finally, the database developed for IEA-R1 has to be specifically used in the applications of this reactor. However, some data may be applied to other facilities if the components are analogous or the operational conditions are alike. In this way, it is expected that all the experience acquired with this research work can be directed toward the project of a new multipurpose reactor being carried out in Brazil.

7. References

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OPERATION AND MAINTENANCE RELATED ISSUES DURING 47 YEARS OF OPERATION

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ABSTRACT

A survey of operational related problems with the 17 years old computerized reactor instrumentation for the TRIGA reactor Vienna and a failure database, based on 47 years of operation, is given.

The TRIGA reactor Vienna became critical on March 7, 1962. At that time, the reactor operated with the original General Atomics (GA) electronic tube-type console. In 1968 this console was replaced by a transistorized instrumentation which worked until 1992. In 1990 it was decided to replace this aged instrumentation by a state of the art console. The new computerized reactor instrumentation was ordered in 1990 from GA and installed and tested during summer 1992. The reactor was first critical after a two months shut-down period on November 10, 1992. During the last 17 years of operation, a number of failures occurred which made the reactor inoperable for about 100 days in total. The instrumentation design originates from the mid eighties and most of the electronic equipment especially the console computer and the Data Acquisition Chamber (DAC) computer was already outdated at the time of installation. Due to the rapid development in data acquisition technology, the problem of spare part availability becomes imminent, and led to the decision to replace the existing instrumentation after 17 years. All relevant documents especially all reactor logbooks have been securely stored since 1962, as they contain valuable data not only about normal operation but also about abnormal occurrences such as automatic reactor shut downs, failure of equipment and maintenance procedures. All these abnormal occurrences have been classified according to a classification proposed at an IAEA Research Reactor ageing meeting in 2008.

1. Introduction

The TRIGA Mark II research reactor at the Atominstitut, Vienna – Austria operates since March 1962. Up to now there are nearly 50 years of successful operation. It is obvious that over this long period several events concerning different systems of the reactor occurred. These events are recorded in the reactor logbooks. During a vacation job [1], the logbooks were digitized in a “MS access database”, and according to IAEA standards, the events were categorized in 9 groups:

- Reactor block, fuel and internals
- Cooling systems
- Confinement/containment
- Instrumentation and controls
- Electrical distribution
- Auxiliaries
- Experimental facilities
- Documentation and configuration management
- Other (non-SSC)

An additional category “Service” was added by the “Atominstiut”.

As result a list with ~ 4500 entries was created. For obvious reasons, such a list is not easy to evaluate. Therefore the purpose of this report is to prepare a clear overview over the different events – categories and their shares of total events using more intuitive charts. Further a look at the chronologic distribution of most significant events is prepared. Practically most of the information can be obtained from the charts. The text will only point out major facts and give some examples of most common events.

2. Main categories – overview

In total there occurred 4483 events in the last 47 years. Fig 1 shows the percentage shares of the different main categories of the total events. It is easy to see, that about half of the events occurred in the Instrumentation and Controls group. With distance the second biggest category are the Service events with 18% followed by Electrical distribution (10%) and “Reactor block, fuel and internals” with 6%. The sum of the remaining categories is about a third of the total events.

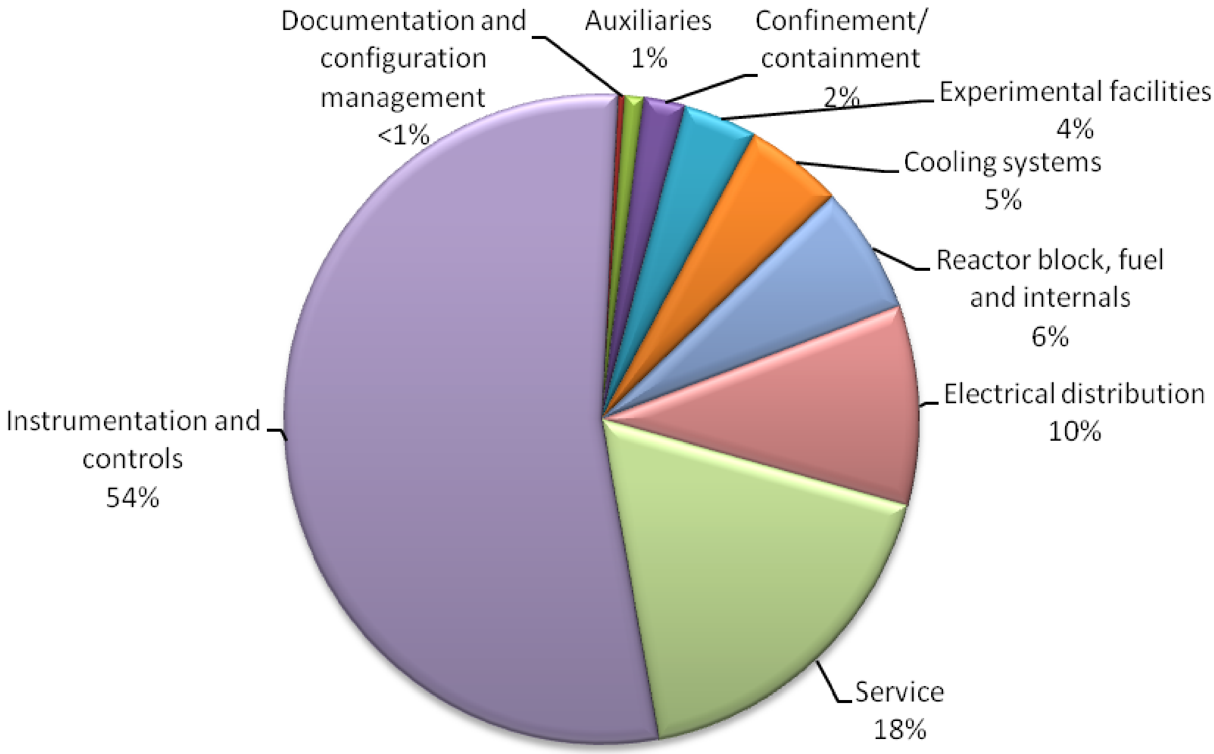

Fig 1. Main categories - percentages

A first impression of the chronological distribution can be taken from Fig 2. The light green bars show the total number of events that happened in the respective year. Beside it there are several bars, describing the shares of the different categories. Noticeable is that most of the peaks appear in connection with a significant raise of the events in a specific category. For example in the years 1981 and 1982 around 175 problems happened with the electrical distribution where normally not more than 5 events per year could be found.

In this paper we will focus on two main groups “Instrumentation and Control” with 54% and the “Service” group with 18%. More details to the other groups can be found in the presentation.

