# POLICY ISSUE
## (Information)

December 27, 1999 SECY-99-290

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: NRC PLANS TO PARTICIPATE IN THE OECD HALDEN REACTOR PROJECT DURING 2000 - 2002

PURPOSE:

To inform the Commission of the results of the Agency's past participation in the Organization for Economic Cooperation and Development (OECD) Halden Reactor Project and staff plans to continue participation during 2000 - 2002.

BACKGROUND:

The Halden Reactor Project is a cooperatively funded international research and development project that operates under the auspices of the OECD - Nuclear Energy Agency with the sponsorship of 21 different countries. The research encompasses (1) nuclear fuels and materials performance and (2) process control development and qualification, which includes both human factors research and the test and development of advanced computerized operator support systems.

The Nuclear Regulatory Commission (NRC) has participated in the OECD Halden Reactor Project since its inception in 1958. During this period, the NRC has received the benefit of numerous research products from this internationally funded cooperative effort. The NRC uses Halden-generated products and information to develop and extend the applicability of analytical tools and as the technical basis for regulatory guidance. The staff plans to continue its participation in the 2000 - 2002 agreement period because of the benefits received and the leverage of resources by participation with 14 signature and 11 associate members representing 21 countries.

CONTACT: J. Persensky, RES
415-6759

The benefits derived from the past and continued participation in the OECD Halden Reactor Project include:

1. Testing of $UO_2$ fuel that has achieved very high burnup in the Halden reactor was done prior to increasing commercial burnup limits. This testing has also resulted in the quantification of important changes in thermal properties. Degradation in thermal conductivity, increases in fission gas release, and changes in gap conductance have all been measured and have been used to upgrade the NRC's fuel rod codes, FRAPCON and FRAPTRAN, for applications up to 65 GWd/t burnup. These codes are used for audit purposes during licensing reviews and for other special analyses.

2. Testing that has been initiated recently on properties of new fuel cladding alloys that are being introduced in U.S. plants to reach higher burnups with more reliability. Corrosion rates and creep rates for low-tin Zircaloy, Zirlo, and M5 cladding will be measured. These results will be used to update the NRC's fuel rod codes for application to the new cladding alloys.

3. Materials that have been irradiated at this facility have subsequently been provided for use by the NRC staff for performing tests to evaluate the effects of irradiation-assisted stress-corrosion cracking on reactor internals. Results from this work have been used by NRR for more accurate assessments of crack growth rates in irradiated components. This has resulted in licensee reduction in inspections and outage times.

4. Products from Halden formed part of the technical bases for the development of new regulatory guides on software quality and life-cycle, as well as review guidance such as revisions of the Standard Review Plan (Chapters 7 and 18) on Advanced Instrumentation and Controls, Man-Machine Interfaces, Software Quality Assurance, and Advanced Control Room Design Reviews.

5. The results of human factors research and experiments performed in the Halden Man-Machine Laboratory (HAMMLAB), which is operated by a highly qualified staff that plans and conducts experiments related to human error, human performance, and the effects of computer-driven interfaces. The results of this research serve as a part of the technical basis for regulatory guidance in areas such as alarm systems, hybrid control rooms, display navigation, control room staffing and measures of human performance. Halden's program on human error analysis is being developed to contribute to improved human reliability information. This will contribute to NRC's risk-informed regulatory activities.

6. Data and information from test and evaluation, led to the development of new review guidance for instrumentation and control (I&C) technology. This information has been and can be used to develop technical bases for realistic safety decisions that will prepare the Agency for the future by evaluating safety issues involving current and new designs and technologies. Some U.S. nuclear utilities have also expressed particular interest in the application of some of the Halden-developed operator support systems and their virtual-environments technology for the development and evaluation of hybrid control stations.

7. A forum for international cooperation and information exchange with 21 different countries.

8. International cooperative funding that provides leverage for the NRC research funds. The NRC gets the benefits of a $40M three-year research program for a contribution of $2.85M.

Attachment 1 is the Proposed Program Plan for the next three-year Joint Program Agreement. Attachments 2 is the Executive Summary of the "Achievements of the Halden Project Programme in the 1997 - 1999 Period," and Attachment 3 is the Executive Summary for the Man-Machine Systems and Fuel Performance and Materials Testing program.

DISCUSSION:

The OECD Halden Reactor Project (HRP) is an internationally funded and staffed nuclear research and development organization located in Halden, Norway. Currently there are 21 countries that cooperate in the HRP. A primary facility of the Halden Reactor Project is the Halden Boiling Water Reactor that currently operates at 18 to 20 Megawatts (MW) and is contained within a mountain. Norwegian authorities have re-licensed the reactor for an additional 10 years, to 2010. The reactor is fully dedicated to instrumented in-reactor testing of fuel and core materials behavior. It also delivers steam to a nearby paper factory.

The research programs at the Halden Project address two broad areas of interest to the NRC: 1) nuclear fuels and materials performance and 2) process control development and qualification, which includes both human factors research and the test and development of advanced computerized operator support systems. The programs are structured to respond to the needs of all member organizations within the international nuclear community. Since the initial startup, the reactor facility has been progressively updated and has now become one of the most versatile test reactors in the world. In the course of this development, over 300 in-reactor experiments have been performed.

Following the early development of in-core instruments on fuel rods, in-reactor testing of basic fuel rod reliability parameters was conducted with the objectives of defining performance data and exploring mechanisms. This in-core measurement capability has been expanded through the development of experimental rig and loop systems in which reactor fuel and materials can be tested under simulated water reactor conditions. Increasingly, fuel reliability and safety considerations are emphasized under the jointly financed program while development and optimization issues are addressed in bi-lateral programs with specific users. The Halden reactor has a large number of experimental channels in the core that are capable of handling many test rigs simultaneously. In the coming 3-year period, approximately 14 different investigations will be conducted on fuel high-burnup capabilities during normal operation, about six efforts will be initiated related to fuel response to transients, and another six studies will be undertaken on fuel reliability issues.

The Halden reactor is now one of the leading research facilities in the world for the study of irradiation assisted stress corrosion cracking (IASCC). The NRC directly benefits from the irradiations they have performed to support the IASCC work being done at Argonne National

Laboratory (ANL) for the NRC. Specimens have been irradiated for slow strain rate testing, crack growth rate testing, and fracture toughness tests at ANL. Halden is unique in its capability to perform the testing under prototypical conditions in a radiation environment. They have also developed unique approaches to fabricating specimens that make it easier to use materials irradiated in power reactors as well as test specimens irradiated in Halden.

As part of the Joint Program, Halden is also participating in a program of research on IASCC developed by the Cooperative International Research Program on IASCC (of which the NRC is a member). They are doing IASCC tests under uniaxial loading and biaxial loading and tests using fracture mechanics specimens.

The OECD Halden Reactor Project also maintains and is expanding one of the most comprehensive facilities in the world for performing experimental research on issues regarding the human-system interfaces for advanced technology in nuclear power plant control rooms. For nearly 20 years, the HAMMLAB has included a full scope VVER-440 simulator, based on the Loviisa Power Plant in Finland. Over the last 3 year-period, the project has acquired a Pressurized Water Reactor (PWR) simulator based on the Fessenheim Plant, a French 900 MW PWR, and is building a Boiling Water Reactor (BWR) simulator based on the 1160 MW Swedish Forsmark-3 Plant, as well as upgrading the VVER simulator. In addition, Halden has developed a set of advanced computerized operator support systems for control rooms, which it tests and upgrades continually. The HAMMLAB has a prototype advanced control room with an integrated surveillance and control system, which is used as a test bed for exploring human-machine issues regarding the role of the operator and interactions with advanced automated controls. The program has also developed a capability in the area of virtual environments, which has been used as a cost-efficient way to design control rooms for nuclear and other process control applications. These facilities are augmented by the largest human factors research staff in the international nuclear arena.

As part of the Joint Program, Halden also conducts research to assure and enhance the quality of computer-based systems. The research addresses topics covering various types of instrumentation and control systems as well as all the life-cycle phases of these systems. The development and application of both formal and conventional methods to verify and validate high integrity software are also subjects of research at the project.

The international organizations actively participating in the Halden project represent a cross section of the nuclear industry consisting of licensing and regulatory interests, national research organizations, reactor and fuel vendors, and utilities. Attachment 4 contains a list of the current members of the OECD Halden Reactor Project and a description of the Steering Bodies.

The results from the research conducted at Halden are distributed to the signatory and associate members in the form of technical reports. Halden publishes approximately 50 reports per year. There are also 3 to 4 workshop meetings per year on specific topics. About every 18 months, Halden conducts an Enlarged Halden Program Group Meeting. At these meetings, research results are presented from programs at Halden as well as from member countries. This provides the attendees with a window to international activities on fuel performance and materials testing, digital instrumentation and control system safety and reliability issues, and human factors issues associated with computer-driven interfaces.

The Halden experimental program is conducted in two ways. The first is the Joint Program, which is a program of work jointly agreed to by members of the project that is planned over 3-year periods and reviewed annually. This part of the program leverages the NRC's funds since all data and information generated from the Joint Program are available to the members for their use. The members can also establish bi-lateral agreements to conduct specific experimental programs. These bi-laterals are funded entirely by the sponsor and the data are only disclosed to other project members at the sponsor's discretion. The bi-lateral efforts and the Joint Program benefit from the synergy of the knowledge that the researchers gain from performing the research.

The Halden project, through its long operation, has proven to be highly versatile and responsive to the changes of research and development needs. It is a relatively small and non-bureaucratic operation where recommendations and priorities by members are accommodated in a flexible manner.

PLANS FOR WORK AT HRP DURING 2000 - 2002:

The NRC staff met with members of the Halden staff on October 20, 1998, and again on October 29, 1999, to discuss the next 3-year program as proposed in the draft report, "Halden Reactor Project Program, Proposal for the 3-Year Period, 2000 - 2002." This report contains proposals for research and development programs on Fuels and Materials and Man-Machine Systems.

The Fuel and Materials Program encompasses four major topics:

- Fuel High Burn-Up Capabilities, Normal Operating Conditions
- Fuel Response to Transients
- Fuel Reliability Issues
- Plant Lifetime Assessments, including IASCC

Experimental and analytical activities will be performed to assess fuel performance capabilities and property changes at high burn-up. The extensive use of refabricated and instrumented commercial fuels is anticipated. The proposed work on safety transients is intended to complement investigations carried out elsewhere on the loss-of-coolant and reactivity-initiated transients. Tests on short-term dry-out associated with anticipated transients without scram (BWR oscillations) are also contemplated. Investigations on fuel performance anomalies arising from current operational experience will be conducted in close collaboration with participants with the purpose of identifying realistic design or operational remedies. Tests are being planned in a sweep-gas rig on release rates of short-lived fission products, particularly iodine and cesium, to provide the basis for an overhaul of the industry standard, ANS-5.4, which describes a standard fission product release model. The NRC participates in this standards activity and uses the standard model for setting source terms for several design-basis accidents. The activities on plant lifetime assessments focus on stress corrosion cracking of structural reactor materials under the combined effect of water chemistry and radiation environment. The proposed program relies on the Halden in-reactor measurement capabilities, on rigs designed to produce a variety of test conditions, and on in-reactor water loops able to create flexible coolant environments.

The Man-Machine Program focuses on four areas:

- Experimental Program and Operation of HAMMLAB
- Human Factors and Control Room Engineering
- Plant Operational Support
- System Safety and Reliability

Computerized operator support systems are to be introduced in the upgraded HAMMLAB for testing to demonstrate the merits of such systems in an integrated control room environment. The activities on human factors and control room engineering area will extend the knowledge of human performance in process control environments and of how this can be incorporated in control room design review guidance. The research conducted in this area will address issues of human error and human reliability. The proposed work on plant-performance monitoring and optimization is intended to develop system solutions that have the potential to improve plant performance as well as operational safety. The aim of the activities on system safety and reliability is to provide methods to enhance the reliability of automated systems and, in particular, safety-critical software for digital I&C. The technical basis for the Man-Machine Program consists of the upgraded test facility HAMMLAB, together with the test methodology, the software, simulation, and control room expertise developed around it.

NRC PARTICIPATION IN 2000 - 2002 HALDEN PROGRAM:

The cost of participation during the 1997 - 1999 agreement period was 23M NoK (Norwegian Kroner) or $3.1M. The NRC's share according to an OECD formula is 23.5 NoK, for the 2000 - 2002 period. During the RES self-assessment and budgeting process $850K was allocated for this effort in FY-2000. It is expected that $1.0 M will be available in FY 2001 and 2002. Halden project management has agreed, that based on the current exchange rate and with payment early in the year, $2.85M would meet our obligation. As always, it is understood and the agreement will be signed with a statement that funding will be subject to the availability of appropriated funds.

Based on our latest contacts, we understand that all of the other member countries have either already signed the agreement or expressed their intent to extend their participation in the Halden Reactor Project for the 2000 - 2002 program period.

COORDINATION:

The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections. The Office of the General Counsel has no legal objection to this paper.

CONCLUSION:

The NRC plans to continue participation by signing an agreement with the OECD Halden Reactor Project for the period January 1, 2000, to December 31, 2002. The work performed at Halden contributes to meeting the agency goals of maintaining safety and regulatory effectiveness by conducting experiments and analyses that are used to develop technical bases for realistic safety decisions and that prepare the Agency for the future by evaluating safety issues involving current and new designs and technologies. In addition, the leverage achieved by sharing the costs with other sponsors makes resource utilization efficient and effective, by getting the benefits of a $40M three-year research effort for $2.85M.

Patricia M Norry for
William D. Travers
Executive Director
for Operations

Attachments: As stated

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OECD HALDEN REACTOR PROJECT

Attachment 1

# HALDEN REACTOR PROJECT PROGRAMME

## Proposal for the Three Year Period 2000-2002

May 1999

**PROGRAMME PROPOSAL FOR THE HALDEN REACTOR PROJECT FOR THE THREE YEAR PERIOD 2000 - 2002**

This document contains a draft proposal for the programme to be carried out in the period 2000 - 2002 aimed at forwarding participants' interests on safety and reliability of nuclear power plants.

*Fuel and Materials Programme*

**Ch. 2. Fuel High Burn-up Capabilities, Normal Operating Conditions**

**Ch. 3. Fuel Response to Transients**

**Ch. 4. Fuel Reliability Issues**

**Ch. 5. Plant Lifetime Assessments**

Experimental and analytical activities will be carried out aiming at assessing *fuel performance capabilities and property changes* at burn-up in excess of current discharge levels. The extensive use of re-fabricated and instrumented commercial fuels is anticipated. The proposed work on *safety transients* is intended to provide unique experimental complements to investigations carried out elsewhere on loss-of-coolant and reactivity-initiated transients. Tests on short-term dryout and anticipated transients without scram are also contemplated. Investigations on *fuel performance anomalies* arising from current operational experience will be conducted in close collaboration with participants with the purpose of identifying realistic design or operational remedies. The activities on plant lifetime assessments are to focus on *stress corrosion cracking* of structural reactor materials under the combined effect of water chemistry and radiation environment. The proposed programme relies on the Halden in-reactor measurement capabilities, on rigs designed to produce a variety of test conditions and on in-reactor water loops able to create flexible coolant environments.

*Man-machine Programme*

**Ch. 6. Experimental Programme and Operation of HAMMLAB**

**Ch. 7. Human Factors and Control Room Engineering**

**Ch. 8. Plant Operational Support**

**Ch. 9. System Safety and Reliability**

Operator support systems are to be introduced in the *upgraded HAMMLAB*, aiming also at demonstrating the merits of such systems in an integrated control room environment. The activities on *human-machine interaction* will extend the knowledge of human performance in process control environments and of how this can be incorporated in control room requirements and design. The proposed work on *plant performance monitoring and optimisation* is intended to develop system solutions having the potential for improving plant performance as well as operational safety. The aim of the activities on *system safety and reliability* is to provide methods devised to enhance the reliability of automated systems and in particular of safety-critical software. The technical basis for the man-machine programme consists of the upgraded test facility HAMMLAB, together with the test methodology, the software, simulation and control room expertise developed around it.

**HALDEN REACTOR**

**PROJECT PROGRAMME**

**Proposal for the**

**Three Year Period 2000 - 2002**

**May 1999**

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# 1. SUMMARY OF PROPOSED PROGRAMME

## 1.1 Previous Programmes

The OECD Halden Reactor Project was established in 1958 as a joint undertaking of the OECD Nuclear Energy Agency through an agreement between national nuclear centres of OECD member countries sponsoring an experimental research programme with the Halden Boiling Water Reactor. Subsequent to the first agreement, which covered a five-and-a-half year period, twelve successive agreements, each of three years duration, have been entered into, the present one terminating on 31st December, 1999.

The research and development programme executed under the first agreement was aimed at demonstrating the operability of a natural boiling heavy water system and included extensive physics and dynamics studies. The emphasis was then gradually shifted to water reactor research of broader scope. Performance studies on light water reactor fuel and development of computerised supervision and control systems have been the main activity areas.

Following the Project's successful development of *in-core rod instruments*, in-reactor testing of *basic fuel rod reliability* parameters were initiated with the major objective of defining performance data as well as exploring mechanisms. This in-core measurement capability has been expanded through development of experimental rig and loop systems where reactor fuel and material can be tested under simulated *water reactor conditions*. Increasingly, fuel reliability and safety considerations are emphasised under the jointly financed programmes while development and optimisation issues are addressed in bilateral programmes.

With the introduction of the water loop systems with representative water chemistry environments, an expanded use of the facilities for characterisation of *core materials* performance has taken place. These activities focus on age related degradation of materials and components for which assessment of residual life is an important safety/licensing consideration. Ageing mechanisms as well as mitigation factors are being investigated. These developments have taken place in response to needs that have gradually emerged in the nuclear community and have been favoured by the availability of expertise and methods for in-core instrumentation.

The development efforts in *man-machine systems research* were initiated on the basis of the experience gained through the reactor dynamics experiments and the use of in-core instrumentation. Initially, efforts were spent on practical demonstrations of advanced concept for closed loop control, and core power distribution and plant load-follow control were successfully demonstrated on the Halden Reactor. Subsequently a shift of priority towards *human-factors* based designs, utilising an experimental control room linked to a full-scope PWR simulator took place. In the later periods, the issues related to upgrading the instrumentation and control systems for the purpose of increased safety and improved operation have been given increased attention. The activities have thus been focused on providing information supporting the design and licensing of upgraded, computer-based control room systems and to demonstrate improvements through system validation experiments carried out in the Projects experimental control room HAMMLAB. This facility now contains full scope simulators for BWR, PWR and VVER systems, complemented with a Virtual Reality Centre.

