SAFETY RE-EVALUATION OF THE HOR REACTOR

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1. Introduction

The HOR reactor of the Interfaculty Reactor Institute (IRI) of Delft University of Technology has been in operation since 1963. This swimming pool type reactor is the main experimental facility of the institute. It is used for a broad range of both fundamental and applied research. Originally the reactor was intended to serve the Dutch universities. While retaining this inter-university character many of the research themes are carried out with both national and international scientific partners.
Gradually the power of the reactor was increased to 3MW maximum operational power. In 1969 a licence was obtained for that power level. Ever since the reactor is operated at a nominal power level of 2 MW. In 1980 a permit was obtained for a new control room and for extension and renewal of the reactor instrumentation. Those modifications were finished in 1982.
In the early nineties it became apparent that further modifications and extensions were necessary. As a consequence of the policy of the US government on proliferation the conversion of HEU to LEU was a prerequisite for the availability of fresh fuel and the return of spent fuel to the US. At the same time the development of new instruments for neutron scattering and positron annihilation asked for a new beam hall. Therefore a renewal of the permit was necessary. After extensive studies in which a safety report and an environmental impact statement were made, the application for a new licence was submitted in 1995. The new licence was granted in 1996. However, due to legal actions by individuals, the licence did not become effective until 1997. The beam hall was inaugurated in 1998. In the same year the conversion from HEU to LEU was started. In June 2000 the "Delft Positron Centre" was officially opened. Here a unique high intensity positron beam serves the 2D-ACAR and Lifetime instruments. The legal procedures lasted until fall 2000 when all objections raised against the new licence were dismissed by the Council of State.
Requirement C16 in the new licence asks for a periodical integral safety re-evaluation of the HOR reactor every 10 years and starting after 2 years.

2. Scope

The main purpose of a safety re-evaluation is to establish the level of safety of the HOR reactor.. This in the context of the established practice of design and operation of this type of research reactor and the developments thereof. Future and lasting safety should be sufficiently guaranteed. In this re-evaluation not only the technical aspects of the reactor as such are assessed, but also operational practice, safety culture and organisation must be taken into consideration.
The following actions are relevant for long term safety:

a. confirm that the nuclear installations are sufficiently safe to allow continued operation;
b. identify and evaluate those factors that might impose restrictions for safe operation of the nuclear installations in the foreseeable future;
c. evaluate the applicable safety standards and their use with the nuclear installations; Propose improvements that can reasonably be implemented;

For the re-evaluation the following stages were foreseen:

1. preparation and drafting of the activity plan;
2. determine the conformity between operational practice and requirements as laid down in the permit;
3. decide on safety reference base;
4. comparison of present situation and safety reference base;
5. recognise possible weaknesses and shortcomings and define actions to increase safety;
6. final report;

3. Conformity between operational practice and requirements of permit;

The permit stipulates a number of requirements that should be met. These requirements cover the following issues:

- Operational management of the IRI
- HOR, reactivity control and fuel elements
- HOR, organisation and operations
- Radiation protection and expertise required
- Radioactive waste
- Environmental impact resulting from nuclear operations
- General requirements for handling radioactive material or ionising radiation emitting equipment
- Safeguards
- Insurance compensation nuclear liability

Systematically every individual requirement was studied and it was analysed in what way these requirements were met. The safety report of the reactor, technical safety specifications, different safety analyses of the HOR, emergency plan, safety instructions, security plan and the various procedures and instructions from the HOR quality assurance system contain the basis for this assessment of conformity.
The resulting document showed that all requirements were met and that the HOR is operated in conformity with the licence.