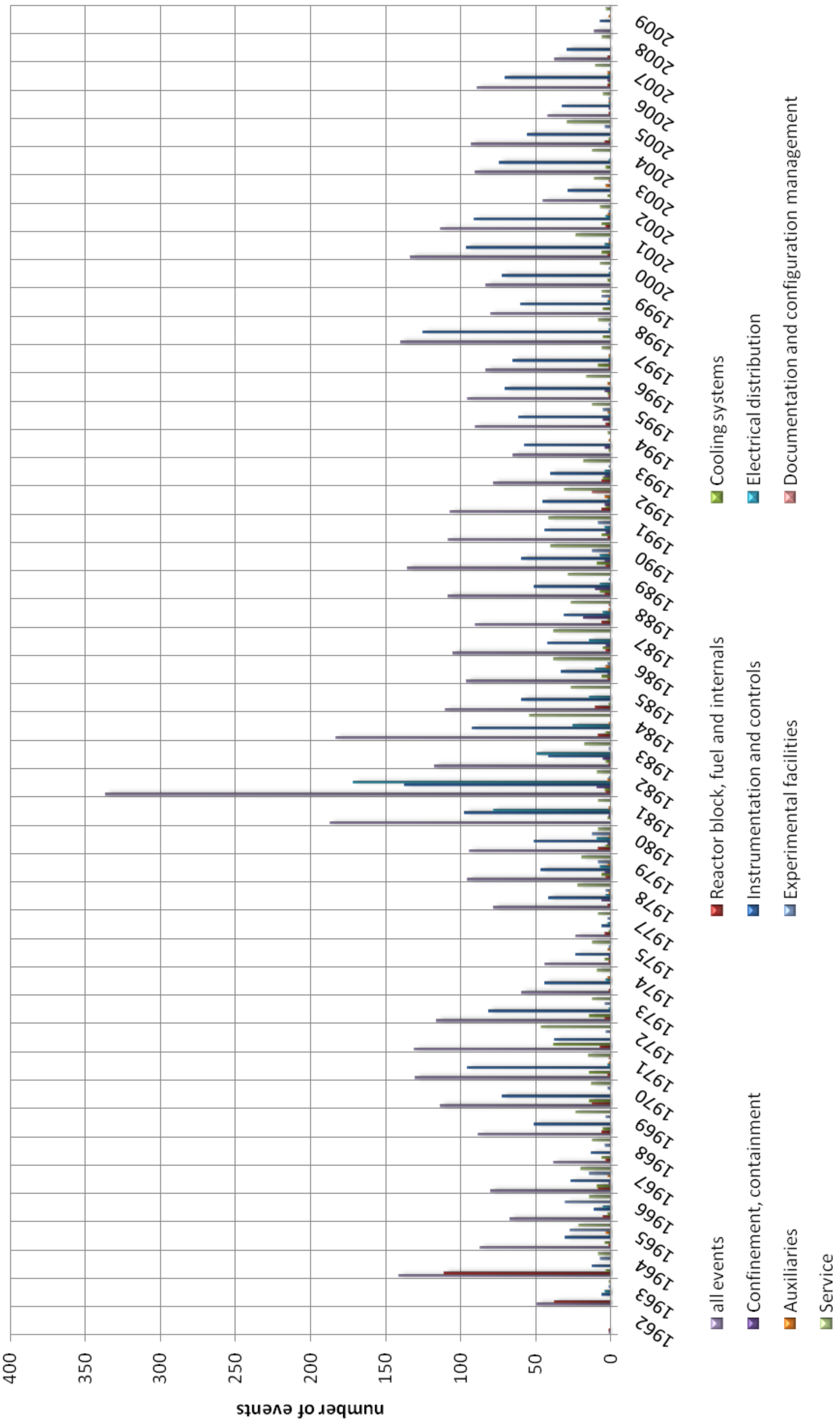

Fig 2. Main categories chronology

3. Instrumentation and control

With a total of 2408 events, the Instrumentation and Controls group is the largest share (54%) of total events. The most important subcategory is Reactor protection with a share of 61% of instrumentation and controls events, where different scrams (fuel-, pool temperature, reactor power monitoring) were the most common events. In the subsections of this chapter, a more detailed investigation of the chronology of reactor protection events, corresponding to the 3 instrumentation periods, is carried out. Radiation monitoring with a share of 16% includes hard- and software problems as well as maintenance tasks of the radiation alarm system. The instrumentation sub category (14%) contains events related to rebuilding and calibrating the different measurement channels.

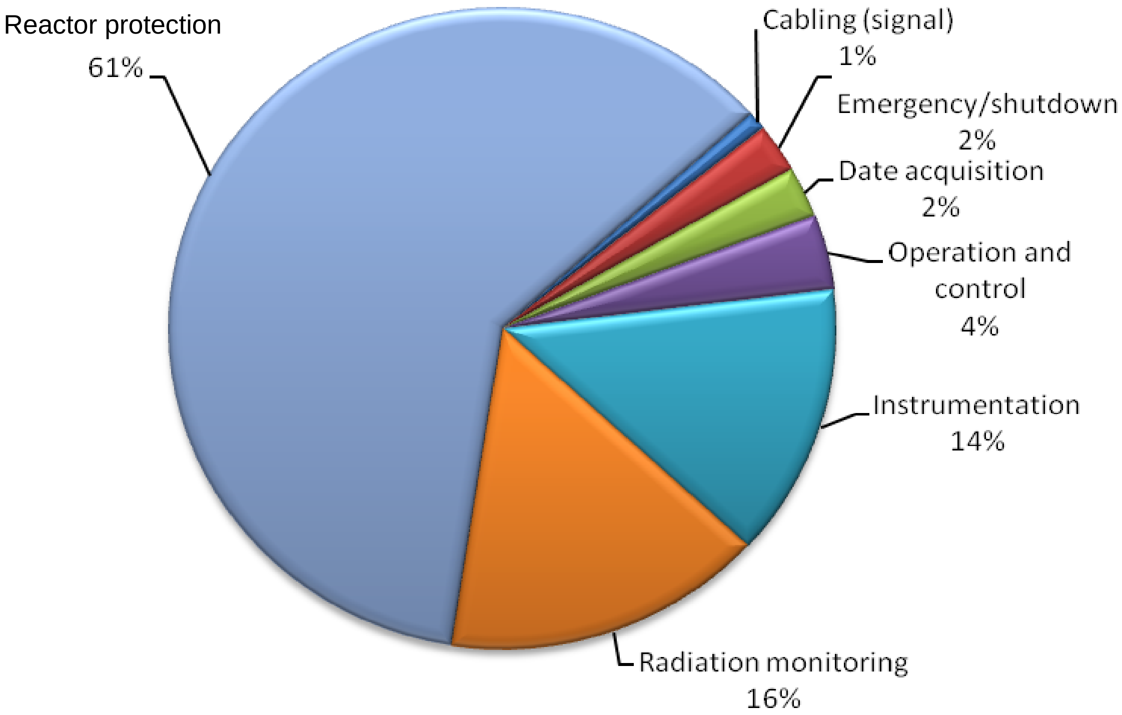

Fig 3. Instrumentation and controls – sub categories

Several reactor protection peaks occurred in 1980 and 1982. Further some smaller peaks in the years 1970/1971, 1980/1981 and 1996 are identified, where unusual many problems with the radiation monitoring facility came up.

Compared to the 2nd and 3rd instrumentation period the 1st period (March 3, 1962 – May 30, 1968) was relatively short. With a total of 14 reactor protection events over 6 years, a mean value of 2 with a standard deviation at the same dimension is obtained. Unnecessary to say, nothing can be concluded from this information. The longest period was the 2nd one which covers the period from June 26, 1968 to July 1, 1992. Here we have as an average 33 reactor protection events per year with a relatively high standard deviation of 29 events because of the unevenly chronological order. The 3rd period, from August 6, 1992 to September 31, 2009 with total 658 reactor protection events has a mean value of 36 events per year with a standard deviation of 20 events and shows similar results to the 2nd instrumentation period.

All these problems, which made the reactor inoperable for about 100 days in the 3rd period, could be solved by the efforts of the operator. Since the beginning of the last year more problems with the NM-1000 and the DAC occurred. These problems have the origin in the ageing of electronic components. Therefore, it was decided to replace the existing instrumentation after 17 years. The new instrumentation should be state of the art and spare parts should be available for at least 10 years.

4. Service

With a total of 800 events the Service group is the second biggest. The relative shares of subcategories can be taken from Fig 4. The biggest part with 39% is taken by “Distilled water auxiliary supply” with common events like cleaning of the distillery and regeneration of the ion exchanger. Nearly all problems with the osmosis and distil facility occurred between the years 1984 and 1993. 13% are maintenance works at the ion exchanger and primary filter like changing the ion exchanger or replacing the filters in the primary and secondary cooling circuit. Typical service company works were crane audits or various works at the reactor hall or service and exchange of pumps. Problems with experimental facilities (11%) were rebuilding of the beam tubes or maintenance of other experimental facilities. Another common event was the exchange of filters of the ventilation system or various bulb exchanges (control-, tank-, hall bulbs).