### 1.2 Basis of the 2000 - 2002 Programme

Enhanced safety and reliability of nuclear power systems will remain the main aims of the envisaged Project programme for the 2000 - 2002 period, however maintaining synergy with non-nuclear environments, especially in the field of instrumentation and control. The results are to be used in support of the safety cases for existing reactors as well as to assist the power plant operation. These aims are to be achieved through active utilisation of the Halden reactor and the HAMMLAB facilities, representing the main research tools of the Project.

The programme planning for the 2000 - 2002 Halden Agreement takes place in a period during which the nuclear industry is facing particular challenges related to deregulation of the electricity markets as well as to ageing and modernisation of nuclear plants. In order to balance the resulting economic impact, the industry is looking for means to improve operational economics and flexibility. At the same time, regulatory authorities have to verify that these measures are implemented in full compliance with safety requirements.

The Halden Board of Management issued, in April 1998, the document "Views on the long Term Direction of the OECD Halden Reactor Project." In that document, the Board emphasises those qualities that will constitute the basis for the continued successful operation of the Halden Project. Detailed technical discussions and enumeration of priorities are omitted, taking the view that an active program will incorporate priorities as they develop, due consideration given to a number of factors such as complementary work performed at other establishment and to synergy emerging from interaction with non-nuclear environments.

The Board notes that safe and reliable utilisation of nuclear energy requires development and verification work in a variety of areas, since questions continue to arise as result of demands on plant performance and regulatory practice. The Board believes that the ability to adapt to a changing environment and to respond effectively to needs as they emerge, constitutes a major virtue of the Project. Consequently, it is essential that the Project retain key values leading to flexibility, primarily in terms of a compact and transparent organisation able to absorb relevant input and convert it into practical programmes.

A continued interaction with and between the Project participants is foreseen for the working out of the more detailed programme plans. Such interaction is considered essential for generating an integrated and as complete as possible understanding of safety issues of common interest. Furthermore, priority will be given to expand on research items for which broader integration of the overall capabilities and expertise at the Project can be beneficial, including increased utilisation of resources and expertise available at the Halden Reactor. Co-operative agreements on the utilisation of participants' facilities are also foreseen, including complementary activities for accomplishing specific experimental or analytical objectives. The Project's international contact net will in this respect be used for catalysing and implementing co-operative activities outside the Halden programmes as well.

Several of the participating organisations maintain an interest also in the non-nuclear area. Although the proposed programmes are primarily focused on nuclear issues, it will be an important aim to extend the interaction with non-nuclear interests in the member countries. This approach has on a bilateral basis been rather successfully pursued in several member countries during the later years, providing technological spin-offs for use in for instance oil and gas exploration and in the process industry in general. Such interactions are important in that they provide valuable feedback to the nuclear area.

The proposed programmes build extensively on priority indications expressed by the participating organisations. The fuel programme is intended to provide basic data on how the fuel performs in commercial reactors both at normal operation and in transient conditions. As extended fuel utilisation is a major industry priority, the emphasis will be on fuel properties after prolonged service time in-reactor. The materials programme addresses irradiation induced material changes and corrosion processes which can lead to progressive degradation of in-reactor components. The possible degradation of material properties is an issue where the industry and the licensing interests have common objectives for exploring mechanisms and for devising mitigating measures. These common interests are reflected in the items being addressed in the proposed programme.

The man-machine activities emphasise human factors work and the development and testing of advanced surveillance and control systems, including establishment of methods and tools for verification and safety enhancement of operator support systems. The merits and constraints when integrating several such systems in control rooms are also evaluated. Further, particular attention is given to issues related to the backfitting of control rooms with computer-based solutions for surveillance and control as well as for man-machine interfaces. Emphasis will be placed on establishing information relevant for formulation of guidelines for design and licensing of new systems, both with respect to software reliability as well as for human factors considerations.

The experimental facilities which are available for the proposed 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. Further expansion of the in-core instrumentation technology is foreseen. The Halden Man-Machine Laboratory and its associated facilities have been upgraded and extended with new technology and improved experimental techniques. The current introduction of modern LWR simulators provides a basis for a wider spectrum of evaluation experiments, and for use of the facilities for testing of hardware and software solutions.

In formulating the programme plans, it has been attempted to balance the activities such that long term priorities are promoted, while at the same time maintaining the ability to develop short term deliverables. It should also be noted that the participating organisations are expanding their use of the Project's facilities and capabilities on a bilateral basis. The synergism between the jointly financed programmes and the bilateral contract activities represents a most important vehicle for supporting the priorities of the different participating organisations in the Project.

In a long term perspective, it will be important to maintain the flexibility for introducing new research elements in the programme as needs and requirements arise. This evolutionary process will depend on continuous guidance and scrutiny by the member organisations, individually as well as through the international steering bodies of the Project.

## SUMMARY OF THE FUEL AND MATERIALS PROGRAMME

Given the maturity of the nuclear industry, the perspective for nuclear energy will, in addition to safety and reliability, be determined by economical factors. In many countries utilities are faced with intense competition due to deregulation and in order to compete effectively, they are aiming to improve operational economics and flexibility, while regulatory authorities have to verify that this is done without detriments to reactor safety.

Licensees are implementing or considering a variety of operational measures as means to reduce operational and fuel cycle costs. This exposes the fuel to increasing challenges, which has prompted the vendors to propose new fuel designs and new materials. Regulatory bodies are faced with the need of qualified models and codes for safety case assessments in a variety of operational conditions, for many different types of fuel designs and at extended burnup. This necessitates new and improved data on fuel properties and fuel behaviour under various normal, abnormal, and accident conditions. At the same time, operational experience demonstrates that unforeseen anomalies can develop as demands on performance become more stringent. Localised corrosion and defected fuel degradation are potential utility concerns. Control rod sticking and anomalous axial power offsets have recently posed limitations on plant capacity factors and caused regulatory concern. Halden experiments can give important contribution in addressing and resolving these issues and those likely to emerge in the future. As the age of power plants increases, safety authorities will need materials property data relevant to in-reactor components at high irradiation doses, as they will form the basis for plant lifetime assessments. Utilities are introducing operational changes that can enhance the reliability of plant structural components - e.g., water chemistry modifications - and are adopting advanced materials where this can be done. Practical verifications and data will be needed to support lifetime predictions of existing and replacement materials as well as to validate measures intended for lifetime extensions.

The present proposal focuses on the following main issues:

- *Fuel high burnup capabilities in normal operating conditions*, aiming at providing fuel property data needed for design and licensing in the burnup range 60 to 100 MWd/kg. Both test fuel and re-fabricated commercial fuels will be used in the proposed investigations.

- *Fuel response to transients*, aiming at providing experimental complements to investigations conducted elsewhere on behaviour in flow starvation and reactivity initiated transients. Further tests on short-term dryout and possibly on power-coolant flow oscillations are also considered.

- *Fuel reliability issues,* aiming at determining the mechanisms and operational conditions that can affect cladding integrity, as well as at identifying realistic design or operational remedies. These investigations are to be conducted in synergy with bilateral activities.

- *Plant lifetime assessments,* aiming at generating validated data on stress corrosion cracking of reactor materials at representative stress conditions and reduction/water chemistry environment. Tentatively, issues related to pressure vessel embrittlement are also addressed in the proposal.

## Schedule of Fuels & Materials Programme 2000 - 2002

| Issue | Scope | 2000 | 2001 | 2002 |
|---|---|---|---|---|
| ***HIGH BURNUP FUEL CAPABILITIES IN NORMAL OPERATION*** | | | | |
| Conductivity degradation, Thermal performance | Gd + UO2 | burnup 80 o ........ | PIE x | |
| | commercial Gd f. | | | burnup 40 o |
| | inert matrix | irradiation | | |
| | PWR UO2 > 50 | ramp, irrad. reinstrumentation | ramp, irrad. reinstrumentation | ramp, irrad. |
| | MOX > 60 | burnup extension o........ | ramp,irrad. o ........ | PIE x |
| | VVER fuel | irradiation | | |
| | MOX | irradiation | | o |
| | new fuels | ........procurement........ | reinstrumentation........ irradiation | |
| Fission gas release | MOX > 60 | burnup extension o........ | ramp o ........ | PIE x |
| | | FGR threshold 30 | FGR threshold 40 x | |
| | UO2 > 50 | threshold, high power ........ | threshold, high power ........ | threshold, high power |
| | commercial Gd f. | irradiation, in-pile measurements | | burnup 40 o |
| | IMF/MOX | | irradiation, in-pile measurements | |
| | VVER | | irradiation, in-pile measurements | |
| | new fuels | ........procurement........ | reinstrumentation........ irradiation | |
| PCMI, Swelling, Load follow, Ramp operation | MOX > 60 | burnup extension o........ | ramp, relaxation o | |
| | PWR UO2 > 50 | power incr., relax o ramp o | ramp o | load follow o load follow o |
| | commercial Gd f. | irradiation, evaluation of in-pile measurements | | burnup 40 o |
| | VVER | irradiation | | |
| Cladding corrosion, Hydriding, Corrosion sensor Development | PWR | continued irradiation IFA-638, oxide thickness and weight gain measurements at regular intervals x | | |
| | | -x- exchange of coupons | | -x- exchange of coupons |
| | BWR | definition of test on local corrosion, design, fabrication | execution | |
| | VVER | design of loop, test definition, loop installation, rig fabrication ........ | | execution |
| | Sensors | development, application of in-pile corrosion sensor, impedance spectroscopy | | |
| Cladding creep | Zry-4 low tin, M5, Zirlo | diameter measurements, stress changes tensile <> compressive | | 15000 fph x |
| | new materials | ........procurements of new materials, replacement test preparation........ | | new test |

## Schedule of Fuels & Materials Programme 2000 - 2002

| Issue | Scope | 2000 | 2001 | 2002 |
|---|---|---|---|---|
| Tolerable rod overpressure | PWR MOX, UO2 | overpressure operation x | | |
| | other PWR clad | procurement of materials, re-fab. overpressure operation x | | |
| | BWR clad | fabrication of rig, procurement of materials, re-fabrication ............ overpressure operation x | | |
| ***FUEL RESPONSE TO TRANSIENTS*** | | | | |
| Loss of Coolant Accident | cladding with high exposure | test def., procurement of material<br>rig fabrication test execution o.................. PIE x | | |
| Reactivity Transient, transient FGR & swelling | rim structure fuel | irradiation of fuel to >100 MWd/kg<br>completion of ramp rig design, fabrication .......... transient ................ PIE .....x | | |
| Short-term dryout | BWR fuel<br>PWR fuel | definition of scope, materials procurement<br>rig fabrication o........ test execution o.............. PIE x | | |
| ATWS power oscillation | high burnup BWR fuel | test definition, identification of fuel<br>..........................................(rig fabrication).................................. power oscil. test x.........(PIE) | | |
| ***FUEL RELIABILITY ISSUES*** | | | | |
| Crud deposition<br>Axial offset anomaly | PWR | installation and testing of dedicated loop<br>(test def.) rig fabrication...................... influence of rating, crud thickness meas., conductivity ................ | | |
| | VVER | installation and testing of dedicated loop ...... study VVER water chemistry effects ................................. | | |
| Localised corrosion | BWR cladding | (rig design) rig fabrication .......... effect of radiation type, material combinations, chemistry...................... | | |
| Fuel failure degradation | BWR clad, fuel | performance of new alloys .....(PIE)... effect of pre-irradiation ...(PIE)... | | |
| ***PLANT LIFETIME ASSESSMENTS*** | | | | |
| Irradiation assisted stress corrosion cracking<br><br>Crack initiation | BWR conditions, SS 304, 347, 316 | stress intensity, fluence, chemistry o ......... PIE ......... x<br>... new rig, fabrication ... low stress intensity, high fluence, surface coating o ....... (PIE) ...... | | |
| | PWR conditions | ... materials ... fabrication ... irradiation in available PWR loop ....... | | |
| | SS dry irradiation | effect of irrad. damage, composition, fabrication on austen. SS<br>(specimens at different fluences, mech. testing, microstructure characterisation) | | |
| | SS - CERT tests | irradiation of pressurised tubes o<br>(CERT tests on sensitised 304 SS in NWC/HWC, different fluence levels) | | |
| Pressure vessel aging | VVER, LWR | irradiation of pressure vessel materials in the HBWR, collaborative arrangements for mechanical testing | | |
| Plant dose reduction | LWR | utilisation of loop systems for activation studies ... effect of NWC/HWC, additives ... (PIE- EDS, DSIMS) | | |

# FUEL HIGH BURNUP CAPABILITIES IN NORMAL OPERATING CONDITIONS

## 2.1 Introduction

Utilities are implementing or considering extended burnup, longer fuel cycles, power upratings and load follow as means to reduce operational and fuel cycle costs. This exposes the fuel to increasing challenges, which has prompted the vendors to propose new fuel designs and new materials. There is also a strong push to use mixed oxide fuels in power reactors. Regulatory bodies are faced with the need for qualified models and codes for safety case assessments in a variety of operational conditions, for many different types of fuel designs and at extended burnup. This necessitates new and improved data on fuel properties and fuel behaviour in the high burnup range and at relevant boundary conditions.

Issues related to extended burnup operation under normal operation conditions are addressed in this chapter. They include:

- Thermal performance and degradation of fuel thermal conductivity with increasing burnup
- Enhanced pellet-cladding interaction due to fuel swelling
- Increase of rod pressure due to fission gas release
- Cladding corrosion and hydrogen pickup
- Cladding creep properties at extended exposures
- Tolerable rod pressure limits

The burnup range considered in the programme varies from 60 to 100 MWd/kg. The programme is based on the utilisation of proven instrumentation capabilities which provide direct in-reactor measurements of relevant fuel performance parameters. The fuel used for the proposed experiments will consist of both test fuel specially designed for a specific test and irradiated in Halden test rigs for the entire lifetime and of commercial fuels previously irradiated in a power reactor and then transferred into Halden instrumented rig.

While $UO_2$ will continue to be addressed relatively extensively, efforts will be made to acquire more knowledge on MOX fuel behaviour. In addition, gadolinia fuel will be used in some of the tests. It is apparent that not all the tests proposed in the following sections can contain the entire range of fuel variants and in this respect the Project has to work out a comprehensive test matrix in which given fuel types are assigned to given test rigs. This will also depend on what types of commercial fuels can be made available to the programme. The same applies to the cladding materials to be included in the corrosion, creep and overpressure tests.

The great majority of the investigations contemplated for the next three year period will involve extended irradiations of test and commercial fuel which already has significant burnup. For the latter, the well proven technology for re-instrumenting with sensors for measurement of fuel temperature, rod pressure and rod length changes will be utilised. However, some tests will be run in moderate burnup ranges, mainly because the irradiation has to be started from unirradiated materials. This is the case for a new VVER fuel irradiation, which has started in the beginning of

1999, and the fuel consisting of fissile materials in inert matrices, which is to start irradiation towards the end of 1999.

The tests devised for cladding property assessments will be conducted in LWR loops which are operated at prototypical thermal-hydraulic and water chemistry conditions. Both coolant temperature and water chemistry can be varied within some limits to accommodate separate effect studies. For tests specifically designed to assess fuel properties, such as thermal conductivity or fuel swelling, normal HBWR conditions provide an ideal, well-defined environment.

Participants have expressed a considerable interest in establishing a loop for simulation of VVER thermal-hydraulic and coolant chemistry conditions. The main utilisation of such a loop would be for corrosion studies, also of cladding alloys used in PWRs, and investigations of chemistry effects, e.g. crud formation.

The proposed experimental and analytical work is based on recommendations from member organisations, on discussions in the Halden Board and Programme Group, and on advice from specialist workshops. The major considerations will be to provide data supporting a mechanistic understanding of potential life-limiting phenomena focusing on representative and modern materials. Plans for destructive and non-destructive PIE will be included in the programme and carried out as appropriate.

The proposal represents a frame programme whose details will be formulated based on further programme discussions in the steering bodies of the Project. While addressing both current and forthcoming performance issues, this proposal is founded on the experience gained in previous programme periods as well as on the in-pile instrumentation and rig designs developed at the Halden Project for meeting a variety of test demands.

## 2.2 Fuel Thermal Performance, Conductivity Degradation

A thorough knowledge of fuel thermal performance is essential for modelling and safety analyses since most material properties and phenomena depend on temperature. The experimental investigation of processes influencing fuel temperatures and accurate measurements are therefore fundamental for fuel safety and reliability assessments. This was recognised in previous programme periods where the database on fuel conductivity degradation and gap conductance was considerably extended for urania fuel. Responding to participants' request to provide a similar data base related to other fuel types in use in commercial LWRs, irradiation programmes have been initiated addressing gadolinia, MOX and VVER reactor fuel.

It is known that the thermal conductivity of oxide fuels tends to decrease with burnup. Temperature data from various tests (e.g. fuel solely irradiated in the HBWR, re-instrumented high burnup LWR fuel) form the basis of the Halden Project's correlation of fuel thermal conductivity changes as well as participants' own derivations. These efforts are supplemented by hot-cell laser flash diffusivity measurements conducted at participants' laboratories on fuel irradiated in the Halden reactor.

It is anticipated that further investigations with respect to fuel thermal performance and especially fuel conductivity changes induced by high burnup will be carried out in the next programme period aiming at extending the burnup range and at including different fuel types. These will also be addressed in other parts of the overall programme, acknowledging the interlinked nature of fuel behaviour. As a result of their data evaluation and needs, different participants have indicated an interest in the following points to improve and extend the correlations obtained so far:

- explore the thermal performance of commercial fuel with burnup > 60 MWd/kgU,
- include other fuel types such as MOX fuel and gadolinia fuel,
- investigate the possibility of a saturation of the conductivity degradation at very high burnup and in combination with additives,
- possibly elucidate the role of irradiation damage and conductivity recovery with respect to relevance for in-reactor fuel performance,
- assess the effect of structural changes induced by extended rim formation.

A number of experiments currently being executed or in an advanced state of preparation will continue for some time into the next programme period. These are related to:

- *Irradiation of gadolinia fuel.* A test with accelerated burnup accumulation addresses the question of differences in thermal conductivity of $UO_2$ and $(Gd,U)O_2$ fuel and whether any reduction caused by the introduction of lattice defects and fission products during irradiation is additive to the influence of gadolinia.
- *Irradiation of MOX fuel.* A test has been initiated to generate temperature and fission gas release data, starting with fresh fuel to cover the low and medium range of burnup as a bridge to the results obtained from pre-irradiated, re-instrumented MOX fuel.
- *Pu burning in an inert matrix.* A test to determine the thermal and stability characteristics of fuel with fissile material in an inert matrix is being prepared in collaboration with participating organisations. The irradiation is envisaged to commence by the end of 1999 and will continue throughout the next programme period.

