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International Atomic Energy Agency

REPORT OF THE

INTEGRATED SAFETY ASSESSMENT OF RESEARCH REACTORS (INSARR)

MISSION

TO THE

Halden Boiling Water Reactor HBWR

Halden, Norway

17-29 June 2007

INTEGRATED SAFETY ASSESSMENT OF RESEARCH REACTORS (INSARR)

DEPARTMENT OF NUCLEAR SAFETY AND SECURITY

DIVISION OF NUCLEAR INSTALLATION SAFETY

INTERNATIONAL ATOMIC ENERGY AGENCY

ORIGINAL: ENGLISH
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Mission date:	17-29 June 2007
Location:	Halden, Norway
Facility:	Halden Boiling Water Reactor, HBWR
Organized by:	IAEA At the request of the Norwegian Radiation Protection Authority (NRPA)

Conducted by:

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CONTENTS

1. INTRODUCTION

1.1 BACKGROUND

The Integrated Safety Assessment of Research Reactors (INSARR) missions are an IAEA safety review service offered upon request to all Member States. The Research Reactor safety reviews are conducted in line with the existing procedures for INSARR missions and they are based on the IAEA safety standards for research reactors.

The operating license of the Halden Boiling Water Research Reactor (HBWR) expires in 2008. The Institute for Energy Technology (IFE) responsible for the operation of HBWR submitted a license application to the Norwegian Radiation Protection Authority (NRPA) to extend the operation license until 2018. The NRPA requested the IAEA to conduct an INSARR mission to assess the safety of the facility. The results of this mission will provide a technical support to the regulatory body in the licensing process. The INSARR mission to HBWR was implemented upon an official request of the Norwegian Radiation Protection Authority (NRPA), submitted to the IAEA on 23 November 2006 by the Permanent Mission of Norway (letter attached in Annex 1).

1.2 ORGANIZATION STRUCTURES

1.2.1 Norwegian Radiation Protection Authority (NRPA)

The Radiation Protection Authority supervises all use of radiation sources in medicine, industry and research. Limiting values have been set for the maximum radiation dose permissible in an occupational context and for the public.

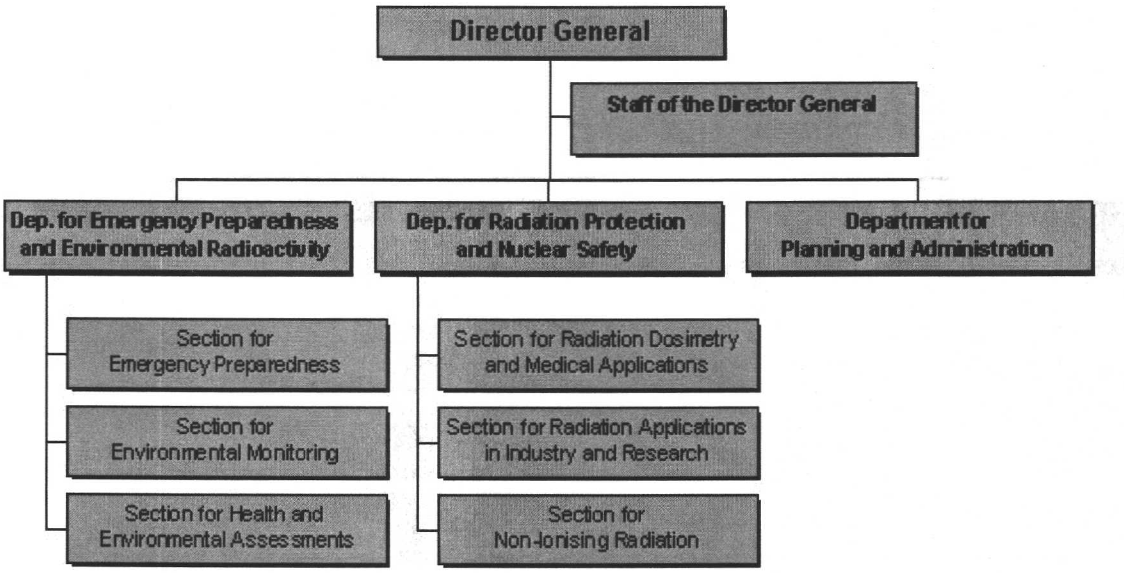


FIG. 1. Organization Chart of NRPA

The NRPA performs its regulatory and control activities based on the Act of 12 May 2000 on Radiation Protection and Use of Radiation and the Act of 12 May 1972 on Nuclear Energy

Activities.

The NRPA supervises the safety of Norwegian nuclear installations: the research reactors (Kjeller and Halden) and the waste facility (Himdalen), all three are operated by the Institute for Energy Technology (IFE). The supervisory responsibility also includes monitoring of the transport of radioisotopes to and from the facilities and the treatment and storage of radioactive waste.

The NRPA organization chart is presented in Figure 1. Under the authority of the Director General of NRPA there are three departments: Department for Emergency Preparedness and Environmental Radioactivity, Department for Radiation Protection and Nuclear Safety, Department for Panning and Administration.

The research reactors are under the supervision of the section for Radiation Applications in Industry and Research.

1.2.2 The Institute for Energy Technology (IFE)

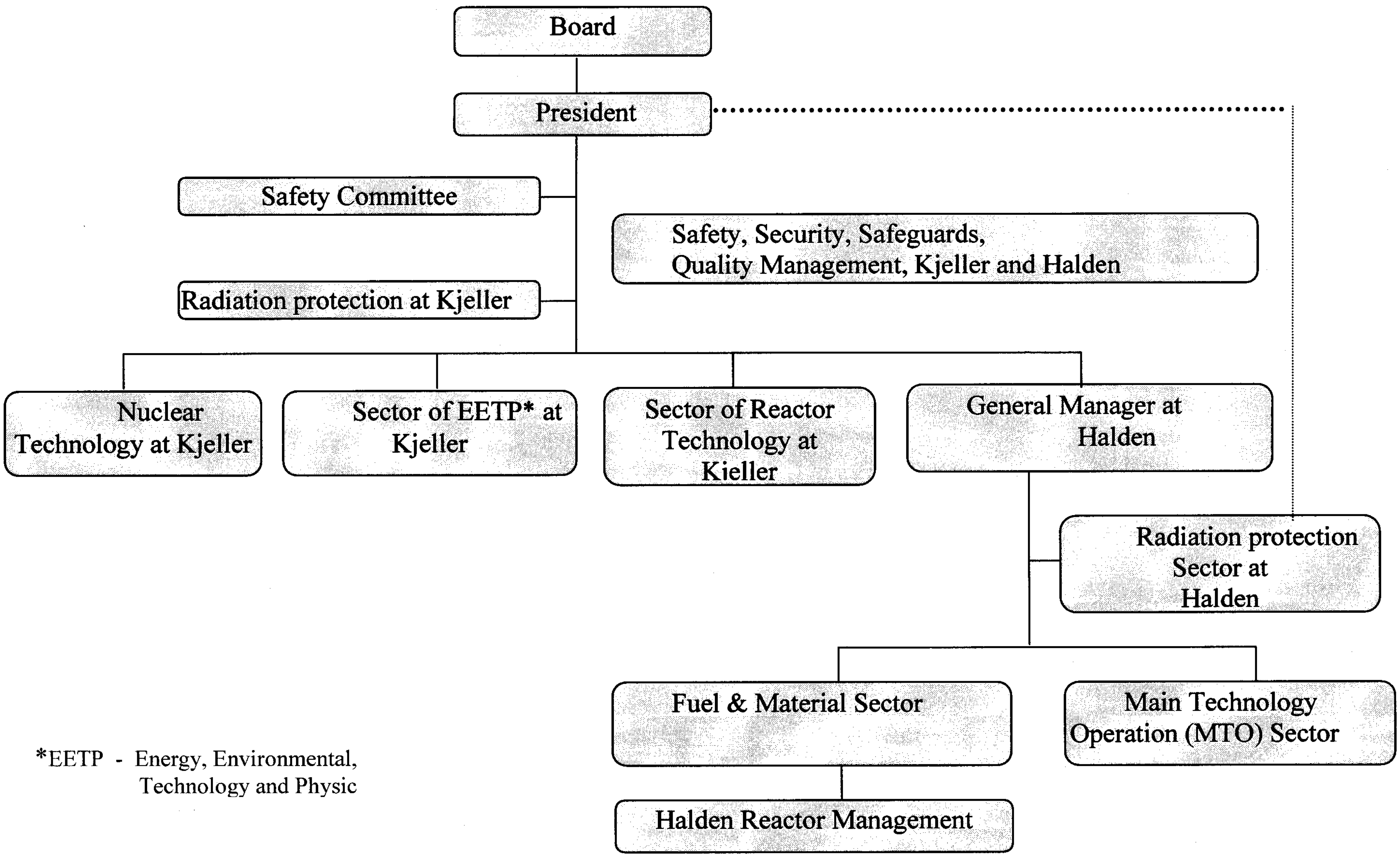
The IFE is an international and independent research institute which was founded in 1948. The IFE installations are located in Kjeller and Halden. The Institute professional activities are divided into five sectors. The Institute runs the international Halden Reactor Project on nuclear safety research. The organizational chart of the IFE is shown in Figure 2.

The supreme body is the IFE Board which has to ensure that the Institute is operated and developed in accordance with the objectives. The Managing Director has the overall responsibility for all the activities of the Institute and he reports to the Board.

The responsibility for the operation of Halden reactor and the conduct of other associated activities lies upon the Reactor Operation and Engineering Division which has four sections:

1. Maintenance and Installation;
2. Reactor Operation;
3. Design and Development; and
4. Safety.

The reactor is operated by six shifts, each of them is composed of a Reactor Engineer, Shift Leader, and two Reactor Operators. The operations are carried out in accordance with written procedures. There are 36 written operating procedures in the Control Room and two in the office of the Senior Reactor Engineer.



*EETP - Energy, Environmental, Technology and Physic

FIG. 2. Organization Structure of IFE

1.3 SHORT DESCRIPTION AND HISTORY OF THE FACILITY

The Halden research reactor was commissioned in 1959 as the 13th reactor in the world. The thermal power of the reactor is 20 MW and its design as heavy water cooled and moderated BWR is quite unique. The steam produced by the reactor facility is used by a paper factory located on the same site. In 1958 an Agreement was signed with the OECD to establish an international research project at the Halden reactor. Presently the Halden Reactor Project (HRP) is supported by approximately 100 organizations in 17 countries.

Figure 3 shows a cross-section of Halden Boiling Water Reactor. The reactor is operated 24 hours a day, seven days per week and 28 weeks per year. The operating schedule mainly depends on the demand of the experimental programme. The non-operational weeks are devoted to maintenance activities and for installation or removal of experimental devices.

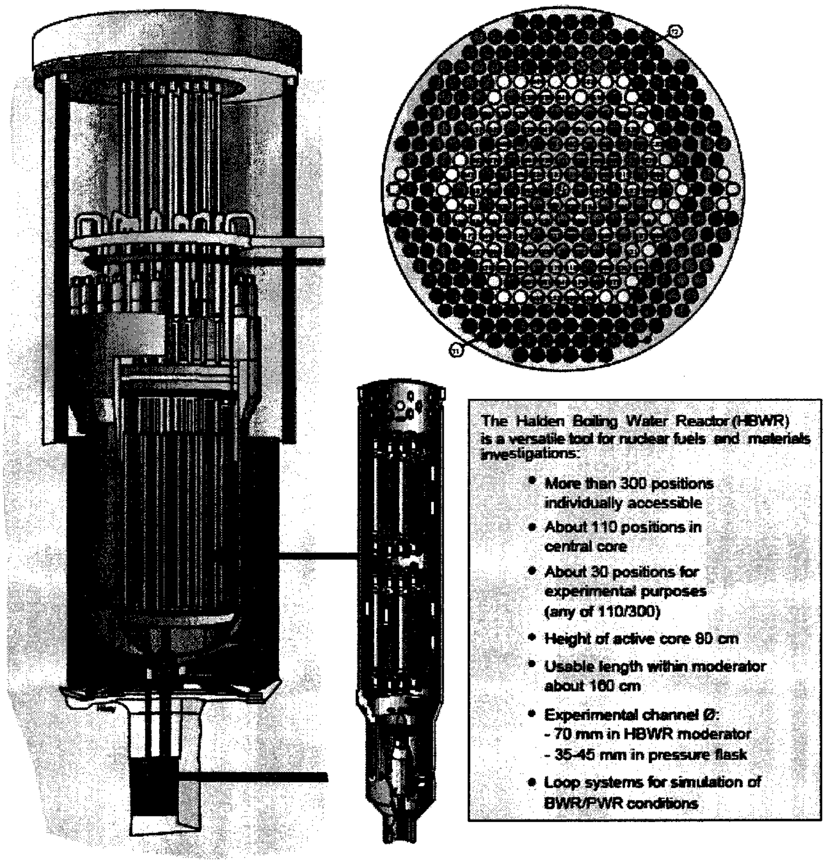


FIG. 3. Halden Boiling Water Reactor

REACTOR SITE

The Halden Boiling Heavy Water Reactor (HBWR) is located in Halden, a coastal town in south-east Norway near to the border of Sweden. The reactor hall is situated within a rock hillside on the north bank of the river Tista. The site area is 7000 m^2. The reactor vessel (space) and the primary coolant system are inside a rock cavern with a net volume of 4500 m^3. The rock covering is 30-50 m thick. Heat removal circuits are either placed inside the reactor hall or in the reactor entrance tunnel. The control room and service facilities are placed outside the excavation. The service buildings contain offices, workshops, and laboratories.

REACTOR SYSTEM

The HBWR is a natural circulation boiling heavy water reactor. The maximum power is 20 MW (thermal), and the water temperature is 240°C, corresponding to an operating pressure of 33.3 bar. Fig. 4 shows a simplified flow sheet of the reactor systems.

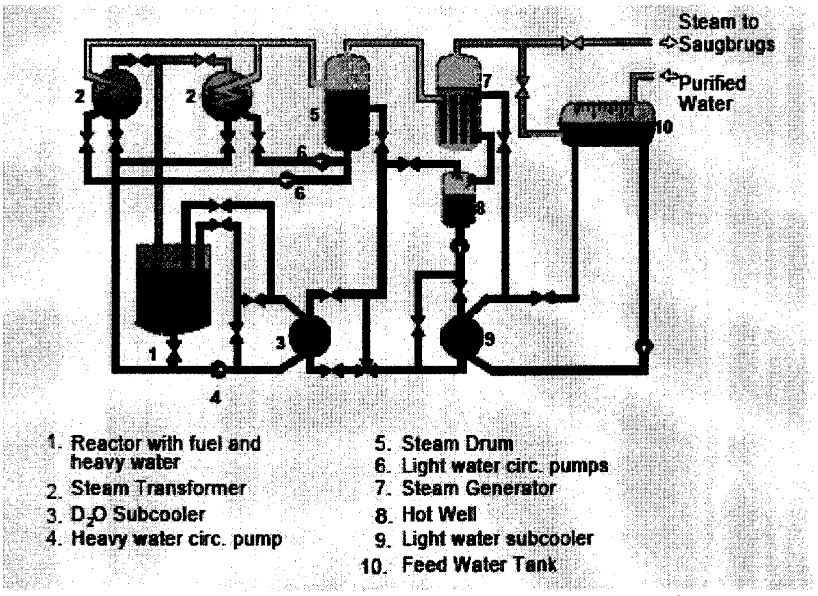


FIG. 4. Simplified Flow Sheet (Graphic Panel)

The reactor pressure vessel is cylindrical with a rounded bottom. It is made of ferritic steel, the bottom and the cylindrical portion are clad with stainless steel. The flat reactor lid has individual penetrations for fuel assemblies, control stations and experimental equipment.

14 tons of heavy water constitute the coolant and moderator. A mixture of steam and water flows upwards by natural circulation inside the shroud tubes which surround the fuel rods. The steam is collected in the space above the water while the water flows downwards and enters the fuel assemblies through the holes in the lower ends of the shroud. The steam flows to two steam

transformers where the heat is transferred to the light water secondary circuit. The condensate from the steam transformers returns to the reactor by gravity.

In the secondary system, two pumps circulate the water through the steam transformers, a steam drum and a steam generator where steam is produced in the tertiary system. The tertiary steam is normally delivered to the nearby paper factory, but may also be dumped to the river.

The access to the reactor hall is not allowed when the reactor is operating, and therefore all control and supervision are carried out from the control room.

REACTOR OPERATING CONDITIONS

The fuel charge of the reactor core consists of a combination of test fuels from the organizations in member countries and driver fuel assemblies, which provide the reactivity for the reactor operation.

Light water and high-pressure loops provide facilities for testing fuels under BWR and PWR conditions.

A selection of nominal reactor operating conditions is given in Table I.

Power Level	up to 20 MW (th)
Reactor Pressure	33.3 bar
Heavy Water Saturation Temperature	240°C
Primary Steam Flow (both circuits)	160 ton/h
Return Condensate Temperature	238°C
Subcooler Flow	160 ton/h
Plenum Inlet Temperature	237°C

Table I. Nominal Reactor Operating Data

Each driver fuel assembly consists of eight UO_2 fuel rods with 6 % enrichment and standard fuel pellet diameter (10.49 mm).

CORE CONFIGURATION

The core consists of about 110-120 fuel assemblies, including the test fuel, in an open hexagonal lattice with a lattice pitch of 130 mm. 30 lattice positions are occupied by control stations. The maximum height of the fuel section is 1710 mm, and the core is reflected by heavy water.

Selected core data are given in Tables II and III. Figure 5 shows a typical core map. The central position in the core is occupied by an emergency core cooling tube with nozzles, and between eight and fourteen core positions contain pressure flasks for light water, high pressure test loops.

Assembly	
Number of rods per assembly	8
Pitch circle diameter, mm	50
Length from lowest pellet in lower rod to	
highest pellet in upper rod, mm	810
Fuel	
Material	UO_2
Enrichment, %	6
Shape	Sintered pellets
Density, g/cm^3	10.52
Pellet diameter, mm	10.49
Pellet height, mm	8.6-10.8
Length of enriched fuel per rod, mm	748-811
Length of natural fuel per rod, mm	12
Cladding	
Material	Zr-2, Zr-4
Inner diameter, mm	10.67
Wall thickness, mm	0.8
Nominal diametral clearance fuel/cladding, mm	0.16-0.18

Table II. Driver Fuel Assembly Design Data

Number of Fuel Assembly	110
Number of Control Stations	30
Core height – usable length for placing fuel	1710 mm
Configuration	open hexagonal lattice
Lattice Pitch	130 mm
Reflector Top: thickness	300 mm
Reflector Bottom: thickness	380 mm

Table III. Selected Core Data

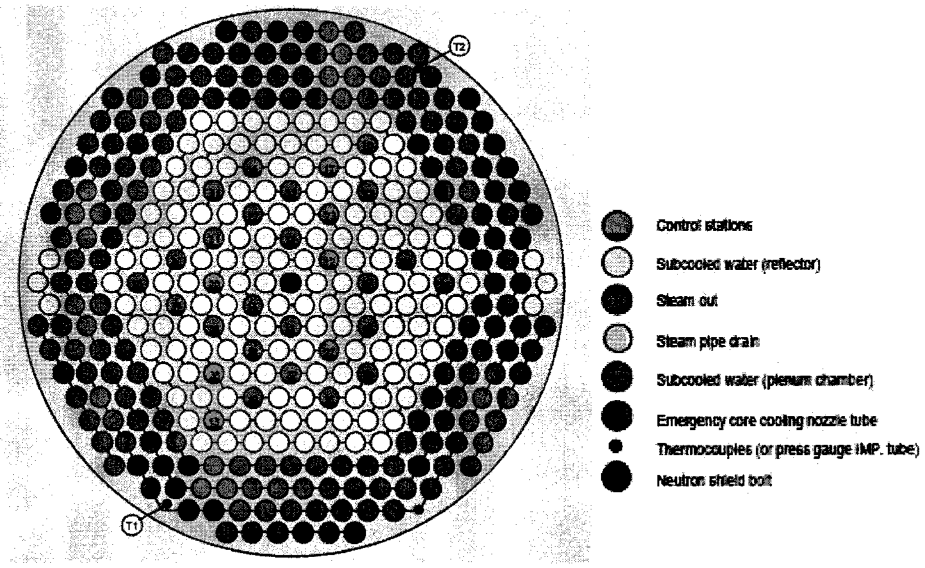


FIG. 5. Plan view of reactor top lid

REACTOR VESSEL

The design working pressure of the HBWR pressure vessel is 40 bar with a saturation temperature of 250°C. The hydraulic acceptance pressure test was carried out at 54 bar, 35 % above the design pressure. The normal operating pressure is 33.3 bar, with a corresponding saturation temperature of 240°C.

There are around two reactor shutdowns per year, dictated primarily by the experimental programmes, and a few additional cooling downs for special tests. The normal heating and cooling rates of the moderator are restricted to 10°C/h.

Inspection and recertification pressure tests are performed every 3rd year at 10 % overpressure. These pressure tests are performed with water/steam at saturation temperature. According to the requirements set by Norwegian Boiler Authority, the inspection and test programmes include ultrasonic examination of vessel welds, lid, bolts, bottom nozzle and primary system piping, and evaluation of radiation induced material changes. Last pressure test was done in 2003. No more tests are planned according to Swedish regulation on ASME code. Nevertheless, leakage tests are done.

All the available welds of the bottom nozzle and of the beltline region of the reactor vessel wall are being ultrasonically examined at the inspections. Also the top lid and the flange bolt are being inspected, the bolts 100 % by ultrasonic. The primary system piping is subject to inspection by NDT methods. No defect indications in the above mentioned inspections have been found.

The irradiation induced changes in the vessel material are being monitored by material testing, flux evaluations and fracture analysis. The Charpy and fracture mechanics test on surveillance specimens are performed by VTTs laboratory in Finland. Neutron flux and fluence assessments enable quantification of the fluence received by the different parts of the vessel, account taken of the changing core loading over the years.

The outcome of the material testing, fluence evaluations, inspections forms the basis for the assessments of vessel integrity. Internationally accepted codes, rules and recommendations are used in a consultative manner. The material tests and the analysis performed indicate that there is no significant problem concerning the safety of the reactor. The issue related to the lifetime of the reactor vessel was the subject of detailed evaluation by the present mission team. The recommendations formulated concerning this issue are presented in Appendix 2 on Issue Paper CLE-01 and Appendix 3.

1.3.1 Utilization programme

The utilization programme for 2006-2008 is defined in the "Halden Reactor Project Programme" document issued in April 2005 by the Institute of Energiteknikk.

The utilization programme of HBWR includes experiments on fuels and materials. The Fuels & Materials programme is defined and executed under the four main categories:

a. Fuel High Burn-up Capabilities in Normal Operating Conditions
b. Fuel Response to Transients
c. Fuel Reliability Issues
d. Plant Lifetime Assessments

Fuel high burnup capabilities in normal operating conditions is done to acquire fuel property data for design and licensing in the burnup range 60 to 100 MWd/kg. Both test fuel and re-fabricated commercial fuels are being used in the investigations. The activities comprise a number of studies of fuels in use in light water reactors. Experiments with heavily instrumented test rods are being performed for studying the performance of fuel with Gadolinium and fuels with other additives. Important activities are also related to characterising the conditions leading to cladding lift-off under fuel rod overpressure.

Fuel response to transients experiments are providing experimental data on fuel behaviour related to reactivity initiated transients and on phenomena occurring during a loss-of-coolant accident. The main activity is the LOCA test series.

Fuel reliability issues are aiming at determining the mechanisms and operational conditions that can affect cladding integrity. The main activities relate to crud deposition and axial offset anomaly studies. Further, long-term corrosion test using commercial alloys are carried out.

Facility lifetime assessments is aiming at generating validated data on stress corrosion cracking of reactor materials at representative stress conditions and water chemistry environments. Issues related to pressure vessel embrittlement are also addressed. Experiments are made to study BWR and PWR crack growth rates under varying chemistry and stress conditions, time to failure, and material embrittlement.

The experimental facilities available for the programmes are continuously being upgraded and expanded. This relates in particular to the loop facilities in which testing can be performed under a variety of well-defined pressure, temperature, water chemistry and irradiation conditions. A total of ten loops are in operation, most of them being capable of serving more than one experiments provided that the required conditions are compatible. A loop with blow-down capability is available for the execution of the LOCA test series.