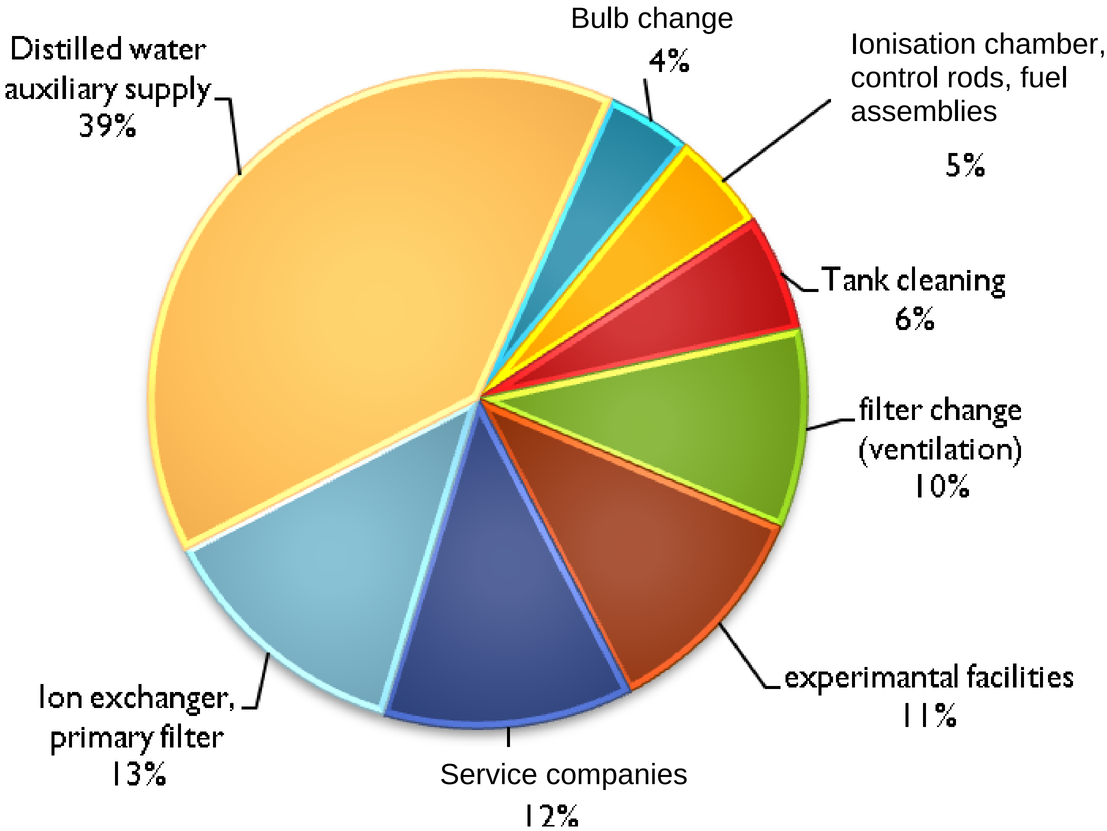

Fig 4: Instrumentation and controls – sub categories

From the chronological point of view it can be said that at the beginning of operation some problems with the ion exchanger and primary filter occurred, later, as mentioned above, there was a decade with an unusual high amount of events concerning “Distilled water auxiliary supply”. Starting with 1969 the tank was cleaned every two to three years.

5. References

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AGEING MANAGEMENT FOR RESEARCH REACTORS

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ABSTRACT

During the past several years, ageing of research reactor facilities continues to be an important safety issue. Despite the efforts exerted by operating organizations and regulatory authorities worldwide to address this issue, the need for an improved strategy as well as the need for establishing and implementing a systematic approach to ageing management at research reactors was identified. This paper discusses, on the basis of the IAEA Safety Standards, the effect of ageing on the safety of research reactors and presents a proactive strategy for ageing management. A systematic approach for ageing management is developed and presented together with its key elements, along with practical examples for their application.

1. Introduction

Ageing is defined as a general process in which characteristics of systems, structures and components (SSCs) gradually undergo degradation with time and use. Ageing management is the engineering, operation, and maintenance activities and practices aimed at monitoring, preventing, and timely detecting and mitigating this degradation. Research reactors experience two kinds of ageing effects: Physical ageing (i.e. gradual deterioration in their physical characteristics); and Obsolescence (i.e. becoming out-of-date in comparison with current knowledge, standards, and technology).

According to the Research Reactors Database [1], about 70 % of existing operating research reactors have been in operation for more than 30 years, with many of them exceeding their original design life. The majority of these reactors are challenged by the negative impacts of SSCs ageing. In this regard, it is worth mentioning that ageing of the SSCs continues to be one of the primary causes of research reactor incidents [2]. Over the years, research reactor operating organizations and regulatory authorities have acquired significant experience in dealing with ageing related problems. However, feedback from IAEA safety review missions showed the need for implementation of a systematic approach to research reactor ageing management [3].

In addressing this need, and in supporting application of the IAEA Code of Conduct on the Safety of Research Reactors [4], a Safety Guide on ageing management of research reactors was developed [5]. It provides guidance on establishing a proactive strategy to ageing management and develops a systematic approach to research reactor ageing management, as well as, on managing of obsolescence, as presented in the following sections.

2. Ageing and safety of research reactors

Ageing degradation may result in a reduction or loss of ability of SSCs to function within their acceptance criteria. Physical ageing reduces safety margins provided in the design of SSCs. If reductions in these margins are not timely detected and adequate mitigation actions are not taken, research reactor safety could be compromised.

Defence in depth is achieved by multiple physical or functional barriers. Ageing degradation may affect one or more of these barriers (e.g. due to corrosion of the fuel cladding material, reactor pool liner, or piping of the primary cooling system). Physical ageing may also increase the possibility of common cause failures (i.e. simultaneous degradation of physical barriers and redundant components) which could result in the impairment of one or more levels of protection provided in the frame of the defence in depth principle. An effective ageing management programme should provide for maintaining the defence in depth through incorporation of good design and engineering features which ensure adequate safety margins. This includes use of design, technology and materials of high quality and reliability, and performance of different operating activities in accordance with approved operating procedures.

Service conditions are major contributors to ageing of SSCs, through chemical and physical processes that affect material properties or functional capabilities. The service conditions are associated with normal operation conditions (e.g. stress/strain, temperature, pressure, and chemistry regime), anticipated operational occurrences (e.g. power excursion and power-flow mismatch) and environmental conditions (e.g. high humidity or presence and use of chemically active liquids and gases). These conditions lead to degradation of SSCs through one or more of the following ageing effects and mechanisms:

- A change in physical properties (e.g. swelling, chemical decomposition, and changes in material strength, ductility, resistivity);
- Irradiation and thermal embrittlement;
- Fatigue, including thermal fatigue;
- Corrosion, including galvanic corrosion, corrosion erosion and corrosion assisted cracking;
- Wear (e.g. fretting) and wear assisted cracking (e.g. fretting fatigue).

Limiting values for service should be included in the operational limits and conditions (OLCs) [6]. Examples include limiting conditions on maximum temperature of the fuel cladding, high (or low) pressure in cooling lines and across filters, high vibration levels of primary cooling pumps, and limits on water coolant chemistry parameters such as conductivity and pH.

3. Proactive strategy for ageing management

Ageing management of SSCs should be implemented proactively (i.e. with foresight and anticipation) throughout different stages of research reactor lifetime, including design, fabrication, construction, commissioning, operation, and decommissioning.

The design and any modifications of a research reactor or associated experimental devices should facilitate inspection and testing aimed at detecting ageing mechanisms and their degrading effects on SSCs, while maintaining the principle that radiation exposure of inspection personnel should be kept as low as reasonably achievable. In the design of a research reactor, consideration should be given to the use of advanced materials (and their production processes) with greater ageing resistant properties (e.g. materials of high resistance to corrosion, or high strength). Considerations should also be given to the maintenance requirements and need for material testing programmes including surveillance specimens dedicated to monitoring of ageing degradation, and to the use of compatible materials, especially those used for welding.

The service conditions and information on possible ageing mechanisms should be properly taken into consideration during the design, fabrication and construction of the SSCs. Baseline data, including manufacturing and inspection records of SSCs as well as records on their shipment and storage conditions should be collected and documented. During the construction process, surveillance specimens for specific ageing monitoring programme should be installed in accordance with the design specifications. The commissioning process should also be used to strengthen ageing management programme. Commissioning tests and verifications should cover identification of hot spots in terms of temperature and dose rate, measurements of vibration levels of rotating machines such as pumps and fans, and characterization of thermal insulation or electrical isolation. All parameters which can influence ageing degradation of the reactor SSCs should be tracked throughout the reactor lifetime.

For implementation of an effective ageing management programme during the research reactor operation phase, the following factors should be taken into consideration, in a proactive manner:

- Minimization of human errors that may lead to premature degradation, through continuing training, and enhancement of the safety culture and sense of ownership;
- Optimal operation of the SSCs to slow down the rate of ageing degradation;
- Proper implementation of maintenance and periodic testing activities in accordance with the OLCs, design requirements and manufacturer's recommendations;
- Follow-up of possible degradation trends in SSCs between successive periodic testing;
- Use of adequate and qualified methods of non-destructive testing and ageing monitoring for early detection of flaws possibly resulting from intensive use of equipment;
- Appropriate storage of spare parts and consumables susceptible to ageing, to minimize degradation while in storage and to control their shelf life properly;
- Feedback of operating experience (both reactor specific operating experience and generic, including operating experience from similar research reactors and industrial plants) to learn from relevant ageing-related events.