Participants have also expressed an interest in including non-standard fuels such as high density fuel, fuel with erbium as burnable poison, and carbide and nitride fuels in the experimental programme. The basic properties of these variants, i.e. thermal and mechanical behaviour as well as fission gas release, can be studied in a straightforward manner using standard irradiation rigs and instrumentation, provided that laboratory prototype fuels can be supplied.

The experimental programme intended for the 2000 - 2002 programme period includes the extended irradiation of re-fabricated, instrumented fuel transferred from commercial reactors and provided to the Project by participants. The desired burnup target for these irradiations ranges between 70 and 90 MWd/kg. The rods will be equipped with fuel centreline thermocouples, pressure transducers and cladding elongation detectors, such that many relevant performance aspects can be addressed at the same time for the same type of fuel. The irradiation in the Halden reactor is expected to continue for tentatively two-three HBWR cycles, such that data are gathered over a significant time and burnup interval.

### 2.3 Fuel Diffusivity and Stored Heat (Heat Capacity)

While the fuel thermal conductivity determines the operating temperature of the fuel at a given heat rating, the fuel diffusivity determines the time constant for heat removal from the fuel during a sudden power drop, such as for instance in the initial phase of a LOCA transient. The heat capacity on the other hand determines, together with the fuel thermal conductivity, the stored heat in the fuel. Also this parameter is relevant for LOCA transients.

It is proposed that the fuel diffusivity/heat capacity be systematically characterised by analyses of the fuel centreline temperature transients following reactor scrams. It should be noted that this programme item does not require ad-hoc experimental equipment, since the same test rigs and fuel used for the thermal assessments described in the previous section will be utilised. The additional requirements are that reactor scrams are carried out at given points in time and that qualified analytical tools are used for the data analyses. The assessment of the heat capacity as such, however, requires knowledge of the fuel density.

### 2.4 Rod Inner Pressure Increase, Fission Gas Release

The fission gas inventory in a fuel rod increases proportionally with burnup, while the decrease of fuel thermal conductivity and the changes of fuel microstructure reduce the ability of the fuel to retain the fission gas. This, together with the rod voids volume reduction caused by fuel swelling, can produce high inner fuel rod pressure at high burnup.

Considering the complex of performance aspects and conditions that must be assessed at high burnup, it is not quite clear that enhanced fission gas release at high burnup does necessarily aggravate the fuel performance. Earlier tests carried out at the Project showed that the fuel performance at normal conditions is not impaired even at high rod pressure levels. On the other hand, when the gas is to a great extent retained in the fuel - especially at grain boundaries - its sudden release may impair the fuel integrity in, e.g. reactivity transients which cause rod overheating.

The programme proposal aims at elucidating the fission gas release behaviour at steady temperatures and in fuel power or temperature uprates. The intention is to provide modellers with qualified data that can be used for analyses of the fuel behaviour at extended burnup, both at normal operation conditions and in safety related transients. In parallel, studies will be continued to determine the extent to which high rod pressure affects the fuel integrity at steady state and in transients.

For normal operating conditions, the fuel rod pressure build-up will be monitored on-line in virtually all the fuel temperature tests outlined in section 2.2. This will enable to characterise the fission gas release as related to fuel temperature in the burnup range 60-100 MWd/kg.

In addition to long term irradiation, studies will be conducted also in power ramp conditions with ramps conducted at rates ranging from seconds to minutes. The aim of these ramps is to "connect" the database on fission gas release at steady state and in transient conditions, this being highly relevant also for safety analyses. The tests dedicated to rod pressure/fission gas release at normal conditions will thus encompass the following categories:

- Long term operation of $UO_2$, MOX and gadolinia fuel conducted on instrumented rods irradiated in the HBWR for the entire service time, burnup range 60 to 100 MWd/kg.
- Long term operation of commercial fuel rods re-fabricated and re-instrumented for extended irradiation in the HBWR, burnup range 70-90 MWd/kg.
- Rapid power ramps of commercial and/or test fuel rods in the burnup range 70-90 MWd/kg.

In addition to the standard commercial fuels indicated above, participants have expressed an interest in fuel with an increased enrichment of 5% U-235 or more. Such fuel is required for extended cycle operation (18 months) and 4 batch core management or other combinations with a fuel residence time of 5 – 6 years. When MOX fuel is utilised in the same way, the Pu content has to be increased accordingly to maintain the parity with the $UO_2$ fuel. In this fuel, more power is generated from original fissile atoms in the hotter parts of the pellets, and the potential for fission gas release is increased compared to low enrichment fuel which has been driven to the same burnup. Furthermore, larger and more interior parts of pellets with peak burnup may exceed the threshold of rim structure formation with possible consequences for fission gas release. Such fuel will require a development and acceptance time of many years and an associated experimental programme should be contemplated.

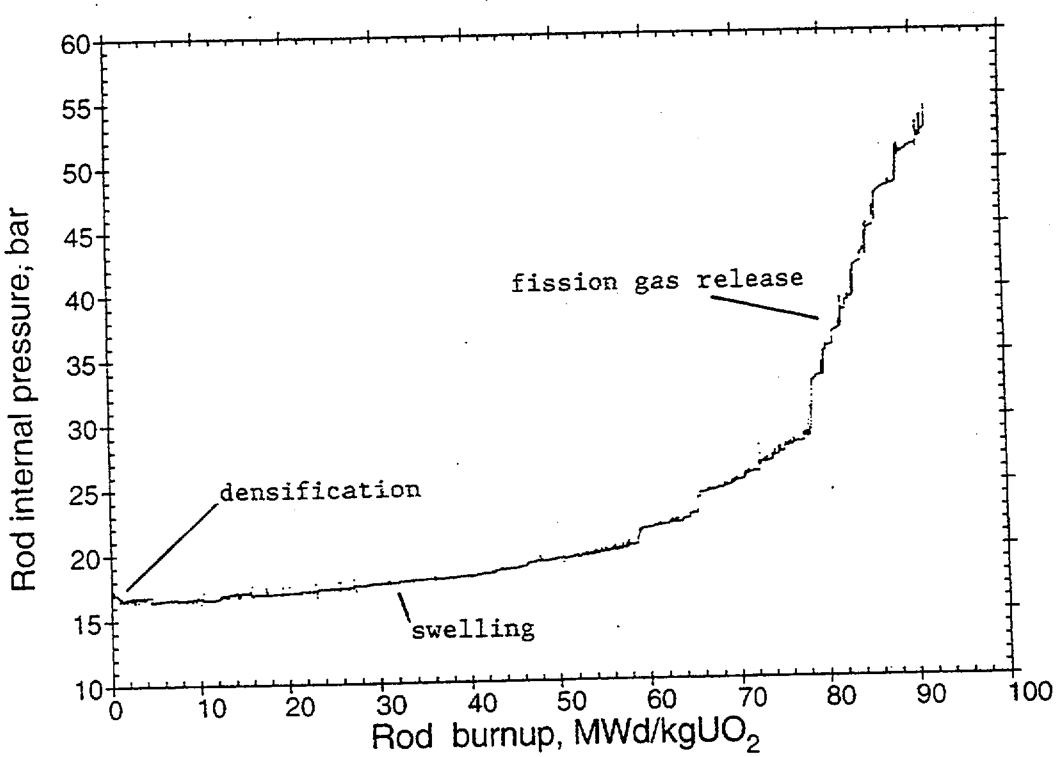



*Fig. 2.1 Rod inner pressure history in a fuel rod irradiated in the HBWR for the entire lifetime. The pressure data are obtained by means of on-line, in-reactor instrumentation*

### 2.5 Pellet Cladding Mechanical Interaction, Power Cycling and Ramp Operation

The incorporation of fission products into the fuel matrix causes swelling of the fuel and when the gap is closed, continued swelling imposes a tensile hoop stress in the cladding, while a pellet-cladding bonding layer is gradually formed which can enhance the pellet-cladding interaction (PCMI). On the cladding side, the move to higher discharge burnup leads to higher waterside corrosion and greater hydrogen pick-up. This reduces the cladding ductility and may result in a limit on discharge burnup or operational flexibility. In a number of countries, PCMI in fact poses restrictions on reactor operation or is seen as a safety concern. It is therefore of continued interest to investigate the phenomena contributing to PCMI in order to characterise the fuel performance at the envisaged higher discharge burnups of > 60 MWd/kg.

Experiments related to PCMI and involving high burnup fuel are essentially identical with those discussed in the sections on fission gas release (2.4) and fuel swelling (2.6). Pellet-cladding interaction produced during power transients can be measured by elongation detectors or, when required, with diameter gauges.

Power cycling has been identified by some participants as a topic for further PCMI studies, in particular as there are only limited in-pile data on the effect of released fission products on fatigue life of the cladding during operations involving a large number of frequent power changes. Further, axial ratchetting as observed in some experiments involving high burnup fuel may lead to an accumulation of strain increments with a possible impact on cladding integrity.

Fuel cladding bonding seems to occur in the same burnup range as the rim structure formation and has the potential of affecting fuel rod behaviour since excessive bonding may lower the PCI failure threshold. Step-wise or fast power ramps of commercial fuel (burnup between 70 and 90 MWd/kg), re-fabricated and instrumented with cladding extensometers, can be used to measure the effect of bonding on PCMI. The size of power increment and hold times should be chosen to avoid failure and to provide information on the cladding stress build-up and relaxation.

It should be noted that both power cycling and ramp operation can be accomplished in the test rig equipped with He-3 power control that is being used for fission gas release investigations described in section 2.4 and fuel swelling in section 2.6. Thus the same experimental setup and irradiated commercial fuels can be used to address performance during ramp operation and power cycling.

### 2.6 Fuel Swelling

In experiments starting with fresh fuel, fuel stack elongation measurements will provide information on fuel densification and swelling. Such data are essential for comparing different fuel manufacturing routes, variants with dopants to influence fuel creep and plasticity, different grain sizes, and the behaviour of MOX and gadolinia bearing fuel. In addition, all joint programme tests addressing fission gas release will also address fuel swelling. This will be achieved by having pairs of rods of equal design and operating history equipped respectively with a pressure transducer and with a fuel stack/cladding elongation detector. Thus, the envisaged programme will encompass:

- Long-term operation of test rods irradiated at Halden for the entire lifetime, burnup range 60 to 100 MWd/kg.
- Extended irradiation in the Halden reactor of commercial fuel (burnup between 70 and 90 MWd/kg) re-fabricated and instrumented with cladding elongation detectors which also indicate fuel swelling in the case of high burnup fuel with fuel-clad bonding.
- Hydraulic diameter measurements providing information on dimensional changes by the gas flow technique.
- Swelling and fuel creep associated with fast power increases and holding periods, predominantly for high burnup fuel.

Participants have also expressed the need of data on cladding diameter changes that may occur as consequence of fuel swelling during irradiation. This can best be achieved by means of diameter profilometries carried out at regular intervals during reactor outages.

## 2.7 Cladding Corrosion and Hydriding

Extended cycle operation, low leakage loading schemes and extended fuel utilisation have increased the duty of cladding in LWRs, in particular in some PWRs. New alloys are therefore being developed to maintain the margins to regulatory requirements. One of the most important considerations is corrosion which weakens the cladding as a result of combined wall thickness reduction and embrittlement by hydrogen pick-up in the metal. These effects are believed also to have an influence on performance during accident conditions, e.g. RIA and LOCA.

Cladding corrosion has been addressed in previous programme periods with experiments designed to assess the separate effects of various parameters on oxidation rate of high and low tin cladding materials. This included hydrogen content of the cladding, irradiated versus unirradiated material, cold spots as sink for migrating hydrogen, cladding stress, and heat rate variations.

A new PWR cladding corrosion rig has been recently installed with the intention to operate it over a long period of time and through most of the 2000 - 2002 programme period. The test matrix and experimental set-up emphasises the comparison of different alloys subjected to varying conditions. The upper part of the test section consists of irradiated cladding tubes from defuelled rods while the lower part contains unirradiated materials, some of them prehydrided. The tubes have been filled with fresh pellets to create a high heat flux and local subcooled boiling. Cladding coupons have been placed above the fuelled tubes. In this way, the corrosion properties of many materials can be studied simultaneously as influenced by temperature, heat flux and neutron flux. The progress of corrosion is determined out-of-pile during interim inspections. It should be noted that the test rods and coupons can be exchanged such that other materials can be introduced at any time. Participants have suggested more developmental alloys for testing, but details have to be discussed in the programme Group and will depend on the progress of the experiment and the obtained results. As planned, the test will proceed for 3-4 years to a burnup of about 50 MWd/kg.

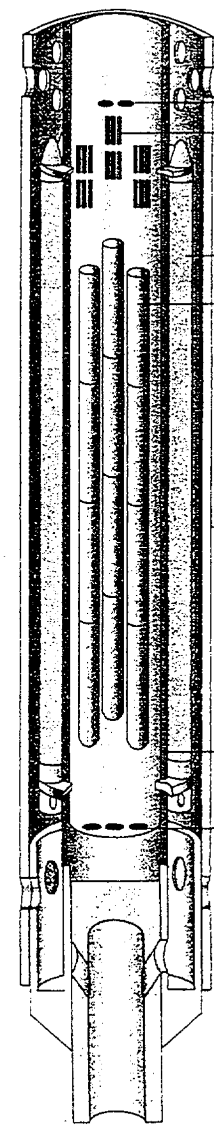



*Fig. 2.2 Schematic of the corrosion test rig.*

The proposed programme envisages a corrosion test also for BWR conditions, intended to explore the separate effect of local radiation fields on the so called "shadows" corrosion. Previous Halden tests carried out in collaboration with participants have shown that corrosion peaks can arise axially in the vicinity of metals which become beta-emitters by neutron activation. The purpose of this test is to clarify the extent to which cladding corrosion can be locally enhanced by elements such as nickel contained both in structural parts and in the crud that may deposit onto the fuel rod. This issue is discussed further in section 4.4.

While the current corrosion tests still rely on established techniques, i.e., eddy current and weight gain measurements, the development of on-line corrosion sensors is being pursued at the Project. One of these is based on the change of electrical resistance as the cladding wall is progressively thinned by the corrosion. The application of impedance spectroscopy to corrosion monitoring is studied in collaboration with participating organisations.

The corrosion tests are carried out in loops operated as prototypical conditions. The Project's established tools and expertise for providing and monitoring the required water chemistry will be used to advantage in carrying out the joint programme as well as participants' tests. Special BWR and PWR water chemistry conditions can be created according to specifications. For instance, PWR loops have been operated with zinc addition to the water or with high lithium concentration. For what concerns BWR loops, a variety of water chemistry conditions have been provided, including hydrogen water chemistry and simulation of various impurities and/or additives. Discussions will be started with VVER users aimed at establishing a VVER loop at Halden addressing materials and water chemistry issues specific to that type of reactors.

## 2.8 Cladding Creep

Creep data on LWR cladding are required in fuel performance modelling for the assessment of clad creep-down and gap closure, stress relaxation during power ramps, and stress reversal at high burnup if the rod pressure exceeds the system pressure due to excessive fission gas release. These phenomena are being addressed in the current programme period with particular attention to stress reversal effects. It is important to note that creep data must be generated in-pile, i.e., in presence of neutrons in order to be adequately applied to real situations. Further, the test specimens must be obtained from pre-irradiated fuel rods in order that data can be used for high burnup assessments. A special irradiation rig operated in a LWR loop has been developed and successfully utilised at Halden for this purpose.

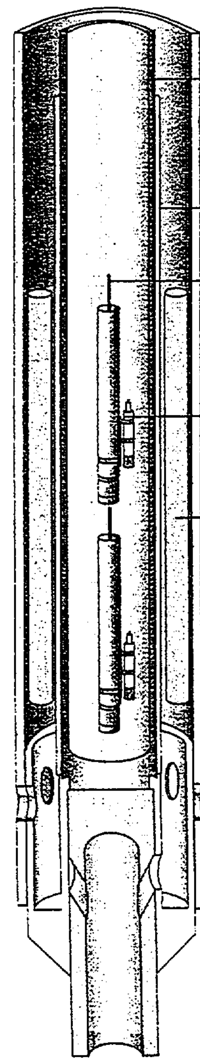



*Fig. 2.3 Rig utilised for creep of high burn-up fuel in stress reversal conditions*

The creep behaviour of high burnup BWR and PWR cladding materials has been investigated at Halden by means of in-pile measurements of the cladding diameter and gas lines mounted to the cladding tubes for control of the stress conditions. The satisfactory experience with in-pile creep measurements at Halden constitutes the technical basis for continued creep tests as recommended by Project participants. A new creep test with modern PWR cladding materials will commence execution during the second half of the current programme period. With an expected ten to fifteen thousand full power hours of irradiation, this creep test will extend into the next programme period.

The new creep test irradiation rig is designed to accommodate multiple loadings of fuelled specimens and will be available during the second half of the next programme period for a possible follow-on creep test. Alternative to the utilisation of pre-irradiated LWR cladding and depending n fast neutron fluence requirements, it would also be possible to arrange for the pre-irradiation of specimens in the HBWR. Practical considerations such as easy availability of a diversity of materials and simplification of transport for re-fabrication would be points in favour of such a solution. Provided that sufficient interest is expressed, the Project will conduct discussions with participants to identify and procure suitable modern materials and/or advanced alloys. For example, the selection may be made to support the study of creep and corrosion properties of the same materials, as is currently the case with the ongoing corrosion and creep tests.

## 2.9 Tolerable Rod Pressure, Normal Operation

The cladding lift-off experiments run at Halden in the recent past have yielded the most direct and convincing set of data on the maximum tolerable rod overpressure. This is defined as the maximum $\Delta P$ above system pressure to which the rod can be operated without causing measurable fuel temperature increases.

Project participants have shown considerable interest in the results and in extending the measurements to different types of cladding, higher burnups and, possible to BWR rods (the tests have so far been performed with PWR fuel). Further discussions with participants will clarify the type of fuels that can be considered and made available for future tests. The Project is prepared to continue the activities on this item and has, in addition to the present rig, initiated the work for producing a replica rig suitable both for PWR and BWR operation. Finally, it should be noted that the rod overpressure tests is designed to produce additional data on fuel temperature, on fission gas release and on fuel swelling.