1.3.2 Recent operational occurrence

IFE implemented a discrepancy system under QA Programme at HBWR. This system represents the internal reporting system of operational occurrences developed to collect lessons learned from incidents at Halden. The "discrepancy" system started to be implemented two years ago.

The lessons learned from past events, occurred before the discrepancy system was implemented, should be collected in a systematic way and the feedback should be included in a training programme and safety assessment of new experiments and operational configuration.

The counterpart mentioned two significant events that occurred after the year 2000:

1. Exceeding the notification level for release of tritium to water in April 2007 due to drainage from an experimental loop and;
2. Fuel failure in a test assembly at Halden reactor due to lack of cooling, in January 2001

Both events are summarized in Appendix 4:

2. OBJECTIVES AND SCOPE OF THE MISSION

During the pre-INSARR mission to HBWR in February 2007, the NRPA discussed with the IAEA representative the objective and the scope of the INSARR mission.

The NRPA requested the INSARR mission to receive an independent opinion from the peer review experts on the safety of the Halden research reactors, a thorough examination of the operation of this reactor in accordance with the agreed safety review areas, especially the assessment of issues related to ageing and advise on how to proceed with the supervision of the safety and operation of the reactor.

2.1 SCOPE OF THE MISSION

As agreed during the pre-INSARR mission, the safety review areas for the main INSARR mission should address:

a. Safety Analysis Report (SAR);
b. Operational Limits and Conditions (OLC);
c. Quality Assurance (QA);
d. Training and qualification;
e. Regulatory supervision;
f. Experiments, modifications and commissioning after major modifications;
g. Radiation protection programme including waste management, airborne and liquid effluents and their radiological impact;
h. Decommissioning plan.

At the request of the counterpart, a specific safety review on life time assessment of reactor pressure vessel has been included in this mission.

2.2 BASIS FOR THE ASSESSMENT

According to the letter received (see Annex 1) and the services provided by the IAEA, the basis for the safety review of research reactors was the IAEA Standards and the INSARR Guidelines. The list of most referred IAEA Safety Standards is listed in References, Part A.

2.3 DOCUMENTS RECEIVED FROM THE COUNTERPART PRIOR AND DURING THE MISSION

During the pre-INSARR mission, the NRPA provided to the IAEA team a CD containing the advance package of information. Those documents are listed in References, Part B.

The list of documents received during the mission and reviewed by the mission team are also included in References, Part C.

2.3.1 Short description of the assessment (way and methods)

The following procedures for the actual conduct of the safety review were used:

1. Examination and assessment of the technical documentation;
2. Performance of a walk-through of the facility;
3. Observation of operational activities, and reactor structures, system and components;
4. Interviews and technical discussions with facility personnel;
5. Interviews and discussions with authorities and personnel from the Operating Organization.

The mission report is based on the so-called ISSUE PAGES developed during the mission by the IAEA team members and the local technical counterparts. The Issue Pages were developed on the basis of the following:

a) They should reflect the transparency of the process;
b) They should facilitate the retrieval of information;
c) They should facilitate follow-up actions;
d) They should be easily understood in the multicultural environment of the IAEA missions to facilitate the exchange of information between the team members and the local technical counterparts.

In the first part of each Issue Page (Issue clarification - Observations) the experts and the local counterparts are requested to isolate the facts that may be considered as a Safety Issue. These are the points in which there should be agreements between both counterparts, avoiding if needed to make judgments or giving any recommendation (just the facts). It follows "possible safety consequences" in which may not be agreement between the team member and the local counterparts. In case of disagreement the local counterparts are requested to write their own comments, explanations, etc. in the section identified as "Counterparts views". The issue is further discussed in the internal team meetings. Recommendations and Good Practices (see below) are team advices.

All Issue Pages are included in Appendix 2.

2.3.2 Review criteria

The INSARR review compares the observations and finding with the IAEA Safety Standards and practices found in other research reactors worldwide. The comparison may result in recommendations, suggestions, comments and good practices presented to the Operating Organization by the team as a whole, in accordance to the following definitions:

Recommendation

Is a team advice to improve the safety, it will be reviewed during the follow-up INSARR mission based on IAEA Safety Standards and good recognized practices. It will focus on WHAT to do. However, under Comments approaches on the HOW can be mentioned.

The recommendations are numbered in the respective Issue page as **R#**

Suggestion

Is a team proposal in conjunction either with a recommendation or may stand on its own. It may indirectly contribute to improvements of the safety but is primarily intended to enhance performance. The suggestions are numbered in the respective Issue Page as **S#**.

Good Practice

Is a proven performance, activity or use of equipment, which the team considers to be markedly superior to that observed elsewhere. It should have broad application to other facilities. The suggestions are numbered in the respective Issue Page as **GP#**.

Comments

Are proposal for the implementation of the recommendations or suggestions. But do not constitute a team advice. The comments are numbered in the respective Issue Page as **C#**.

The issue number is composed by a code of 3 letters according to the review area:

Scope of the Mission Safety Review Areas from INSARR Guideline

Areas or topics of the assessment	**Code**
Safety Analysis Report	**SAR**
Safety analysis	**SAN**
Operational limits and conditions	**OLC**
Regulatory supervision	**REG**
Safety committees	**SCO**
Safety culture	**SCU**
Operating organization and reactor management	**OOR**
Management system	**MSY**
Conduct of operations	**COP**
Utilization and experiments	**UEM**
Training and qualifications	**TRQ**
Ageing management	**AMG**
Maintenance and periodic testing	**MPT**

Modifications	**MOD**
Commissioning after modifications	**COM**
Operating Procedures	**OPP**
Radiation protection programme	**RPP**
Radioactive waste management	**RWM**
Emergency planning	**EMP**
Decommissioning	**DEC**
Fire protection	**FIR**
Environmental Impact Assessment	**EIA**
Life extension of the reactor pressure vessel	**LER**

3. CONDUCT OF THE MISSION

The mission was conducted following the programme attached in Annex 1.

3.1 INSARR TEAM

The INSARR review team was composed of two IAEA staff members: Mr. Hassan Abou Yehia, (Team Leader) and Ms. Cristina Ciuculescu (Deputy Team Leader); five external experts: Mr. Nils Johan Lennart Gustafson (Studsvik Nuclear, Sweden), Mr. Sergey Morozov (Rostechnadzor, Russian Federation), Mr. Denis Rive (Institute for Radiation Protection and Nuclear Safety, France), Mr. Ricardo Waldman (NRA, Argentina), and Mr. Fred. J. Wijtsma (NRG Petten, Netherlands). The details of the review team are presented in Annex 3.

The mission started with a training and team assembly session in Oslo on Sunday, 17 June 2007 with all the members of the review team (RT). During this session, the IAEA staff presented the INSARR structure, the procedure to write the issue pages during the mission, the reporting activities during the mission, and general information concerning logistic aspect. Each expert provided presentations and feedback on the documents (References, Part B) received in advance.

3.2 ENTRY MEETINGS

An entry meeting was held on Monday, 18 June 2007 for the mutual introduction of NRPA staff, the main counterparts from Halden, the review team members, and the welcome address from the requesting organization NRPA. The list of participants to the opening meeting is provided in Annex 4.

The NRPA director presented the regulatory frameworks and expresses his expectations from the INSARR mission.

After the opening remarks, the team leader presented the IAEA Programme on the Safety of Research Reactors and the main provisions of the Code of Conduct on the Safety of Research

Reactors. The team leader presented also the summary of the preliminary remarks on the safety of Halden research reactor based on the assessment of the advance package of information.

From the NRPA headquarter the review team travelled to Halden in the afternoon.

On Tuesday morning, 19 June 2007, the review team and the technical counterparts met at Halden site. After introduction of all participants, the manager of Halden Reactor Project presented the utilization programme for 2006-2008.

FIG. 5. Photo from the Entry Meeting at Halden Site.

After the opening meeting, the review team performed the walk-through of the Halden research reactor. The detailed comments and observation raised during this technical visit are included in Appendix 1.

3.3 BILATERAL MEETINGS WITH THE COUNTERPARTS

Each morning there was a bilateral meeting between the team leader and the main counterpart. All the issues and findings from previous day were discussed openly.

Following the programme presented in Annex 2, the review team was split in several working groups to analyse the safety review area and safety documents with the technical counterparts.

At the end of the working day, each afternoon, the review team held at the hotel a meeting to brief the team leader on the main issues identified during the working sessions and to discuss the safety issues. Based on the discussions with the counterparts, the review team drafted the issue pages presented in the Appendix 2.

Several specific sessions were organized during the mission to discuss and clarify items important to safety:

1. On 19 June, Tuesday afternoon there was a plenary session on the studies for pressure vessel ageing;
2. On Monday 25 June there was a dedicated meeting with Safety Committee members.

4. MAIN CONCLUSIONS

The INSARR mission team was pleased to notice good practices of the Operating Organization (IFE) in several review areas such as the safety culture, public information, strong internal and external communication, transparency on safety matters and motivation to improve the safety of HBWR.

The INSARR mission team found that a good Quality Assurance programme is implemented by the Operating Organization. The team recognised the effective strategy adopted by the IFE to transfer and maintain the knowledge related to the facility.

The INSARR mission team has recommended to IFE several improvements at the organizational level to increase the independence of the Safety Committee and to reinforce the position of the Health Physics function within the Operating Organization. Other improvements have been recommended to formally establish a technical support group advising the reactor manager on safety issues, and to improve the communication from the Safety Committee to NRPA.

After a detailed review of the Safety Analysis Report, Operational Limits and Conditions and Emergency Preparedness Plan, the team recommended that these safety documents should be completed to incorporate important information and analyses currently presented in other documents. The most relevant information that needs to be incorporated is the bases of the design requirements, comprehensive list of Postulated Initiating Events including human errors and external events; analysis of these events, limiting conditions for the experiments; list and location of different monitoring systems important for safety with the associated alarms.

Concerning the important safety issue related the lifetime assessment of the reactor pressure vessel, the team has assessed thoroughly the different documents provided by the counterparts and recommended to enhance the monitoring of the integrity of the vessel. This consists mainly in increasing the frequency of controls and In Service Inspections and to enlarge the controlled area of the base material.

Another important safety issue addressed by the team is related to the prevention and protection against fire. An urgent need was identified to perform a new comprehensive fire analysis and to implement improvements of protective measures including fire barriers and compartments, reduction of combustible load and installation of additional fire detectors.

The effective implementation of the above mentioned improvement could be reviewed during a future follow-up INSARR mission.

In conclusion, from a safety point of view, the team considers at present, that there is no major problem which may constitute a hold point against the continuation of the HBWR operation.

5. RECOMMENDATIONS

.1 RECOMMENDATIONS TO THE GOVERNMENT OF NORWAY

- The resources of regulatory body (NRPA) dedicated to the control and supervision of the safety of research reactors should be increased in order to establish regulatory documents covering in particular the licensing process for research reactors and new experiments, the assessment capability and acceptance criteria of the safety analysis report (SAR) and the operating limits and conditions (OLCs) as well as the criteria for event and incidents investigation and reporting.

- A solution for long term storage of spent fuel or reprocessing should be investigated.

5.2 RECOMMENDATIONS TO THE NORWEGIAN RADIATION PROTECTION AUTHORITY (REGULATORY BODY)

- The safety assessment ability of NRPA staff should be enhanced. This could be ensured either by providing extensive training for the staff or by using external expertise as technical support. In this regard the creation of a "Standing Safety Group", composed of external experts to advise the regulatory body on important safety issue should be considered.

- It is suggested that NRPA take the initiative to establish with the Operating Organization a common working group with the task of drafting the needed technical regulatory guidance.

- It is suggested to NRPA to establish a planned and systematic inspection programme. The scope of this programme and the frequency of inspections shall be commensurate with the potential hazard posed by the research reactor.

5.3 RECOMMENDATIONS TO THE OPERATING ORGANIZATION (INSTITUTE OF ENERGY TECHNOLOGY - IFE)

n order to ensure rigour and thoroughness at all levels of the staff in the achievement and aintenance of safety it is recommended to IFE to

- ensure that it has sufficient staff with appropriate education and training at all levels;
- strictly adhere to sound procedures for all activities that may affect safety, ensuring that managers and supervisors promote and support good safety practices while correcting poor safety practices;
- review, monitor and audit all safety related matters on a regular basis, implementing appropriate corrective actions where necessary;
- be committed to safety culture on the basis of a statement of safety policy and safety objectives which is prepared and disseminated and is understood by all staff.

- In order to further enhance the transparency and communication on safety issues, the minutes of the safety committee meetings should be transmitted to NRPA

5.3.1 Summary of recommendations and good practices

Safety Analysis Report

- The Operating Organization should review and update the information in the SAR for the reactor cooling systems. According to the IAEA SS-35-G1, A.602 the SAR shall describe in detail the design and the operation of the primary cooling system. The design and performance characteristics of the main components (pumps, valves, heat exchangers, piping) should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials of which the components are made and the effects of irradiation on these materials shall be specified. The primary vessel, together with in-service environmental factors such as corrosion, fatigue and thermal stress cycling shall be described.
- The chapter on Radiation protection and activity release should be updated in accordance to the IAEA SS-35-G1, para.1201-A.1241, describing for normal operation conditions:
    1. The radiation protection programme
    2. Sources of radiation at the facility;
    3. Facility design for radiological safety;
    4. Waste management system;
    5. Dose assessment for normal operation.
- The chapter on facility design for radiological safety should include a description on how the implemented radiological provisions (e.g. zoning, shielding, radiation monitoring, etc.) reduce exposure to personnel, minimize the undesired production of radioactive material, reduce the time spent for maintenance and operational activities in which the possibility exists of internal or external exposure, and maintain releases of radioactive material to the environment as low as reasonably achievable.
- This section should describe the permanent radiation areas, effluent and airborne radiation monitoring systems and should include in particular the following information:
    - location of monitors and detectors;
    - type of monitor and instrumentation (stationary or mobile; sensitivity, type of measurement, range, accuracy, and precision);
    - type and location of local and remote alarms, annunciators, readouts and recorders;
    - alarm set points;
    - provision of emergency power supplies;
    - requirements for calibration, testing and maintenance; and
    - automatic actions initiated or taken.

According to IAEA SS-35-G1 A.1229-A.1232, The information on all levels of radioactive waste should be added in the SAR.

The criteria and safety principles adopted for the facility must be incorporated in the SAR.

It is recommended to perform periodic measurements of the control rod worth in order to ensure that there is no significant variation due to neutron absorption.

The burn-up limit for unloading of the core drive fuel elements should be determined, justified and implemented The limit value should be integrated in the OLCs.

The measurement of the reactor power and associated uncertainties as well as the calibration of measuring neutron channels should be described in detail and justified in the SAR.

The chapter on Safety Analysis in the SAR should be completed by the following:

- Comprehensive list of Postulated Initiating Events including human errors;
- External events, loss of flow, start-up accident and loss of support systems such as loss of compressed air ;
- Analysis of the risk of rupture of the structure supporting the core;
- LOCA analysis considering the complete melting of the core;
- Reactivity insertion accidents starting at the minimum initial reactor power.

Operational Limits and Conditions (OLCs)

The Operating Organization should prepare a set of OLCs which could either be included in the SAR or presented in a separate document. On the basis of the Safety Guide 35-G1, these OLCs should include:

- Safety limits;
- Safety system settings;
- Limiting conditions for safe operation;
- Surveillance requirements;
- Administrative requirements.

Each of these OLCs should be justified or be used as condition for the execution of the safety analysis.

The requirements (frequency and acceptance limits) for the control rod drop time measurements should be included in the OLCs.

Regulatory Supervision

To enhance the supervision of the safety of the HBWR, the NRPA should develop an inspection programme and increase the number of inspections at the Halden site. This programme should also be established for other nuclear facilities.

- The NRPA should develop Regulatory Guides on Safety of Research Reactors in line with the IAEA Safety Standards.

Safety Committee

- To improve the independency of the Safety Committee from the Operating Organization, it is recommended to add and formally nominate external experts in the Safety Committee advising the IFE Director or to establish an independent group of external experts to supervise and assess the Safety Committee work on important safety matters.
- The Operating Organization should establish an internal advisory group with clearly defined terms of reference to advise the Reactor Manager on safety aspects. The expertise of this group should cover the design, operation, modification and utilization of the facility including new experiments.
- The Safety Committee should periodically review:
  - The operational and safety performance of the facility;
  - Reports on routine releases of radioactive material to the environment;
  - Reports on radiation doses to the personnel and the public.

Operating Organisation

- To reinforce the independence of the Health, Physics function, which is currently under the General Manager of Halden, it is recommended to assign this function under the supervision of the Director of IFE (like in the case of the Kjeller site), or under the direct supervision of the Head of Safety, Security and Quality Management Department.

Utilization and Experiments

- The review and the licensing of experiments and associated experimental devices should be improved by defining and implementing a clear licensing process involving the Regulatory Body and supported by competent external experts.
- In the SAR a list of requirements and limitations related to the experimental utilization is to be included. This list should contain the enveloping values of parameters important to safety and should facilitate the review process of the feasibility of the foreseen experiments.

Training and Qualification

- The Regulatory Body and the Operating Organisation should establish a clear procedure for the authorization of Operating Personnel.
- The existing training and retraining requirements for all Operating Personnel need to be formalized.

Ageing Management

- The Operating Organisation should establish a comprehensive ageing management programme for all systems, structures and components which are either important for safety or which could influence the long term availability of the facility. This programme should integrate the existing ageing programme for electrical systems, structures and components.

Maintenance and Periodic Testing

- The operating organisation should establish for all items important to safety a comprehensive maintenance and inspection programme defining the maintenance requirements, the frequency of inspections and associated acceptance criteria.

Radiation Protection Programme

- The Radiation Protection Programme at of the Halden site should be improved in accordance with NS-R-4 (para.7.97) and BSS No. 115.

- The radiation protection zoning and barriers should be reviewed and upgraded in the light of the BSS and international good practices.

Radioactive Waste Management

- The waste management programme at Halden should formally define the person responsible for waste management. Periodic retraining should be done to all staff involved in handling of radioactive waste. Radioactive waste categorization should be clarified and included in an appropriate document for handling and storage of radioactive waste.
- The procedures on handling of liquid effluents should clearly state that mixing of different levels of waste is not allowed.
- The estimation, handling and disposal of waste generated by the experiments and experimental devices should be incorporated in the experiment reports; the compatibility to handle the waste originating from experiments with existing waste management procedures should be assessed.

Emergency Preparedness

- On the basis of the next updating of the Safety Analysis Report, the on-site and off-site emergency preparedness plans should be revised to integrate the conclusions of the analysis concerning in particular the Design and Beyond Design Basis Accidents.
- The classification based on the severity of the accidents and the conditions for starting and termination of the emergency situation should be included in the Emergency Plan.

Decommissioning

- According to IAEA SS-35-G1, para.A.19, the SAR should include a chapter on the decommissioning of the facility. This chapter could be a summary of the existing decommissioning plan.

Fire analysis

- The Fire Analysis should be updated as soon as possible to take into account the present conditions of the facility. The improvements derived from this analysis should be implemented on the basis of the defence in depth principles (compartments, barriers, detectors, etc.). The fire analyses and fire protection provisions should be integrated in the SAR
- The Operating Organisation should improve the housekeeping with the aim to minimize the combustible loads in order to limit the fire risk and propagation.

Environment

- The SAR should be completed with part II of the Impact Analysis. According to IAEA SS 35 G1, para.205 one of the ways in which the operating organization demonstrates that it has achieved adequate safety is through the information normally incorporated in the SAR.

Vessel Life Time

- Due to the age of the reactor vessel, the embrittlement under irradiation and the different sources of uncertainties, the operator should improve as much as possible the prevention provisions to assure the integrity of the reactor vessel. In this regard, the operator should:
    - continue to improve the knowledge about the behaviour of the reactor vessel material under irradiation;
    - increase the frequency of controls and In Service Inspections in such a manner that 50% of the welds be controlled every three years and the totality of the welds controlled every six years instead of nine years currently applied;
    - extend the control area of the reactor vessel to include a larger part of the base material around the heat affected zone.

  These improvements should be implemented in order to have more confidence and guaranty for the conservation of the mechanical integrity of the vessel. Nevertheless, in the framework of the defence in depth principle, the consequences of a sudden rupture of the reactor vessel should be evaluated in order to verify that this event is not more severe than those already analysed in the SAR.

Good Practices

The good practices observed by the review team are summarized as follows:

- Implementation of a policy to ensure the transfer of knowledge from experienced staff at key positions by early replacement and job overlap of 3, 6 or even 12 months.

- Work orders for corrective maintenance are issued in a structured way. All sections (maintenance, operations and health physics, etc.) are involved in the process of preparation, isolation, and execution and restoring of the systems maintenance. Safety risks and related precautions are clearly identified.

- A common meeting is held each morning within the Operating Organization with the participation of the operating group, the maintenance group and the radiation protection group.

- Weekly reports, including radiation protection issues, are distributed internally within the site and externally to the NRPA.

- The transportation of radioactive materials is performed in accordance with international regulations. The transport casks are licensed and each transport is authorized.

- Timely development of the decommissioning plan.

- Presentation of the results of the Impact Analysis to the authorities, community and local population which had a chance to put forward their views and their comments.

APPENDIX 1: WALK-THROUGH FACILITY

1. General

During the morning of the fist day at Halden, 19 June 2007, the review team performed a walk-through of the facility. During this visit several members of the counterpart participated giving dedicated and specific explanations of the lay-out, status of the reactor and on-going activities.

The members of the counterpart present were:

- Chief of operation/Division head;
- Head of maintenance and installation section;
- Senior reactor chemists;
- Deputy head of radiation protection division.

Since the facility was shut down for core reloading and maintenance several activities (repair, replacement and upgrading) were ongoing. It should be noted that during the reactor operation the hall is inaccessible. After the walk-through of the reactor hall, the team visited the control room, where the counterpart explained the lay-out of the control and safety systems, the general design philosophy, and the separation between reactor operation and experimental activities.

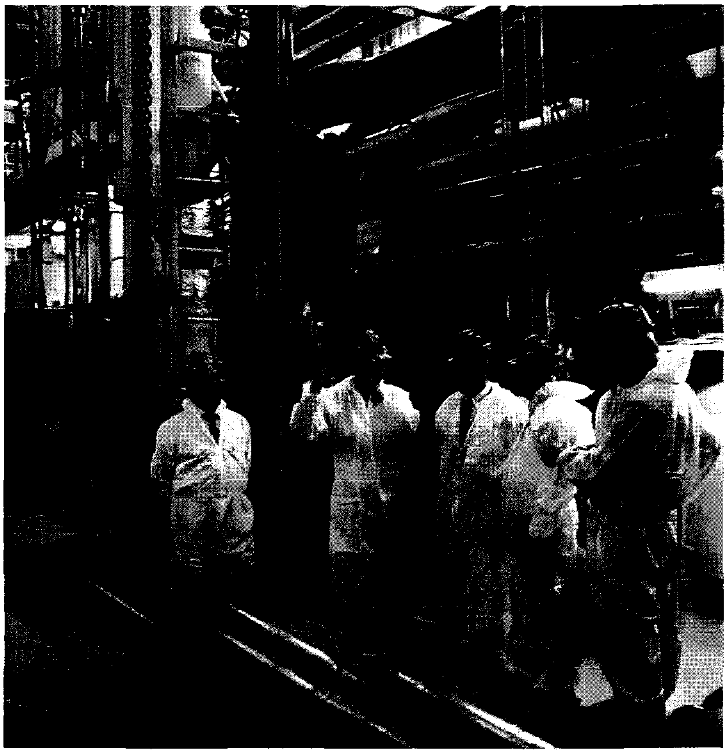


FIG. 6. Photo during walk-through facility.

Finally, two team members visited the radioactive waste storage facility on Thursday, 21 June 2007 in the framework of the review concerning the radioactive waste handling and storage. Details of the findings during the different parts of the walk-through are summarized in the following sections.

2. Reactor Hall

After the fulfilment of the required radiation protection measures the reactor hall was visited. Due to the shut down of the reactor a lot of maintenance activities were ongoing such as work on the primary and secondary systems, I&C-systems and electrical systems. Also the handling of driver fuel elements in preparation of the next core loading had been scheduled. This activity was temporarily interrupted due to the visit since internal procedures do not allow the handling of fuel during visits or other activities.