4. Safety reference base

The assessment of the safety aspects and the determination of the requirements that represent the state of the art in safety can only be carried out satisfactorily if there is a common understanding between the regulating authority, the Dutch nuclear safety authority KFD and IRI about the safety reference basis. It was agreed that the following documents form the safety reference base:

- Code of the Safety of Nuclear Reactors: Design, IAEA Safety Series, Safety Standard No. 35-S1
- Code of the Safety of Nuclear Reactors: Operation, IAEA Safety Series, Safety Standard No. 35-S2
- IAEA-publications: Safety Series: SS 35-G1, SS 35-G2, SS 35-G6, SS 35-G7, SS 50-SG-O12
- Services Reports Series: IAEA-SVS-1
- TECDOC-Series: TECDOC-792
- Dutch Nuclear Safety Rules. These rules apply for Power reactors and are to be interpreted for a research reactor of the HOR size.
- Previous safety evaluations and analysis.
- Representative examples from other university research reactors of comparable size and application ("good practices")
- Knowledge and insight obtained through regular contacts with colleagues operating other research reactors : AFR-, TRTR-, IGORR

With these documents a systematic self assessment was carried out of the operational management of the HOR reactor. The different issues as they appeared in the documents were methodically investigated and shortcomings and weaknesses were identified. Also the strong points of the HOR practice were identified. This exercise resulted in a number of tables where the issues, shortcomings and remedial actions were given. See Tables 1 and 2.

During this analysis it became clear that it would be extremely difficult to find "representative examples from university research reactors of comparable size and application" with which the HOR could be compared on a one to one basis. Also the lack of documentation in English played a part. Therefore IRI proposed to the Dutch authorities to ask the IAEA for an independent assessment by means of an Integral Safety Assessment of Research Reactors: INSARR. These missions are carried out by a team of international experts according to standard IAEA procedures. It was initiated in February 2000, carried out in May and reported in July 2000.

Table 1

Review of present situation of HOR with IAEA SS No. 35 and NVR 1.3

Safety Series No.35-S1 (Design)

§	RESULT/REMARK	ACTION/FOLLOW UP
DESIGN LIMITS OF PARAMETERS		
518	Design parameters of original system are not always retrievable.	Where relevant to safety: reconstruct original information
DESIGN FOR RELIABILITY		
521	Survey of safety relevant instrumentation is given in the safety report, however maximal non availability is not explicitly mentioned..	Where relevant: state explicitly.
DESIGN FOR RELIABILITY, **Redundancy and single failure**		
523	Safety function containment isolation: one set of isolation valves only.	Planned installation of double set.
CODES AND STANDARDS		
533	Sometimes unknown, non retrievable or missing in parts.	Approach to be determined according to § 534.
DESIGN FOR OPERATIONAL STATES, **Material selection**		
543	Design data not always retrievable (see §518).	Safety relevant data reconstruct/check and record.
DESIGN FOR ACCIDENT CONDITIONS		
545	Loss of containment isolation function when single isolation valve fails. (see § 523).	Execute the planned installation of double valves.
PROTECTION SYSTEM		
634	Insufficient documentation in TIP of bypassing of eigt safety actions by key-swiches (used for measurements or maintenance	Document and record conditions in safety report. Check procedures.

5. Safety relevant operational performance

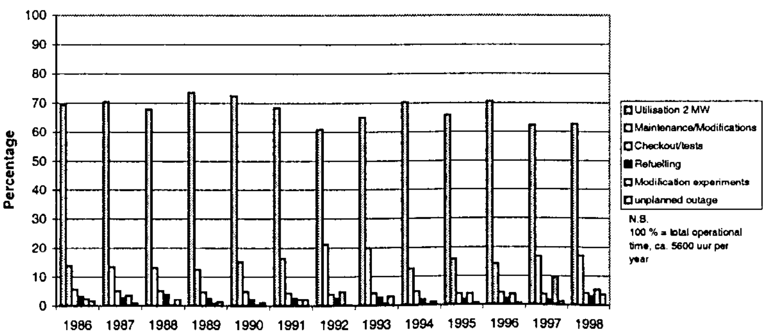
For a safety evaluation it is of importance to know the data of past performance. Especially long term trends in the value of the various key indicators give information on the development of the technical state of the installations and ageing effects. Combined with nuclear engineering judgement, expectations for future safety can be deduced. Not only technical aspects can be subject to such an evaluation. Also the development of the learning organisation and in the safety culture are reflected in the appropriate key indicators.