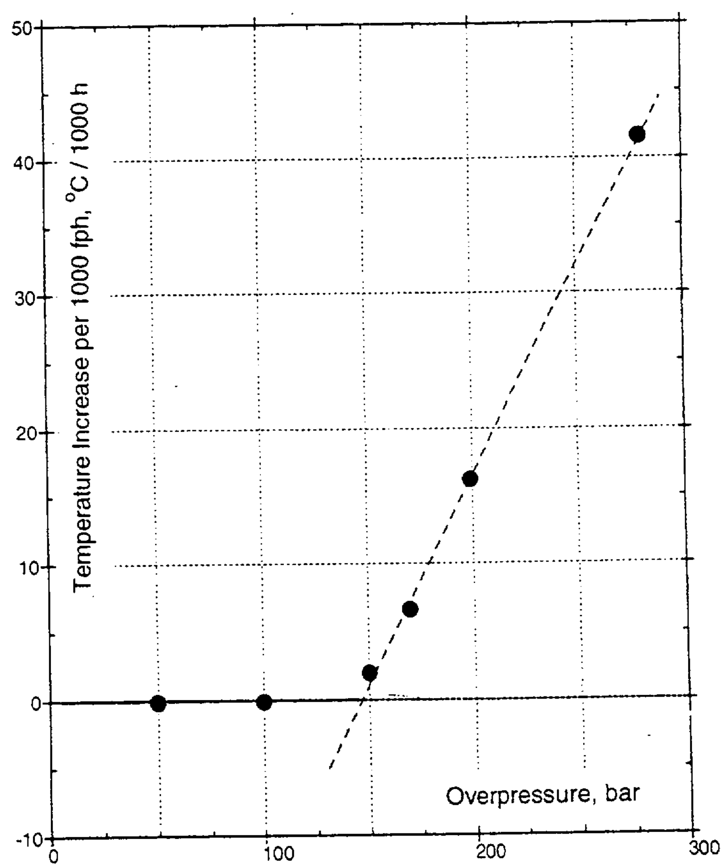



*Fig. 2.4 The diagram shows the fuel temperature increase rate as a function of rod over-pressure. (i.e., the $\Delta P$ above the sys-tem pressure of 165 bar). It can be observed that there is no temperature change for 50 and 100 bar overpressure. A linear fit of the data with higher overpressure shows that one must exceed 150 bar over-pressure to monitor measurable fuel temperature increases. (PWR rod, 59 MWd / kg)*

## 3. FUEL RESPONSE TO TRANSIENTS

### 3.1 Introduction

The programme items discussed in the previous section address key properties that are needed in order to understand and predict the fuel behaviour at normal conditions, this being a first essential prerequisite for the licensing of extended burnup fuel. At the same time, industry efforts are directed towards the development of new design able to better meet the challenges that arise at high burnup, by means of demonstrative irradiations carried out in parallel programmes, both in Halden bilateral programmes and elsewhere.

This endeavour does however require that the fuel high burnup capabilities be demonstrated not only at normal operating conditions, but also in safety transients. As these transients are by definition undesired and extremely rare events, the related experimental basis must be gathered by dedicated investigations conducted in a test reactor and in hot cells. In order to be cost-effective, it is important that the intended experiments focus on issues that are essential to safety and that cannot be resolved otherwise. The guidance of Project's participants for defining the key requirements for the tests in this area is thus of primary importance.

Tentatively, the Project programme for the period 2000 - 2002 may encompass the following items:

- Tests in support of mechanistic understanding of reactivity transients (RIA), primarily in terms of fission gas release and fuel swelling during rapid temperature transients.
- Fuel behaviour in short-term dryout transients, mainly aimed at verifying the fuel integrity and operability after such transients.
- Integral LOCA tests, devised to demonstrate the extent to which high burnup affects the overall fuel behaviour in loss of coolant conditions.
- Fuel response to anticipated transients without scrams (ATWS), i.e., in transients characterised by power oscillations consequent to coolant flow instabilities.

It should be noted that the experimental programme described in the previous chapter (i.e., at normal operating conditions) already constitutes an important contribution to the understanding of fuel behaviour in transients, at least in that it provides the initial conditions and property data needed for the analyses.

Consultations with Project participants are needed in order to set priorities on the above subjects, clarifying what need to be done also in light of the technical solutions envisaged at Halden and of the complementary work planned at other laboratories. Previous experience and conceptual solutions are available in relation to possible LOCA tests, as such transients were carried out extensively in the past. The same experience shows, however, that such tests are rather demanding and only well selected experiments should be carried out. As to RIA, the intention is not to duplicate pulse reactor integral tests, but to contribute with specific investigations on the understanding of the mechanisms involved. Good experience exists at Halden on short-term dryout, whereas flow/power instability tests need to be further discussed in order to arrive to a feasible experimental concept.

### 3.2 Reactivity Transients

Experimental data on reactivity initiated transients (RIA) have shown that damage may occur in high burnup fuel at relatively low energy depositions. However, a general trend of the experimental data is hard to identify since only very few failures have been obtained at enthalpies below 100 cal/g and in the burnup range 30-65 MWd/kg. Moreover, the failures which occurred at the lowest deposited energies seem to be associated with cladding in an atypical state, i.e., severely hydrided and with thick and spalled oxide layers. Different modified failure criteria have been proposed to take into account the new results, but additional experimental data at high burnup are still needed in order to understand the mechanisms involved and what may be done to improve the fuel to preclude failure.

In a RIA test conducted in a pulse reactor, the entire reactor is subjected to the transient and very high reactivity pulses of a few milliseconds duration are produced. In a material test reactor, very high and very short power pulses are precluded by the overall reactor design, and power peaks will by necessity be of smaller amplitude and longer duration. As an indication, the maximum achievable energy deposition rate for a commercial high burnup rod in the Halden reactor is of the order of 25-40 cal/g per second depending on conditions and design. If highly enriched rods are used, the possibility exists for increasing the energy deposition rate to ~70 cal/g per second. For comparison, in pulse reactor, energies of ~100 cal/g are deposited in milliseconds.

In a workshop on accident conditions conducted in 1997, Project participants have recommended to focus joint programme work on supplementary analyses and information for understanding the mechanisms involved in reactivity transients. One point to be clarified is the role of the pellet outer rim, where the highest inventory of fission gas is present and where the highest temperature is experienced during the transient. This combination may produce extensive fuel swelling and/or excessive local gas pressure, especially in a closed gap situation.

The fuel swelling and the gas release process can be addressed at Halden by utilising highly irradiated disks which would be subjected to rapid power increases by moving an absorber shield surrounding the fuel. It is anticipated that fuel disks with a burnup in the range 100-120 MWd/kg could be made available for this purpose. This burnup range is representative of the burnup in the rim region of LWR fuel rods with an average burnup of 50-60 MWd/kg. The possibility exists to attach gas lines to the tubes containing the fuel disks with the aim of monitoring the fission gas release behaviour under normal conditions, in particular as a function of the progressive evolution of the rim structure for burnups beyond 70 MWd/kg.

Rod pressure/fission gas release measurements on high burnup fuel have shown that axial gas transport is severely impaired. One advantage of using fuel disks is that gas transport to the free volume and the pressure sensor is facilitated, thus the possibility exists to measure the pressure evolution during the transient.

Participants have pointed out that the fuel should be under constraint to closely represent the real situation. This can be achieved by having a ring of cladding material around the fuel disk during irradiation. In the axial direction, a compressive force can be provided which would also support the swelling measurement during the transient using an axial elongation detector.

An alternative to the disk design would be to utilise a segment of high burnup commercial fuel. Since the low reactivity of such fuel precludes the generation of sufficiently high power in Halden reactor flux conditions, it is intended to evaluate the possibility of imposing a rapid temperature transient by high current electrical heating. In this set-up, the gas release from the rim would be monitored by means of a gas line reaching the fuel surface through a penetration of the cladding.

### 3.3 Fuel Performance in Flow Starvation Conditions

*Short-Term dryout*

Light water reactor fuel may be subjected to thermal-hydraulic transients resulting in inadequate cooling for short periods of time as departure from nucleate boiling in PWRs and as short-term dryout in BWRs. The issue of fuel dryout is treated in different ways by different countries. A consensus exists, however, that the current criterion of a minimum critical power ratio (MCPR), used to assure fuel rod integrity in the event of off-normal transients which may lead to fuel dryout, imposes operating restrictions on the running of LWR plants. An incentive therefore exists to revise and improve the MCPR criterion, and it has been suggested that a rod failure criterion based on clad temperature history offers potential benefits. This requires extensive thermal-hydraulic data and analyses, complemented by in-pile test data and analysis, post dryout PIE and sample testing. The whole body of new information then must be applied and incorporated into safety analysis methods.

An experimental series has been undertaken in the HBWR from 1996 - 1998, designed to produce dryout and a concomitant rise in clad temperature to study the behaviour of irradiated BWR fuel rods under conditions of reduced coolant flow. The test rig comprises three individual flow channels, each containing one instrumented test rod. The flow transients can thus be imposed separately on each rod. Review meetings with member organisations have been held for discussing the test matrix, the test conditions and priority of the experiments. Another important issue has been to define the scope of the pre- and post-test material characterisation programme.

The test monitoring and control is based on measurements by cladding thermocouples and elongation transducers fitted to each rod. The first two loadings included one fresh and five pre-irradiated BWR fuel segments with burnups in the range 24-44 MWd/kg $UO_2$ and were aimed at peak target cladding temperatures of 550, 650 and 750°C.

As this programme proceeds and in-pile and PIE results become available, a follow-on series to supplement and extend the database will be discussed with member organisations. The scope of such an extension programme on dryout fuel behaviour will have to be defined at a later stage, but the Project has received proposals suggesting that these future tests should emphasise:

- use of different fuel rod diameters, BWR versus PWR dimensions,
- flow instability tests close to dryout at DNB conditions (cycle periods: 2-3 s),
- power increase tests, one-sided dryout and bundle type tests.

*Loss of Coolant*

The safety criteria for loss-of-coolant accidents were defined to ensure that the core would remain coolable. Since the time of LOCA experiments, which were largely conducted with fresh fuel, changes in fuel design, the introduction of new cladding materials and in particular the move to high burnup have generated a need to re-examine these criteria and to verify their continued validity. To this end, hot cell programmes concentrating on embrittlement and mechanical characteristics of high burnup cladding have been initiated in some participating countries. One of the points to be addressed is for instance how the 15-17% oxidation limit has to be applied to fuel rods that may already be considerably oxidised ahead of the transient.

In a workshop on transients conducted in autumn 1997, Project participants recommended integral in-pile tests on issues related to fuel behaviour under LOCA conditions. In particular, the interaction of bonded fuel and cladding and the behaviour of fragmented fuel around the ballooning area were seen as being of major interest for investigations. Other aspects to be addressed are the quench behaviour of corroded and hydrided fuel cladding under axial stresses and axial gas communication in high burnup rods as affected by gap closure and fuel-clad bonding.

The Project suggests that one parameter be considered with some attention, that is, the rod internal pressure at beginning of the transient, which may be considerably greater than the beginning of life pressure. Halden tests have shown that at normal operating conditions, high burnup rods can withstand considerable rod pressure - above system pressure - without impairing the fuel performance. For LOCA transients at high burnup it is not proven that high pressure has deleterious effects, or the extent to which deleterious effects may affect safety. In fact, the lower heat up rate, the very bad axial communication of gas from the plenum and local differences in cladding state may result in different failure modes than the ones observed in earlier tests with fresh fuel, and thus in different post-transient fuel geometries.

Taking into account the complexity of the test channels envisaged for LOCA studies and the efforts involved in fuel re-fabrication, single rods tests would be preferred. The test section can be designed to simulate the thermal boundary conditions and constraints provided by neighbour pins. Cladding temperatures have to be monitored and controlled during the overheating event, e.g. by surface thermocouples. A cladding extensometer will provide data on dimensional changes, supplemented by PIE. In considerate-

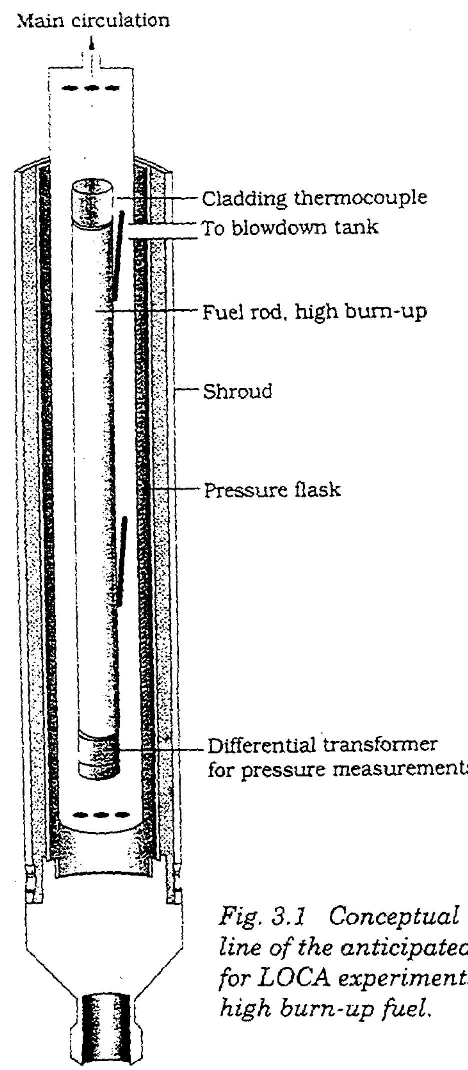



*Fig. 3.1 Conceptual ou: line of the anticipated ri; for LOCA experiments o high burn-up fuel.*

ion of the importance that axial gas transport may have, it is suggested that the test rodlets should not be too short. One point that needs to be clarified is the type of fuel rods and cladding material that would be used for the test. For a number of reasons it would be desirable that LOCA tests be conducted on the same fuel used for long-term irradiation at normal conditions.

### 3.4 Anticipated Transients without Scram

In a BWR, power oscillations can occur because of coolant flow instabilities in the core during an ATWS. According to simulation calculations, the power peaks can be an order of magnitude greater than the pre-transient power. In ATWS safety assessments, the enthalpy limit for reactivity-initiated-accidents (RIA) is applied to demonstrate the absence of fuel dispersal and related phenomena. Since cladding failures have occurred at relatively low enthalpies during recent RIA tests, the question has been raised whether the ATWS enthalpy limit should be revised.

ATWS simulations have shown that the transient is slow compared to the RIA event and that the cladding temperature remains relatively stable and low unless a much decreased cladding-coolant heat transfer is introduced. These conditions make it feasible to conduct a power oscillation test with impeded cooling in the HBWR, using a design similar to that of the dryout test. Flow oscillations have already been executed as part of the dryout test series, and an adverse effect on the cladding (e.g. failure) has not been observed.

Another point under discussion and evaluation is the development and extent of PCMI during power oscillations. Experimental data show that changes between zero and maximum operation power tend to impose an extra PCMI stress on the cladding which, however, is removed wholly or partially by fuel creep during steady state operation. This relaxation capability is reduced in an ATWS power oscillation event due to the short available time (seconds). On the other hand, the fuel has a thermal time constant comparable to the duration of one oscillation cycle and therefore the thermal contration and expansion is damped. This could have a mitigating influence on the amount of PCMI which can be quantified with cladding elongation measurements in an experimental setup.

Because of the low probability of the event, regulation authorities have not expressed an immediate safety concern. The Project intends to follow the developments which may emerge related to the subject, and if possible to include relevant testing capabilities in a new dryout rig and test series.

### 3.5 Axial Gas Transport in High Burnup Fuel

One aspect that may affect both LOCA and RIA transients is the ability of the high pressure gas contained in the fuel rod to migrate axially under the effect of local differential pressures. This was deemed of importance at the time when extensive studies were dedicated to LOCA (on fresh fuel), to the point that in the seventies tests sponsored by the USNRC were carried out at Halden (on fresh or low burnup fuel) for the purpose of characterising the axial gas transport. This item should be of greater importance for fuel having considerable burnup. The Project has sporadically performed, when suitable and feasible, measurements of axial gas transport in high burnup fuel. It is proposed that such measurements be carried out systematically in pre-defined sets of rods, in order to gather an organised data base on axial gas transport which can effectively be used by modellers for computer simulations of safety transients.

# 4. FUEL RELIABILITY ISSUES

## 4.1 Introduction

The operational performance of nuclear power plants has improved considerably in recent years in terms of fuel cycle economy, average capacity factor as well as personnel exposure. Increased burnup, extended operational cycle and power upratings have been introduced, or are being considered, in order to improve the fuel economy.

New technical solutions are being proposed by fuel vendors aimed at supporting the utility efforts to improve fuel reliability in a variety of demanding service conditions. At the same time, authorities have to verify that the trend towards more varied fuel materials and designs and more varied and challenging conditions occurs without detriment to safety. Authorities and industry tasks are however concurrent in many respects, at least in that fuel performance reliability and predictability is in the interest of both safety organisations and utilities.

Operational experience shows that unforeseen deviations form normal fuel performance do occasionally occur in some conditions, and in some plants causing limitations to the power output from the plant and even premature shutdown in order to discharge the fuel affected by the anomaly. Such events appear to be related to a combination of causes - for instance specific water chemistry and/or power history - and may sometimes have implications on safety, for instance on control rod efficiency.

Remedies may involve design modifications or adoption of new materials or modifications of the boundary conditions. In any case, a proper understanding of the phenomena involved is essential in order to arrive to cost-effective solutions and reduce the occurrence of similar anomalies in the future.

This section addresses the following issues:

- Crud deposition as affected by water chemistry conditions and operating power.
- Axial offset anomalies caused by local boron accumulation on PWR fuel rods.
- Localised corrosion, possibly caused by the presence of local radiation fields.
- Degradation of failed fuel resulting in severe exposure of the fuel to the coolant and in consequent high radiation levels in the coolant.
- Control rod sticking as results of axial growth of guide tubes during service.

It should be clarified that the joint programme cannot address all these issues in their entirety and that Halden work must be put in the context of parallel PIE and laboratory activities conducted in other programmes. Further, a complementarity between joint and bilateral work scope at Halden must be sought in order to pursue the above issues with the necessary resources. Dialogue with many Project participants is in this respect already taking place as related for instance to the issues of axial offset anomalies and of control rod sticking. As to other items such as localised corrosion and degradation of failed fuel, it is expected that the joint programme will address generic studies on basic mechanisms, while the behaviour of commercial materials will need to be covered by tests conducted on behalf of participants.