During the visit of the reactor hall the following areas were visited:

- Entrance tunnel with radiation protection equipment;
- Ground floor with among other the reactor top flange and fuel handling and storage facilities;
- Basement levels where several experimental devices and structures, systems and components were being maintained.

After the execution of the check on possible contamination, the walk-through of the facility was continued with a visit of the electrical supply and emergency systems.

3. Electrical Supply Systems

The visit of the electrical supply systems was conducted in presence of the responsible staff member for electrical installations of the HBWR. The visit was executed in a logical order. The normal supply route of electrical power to the facility was followed i.e.

- Supply of power from the external grid through the paper factory;
- Three main transformers where 500 kV is transformed to 400 V and 230 V;
- Two sets of Uninterrupted Power Supply (UPS);
- Three sets of battery packs with room ventilation, but no Hydrogen monitoring;
- Diesel generator room where one major diesel and one small (back-up) diesel are located.

4. HBWR control room

The visit to the electrical supply system was immediately followed by a visit to the control room. During this visit, explanations of the control room lay-out, the use of PLC-systems and data logging for both experiments and operational data, interlock features and warning systems were given. Also the required shift constitution (specialization and quantity) of the operators at different operational states was addressed.

5. Waste storage

On 21 June 2007 two team members, accompanied by a senior radiation protection staff, visited the waste storage area. In this area both low active waste and intermediate active waste are stored. If needed, additional shielding is applied to minimize the possible dose to the staff.

6. Findings

After the walk-through the individual impressions, observations and findings of the team members were discussed. This section provides a condensed summary of these findings and has been used during the detailed discussions between the INSARR-team members and the HBWR-counterpart.

6.1. Fire protection

Remarks on the fire protection strategy and implementation are summarized as follows:

- Insufficient fire detectors installed;
- Only limited compartment applied;
- No fire barriers in cable trays;
- No fire detectors associated with the cable trays;
- No programme and no strategy to reduce combustible loads;
- Housekeeping with respect to fire reduction needs to be improved, examples are:
    - Combustible load due to use and storage of paper, wood;
    - Storage of oil in glass bottles in reactor hall.
- Overloaded cable trays without physical separation;
- Limited measures to prevent segregation;
- Limited use of fire retardant cabling;
- Cutting and welding with gas bottles stored at different elevations;
- Inspection dates of fire distinguishers expired.

The fire protection and prevention programme was discussed in a dedicated session; issues can be found in Annex 2, Section 18 (appendix p. 87 (FIR-01). However, there was consensus between the team members that the fire analyses should be updated and more attention to the fire protection programme by the HBWR-staff was needed. In addition the team stressed the need for the implementation of practical modification to improve the prevention and protection against fire.

6.2. Housekeeping

Several observations and findings related to housekeeping were made by the team members, which are summarized as follows:

- Insufficient separation between different zones;
- Overload of equipment and tools in several areas;
- Too much paper, cloths and other combustible materials stored in areas important to safety;
- Removal of obsolete equipment and tools and parts was needed;
- Leakage found on grounds and walls not properly cleaned;
- Painting and repair of walls needed after facility modifications.

The team members agreed that, despite the operational state of the facility conditions ("Shutdown"). Important attention to housekeeping has to be given by the HBWR staff and Operating Organization.

6.3. Radiation Protection

Although the important aspects of radiation inspection/health physics is addressed during dedicated sessions of the INSARR in this section the main findings of the facility walk-through are presented:

- Radiation zoning is insufficiently implemented;
- No entrance signs on radiation /contamination levels,
- No warning signs at doors of compartments;
- Alarming by increased radiation levels in reactor hall is achieved by a gamma monitor close to the rotating lid above the reactor tank;
- No signs indicating the measurement values (radiation/contamination) were updated since one week after reactor shutdown;
- Contaminated water stored in the reactor hall without leak detector nor cover on the top;
- Coloured tags not consistently applied;
- Clothing procedure (white coat or coverall) not used by staff member of the Operating Organization;
- Location of lockers in the tunnel in relation to contamination measurement equipment is wrong;
- Easy by-pass possibility of the mandatory contamination check.

APPENDIX 2: ISSUE PAGES

ISSUE SAR-01: DESIGN REQUIREMENTS OF REACTOR COOLANT SYSTEMS

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, Section A.14

[3] SAR Part I – Chapter 5 – Heavy Water Circuits

[4] SAR Part I – Chapter 6 - Light water circuits;

ISSUE CLARIFICATION

A.601. This chapter of the SAR shall provide a description of the reactor coolant systems which transfer the heat from the reactor to the ultimate heat sink. The description shall contain the main design and performance characteristics. It shall be supported by schematic flow diagrams and an elevation drawing.

OBSERVATIONS

The water-cooling systems of Halden reactor include tree circuits: a primary heavy water circuit, a secondary light water circuit and a tertiary light circuit, and a separate shielding water circuit.

Chapter 5 of SAR part I includes the description of heavy water circuits covering: closed primary circuit, steam circuit, sub cooling circuit, pressure release system, D_2O purification circuit and heavy water handling, D_2O recombination circuits, water chemistry circuits, emergency core cooling system and the gas collecting system.

Chapter 6 of SAR part I includes the description of light water circuits: closed secondary circuit, feed water circuit, steam output circuit, raw water and coolant circuit, water treatment circuits, and the separate shield circuit.

Those chapters do not address the design requirements for each water system, the design and performance characteristics of the main components.

The flow and instrumentation diagrams are included. The drawings of main components are not included.

All the drawings from SAR part I, are not updated. There are working flowcharts, drawings that are updated continuously and properly reviewed, authorized and released according to a distribution list.

The flowcharts and drawings are called “safety standards”. According to the counterpart comment, this term is used due to historical reason.

From discussion with operating group resulted that there is a drainage system for heavy water leakage from primary circuit. This system is not described in the SAR.

The pipes of primary circuit were changed few years ago. In addition, valves from this circuit were changed with a new type which according to the counterpart some of those new valves do not require leakage detectors. Those design modification data are specified in the SAR

There are two steam transformers and one heat exchanger in the primary circuit. The initially design specifications required a capacity of totally 25 MW heat removal. During in-service-inspection activities, several pipes of two-steam transformers were plugged due to non-conformity of pipe thickness. The heat transfer capacity of those are lower (7 MW instead of 9 MW respectively 8 MW instead of 11 MW. Now the primary circuit could remove maximum 20 MW, but the normal operation is kept at maximum 18 MW.

The operational limits and conditions of water chemistry parameters were not included in the SAR.

In the next two years period it is planned to replace both stream transformers.

The material in the steam circuit for heavy water system follows the ASTM standard. However, there is no similar specification for any other components of cooling systems.

The chemistry data for the primary coolant is not presented, including the effects of irradiation of the primary coolant.

The reactor vessel is surrounded by two closed water-shielding circuits for cooling the concrete. The water level is measured on-line in order to detect any eventually leakage

In chapter 5 of the SAR part III the list of facility regulations is inconsistent with the information given in appendix A. While in appendix A PR-5 (“Emergency regulations") and PR-7 ("Transport of radioactive materials") are used as references they do not appear in the list of facility regulations.

The term "Safety Standards" in SAR part III chapter 6 and appendix B is confusing. Safety Standards are the IAEA documents to be implemented by the Operating Organization. The term "Principle design lay-out" is suggested.

POSSIBLE SAFETY CONSEQUENCES

If the information in SAR is not updated or accurate, it may be a source of wrong data that could induce errors in a quick assessment in case of an emergency.

Modifications performed without comparison with design requirements may lead to installation of inadequate equipment, which may affect the safety of the installation.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The operating group does not use the information from the SAR.

RECOMMENDATION

R1: The IFE should review and update the information in the SAR for the reactor cooling systems. According to the IAEA SS 35 G1, A.602 the SAR shall describe in detail the design and the operation of the primary cooling system. The design and performance characteristics of the main components (pumps, valves, heat exchangers, piping) should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials of which the components are made and the effects of irradiation on these materials shall be specified. The primary vessel, together with in service environmental factors such as corrosion, fatigue and thermal stress cycling shall be described.

COMMENT

C1: The term "safety standard" is used by the Operating Organization as a term for flowcharts and drawings. This is not properly used according to the international term of "safety standard".

SUGGESTION

S1: It is suggested to the Operating Organization to adopt the international technical language especially for safety standards.

ISSUE SAR-02: ADDITIONAL INFORMATION NEEDED ON RADIATION PROTECTION

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA Safety Standard No.115 International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, 1996

[3] IAEA Safety Requirement NS-R-4, Safety of Research Reactors, 2005

[4] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report

[5] SAR – Part IV Chapter 5 – Radiation protection and activity release

ISSUE CLARIFICATION

The purpose of the chapter on operational radiological safety is to provide information, as appropriate, on radiation protection policy of the organization; overall radiation protection programme:

- quantitative account of sources of radiation at the facility;
- facility design for radiological safety; handling and movement of radioactive materials;
- dose assessment for normal operation; procedures and training; facilities, equipment, and instrumentation;
- environmental monitoring;
- access control and zoning;
- shielding;
- ventilation for radiological control;
- area and effluent radiation monitoring;
- solid, liquid and gaseous waste; and
- anticipated direct radiation exposures within the facility.

OBSERVATIONS

The Chapter 5 of the SAR (Part IV) on Radiation protection and activity release addresses radiation exposure, air contamination within the research reactor facility, personnel monitoring and doses, activity release to the atmosphere, volatile fission products (release due to sudden cladding failure, release during continuous operation with fuel failure), argon, activity release to the water. One brief paragraph deals with the environmental monitoring.

The annual effective dose limit for maintenance and loading staff at the Halden reactor is 100 mSv in a five year period. Not exceeding 50 mSv in one year. For all other staff the national limit of 20 mSv/year applies.

The legal dose limits for public are not specified in this chapter. The Organization, staffing and responsibilities are not addressed.

The facility, equipment and instrumentation are also not addressed.

An overview of written procedures for the radiological protection programme is not provided.

The provisions for controlling the conduct of the radiation protection programme and its review are not addressed in the SAR.

The facility design for radiological safety is also not addressed. The description of radiation sources at the facility is not included. However, a separate document containing the list of radiation sources was provided by the counterpart.

According to the discussion with the counterpart the shielding calculations are done with MIRCROSHIELD computer code. This information should be included in the SAR.

The information on radiation protection procedures, measurements, monitors, organization, staffing and responsibilities for radiation protection, procedures for training in radiation protection, radiation protection policy, is described in different internal documents of Halden Reactors, e.g. in the top level quality documentation of the division QA-013-no.

Training is not implemented for in-service maintenance and repair staff. The Institute is planning to do that in the near future.

Permission to release radioactivity to the effluents is given by the National Radiological Protection Authority. The limiting criterion is the potential dose to the members of the public. Because of their living habits, they are likely to receive the maximum dose from the releases.

Annual reports are sent to the authorities, presenting radiation doses, activity releases and the results of the environmental monitoring programme.

POSSIBLE SAFETY CONSEQUENCES

Incomplete information on operational radiological safety may lead to insufficient radiation monitoring of workers and as a consequence to overexposure.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The counterpart agreed with the recommendations. The counterpart agreed that there is a need to retrieve the information on design for radiation safety.

RECOMMENDATION

R1: The chapter on Radiation protection and activity release should be updated in accordance to the IAEA SS-35-G1, para.1201-A.1241, describing for normal operation conditions:

1. The radiation protection programme
2. Sources of radiation at the facility;
3. Facility design for radiological safety;
4. Waste management system;

5. Dose assessment for normal operation.

R2: The section on facility design for radiological safety should include a description on how the implemented radiological provisions (e.g. zoning, shielding, radiation monitoring, etc.) reduce exposure to personnel, minimize the undesired production of radioactive material, reduce the time spent for maintenance and operational activities in which the possibility exists of internal or external exposure, and maintain releases of radioactive material to the environment as low as reasonably achievable.

This section should describe the permanent radiation areas, effluent and airborne radiation monitoring systems and should include in particular the following information:

- location of monitors and detectors;
- type of monitor and instrumentation (stationary or mobile; sensitivity, type of measurement, range, accuracy, and precision);
- type and location of local and remote alarms, annunciators, readouts and recorders;
- alarm set points;
- provision of emergency power supplies;
- requirements for calibration, testing and maintenance; and
- automatic actions initiated or taken.

SUGGESTION

S1: This chapter of the SAR shall contain a section on the conclusion regarding the acceptability of the operational radiological safety programme and design features of the facility.

ISSUE SAR-03: INCOMPLETE DESCRIPTION OF THE RADWASTE SYSTEM IN THE SAR

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA NS-R-4 Safety of Research Reactors, 2005

[3] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report.

[4] SAR Part I – Chapter 13 Radioactive waste - handling and disposal.

ISSUE CLARIFICATION

According to NS-R-4, paragraphs 7.104-7.107, the reactor and its experimental devices shall be operated to minimize the production of radioactive waste of all kinds, to ensure that releases of radioactive material to the environment are kept as low as reasonably achievable and to facilitate the handling and disposal of waste. Written procedures shall be followed for the handling, collection, processing, storage and disposal of radioactive waste. An appropriate record shall be kept of the quantities, types and characteristics of the radioactive waste stored and disposed of or removed from the reactor site.

OBSERVATIONS

SAR Part I, Chapter 13 provide information on radioactive waste - handling and disposal;

From the discussion with the counterpart resulted that off-gas system is under upgrading process. The gas pipes from the heavy water system and the pipes from the light water system will be separated in the new upgraded system. This will allow tritium recovering from gaseous phase and recirculation back into the heavy water inventory. There was a working group designing the new off-gas system. According to the counterpart there are used the same design data (pipe diameter, material, etc.) as the existing off-gas system.

The efficiency of filters is not controlled. According to the counterpart, the filters are changed every 2.25 year without checking the efficiency of new filters.

However there is a system at Kjeller to control the quality of the HEPA filters and charcoal filters. This system was used in the past for checking the filters that were used at Halden reactor.

In the off-gas system there are two monitors, one measures tritium and noble gases and the other measures only noble gases. According to the two measurements, it can be assessed if the airborne effluent contains tritium or noble gases.

The gas monitors are used only for detecting, but not for measuring, so they are not regularly calibrated, only a functioning check is done every half a year.

Liquid waste

There are 3 delay tanks physically separated. Normally the liquid waste goes to one of those tanks and after that to the release point and out to the Tista river. If the liquid waste contains high activity it is directed through the other tanks to purification system and when the measurements shows that becomes "clean" it is released through the delay tank

POSSIBLE SAFETY CONSEQUENCES

If there is no clear summarized information on the radioactive effluents, this may lead to inaccurate assessment of dose to the critical group.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The counterpart agreed to include the additional information in the SAR.

RECOMMENDATION

R1: According to IAEA SS 35-G1 A.1229-A.1232. The information on all levels of radioactive waste should be added in the SAR.

Solid waste

(a) description of types of waste, class, the sources and quantity of solid wastes, including physical form, volume, isotopic compositions and measured or estimated activity;
(b) method of collection, processing, packaging, storage and shipment.

Liquid waste

(a) description of types and quantity of liquid wastes, their sources, location, form and estimated activity;
(b) process equipment, storage tanks and release points to the environment;
(c) measures to separate radioactive and non-radioactive effluents;
(d) release goals; and
(e) requirements for system capacity, redundancy, flexibility and the capability required to facilitate maintenance, reduce leakage and prevent uncontrolled releases to the environment.

The criteria for determining whether processed liquid wastes will be recycled or discharged shall be described, including the expected effluent concentrations by radionuclide and the total annual release to the environment. Identify the dilution factors considered upon release.

Gaseous waste

(a) description of the types and quantities of gaseous wastes and the sources, location, form and calculated radionuclide quantities;

(b) process equipment and release points to the environment;
(c) measures to separate radioactive and non-radioactive effluents;
(d) release goals; and
(e) requirements for system capacity, redundancy, flexibility and the capabilities to facilitate maintenance, reduce leakage, and prevent uncontrolled releases to the environment.

Design provisions to handle gaseous material with a potential for explosion should be described.

ISSUE SAR-04: CONTAINMENT AND CONFINMENT

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA Safety Standard No.115 International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, 1996

[3] IAEA Safety Requirement NS-R-4, Safety of Research Reactors, 2005

[4] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report

[5] SAR Part I – Chapter 7 Containment

ISSUE CLARIFICATION

The confinement or containment of research reactors should:

1. be capable of withstanding extreme loading from accident events including those arising from all postulated internal and external events considered in the reactor safety analysis;
2. provide proper margins for the highest calculated pressure and temperature loads expected during design basis accident conditions;
3. provide suitable means to control the release of radioactive materials during design basis accident conditions; and
4. have a degree of leak tightness commensurate with the requirements of the reactor safety analysis.

In addition, an initial and periodic leak tests, routine testing, and filter replacement should be performed.

OBSERVATIONS

The tests performed are useful to know that the value of leak pathways is similar to the values obtained in previous test.

A variation of test parameters such as ambient pressure, temperature and humidity could affect the result of the leak test. For this reason it is necessary to justify the quantity and position of detectors used during the execution of the test.

The leak test is made at a static pressure of 1.3 bars which is lower than the calculated pressure for the reference accident (1.7 bars). In the SAR there is no reference to the validation of the model used to extrapolate the measured leak rate to the leak rate given in the reference accident calculations.

The calculated value for the maximum static overpressure during the reference accident is 1.7

bars. In the SAR there is no justification how this value was obtained.

It is necessary to provide operational limits for RA1 and RA2 gamma monitors. There are set point values for those monitors in count per second but not in engineering units (dose rate). There is no justification for the choice of these setting points.

It is necessary to provide clear criteria for changing filters (activity and measured pressure difference).

POSSIBLE SAFETY CONSEQUENCES

If the leak rate of the containment is not well known, during abnormal occurrence, the public exposure could be different from those estimated by calculations.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The Operating Organization has no comments on this issue.

COMMENT

C1: A justification of the number and position of detectors used during a leak test should be presented.

C2: The validation of the model used to extrapolate the measured leak rate to the leak rate given in the reference accident calculations should be justified.

C3: The justification in the SAR of the 1.7 bar for the maximum static overpressure should be given.

SUGGESTION

S1: The operational limit values for RA1 and RA2 gamma monitors should be defined in engineering units and justified.

ISSUE SAR-05: SAFETY ANALYSIS REPORT

Lack of safety philosophy and implementation

BASIS AND REFERENCES

[1] IAEA guideline NS-R-4;

[2] IAEA safety guide 35-G1

[3] SAR part I, II, III and IV.

ISSUE CLARIFICATION

The SAR should reflect the criteria and safety philosophy of the design, operation and utilization of the RR. This should amongst others be based on:

- Defence in depth principle;
- Common cause/common mode failures;
- Single failure proof;
- Redundancy and diversity;
- Physical separation.

The presence and implementation of a safety philosophy and related implementation are not described.

OBSERVATIONS

In the current design of the RR there is a lack of redundancy, diversity, physical separation, etc. The possibilities for common cause and common mode failures are presented.

In particular there is no clear segregation between the protection and control systems for the signals of the experimental devices.

POSSIBLE SAFETY CONSEQUENCES

The lack of clearly defined safety philosophy and implementation could lead to unsafe situations.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The safety philosophy and the implementation of the philosophy are included in the HSE strategy, HSE-manual and administrative regulations. The philosophy and its implementation will be included in the SAR and the SAR will be revised with the intention to have a clearer link between steering documents and the Presidents' level and other documents.

RECOMMENDATION

R1: The criteria and safety principles adopted for the facility must be incorporated in the SAR.

ISSUE SAR-06: SAFETY ANALYSIS REPORT, PART II – CORE PHYSICS

BASIS AND REFERENCES

[1] IAEA Safety Requirement NS-R-4

[2] SAR Part II, Nuclear Core, Core Reactivity Characteristics and Codes for core physics calculations.

ISSUE CLARIFICATION

An analysis shall be provided which shows that the nuclear conditions in the core physics are acceptable throughout the anticipated core cycle. The analysis shall include basic design on the nuclear design, reference to the calculation methods and codes, experimental verification of the basic input data, or other information that can support the validity of the nuclear properties, control rods characteristics, reactor stability, and thermal and hydraulic characteristics.

An analysis shall be provided which shows that the effectiveness, speed of action and shutdown margin of the reactor shutdown system are acceptable and that a single failure in the shutdown system will not prevent the system from completing its safety functions when required.

Information shall be provided to prove that, during operational states, adequate core cooling capacity will be available to keep the reactor fuel in a thermally safe condition and that an adequate thermal safety margin will be available to prevent or minimize fuel damage in accident conditions.

OBSERVATIONS

Validation of codes was made by comparison between calculation results with experimental data during the commissioning of the reactor and after some special measurements and by comparison between different codes used. Some codes (HELIOS, WIMS, MNCP) are internationally very well known for enriched uranium-heavy water core calculations. The comparison with experimental data of control rods worth was not made for a long time.

The different neutronic calculations are made by three users with adequate experience.

About reactivity factors, the calculated power coefficient for reactivity was compared with operational data. Fuel, moderator and coolant temperature reactivity coefficients are in general calculated values.

The responsibilities of the physicist group consist in:

- verifying thermo-hydraulics safety conditions (burnout- fuel centre melting limit); and
- reactivity conditions:
  - safety: maximum control rod worth, burn-up of control rods and shutdown reactivity, criticality calculations for transportation and HBWR fuel storage;
  - operation: determination of the minimum reactivity needed and implementation

of the experimental device conditions.

- Recommending new configurations, modification of control rod positions during operation due to testing requirements, and replacement of control rods.

The tasks performed by the group are the following:

- Neutronic calculations during operation: following control rod critical positions, drive fuel element burn-up and reactor vessel and coupon neutron fluences.
- Interaction with the users to discuss safety issues and test requirements.
- Verification by calculation that the requirements for experimental devices are compatible with the specified occupational dose and release limits.

All the above mentioned tasks are made following adequate internal procedures.

Conditions for thermal-hydraulic margins, shutdown margin and stuck rod criteria are fulfilled.

1. Control rod characteristics

Calculation of individual control rod worth is made in a configuration with all control rods withdrawn and at normal temperature and power conditions (240 °C and 20 MW).

In case of control rod ejection the calculation showed that the consequences will be limited if the worth of the control rod is less than 1.8%. However, the worth of some control rods is higher than this value and the ejection of such control rod may lead to significant consequences. For such control rod there are administrative rules to limit its insertion and consequently to limit the reactivity addition in case of its ejection. The limitations of the control rod position are surveilled and trigger a visual alarm in case of exceeding the set value.

The maximum reactivity excess (from cold core without Xe-135 and considering 100 days of operation plus 30 days more in order to take into account the uncertainties) is about 10%. The shutdown reactivity is about 25% in cold state and 37% in hot state.

The safety criterion of at least 5% of reactivity shutdown is verified as well as the stuck rod criterion.

The material of control rods is 30% Cd and 70% Ag. The silver is used to both improve mechanical properties and to maintain a basic absorption in more burnt parts of a control rod. The burnup of control rods absorber material is continuously calculated. The result is that most of the control rods have burnup only in the lower part and the effect on their worth is small.

2. Reactor Stability

An oscillation frequency in the range of 0.005 Hz to 0.009 Hz was observed at certain conditions during operation with the second fuel charge. The amplitude increases with increasing power, pressure, moderator level and decreasing subcooling. As a consequence, the system of introduction of subcooled water to the reactor was modified in 1967 before starting the third charge operation. The modification essentially implied that subcooled water could be supplied to the side reflector from tubes extending down through the reflector. It was expected that the subcooled water supplied to the side reflector would be sucked into the core region, reducing the moderator void and thereby having a stabilizing effect on the dynamic behaviour of the reactor. Operational experience and experimental evidence from the operation of the reactor with the third and fourth fuel charges confirmed the stabilizing effect of introducing subcooled water to the side reflector.

3. Thermal and hydraulic characteristics

The power of the individual fuel assemblies is calculated prior to start-up of each core loading. Uncertainties are 5% for measured values and 10% for calculated ones.

Conditions for thermo-hydraulics safety conditions for all fuel elements (burnout- fuel centre melting limit) were fulfilled.

There is no information in the SAR related to reactor power measurement, its uncertainties as well as power calibration of neutron detectors.

Tables 7.2, 7.3 and 7.4 of SAR Part II, should be clarified in order to distinguish between safety limits and operation values.

POSSIBLE SAFETY CONSEQUENCES

The nuclear safety of the reactor could be affected if neutronic and thermal-hydraulic design calculations are not accurate. This could lead to a situation where safety requirements are not fulfilled.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The Operating Organization will implement the recommendations.