It is also important to identify and account for possible changes in operational conditions (e.g. radiation levels, coolant flow distribution and velocity, and vibration level) that could cause accelerated or premature ageing and failure of some SSCs, in the event of upgrading of reactor power, installation of new experimental device or changes in the utilization programme, and replacement of SSCs.

4. A systematic approach for ageing management

A systematic approach to ageing management for research reactors should include the following main elements:

- Screening of SSCs for ageing management review;
- Minimization of expected ageing degradation;
- Monitoring, detecting, and trending of ageing degradation;
- Mitigation of ageing degradation;
- Continuous improvement of the ageing management programme.

A methodology based on importance to safety should be applied for screening of SSCs for ageing management. For efficiency, similar components (e.g. pumps, valves, small diameter piping, and electrical cables) that operate in comparable service conditions (e.g. pressure, temperature, and water chemistry) could be grouped. The list of SSCs important to safety should have been identified in the design stage. This list should be reviewed for updating/completeness (or developed if it was not performed in the design stage). For each of these SSCs, the screening process should be performed to identify the elements that in case of a failure could lead directly or indirectly to the loss of a safety function. This list should be reviewed to identify specific elements for which ageing degradation has the potential to cause SSCs failure. Justification should be provided for excluded elements.

In order to minimize the effects of ageing degradation, preventive actions should be applied. These actions should be determined in the design stage and applied through the reactor lifetime. They include establishment of appropriate operating and service conditions, routine maintenance programme that includes periodic replacement of components and consumables, and when necessary, change of SSCs design and materials. The preventive actions should be continuously improved with account taken of relevant operating experience.

Reactor operating parameters that can be predictive of ageing degradation should be routinely monitored. These parameters include, for example, control rod drop time, water chemistry parameters, temperature, flow rate and pressure. These parameters should be continuously monitored, either online or periodically (in this case, the frequency of measurement should be defined in the OLCs). Records of these measurements should be maintained, assessed and trended in order to predict the onset of ageing degradation in a timely manner.

Routine operating activities should be used to detect ageing degradation. Examples of these activities include observation of the conditions of SSCs (e.g. leaks, noise, and high vibration) during routine walkdowns to the reactor facility. Sampling of water coolant for chemical or radiochemical analysis is another important routine activity that could provide for detection of ageing degradation of some SSCs such as fuel cladding. Non-destructive testing is an efficient way to detect ageing degradation through application of visual, surface, and volumetric examinations. These examinations allow for detection of scratches, wear, cracks, corrosion, or erosion of surfaces, as well as near-to-surface and deep flaws or discontinuities. Ageing degradation can also be detected by checking the performance of SSCs. These checks should be part of the routine maintenance and inspection activities. The results of periodic testing performed to verify compliance with OLCs should be evaluated to detect and correct the operating and environmental conditions before they give rise to significant consequences for safety.

Actions for mitigating ageing degradation include modifications of operating conditions and practices that may affect the rate of ageing degradation. The most common actions for mitigation of ageing degradation are replacement, refurbishment, and modification of SSCs. Replacement of SSCs is performed to achieve the original design intents or service objectives. Implementation of refurbishment and modification projects should be subjected to justification, project management, and procedures for design, construction and commissioning equivalent to those applied to the reactor itself [7]. The projects with safety significance require safety analysis and authorization from regulatory authorities. References [8 and 9] provide the specific experience acquired by many research reactor operating organizations and regulatory authorities from planning, management and implementation of refurbishment and modification projects.

The effectiveness of the ageing management programme should be periodically reviewed for continuous improvements in the light of current knowledge and experience acquired from the facility and other similar facilities. In addition, evaluation of cumulative effects of ageing on safety of research reactors should be treated as ongoing process and should be assessed in periodic safety reviews, which are used to determine whether the research reactor or individual SSCs can be operated safely for a specific period of future operation. Periodic safety reviews are also used to provide inputs for improvement of the scope, frequency, and procedures for maintenance, surveillance, and inspection, for updating of the reactor safety analysis, and for modifications of operating conditions or design.

In order to support implementation of the approach discussed above, records should be kept to provide for identification and evaluation of failures caused by ageing effects, prediction of future performance of SSCs, and decisions on the type and timing of preventive maintenance actions as well as for identification of new emerging ageing effects before they jeopardize the safety, reliability, and service life of the reactor. These records should include baseline information consisting of data on the design and conditions at the beginning of life service, reactor operating records covering service conditions, availability testing and failure data of SSCs, and maintenance records.

5. Management of obsolescence

During the lifetime of a research reactor, advances will occur in technology which may lead to difficulties in getting spare parts. Installation of new components may also lead to changes in failure modes (e.g. modern instrumentation contains microprocessors that have different failure modes from those of their older components). Changes in safety requirements and regulations and advances in knowledge may necessitate important modifications of the existing facilities. This includes, for example, improvement in the physical segregation of SSCs or in the resistance of the facility against the effects of internal and external hazards. Research reactor safety and operating documents may also become outdated or even obsolete. Periodic updating of such documents is needed to maintain their conformity with the actual status of the reactor facility and to take into account feedback from operating experience. Updating of these documents is also necessary when modification of existing (or installation of new) experimental devices is being performed.

Similar to the management of physical ageing, actions to manage obsolescence for research reactors should also be performed in a proactive manner before occurrence of any decline in reliability or availability of the reactor. Table 1 presents possible obsolescence conditions and recommended ageing management actions.

Table 1: Obsolescence conditions and recommended ageing management actions

Condition	Ageing management actions
Changes in technology	- Ensure systematic identification of useful service life and anticipated obsolescence; - Prepare modification projects for future replacement of obsolete SSCs; - Provide spare parts for the planned service lifetime or identify alternative suppliers.
Changes in safety requirements and regulations, and advances in knowledge	- Ensure compliance with current safety standards and regulations; - Consider modification of SSCs important to safety, as required.
Documentation becoming out of date	- Establish and implement an effective integrated management system.

6. Conclusion

Effective ageing management for research reactors could be accomplished by integrating existing operating programmes including maintenance, periodic testing and inspection, as well as applying good operational practices, using the results of research and development, and incorporating lessons learned from operating experience, performing periodic safety reviews as a central tool to confirm continued safe operation of the reactor. Nevertheless, it is important to recognize that effective management of ageing requires the use of a systematic approach that provides for minimizing in a proactive manner the ageing degradation, as a consequence of service conditions, monitoring, trending and early detecting SSCs degraded conditions, and timely implementing the necessary mitigation measures.

Refurbishment and modernization projects of research reactors are important activities for ageing management. These projects should not only be limited to pure replacement of systems and components, or to improve reactor availability and meet the users' requests but should also be performed to ensure compliance with the up-to-date safety requirements and criteria, including the IAEA Safety Standards.

7. References

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[7] International Atomic Energy Agency, Safety in Utilization and Modification for Research Reactors, IAEA SS No 35-G2, IAEA, Vienna, 1994.

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INSPECTION PROGRAM FOR U.S. RESEARCH REACTORS

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ABSTRACT

This paper presents an established program for inspection of nuclear research reactors to ensure that systems and techniques are in accordance with regulatory requirements and to provide protection for the health and safety of the public. The inspection program, implemented from the time a facility gets licensed, remains in effect through operations, shutdown, decommissioning, and until the license is terminated. The program establishes inspection methodology for operating, safeguards, and decommissioning activities. Using a performance-based approach, inspectors focus their attention on activities important to safety. Inspection procedures allow the inspectors to assess facility safety and compliance to applicable requirements. A well designed inspection program is an integral part of the mechanism to ensure that the level of performance in the strategic areas of reactor safety, radiation safety, and safeguards is acceptable and provides adequate protection of public health and safety.

U.S. Inspection Program

The general policy for regulation of Research and Test Reactors (RTR) is described in the Atomic Energy Act of 1954, as amended, Section 104.c which states:

"The Commission is directed to impose only such minimum amount of regulation of the licensee as the Commission finds will permit the Commission to fulfill its obligations under this Act to promote the common defense and security and to protect the health and safety of the public and will permit the conduct of widespread and diverse research and development."