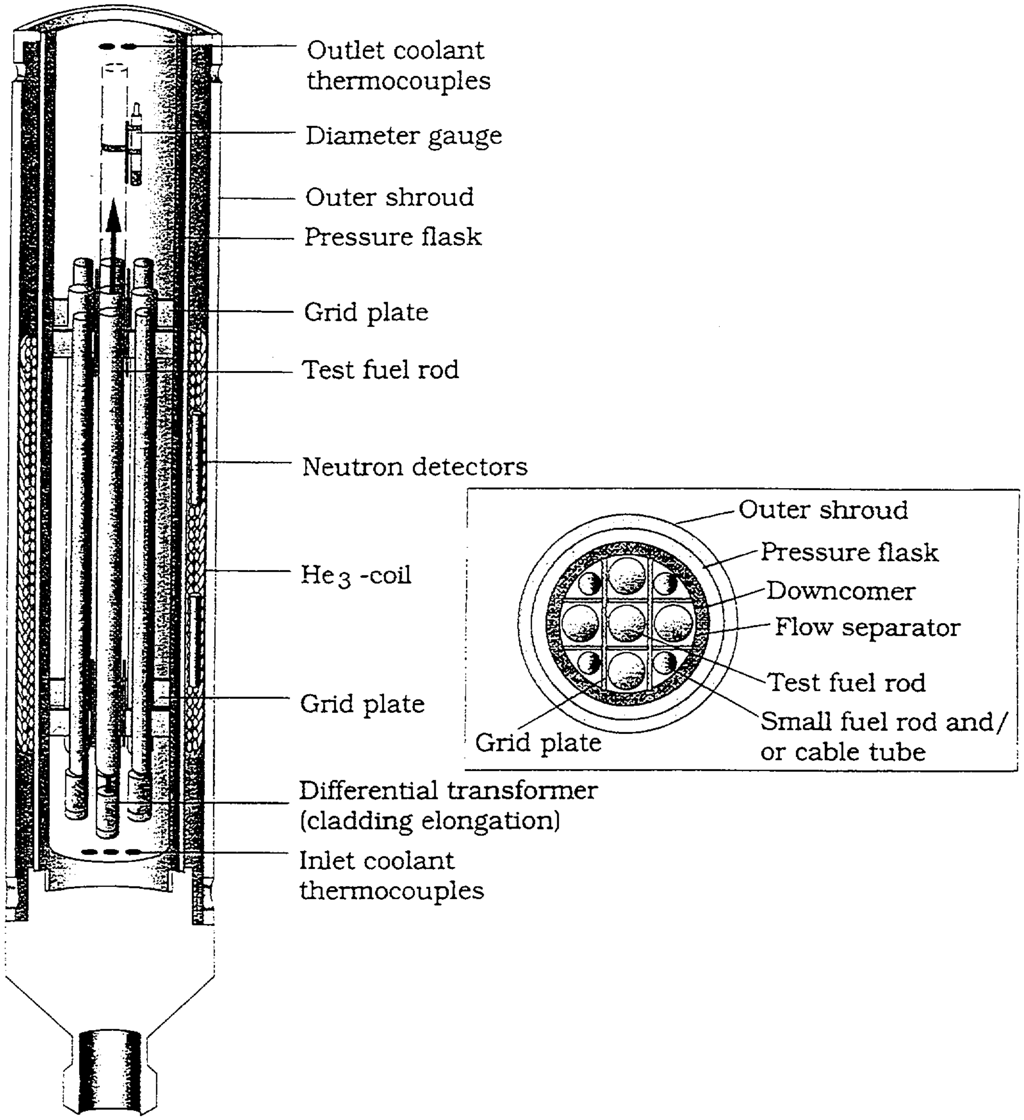



*Fig. 4.1. This figure shows a tentative design of the crud deposition rig, in which the crud layer is measured by means of diameter profilometry of the moveable centre rod. This technique has shown to be accurate for the purpose of the test, as demonstrated by earlier experience. The rig is rather sophisticated in that it contains a square arrangement of rods aiming a simulating an actual sub-channel. However, a simplified design may consist of a single rod arrangement, with the diameter gauge travelling along the (stationary) rod. Discussions are taking place for the use of such design for instance in relation to axial offset anomalies studies which are intended to be performed on behalf of a participant.*

## 4.2 Crud Deposition

Soluble and insoluble impurities in the coolant can deposit as crud on the cladding surface. The process is favoured by local boiling and high coolant temperature, depending also on the water chemistry conditions of the water. The crud deposit may disappear or decrease during shutdowns since the solubility of the crud components is generally higher at low temperature. Some of the consequences of crud formation may be increased pressure drop, increased cladding corrosion and radiation build-up. These effects may be accentuated by the trend towards increased rating and low leakage cores.

Operation with additives to the coolant, which is considered both for BWRs and PWRs, may to some extent influence the crud deposition onto the fuel and questions have been posed as to the consequent cladding corrosion rate and hydrogen pickup. Crud deposits have caused uneven pressure drops and coolant temperatures also in VVER cores.

Earlier tests at Halden have shown that crud deposition rate may be strongly affected by local heat flux, i.e., by local power conditions. For the period 2000 - 2002, it is suggested that the heat rating effect under given water chemistry conditions be addressed more in detail, since it determines the ability of the fuel to withstand power upratings.

Besides investigating the mechanisms leading to crud deposits, utilities are very interested in determining the conditions under which the crud may be re-dissolved, for instance during power decreases and/or shutdowns.

The possibilities of a crud formation study to be conducted in Halden loops have been discussed in a workshop and in an expert group meeting in 1997. A challenge in crud formation studies is the on-line determination of the effect of parameter variations on crud build-up and resolution. The proposed test rig as outlined in Fig. 4.1 contains a centre rod that can be moved for measuring the diameter change and thus the crud layer thickness. The resolution is typically 1 μm.

It has also been suggested to include a thermocouple in one of the stationary rods to assess the consequence of deposition in terms of cladding temperature changes, which may be useful for appraising the thermal resistance of the crud layer. Due to the thermal resistance, increasing oxide and crud layers can cause an increase of cladding temperature and thus enhance the corrosion process. In the previous programme periods, tests were carried out aimed at determining the conductivity of the oxide film. The issue of thermal conductivity can be considered again in conjunction with the crud formation test, this time for determination of the conductivity of the combined crud and oxide layer.

For studies at BWR and PWR conditions, existing loops at Halden already have the built-in capability of varying the water chemistry parameters on-line as well as the test section power and the coolant flow rate and temperature. Pending further discussions in countries using VVER reactors, the possibility of realising a VVER loop at Halden aimed at materials and water chemistry tests will be explored.

As a final remark, it should be noted that studies on radiation build up and on cobalt deposition in the primary loop surfaces have also been carried out in the past and may return of actuality in consideration of possible variations of coolant chemistry in the future.

### 4.3 Axial Offset Anomalies

In the past, problems with excessive deposition of crud on PWR fuel cladding were fairly common. However, after taking account of corrosion product solubility studies, coolant pH was much better controlled, leading to the virtual disappearance of crudding problems until the recent spate of axial offset anomalies (AOA).

AOA arises from a lowering of the neutron flux in the top half of the core due to boron "hide-out" in relatively thick porous crud on areas subject to nucleate boiling. Nucleate boiling has increased recently in many plants due to higher peaking factors, (low leakage cores), power uprating and higher boron concentrations (18- and 24-month cycles). Heavy boiling crud deposition can occur even when chemistry control is good.

An increasing number of plants have suffered from axial offset anomalies, which may become a greater potential problem in the future since intentions exist to uprate some of the PWR plants. The consequences of AOA include errors in estimated control rod position and shutdown margins, reduction of operating power, increase in dose rate and possible association with fuel failures.

The specific issue of axial offset anomalies is being addressed in other international programmes and it is envisaged that tests in a Halden loop will be conducted under bilateral arrangements with Project participants. The objectives of such tests would primarily be to assess the conditions leading to the initial crud build-up at conditions representative of plants in which AOA has occurred and to investigate realistic options to remedy AOA by means of operational measures.

### 4.4 Localised Corrosion

In LWRs and in particular in BWRs local corrosion enhancements have been observed on fuel rods and on fuel assembly structures. This phenomenon is not fully understood, although it is believed to be related to the vicinity of metals which become radioactive under neutron irradiation. The consequence can be reduced service time and potentially failure of the fuel.

Previous tests at Halden conducted on behalf of a participant have demonstrated that corrosion spikes can arise along the fuel rod as consequence of the local radiation field caused by the presence of platinum, which becomes a beta-radiation emitter in the reactor environment. It is possible that the occurrence of local enhanced corrosion observed in those tests is related to local electrochemical potentials arising from the flux of charged radioactive particles.

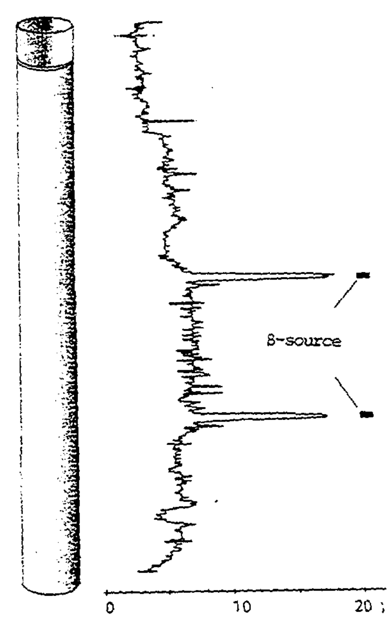



*Fig. 4.2 Oxide thickness a: affected by local beta sour*

As a further step to understand local corrosion, it is suggested that investigations be conducted to assess the effect of metals/alloys more representative (than platinum) for LWR fuel assembly, for instance alloys containing nickel. It can also be of interest to determine possible detrimental effects on corrosion caused by local crud deposits, which also can be beta-emitters (and alpha-emitters if boron can accumulate on PWR fuel surfaces).

The Project work in this field should be directed to a generic better understanding of the conditions leading to enhanced corrosion, possibly in conjunction with corrosion and/or crud studies carried out in other parts of the programme. More specific tests addressing the response of commercial alloys are beyond the joint programme scope and are to be carried out separately.

### 4.5 Fuel Failure Degradation

Improved quality control and fuel designs have virtually eliminated defects from the fabrication process and from pellet-cladding interaction. Better control of the core operation has also contributed to reducing the occurrence of failures. In spite of these improvements, however, a limited number of failures still occur in LWRs, primarily because of cladding fretting caused by small foreign objects transported by the coolant and trapped in the fuel assembly.

Parallel with the introduction of fuel assembly design remedies to further decrease the possibility that primary failures occur, efforts are being made to understand the mechanisms involved in the fuel degradation subsequent to primary failure. Efforts have also been made at Halden to provide test conditions which are representative for cases in which the degradation of a primary failure can occur and/or a test environment suitable for separate effect studies. Integral studies of fuel failure degradation are being performed in bilateral programmes, with the objective of demonstrating the resistance of improved products to secondary fuel failure degradation.

The joint programme work is being focused on producing repeatable test conditions in which the oxidation rate and the hydrogen pick-up rate can be determined by means of on-line measurements. To this end, the Project has initiated a test series in the previous programme where data on steam reaction and hydrogen pick-up kinetics are being obtained from a simple and controlled set-up. The results should provide valuable information for codes which are being developed for modelling of the process of secondary failure degradation. It is proposed that selected tests on fuel failure degradation be continued in the next three year programme, possibly making use of pre-irradiated materials.

### 4.6 Control Rod Sticking

In some conditions, PWR assemblies develop a gradual S-shape bending along their vertical axis, probably as results of in-reactor growth of structural materials. This phenomenon has been observed in a number of US and European plants and has caused authority concerns in that it hinders the axial movement of the control rods.

This issue is being addressed by the industry by means of assembly design modifications. For this to be effective, data on axial creep of structural materials up to extended burnup are needed. While this report is being written, consultations are taking place for a possible investigation on this subject in one of the Halden loops within the scope of an international fuel programme.

# 5. PLANT LIFETIME ASSESSMENTS

## 5.1 Introduction

As the age of nuclear power plants increases, safety authorities will need materials property data relevant to in-reactor components at high irradiation doses, as these will form the basis for plant lifetime assessments. Utilities are introducing operational changes that can enhance the reliability of plant structural components, e.g. water chemistry modifications, and are adopting advanced materials where this can be done. Pressure vessel annealing is a possible option for mitigating the effect of radiation embrittlement. Practical verifications and data will be needed to support lifetime predictions of existing and replacement materials as well as to validate measures leading to lifetime extensions.

The aim of the experimental work proposed for the 2000 - 2002 programme period is to improve the understanding of the materials ageing and degradation processes as well as to demonstrate the merits of mitigating methods and measures. Studies are to be performed in loops able to simulate light water reactor conditions and in particular specific water chemistry conditions.

The work initiated in previous programme periods on irradiation assisted stress corrosion cracking will be extended to include highly irradiated materials. The programme is intended to clarify the extent to which remedies introduced to alleviate the stress corrosion of in-reactor components remain applicable to components which have been in service for a long time. One focus will be on representative BWR materials which have been retrieved from commercial reactors and on the use of these materials for in-core measurements of crack growth rates at given stress intensities. A second focus will be on stress corrosion studies under PWR conditions, as it is anticipated that cracking of in-reactor materials in PWRs may also become an issue of concern.

Water chemistry is important both for fuel cladding corrosion and stress corrosion cracking of in-reactor and primary circuit components. Further, water chemistry affects the crud deposition on fuel rods and the radiation level of the plant. Based on input from participants, studies on water chemistry effects will be carried out to assess the merits of different water chemistries.

The embrittlement of reactor pressure vessel materials due to neutron irradiation is an important issue as nuclear plants age. The Project intends to support collaborative programmes with participants in this area as needs arises, notably by utilising the Halden reactor as a course of neutrons under a wide variety of temperature and flux and fluence conditions.

## 5.2 Irradiation Assisted Stress Corrosion Cracking

The Project has since 1991 addressed Irradiation Assisted Stress Corrosion Cracking (IASCC), the materials degradation phenomenon which affects the structural integrity of core component structural materials such as stainless steels and nickel base alloys. The in-pile investigations are performed in facilities simulating radiation, thermal-hydraulic and water chemistry conditions typical of those found in commercial BWRs. The studies focus on the IASCC susceptibility of stainless steels and nickel base alloys, the effects of radiation and water chemistry on IASCC, the quantification of crack growth rates as a function of stress intensity, and the benefits of mitigation measures such as Hydrogen Water Chemistry (HWC) and noble metal or other coatings.

The test objectives and scope of the IASCC investigations are defined in co-operation with Project participants at annual review meetings where experimental needs are reviewed and updated, the short-term goals for on-going investigations are defined and recommendations for new experiments are discussed and prioritised. At the most recent review meeting in May 1998, participants endorsed further IASCC studies in the 2000 - 2002 programme period and recommended specific topics for programme continuation and extension.

The first IASCC study conducted at Halden was aimed at determining whether HWC could suppress crack propagation in stainless steels and nickel base alloys commonly found in commercial LWRs. The results clearly demonstrated the beneficial effect of hydrogen additions in mitigating crack growth in thermally sensitised materials as well as in materials with an irradiation induced susceptibility to cracking.

In order to overcome the limitations of constant displacement wedge loaded Double Cantilever Beam (DCB) specimens, actively loaded DCB and Compact Tension (CT) specimens were developed and produced in co-operation with Project participants for crack growth versus stress intensity studies. The specimens are equipped with bellows which enable the on-line control and variation of the stress intensity. The feasibility of the bellows loading concept for crack growth specimens was demonstrated with an in-pile qualification test.

As important step in providing for more versatile IASCC studies has also been made with the development of CTs containing active material. The geometry and design of the specimens is based on the same principle as the bellows-loaded CTs. Depending on the amount of available irradiated material, square bodies or circular inserts electron beam welded into carrier material form the crack growth region. Since the more complicated parts of the specimens are machined from unirradiated material, the difficulties associated with the preparation and handling of corresponding, full-size crack growth specimens is considerably reduced. The utilisation of welded-in irradiated material as the "test piece" has proven particularly valuable in studies aimed at assessing the benefits of IASCC countermeasures in materials with high end-of-life fluences.

The IASCC experiments conducted or planned in the current programme period have taken advantage of the developments described above and provided results and experience on which to base the plans for the 2000 - 2002 programme period.

The effects of water chemistry and stress intensity on crack growth in pre-irradiated materials are studied with CT specimens with bellows loading and potential drop wires for crack growth monitoring. The test has demonstrated that the technology for utilising materials with high accumulated fluences can be used to advantage in the Halden reactor.

The data from this test show:

- higher crack growth rates in the high fluence 304 SS material compared to the low fluence material,
- the mitigating effect of HWC which, however, is less effective for high fluence material,
- a delayed response to water chemistry changes.

A follow on test with similar objectives will be irradiated during 1999 and 2000. The rig layout and the test matrix as currently foreseen are shown in Fig. 3.1. For this test, participants have recommended to concentrate on the difference between high and low fluence materials and repeatability of crack growth data. Specifically, the test matrix selected will:

. provide additional crack growth data on the high fluence control blade handle material (304 SS, fluence $9x10^{21}$ n/cm$^2$),

- enable comparison of reproducibility in cracking behaviour of the two 347 SS specimens at similar stress intensity levels and/or comparison of cracking behaviour at different stress intensity levels,

- evaluate the cracking behaviour of two different materials with similar fluence (347 SS versus 316 NG),

- compare the crack growth rates of high and low fluence specimens ($9x10^{21}$ versus $0.9\text{-}1.5x10^{21}$).

It should be noted that one CT specimen is equipped with an alternative way for crack growth measurement. The *C*rack *M*outh *O*pening *D*isplacement (CMOD) method will be compared with the DC potential drop technique. The advantage of using CTs with CMOD measurement is that the specimens can be exchanged, thus allowing the reuse of irradiation rig.

As for future studies in the 2000 - 2002 period it is recommended that, in addition to the execution of IASCC related experiments as described above, the following points should be considered:

- *Determine crack growth rates at low stress intensities* ($k \leq 15$ MPa m$^{1/2}$) for normal and hydrogen water chemistries.

- *Provide data derived from long-term measurements.* In order to decrease the influence of transient effects produced by change of water chemistry or stress intensity, and to increase the statistical significance of crack growth data, constant conditions should be maintained for long periods (> 1 month).

- *Concentrate on high fluence BWR materials.* Exposures of > 10 dpa (>$6x10^{21}$ n/cm$^2$) are of particular interest. They can occur, e.g. in control rod materials to which attention has been drawn again more recently.

- *Assess the effect of surface coatings on crack growth.* The benefits of noble metal coating in enhancing the effectiveness of HWC has already been studied by comparing the crack growth behaviour of coated versus uncoated DCBs under partial HWC conditions. Participants have suggested conducting similar investigations using a zirconium oxide film on the specimen surface. As a first step, the layer will be produced in an autoclave before mounting the specimen(s) in the irradiation rig. After demonstration of the effectiveness of this crack growth countermeasure, attempts may be made to produce the coating in-pile. Candidate material could be 347 SS from the Würgassen power plant top guide - the same material will be used uncoated in IFA-639 thus allowing a comparison with the cracking behaviour of coated material.