RECOMMENDATION

R1: It is recommended to perform periodic measurements of the control rod worth in order to ensure that there is no significant variation due to neutron absorption.

The burn-up limit for unloading of the core drive fuel elements should be determined, justified and implemented The limit value should be integrated in the OLC's.

The measurement of the reactor power and associated uncertainties as well as the calibration of measuring neutron channels should be described in detail and justified in the SAR.

ISSUE SAN-01: SAFETY ANALYSIS

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA Safety Standard No.115 International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, 1996

[3] IAEA Safety Requirement NS-R-4, Safety of Research Reactors, 2005

[4] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report

[5] SAR – Part IV, Chapter 6 Incidents

[6] SAR – Part IV, Chapter 7 Accidents

ISSUE CLARIFICATION

In this chapter, the effects of anticipated process disturbances and postulated component failures and human errors (postulated initiating events) shall be described, including their consequences to evaluate the capability of the reactor to control or accommodate such situations and failures.

To ensure completeness of presentation and to facilitate the review and assessment by the regulatory body, this chapter of the SAR should include the following:

- General approach and methods used in the safety analyses; consideration should be given to a brief summary under the following headings:
  A. Methods of identification and selection of initiating events.
  B. Methods of analysis, including where appropriate: (a) event sequence analysis; (b) transient analysis; (c) evaluation of external events and special internal events; (d) radiological consequence analysis.
  C. Acceptance criteria.
- Selection of Initiating Events: The spectrum of accident initiating events considered in the analyses. The list shall be comprehensive and justification for rejection of particular initiating events shall be provided.
- Summary of significant results and conclusions regarding acceptability, including a brief description of the dominant accident sequences. A comparison of the results of the analyses against appropriate acceptance criteria shall be given. An evaluation should be presented to demonstrate that the design is acceptable and to confirm the validity of the operational limits and conditions.

OBSERVATIONS

The SAR – Part IV, Chapter 6 deals with Incidents and chapter 7 deals with Accidents.

Selection of postulated initiating events (PIEs) should be justified. For those events that are not taken into consideration a qualitative justification should be provided in the SAR.

From the list of PIEs recommended in Appendix to the IAEA NS-R-4, Human errors, External Events (e.g. the effects of toxic materials, fire or explosion from Paper Plant), loss of flow (LOFA), start-up accidents, loss of support systems (e.g. compressed air system, rupture of structures supporting the reactor core) are not addressed. Other events, specific to Halden reactor are address in the SAR.

Under the chapter on Incidents are qualitatively addressed: Control Rod Withdrawal; Sudden Introduction of Subcooling; Sudden Introduction of Cold Water; Sudden Introduction of Light Water into the Moderator; Other Reactivity Incidents (unscheduled stopping of the subcooler pump, PA 1; faulty operation of the purification circuit; and unscheduled starting of the hot well pump, PB 8 or PB 14); Failure of Heat Removal Circuits; Experimental Equipment Failure (Light Water Forced Circulation System; Fuel Rod Gas Flow Control System; Helium-3 Flux Control System; Ultra High Gas Pressurizing System); Computer Failure; Fire (Fire in the Control Room or Control Room Basement; Fire in Cable Gates; Fire in the Reactor Hall/Reactor Tunnel; Fire in the Fuel Storage Building); Loss of Electrical Power (Short Circuits; Earth Faults; Consequences in Case of Failure in Electrical Power Distribution System).

Under the chapter on Accidents are analyzed the Loss of Coolant Accident; Control Rod Ejection; Fuel Handling Accident; Post LOCA clean-up.

During discussion with the counterpart, it was recommended to include the analysis of the BDBA accident corresponding to pressure vessel rupture.

Under the section on Control Rod Withdrawal, it should be clarified if the maximum reactivity insertion of 85 pcm/sec covers any core configuration. A chart of fuel temperature evolution during this incident should be provided to justify the statement that "no hazardous overheating of the fuel will occur" especially if the accident occurs during the start-up.

The section on Sudden Introduction of Light Water into the Moderator should be clearly separated in two different incidents:

1. Introducing water from the subcooler to the moderator; and
2. Sudden introduction of water from the emergency core cooling system.

Section on Failure of Heat Removal Circuits should include LOCA as well LOFA analysis.

During discussion with the counterpart, it was emphasized that the tertiary water system is used to control the reactor pressure. The loss of this system could lead to pressure increase in the reactor vessel. Taking into account that there is only one-release safety valve, this may fail to operate in such situation. This may lead to high stress on reactor pressure vessel and affect rupture of the pressure vessel. To avoid such situation it is recommended to install one additional safety release valve as a redundancy of the existing equipment (this issue is addressed in another part of this report).

In the LOCA analysis it is considered that up to 50% of the core fuel may melt. The list of melting points of core materials (page 7-5) does not include compounds that may form eutectic mixture, which may have lower melting points than those considered in the LOCA analysis. For this and other reasons the choice for the percentage of core fuel melting should be justified. The counterpart indicated that, regardless of the assumed melted percentage of the core, the assessment of the fission product releases from the reactor building was based on the release fraction from the core required by NRC in Regulatory Guide 1.183. This is satisfactory.

The equation (in page 7-3) used to calculate the reaction heat rate between D_2O steam and zirconium should be corrected to be applicable for t=0.

In the accident analysis, after LOCA, it was considered that the control rods may collapse and disappear from their original core positions and the there could be circumstances for a second criticality accident. The conclusion is that after this accident the insertion of water will be feasible and addition of natural boron, can keep the system subcritical. However, if the ECCS is unavailable, it should be shown the way to add boron into the melting core.

From discussion with the counterpart it was mentioned that boron existing at Halden is only available for experiments not for emergency situation in case of LOCA.

Concerning the assessment of activity concentration in reactor hall and air lock during LOCA, it is considered that the charcoal filters will work properly to retain the I2. However, it should be analyzed if the humidity resulting from LOCA will affect the efficiency of the charcoal filters.

About the control rod ejection accident the following consideration was mentioned:

The reactivity (in dollar units) during the transient is calculated by:

$\$(t) = \$cre(t) + \$fdb(t) + \$scr(t)$

Where:

$cre = control rod ejection reactivity (positive value)

$fdb = feedback reactivity (negative value due specially to the increase of the fuel temperature and void formation)

$scr = scram reactivity

The worst condition is $\$(t) = \$cre(t)$ means a very short control rod ejection time, small feedback contribution (low initial power and/or low feedback parameters) and small scram contribution (long electronic delayed, long control rod drop time).

In this situation, the minimum period T is (inhour equation)

$T = (l/\beta)/(\$ - 1) = (44xE\text{-}5/6.5E\text{-}3)/((0.18/0.0065 - 1) \sim 0.04$ s

In the SAR analysis of the accident maximum feedback reactivity was considered (maximum initial power). The reason is that in this condition the flow coolant, as the driven force for control rod ejection, is maximum.

The β value of 0.0065 (Keepin value) should be justified. It corresponds to the nuclear β of U-235. The contribution of the Pu-239 that decreases this value, the difference between effective and nuclear value that increases the β value and others values for nuclear β (0.00685 Tuttle value) should be considered in the justification. The value of the mean life l (44E-5 s) that decreases with the content of light water in heavy water should also be considered.

POSSIBLE SAFETY CONSEQUENCES

Mitigation action should be planned for any accident or incident situation; additional safety barriers could be added to prevent or to mitigate their consequences. For this reason all, the possible consequences of incidents and accidents should be well analyzed. Accidents or incidents which are not addressed in the SAR, but which may occur, could lead to public doses different from those estimated in existing calculations.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The counterparts commented that in their view the pressure vessel rupture might lead to similar consequences as LOCA.

The installation of a second relief safety valve is possible to implement.

Concerning the paragraph on control rod ejection analysis, the calculations using the UK Panther code, do not consider the moderator temperature coefficient and the moderator void coefficient and are thus conservative. Regarding the fuel temperature reactivity coefficient, this is more negative at lower temperature such that the feedback reactivity is not maximum in the analysed situation. The selected situation (high power) is judged to be conservative because
- The driving force for control rod ejection is fully developed
- The possible temperature increase leading to fuel melting is at its minimum
- The fuel temperature reactivity feedback is less efficient than at low power and temperature.

For the paragraph discussing the β value the counterpart commented that the numbers mentioned in the SAR are only for illustration of the issue, but are not those used in the analysis. The RIA analysis is using the code Panther applied in the UK for similar purposes. The beta values and lifetime are calculated based on the actual materials in the core. The burnup of the fuel elements is included in the analysis and thus the effect of Pu on beta value.

RECOMMENDATION

R1: The chapter on Safety Analysis in the SAR should be completed by the following:
- Comprehensive list of Postulated Initiating Events including human errors, external events, loss of flow, start-up accident and loss of support systems such as loss of compressed air ;
- Analysis of the risk of rupture of the structure supporting the core;
- LOCA analysis considering the complete melting of the core;
- Reactivity insertion accidents starting at the minimum initial reactor power.
- The numbers mentioned in the SAR should be those used in the analysis.

SUGGESTION

S1: For the second criticality accident scenarios in case of LOCA it should be assessed the consequences in the case that boron could not be added or actions should be taken by IFE to supply the necessary quantity of boron at Halden for emergency conditions. A technical solution should be found to allow insertion of boron in case of LOCA accident.

S2: The SAR should include all hypotheses based on which the calculations were performed and the input data considered, codes used and its validation for the facility characteristics. The justification of the accident scenarios analysed should be also described in the SAR.

ISSUE OLC-01: OPERATING LIMITS AND CONDITIONS

Lack of comprehensive set of OLCs

BASIS AND REFERENCES

[1] IAEA Safety Guide no. 35-G1 (section A.17)

[2] SAR part I, II, III and IV as well as supporting documents such as operating certificates

ISSUE CLARIFICATION

In section A17 of IAEA Safety Guide no. 35-G1 (A.1701 to A.1708) the requirements and preferred content of a set of Operational Limits and Conditions (OLCs) are given. The OLCs represent an envelope of parameters, developed by the Operating Organization which will protect the reactor, the staff, the general public and the environment from undue exposure if they are not exceeded.

The OLCs shall either be presented in a separate document or included in a chapter of the SAR.

OBSERVATIONS

During several sessions the review team have made the following observations:

- Safety limits, safety settings and other limiting conditions for safe operation are widely spread through the documents supplied. Some of them are included in different chapters of the SAR, Operating certificates or operating procedures while others can only be found in the supporting documents;
- There is no specific chapter on OLCs included in the SAR nor is it condensed in a separate document;
- There is no overall surveillance programme including frequency and scope of periodic tests showing that the performance level set by the conditions for safe operation are met;
- There is no dedicated list of administrative limiting conditions and requirements associated with the operation of the facility;
- The basis of the OLCs applied is not defined; it is not clear if the present OLCs have served as enveloping conditions during the execution of the Safety Analysis, if the OLC is a license condition or if the OLC derives from the safety analysis performed;
- The Operating Organization does not apply the differences between safety limits, safety system settings and limiting conditions for safe operation;
- It is not clear if the limits applied by the Operating Organization are exhaustive and by that if all parameters relevant for the safety of the facility have been addressed; for example pH-values, water conductivity are not measured periodically and are missing in the OLCs;
- A list of enveloping parameters important for the safety of the experimental devices, the interaction with the reactor and their influence on the safe operation of the reactor is

missing.

POSSIBLE SAFETY CONSEQUENCES

Because of the important role of the OLCs in safe operation, a lack of a dedicated set of OLCs could influence the safe operation of the reactor and of the experimental devices. It could also lead to inconsistency between the license requirements, the OLCs and the operating procedures.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The safety limits, safety settings and other limiting conditions are included in the SAR, but widely spread in the SAR. However, some relevant limits, settings and conditions are not in the SAR but in other documents.

IFE will revise the SAR so that OLCs are in one specific chapter or document and make sure that the limits, etc. in the list cover OLCs including surveillance requirements of systems relevant to safety.

RECOMMENDATION

R1: The Operating Organisation should prepare a set of OLC's which could either be included in the SAR or presented in a separate document. On the basis of the Safety Guide 35-G1, these OLC's should include:

- Safety limits;
- Safety system settings;
- Limiting conditions for safe operation;
- Surveillance requirements;
- Administrative requirements.

Each of these OLC's should be justified or be used as condition for the execution of the safety analysis.

COMMENT

C1: The importance of the OLC's for safe operation of the reactor and the experimental devices should be explained and understood by the staff of the Operating Organization.

ISSUE OLC-02: OPERATING LIMITS AND CONDITIONS

Back-up and data storage

BASIS AND REFERENCES

[1] IAEA guideline NS-R-4 (7.84)

[2] SAR part I chapter 9 "Main computer"

ISSUE CLARIFICATION

In chapter 9 of the SAR the main computer system and its applications are described.

OBSERVATIONS

The storage of data and back-up of data are performed in a structured manor; the responsibilities and frequency of making back-ups of the facility data is described in Operating procedures which were provided by the counterpart. The required period for storage of data is not indicated.

POSSIBLE SAFETY CONSEQUENCES

Without a prescribed period to maintain the stored data a loss of information could occur.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The storage requirements for data are presented in procedures describing back-up routines. Yearly back-ups are kept forever. This requirement will be included in the SAR.

SUGGESTION

S1: The storage requirements (records, period, etc.) for the facility data collected by the main computer should be included in the OLCs. The Operating Organization is advised to ensure that the data relevant for safety and/or decommissioning can be retrieved after a long term period due to computer technology evolution.

ISSUE OLC-03: OPERATING LIMITS AND CONDITIONS

Control rod drop times

BASIS AND REFERENCES

[1] IAEA guideline NS-R-4 (6.90 to 6.94)

[2] SAR part I chapter 4 "Control station"

ISSUE CLARIFICATION

In chapter 4 of the SAR (part I) the control station (control rods and drive mechanism) is described.

OBSERVATIONS

The control rod drop times are measured by the Operating Organization by means of a dedicated procedure. The frequency for performing the measurements and the acceptance criteria as part of the OLCs are however not indicated.

The measurement of the control rod drop times is an efficient mean to detect possible degradation of the reactor shutdown system.

POSSIBLE SAFETY CONSEQUENCES

Abnormal functioning of the shut down system could lead to significant safety consequences.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The Operating Organization has no comments to this issue. The recommendations will be included as proposed.

RECOMMENDATION

R1: The requirements (frequency and acceptance limits) for the control rod drop time measurements should be included in the OLCs.

ISSUE REG-01: REGULATORY SUPERVISION

BASIS AND REFERENCES

[1] IAEA NS-R-4, Safety of Research Reactors

ISSUE CLARIFICATION

Regulatory Supervision is part of the overall system of safety assurance and must play a major role in nuclear and radiation safety. A structured nuclear and radiation safety supervision has been established in Norway.

OBSERVATIONS

Norwegian Radiation Protection Authority (NRPA), implementing the Halden Reactor Site Supervision, was established by the Norwegian Government and its duties, responsibilities and powers are expressed in the Act No. 28 of 12 May 1972, Nuclear Energy Activities, chapter II, section 10. NRPA is the highest agency in the nuclear, radiation and industrial safety. It functions as the institution making recommendations and giving advices to the Ministry of Health and Social Affairs.

NRPA has about 100 employees working in 3 departments:

- Department for Radiation and Nuclear Safety;
- Department for Emergency and Environmental Issues;
- Department for Planning and Administrative Matters.

The Department for Radiation and Nuclear Safety consists of 4 sections:

- Non-Ionizing Radiation;
- Radiation Safety in Medicine;
- Medical Diagnostics; and
- Industrial and Research Application of Radiation.

From them, the last section is dealing with supervision of the nuclear and radiation safety at Kjeller and Halden reactor sites.

It consists of 13 specialists distributed as follows:

- Nuclear and Radiation Safety at Kjeller and Halden – 4;
- Industrial Safety – 4;
- External Relations – 2 (e.g. external projects under the Ministry of Foreign Affares);
- Personal Dosimetry (Medical and Industrial) – 3;
- 3 posts in vacancy.

These specialists are dealing with the licensing and inspection issues for the nuclear reactors in the following areas:

- Emergency Planning;
- Environmental Monitoring;
- Research of Radiology and Emissions;

- Applications for Radioactivity Releases.

In the areas of emergency planning and environmental monitoring, specialists from other sections are available, contributing with approximately 1 man-year. As people have other tasks in addition to their main location, the effective workforce allocated to nuclear safety is approximately 2 man-year from the section and 3 man-year in total.

In implementing Regulatory Supervision at Halden there are only few acts and regulations (laid down by Royal Decrees and introduced by the Ministry of Health):

- Act 28 of 12 May 1972 concerning Nuclear Energy Activities [reviewed 18 August 1995];
- Act on Radiation Protection and Use of Radiation, No. 36 of 12 May 2000); and
- Regulations No. 1362 of 21 November 2003 on Radiation Protection and Use of Radiation (Radiation Protection Regulation).

There is no formally developed and adopted Nuclear Safety Standards. In the regulatory supervision, including licensing process, NRPA has adopted for use, on an informal basis, the IAEA Standards and Guides, as well as the other national standards (e.g. USA, CFR Codes), (see attached letter from the Head of Department, Mr. Sverre Hornkjøl).

Since NRPA has limited man-power resources it is forced to invite the external experts from abroad (e.g. Sweden) to review safety analysis issues.

The line of licensing looks as follows: The Halden site is to submit the application license for the reactor operation to the Ministry which asks NRPA to start licensing process (review, inspection, etc.). It takes about 1,5 years, with the License Conditions to be attached to the license. When the license is received, the Halden site shall develop an Action Plan to be further inspected on a regular basis by NRPA.

The second important part of NRPA activities is to conduct inspections at the site and meetings with the IFE. Usually no more that 2 meetings/inspections per year.

POSSIBLE SAFETY CONSEQUENCES

The regulatory supervision is a major actor for ensuring the Nuclear and Radiation safety at the Halden Reactor site.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

Agreed. For development of safety standards systems NRPA will need more resources.

RECOMMENDATION

R1: To enhance the supervision of the safety of the HBWR, the NRPA should develop an inspection programme and increase the number of inspections at the Halden site. This programme should also be established for other nuclear facilities.

The NRPA should develop Regulatory Guides on Safety of Research Reactors in line with the IAEA Safety Standards.

ISSUE SCO-01: SAFETY COMMITTEE

Limited independency of the Safety Committee and lack of advisory committee to the Reactor Manager

BASIS AND REFERENCES

[1] IAEA guideline NS-R-4 (4.15)

[2] SAR part III chapter 3 "Administrative control"

ISSUE CLARIFICATION

In chapter 3 of the SAR (part III) the role of the Safety Board (Safety Committee) is described. This Safety Board provide advises to the Director of the IFE on different safety issues related to the reactor and experiments.

OBSERVATIONS

- To advise the Director of IFE, there is a well established safety committee with competent members covering the required areas of interest.
- Although the members of this safety committee are not directly involved in reactor operation, they functionally are staff members of the Operating Organization (IFE) and therefore formally not independent.
- There is no formal advisory group available for the Reactor Manager with experts in different fields associated with the operation and modification of the facility, the design and the conduct of experiments.

POSSIBLE SAFETY CONSEQUENCES

- The limited independence of the safety committee could lead to public doubts on the existence of "conflicts of interest" and on the effectiveness of this committee to review and advise correctly on safety matters.
- The lack of an advisory group for the Reactor Manager may affect the quality of the internal review procedure of safety documents and issues. This limitation related to the internal review process could lead to possible safety issues or operational problems.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

For the Reactor Manager there are several advisors in the staff of the organization. However, these advisors are not formally put together in a committee.

When a safety issue is presented to the safety committee, the responsible personnel for the issue is only presenting the issue and is not part of the safety evaluation. This also includes members of the committee presenting safety issues.

RECOMMENDATIONS

R1: The Operating Organization is recommended to add and formally nominate external experts in the Safety Committee advising the IFE Director or to establish an independent group of external experts to supervise and assess the Safety Committee work on important safety matters.

R2: The Operating Organisation should establish an internal advisory group with clearly defined terms of reference to advise the Reactor Manager on safety aspects. The expertise of this group should cover the design, operation, modification and utilization of the facility including new experiments.

ISSUE SCO-02: SAFETY COMMITTEE

BASIS AND REFERENCES

[1] IAEA, NS-R-4, Safety of Research Reactors

[2] Requirements for the HBWR Safety Analysis Report

[3] Strategy for HSE

[4] HSE Manual

ISSUE CLARIFICATION

According to the IAEA Safety Standards and international practices a Safety Committee inside the Operating Organization should be established as an independent reviewer and advisor on all safety administrative and technical issues, related to operation of HBWR. Such a committee has been established inside IFE to help the Operating Organization management to ensure the required safety level.

OBSERVATIONS

Safety Committee in all documents is called Safety Board. It was established around 1948 and appointed by the IFE Director. Now it consists of 9 members, 2 observers, 1 leader as follows:

1. Mr. Steinar Backe, Deputy Head of Safety, Department Head of Radiation Protection and Environmental Monitoring and Surveillance, Head of Radiation Protection Kjeller
2. Mr. Thomas Elisenberg, Chief of Operations HBWR (Reactor Manager)
3. Mr. Nils Førdestrømmen, Reactor Physicist and Expert MTO (HF)
4. Mr. Sverre Hval, Deputy Chief of Operations, JEEP II, Reactor Physicist, Criticality
5. Mr. Erik Kolstad, Department Head Experimental Analysis, Fuel Behavior
6. Mr. Knut Lunde, Senior Advisor, Expert Metallurgy
7. Mr. Arnulf Wahlstrøm, Deputy Department Head, Experimental Operation, Expert in Instrumentation
8. Mr. Erlend Larsen, Secretary, Expert Chemicals and Security
9. Ms. Evelyn Foshaug, Head of Radiation Protection Halden
10. Mr. Gordon C. Christensen, Section Head HSE Kjeller, Chemical Expert
11. Mr. Jon Per Rambæk, Chief of Operations (Reactor Manager) JEEP II – Kjeller
12. Mr. Atle Valseth, Head of Safety, Safeguards, Security and Quality Management Department, Head of the Safety, Head of Safety Committee

Its duties, responsibilities and powers are set up in the Terms of Reference of the Safety Committee. The Committee is reporting directly to the IFE President, but is not a fully independent from the operating organization. The issues to be reviewed and analyzed are the following:

- Notifications, license condition changes, new systems;
- OLC, incidents and accidents to be reported to NRPA.

The Safety Committee works together with external bodies and physical protection. All resolutions made by this Committee are achieved in an appropriate system. The head of this Committee is the Head of Safety, Security and Quality Management Department in the organizational structure of the institute, which is, at the same time, independent from the Operator. The Head of Safety, Security and Quality Management Department is also managing the QA section of IFE. So, all issues related to safety are also included in a QA system and programme.

The issues like periodic review of operational performance, results of routine releases and radiation doses to the personnel and the public were not explicitly stated in the Terms of Reference of the Safety Committee.

POSSIBLE SAFETY CONSEQUENCES

The Safety Committee plays an important role in maintaining the adequate safety level of the reactor and associated activities. It acts as the "internal regulatory authority" of the Institute.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

Agreed during discussion.

RECOMMENDATION

R1: To improve the independency of the Safety Committee from the Operating organization, it is recommended that external experts be added and formally nominated as members of this Committee.

R2: The Safety Committee should periodically review:

- The operational and safety performance of the facility;
- Reports on routine releases of radioactive material to the environment;
- Reports on radiation doses to the personnel and the public.

ISSUE SCU-01: SAFETY CULTURE

BASIS AND REFERENCES

[1] IAEA NS-R-4, Safety of Research Reactors

[2] INSARR Guidelines (para 3.9)

ISSUE CLARIFICATION

Safety culture issues are getting more and more important at present time and closely related to quality assurance to create for each participant in the reactor safety the deep belief that the safety issues must be of the first priority.

OBSERVATIONS

The review showed that most of the recommendations of the IAEA concerning the safety culture are being properly introduced and implementing by the Operating Organization.

The organizational structure and the organizational policy have provided to all staff the clear distribution of responsibilities of functions such that all staff have well understanding of them and their attitudes to the safety are at adequate level. The safety culture is being implemented by the operating staff with a high level of quality.

The only remark that can be mentioned on this subject is to include in the annual performance appraisal a section on "attitude towards safety".