This general policy is reinforced by the U.S. Nuclear Regulatory Commission (NRC) inspection program which performs a basic mission of determining whether a licensee’s reactor is acceptably safe and meets current regulatory requirements and commitments. The program establishes inspection methodology for operating, safeguards, and decommissioning activities and conditions. Using a performance-based approach, inspectors focus their attention on activities important to safety. Performance-based inspection emphasizes observing activities and the results of licensee programs over reviewing procedures or records. For example, an inspector may identify an issue through observing a facility activity in progress, monitoring equipment performance, or the in-facility results of an activity (e.g., an engineering calculation), and then let the observation lead to evaluation of other associated areas. Discussions with facility personnel and reviewing documents should be used to enhance or verify performance-based observations. This approach is designed to emphasize observation of activities.

While the licensee is responsible for facility safety and compliance with regulatory requirements, the NRC inspector is responsible to independently assess the licensee's fulfillment of his/her responsibilities. Advice or recommendations are not to be given to the licensee.

The U.S. research reactors are classified according to their maximum thermal power which determines the frequency and depth of the inspection program

Class I

- Thermal power is 2 Megawatts or more
- Inspection effort consists of 10 modules (inspection procedures)
- Entire program is repeated annually (2 one-week visits annually)

Class II

- Thermal power is less than 2 Megawatts
- Inspection effort is one module with 11 sub-topics
- Program is completed over a two year interval (2 one-week visits over two years)

Class III

- Permanently shut down (Possession Only License or Actively Decommissioning)
- Inspection effort is one module
- Inspected once every three years

Safeguards Inspections

- Licensees are required to implement an NRC approved physical security program tailored to the safeguards categories of material that the licensee may possess. In addition, licensees must report inventory and transactions at regular interval. The security portion of the RTR inspection program and the inspection frequency use a graded approach based on the amount of Special Nuclear Material on-site and type of fuel used - Highly Enriched in Uranium (more than 20% U-235) (HEU), or Low Enriched in Uranium (less than 20% U-235) (LEU).

Non-Routine and Reactive Inspections

- OSHA Industrial Safety
- DOT Transportation Requirements 49 CFR 172-178
- IP 86730 - Transportation of Radioactive Materials
- IP 86740 - Inspection of Transportation Activities
- Event/Accident root cause analysis (special inspections)
- Decommissioning Plan and License Termination Plan
- Worker allegations
- Public meetings
- Escalated enforcement

Inspection Procedures

As a general rule, inspections should be conducted in accordance with inspection procedures. Inspection procedures identify requirements that the inspector considers while evaluating the associated area. These requirements may not be the same as NRC requirements placed on a specific licensee. As such, it is not implied or intended that inspection program requirements are to be levied on the licensee. Any attempt to force inspection program requirements on the licensee constitutes misinterpretation of NRC inspection philosophy and misuse of inspection requirements.

It is not possible to anticipate al the unique circumstances that might be encountered during the course of a particular inspection and, therefore, inspectors are expected to exercise initiative in conducting inspections, based on their expertise and experience, as needed, to assure that all the inspection objectives are met.

Inspection Procedure Format

- Each inspection procedure follows a common format which includes:

  - Objective
  - Inspection requirements
  - Inspection guidance
  - Specific guidance
  - Resource estimate
  - References

- The inspection procedure is designed to confirm that the licensee's programs are consistent with the regulatory requirements.

- Examples of regulatory requirements are:

  - Code of Federal Requirements (CFR) Title 10 Parts 20, 50, 61, 71, and 73
  - Operating License (OL) / Technical Specifications (TS)
  - Confirmatory Action Letters (CAL)
  - Licensee commitments in Security Plan, Emergency Plan, and Reactor Operator Requalification Plan.

- Additional statements are found in the Final Safety Analysis Report, in national consensus standards (ANSI 15), and NRC NUREGs and Regulatory Guides.

Inspection Planning and Conduct

- Select the inspection procedures to be used for the appropriate Class of RTR.

- Schedule the inspection with licensee management. The NRC generally announces inspections at RTR, and for the most part does not conduct unannounced / surprise inspections).

- Review the appropriate material, such as previous inspection reports, licensee's TS and FSAR. Note any open issues. The results of past inspections, events evaluations, and inspector and management reviews shall be used to determine the focus of the inspection.

- Interviewing skills and careful records review may indicate an unanticipated program weakness. Take the time to determine the safety significance of the unanticipated issues at the expense of completing the routine IPs.

- Note taking is very important for maintaining official verifications of document review in the inspection report. For example, the inspector will need to record the full name, revision number, and effective date of each procedure, policy, report, or letter that was reviewed.

- An entrance meeting helps in scheduling the inspection time. Inspectors should hold an entrance meeting with the senior licensee representative who has responsibility for the areas to be inspected.

- An exit meeting sums up the preliminary results. At the conclusion of an inspection, inspectors must discuss their preliminary findings with the licensee's management at a scheduled exit meeting.

Inspection Reports

- Communicating inspection observations is an integral and important part of every inspection, whether done daily during the course of an inspection, or periodically with status meetings. A final inspection report must be issued to clearly communicate inspection results to licensees, NRC staff, and the public. The final report should describe the inspection scope, including the inspection procedure used, identify how the inspection was conducted (i.e., the methods of inspection), what was inspected, the criteria used to determine whether the licensee is in compliance, and describe the observations and findings. Whenever possible, an observation should be related to a requirement or commitment. Findings are an assessment of the significance and context of the observations. The inspection report should clearly relate how the finding relates to the observations, whether the finding is neutral, positive, or negative, and how significant the finding is. A report with no negative findings is acceptable. It is NRC policy that the licensee is encouraged to identify and permanently fix any program weaknesses. Safety significant program weaknesses will be characterized as Inspector Follow-up Items, Unresolved Items, or Violations. The goal is to formally issue routine inspection reports within 30 days and special inspection reports within 45 days.

Enforcement

- Program weakness at U.S. RTR are assigned a severity level (Severity Level I-IV) in the inspection report to identify its safety significance:
  - Actual Safety Consequences
  - Potential Safety Consequences
  - Impacting the Regulatory Process
  - Willfulness

- The level also determines the amount of the monetary fines and extent of corrective actions to be taken by licensee management.

- Violations may be written against individuals as well as facilities.

- Enforcement guidance and requirements are found in 10 CFR 2 and the NRC Enforcement Manual. The format for violations in the inspection report is provided in the Manual.