- *Extend the IASCC programme to PWR conditions.* Material irradiated to 35 dpa in a PWR can be made available by a participant for IASCC testing purposes. A loop suitable for providing PWR conditions is available, and other low and high fluence materials now being studied under BWR conditions can be included for comparison of e.g. 316L used for baffle bolts.

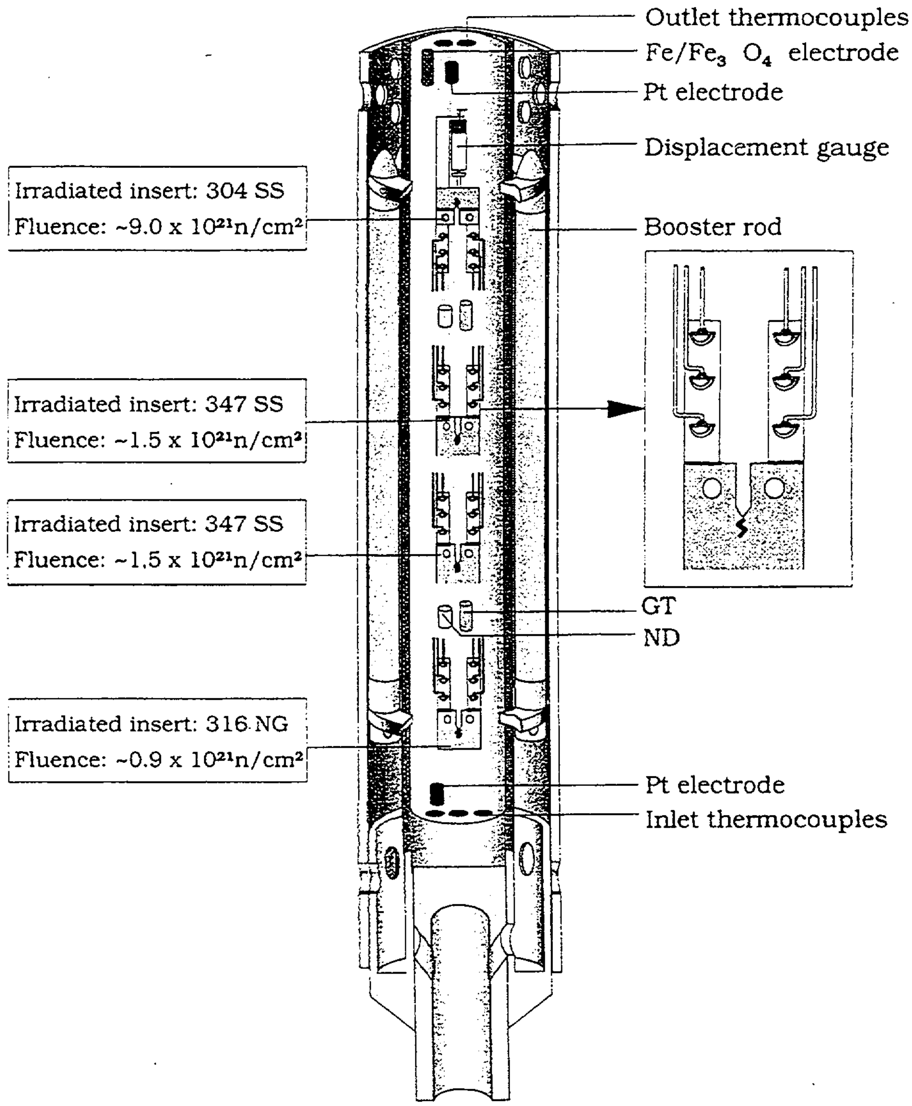



*Fig. 5.1 View of the IFA-639 rig, designed to determine crack growth rate of high dose BWR structural materials*

Given that participants express continued interest in PWR-IASCC investigations during discussion of this programme proposal, the Project intends to provide a rig design by the beginning of 1999. An irradiation rig can then be available at the beginning of the 2000 - 2002 programme period, while particulars of the experimental programme will be defined in discussions within the IASCC advisory group at the HPG.

In addition to studying the effects of environmental parameters on the crack growth behaviour of structural materials, dry irradiation of specimens and subsequent mechanical testing and microstructural characterisation at specialised laboratories will continue during the 2000 - 2002 programme period. The main objective is to determine the effects of irradiation damage, chemical composition and fabrication procedure on the mechanical properties of austenitic stainless steels. The study, which is being performed on a joint programme basis, is carried out in co-operation with the Argonne National Laboratory (ANL) and is sponsored by the USNRC.

The investigation, a total of 96 slow strain rate tensile and 24 compact tension fracture toughness specimens, machined from 19 laboratory heats and 8 commercial heats of austenitic stainless steel and arranged in six stainless steel capsules are being irradiated to low (~0.4x $n/cm^2$), medium (~.9x$10^{21}$ $n/cm^2$) and high (> 2.5x$10^{21}$ $n/cm^2$) fluence levels.

One low fluence and two medium fluence capsules have already been discharged and shipped to ANL, where the specimens are being subjected to hot call tests to determine the effects of irradiation on the fracture toughness and tensile properties. In 2000, target fluences will have been accumulated in the remaining 3 capsules, which will be shipped to ANL. The final results of the PIE are expected to be available in the course of the following year.

The effects of fluence and stress level on crack initiation are studied by means of pressurised tube specimens. One half of the tubes is surrounded by booster rods which provide a high fast neutron flux, while the other half is located above the booster section where the fast neutron flux is negligible.

In parallel with the in-pile study, out-of-pile investigations on selected test materials are conducted in co-operation with a participating organisation. Two phases are foreseen:

*I* *Effect of NWC and HWC water chemistry on SCC*

Constant extension rate tensile (CERT) tests will be conducted using tensile specimens prepared from sensitised and solution annealed 304 SS. The specimens will be tested under NWC and HWC conditions. This phase will to a large extent be completed in the current programme period.

*II* *Influence of water chemistry and fluence level on SCC behaviour*

CERT specimens will also be prepared from tube materials irradiated to different fluence levels. It is expected that this phase will mostly be a part of the 2000 - 2002 programme period.

### 5.3 Pressure Vessel Ageing

The embrittlement of reactor pressure vessel materials due to neutron irradiation remains an important safety correlation as nuclear plants age. Research work can identify the mechanisms involved in the embrittlement process and devise possible mitigation measures. Further, experimental findings can improve the predictability of the residual service time.

Extensive work has been and is being performed in a number of organisations in relation to pressure vessel embrittlement and to the mechanisms involved at alloy microstructural level. In discussions with participants, new research needs have been pointed out specifically in the area of neutron flux and spectrum effects and for what concerns the validation of surveillance methods.

Neutron embrittlement effects on pressure vessel materials is traditionally assessed on the basis of testing Charpy-type specimens irradiated as part of surveillance programmes. One shortcoming of such programmes is the limited amount of material available which is being resolved through reconstitution techniques and miniaturisation of test specimens. However, it is essential to establish proper correlations between the data from subsize specimens with those of standard size, an issue which requires further experimental verification. This is of importance in relation to assessments of the effects of pressure vessel annealing and subsequent re-embrittlement and in particular with respect to the establishment of fracture mechanics data.

The damaging effect of neutron irradiation is primarily attributed to the high energy component of the neutron spectrum. Questions exist, however, on the possible effects of epithermal and thermal neutrons which might have been underestimated so far. Another issue is whether accelerated irradiation produces conservative results.

For what concerns irradiation conditions, techniques are available, and in use at Halden for bilateral tests, for providing irradiation at representative temperature and flux conditions. These are achieved by encapsulating the specimens in a metal capsule having well defined geometry and filler gas (normally helium or argon), such that the specimens are brought to the desired operating temperature. With respect to the fast neutron flux and spectrum conditions, a large variety of irradiation conditions can be realised within the HBW core.

For establishing of experimental work in this area, the Project will continue consultations with participants, in particular with a view to identify issues of generic character as a complement and enlargement of activities performed elsewhere. In this respect, participation in international co-operative programmes will be given priority. Furthermore, while it is anticipated that the irradiation is carried out in the Halden reactor, collaborative arrangements for the mechanical testing are to be established with participating organisations.

### 5.4 Plant Dose Reduction

The Project is operating several loop systems under BWR and PWR conditions, in which water chemistry parameters can be varied. Although the loops are primarily in support of the fuel and cladding programmes as described above, they have during recent years also provided space and conditions for investigations related to activity build-up on materials used in the primary systems of power reactors. In these studies, the environments used were representative of BWRs with NWC and HWC as well as switching between them, and of PWRs with different lithium concentrations. Coupons of materials such as 304 stainless steel, VVER steel (titanium stabilised stainless steel), Inconel-600 and -690, as well as Incoloy-800 were included in the tests. Also, zinc depleted in Zn-64 has been added to these environments and materials. The effect on the activity uptake of cobalt has been studied during several reactor cycles in order to follow the long term behaviour of cobalt and zinc. The effect of interrupting zinc addition on the behaviour of cobalt has also been studied.

The incentive of these studies is to provide knowledge on how to achieve a dose reduction as the activity builds-up on the system surfaces increases in ageing nuclear power plants. A promising way to reduce the build-up is by changing the water chemistry, e.g. by using additives such as dissolved zinc or magnesium in concentrations of a few ppb. The short and long term consequences of changed water chemistry must be assessed by experiments. As an example, the interruption of zinc injection can cause a situation that is worse than without injected zinc. This can be verified by well controlled experiments in high temperature water, simulating the environment in a nuclear power reactor and addressing the mechanisms involved in the transport of radioactive nuclides.

Alternatives to zinc as additive have been suggested: magnesium, chromium, nickel, silicon, aluminium etc. Magnesium is a prime candidate for further investigations since, unlike zinc it does not have the drawback of long-lived radioisotopes.

It is foreseen that activities in the field of crud build-up, corrosion and water chemistry will continue in the 2000 - 2002 period. With respect to the benefits of zinc addition on the activity build-up, the work will be based on coupons installed in loops in the HBWR. The long time effects on the activity build-up after interruption of zinc addition will be studied in PWR environments. Hydrothermally zinc doped oxides of different ages will be used for studying the release of zinc from the oxides as well as the build-up of radioactivity. As reference, non-zinc doped hydrothermally grown oxides of the same age and environment will be used. In the BWR environment the effect of magnesium in NWC and HWC on the activity build-up will be compared to coupons from a non-magnesium containing coolant.

The morphology and microstructure of the oxides on the coupons will be studied with SEM (Scanning Electron Microscopy). The chemical elemental composition in the grains and grain boundaries will be determined by EDS (Energy Dispersive (x-ray) Spectroscopy). Depth profiles of the elemental composition will be determined by DSIMS (Dynamic Secondary Ion Mass Spectroscopy) or GDMS (Glow Discharge Mass Spectroscopy). The radioactivity uptake will be determined by $\gamma$-spectroscopy.

Acknowledging the fact that water chemistry parameters have strong impact on all materials in the core as well as in the primary system of a reactor, the Project anticipate extended activities in this area. To this end it is intended to further qualify and expand the experimental infrastructure, the analytical capabilities and the monitoring means. It is foreseen that details of the experimental work will be established in close consultations with participants and in particular with utilities.

# MAN-MACHINE PROGRAMME

**HALDEN REACTOR PROJECT**

**MAN-MACHINE PROGRAMME**

**Proposal for the**
**Three Year Period 2000 - 2002**
**May 1999**

CONTENTS

## SUMMARY OF THE MAN-MACHINE PROGRAMME

Advanced computer-based human system interface technologies are being introduced into existing nuclear power plants to replace the existing interfaces. These developments can have significant implications on plant safety in that they will affect the overall function of the personnel in the system: the amount, type, and presentation of information; the ways in which personnel interact with the system; and the requirements imposed upon personnel to understand and supervise an increasingly complex system.

The programme for 2000 - 2002 is intended to address the above issues by means of extensive experimental work in the human factors, control room design and computer-based support system areas. The work will be based on experiments carried out in the upgraded Halden Man-Machine Laboratory facility (HAMMLAB) which will become an even stronger nucleus of the research programme. The proposal is to a great extent based upon input from Project's participants and contains four main areas of activity.

One activity relates to the *Experimental Programme and Operation of HAMMLAB*. The use of the laboratory will increase in terms of type and size of experiments, extended operational regimes and more realistic work settings. Installations of several advanced operator support systems on the new simulators are proposed to demonstrate the benefits of such systems in an integrated control room environment. An ambitious experimental programme is planned that will make extensive use of the *HAMMLAB* facility. The programme will address a wide range of research and development issues including control room layout, interaction modes, information presentation and display design, levels of automation, human error and collaborative work.

The activity on *Human Factors and Control Room Engineering* aims to extend the knowledge about the characteristics of human performance in process control environments and to demonstrate how this can be used in the specification and design of solutions to specific problems. The proposed programme addresses hybrid and advanced control rooms, contemporary and future man-machine interaction, interface design, alarm handling, virtual reality and training, function allocation and modelling, human reliability and further human performance method development.

The *Plant Operational Support* activity aims to explore and demonstrate system solutions that have potentials for improving plant performance and optimising plant operation as well as improving operational safety. The proposed activities address issues such as plant performance, maintenance, readiness and procedures as well as accident and emergency management. Also, it is proposed to investigate how new technology as e.g. Virtual Reality can be used in decommission planning.

One focus of the *System Safety and Reliability* activity is to investigate the benefit of formal software development methods to develop computer systems with high reliability requirements. It is proposed to continue these investigations and to extend the applications to new areas. The integration of computer systems in plants makes it necessary to evaluate these in the total safety assessment context. It is proposed to study the incorporation of different evaluation methods for safety assessment of programmable plant control and supervision systems in the coming period.

## Schedule of Man-Machine Systems programme 2000-2002

| Issue | 2000 | 2001 | 2002 |
|---|---|---|---|
| **EXPERIMENTAL PROGRAM AND OPERATION OF HAMMLAB** | | | |
| **HAMMLAB and VR-lab** | HAMMLAB operation development | | |
| | Laboratory development | | Consider pure VVER simulator ▽ |
| **Experimental and Demonstration program** | Integr. of alarm systems and demo | Alarm system tests ▽ | Advanced control room experiment ▽ |
| | Integr. of signal validation and demo | Integr. of acc. man. system and demo ▽ | Hybrid control room experiment #1 |
| | Integr. of procedure system and demo | Procedure experiment ▽ | ▽ |
| | Man-machine interaction modes tests | | ▽ |
| | Interface research experiment | | ▽ |
| | | Level of automation tests | ▽ |
| **Simulator MMI and Communication software** | User Interface management system development | ▽ | |
| | Integration platform development | ▽ | |

## Schedule of Man-Machine Systems programme 2000-2002, cont.

| Issue | 2000 | 2001 | 2002 |
|---|---|---|---|
| **HUMAN FACTORS AND CONTROL ROOM ENGINEERING** | | | |
| **Control Room Design**<br>Advanced control room<br>Hybrid control rooms<br>Virtual mock-up's | Establish PWR CR<br>Decide approach<br>Apply in CR design | Tests/experiments<br><br>Tools for design criteria check | improvements<br>Test<br>Coupling VR/simulator |
| **Interface design**<br>Interaction modes<br>Interface R&D | Test, navigation ,Interaction devices Guidelines<br>Test BWR/PWR overview display | Test task-oriented, adaptive displays | |
| **Alarm handling**<br>Alarm philosophy, system imp.<br>Tool development | Alarm Philosophy-PWR system devlm.<br>Development as needed for | and testing Philosophy and system<br>COAST | improvement Guidelines |
| **VR for training** | New training environment | VR for maintenance | VR for outage planning |
| **Function allocation and Human Centred Automation**<br>Human centred automation<br>Function allocation | Information gathering, documentation<br>Model development Function allocation | Tests/experiments Improved<br>strategies | documentations, guidelines<br>Continued development |
| **Human reliability**<br>Human reliability assessment<br>Hum. error and perf, teamwork<br>Oper. Perf. during night shifts | Review existing methods, identify gaps<br>Broadening scope Team performance<br>Tests, HBWR operators | Improve HRA methods practical use<br>experiment Performance | <br>during maintenance. Effect of management |
| **Human performance measures and data analysis methods** | Continued improvements as needed | (more efficient methods, improved | situation awareness, workload) |

Schedule of Man-Machine Systems programme 2000-2002, cont.

| ISSUE | 2000 | 2001 | 2002 |
|---|---|---|---|
| **PLANT OPERATIONAL SUPPORT** | | | |
| **Plant Performance**<br>Thermal Performance Monitoring | Establish user needs | System requirements ▽ Develop | modular, configurable workbench ▽ |
| **Plant Maintenance**<br>Condition Monitoring, Maintenance Strategies<br>Diagnosis and Maintenance Data | Specify PEANO extension ▽<br>Specify OO-database ▽ | Maintenance strategy selection<br>Integrate service data and diagnostics | Develop integrated system ▽<br>Applications ▽ |
| **Plant Readiness and Operability**<br>Low Power Operation<br>Plant Readiness and Operability | Specify support system requirements ▽<br>Reqs. to configuration management ▽ | Develop specific COSSes (shutdown<br>Electronic TechSpecs and procedures | SPDS, procedure system etc.)- ▽<br>Status checking COSSes ▽ |
| **Plant Procedures** | Electronic procedure tools in HAMMLAB ▽ | Procedure evaluation | Guidelines for procedure design ▽ |
| **Accident and Emergency Management** | Tracking simulation development | CAMS-BWR integration ▽ | |
| **Decommissioning Planning** | Integration of CAD and VR | Dismantling planning ▽ | Radiation dosimetry ▽ |

## Schedule of man-machine systems programme 2000-2002, cont.

| Issue | 2000 | 2001 | 2002 |
|---|---|---|---|
| **SYSTEM SAFETY AND RELIABILITY** | | | |
| **Formal Development Methods** | | | |
| Formal Specification | Evaluation of languages and tools | Recommendations and guidelines for | different applications |
| Formal Specification of MMIs | Specification of language for MMI design | Development of language and tools for | MMI design |
| Formal Methods Development Tool | Further development of HRP Prover | Instruction material, tutorials and verific. | Applications |
| Formal Methods and SW Quality | FMs in integrated QA | Evaluation criteria, metrics | Effectiveness of FMs |
| **Verification Methods** | | | |
| Program Analysis | Specification of tools for OO-programs | Development and assessment of tools | |
| Formal Verification | Investigate model checking in verification | Case studies, procedure verification | |
| Testing | Testing strategies selection | Measure cost-effectiveness of strategies | |
| **Assessment of COTS and Legacy Software** | Assessment methods | Standards compliance, retrospective analysis | Case studies |
| **Risk based Software Assessment** | | | |
| Specification based Risk Analysis | Formal specifications in risk analysis | Case studies | |
| Probabilistic Methods | Use of BBN in PSA | Applications to real systems | Assessment of embedded SW |

# 6 EXPERIMENTAL PROGRAMME AND OPERATION OF HAMMLAB

## 6.1 Introduction

The Halden Man-Machine Laboratory, *HAMMLAB*, has for a number of years played an important role in the man-machine systems area. A considerable extension of the functionality and capability of the laboratory is performed during the present programme period through the *HAMMLAB* 2000 project in terms of procurement and installation of two more nuclear simulators and the establishment of the complementary Virtual Reality Laboratory.