The following aspects were studied:

Utilisation and unavailability

Since its start in 1963 the use of the reactor has changed significantly. This is a direct consequence of the evolution of the research programmes. In the first years most of the research was

Figure 1:
HOR Performance Indicators
Utilisation, planned and unplanned outages



concentrated on reactor physical topics. More recently neutron beam research and neutron activation became more prominent. The latest developments are in the field of materials research with the new intense positron beam. This positron beam is generated in the reactor and by magnetic guides the particles are transported to the instruments in the new beam hall.
The availability of the reactor is an important performance indicator. It is an indication of the percentage of the total working time of the reactor (5600 hrs) available for experiments. Especially the non planned outages are a good indicator for the operational performance. In figure 1 the development from 1986 - 1999 is given.

Performance of fuel and containment

An important performance indicator is the concentration of Iodine-131 in the pool water. This concentration directly reflects the quality of the fuel elements and the integrity of their aluminium cladding with respect to the retention of fission products from the fuel matrix and ion exchanger functioning. The iodine-131 concentration in the pool water is determined weekly by measuring water samples. The fission product monitor continuously measures iodine-131 in the pool water and every two weeks samples of the ion exchange resin bed of this monitor are analysed for fission products.

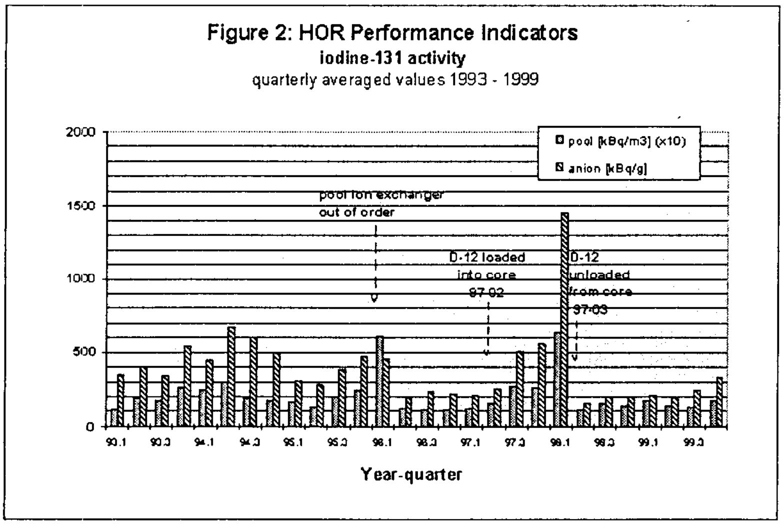
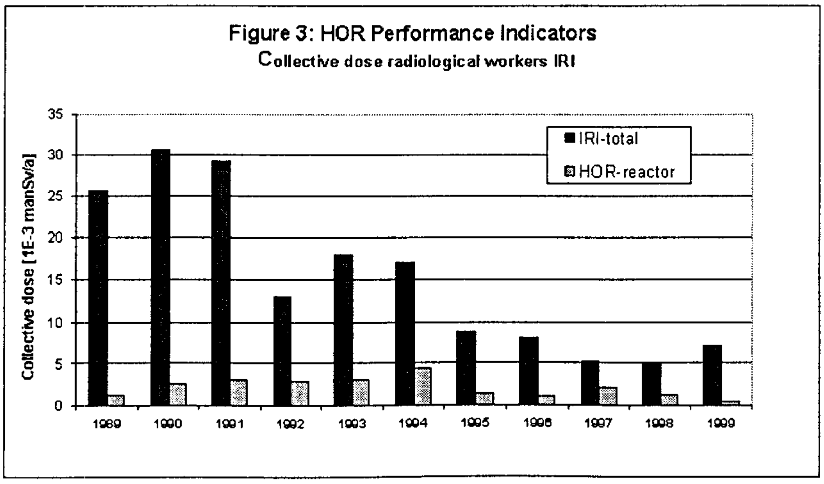


Figure 2 shows clearly that early 1998 the iodine concentration was significantly elevated. Presumably during 1997 a small defect in the cladding of one of the fuel elements has developed. An extensive diagnostics programme identified element D-12 as the faulty one. It showed one small spot of pitting corrosion. The element has been permanently removed from the core. The leakage was so small that only with very sensitive equipment and under special operating conditions and after integration over prolonged periods that the defect could be proven. It never had any radiological consequences either for personnel or surrounding. Of all 158 fuel elements used so far this was the first one to show this slight defect.