Civil Penalty Flow Chart

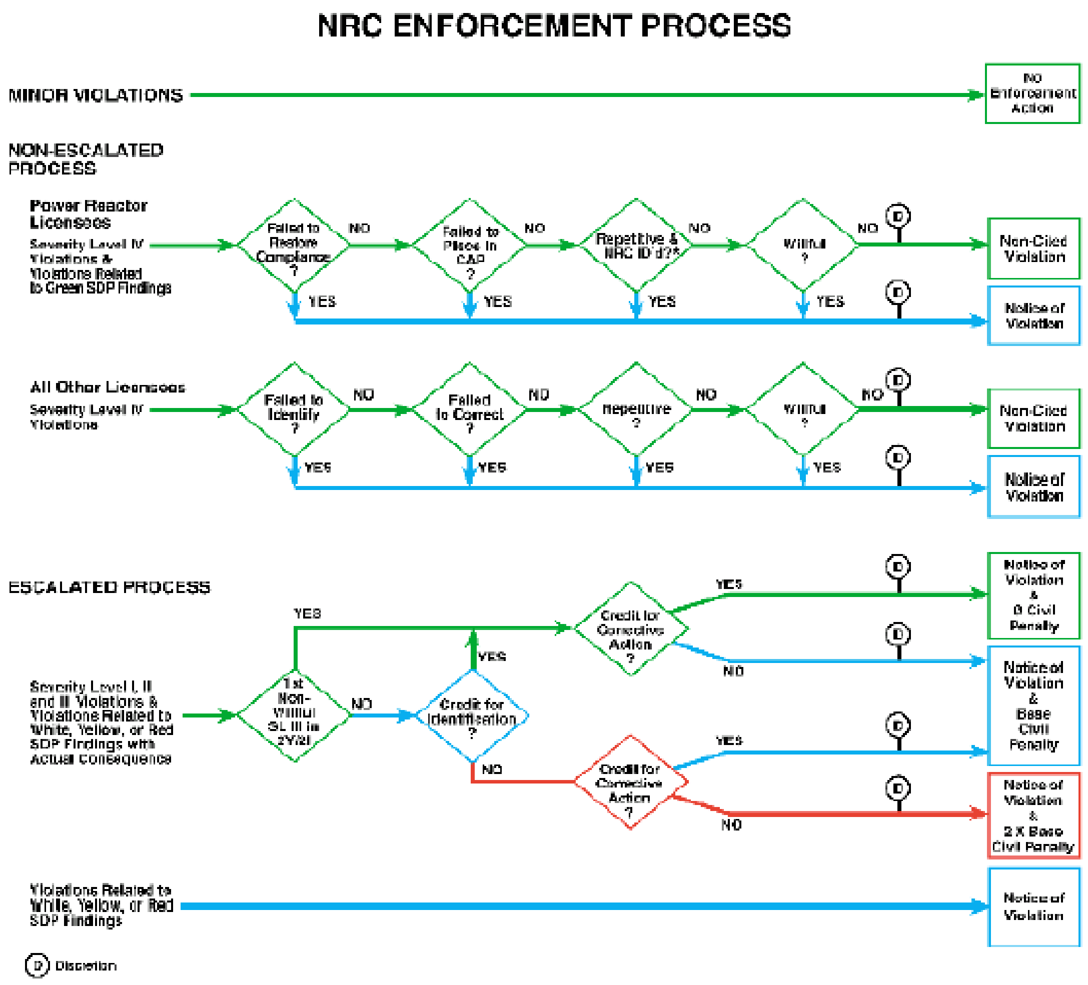

Figure 1 - This graphic represents the NRC's graded approach to dealing with violations, both in terms of addressing their significance and developing sanctions.

FINDINGS FROM WORKING FOR THE IAEA INITIATIVE ON RESEARCH REACTOR AGEING AND AGEING MANAGEMENT

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Abstract

1995 the last sharing and compiling the existing knowledge about of the Research Reactor (RR) Ageing and the respective Fighting took place during a well attended conference at Geesthacht, Germany, documented in a bulky conference report. In 2008, the International Atomic Energy Agency has initiated another collecting and evaluating in order to make the recent experience in that field available to the entire RR Community. In this respect, RR operators, plant and system fabricators, and authorities as well as independent experts have been approached worldwide for providing contributions and fortunately about every second member of the RR Community replied.

The paper is going to inform on the experience gained by the contacts and communication, the replies as well as the non-replies, underlying motives as problems, and mainly, some statistical evaluation of the findings. The respective IAEA data base being accessible to all members of the RR Community will be briefly characterised in structures and contents.

1. INTRODUCTION, HISTORY OF THE INITIATIVE

15 years ago, in 1995 a conference on RR Ageing took place, commonly organized by the German Research Center GKSS and the IAEA and hosted by the GKSS at Geesthacht[1]. It had been attended by more than 100 participants.

Parallel to preparing that conference the IAEA supported by an International Working Group plus a Symposium at Chalk River and a Seminar at Bangkok prepared a respective TECDOC dealing with the Management of Research Reactor Ageing[2].

In the period between 1995 and 2009, the IAEA has supplemented the documents dealing with the Ageing & Ageing Management of Research Reactors either directly or indirectly (for the related terms dealing in parts with ageing see Fig.1); some examples of valid documents are:

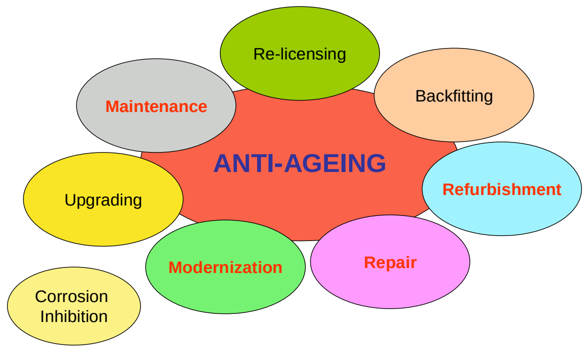

Fig.1: Fighting Ageing and closely related activities

- IAEA-TECDOC-448 “Analysis and Upgrade of I&C Systems for the modernization of RRs” (1988)

[1] The proceedings have been published as GKSS 95/E/51 containing 31 contributions from RRs + 10 other (authorities, RR-groups)
[2] IAEA-TECDOC-792 ”Management of Research Reactor Ageing“, issued 1995

- IAEA-TECDOC-1263 “Application of non-destructive testing and in-service inspection to research reactors” (2001)
- Technical Report Series No. TRS-443 “Understanding and Managing Ageing of Material in Spent Fuel Storage Facilities” (2006)
- Technical Report Series No. TRS-418 “Corrosion of Research Reactor Aluminium Clad Spent Fuel in Water” (2004)
- IAEA-SSS No. NS-G-4.2 “Maintenance, Periodic Testing and Inspection of RR Safety Guide” (2006)
- IAEA-Nuclear Energy Series No. NP-T-5.4 “Optimization of RR Availability and Reliability: Recommended Practices” (2008)
- IAEA SS Draft Safety Guide DS 412 “Ageing Management for Research Reactors” (2008)

2. STRUCTURE OF THE INITIATIVE

End of 2008 the IAEA invited some experts for a brainstorming on what can be done to intensify the fighting of ageing at RRs. In evaluating the existing rules and the numerous parallel initiatives such as on Modernisation & Refurbishment [3] and Availability and Reliability[4] as well as considering that the IAEA has renewed the general guiding for Managing of Research Reactor Ageing already[5], the expert group came to the conclusion that a actual compilation of the actual knowledge and experience of the RR operators with ageing effects and the respective curing might be a very helpful effort for

- Making all existing knowledge in that field available to all members of the RR community, and
- Making - by approaching these members under the initiative - all operators aware of the importance of considering and fighting ageing.

Such compilation of what the community knows in total and what was assumed having substantially grown during the 15 years after the last appraisal would support understanding of ageing and channelling the necessary support to the most important subjects. It was never understood to reduce the importance or replace the systematic fighting of ageing at an RR and the application of the comprehensive compilation of tools and systematics contained in the IAEA guides, standards, and TECDOCs.

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The expert group recommended inviting all members of the RR community (operators, authorities, independent technical experts, RR industry (plant designers/suppliers, component manufacturers)) for contributing with their experience and applied methods to the intended compilation, planned as a data bank accessible to every member of the RRC via the INTERNET. For the necessary systematics of the information a template and a scheme for categorizing the ageing issues were created. This scheme consisted of 76 RR systems structured into 9 system groups (reactor block & fuel, cooling, confinement, I&C, power supply, auxiliaries, experimental facilities, documents & configuration management, others (staff, codes, etc.) and 13 Ageing Mechanisms which were:

(A) Radiation induced change of properties
(B) Temperature induced change of properties
(C) Creep due to stress / pressure

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[3] Meeting at Delft 10/2006, published as IAEA-TECDOC-1625 08/2009
[4] IAEA-Nuclear Energy Series No. NP-T-5.4 “Optimization of RR Availability and Reliability: Recommended Practices” (2008)
[5] The 1995 IEAE-TECDOC-792 (Management of Research Reactor Ageing) has been replaced by the IAEA Draft Safety Standard IAEA-DS412 (Ageing Management for Research Reactors) in 2007

(D) Mechanical displacement / fatigue / wear from thermal cycling, flow induced vibration, .
(E) Material deposition
(F) Flow induced erosion
(G) Corrosion
(H) Damage due to power excursions
(I) Flooding – deposition and chemical contamination
(J) Fire – effects of heat, smoke, or reactive gases
(K) Obsolescence / technology change
(L) Changes in requirements / acceptable standards
(M) Others (time dependent, to be named)

The members of the RR community were asked to report on the issues by selecting the category, writing a short report comprising the issue and the mitigating or corrective actions, and providing a suitable contact address, all in English language.

3. TEN MONTHS OF WORK FOR THE INITIATIVE

Subsequently the main findings of that endeavour and the experiences from working for the initiative are depicted.