During the upcoming period *HAMMLAB* will be further developed in terms of its organisation, and installation of advanced support systems. An ambitious experimental programme is proposed that will make extensive use of the laboratory.

## 6.2 HAMMLAB and Virtual Reality Laboratory

- *HAMMLAB*

HAMMLAB as of 1999 consists of one full-scope simulator, an advanced control room with a number of CRTs and a Large Overview screen. It presents a unified MMI and several operator support systems are available. Various types of experiments are controlled from the experimenters gallery. Test subjects are operators from commercial NPPs or the Halden reactor.

After the completion of the HAMMLAB 2000 project, HAMMLAB contains three nuclear simulators and one oil production platform simulator, for more details see chapter 11.2.1.

- *NORS* is a "westernised" PWR of VVER-440/213 type.
- *CP-0* is a Westinghouse type 900MW 3-loop PWR.
- *F-3* is an ABB type 1160 MW BWR.
- *Oseberg* is a full-scope simulator of the Oseberg A oil production platform in the North Sea.

- *Virtual Reality Laboratory*

There is a growing recognition of the need to integrate human factors principles into planning and design as early as possible. Virtual Reality technology offers strong potential for the achievement of this objective. Use of a VR design tool allows design engineers to interact directly with their developing designs by taking the viewpoint of the end-user. Future operational problems that may be design-induced, and which historically do not become apparent until very late in the design process, can be identified at an early stage. A VR Laboratory with necessary hardware, software and competent staff was recently established to complement the HAMMLAB activities during the present period.

- *Use of HAMMLAB*

It is presumed that the activity level in HAMMLAB will increase considerably in the next programme period, however, the type of activities will remain about the same as today, i.e.:

- human factors experiments,
- studies related to control room design, and
- studies and demonstrations related to operator support system design and development.

Some characteristics of the use of HAMMLAB are given in the following:

*Type and Size of Experiments*. A prudent mixture of a few large and more smaller experiments per year are foreseen in HAMMLAB in the new programme period. The laboratory has been developed to meet such requirements.

*Extended Operational Modes*. The introduction of new simulators with extended capabilities improves the possibilities for performing studies in a broader operational domain. Low power operation and accident states can now be addressed.

*More Realistic Work Settings*. Two different schools exist in human factors research: the well-controlled laboratory experiment and the field study. Both have provided useful results, but they have sometimes been in apparent conflict in terms of the concepts and theories they have used. It may therefore be an appropriate task for an experimental facility to demonstrate that the two schools can be combined. This will lead to experiments - or "simulated field studies" - with a higher degree of realism (longer scenarios, more realistic tasks, more naturalistic and complex work settings, work in teams), to supplement the more conventional hypothesis testing experiments. This will require the development of a different set of methods, e.g. for analysing complexity, understanding how work becomes organised, modelling the interaction between people (communication and control), etc.

*Increasing the Pool of Operators*. The availability of few commercial power plant operators has previously been one of the limiting factors of HAMMLAB. With the introduction of new simulators of common western type of nuclear power plants, the availability of a broad pool of operators will facilitate access to experimental subjects.

- *HAMMLAB Operation*

Since HAMMLAB has been substantially expanded the organisation around the laboratory needs to be strengthened in the coming period. Operation of the facility, performing modifications and upgrades, and conducting experiments need considerable resources.

*Organisation*. The new HAMMLAB organisation needs expertise in areas such as process, computers, simulators, and human factors. The aim is to provide a flexible and up-to-date laboratory, which can be used for various experimental purposes.

*Test Subject Training*. It is necessary to provide suitable test subjects to ensure validity of experiments performed in HAMMLAB. The HAMMLAB staff will focus on the type of competence required of test subjects, the modification of existing or creation of additional training modules, their execution, and the development of a thorough assessment programme to assure the attainment of the training objectives. Particular emphasis will be placed on training material for operating the new CP-0 and F-3 simulators.

*Experiment Preparation and Data Analysis*. The growing activity in HAMMLAB demands more time and cost effective routines. The potential for reducing the amount of resources needed to set up experiments will be exploited. One will look into methods for improving data collection of simulator and audio/video data, visual activity and usability of the physical laboratory infrastructure. Other issues are development of standard/automatic data collection procedures for better resource management, improved experimental routines and better accuracy in time estimates for personnel resources.

*Laboratory Development Programme.* The HAMMLAB 2000 project is planned completed by the end of 1999, but the next period will also contain development activities related to the facility. The new BWR and PWR simulators need to be brought up to the same level of operation as the current NORS simulator which means that the simulators should have a complete mimic-based MMI, overview displays, and an advanced alarm system.

A pure VVER-simulator will be considered for installation in HAMMLAB towards the end of the three year period.

Each of the simulators in HAMMLAB will initially have their own experimenters system. A project to develop a unified experimenters system, as a front-end to all simulators, was initiated in the present period and the design and implementation of such a system, including the linking to all simulators is proposed for the next period.

The infrastructure of the laboratory, i.e. hardware and software systems and experimental equipment will continuously be upgraded to meet the requirements set forward by the experimental programme.

## 6.3 Experimental and Demonstration Programme

An ambitious *experimental programme* is planned that will make extensive use of HAMMLAB during the coming period. The program addresses a wide range of issues including, but not restricted to;

- *Advanced and hybrid control rooms.* The HAMMLAB control rooms will serve as examples of efficient use of modern technology in control room design. Experiments to evaluate the quality of solutions may either be performed internally in HAMMLAB or co-ordinated with studies elsewhere, see also 7.2.
- *Integration and evaluation of support systems.* Support systems (COSSs) will be implemented and tested out as individual systems, but also integrated in a uniform manner into the total, unified MMI as a whole. Integrated and combined use of a number of COSSs will be investigated.
- *Procedure evaluation.* Procedures play a particularly important role in the control room. Experiments will be conducted to investigate efficient use of procedures in various operational situations, see also 8.5.
- *Man-machine interaction modes.* New technology opens for new ways of communicating between operator and process, such as sound based systems and touch based interaction. Studies will investigate the potential of these methods in the NPP control room environment, see also 7.3.
- *Interface research and development.* Possibilities exist for improved information presentation in hybrid and advanced control rooms. Types of displays to be developed and

experimentally evaluated include overview displays, task oriented and adaptive displays, see also 7.3.

- *Varying level of automation.* Various levels of automation are found in existing nuclear plants. An extensive programme will investigate the total effect on efficiency and safety of level of automation and presentation of automatics in the control room, see also 7.6.
- *Human performance.* Human error studies have been an ongoing activity in HAMMLAB. In the future experimental programme, human performance in general will be studied. Special emphasis will further be put on team performance, see also 7.7.
- *Experimental methodologies.* To meet the goals of the experimental programme, efficient evaluation methodologies must be available. Also in the future, methods will be developed or adapted to serve the programme needs, see also 7.9.

The aim of the *Installation and Demonstration Programme* is to gradually build up the advanced control room facility with new functions to demonstrate the advanced features in an integrated control room environment, and to provide new and important functions for the experimental programme.

- *Alarm Systems and Event Identification COAST/ALADDIN.* The alarm system toolbox COAST has previously been used to build and successfully demonstrate the CASH advanced alarm system for the NORS simulator. The main features of the event identification system Aladdin, built on neural net and fuzzy logic techniques, have recently been demonstrated. It is proposed to install and demonstrate these combined modules on both the CP-0 and F-3 simulators, see also 7.4.
- *Signal Validation - PEANO.* The signal validation module PEANO has recently successfully demonstrated its capabilities to identify failing or drifting measurement readings by means of neural net and fuzzy logic technology. It is proposed to integrate PEANO in the unified MMI of the new simulators to demonstrate its usefulness in a control room environment, see also 8.3.
- *Computerised Procedures - COPMA.* During the present programme period the Project is developing a computerised procedure system built on international document standards and web-technology. It is proposed to integrate this system called COPMA-III in the new control room environment for CP-0. This will enable the Project to demonstrate the feasibility of the system as well as using it for experimental purposes. The purpose of the experiment is to evaluate the ability of operators to handle daily work, disturbances and accidents by means of computerised procedure systems, see also 8.5.
- *Accident Management - CAMS.* During previous programmes the Project developed a BWR prototype version of an accident management system called CAMS. It is proposed to integrate this system in the control room environment for the F-3 simulator. The purpose of this exercise is to evaluate the benefit of such a computerised tool in handling complicated transients and accident situations, see also 8.6.

## 6.4 Simulator MMI and Communication Software

- *User Interface Management System*

The Picasso-3 system offers the user interface designers an efficient and versatile tool that reduces the efforts in design and experimental evaluation of advanced presentation formats and techniques.

It is proposed to develop the Picasso-3 system further to comply with the requirements of the HAMMLAB software structure and the human factors research. The rapidly changing software environments, i.e. operating systems, new languages and technology, do require that the Picasso-3 system is updated accordingly, for example the integration and usage of JAVA, VRML and ActiveX will be evaluated. In addition the system will be integrated with 3D (VR) software packages to investigate presentation methods not available today.

The Picasso-3 system is primarily used for real time process control and supervision. It has been seen that an increasing number of administrative functions are introduced and integrated in the supervision systems. An ODBC (Open DataBase Connectivity) module will be developed to enable the Picasso-3 system to access data in database systems using this interface standard.

- *Integration Platform*

The work on the Integration Platform which is a concept for simplifying the building of the simulator systems in HAMMLAB will continue. It is a goal that this system shall reduce the maintenance and development costs of the laboratory as well as increase the availability of the simulator systems for experiments.

To investigate the usability of remote supervision and control the Picasso-3 system will be expanded with functionality to simplify the integration of this technology in the HAMMLAB laboratory and in principle, at utilities.

Also communication hardware and software develop rapidly and there will be a demand to adhere to standards preferred by the industry. For simulator systems which need to transfer large amounts of data, new communication hardware may remove the bottlenecks which sometimes occur in Ethernet based networks today. The SoftwareBus system developed in the Picasso-3 project will be further developed and bridges to the CORBA standard will be considered for interfacing to software systems using this technology.

# 7 HUMAN FACTORS AND CONTROL ROOM ENGINEERING

## 7.1 Introduction

The objectives for Human Factors and Control Room Engineering research are to extend the knowledge about the characteristics of human performance in process control environments and to demonstrate how this can be used in the specification and design of solutions to specific problems. Prudent decisions on issues relating to process control and NPP safety require adequate understanding of the relations between new technology, operator, and performance.

The basis for achieving these objectives will be a combination of research, design, and test and evaluation activities that addresses the central issues in process control and man-machine interaction. The proposed research programme therefore addresses issues such as hybrid and advanced control rooms, contemporary and future man-machine interaction, interface research and development, alarm handling, function allocation and automation, human reliability and further method development.

The emphasis will be firmly placed on producing practical advice for member organisations to demonstrate how the Project develops and applies human factors principles and techniques to the design and evaluation of modern process control environments. This advice will be supported by a combination of sound experimental findings, theoretical understanding, and practical design experience.

## 7.2 Control Room Design

The nuclear industry is today faced with important decisions concerning whether to transform the original conventional control rooms into hybrid control rooms, or to completely replace the older analogue I&C equipment with modern digital equipment, thus introducing a modern advanced control room into an old plant. Against this background, the programme proposal addresses aspects important for both advanced and hybrid control rooms.

- *Advanced Control Rooms*

Preliminary experiences with modern VDU-based control rooms indicate that they offer many advantages over older conventional types. However, there remain many issues that require further research and clarification. The basis for the proposed work is the experimental advanced control room in HAMMLAB, refer Fig. 7.1. The current HAMMLAB has been designed around a Unified Control Room MMI Concept, where all interfaces to the process and the support systems are designed to appear and function as parts of one single interface system for the operators. The functionality and interface within HAMMLAB is similar to that of the new advanced control rooms, such as the N4 in France and ABWR in Japan. Thus the current HAMMLAB is a realistic prototype of a modern, advanced control room where further development work can be done and comparisons made to present design solutions. Consequently the following programme items will be investigated:

*Overall Control Room Efficiency*: The CP-0 PWR simulator in HAMMLAB is equipped with CRT-based photo panels which emulate all the panels in the conventional control room of the

## 7.3 Interface Design

- *Contemporary and Future Man-Machine Interaction Modes*

Rapid technological developments have made it possible to extend the way in which humans interact with technological systems. The successful design and use of such technology requires an understanding of both the basic theoretical underpinnings of man-machine interaction and of the issues specific to particular means of interaction. This element of the programme will investigate and document these issues by considering three aspects: (1) The mode that the interaction takes, e.g., conventional visual-tactical/motor based interaction, as well as newer innovative interaction modes such as sound based systems, touch based interaction; (2) The interaction devices themselves, both advanced (virtual reality) and traditional, and their benefits and limitations; (3) Broader aspects which cut across the above, including, navigation within computerised information spaces, workload, the feasibility of advisory systems, etc. Deliverables would include both experimental results and practical advice to member organisations concerning the relative merits of both contemporary and future interaction modes, interaction devices and their suitability and applicability, and guidelines for both design and evaluation of wider interaction issues such as navigation.

- *Interface Research and Development*

The presentation of information within hybrid and advanced control rooms is likely to be via so-called soft controls combined with CRT-based mimic displays. The presentation and combination of the many possible types of information and display elements present challenges to both designers and operators. In addition, the potential impacts of newer forms of displays including overview displays, task oriented displays, and adaptive displays should be explored in a systematic manner. The flexibility to present information in a wide variety of ways has the potential to either enhance operator and team performance or introduce problems. This programme element will investigate these issues from both a theoretical and practical standpoint. Benefits and limitations of both conventional and newer forms of display and information presentation will be explored by a process of research, design, and evaluation. Work will be reported exploring both the theoretical justification for, and practical implications of, the findings for process control display design.

## 7.4 Alarm Handling

Existing alarm guidelines and philosophies make many assumptions concerning how and why operators make use of alarm systems. Empirical support for some of these assumptions is lacking, and important issues relating to what the operators actually use the alarm system for, what the operators wish to use it for, or what is the best way to use the alarm system, remain inconclusive or unanswered. HRP aims to provide empirical support to alarm philosophies and guidelines - to improve the technical basis for their content. The next three year period will build on previous HRP work in this area and through a combination of new experiments, and further analysis of data from previous experiments, essential and specific questions related to alarms will be investigated. Research will include the use of alarm systems and their effects on operators as individuals and as a team, and seek to identify the best way to generate, structure, and present alarms.

Several activities will be performed. The first task will be the development of an alarm philosophy that includes overall alarm design standards. This should be a living document, based on results from new experiments, reanalyses of old experiments, bilateral work, input from industry practice and other research facilities. The document should outline the state of the art of alarm systems, and be an alarm system guideline applicable for designing alarm systems. The second task will apply the findings from the work outlined above to the application and realisation of the alarm philosophy in the design of the alarm systems for the CP-0 Fessenheim simulator in HAMMLAB.

The third activity is related to COAST - the computerised alarm system toolbox developed at the Project. It has shown to be a flexible tool for implementing advanced alarm systems and facilitate human factors experiments. Many member organisations also use the tool, and the tool is proposed to be further developed in response to the needs and requests from the member organisations. The whole environment around the tool will be steadily improved with respect to set-up and configuration, and in terms of user guidance, courses and tutorial material.

Development of a formal (or semi-formal) description language directed towards the description of the contents and dynamics of alarm systems user-interfaces will also be considered.

### 7.5 Virtual Reality Technology for Training

VR can be used as a tool for training personnel in different tasks, for example process supervision, maintenance and outage work, process understanding or emergency routines.

It is proposed to investigate how a combination of traditional operator display stations and VR-technology can be used to make a simulator system where e.g. panels are dynamically coupled to the simulator models and the physical infrastructure of the control room is modelled.

A VR-model is proposed designed and implemented with the purpose to investigate the effect of training personnel in maintenance and outage work. Experiments will be carried out using crew from the HBWR, training in a practical work situation at the reactor plant.

Experiments are proposed to investigate the usability of VR for modelling different types of environments. The intention is to collect and analyse data that can be used to develop basic guidelines for the design of virtual environments and VR systems.

These experiments will focus primarily on the effect of different VR interface techniques and hardware options on the users' ability to comprehend spatial information. These experiments are important as it is anticipated that they will provide information that can be used both to improve the design of virtual environments and also provide an indication of which display technologies are adequate for different applications. The choice of display technology is an important cost factor in Virtual Reality projects. It is important that a cost-effective choice is made, at the same time meeting the demands of an application, both with regards to spatial understanding and image resolution.

## 7.6 Function Allocation and Human Centred Automation

During the next three year programme work will be carried out on two complementary projects investigating the role of automation and its relationship to, and effect on, individual operators and shift teams. With the objective of developing a set of models and tools to investigate and support the design of human centred automation the programme will combine results from both the human centred automation project and the function modelling and allocation project.

- *Human Centred Automation*

Automation has traditionally been driven by engineering needs and concerns, sometimes to the detriment of operator performance. Work is needed to establish a basis for automation design that acknowledges the needs of the operators and the demands on the human-machine system. A central concept here is the notion of controllability of a system, that is the extent to which operators subjectively and objectively control the system, and how this can be analysed and designed. This work will build on the work carried out during 1998-1999 involving a method for classifying levels/types of automation, new experimental measures, and the initial work on controllability. The project will seek to verify and validate these aspects in both experimental and applied settings. An important basis for such work is the systematic study of experiences, for instance, as event reports or field studies. Experiments will be performed in HAMMLAB to test the theoretical models and establish their explanatory power in light of information from event reports and field studies.