POSSIBLE SAFETY CONSEQUENCES

Maintaining the safety culture of the staff is a prerequisite of the Nuclear and Radiation Safety Assurance.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

Agreed to reconsider the remark on the annual staff appraisal.

SUGGESTION

S1: It is suggested to include in the "Annual Performance Appraisal" a specific section on staff attitude to safety.

ISSUE OOR-01: OPERATING ORGANIZATION AND REACTOR MANAGEMENT

BASIS AND REFERENCES

[1] IAEA, NS-R-4, Requirements for the safety of research reactors

[2] HBWR Safety Analysis Report, Part III-Operation

ISSUE CLARIFICATION

It should be noted that in HBWR Safety Analysis Report (SAR) this issue is presented in sufficient manner to understand and evaluate the structure and administrative issues of the reactor management in term of their influence on safety.

OBSERVATIONS

The organizational chart of Operating Organization and Reactor Management (OORM) is clearly established throughout the whole IFE. There is a set of supplemented documents like QA functions, duties, responsibilities and authorities in the process of the reactor operation. All functions, duties, responsibilities and authorities are clearly stated. The following shortcomings have been mentioned during review:

- No formalized system for job motivation and position
- The operational chart concerning Radiation Protection is slightly different from the IAEA recommendations and international practices concerning the degree of independence of Health Physics from Operator.
- There is no System of Safety Standards and Guides for Nuclear and Radiation Safety. OORM is using informally the IAEA Standards and Guides together with other national standards on a case by case basis to be approved by NRPA.

POSSIBLE SAFETY CONSEQUENCES

The proper organization and reactor management play a major role in ensuring safe operation of the reactor.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The Operating Organization will implement the recommendation.

RECOMMENDATION

R1: To reinforce the independence of the Health, Physics function, which is currently under the General Manager of Halden, it is recommended to assign this function under the supervision of the Director of IFE (like in the case of the Kjeller site), or under the direct supervision of the Head of Safety, Security and Quality Management Department.

SUGGESTION

S1: Create formal system for job position and motivation.

ISSUE MSY-01: QUALITY ASSURANCE; STRATEGY FOR HSE; HEALTH, SAFETY AND ENVIRONMENT MANUAL

BASIS AND REFERENCES

[1] IAEA, The Management System for Facilities and Activities (GS-R-3)

[2] IAEA, Application of the Management System for Facilities and Activities (GS-G-3.1)

[3] HWR Safety Analysis Report, Part I-Description

ISSUE CLARIFICATION

This is to review the IFE QA system and programmes to verify that activities important to safety are managed, performed and evaluated in accordance with the QA programme and procedures. Finally, it was noted that the Institute has a very well developed QA system for all activities related to nuclear and radiation safety.

OBSERVATIONS

During the review, several documents have been presented on QA system of the IFE. This system encompasses not only nuclear and radiation areas, but other industrial areas, which are part of the IFE field of activity.

In nuclear and radiation safety the QA system is set up on a hierarchical basis level:

1. Red;
2. Amber;
3. Yellow; and
4. Green.

It means from the top level of general requirements (red) to the bottom level of specific requirements (green) for operation sections and the staff of the reactor. All justifying documents are presented. The QA system got "Certification of Qualification" from Norway and Denmark Oil Industry.

POSSIBLE SAFETY CONSEQUENCES

Remote possible consequences.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

Agreed to take into account the suggestion indicated below.

SUGGESTION

<u>S1:</u> It is suggested that the HSE Manual includes a diagram of responsibilities and motivation aspects, and Safety Strategy integrates more explicitly the safety culture aspects.

ISSUE COP-01: CONDUCT OF OPERATION

BASIS AND REFERENCES

[1] IAEA, NS-R-4, Safety of Research Reactors

[2] HBWR, Safety Analysis Report, Part III: Operation

ISSUE CLARIFICATION

On the basis of the comprehensive review of this subject at the reactor it was understood that all operations at the reactor related to safety are being conducted in accordance with written and approved procedures and under the dual supervision and control of the reactor management.

OBSERVATIONS

The Halden Reactor Project Management Chart is presented. The responsibility to safely conduct the reactor operation and other associated activities lies upon the Reactor Operation and Engineering Division having 4 sections:

5. Maintenance and Installation;
6. Reactor Operation;
7. Design and Development; and
8. Safety.

The reactor is operated by 6 shifts, each of them being composed of a Reactor Engineer, Shift Leader, and two Reactor Operators. The operations are carried out in accordance with written procedures, the records and reports are well maintained. There are 36 written operating procedures in the Control Room and two in the Senior Reactor Engineer Office. Written operating procedures include adequate, technically accurate and complete written instructions to the staff for all activities related to the reactor operation, including radiation protection, needed authorizations, instruction on the staff actions in case of emergency. Also included are the instructions on the activities during outage periods for doing in-service inspection, maintenance, repair and fuel handling. Staff is properly trained and qualified. All of them got a permission by the Operating Organization Management to operate the reactor. The operating documents and records are kept and stored during 10 years. Most current version of instructions is used by operators and checked at every reactor start-up. However, it is noted that experiments to be conducted at the reactor have not been categorized in term of impact on safety, as well as possible incidents (or failures) during in-service inspections or maintenance are not explicitly stated. The signs located on the components and instrumentation of the primary and secondary coolant systems need to have more clear appearance and be coloured in accordance with the safety class assignment to the components.

POSSIBLE SAFETY CONSEQUENCES

The lack of clear operating procedures may result in the misunderstanding by the staff of the

Operating Organization leading to incidents and accidents.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

Instructions will be updated to include the above mentioned remarks.

SUGGESTION

S1: It is suggested that:

- the safety category of experiments be defined;
- safety classification of primary and secondary coolant system components be represented by visible signs.

ISSUE UEM-01: UTILIZATION AND EXPERIMENTS

Safety assessment and requirements

BASIS AND REFERENCES

[1] IAEA guideline NS-R-4

[2] SAR part I chapter 11 "Experiments"

ISSUE CLARIFICATION

In chapter 11 of the SAR (part I) the experimental devices, loops and their specific applications are described.

OBSERVATIONS

- Despite the existence of experimental procedures defining the conditions for experiments, the SAR does not provide design requirements for experimental devices (enveloping values) such as allowed linear power, maximum power, allowed materials presented in the SAR.
- The assessment of the experimental devices is not done taking into account the "defence in depth principles".
- The waste generated by experiments and other utilizations are not included in the design reports of the devices.
- The safety assessment of the experiments is performed by the safety board without the support of other knowledgeable experts.

POSSIBLE SAFETY CONSEQUENCES

The utilization of the facility is the first reason for its existence; failure of experiments with possible consequences for the facility could influence the safety and availability of the facility.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The SAR covers which type of experiment can be performed at HBWR. These types of experiments are therefore approved and included in the licence conditions. If other types of experiments are planned to be done, new approval by the Regulatory Body must be given.

The safety philosophy and the implementation of the philosophy are the same for experiments as for the facility, which also includes the principle of defence in depth.

RECOMMENDATION

R1: The review and the licensing of experiments and associated experimental devices should be improved by defining and implementing a clear licensing process involving the Regulatory

Body and supported by competent external experts.

R2: In the SAR a list of requirements and limitations related to the experimental utilization is to be included. This list should contain the enveloping values of parameters important to safety and should facilitate the review process of the feasibility of the foreseen experiments.

R3: The estimation, handling and disposal of waste generated by the experiments and experimental devices should be incorporated in the experiment reports; the compatibility to handle the waste originating from experiments with existing waste management procedures should be assessed.

ISSUE TRQ-01: TRAINING AND QUALIFICATION

Lack of authorization of operators

BASIS AND REFERENCES

[1] IAEA guideline NS-R-4 (7.3 and 7.4)

[2] SAR part III chapter 4 "Operating Personnel"

ISSUE CLARIFICATION

In chapter 4 of the SAR the requirements and the specific education levels of the operating personnel are presented.

OBSERVATIONS

The education and training of Reactor Operators, Shift leaders and Reactor Engineers are established in different documents describing the required education level, training plan and experience needed, however:

- Operating Personnel does not receive a formal license to operate ("authorization") after successful completion of the training programme.
- The Regulatory Body is not formally involved in authorization of Operating Personnel.

POSSIBLE SAFETY CONSEQUENCES

The established content of training, training requirements and authorization by the Operating Organization as well as their implementation are not formalized. This may lead to training deficiencies which in terms might lead to insufficient level of control room staffing.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The Operating Organization has no comment on this issue.

RECOMMENDATIONS

R1: The Regulatory Body and the Operating Organisation should establish a clear procedure for the authorization of Operating Personnel.

R2: The existing training and retraining requirements for all Operating Personnel need to be formalized.

SUGGESTION

S1: It is advised that the list of authorized Operating personnel with their specific level of training be send annually to the Safety Board and the Regulatory Body for information.

ISSUE AMG-01: AGEING MANAGEMENT

BASIS AND REFERENCES

[1] IAEA guideline NS-R-4 (section 7.108 to 7.110)

[2] IAEA Safety Guide SS 35-G1

[3] SAR, Part IV (Chapter 4)

ISSUE CLARIFICATION

In chapter 4 of the SAR (part IV) the ageing requirements and measures taken to ensure the long term availability of the reactor equipment and structures are described. The establishment of an ageing programme for all systems, structures and components which are either important for safety or which could influence the long term availability of the facility is essential to ensure safe operation and to prevent unwanted or unforeseen degradation of systems, structures and components.

OBSERVATIONS

The Operating Organization has established an ageing management programme for mechanical systems, structures and components of the primary cooling system which are important for safety of the facility; this programme is in place and used to execute periodic inspection, testing, maintenance as well as refurbishment.

An internal document describes the philosophy of the Operating Organization to prevent ageing of electrical systems, structures and components which are important for safety or the long term availability of the facility. It contains in an structured manner the following components:

- Main transformers;
- Distribution cabinets;
- Uninterrupted Power Supply systems;
- Connection panels; and
- Cables and wiring.

Although this document is available, it is not part of the ageing programme nor has it formally been implemented by the Operating Organization.

The Operating Organization also applies a policy to ensure the transfer of knowledge from experienced staff at key positions by early replacement and job overlap of 3, 6 or even 12 months. This strategy is supported by upper management and funds are made available.

POSSIBLE SAFETY CONSEQUENCES

A non-comprehensive ageing management programme may lead to unforeseen degradation of systems, structures and components of the facility reducing its availability and affecting its safety

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The Operating Organization has no comment on this issue.

RECOMMENDATION

R1: The Operating Organisation should establish a comprehensive ageing management programme for all systems, structures and components which are either important for safety or which could influence the long term availability of the facility. This programme should integrate the existing ageing programme for electrical systems, structures and components.

GOOD PRACTICE

G1: The policy to ensure the transfer of knowledge from experienced staff at key positions by early replacement and job overlap of 3, 6 or even 12 months is recognized as a good practice.

ISSUE MPT-01: MAINTENANCE AND INSPECTION

Requirements for maintenance and periodic testing of SSC's important for safety.

BASIS AND REFERENCES

[1] IAEA, NSR-4

[2] IAEA safety guide NS-G-4.2

[3] Safety Analysis Report, part IV, chapter 4

ISSUE CLARIFICATION

In chapter 4 of the SAR (part IV) the requirements for maintenance and periodic testing are presented. The inspection programme concerning the items important for safety is not exhaustive.

OBSERVATIONS

In chapter 4 of the SAR (part IV) the categorization of some of the mechanical systems, structures and components is presented on basis of their importance to safety. The inspection frequency and applied inspection methodology for these components are given in the inspection programme while other supporting documents specify the maintenance and periodic testing in more detail.

The formal periodic maintenance and inspection programme is limited to the mechanical systems, structures and components which are important to safety. There is an exhaustive periodic maintenance and inspection programme established for other systems, structures and components important for safety such as the control rod drive mechanisms, the air monitoring systems, electrical Uninterrupted Power System (UPS) and nuclear instrumentation. This draft is in process of being formalized.

POSSIBLE SAFETY CONSEQUENCES

An established maintenance programme covering all items important for safety, including periodic testing and acceptance criteria, is essential to ensure safe and long-term availability of the facility.

The purpose of periodic testing and maintenance is to confirm that the systems, structures and components continue to meet the design intent as expressed in the SAR and the OLCs.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The Operating Organization has no comment on this issue.

RECOMMENDATION

R1: The Operating Organization should establish for all items important to safety a comprehensive maintenance and inspection programme defining the maintenance requirements, the frequency of inspections and associated acceptance criteria.

GOOD PRACTICES

GP1: All components of the facility have a unique identification number which enables to retrieve historical data on failures and malfunctioning. This also allows well planned and organized maintenance and periodic testing.

GP2: Working orders for corrective maintenance are issued in a structured way. All sections (maintenance, operations and health physics, etc.) are involved in the process of preparation, isolation, and execution and restoring of the systems maintenance. Safety risks and related precautions are clearly identified.

ISSUE RPP-01: IMPROVEMENTS FOR RADIATION PROTECTION PROGRAMME

BASIS AND RFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA Safety Standard No.115 International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, 1996

[3] IAEA Safety Requirement NS-R-4, Safety of Research Reactors, 2005

[4] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report

[5] Halden Procedure Sv-krav-08/30.09.2004 - Limits of doses and the actions to limit the radiation exposure.

[6] Requirements on radiation protection related to the waste handling Sv-krav-06/24.05.2006

[7] Procedure for handling radioactive waste at HBWR – QA-P-852/30.11.2005

[8] Hard copy of list of radiation protection monitors, printed from electronic file of the Picasso software – Stv 23.09.2005 – ask under which procedure/instruction is this file updated/or belonging to.

[9] The criteria for reporting and routines for reporting on radioactive releases – QA-RPI-33/16.04.2007

[10] Release to the delay tank in special conditions – QA-RPI-56/31.07.2005

[11] Handling instruction for cutting pipes and filling steel waste boxes and concrete boxes ROE-A-045/03/03/2006.

[12] Basic rules and principles for handling radioactive materials – Sv-SAR-001/26.01.2006

[13] Control of the contamination of non-active waste to be discharged from the site – QA-RPI-111/28.07.2006

[14] Radiation protection education level 1- Rules for working in controlled area – QA-RPI 95/27.03.2007

[15] Overview on Radiation protection education at the HBWR – Level 1, 2, and 3 – QA-P-953/17.04.2007

[16] Flowchart no.379507, of the cooling system of the D_2O steam circuit and subcooler circuit

[17] Flowchart no.379244, of D_2O purification circuit and recombination circuit low and high pressure

[18] Draft flowchart on Off-gas and fission gas collection system – proposal for design modification - Drawing number is missing.

[29] Listing of date/time/nuclides measured during an abnormal occurrence release case – 6

April 2007 - there is no reference to any procedure/computer programme.

[39] The operation license for IFE Atomic Facilities in Halden and Kjeller – ref.97/08715 FOS HEJ from 22.12.1999, the validity is from 01.01.2000 until 31.12.2009 and press release.

[40] The Permit for release of radioactive effluents to the Institute for Energy Technique. Doc.GO05-11/Application number 2005/00794 from January 2006 until 31 December 2009

ISSUE CLARIFICATION

The radiation protection programmes, procedures and practices at research reactors should assure that radiation protection of workers and members of the public is in accordance with international recommendations and national regulations and laws. The radiation protection programme should exist and be implemented through appropriate practices. It should clearly define the authority and responsibilities of the radiation protection group. The radiation protection programme is subject to the requirements of the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources and is subject to the approval of the regulatory body.

OBSERVATIONS

The policy on radiation protection in IFE is included in the Administrative document no.049/27.03.2007.

The organization chart for the radiation protection group, presented during the discussion, shows that a clear line of communication and reporting exists within the group and directly to the director.

However, the radiation protection group is organized under the department of experiments at Halden. The independency of the radiation protection organization from the Operating Organization at Halden is to be discussed.

There is a written overall radiological protection programme.

There are clear written requirements on the education needed to hire a new person as radiation protection officer. The basic education programme after employment is included in procedures. There are both theoretical and practical education programmes. For all staff employed by IFE and for guest workers there is a basic education programme on radiation protection. For the operational staff there is an advanced-training programme. However there is no examination concerning the theoretical knowledge.

The release limits are not established in the operating license. There are release limits of effluents and reportable limits of the effluents. The release limits are set in the Appendix to the Radiation Protection Act no.1362.

The reportable limits are stated in the Permit for release of radioactive effluents to the Institute for Energy Technique. Doc.GO05-11/Application number 2005/00794 from January 2006 until 31 December 2009. According to the counterpart until now there was no situation in which the release limits were exceeded. However a procedure should be developed for actions/measures to be applied in such case.

For the time being, there is a shortage of staff in the radiation protection group. The hiring process to fulfill the vacancies on radiation protection staff is ongoing.

The discussion with the counterpart showed that there is no financial constraint concerning the procurement of radiation protection equipment.

All the dosimetry system is monitored by IFE/Kjeller. The responsible from Kjeller evaluate the dosimeters and report the doses every six-month to NRPA.

Within normal operation the radiation dose rates in the reactor hall could be up to 20mSv/h. More often the radiation dose rates in the reactor hall during operation do not exceed 10mSv/h. During a serious fuel failure the dose rate in the reactor hall, measured in the sink reached 1Sv/h. It should be noted that during normal operation, the access of the personnel in the reactor hall is forbidden.

The fix radiation monitors are not calibrated, only the functionality is checked regularly. The portable radiation monitors are calibrated once per year.

According to the counterpart, the radiation protection staff is part of the evaluation group of any new experiments. This should be reflected in the internal procedure.

The radiation protection counterpart mentioned that all cases of contamination or overexposure are investigated immediately by radiation protection staff. However, it was not clear how the feedback from the lessons learned from that investigations are implemented further on.

There is no ALARA programme and there is no ALARA committee.

POSSIBLE SAFETY CONSEQUENCES

If a clear Radiation Protection Programme is not in place, there could be practices and activities, which may lead to improper radiation surveillance, and as a result the workers involved may receive higher dose or radioactive waste may be released without control.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

For the time being, there are no requirements for examination at the end of the training of Radiation Protection Staff. The elements of the radiation protection programme are dispersed in different documents.

The counterpart mentioned that there is a procedure for new experiments and design modification which requires that the project leader call for the radiation protection specialist to participate in the working group for safety assessment of new experiment.

RECOMMENDATIONS

R1: The Radiation Protection Programme at of the Halden site should be improved in accordance with NS-R-4 (para.7.97) and BSS No. 115. This plan in particular should include provisions for the following:

(a) Ensuring that there is co-operation between the radiation protection staff and the operating staff in establishing operating procedures and maintenance procedures when radiation hazards are anticipated, and ensuring that direct assistance is provided when required.
(b) Providing for the decontamination of personnel, equipment and structures.
(c) Controlling compliance with applicable regulations for the transport of radioactive material.
(d) Detecting and recording any releases of radioactive material.
(e) Recording the inventory of radiation sources.
(f) Providing adequate training in practices for radiation protection.
(g) Providing for the review and update of the programme in the light of experience.

SUGGESTION

S1: The IFE should define and implement an examination process for all the operation staff after completion of the radiation protection training programme.

GOOD PRACTICE

GP1: The team noticed the existence of the following good practices:

- A common meeting is held each morning within the Operating Organization with the participation of the operating group, the maintenance group and the radiation protection group;
- Weekly reports, including radiation protection issues, are distributed internally within the site and externally to the NRPA in a shorter version.

ISSUE RPP-02: IMPROVEMENT NEEDED FOR ACCESS CONTROL AND ZONING

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA Safety Standard No.115 International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, 1996

[3] IAEA Safety Requirement NS-R-4, Safety of Research Reactors, 2005

[4] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report

[5] Halden Procedure Sv-krav-08/30.09.2004 - Limits of doses and the actions to limit the radiation exposure.

ISSUE CLARIFICATION

In determining the boundaries of any controlled area, registrants and licensees shall take account of the magnitudes of the expected normal exposures, the likelihood and magnitude of potential exposures, and the nature and extent of the required protection and safety procedures.

The boundaries of restricted access to controlled areas should be established by means of administrative procedures, such as the use of work permits, and by physical barriers, which could include locks or interlocks; the degree of restriction being commensurate with the magnitude and likelihood of the expected exposures; Periodically there should be a review of conditions to determine the possible need to revise the protection measures or safety provisions, or the boundaries of controlled areas.

OBSERVATIONS

The zoning is divided in clean area and controlled area. In the controlled area, there is contaminated area and non-contaminated area.

At the entrance in the reactor tunnel there is a whole body monitor. However, this could be bypassed because there is no physical barrier in place. There is only an administrative barrier (procedure).

The radioactive waste is stored in a parallel tunnel. To transfer radioactive waste from the reactor to that tunnel, the waste should be taken through the clean area. There is no direct connection between reactor tunnel and waste tunnel.

The chemistry laboratory is located on the first floor above the reactor tunnel. The access to this laboratory is from the clean area. The staff coming out from chemical laboratory could leave without monitoring control.

There is no layout drawing concerning the radiation protection zoning, which is only described in written text. However, there is a schematic picture in the basic information brochure.

During walk-down through facility the warning symbols and signs for radiation protection zoning (contaminated and non-contaminated area) were missing. Also in the waste storage tunnel there was no warning sign on the shielding where the high contaminated liquids are stored.

POSSIBLE SAFETY CONSEQUENCES

Contaminated personnel and equipment may access the clean area without control of the contamination.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The counterpart is aware of the improper boundary and barrier for the contamination area. There is a draft programme for upgrading the barrier between contaminated and non-contaminated area.

RECOMMENDATION

R1: The radiation protection zoning and barriers should be reviewed and upgraded in the light of the BSS and international good practices.

ISSUE RWM-01: TRANSPORT OF RADIOACTIVE MATERIALS

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report

[3] SAR Part III – Chapter 9 – Transport of radioactive materials

[4] Transport cask, calculations for transport cask for fuel elements, Doc. RD-42, 1967-09-13.

[5] Packaging for irradiated fuel, Doc. RD-75, 1969-03.

[6] Criticality evaluation of transport container for non-irradiated fuel, Doc. SiKval-99-5, 2000-02-18

ISSUE CLARIFICATION

According to IAEA TS-R-1 Regulations for the Safe Transport of Radioactive Material, 2005 Safety measures should be in place during the transport of radioactive material. This safety measures are achieved by requiring:
(a) containment of the radioactive contents;
(b) control of external radiation levels;
(c) prevention of criticality; and
(d) prevention of damage caused by heat.

These requirements are satisfied firstly by applying a graded approach to contents limits for packages and conveyances and to performance standards applied to package designs depending upon the hazard of the radioactive contents. Secondly, they are satisfied by imposing requirements on the operation of packages and on the maintenance of packagings, including a consideration of the nature of the radioactive contents. Finally, they are satisfied by requiring administrative controls including, where appropriate, approval by competent authorities.

OBSERVATIONS

In SAR chapter 9, Transport of radioactive materials it is mentioned that the Institute for Energy Technology has permission to transport radioactive materials according to the transportation license for the Institute. The certificates and license for transportation were available for consultation during discussions.

The system for transport of radioactive materials is described in the following documents:

Administrative regulation 065; Transport of radioactive materials at IFE.

Procedure QA-P-848; Procedure for transport of radioactive materials.

Procedure QA-P-814 and Operating Procedure OPE-16; Rules for handling of irradiated fuel.

Procedure QA-P-836; Procedure for preparing Loading Programme.

Additional information was received from the counterpart as follow:

The transport activates of IFE consist of transport of radioactive materials (non-irradiated and irradiated fuel, heavy water, material specimens), transport of radioactive waste (mix bed resin, filters, plastic waste, paper, etc.), transport of isotopes, and transport of low and middle active waste (from IFE, other industries, hospitals, etc.) to storage repository at Himdalen.

The reviewers are pleased to notice that the transport activity is done following IAEA SS TS-R-1 "Regulations for the Safe Transport of Radioactive Material" 2005.

Type of transport containers includes exceptional packages e.g. industrial package type 1 (IP-1), type 2 (IP-2) type 3 (IP-3), package type A / AF, package type B(U) /B(U)F, package type B(M) /B(M)F and package type C.

The Kjeller flask N/0003/AF-96, Package type A is used for transport of non-irradiated fuel between Halden and Kjeller.

The Kjeller flask N/0001/B(M)F is used for transport of irradiated fuel between Halden and Kjeller. During discussion with the counterpart, it was mentioned that the container was manufactured in 1998.