Radiation dose



The radiation dose received by IRI personnel is monitored for gamma and neutrons. Data are available over a 10 year period. In figure 3 we give these data. The average collective dose over this period was 15×10^{-3} man Sv/a for the whole IRI population. For people working in the reactor containment this figure is 2.1×10^{-3} man Sv/a. The trend over this period shows a 4 - 5 fold decrease in the collective dose. Scientists and technicians, working with neutron beams receive the highest individual dose while reactor operators show radiation dose that are at the lower end of the distribution.

Emissions of radioactive substances to the environment

Waste water from laboratories and the reactor installations are collected in large tanks. Only very slight quantities are found in the waste water as most of the radioactive waste is collected separately. Before the wastewater is discarded to the sewer, alpha, beta and gamma activity is determined. The beta activity annually discarded is 11.0 MBq/a and the gamma activity is 6.25 MBq/a. Both quantities show a downward trend. The alpha activity was always below the detection limit of 0.5 kBq/m^3.

Emissions to the air are dominated by argon-41. During the whole period the average argon-41 concentrations were below the detection limit.

Other performance indicators

Other performance indicators were evaluated such as: experience and technical qualifications of reactor staff and their age distribution, operational safety and emergency planning.

All the data collected under this part of the safety re-evaluation were reported to the regulating authority for further evaluation and comparison with state of the art installations.

6. Integral Nuclear Safety Assessment of Research Reactors

The INSARR mission was carried out according to the guidelines for the Review of Research Reactor Safety, IAEA Services Reports, SVS-1, IAEA, Vienna 1997. The following aspects were reviewed by a international team of experts.

Operating organisation and reactor management
Modifications to the reactor
Design
Conduct of operations
Operational limits and conditions
Safety analysis
Safety analysis report
Maintenance and periodic testing
Radiation protection
Radioactive waste management
Emergency planning

The results were laid down in: Report of the INSARR Mission to the HOR Research Reactor, Netherlands, 7-12 May 2000, IAEA, Division of Nuclear Installation Safety,IAEA-NSNI/INSARR/00/01, Vienna 2000.

The conclusions of this mission were recommendations, items that should be addressed at short notice, suggestions, items which are advised to carry out, and good practices, items that are recommended to other organisations.

- RECOMMENDATIONS

  All written procedures and guides should be submitted to the Reactor Safety Committee for its review and recommendation as required by the Committee's procedure

  The procedure for approval of an irradiation should be modified to include the approval of the reactor manager

  The SAR should provide adequate references for the review and assessment of the material discussed in the SAR

  An annual inspection programme should be instituted for judging the condition of the reactor pool internals and provide a basis for the evaluation of ageing phenomena

  The emergency plan should be modified to follow the recommendations of the IAEA emergency planning document especially in the area of accident classification and response actions for each classification

- SUGGESTIONS
  The safety analysis should use IAEA terminology in the discussion (i.e. DBA and BDBA)
  A system of work permits should be inaugurated
  A radiation protection refresh training program should be developed for all workers in categories A and B
  The written radiation protection procedures and instructions should be incorporated into the QA program to assist in making a distinction between the two
  The written radioactive waste management procedures and instructions should be incorporated into the QA program to assist in making a distinction between the two
- GOOD PRACTICES
  Mentoring takes place between the chief operator in training and the professional staff at the facility. This training has shown to be very effective in nuclear power training and is infrequently used for research reactor training
  In order to combat isolationism, the IRI sends experienced operators and radiation protection personnel to other reactors for internships. While this is done frequently in developing countries, it is infrequent in industrialised countries
  IRI has instituted a programme for optimisation of operational and experimental activities in order to reduce the amount of radioactive waste produced
  IRI has made clear efforts and succeeded in keeping the facility updated and using information technology to enhance operational safety

7. Conclusion

According to the initial planning IRI has identified all corrective actions that are pursuant from the different investigations. The implementation of a number of improvements has already taken place, others are under way. A final report has been submitted to the regulator/inspectorate, who has accepted it as a basis for a discussion on the corrective actions. A formal approval of the results of the safety re-evaluation is expected for mid 2001.