First the gross figures of the initiative:

- 133 RRs approached (+ 28 authorities / industrial suppliers etc.)
- 77 RRs replied (+ 6 authorities / industrial suppliers)
- 367 ageing issues reported distributed over 62 out of the 76 different systems

All the related contacts were made by Email; without that tool the entire initiative would have been impossible[6]. Numerous Emails became necessary to get a final version of a completed template; the record in number of Emails per one completed template has been 55 (forward + backward). Altogether the number of different versions of completed templates at my PC arrived at 851, just to illustrate what it means to work for the IAEA and the RR community. On average, every template had a revision rate of 4.5 times.

Why are such initiatives so demanding, for the contributor as well as for the receiving party? To me a contributor faces one or more of the following problems:
The language: Having read 851 templates in 10 month I am quite aware of the problem which many operators have to read bulky IAEA documents or understand the instruction sheet of an IAEA template. Thus, there are special thanks to all those contributors who mailed completed templates despite major language problems. And I hope I could transfer most of them to versions acceptable to the author and understandable to the community. However, the IAEA should not underestimate the language issues when expecting that their numerous guides and standards are considered adequately.
The template: Many contributors interpreted the binding template as some hint on how to proceed. Some drafted there own template to overcome the biding limitations. Many did not reply to the specific items listed there. And sometimes the template, that has to be admitted, was not suitable, e. g. for authorities and their contributions.
The advisor: Frequently it took quite some efforts to convince an operator to complete a template for a rather unknown advisor, telling him the specific problems of his plant

[6] It should be mentioned here that the author expected a list of Email-addresses of the RR operators being available at the IAEA. This expectation was completely wrong. The search for suitable addresses and contact persons was a substantial effort at the beginning of the work and continued to be a problem as such addresses are ageing fast as well, e. g. by changing everything except the RR (the name of the operator, the provider, the names of staff, etc.). Older publications are of rather limited support, too.

originating from its age. This reaction was understandable and accepted but created supplementary efforts.

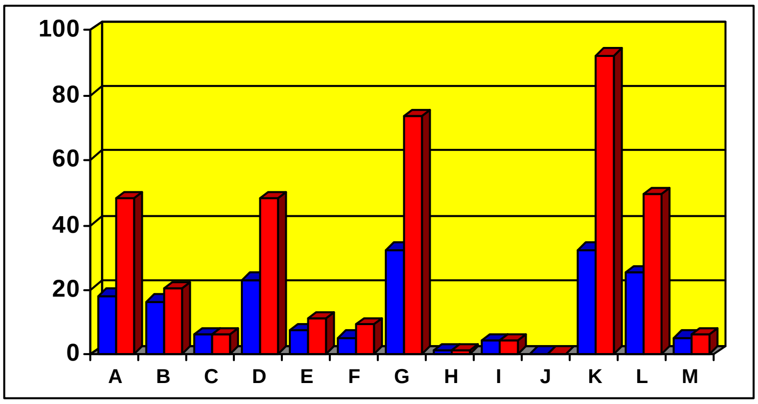
The 'secrets': Opposite to other institutions the nuclear facilities have been trained over decades to report on their problems always and completely, even when the problem could be cured easily before it became a risk. However, ageing issues are frequently in the grey area between need to report (e. g. to the supervising authority) and the demand to keep the RR in operation for clients or the own research tasks for a certain period. Certainly – also learned from the completed templates – budgets for fighting / curing ageing issues are not available easily. Thus, the decision for frankly reporting on the issues might have been not an easy-one at many plants and some operators have decided against publishing their experience, e. g. on problems long ago[7].

The effort: Completing a template was a simple task: 15 minutes for an US operator, 2 – 5 hours in case of language problems. Only very few were allowed to give up during the revision period or withdraw their input as the specific problem could not be accepted as caused by ageing.

In summing up, I consider language problems[8] and keeping 'secrets' secret have been the major reasons for not replying or rejecting the request. Also, the (too) frequent approaches by the IAEA and authorities make operators less willing to participate in just another initiative.

4. ABOUT RESULTS

Replying of every second institution that had been approached seems to be a rather positive result and allows – besides the valuable experience as reported in any specific case – some statistical considerations. The number of reported cases available for statistics is 367 which have been filled into the system – mechanism matrix of 76 systems × 13 mechanisms (about 1000 elements) and looked at in terms of the frequency of the named mechanisms (Fig. 2) as well as systems and system groups (Fig. 3).



A=Radiation induced
B=Temperature induced
C=Creep due to stress...
D=Mech.displ./fatique/wear
E=Material deposition
F=Erosion
G=Corrosion
H=Damage ← power exc.
I =Flooding consequences
J=Fire consequences
K=Obsolesc./techn.change
L=Requir./standard changes
M=other (staff, PSA, ...)

Fig.2: Frequency of named Ageing Mechanisms
blue: nominated systems (out of 76) per mechanism
red: total named issues (out of 367) per mechanism

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[7] The author recommends the IAEA to consider approaching the authorities at those countries which operate RRs prior to starting such initiatives in order to avoid hiding experience with ageing due to the described conflict of interests

[8] The author admits that statistical evaluations for the different regions (continents) did not show a major difference in the reply rate from the region, opposite to what the language problem seems to suggest. The lowest response rate was from the US RRs.

From Fig. 2 one gets the most frequently named mechanisms being obsolescence / technological change (92 out of 367) and corrosion (70 out of 367), whereas damages from power excursions, flooding & fire are of no importance obviously.

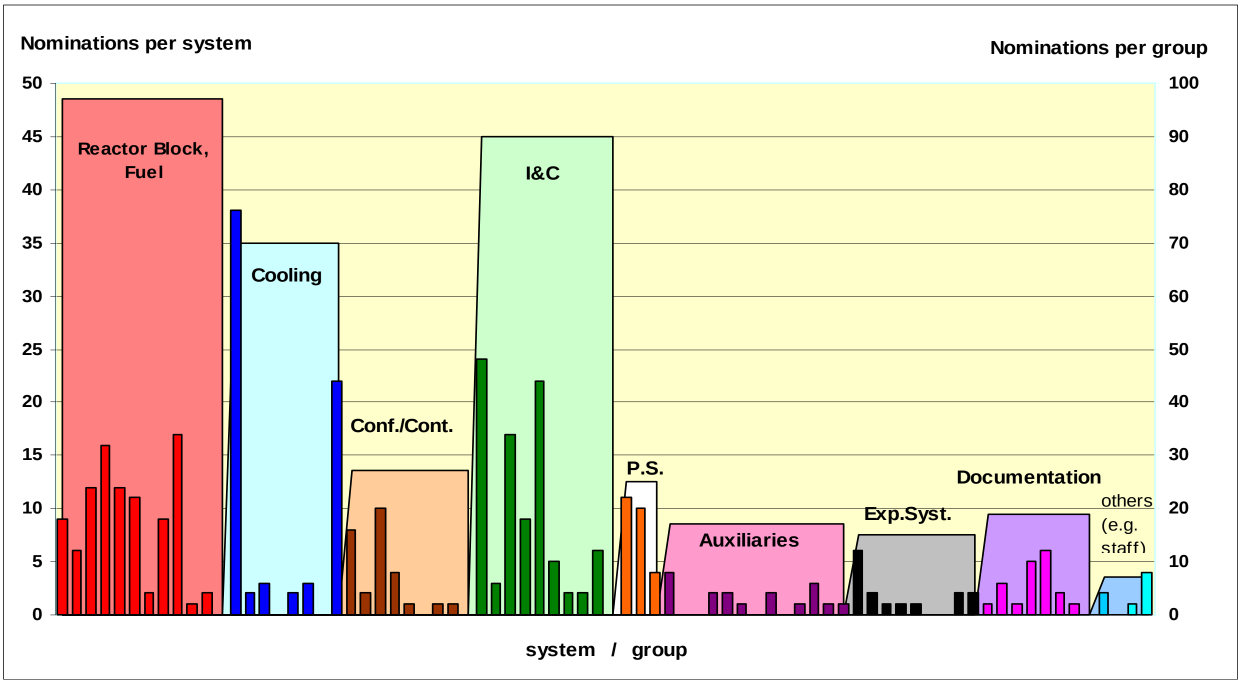

Fig. 3: Frequency of named single systems (columns) and system groups (coloured areas)

Fig. 3 shows that the most named systems are Primary Cooling (38) and Reactor Protection (24), followed by Secondary Cooling and Control Console (22 each). This is of special importance as all these systems are safety relevant. Fig. 3 adds the information that among the system groups, Reactor Block /Fuel (97) dominates the nomination against I&C (90) and Cooling (70).