The counterpart provided safety reports for containers (e.g. criticality, shielding and heat transfer calculations): [4] Transport cask, calculations for transport cask for fuel elements, Doc. RD-42, 1967-09-13. [5] Packaging for irradiated fuel, Doc. RD-75, 1969-03. [6] Criticality evaluation of transport container for non irradiated fuel, Doc. SiKval-99-5, 2000-02-18.

Calculations for the transport cask for fuel rod are prepared by radiation protection and the physics group.

Instructions and procedures for waste transports on site exist.

The transport container IFE/2600/A is used for transport of material specimens to foreign countries.

Containers N-IFE/2801/IP-2 is used as package for drums containing waste or heavy water.

Depending of type of container, the following test were requested: test of water penetration (for containers IP-3, A, AF, B), drop test on hard ground (for IP-2, IP-3, A, AF, B), stacking test (for containers IP-2, IP-3, A, AF, B), penetrating test (for containers IP-3, A, AF, B), drop test on steel column (for containers AF, B), fire test (for containers AF, B). Alternatively, the requirement accepts calculations or based arguments instead of tests.

The counterpart provides examples of transport documents for a shipment.

From organization point of view there is a transport coordinator, a dangerous Goods Safety Advisor (necessary for all activities which include transport of dangerous goods on road or railway (EU-directive 96/35/EF)).

After a specific training, it is issued a certificate for drivers of vehicles carrying dangerous goods.

Physical Protection and Emergency Preparedness measures are in place for any transport. In this regard there are detailed strategy plans/instructions for personnel who carry out transports (ref. ROE-A-008) for driver, escort personnel, internal HBWR staff on call, the reporting of position codes (ref. ROE-A-008) and the handling of accidents (ref. "Emergency preparedness plan for IFE

Halden" and ROE-A-008).

There are in place procedures for handling of containers and implementation of transports (ROE-A-015 and ROE-A-008), program for training of drivers, escort personnel and internal HBWR staff on call (ROE-283), emergency preparedness plan for IFE Halden.

At the walk through of the waste tunnel there was one concrete package with waste that was broken. The counterpart mentioned that this was a mistake from the manufacturer that the package was not properly armed. The counterpart has discussions with the external manufacturer about this mistake.

POSSIBLE SAFETY CONSEQUENCES

All measures are in place to coupe with any emergency situation.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The counterpart agreed with comments done during discussion on Transportation.

SUGGESTION

S1: The Chapter 9 - Transport of radioactive materials form SAR Part III is suggested to be completed with additional information as recommended in IAEA SS 35 G1, paragraph A.1211 with methods of handling and storage of sources, radioisotopes or other radioactive materials; and handling and disposal of radioactive waste.

GOOD PRACTICE

GP1: The transportation of radioactive materials is performed in accordance with international regulations. The transport casks are licensed and each transport is authorized.

ISSUE RWM-02: RADIOACTIVE WASTE MANAGEMENT

Lack of critically assessment

BASIS AND REFERENCES

[1] IAEA guideline NS-R-4

[2] SAR part I chapter 12 "Handling and storage of fuel"

ISSUE CLARIFICATION

In chapter 12 of the SAR (part I) the criteria and methodology to handle and store fresh and spent operational and experimental fuel are described.

OBSERVATIONS

Existing documents on criticality and analyses on fuel storage are currently not included in the SAR. Despite the fact that these documents have been assessed by the Regulatory Body in the framework of the previous license, the SAR should provide the following detailed information:

- Design requirements of all tools and equipment used;
- Criticality analyses under normal and accidents conditions;
- Storage capacity with possible limitations of positions
- Ability to unload the core under all circumstances
- OLC's of the fresh and spent fuel handling and storage

POSSIBLE SAFETY CONSEQUENCES

The absence in the SAR of sufficient information on criticality available in supporting documents, could lead to possible unsafe situations during handling and storage of fuel.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The revised safety document for the fuel storage is been approved by the Regulatory Body and the existing document is covered by the current license.

Important documents and safety issues are either referred to and analyzed in the safety document for the storage:

- Criticality under normal and accident conditions;
- Accidents;
- Capacity, and that there is storage capacity for core evacuation;
- Limits and conditions for spent fuel to be stored;
- Tools and equipment which are being used.

COMMENT

<u>**C1:**</u> The Operating Organization should implement all requirements specified under "Observations" in the SAR.

ISSUE RWM-03: IMPLEMENTATION OF WASTE MANAGEMENT PROGRAMME

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA NS-R-4 Safety of Research Reactors, 2005

[3] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report.

[4] SAR Part I – Chapter 13 Radioactive waste - handling and disposal;

[5] Waste Handling. Doc.QA-RPI-35/07.03.2007

[6] Quality Assurance Book for Radiation Protection Department – QA-013-no/14.02.2006

[7] Information letter from IFE to NRPA for exceeding the Reporting limits for tritium/ 23.04.2007

[8] The permit for release of radioactive effluents to the Institute for Energy Technique. Doc.GO05-11/Application number 2005/00794 from January 2006 until 31 December 2009.

[9] Process description for design, production, installation and modification of systems at HBWR. DocROE-P-001/24.08.2005

[10] Radiation Protection Declaration for installation and modifications of systems at HBWR. Doc.Sv-krav 17/18.07.2005

[11] Calculation of release to air of Tritium and Noble Gases.Doc QA-RPI-73/27.11.2003

[12] Instrumental check before star-up. Doc.QA-RPI-45/18.06.2007.

[13] Requirements on radiation protection related to the waste handling Sv-krav-06/24.05.2006

[14] Procedure for handling radioactive waste at HBWR – QA-P-852/30.11.2005

ISSUE CLARIFICATION

The operation of reactor and its experimental devices should minimize the production of radioactive waste of all kinds, to ensure that releases of radioactive materials to the environment are kept as low as reasonably achievable and to facilitate the handling and disposal of waste.

OBSERVATION

There is a written overall waste management programme at Halden.

During the discussions with different counterparts (radiation protection, transport) there was no clear description of the waste transportation (procedures, responsible person, and criteria) within the site e.g. from reactor hall to the storage tunnel, transfer of contaminated equipment between different controlled areas (from the reactor hall to the workshop). For clarifications, the counterpart provided the Requirements on radiation protection related to the waste handling Sv-krav-06/24.05.2006 [13] and the Procedure for handling radioactive waste at HBWR – QA-P-852/30.11.2005 [14]. The handling procedures apply to the entire site. There is no specific procedure for waste management within the reactor building.

There was no clear explanation on waste classification as low level waste and medium level waste neither procedure on segregation of waste. Records on quantities, types, and characteristics of stored solid and liquid radioactive waste and the waste that was removed from the reactor site were not available.

No goals have been set up by the Operating Organization to minimize generation of solid waste.

However at the walk through of the waste tunnel there was one concrete package with waste that was broken. The counterpart mentioned that this was a mistake from the manufacturer that the package was not properly armed. The counterpart has discussions with the external manufacturer about this mistake.

The radioactive waste is always sent to Kjeller for checking and handling and then it is sent from Kjeller to Himdalen. The waste packaging is decided based on dose rate limits.

The counterpart mentioned that there are descriptions for all waste packages including those stored in the waste tunnel. There are no radiation monitors in the waste storage tunnel.

All the experimental fuel rods are prepared at Kjeller and then sent to the reactor at Halden. For all the transport between Kjeller and Halden the responsibility for transport is on the Halden site.

There is a responsible person for waste management at Halden. There is also a supervisor for control of the waste management programme at Halden.

The counterpart mentioned that every person at Halden is responsible for measuring waste for correct categorizing.

The main source of liquid waste by volume is the drainage of ground water from the reactor containment. The water is pumped from the sink and is passed through one of the three delay tanks and then released to the river Tista. There is no on-line monitoring system for the activity of the liquid waste inside the delay tank. , but the water stream in the output line from the tank is monitored by two gamma detectors. Should the gamma monitors increase above a prescribed level, the output line will automatically be closed and the water is collected in the delay tank. The released activity from the delay tank is assessed by analysis of samples of the liquid taken two times per day from the output line from the tank. Other sources of liquid waste are normally purified, and the remainder activity is assessed before the water is released to the delay tank and after the result of analysis, the liquid waste is released into Tista river. If there is a need to use the purification system for liquid collected in the delay tank, there is a special working procedure for water circulation through the tank. There is one release point from only one tank.

There is a procedure to remove the sludge from the delay tank, which address only radiation protection requirements. The written instruction for this work is under development.

There is no active liquid effluent waste system in the reactor building. All radioactive liquids are not handled in the same way, taking into account the chemical compatibility, radioactive activity (segregation between low level, medium level liquid waste and high level activity waste) and dose rates.

There is no estimation of the waste that is going to be generated for the forthcoming period, for example the estimation of radioactive waste that will be generated from July to December 2007. This estimation should be checked against the generated waste.

For the normal ventilation, the efficiency of filters is not measured. The filters are replaced every 2.25 years.

The counterpart mentioned that there is a project under development for design modification of off-gas system in order to separate the heavy water and light water pipes from each other so that the heavy water could be recovered and returned to the primary circuit. A working group is taking care of the design requirements for this modification. The intention is to provide the same flow rate through the new system.

Within the off-gas system, there are two monitors, that are measuring both tritium and noble gases, and two monitors that measure particulate activity and iodine respectively. The two latter monitors will also detect noble gases. Those four monitors are not for measuring only for detecting. The measurements of airborne releases are done through sample analysis from charcoal filter, liquid scintillation counting of humidity samples and gamma spectral analysis of air samples. The monitors are not regularly calibrated only the functionality is tested. The functioning of those monitors is controlled twice per year.

According to the counterpart, this year it was a leakage from the He-3 experimental system. The pipes that relates to the system was contaminated with tritium. The counterpart mentioned that the gaseous release does not reach the release limits but it was over the reporting limits.

POSSIBLE SAFETY CONSEQUENCES

Mixture of liquid waste, without taking into account the chemical compatibility and radioactive activity, may lead to an increase of radiation exposure for workers as a result of wrong storage conditions.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The counterpart agreed that the filter efficiency should be checked and they will contact an external company to do this test.

RECOMMENDATION

R1: The waste management programme at Halden should formally define the person responsible for waste management. Periodic retraining should be done to all staff involved in handling of radioactive waste. Radioactive waste categorization should be clarified and included in an appropriate document for handling and storage of radioactive waste.

The procedures on handling of liquid effluents should clearly state that mixing of different levels of waste is not allowed.

GOOD PRACTICE

GP1: The design modification to separate the off-gas pipes for heavy water and light water is considered a good practice to recover the tritium.

ISSUE EMP-01: EMERGENCY PLAN

BASIS AND REFERENCES

[1] IAEA Safety Service No.1 INSARR Guideline, 1997

[2] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report

[3] IAEA SS NS-R-4 Safety of Research Reactors, 2005

[4] IFE Documentation as listed in the documents supplied by the Counterpart

ISSUE CLARIFICATION

A well established and implemented Emergency Preparedness Plan is essential to control on-site and off-site emergency situations and to mitigate consequences to the staff, members of the public and the environment. It should also include the administrative, technical and training requirements.

OBSERVATIONS

Emergency Preparedness:

For the emergency preparedness it is important to implement, a well defined emergency organization, with its responsibilities, the communications and reporting, the technical means, the emergency plan and its application through periodic exercises.

Emergency Organization:

Crisis Handling Group	Set up by the Government
↑	
Crisis Committee	The NRPA is in charge of the Crisis Committee.
↑	
LEVEL 1	Strategic Management IFE
↑	
LEVEL 2	Internal Rescue Staff (IRS) Halden/Kjeller
↑	
LEVEL 3	Building/Installation/Laboratories

Crisis Handling Group:	The crisis handling group is a non-technical Committee activated by the Government in case of any serious accident.
Crisis Committee:	The Crisis Committee was created by Royal Resolution of 26 June 1998; it deals with nuclear accidents in Norway or in another country with significant consequences in Norway.
LEVEL 1:	Timely dissemination at the appropriate level of accurate information to the public and the media. Plan for re-establishment of a normal situation and to handle public relations aspects. This Committee acts for both Halden and Kjeller sites.
LEVEL 2:	Its role is to assess the potential on-site and off-site consequences, to notify the appropriate local and national organizations, to monitor the radiological and physical state of the facility. There is a separate IRS for Halden and Kjeller sites.
LEVEL 3:	Implement protective measures related to the emergency condition to limit the consequences, to correct it and to return the reactor to a safe standard condition.

Emergency Plan:

The Emergency Plan includes:

- a statement of the objective;
- a description of the reactor including fission product inventory,
- a description of the location of the reactor facility including surrounding population density, nearby industrial activity and access routes;
- emergency organization responsibilities;
- off-site organizations to be notified;
- a description of emergency facilities and equipment, including location and communication equipment.

It does not contain a classification based on severity levels of emergencies, nor a checklist of assessment actions, nor the conditions and indications for starting and termination of the emergency.

In case of emergency:

- The acting officer in charge (Level 3) shall contact the acting emergency preparedness officer and Radiological Officer who are on call (Level 2). The acting radiological officer has his own emergency instructions. The acting emergency preparedness officer shall call the Head of Safety (in charge of Level 2).

- The Head of Safety calls other members of Level 2 and communicates to the Chairman of Level 1 (NRPA Director).

In an emergency situation, the paper factory situated beside the HBWR will be contacted by telephone and will be informed on the situation. The paper factory then has its own procedures for further actions when deemed appropriate.

There is an emergency monitoring room at HBWR site in the maintenance building that can be used in case of evacuation of the main control room. Near the main control room there is a storage for emergency equipment.

The Institute shall warn the authorities if an accident occurs or if an event occurs that could lead to off-site consequences. In this case, the authorities will take initiatives in accordance with the "Organization plan for rescue services in Norway". The chief of police for the Romerike district or the chief of police for Østfold and Follo district will be in charge for the local planning and execution of the actions.

There are Appendixes to the Emergency Preparedness Plan that deal with different types of emergencies, including the incidents and accidents covered by the present SAR. These Appendixes also take into account that more than one emergency situation could occur at the same time.

There are clear established agreements with local hospitals and the local ambulance for treatment of contaminated workers.

The IFE Service provides the following services:

- advices IFE personnel and gives guidance on questions of health;
- provides first aid;
- performs regular health checks;
- gives training in health-related aspects;
- participates in safety inspections walks through.

The Institute has access to a doctor and nurses employed at both Kjeller and Halden, and a physiotherapist at Kjeller too.

The Fire Protection Service performs:

- periodic testing of fire alarm equipments, extinguishers, escape routes and signposting;
- fire training including fire fighting techniques.

A discussion is ongoing within national emergency organization concerning the distribution of iodine tablets to the public.

At present, only the Level 1 to 3 communications of the emergency organization were the subject of emergency drills. Local, regional and national authorities have participated in exercises latest in 2006 at the drill at the Halden reactor. There is an ongoing process to include the other two levels in the periodic emergency exercises.

Emergency Exercises:

The overall exercises take place one or two times per year. External observers are present during the execution of the exercises. The control room staff is regularly trained for different emergency situations.

It was verified that regular training drills, including off-site organizations are conducted to check the operational character of the emergency plan with records of the drills and the emergency organization, findings and recommendations for improvement.

No external events or severe natural phenomena, nor off-site fire have been considered to be included in the emergency exercises.

POSSIBLE SAFETY CONSEQUENCES

A well established and implemented Emergency Preparedness Plan is essential to control on-site and off-site emergency situations and to mitigate consequences to the staff, members of the public and the environment.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The Operating Organization will implement the recommendations.

RECOMMENDATION

R1: On the basis of the next updating of the Safety Analysis Report, the on-site and off-site emergency preparedness plans should be revised to integrate the conclusions of the analysis concerning in particular the Design and Beyond Design Basis Accidents.

The classification based on the severity of the accidents and the conditions for starting and termination of the emergency situation should be included in the Emergency Plan.

COMMENT

C1: The Operating Organization should investigate if the height of the chimney and its specific location very close to the mountain are being taken into account in the distribution model for the release of activity in case of accidents and the subsequent dose consequences.

C2: In case of revision, there should be a procedure for returning the obsolete emergency plan back to the Operating Organization.

SUGGESTION

S1: The Operating Organization is advised to agree with the paper factory on the installation of loud speakers to be used for information to their personnel in case of an emergency situation at the HBWR.

ISSUE DEC-01: DECOMMISSIONING PROGRAMME

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA Safety Standard No.115 International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, 1996

[3] IAEA Safety Requirement NS-R-4, Safety of Research Reactors, 2005

[4] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report

[5] Decommissioning plan for the nuclear facilities for the Institute for Energy Technology, SU-2004-234, December 2004.

ISSUE CLARIFICATION

For some operating research reactors, the need for their ultimate decommissioning was not taken into account in their design. Nevertheless, all operational activities at research reactors, including maintenance, modifications and experiments, shall be conducted in a way that will facilitate their decommissioning. Documentation of the reactor shall be kept up to date and information on experience with the handling of contaminated or irradiated Safety Systems and Components (SSCs) during the maintenance or modification of the reactor shall be recorded to facilitate the planning of decommissioning. In developing the decommissioning plan, aspects of the reactor design to facilitate decommissioning shall be reviewed, such as the selection of materials to reduce activation and to facilitate decontamination, the installation of remote handling capabilities for the removal of activated components, and the incorporation of facilities for the processing of radioactive waste. In addition, aspects of the facility operation that are important in relation to decommissioning, such as any unintentional contamination whose cleanup has been deferred until the reactor decommissioning, and any modifications that may not have been fully documented, shall also be reviewed.

OBSERVATIONS

In the SAR there is no chapter presenting a summary of Decommissioning of Halden research reactor.

There is a strategic decommissioning plan for all IFE nuclear installations and facilities. There are also detailed preliminary decommissioning plans for each facility including the Halden research reactor (ongoing plan). The previous initial plan was submitted for the current license. The strategic decommissioning plan was developed as a result of the requirement from operating license and was submitted with the preliminary decommissioning plan to NRPA in 2005. By the end of 2005, the NRPA requested that a more developed preliminary decommissioning plan shall be submitted to the Operating Organization by the end of December 2007. This plan should include detailed decommissioning activities for the case that the spent fuel will be removed from the site. It should also indicate the final destination of the facility. The dismantling activities should be sufficiently detailed to estimate the cost of the decommissioning plan of Halden research reactor and the other nuclear facilities.

An external consultant was contracted to perform the cost calculations for decommissioning until free release of all facilities.

For the time being there is no requirement to archive for long term the main and important documents that could contain pertinent information and should be available in the future even after the decommissioning process is finalized.

POSSIBLE SAFETY CONSEQUENCES

All important documents containing pertinent information shall be archived for the future. They can be lost after decommissioning if there are no appropriate conditions for safe archiving.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The counterpart agreed to include in the SAR a summary of the decommissioning plan and will implement a long term archiving process.

RECOMMENDATION

R1: According to IAEA SS-35-G1, para.A.19, the SAR should include a chapter on the decommissioning of the facility. This chapter could be a summary of the existing decommissioning plan.

SUGGESTION

S1: All important documentation should continuously be archived.

GOOD PRACTICE

GP1: The approach used for developing the decommissioning plans gradually, first, establishing a decommissioning strategy for all facilities and then developing the preliminary decommissioning plan for each facility is a good practice.

ISSUE FIR-01: FIRE PROTECTION

Renewal of fire analysis and implementation

BASIS AND REFERENCES

[1] IAEA guideline NS-R-4 (7.71 and 6.21 to 6.25);

[2] IAEA safety guide No. 35-G1, A.211;

[3] SAR part I chapter 14 "Fire protection"

ISSUE CLARIFICATION

In chapter 14 of the SAR the fire prevention and warning systems are presented.

OBSERVATIONS

- Operating Personnel perform regular inspections by means of facility walk-downs of different parts of the facility. These inspections however are not based on work instructions and are not integrated in the document structure.
- The fire analysis performed by an external company originates from 1994 and does not reflect the present status of the facility.
- In 2006 during the periodic inspection by the Halden fire brigade, no remarks on the fire protection and implementation were made; however the team members identified several significant areas for improvement during the facility walk-down.
- The results of the internal fire audit performed in 2007 by the Operating Organization were reported to the Halden fire brigade. These audits are performed every 6 months; however there is no established programme.
- The fire protection and equipment are inspected on a regular basis by means of a written programme.
- The presence of a high combustible load combined with the lack of physical separation, barriers, compartments and fire/smoke detectors in sensitive areas as well as the improvements needed for housekeeping form a major fire risk which could jeopardize the safety of the facility.

POSSIBLE SAFETY CONSEQUENCES

A possible fire in the facility will cause a major safety risk with potential off-site consequences.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

Chapter 6.9 of SAR, Part IV includes safety assessment of fires at the facility. Fires in the storage for spent fuel are also included in the safety documents for the fuel storage.

The fire analysis done by an external company is planned to be updated. Conclusions from this revision will be included in the SAR.

According to operating procedures the operating personnel performs walk-downs of different parts of the facility each second hour.

RECOMMENDATIONS

R1: The Fire Analysis should be updated as soon as possible to take into account the present conditions of the facility. The improvements derived from this analysis should be implemented on the basis of the defence in depth principles (compartments, barriers, detectors, etc.). The fire analyses and fire protection provisions should be integrated in the SAR.

R2: The Operating Organisation should improve the housekeeping with the aim to minimize the combustible loads in order to limit the fire risk and propagation.

SUGGESTION

S1: The facility walk-downs performed by the Operating Personnel should be formalized and implemented by means of an operating procedure.

ISSUE ENV-01: ENVIRONMENTAL IMPACT

BASIS AND REFERENCES

[1] IAEA Safety Services No.1, INSARR Guideline 1997

[2] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, Section A.14

[3] Impact Analysis for continued operation of the licensed facility.

ISSUE CLARIFICATION

The assessment of environmental impacts has to prove that the exposure limits for the public to effluents and external exposure are not exceeded. Thus the assessment has to be based on the maximum permissible release of effluents according to the license and on the critical group of the public in the vicinity of the site.

OBSERVATIONS

The Impact Analysis (IA) was one of the conditions requested through the operating license in 1999. The license given by the Regulatory Authority to the Operating Organization was subject to certain conditions, and one of these conditions is that operation of the reactor is subject to an Impact Analysis in accordance with the Planning and Buildings regulations (PBL), before the end of 2004.

The Impact Analysis consists of two parts. The Environmental Impact Analysis is included in part one. This document was issued as an open document easy to understand for the public.

The IA was sent to the Norwegian Radiation Protection Authority (NRPA, Strålevernet) on July 2nd 2002. The report was then sent by NRPA as hearing material to the relevant departments and supervisory authorities, to regional and communal authorities, to local organizations (LO), local residents associations and environmental organizations, community and local population, which had the opportunity to make comments on this report to put forward their views.

NRPA has received a number of comments from central authorities, county councils and municipal councils, as well as interested organizations, environmental organizations and from the Swedish Naturvårdsverket (Environmental Protection Agency). The Institute of Energy Technology has also held open meetings with the local population around the facilities, both in Kjeller and in Halden. These meetings were announced in the press, and invitations where also sent to local residents associations and environmental organizations.

The openness and interest that is shown by the increase in visits by school classes, associations and interested organizations to the Institute, not in the least to such arrangements as the "Open day" where people are given the chance to find out about the research reactors.

POSSIBLE SAFETY CONSEQUENCES

Public acceptance is an important factor for further operation of the research reactor. Pressure from the public may lead to early shut down of the facility.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The counterpart complied with the request of regulatory authority to develop an Impact Assessment for both reactors.

GOOD PRACTICE

GP1: The results of the Impact Analysis were presented to the authorities, community and local population which had a chance to put forward their views and their comments. This public hearing activity has a positive impact on public acceptance of nuclear activities of the site.

ISSUE ENV-02: ENVIRONMENTAL IMPACT

BASIS AND REFERENCES

[1] IAEA Safety Guide 35 G1- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report.

[2] Impact Analysis for continued operation of licensed facility at the Institute for Energy Technology, Kjeller and Halden, December 2004

[3] Impact Analysis, additional considerations, Part II, 24 December 2006

ISSUE CLARIFICATION

During phase two of developing of Impact Analysis, IFE processed the comments on Part I of this document. The Institute has been given a license by the Government to operate the research reactors at Kjeller and Halden until January 1st 2009, /1/. The purpose of the Impact Analysis is to clarify any effects of research reactor operation that can have important radiological consequences for the environment, natural resources and the community.