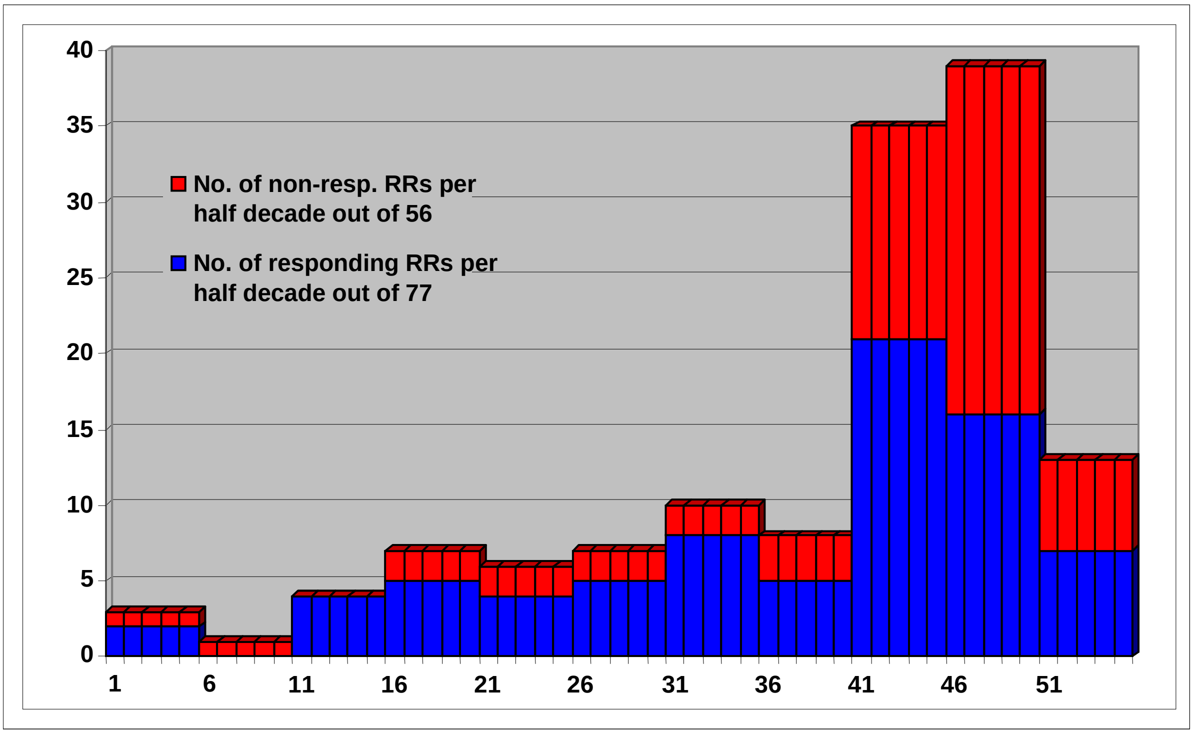

Fig. 4: Age distribution of approached (participating and non-participating) RRs per half decade

As known to everybody in the RR-community, the RRs worldwide are really old mostly. This is also reflected by Fig. 4, showing the age distribution of the participating (and non-participating) RRs. The average age (status October 2009) of the approached 133 RRs was 39.5 years past first criticality, the respondents had 38 years since, the non-responders 42.

An analysis of the average thermal power of all RRs having been involved in the initiative resulted in 13.48 MW per approached RR. The average power of those having replied is very nearby (13.21 MW). Further Fig. 5 clearly indicates that there is no dependence of the reply rate from the power of the RR, the power being also a measure for the number of staff at the plant generally.

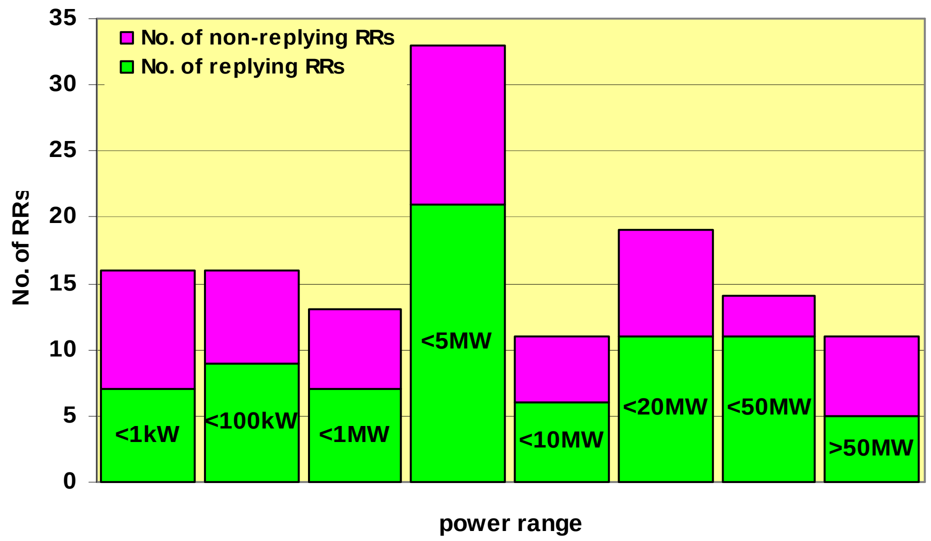

Fig. 5: RR power dependence of replies

5. THE TECHNICAL MEETING AT THE IAEA

Thirty operators[9] presented their reported ageing issues at a Technical Meeting in Vienna[10] early October; in these cases the ageing was illustrated in real detail, far beyond the contents of the completes templates[11].

At that meeting the increasing shortening of lifetime of modern I&C systems has been emphasized by some participants. Having been already an all-time-problem item this decrease of lifetime by e. g. non-deliverable spare parts becomes even more concerning. On the other hand, projects for renewal of control consoles for certain RR-types such as TRIGAs or recently SURs[12] might be a way out for some of the low power reactors. Generally, the short lifetime of some systems and the early termination of spare part supplies are contradictory to the long life of most of the RRs. The statistics of the contributions shows that obsolescence is already the most frequent ageing mechanism (see Fig. 2) and I&C systems are the second most mentioned system group (see Fig.3). Also it should be mentioned that the specific support by the IAEA in this field is a rather aged document.[13]

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[9] Beyond the 30 contributions on sole ageing at RRs there were 10 more contributions under the headline of Modernization & Refurbishment at RRs as well as some general contributions

[10] A completed template was a pre-condition for getting invited to the Technical Meeting

[11] The contributions are foreseen to be published by the IAEA in a revised edition of TECDOC -1625: Research Reactor Modernisation and Refurbishment which contains the results of a workshop held at the HOR at Delft, The Netherlands in October 2006 only for the time being

[12] SUR stands for Siemens UnterrichtsReaktor

[13] IAEA-TECDOC-443, Analysis and Upgrade of Instrumentation and Control Systems fort he Modernisation of Research Reactors, Vienna (1988)

6. THE DATA BASE & CONCLUSIONS

The importance of the collection of the experience on ageing is however not in the statistics. It is contained in the created data base. Reporting on how the ageing problem has been discovered / detected, how the consequences have been handled, how budget constraints had influenced curing, whether external help has been needed, whether the treatment of the ageing issue was part of a broader context on dealing with ageing at the plant[14] and how the authorities have been involved[15]. All those features are reported for 367 cases, and additionally there is always a contact person (with Email address) for more information.

The data base is aimed as a living document[16]. The access to the data base has been restricted to members of the research reactor community[17], to avoid misuse of the contained information. But frequent use as well as frequent updates and supplements (without being pushed, squeezed and tortured by persons as the author) are what the expert team aimed at when this initiative had been started in October 2008.

[14] It should be mentioned here, that the replies to this aspect in the completed templates – if provided – clearly demonstrated a lack of systematics in approaching ageing issues at the majority of RRs despite the support given by the IAEA since 1995 latest. Mostly staffing seems to be inadequate for a systematic following up of all existing supportive recommendations plus safely operating the RR.

[15] The author considers it being an interesting task to evaluate the 367 reported cases with regard to the aspects mentioned here, e. g. by a student performing a practical course

[16] New input to the data base can be fed in via < E.Bradley@iaea.org>

[17] The data base can be accessed via the link <http://filenet.iaea.org/OurWork/ST/NE/NEFW/AD/index.html>

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