OBSERVATIONS

In part II, the radiological consequences of both normal operations and of serious accidents are considered. The natural surroundings around the facilities have been monitored during several decennia, statistics of radioactive releases are presented, and information about the radioactive environment in the area around the facilities is presented.

The chapter on accident analysis was changed taking into consideration more types of accidents and more types of failures. This assessment is not included in the present version of the Safety Analysis Report, which is the base for the next license renewal.

The counterpart mentioned that the next update of the SAR will be done in 2009 after the new license will be issued. The accident analysis developed in Part II of the Impact Analysis will be revised and will be added in that new SAR version.

The accident analysis presented in this document is based on the US NRC Regulatory Guideline no.1.183 “Alternative Radiological Source Term for evaluating design basis accidents for BWR”.

Impact Analysis is basically devoted to one selected accident scenarios – guillotine rupture of the lower part of primary coolant circuit plus emergency cooling system complete failure (called the reference accident) plus superposition of failure of filter and ventilation system. This was a requirement of NRPA. The reason to do so is to show the public how the facility is safe even with all safety protection systems out of operation. The results of calculations with such assumptions should be reviewed carefully implementing best estimate approach

POSSIBLE SAFETY CONSEQUENCES

The result of the accident analysis may lead to modification of OLC and as a consequence to violation of the license conditions.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The SAR will be updated in 2009 after the new license will be issued. Part II of the Impact Analysis will be included in the new version of the SAR.

RECOMMENDATION

R1: The SAR should be completed with part II of the Impact Analysis. According to IAEA SS 35 G1, para.205 one of the ways in which the operating organization demonstrates that it has achieved adequate safety is through the information normally incorporated in the SAR.

SUGGESTION

S1: The development of the SAR should be based on the IAEA Safety Standard 35 G1-Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report and the new Safety Report Series on Source Term Derivation and Radiological Consequences Analysis of Research Reactor Accidents, 2007.

COMMENT

C1: The institute should have Health, Environment and Safety indicators clearly defining the following thresholds:

- Regulatory limits;
- Control level limits; and
- Administrative limits.

Commonly used international practice is, e.g.

- 10 mSv;
- 20 mSv; and
- 5 mSv (ALARA goal).

ISSUE LER-01: LIFETIME ASSESSMENT OF REACTOR VESSEL

BASIS AND REFERENCES

[1] Safety Analysis Report – Part IV chap. 4

[2] Halden Reactor Vessel Surveillance 2006 – n° VTT-R-10329-06

[3] Halden Reactor Vessel Surveillance 2006, study of an additional Sample – n° VTT-R-00912-07

[4] Oppdatering av beregnede verdier for noytronbestraling av reaktortanken ved HBWR pr. Oktober 2006 - ROE-HBWR-025

[5] Material surveillance program for the pressure vessel at HBWR - ROE-HBWR-158

[6] Trykk og temperaturgrenser for reaktordrift - ROE-HBWR-159

[7] Prosedyre for bestemmelse av trykk og temperaturgrenser for reaktordrift - QA-P-855

[8] Temperaturberegninger for reaktortanken ved LOCA-situasjon - FC-Note 0000

[9] Kontrollrapport ITO-HBWR-070522-1

[10] ASME code Appendix G – article G-2000

ISSUE CLARIFICATION

The reactor vessel uses as material ferritic steel which is very sensitive to irradiation effects.

The base material reference temperature (brittle/ductile transition for non irradiated material) was reduced from 47°C to -37°C obtained by using master curve instead of Charpy testing methodology.

The reactor vessel is now about 50 years old and was the subject of only 25 years full power irradiation.

It should be noted that the knowledge acquired from nuclear power reactors is not fully relevant as the material and the operating conditions (temperature, neutron flux) for the HBWR reactor vessel material are different from those of light water power reactors.

Complete closure of the isolation valves installed on the reactor vessel is necessary in order not to exceed in absolute value the temperature rate (-10°C/h) used in the demonstration of the integrity of the vessel in thermal transient conditions corresponding to categories A, B, C (normal, upset, emergency). Indeed in case of failure of the complete closure of the isolation valves, the pressure temperature shock (PTS) could be higher.

From the beginning, many samples of the vessel material and welding were irradiated in the framework of a surveillance programme related to this structure. In addition, In Service Inspection Programme was established and implemented in accordance with ASME Code.

The main issue is the conservation of the mechanical integrity of the vessel in the future taking into account the effect of irradiations.

OBSERVATIONS

See issue clarification.

POSSIBLE SAFETY CONSEQUENCES

Loss of the mechanical integrity of the HBWR reactor vessel in the future due to ageing effects related to irradiation.

COUNTERPART VIEWS AND MEASURES ON THE FINDINGS

The Operating Organization has no comment on this issue.

RECOMMENDATIONS

R1: Due to the age of the reactor vessel, the embrittlement under irradiation and the different sources of uncertainties, the operator should improve as much as possible the prevention provisions to assure the integrity of the reactor vessel. In this regard, the operator should:

- continue to improve the knowledge about the behaviour of the reactor vessel material under irradiation;
- increase the frequency of controls and In Service Inspections in such a manner that 50% of the welds be controlled every three years and the totality of the welds controlled every six years instead of nine years currently applied;
- extend the control area of the reactor vessel to include a larger part of the base material around the heat affected zone.

These improvements should be implemented in order to have more confidence and guaranty for the conservation of the mechanical integrity of the vessel. Nevertheless, in the framework of the defence in depth principle, the consequences of a sudden rupture of the reactor vessel should be evaluated in order to verify that this event is not more severe than those already analysed in the SAR.

COMMENTS

C1: In application of the redundancy to avoid the reactor vessel thermal shock, the operator has already planned the implementation, before the end of the year, of second isolation valves on the main primary pipes and a second safety valve on the reactor vessel in the next year.

C2: Concerning category D (faulted), the operator started some studies with consideration of several leakage rates, but without taking into account the emergency core cooling system. So a detailed safety analysis, taking into account the results of the above mentioned studies, should be carried out in order to demonstrate the integrity of the reactor vessel in category D. This analysis should consider the operation of the ECCS in this situation as it could lead to more severe pressure temperature shock.

C3: Application of the leak before break (LBB) concept could be useful to prevent a complete rupture of the reactor vessel as the detection of tritium leakage is very sensitive and efficient.

APPENDIX 3: AGEING OF THE REACTOR VESSEL UNDER IRRADIATION

Effects of irradiation on the mechanical characteristics

The mechanical properties of steel material, in particular ferritic steel used in the reactor vessel, change under irradiation. The irradiation damage provokes an increase of the hardness as well as an increase of the traction characteristics. On the other hand, this increase also comes with an embrittlement of the material. This embrittlement under irradiation appears by a reduction of the toughness of the steel vessel, that means that the critical stress under which a brittle rupture can be triggered at a given temperature decreases with the significance of the irradiation, therefore with the duration of the reactor operation. This embrittlement leads to an increase of the transition temperature that separates the brittle behaviour of the steel from the ductile behaviour.

Issues

The design of the vessel thickness that relies on the observance of an allowable stress defined from the steel traction characteristics is not brought into question by the damage irradiation. But this embrittlement must remain acceptable until the end of the reactor life, to lead to a toughness of material sufficient to allow the justification of the resistance to abrupt rupture of the vessel under all operating situations, considering adequate safety margins. Indeed the rupture assumptions of the vessel have not been taken into account in the design of different reactors for the definition and the design of the back-up systems. So the embrittlement due to the irradiation must not under any circumstances lead to a risk of rupture of the vessel at the time of a pressure thermal shock (PTS) transient. In other words, the main issue is the conservation of the mechanical integrity of the vessel in the future taking into account the effect of irradiations.

However, it should be noted that the knowledge acquired from nuclear power reactors is not fully relevant for the HBWR reactor as the material and the operating conditions (temperature, fast neutron flux) for the reactor vessel material are different from those of light water power reactors.

Method for keeping the vessel in service

The method is mainly based on US NRC Regulations 10 CFR and on ASME code section XI. The safety demonstration is based on the calculation of the margins with respect to the brittle and ductile rupture risk. In other words, these margins are between the toughness of the vessel material and the stress intensity factor that characterizes the stress field at the level of defect considered under the given loading. For all operating situations and all points of the structure, it must be verified that the safety coefficients are greater than the minimum values required.

The key points are the following:

1. A good knowledge of the initial material characteristics and their evolution under irradiation,
2. The fast neutron fluence evaluation,
3. The verification that an assumed defect has safety coefficients with respect to an abrupt rupture under all situations (PTS transients).

Concerning the first point, a material surveillance programme was established as early as 1958 and was implemented in parallel with the construction of the pressure vessel. Many samples of the original vessel material have been irradiated with an anticipation factor in order to know in advance what the characteristics of the vessel steel will become. Then, Charpy tests allowed establishing the brittle/ductile transition curves which are used to assess the toughness of the reactor vessel material. A revised material surveillance programme was established in 1988. It was based on the need to supplement the initial programme with respect to an increase of the irradiation. More recently in 2006, direct toughness measurements have been made in order to reduce uncertainties and conservatisms. The criteria for the surveillance of the vessel are:

- Respect of ASTME 185-82,
- Samples must be located close to the vessel wall in order to represent correctly the neutron spectrum and the temperature.

According to the regulation, the reactor pressure vessel material should be proved 10 years ahead of the next operational permit.

The acquisition of data in a continuous process in order to increase the knowledge and the use of the more accurate testing method is a good practice. Indeed, the last results show the very important conservatism of the evaluation of the toughness with the Charpy testing. Based on recent testing results (9 measurements on non irradiated base material), the reference temperature was significantly reduced from 47°C to -37°C. Then new experiments in combination with previous results confirm this. Indeed, the testing performed prior to 1988 showed that the weld has a considerably lower reference temperature than the base material and the HAZ (heat affected zone). According to this revised material surveillance programme, the last material testing was performed in 2006, and the following are forecast in 2009, 2015, 2021, 2027 and 2033. Of course, this programme has to be carried out and periodic verification should be made using the new measurement method. As many samples are available, the operator should take the best benefit from all the samples to improve the knowledge on the vessel material behaviour. So, in order to increase the knowledge about the embrittlement of the reactor vessel and to have more confidence and guaranty in the conservation of the mechanical integrity of the vessel, the test period should be evaluated and implemented on basis of the test results obtained in 2009. An important operational limitation linked to the reference temperature is the minimal temperature of the primary coolant: 70°C. As many other operational limitation, the basis of this value has to be justified.

Concerning the fluence evaluation of the reactor vessel and of the samples (item 2 above mentioned), the table 1 gives the current status. These evaluations are mainly based on calculations. Even if few measurements have been done, uncertainties remain; factors between 1.2 and 1.5 depending on the core and the position in the vessel. Since 1988, calculations take into account the real core history and are updated after each core reloading. Nevertheless it is must be assured that the uncertainties develop in the same way for the reactor vessel and the samples and so that the fluence evaluation of the reactor vessel and of the sample is conservative. Indeed, from a safety point of view, the fluence of the reactor vessel has to be under estimated and in the opposite side the fluence of the sample over estimated. The operator should justify the conservatism of the methodology used. Moreover, the lead factor of the samples which allow predicting the behaviour of the reactor vessel material above 10 years ahead of the next operational permit is upper than the values recommended by US standards (between 1 and 3).

The results of the test performed at the beginning of the reactor life show that an important lead factors under estimate the effect of the irradiation. Those uncertainties and effects are also in favour of new measurements in addition of those already planned in the surveillance programme as mentioned above.

Concerning the last point (item 3 above mentioned), the defect used in the safety analysis:

- presents dimensions (25 mm thick) greater than the detection threshold of the method by ultrasound used for the in service inspection programme of the vessels,
- is positioned in the maximum irradiation zone of the reactor vessel (hot point).

The safety analysis also considers a temperature rate of -10°C/h.

In the demonstration of the reactor vessel integrity, the analysis shows that the complete closure of the isolation valves installed on the reactor vessel (programme unit 1) induces less severe temperature rate in PTS transients corresponding to categories A, B, C (normal, upset, emergency) than without this closure. So the possibility of improvement of the reliability of this closure system should be done. The operator has already planned the implementation, before the end of the year, of second isolation valves on the main pipes in application of the redundancy principle.

Concerning category D (faulted), some studies with several leakage rates have been started, but without taking into account the emergency core cooling system. Nevertheless, a detailed safety analysis should be carried out in order to demonstrate the integrity of the reactor vessel in category D. This analysis should consider the operation of the ECCS which could induce a more severe PTS during the transient. It should be underlined that the operation of the ECCS has important consequences on the fuel behaviour in case of loss of coolant accident. It is obvious that there is a strong link between the integrity of the reactor vessel and the potential risk of melting of the core. The operator considers the leak before break (LBB) concept as an element to prevent a complete rupture. Indeed the detection of tritium leakage could be very sensitive and efficient for detecting vessel leakage.

For the previous studies, in order to assess the temperature rate during PTS transients, it could be useful to establish a thermo-hydraulic model of the reactor vessel and primary circuit. Such model which allows simulating transient with different failures should be qualified with the results of the significant experimental programme done at the earlier 60s. Such generally request is formulated during periodic safety review (PSR) in many countries: “reassessment of transients with up to date tools and knowledge”.

Nevertheless in the frame of the defence in depth principle, the consequences of the abrupt rupture of the reactor vessel should be evaluated in order to verify that this event is not more severe than those already studied in the SAR.

Moreover, due to the redundancy principle (described in IAEA Safety Standards) and in order to reduce the risk of reactor vessel over pressure, a second safety valve should be implemented. Such redundancy is requested by regulation in many member states (for example, European Directive 97/23, French ministerial decision ESPN, French decree 13/12/99) for pressurised systems or components. Those texts specify that the set value for the opening pressure of the safety valve should be equal or below the maximal allowable pressure. The French ministerial decision ESPN also specifies that a periodic re-qualification (pressure testing, verification of safety components, inspection ...) should be done every 10 years. The last pressure test of the reactor vessel has been performed in 2003. No more tests are planned according to Swedish regulation and ASME. The operator has already planned the implementation, before the end of next year, of a second safety valve on the reactor vessel in application of the redundancy principle.

In addition, a Service Inspection programme was established and implemented in accordance with ASME Code. This programme considers partial inspection every three years so that all the welds will be inspected every nine years. Considering the results of the above analysis, the age of the reactor vessel (about 50 years old even if it was the subject of only 25 years full power irradiation), and the fact that an abrupt rupture of the reactor vessel has not been taken into account for the definition and the design of the back-up systems, an improvement of the ISI programme should be implemented (for example, the control of 50% of the welds every three years instead of 1/4, 1/4 and 2/4 as done at present, and the extension of the area of control to a larger part of the base material).

<u>Summary</u>

Concerning the embrittlement of reactor vessel and its integrity, the objective of the above mentioned assessment is to improve:

- the knowledge about the behaviour of the HBWR reactor vessel material under irradiation (more data, best benefit from all the samples irradiated, ...),
- the surveillance (increase frequency, extended area of control, ...) and
- the reliability of the safety systems implied in the protection of the vessel (second safety valve, closure system, ...).

Table 1: Fluence – Current status, June 2007

Reactor vessel (June 2007):

	Fluence (10^{18} n/cm²)
Average of base material	3.13
Maximal of base material	8.17
Maximal of HAZ	3.29
Maximal of weld position	3.29

Samples results: base material

	Accumulation rate (10^{18} n/cm² per year)	Lead factor	Fluence (10^{18} n/cm²)	ART_{NDT} (°C)	ART_0 (°C)	Years of equivalence to fluence of vessel
Irradiated material	0.14	0.60	4.36	167	83	< 1988
	0.49	2.08	6.51		131	< 1988
	0.73	5.12	9.98		151	2045
Initial value	0.00	-	0.00	47 *	-37 **	-

Charpy method (*) has been used before 2006, then direct method (**) is used.

Forecast (direct method): base material

	Accumulation rate (10^{18} n/cm² per year)	Lead factor	Fluence (10^{18} n/cm²)	Years of equivalence to fluence of vessel
2009	0.39/0.49	3.60	9.23	2030
2015	0.08	3.15	9.72	2040
2021	0.08	2.88	10.20	2050
2027 ***	0.13	2.78	6.68	2061
2033	0.08	2.68	10.60	2060

*** HAZ (135°)

APPENDIX 4: RECENT OPERATIONAL OCCURENCE

1. Exceeding the notification level for release of tritium to water.

For the purpose to make it possible for drainage of the experimental loops 6 and 8 under the reactor shut down in week 17 of 2007, water was emptied to the dump tank that already contain 0,87 TBq tritium (87% of notification level of tritium to water) 21 April 2007. The water that was released was from emptying several experimental loops under week 11 and week 12. The reason for the high concentration of tritium in the water was supposed originates from a leakage from the He-3 system in one test rig in the experimental loop 6. During operation the pressure in the experimental loop is always higher than the pressure in the He-3 system. No changes in pressure that could relate to a leakage were indicated during operation.

The most likely reason is that the leakage to the experimental loop was during a test of the experimental rig under week 12. During this test the He-3 system have a higher pressure than the experimental loop. The test of the rig-/loop system was carry through with He-4 gas but the pipes that relates to the system was contaminated with tritium from the He-3 system. Most likely tritium can have been transported into the experimental loop this way.

Before further use the experimental rig will be properly tested.

Earlier release of tritium to water was 18 % of the notification level. The total level have after this increased to 105 %. In addition water with a tritium content of 1.1 TBq will be released after drainage of experimental loop 6. This will increase the total level to 215% of the notification level. The total release together with this occurrence is 1.79 TBq of tritium. This means a dose of 0.00039 μSv to one person in the critical group and this means 0.039% of the total limit for release to water.

2. Fuel failure in a test assembly at Halden reactor

On 28 January 2001 lack of cooling resulted in failure of several fuel rods in a test assembly in the Halden reactor. This resulting in contamination of the primary circuit and lead to very high radiation levels in the containment of the reactor. There were no releases of radionuclides to the containment or to the environment around the research reactor. In the subsequent handling of failed fuel and bringing the reactor back to a normal state, the primary aim was to keep radiation doses to the workers and releases to the environment as low as possible. Every option was analyzed and carried out according to these criteria. Fourteen days after the incident, shielding and disintegration had brought the dose rate in the working area down to 0.05 to 0.1mSv/h. The primary circuit and the purification system had been fully operational throughout the period, and the activity concentration in the primary circuit water was brought down to values comparable to values experienced in shut down periods before incidents. It was therefore considered justifiable to start the clean up operation. The collective dose from the whole operation was 0,052 manSv, which is 10% of the average annual collective dose at the Halden reactor, and there were no additional releases of radionuclides to the environment.

ANNEX 1: LETTER REQUESTING THE MISSION

746-J7.05.1

PERMANENT MISSION OF NORWAY
TO THE INTERNATIONAL ORGANIZATIONS IN VIENNA

045587

Mr. Ken Brochman, Division Head
Division of Nuclear Installation Safety
Department of Nuclear Safety and Security
International Atomic Energy Agency
P.O. Box 100
1400 Vienna

Vienna, 23 November 2006

Dear Mr Brochman,

I have the honour to refer to the Agency's http://www-ns.iaea.org/reviews/rr-safety-reviews.htm - amat#amatIntegrated Safety Assessment of Research Reactors (INSARR) programme. We are aware that this programme can be useful in assisting States in ensuring and enhancing the safety of operating research reactors. In that respect, my government would like to request that the Agency conduct an INSARR mission to Norway at the earliest opportunity.

Please accept, Sir, the assurances of my highest consideration.

4

ACTION
27. NOV. 2006

Bengt O. Johansen
Ambassador Perm. Representative

M: Guerpinar
Viktorsson

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ANNEX 2: AGENDA

SUNDAY 17 June	
14:00 – 15:30	**Briefing for the team members: INSARR Methodology: INSARR Structure, INSARR Reporting, General recommendations including logistic and security.** H. Abou Yehia (Mr.)+ C. Ciuculescu (Ms.)
15:30- 16:00	**Break**
16:00 – 20:00	**Comments on available materials – SAR and Technical Report** Each expert prepared 20 min. presentation on safety issues identified from the material received as advance information Fred Wijtsma (Mr.) Denis Rive (Mr.) Lennart Gustafson (Mr.) Sergey Morozov (Mr.) Ricardo Waldman (Mr.)
20:00 – 21:00	**Preparation of presentation on main safety issues identified from advance information - all**
MONDAY 18 June	
8:30-9:00*	**Transfer from the Hotel to NRPA**
09:00 - 10.30	Entry meeting: NRPA Staff, Review Team, Counterparts;
10:30 – 11:00	Break
11:00 – 11:30	IAEA Programme on the Safety of Research Reactor (IAEA Structure for RR, IAEA SS, Services – INSARR mission, Code of Conduct) H. Abou Yehia
11:30 – 13:00	Presentation of NRPA on the expectation of INSARR Mission
	Presentation on Regulatory Framework in Norway
	Presentation on Regulatory Supervision of Halden Research Reactor (license, conditions, mandatory regulatory documents, inspection programme)
13:00 – 14:00	Lunch
14:00	Departure to Halden Site
14:00 - 16:00	Transfer to Halden Site
17:00 – 19:00	Team members assembly to discuss the schedule for the following day

TUESDAY 19 June	
8:45-9:00*	**Transfer from the Hotel to Halden RR**
09:00-10.30	Entry meeting: Opening Presentation of Review Team, Counterpart, Senior Management of Halden Research Reactor Presentation of the purpose of the mission
10:30-10: 45	COFFEE BREAK
10:45:12:30	Walk through the facility, • Reactor Hall: area monitors; reactor tank; visual examination of internals, water quality; thermal column; shielding plugs of the beam ports; • Control Room: control console, radiation protection monitors; radiation panel; • Explanation of star-up and operation procedures • Primary circuit – pumps room; heat exchangers; • Ventilation system: pumps, filters, stack monitor; pressure indicator panel; • Secondary circuit – pumps, cooling towers; water purification system; • Radioactive Waste drainages system; liquid waste storage; • Diesel Generator, UPS system
	Review Team (RT): All Counterparts (CP): All
12:30-13:30	**Lunch break**
13:30 – 14:00	Plenary meeting – IFE presentation on the progress of the study on lifetime extension of reactor pressure vessel

14:00 – 15:20	Regulatory Supervision	Impact Analysis for continues operation of licensed facility	SAR- Part III Chapter 1 Introduction Chapter 2 Definition of reactor Chapter 4 Operating personnel	Calculation on lifetime extension of reactor pressure vessel
	RT: Sergey Morozov CP: Mr. A. Valseth	RT: L. Gustafson + C.Ciuculescu CP: Mr. T. Walderhaug	RT: F. Wijtsma CP: Mr. S. E Christiansen	RT: D. Rive +Ricardo Waldman CP: Ms. L. Gjønnes
15:20-15: 40	COFFEE BREAK			
15:40-17:00	Impact Analysis for continues operation of licensed facility	SAR-Part III – Chapter 3 Administrative Control Chapter 5 Facility Regulations Chapter 6 Safety Standards	Calculation on lifetime extension of reactor tank	
	RT: L. Gustafson + C. Ciuculescu CP: Mr. T. Walderhaug + Mr. A. Valseth	RT: S. Morozov and F. Wijtsma CP:Mr. T. Elisenberg + Mr. S-E. Christiansen	RT: D. Rive + R. Waldman CP: Ms L. Gjønnes	
17:00-17:30	**Transfer to the Hotel**			
19:00-21:00	**Finalization and discussion on issue pages within the team**			

WEDNESDAY 20 June				
8:00-8:45	**Morning meeting of the review team –Briefing on previous day findings**			
8:45 -9:00	**Transfer from the Hotel to Halden RR**			
	Team Leader briefing to the Main Counterpart on previous day findings			
09:00- 10:30	SAR - Part I - Chapter 1 –Introduction Chapter 2 – General Description Chapter 3- Reactor	SAR-Part I Chapter 5 – Heavy Water Circuits Chapter 6 – Light water circuits	SAR Part II Chapter 1 – Introduction Chapter 2 –Summary Chapter 3 –Nuclear Core Parameters	SAR- Part IV Chapter 1- Introduction Chapter 2- Design, testing and operational experience Chapter 3 – Facility safety features
	RT: Sergey Morozov CP: Mr. S. E Christiansen + Mr. G. Mjønes	RT: L. Gustafson + C. Ciuculescu CP: Mr. P. Thowsen + Mr. J.S. Mjølnerød + O.M.Davidsen	RT: Ricardo Waldman CP Ms. L.A. Moen + Mr. W. Wiesenack	RT: F. Wijtsma + D. Rive CP: Mr. T. Elisenberg + Ms. L. Gjønnes

10:30-10:50	**Coffee break**			
10:50-12:30	Operating Organization	SAR-Part I Chapter 13 Radioactive waste – handling and disposal	SAR Part II Chapter 4 – Core Reactivity Characteristics Appendix: Codes for core physics calculations	Chapter 4 – Maintenance and Inspection
	RT: Sergey Morozov+H.Abou Yehia CP: Mr A. Valseth + Mr. T. Elisenberg	RT: L. Gustafson + C.Ciuculescu CP: Mr. T. Walderhaug	RT: Ricardo Waldman CP:Ms. L.A. Moen + Mr. W. Wieseanack	RT: F.Wijtsma + D.Rive CP:Mr. G. Mjønes + Ms. L. Gjønnes
12:30-13:30	**Lunch break**			
13:30 – 15:00	Safety Committee	SAR – Part I Chapter 12 – Handling and storage of fuel	SAR – Part I Chapter 7 – Containment	
	RT: Sergey Morozov CP: Mr A. Valseth	RT: + F. Wijtsma + C. Ciuculescu CP: Mr. G. Mjønes	RT: Ricardo Waldman + L. Gustafson CP: Mr. T. Elisenberg + Mr. P. Thowsen	
15:00-15:20	COFFEE BREAK			
15:20 – 17:00	Act No.28 Concerning Nuclear Energy Activities	SAR – Part III Chapter 9 - Transport of radioactive materials	SAR – Part I Chapter 4 – Control Stations Chapter 8 – Control and Instrumentation	Assessment of the results on the study on lifetime extension of reactor pressure vessel
	RT: Sergey Morozov CP:Mr. A. Valseth	RT: L. Gustafson + C.Ciuculescu CP:Mr. T.A. Gulbrandsen + Mr. T. Elisenberg	RT: Ricardo Waldman + F.Wijtsma CP:Mr. T.L. Skorpen + Mr. P.A. Wiig	RT: D.Rive CP: Ms. L. Gjønnes
17:00-17:30	**Transfer to the Hotel**			
19:00-21:00	**Finalization and discussion on issue pages within the team**			

THURSDAY 21 June	
8:00-8:45	**Morning meeting of the review team –Briefing on previous day findings**
8:45 -9:00	**Transfer from the Hotel to Halden RR**
	Team Leader briefing to the Main Counterpart on previous day findings

09:00- 10:30	Health, Safety and Environment Manual	SAR – Part IV Chapter 5 – Radiation Protection and activity release	SAR – Part I Chapter 9 – Main process computer system
	RT: Sergey Morozov CP:Mr. A. Valseth	RT: L. Gustafson + C.Ciuculescu CP:Mr. T. Walderhaug	RT: R.Waldman + F.Wijtsma + D.Rive CP:Mr. A. Rånes + Mr. P.A. Wiig
10:30-10:50	**Coffee break**		
10:50-12:30	Strategy for HSE	Airborne and Liquid Effluents	SAR – Part I Chapter 10 – Power Supply
	RT: Sergey Morozov CP: Mr. A. Valseth	RT: L. Gustafson + C.Ciuculescu CP: Mr. T. Walderhaug	RT: R.Waldman+ F.Wijtsma + D.Rive CP:Mr. H.A. Vangseng + Mr. T. Skorpen
12:30-13:30	**Lunch break**		
13:30 – 15:00	Training and Qualification	SAR – Part III Chapter 8 Emergency Preparedness Plan	SAR – Part I Chapter 11 – Experimental circuits
	RT: Sergey Morozov CP:Mr. T.Elisenberg + Mr. P. Thowsen	RT: L. Gustafson + R.Waldman CP :Mr. A.Valseth + Mr.T. Walderhaug	RT: F.Wijtsma+ C.Ciuculescu + D.Rive CP:Mr. Mr.T. Skorpen + Mr. R. Suther
15:00-15:20	COFFEE BREAK		
15:20 – 17:00	SAR – Part III Chapter 4 – Operating personnel and, Chapter 10 – Operating Certificates Appendices A, B, C	SAR – Part I Chapter 14 – Fire Protection	OLC for commissioning Commissioning after modifications Part I – Chapter 15 Limitations Part II – Chapter 8 Operational limitations
	RT: Sergey Morozov CP:Mr. Mr. P.Thowsen	RT: L. Gustafson, F. Wijtsma CP:Mr. O.M. Davidsen	RT:R. Waldman+ F.Wijtsma+D.Rive+H.Abou Yehia CP:Mr. Mr. S.E.Christiansen + Mr.T Elisenberg
17:00-17:30	**Transfer to the Hotel**		
19:00-21:00	**Finalization and discussion on issue pages within the team**		

FRIDAY 22 June	
8:00-8:45	**Morning meeting of the review team –Briefing on previous day findings**
8:45 -9:00	**Transfer from the Hotel to Halden RR**
	Team Leader briefing to the Main Counterpart on previous day findings

09:00- 10:30	Conduct of Operation	LRS Rescue Action Plan	SAR Part II Chapter 5 -Control rod characteristics	SAR – Part III Chapter 7 – Operating Manual
	RT: Sergey Morozov CP:Mr. T. Elisenberg +Mr. P.Thowsen	RT: L. Gustafson CP:Mr. A. Valseth	RT:R.Waldman+ D.Rive CP: Ms. L.A. Moen	RT: F.Wijtsma + C.Ciuculescu CP:Mr. T. Thowsen + Mr. R. Kirkerød
10:30-10:50	**Coffee break**			
10:50-12:30	Safety Culture	Contingency Plan for IET	SAR Part II Chapter 6 – Reactor Stability – Reactivity feedback	SAR – Part III Chapter 11 – Operating Data
	RT: Sergey Morozov CP:Mr. T. Elisenberg +Mr. A. Valseth	RT: L. Gustafson CP:Mr. T. Walderhaug	RT:R.Waldman+ D.Rive CP:Ms. L.A. Moen	RT: F.Wijtsma +C.Ciuculescu CP: Mr. T. Thowsen + Mr. R. Kirkerød
12:30-13:30	**Lunch break**			
13:30 – 15:00	Quality Manual	Decommissioning Plan for the Nuclear Facilities for IFE	SAR Part II Chapter 7 – Thermal and hydraulic characteristics	Operating Procedures
	RT: Sergey Morozov CP:Ms U.Friis Skogstad +Mr C-V. Sundling	RT: L. Gustafson + C.Ciuculescu CP:Mr. A. Valseth	RT:R.Waldman + D.Rive CP : Ms. L.A. Moen	RT: F.Wijtsma CP:Mr. T. Thowsen + Mr. R. Kirkerød
15:00-15:20	COFFEE BREAK			
15:20 – 17:00	Quality Manual	Decommissioning Plan for the Nuclear Facilities for IFE	SAR – Part II Chapter 8 –Operational limitations Part I – Chapter 15 Limitations	Operating Procedures for experiments
	RT: Sergey Morozov+ RT: F.Wijtsma CP: Ms U.Friis Skogstad +Mr C-V. Sundling	RT: L. Gustafson+ C. Ciuculescu CP: Mr. A. Valseth	RT:R.Waldman + D.Rive CP: Ms. L.A. Moen	RT: F.Wijtsma CP: Mr. T. Thowsen + Mr. R. Kirkerød
17:00-17:30	**Transfer to the Hotel**			
19:00-21:00	**Finalization and discussion on issue pages within the team**			

MONDAY 25 June	
8:00-8:45	**Morning meeting of the review team –Briefing on previous day findings**

8:45 -9:00	**Transfer from the Hotel to Halden RR**		
	Team Leader briefing to the Main Counterpart on previous day findings		
09:00- 10:30	Clarifications on Safety Culture	SAR – Part IV Chapter 6 Incidents	Maintenance Periodic Testing and Inspection Programme
	RT: Sergey Morozov + L. Gustafson CP:Mr. T. Walderhaug + Mr. A. Valseth	RT: R.Waldman + C.Ciuculescu CP: Mr. T. Elisenberg + Mr. T. Skorpen	RT: F.Wijtsma + D.Rive CP: Mr. G. Mjønes + Ms. L. Gjønnes
10:30-10:50	**Coffee break**		
10:50-12:30	Safety Committe		SAR – Part IV Chapter 7 Accidents
	RT: Sergey Morozov+ L. Gustafson RT: F. Wijtsma + D.Rive+H.Abou Yehia CP: Safety Committee members		RT:R.Waldman + C.Ciuculescu CP: Mr. T. Walderhaug + Mr. T. Elisenberg
12:30-13:30	**Lunch break**		
13:30 – 15:00	Radiation protection Act No.36 Regulation No.1362 Radiation Protection Annex to Regulations on Radiation Protection	Clarifications and further discussions on Emergency Preparedness	Ageing Management Programme
	RT: Sergey Morozov+C.Ciuculescu CP: Mr. T. Walderhaug	RT:R.Waldman + L. Gustafson CP: Mr. A. Valseth + Mr. T. Elisenberg	RT: F. Wijtsma + D.Rive CP:Ms. L. Gjønnes
15:00-15:20	COFFEE BREAK		
15:20 – 17:00	Radiation Protection Programme		Ageing Management Programme
	RT: Sergey Morozov+L. Gustafson +R.Waldman CP: Mr. T. Walderhaug+ Mr. A. Valseth + Mr. T. Elisenberg		RT: F. Wijtsma + D.Rive+H.Abou Yehia CP: Ms. L. Gjønnes
17:00-17:30	**Transfer to the Hotel**		
19:00-21:00	**Finalization and discussion on issue pages within the team**		

TUESDAY 26 June	
8:00-8:45	**Morning meeting of the review team –Briefing on previous day findings**
8:45 -9:00	**Transfer from the Hotel to Halden RR**
	Team Leader briefing to the Main Counterpart on previous day findings
09:00- 10:30	Development of issue pages, clarifications on safety issues identified; Counterparts view on safety issues and recommendations
	RT: All CP: As need for clarifications
10:30-10:50	**Coffee break**
10:50-12:30	Development of issue pages, clarifications on safety issues identified; Counterparts view on safety issues and recommendations
	RT: All CP: As needed for clarifications
12:30-13:30	**Lunch break**
13:30 – 15:00	Development of issue pages, clarifications on safety issues identified; Counterparts view on safety issues and recommendations
	RT: All CP: As needed for clarifications
15:00-15:20	Coffee break
15:20 – 17:00	Development of issue pages, clarifications on safety issues identified; Counterparts view on safety issues and recommendations
	RT: All CP: As needed for clarifications
17:00-17:30	**Transfer to the Hotel**
19:00-21:00	**Finalization and discussion on issue pages within the team**

WEDNESDAY 27 June	
8:00-8:45	**Morning meeting of the review team –Briefing on previous day findings**
8:45 -9:00	**Transfer from the Hotel to Halden RR**
	CP: As needed for clarifications

09:00- 10:30	Work on mission report Clarifications on safety issues. RT: All CP: As needed for clarifications
10:30-10:50	**Coffee break**
10:50-12:30	Work on mission report Clarifications on safety issues. TM: all CP: as needed
12:30-13:30	**Lunch break**
13:30 – 15:00	Review Mission Report TM: all CP: as needed
15:00-15:20	COFFEE BREAK
15:20 – 17:00	Preparation of the most important safety issues to be highlighted during the exit meeting Each member of the review team TM: All CP: as needed for clarifications
17:00-17:30	**Transfer to the Hotel**
19:00-21:00	**Finalization and discussion on issue pages within the team**

THURSDAY 28 JUNE	
8:00-8:45	**Morning meeting of the review team –Briefing on previous day findings**
8:45 -9:00	**Transfer from the Hotel to Halden RR**
09:00-13:00	**Exit meeting at Halden RR** Briefing on the Mission Report content Concluding Remarks Review Team, Counterpart, Senior Management of Halden RR, Regulatory body (senior staff)
13:00	**Mission Finalization at Halden RR**
13:00-14:00	**Lunch**

14:00–16:00	Transfer from Halden to Oslo
18:00-20:00	Team feedback

FRIDAY 29 JUNE	
8:00-8:45	**Morning meeting of the review team –Briefing on previous day findings**
8:45 -9:00	**Transfer from the Hotel to NRPA**
09:00-13:00	**Exit meeting at NRPA** Briefing on the Mission Report content Concluding Remarks – Review Team, Counterpart, Senior Management of Halden RR, Regulatory body (senior staff)
13:00	Mission Closure
13:00-13:30	Transfer to the hotel
13:30 –14:30	Lunch
14:30-16:30	Team feedback

ANNEX 3: IAEA REVIEW TEAM

Review Team (RT):					
Name	**Job Title/ Section/Organization**	**Education**	**Office Address**	**Telephone Number:**	**E-mail**
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ANNEX 4: ATTENDANCE LIST OF THE OPENING MEETING OF INSARR 2007

Name	Job Title
Hassan Abou Yehia	SH-RRSS, IAEA
Cristina Ciuculescu	RRSS, IAEA
Lennart Gustafson	Studsvik Nuclear AB, Sweden
Wolfgang Wiesenack	General Manager, IFE-HRP
Atle Valseth	Head of Safety, IFE
Thomas Elisenberg	Chief of Operation, IFE-HRP
Tord Walderhaug	Deputy Head of Radiation Protection, IFE-HRP
Ricardo Waldman	Nuclear Safety, ARN Argentina
Denis Rive	Institute for Radiation Protection and Nuclear Safety, France
Fred J. Wijtsma	Reactor Manger, Low Flux Reactor/High Flux Reactor
Sergey Morozov	Division Head for Safety Assessment, Licensing and Inspection of RR of the Federal Environment, Industrial and Nuclear Safety Supervision of Russia
Tonje Sekse	Safeguard Officer, NRPA
Håkan Mattsson	Advisor, NRPA
Sverre Hornkjøl	Acting Head of Section, Industrial and Research Applications of Radiation, NRPA
Ole Harbitz	Director General, NRPA
Gunnar Saxeeøl	Director of the Department of Radiation Protection and Nuclear Safety, NRPA

ANNEX 5: LIST OF COUNTERPARTS

Name	Job Title/Section	Phone Number	E-mail
Mr. Wolfgang Wiesenack	Project manager	**+47 69212347**	*wolfgang.wiesenack@hrp.no*
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ANNEX 6: LIST OF PARTICIPANTS TO THE SAFETY COMMITTEE MEETING

Name	Position	Organization
Fred J. Wijtsem	Reactor Manager HFR/LFR	NRG Petten, NL
Erik Rolstad	HRP, Division Head DPE	HRP
Hassan Abou Yehia	SH-RRSS	IAEA
Atle Valseth	Head of Safety	IFE
Nils T. Fördeströmmen	Senior Advisor	IFE
Thomas Elisenberg	Chief of Operation	IFE
Lennart Gustafson	Manager Decommissioning/Radiation Protection	SNAB
Håkan Mattson	Advisor	NRPA
Sergey Morozov	Division Head	Rostechnadzor
Arnulf Walström	Dep. Division Head EE	IFE

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[3] IAEA Fundamental Safety Principles, IAEA SF-1 (2006).

[4] IAEA Safety Standard No.115 International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, 1996

[5] IAEA Safety Guide, Safety Assessment and Preparation of the Safety Analysis Report for Research Reactors, SS No. 35-G-1, IAEA (1994).

[6] IAEA Safety Guide, Safety in the Utilization and Modification of Research Reactors, SS No. 35-G-2, IAEA (1994).

[7] IAEA Safety Report Safety Analysis for Research Reactors, Safety Report, IAEA (2007).

[8] Safety of New and Existing Research Reactor Facilities in Relation to External Events, IAEA Safety Report No.41 (2005).

[9] Standard Format and Content for Safety Related Decommissioning Documents, IAEA Safety Report No. 45 (2005).

[10] The Management System for Facilities and Activities, GS-R-3, IAEA (2006).

[11] Application of Management System for Facilities and Activities, GS-G-3.1, IAEA (2006).

B. Advanced Package of Information

[1] Health, Safety and Environment (HSE) Manual

[2] Strategy for HSE

[3] Part I – SAR – detailed design, construction and the mode of operation of the reactor and plant

[4] Part II - Nuclear and thermal characteristics of the core

[5] Part III – Operation

[6] SAR – Part IV - analysis of the adequacy of the inherent safety and the engineered safety system

[7] LRS Rescue Action Plan

[8] Contingency Plan

[9] Decommissioning Plan

[10] Decommissioning-Figures

[11] Impact Analysis

[12] Regulations No.1362, Radiation Protection

[13] Annex to Regulations on Radiation Protection

[14] Act No. 28 - Concerning Nuclear Energy Activities

[15] Act No. 36 – Radiation Protection

C. Documents received from the Counterpart during the Mission

[1] The operation license for IFE Atomic Facilities in Halden and Kjeller – ref.97/08715 FOS HEJ from 22.12.1999, the validity is from 01.01.2000 until 31.12.2009 and press release

[2] Inspection report no.2/04, 29.10.2004 as a result of the inspection performed on 28.10.2004 at HBWR.

[3] Minutes of Meeting between NRPA and IFE on licensing meeting on 14.02.2007 concerning clarifications related to the application documents (SAR part I, II, III, IV).

[4] NRPA Report to the Ministry of Health on the IFE application for the operation license of nuclear installations granted from 1st January 2000, Ref.number 97/00879/520.3/Sho.

[5] Application for license for continues operation of IFE nuclear installations after 2008. Ref. number KD-2006-01 from April 2006

[6] Halden Procedure Sv-krav-08/30.09.2004 – Limits of doses and the actions to limit the radiation exposure.

[7] Requirements on radiation protection related to the waste handling Sv-krav-06/24.05.2006

[8] Procedure for handling radioactive waste at HBWR – QA-P-852/30.11.2005

[9] Hard copy of list of radiation protection monitors, printed from electronic file of the Picasso software – Stv 23.09.2005 – ask unde which procedure/instruction is this file updated/or belong to

[10] The criteria for reporting and routines for reporting on radioactive releases – QA-RPI-33/16.04.2007

[11] Release to the delay tank in special conditions – QA-RPI-56/31.07.2005

[12] Handling instruction for cutting pipes and filling steel waste boxes and concrete boxes ROE-A-045/03/03/2006

[13] Basic rules and principles for handling radioactive materials – Sv-SAR-001/26.01.2006

[14] Control of the contamination of non-active waste to be discharged from the site – QA-RPI-111/28.07.2006

[15] Radiation protection education level 1- Rules for working in controlled area – QA-RPI 95/27.03.2007

[16] Overview on Radiation protection education at the HBWR – Level 1, 2, and 3 – QA-P-953/17.04.2007

[17] Flowchart no.379507, of the cooling system of the D_2O steam circuit and subcooler circuit

[18] Flowchart no.379244, of D_2O purification circuit and recombination circuit low and high pressure

[19] Draft flowchart on Off-gas and fission gas collection system – proposal for design modification - Drawing number is missing

[20] D2O Drain tank circuit - Drawing number is missing

[21] Closed secondary circuit - Drawing number is missing

[22] Flowchart of Feed water circuit - Drawing number is missing

[23] FWT- Water supply and steam heading - Drawing number is missing

[24] Main steam line - Drawing number is missing

[25] The raw water and coolant circuit outside the reactor hall - Drawing number is missing

[26] The raw water and coolant circuit in the rector hall and air lock - Drawing number is missing

[27] The raw water and coolant circuit in 1st-/2ndt floor and sink - Drawing number is missing

[28] Certificate of the Kejller cask – 15.02.2006 – ref.number 98/01097/305.5/Sho

[29] Checklist for instrumentation before starting the operation – QA-RPI-45/18.06.2007 – Draft checklist – no signature

[30] Listing of date/time/nuclides measured during an abnormal occurrence release case – 6 April 2007 - there is no reference to any procedure/computer programme

[31] Flowchart of Emergency Spray and Filter Circuit – checked and approved but no number.

[32] Example of Safety Document from Project Leader at Halden for a new metallurgical laboratory submitted to Safety Board in 1994 - pls. provide a similar document submitted after year 2000.

[33] Test results diagram and calculations for containment tests for leakage from the reactor hall/ 2007-14 to 15 January.

[34] Organization Chart for the Institute Emergency Preparedness – fig.1 from instruction 081

[35] Crisis committee organization chart on governmental level – no reference to any procedure or document

[36] Administrative instruction for the institute emergency organization – Ref. to emergency instruction 081 (explains the emergency organization levels and responsibilities in each level) – 2006/04.07

[37] Emergency plan level 2 at IFE Halden – Administrative document 081 ver. 4, appendix 4 from 10.03.2006

[83] Waste Handling. Doc.QA-RPI-35/07.03.2007

[84] Quality Assurance Book for Radiation Protection Department – QA-013-no/14.02.2006

[85] Information letter from IFE to NRPA for exceeding the Reporting limits for tritium/ 23.04.2007

[86] The permit for release of radioactive effluents to the Institute for Energy Technique. Doc.GO05-11/Application number 2005/00794 from January 2006 until 31 December 2009.

[87] Process description for design, production, installation and modification of systems at HBWR. DocROE-P-001/24.08.2005

[88] Radiation Protection Declaration for installation and modifications of systems at HBWR. Doc.Sv-krav 17/18.07.2005

[89] Calculation of release to air of Tritium and Noble Gases.Doc QA-RPI-73/27.11.2003

[90] Instrumental check before star-up. Doc.QA-RPI-45/18.06.2007.

[91] NRPA – Information Letter "License handling of IFEs facilities. The plan from autom 2006 until spring 2008". 21.09.2005.

[92] NRPA – Information Letter, Documents used in regulatory activities vs.IFE/19.06.2007

[93] Quality Manual for IFE.Doc IFE/10/10/00/23.06.2006. Approved by Kjell Bendiksen

[94] Certificate of qualification for suppliers to the Oil Industry awarded to IFE.No.959432538/01.07.2006

[95] Quality System contacts .Doc.QA-037/09.01.2007

[96] Halden Reactor Project (HRP) Quality Policy. Doc. QA-005/07.11.2005

[97] Instruction for the Head of Safety.KD-2006-07

[98] Instruction for Manager for Reactor Operation KD-2006-07-IV

[99] Orange Book for Halden. Quality Assurance System.QA-054/02.01.2007

[100] IFE Quality Assurance System for Department of Reactor Operation. QA-017/04.01.2007

[101] Plan for internal QA Programme Revision. Doc.KSRS-K07/26.01.2007

[102] Process description for early Quality Assurance Goals and Reporting of Quality.Doc IFE10/10/40/10.04.2007

[103] Project Handbook for IFE. Doc.IFE10.20.50/30.06.2006

[104] List of Safety Reports and Permits

[105] Central procedure for non-conformity handling. Doc. IFE 01/03/20.27.11.2006

[106] Instruction for registration of a non-conformity. Doc. IFE 01/03/20_27.04.2006

[107] Organization Plan for Staff and General functions/1st October 2006

[108] Administrative Instruction No.058 – Matters to be handled by Safety Committee. 01.04.2006

[109] Clarification of reactor facility and experimental loops before and after handling. Doc. RE-052_ver.11 – Draft document no data no signature

[110] Calibration and function testing of mobile instruments. QA-P-927/29.04.2003

[111] Pressure and Temperature limits for reactor operation ROE-HBWR-159/15.06.2007

[112] Organization Plan for NRPA, 18 June 2007

[113] Central procedure, process description for discrepancy handling, Doc. IFE 01/03/20, 27 November 2006

[114] Example of list of discrepancies, no date

[115] Example of discrepancy report, waste handling, Doc. A 395, 7 December 2006

[116] Example of discrepancy report, Doc. A 396, 7 December 2006

[117] Radiation Protection in the 2000s – Theory and Practice, Doc. STUK-A195, 25-29 August 2002

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[119] Halden Reactor Vessel Surveillance 2006, Study of an Additional Sample, Doc. VTT-R-00912-07, 31 January 2007

[120] List of feed-back reports, 29 February 2000 until 18 January 2007-06-28

[121] Control Report, Doc. ITO-HBWR-070522-1, 6 June 2007-06-28

[122] Updating calculation for neutron irradiation of the reactor tank at HBWR in October 2006, Doc. ROE-HBWR-025, Version 5, 29 March 2007

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