- *Functional Allocation and Human Centred Automation*

A common denominator when introducing operator and decision support systems is that they change the automation degree and the role of the operating crew. The Project has in the current three-year period started a project called FAME, Function Allocation Methods, to study this from an analytical point of view. FAME is a tool utilising functional modelling or goal-oriented methods called Goals-Means Task Analysis, GMTA, to model the operator and the automation.

The FAME tool and related methods may be used in different applications relating to operator support, control and automation systems. In the coming three-year period the following topics will be studied:

*Basic Study of Function Allocation.* This study is closely co-ordinated with the Human Centred Automation studies, (see above) and aims to test different function allocation strategies through simulations using the FAME tool. It can also assist in planning and analysing experimental activities. In this way a synergy effect between the experimental and analytical approaches can be created, strengthening both sides.

*Interaction and Information System Analysis.* Given a specific level of automation or a set of support systems, different human-system interaction strategies and interfaces may be tested and evaluated at an early stage. This reflects the fact that the type of information given to the operator, and in what way, is as important as the actual level of automation. In the same way tools and goal-oriented methods like GMTA could be used directly in task oriented MMI design.

Based on the GMTA models and simulations one may analyse the MMI before it is built, not only afterwards as with traditional task analysis methods.

In order to support the above research topics, the FAME tool itself will be further developed. FAME is based on normative models, and these should be evaluated against experimental data.

### 7.7 Human Reliability

- *Human Reliability Assessment*

During the coming period the Project will establish a new programme to investigate human reliability assessment (HRA). This will be closely coupled and complementary to the research work already carried out at Halden in the area of human error and performance.

The Nuclear Energy Agency (NEA) reported in 1998 that the use of standard PSA techniques has matured in recent years. Nevertheless, there were significant differences between PSAs in the implementation of the same HRA methods. The programme will consider the implications of recent and ongoing theoretical work for existing and emerging methods, and provide practical help for assessors.

Firstly, emerging issues in the HRA area, such as errors of commission, the treatment of diagnosis or decision-based errors, and dependencies and techniques will be investigated. The work will contribute to this area in terms of theoretical and conceptual development, and in the evaluation of how current and emerging techniques are able to deal with these issues. Secondly, we aim to help improve HRA methods and how these are used in practice in safety cases and probabilistic safety assessment. Even if existing techniques are theoretically sound, there is often considerable variation in how they are interpreted and used. This can threaten the practical reliability and validity of the techniques. In both of these areas full use will be made of work within the Human Error Analysis Project (HEAP), in member countries, and in other research institutes. An objective for the work will be to improve the reliability and confidence in existing probability values, and where possible generate new data.

It will be important that the work does not duplicate efforts in member countries and organisations, or other research organisations. Halden will therefore establish links to the activities in member countries and focus on the needs of members, especially during the initial planning of the programme. The need for multidisciplinary competence in this area means that international co-operation will be very important.

- *Human Error and Performance*

The concern for the quality of human performance has traditionally focused on the negative aspects, in the form of human error and human reliability. Most of the developments have focused on quantifiable models for individual performance. The HEAP programmes of 1994-1999 have demonstrated a clear need to extend this approach. Consequently the Project will research and report on a number of important aspects relating to human error and performance during the coming period, including; (1) both positive and negative aspects of performance, i.e., performance quality in general, with a distinction made between performance efficiency and performance reliability; (2) performance outside of the main control room and operational activities supporting maintenance; (3) extending the focus for investigation from the individual operator to team (collaborative) performance and organisation; and (4) its relation to integrated performance and safety analyses rather than PSA-dominated analyses.

- *Team Work*

In addition to the reliability aspects, there is a strong industry need to investigate collaborative or co-operative work. This will require the development of concepts and models to describe team and individual performance and shared performance management, as well as proposals for experiments on various scales to explore specific ideas further.

- *Operator Performance During Night Shifts*

Prior research performed at night addressing biological rhythms, performance and shift work has dealt with the simpler mental functions, such as reaction time and performance of simple psycho-motoric tasks, and simple cognitive tests. The research on higher order cognitive functions is apparently always done during the day when human performance is at its peak. There is an increased risk for performance failures at night due to the body's circadian rhythms and their effects on mental and physical functions, and the sleep deficits often incurred as a result of working rotating shifts. Although causal relationships are difficult to establish, many industrial accidents happen during night shifts.

During the present period the Project established a programme to investigate higher order cognitive functions during night shift. The contents of a longer term programme will be determined by the results of this work. These studies, currently being analysed, seek to establish a baseline of performance for HBWR operators as well as exploring the feasibility of investigating their cognitive performance within their natural environment, i.e., during their normal shift cycle whilst operating the HBWR.

## 7.8 Human Performance Measures and Data Analysis Methods

HAMMLAB has proven to be a centre for authentic and systematic Human-Machine interaction research, see figure 7.1. However, the combination of high realism and experimental control remains a scientific challenge. To meet this challenge, the development of methods and tools for systematic experimental investigation in HAMMLAB will be continued.

In the context of ongoing HAMMLAB activities, different types of experimental design and statistical analyses will be tested in order to establish techniques for compensation of crew variation and unwanted scenario effects. A parallel approach will be further refinement of the complexity profiling system developed within the Human Error Analysis Project. These efforts are meant to increase experimental control without sacrificing the realism and ability to generalise results from HAMMLAB experiments.

Effective experimental manipulations in dynamic operating environments presuppose an advanced methodology for human performance measurement. Existing measures of situation awareness and cognitive workload based upon visual activity, and indicators of operator and plant performance, will be improved continuously. In this respect, one high priority is validation of current human performance assessment systems. This, in turn, will lead to an increased understanding of underlying performance criteria and the relationship between different types of measurements.

# 8 PLANT OPERATIONAL SUPPORT

## 8.1 Introduction

A large number of existing nuclear power plants are planning substantial upgrading of their I&C systems and control rooms. The trend is to replace analogue I&C equipment becoming obsolete with digital systems, and in the control rooms computer-driven VDU systems are replacing conventional panels. At the same time new plant designs have nearly fully digital I&C systems and compact, computer-based control rooms.

Another trend in the electric power industry is the increasing competition between the different producers due to deregulation of the electricity market. Thus, efficient plant operation and maintenance while at the same time maintaining plant safety requirements becomes increasingly important to nuclear utilities.

The computer-based I&C systems and control room solutions have potentials for increasing plant performance and optimising plant operation as well as improving operational safety of nuclear power plants. The Plant Operational Support programme for the 2000-2002 period will explore and demonstrate these possibilities through development and assessment of methods and systems for plant operation and maintenance. The planned activities comprise development of new and more robust plant monitoring systems, thermal performance optimisation, better planning tools including systems for low power operation, accident management and procedure management systems.

Further, work will be initiated on how new technologies as e.g. Virtual Reality, can be utilised in planning a decommissioning project.

## 8.2 Plant Performance

- *Thermal Performance Monitoring*

Monitoring of the thermal efficiency of nuclear power plants is expected to become increasingly important as energy-market liberalisation exposes plants to increasing availability requirements and fiercer competition. The general goal in thermal performance monitoring is straightforward: to maximise the ratio of profit to cost under the constraints of safe operation. One may perceive this goal to be pursued in two ways, one oriented towards fault detection and cost-optimal predictive maintenance, and another determined at optimising target values of parameters in response to any component degradation detected, changes in ambient conditions, or the like.

Work on thermal performance monitoring have been initiated in the current three years period. This work suggests the design of a flexible workbench for easy assembly of an experimental thermal performance monitor. The suggested design draws heavily on the Project's experience in implementing control-room systems featured by assets like high levels of customisation, flexibility in configuration and modularity in structure.

The proposed work on thermal performance monitoring comprises the following activities:

- Establish user's needs through close co-operation with utilities.

- Investigate novel techniques for thermal performance monitoring, in particular the application of neural nets as alternatives to models based on physical balances.
- Study accuracy and other requirements for thermal performance monitoring.
- Design and establish a modular, configurable workbench for integrating various types of software needed in a thermal performance monitor.

The results of these activities are expected to be an increased awareness among Project member organisations of the possibilities for computer-assisted monitoring of plant performance, better exploitation of existing software, and in the end, improved plant economy.

### 8.3 Plant Maintenance

- *Condition Monitoring and Maintenance Strategies*

Condition monitoring of a system or component means to assess the current state and estimate the future state of the system, using real time measurements and calculations. The difference between actual and optimal conditions, caused by system or components degradation, is used to evaluate the need and the time for maintenance, where the cost-benefit ratio for the operation is the lowest possible. Increased availability, performance and life time are the reasons why condition monitoring could be the preferred maintenance strategy for a plant subsystem or component.

During the last period a system for process signal validation called PEANO, has been developed. This system is based on AI techniques and its main feature is that there is no need for development of a physical model of the process to be monitored. PEANO can be used as it is, to diagnose instrumentation failures or drifts during normal and/or abnormal process conditions. An extension of this program is proposed for the upcoming three-year programme, that is to use PEANO as a building block for a condition monitoring system.

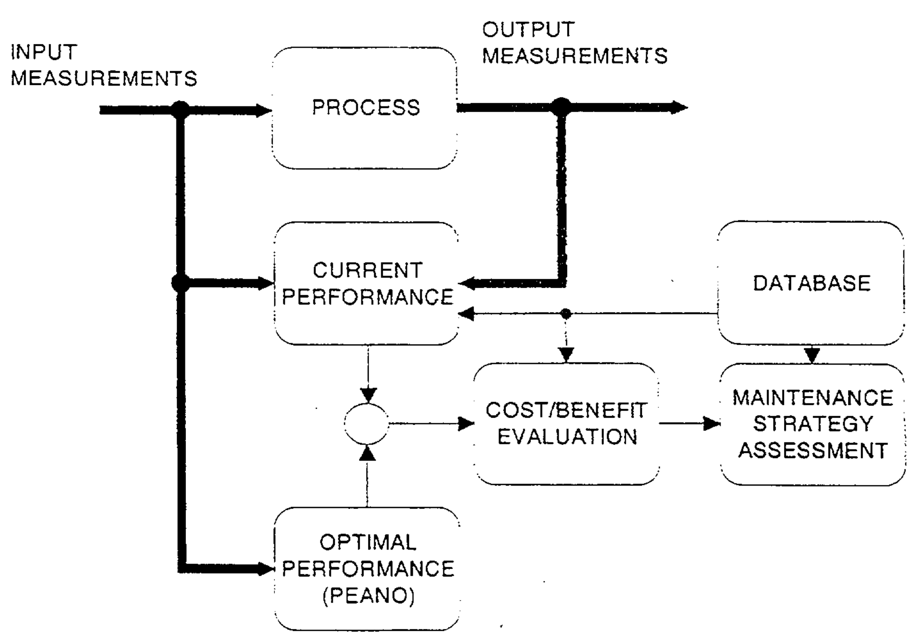



*Figure 8.1: A Condition Monitoring System Using PEANO*

Fig. 8.1 shows how PEANO can be used to track the optimal performance conditions of the component (system), in terms of efficiency, heat load, vibration level, etc, (depending on the monitored system), while the current performance block estimates the same parameters using the actual input-output measurements. The loss of performance trend is the input to the Decision Making block, which based on cost/benefit evaluation shall derive the right maintenance strategy for this component (system). Optimisation of the overall performance is the target of the Strategy Assessment block. It is proposed to develop a condition monitoring and maintenance system as outlined in Figure 8.1 in the 2000-2002 period.

- *Diagnosis and Plant Maintenance Data*

It is proposed to continue the work on knowledge based diagnosis systems. An essential quality of such systems is their organisation of knowledge. The Integrated Diagnosis System project (IDS) aims to develop an application independent knowledge framework. This framework can be used for storing and organising the knowledge of both diagnosis systems and other types of operator support systems.

For this purpose object-oriented databases which make it possible to integrate diagnosis systems and general plant data will be investigated. The plant data should contain information related to the service history of the various plant components. Plants are collections of inter-related objects (pumps, valves, heat-exchangers) and their data should be stored in an object-oriented database. A seamless integration between diagnostic knowledge and plant maintenance data within one object-oriented database opens up for more intelligent follow-up of service plans. Oppositely, knowledge about previously serviced components can help both operators and automatic diagnosis systems in finding a good diagnosis for an upcoming fault.

"Data Mining" is an area within the AI field that aims to extract interesting observations from a monitored system. This is particularly interesting for diagnostic systems in areas where model-based systems have usually performed poorly. Such areas are:

- Vibration monitoring of rotating equipment, like pumps and turbines
- Noise analysis. Here process noise is the data source, including reactor noise for BWR core stability monitoring.

Vibration monitoring is a fundamental block for predictive maintenance of rotating machinery, where, for example, the only currently available alternative for bearings state assessment is to do preventive maintenance.

In the current three-year programme a state identification project, ALADDIN, has been started, and applied in alarm filtering systems. The results from this project is proposed to be used as a basis for development of a diagnostic system with the above mentioned goal in the next three-year period.

### 8.4 Plant Readiness and Operability

- *Low Power Operation*

A preliminary study on information requirements and development needs in nuclear power plants during outages has been conducted in the current period. The focus has been on refuelling outages from the viewpoint of the control room personnel.

Support systems are needed both for planning and execution of refuelling and maintenance outages. Key issues in outage planning are the requirements of technical specifications for the availability of operational, safety and support systems in different plant configurations. For example, the possible high-risk configurations and tasks should be clearly identified. A well-structured model of the plant and outage tasks allows the use of computer supported planning and estimation of the risk levels associated with different plant configurations. It can also be used for optimising the outage time and utilisation of resources.

The study will be used as a basis for identifying areas where the Project could contribute to safer and more efficient low power operation. Examples of research topics which will be pursued are computerised procedures for maintenance operations, special displays for different tasks and plant modes e.g., special safety parameter display systems for shut-down modes, and the use of virtual reality for maintenance training.

- *Plant Readiness and Operability*

Maintaining the correct, safe status and alignment of all plant systems is a very complex task. Configuration management is a structured method to obtain this goal applicable to plant design, construction and modifications, as well as plant operation and maintenance.

In plant operation and maintenance configuration management is particularly important during outage periods and return to power (warming up, starting-up). During low power operation the plant configuration and operational states of different plant systems change very frequently. Safety requirements and rules of Technical Specifications create complex dependencies and restrictions limiting the number of allowed configurations. Good configuration management shall assure that maintenance work tasks are not performed before the necessary safety umbrella has been established, i.e., the work permits are available and the systems concerned are isolated and labelled by the operators. In returning to power it is equally important that all operations, safety and support systems have been reset and made ready for operation.

Currently, configuration management is to a large extent based on paper work and manual actions. Procedures and manual checks are performed to prepare for changing plant state. There is a need for developing more automatic approaches to configuration management and the rapid development in information technology makes more efficient systems for configuration management possible. In the new period the Project will address development of tools for plant configuration management. This work may include computerisation of the rules and documents defined in Technical Specifications and linking this information to real-time data from the process control systems and task data from the outage management system. Linking to living PSA in the form of a risk monitor could also be considered.

Further, new operator support systems to check plant system configurations are needed. The Project will address the development and inclusion of a new operator support function within the plant configuration management system. This operator support system should check plant system status before changing plant state. Such a function could be initiated either automatically from a computerised configuration management system, or manually on request from the operators.

### 8.5 Plant Procedures

Procedures are an essential part of the working environment in any NPP control room. The traditional way to manage procedures is through using volumes of printed paper. However, the usage of printed procedures poses several problems, e.g., the possibility that the operator fails to execute instructions in the procedure. Further, maintenance of printed procedures is cumbersome and error-prone. The introduction of computerised procedures may solve many of the problems associated with paper-based procedures, in particular in modern VDU-based control rooms, and the Project proposes to continue its development efforts on procedure management and procedure tools addressing the following three topics:

*Procedure Development and Implementation Tools*. The work on computerised procedure tools will be focused on how current and emerging computer-based technologies, e.g., Internet/intranet standards and technologies can be used for addressing problems experienced in procedure handling in the industry. At the end of the present three-year period a prototype (COMPA-III) together with a specification of how to build an "industrial" computerised procedure system (CPS) will be available. This system will consist of two main parts: an integrated procedure development and administration system serving the needs of all actors involved in the preparation and subsequent life-cycle phases of procedures, and a system supporting the control room operator in on-line execution of procedures. In the upcoming three-year period a main task will be the implementation of this "industrial" CPS in the HAMMLAB environment and promotion and support of its use in Project member organisations.

*Procedure Evaluation*. The work at the Project has so far concentrated on providing assistance to the operator in safe execution of the procedure. It has been taken for granted that if the procedure is correctly implemented the desired goal state will be reached. Also, the purpose of using procedures in control of the plant is to enforce the behaviour of an "ideal operator" in the plant situation for which the procedure is written, thereby facilitating the operator's tasks and decrease his/her workload, and these features of the procedure have also been taken for granted.

In the upcoming three-year period it is proposed to initiate research on the procedures intended beneficial effects on operator behaviour and desired goal state. This work will draw upon the planned work on Function Allocation Methods and operator modelling (the FAME-project) (see 7.6) and will also utilise formal methods used in software specification (see section 9.2).

*Guidelines for Procedure Design.* With the planned flexibility in COPMA-III it will be possible to perform human factors research on alternative ways of designing and presenting procedures. The COPMA-III system will contain configuration possibilities that are powerful enough to create quite different kinds of procedure systems in HAMMLAB. It is proposed to perform comparative human factors studies both with respect to procedure structure and procedure presentation to establish guidelines with respect to what type of procedure system should be utilised in different contexts.

It is important that a computerised procedure system is easily integrated with the remaining systems in the control room. Since COPMA-III is based on upcoming standards for component-based software, it is able to run any unknown software module observing the standard. As an example, availability of a component-oriented version of the Picasso-3 system, would enable direct integration of Picasso-3 pictures in the procedures. It is proposed to do research on how a seamless integration of procedure information and traditional mimic-diagrams can enable development of task oriented displays (see also section 7.3).

### 8.6 Accident and Emergency Management

Severe accident management involves many activities that aim to minimise the off-site consequences of an event. The long-term target of the Halden Project is to develop a support system to help the operators, plant technical support and emergency organisations in adopting the right strategies in order to bring a plant in trouble back to a safe shutdown.

A working prototype of the CAMS accident management system has been developed in previous programme periods. The benefit of such a tool should be demonstrated and the plan is to integrate CAMS in the new BWR simulator for HAMMLAB.

In consideration of the importance of real-time process tracking in normal and abnormal conditions, for any system with predictive and diagnostic capability, the main objective is to continue the research on tracking simulation in particular related to accident management. A second objective is to integrate the current CAMS prototype with the BWR simulator of HAMMLAB and demonstrate its feasibility.

The expected results are improved methods and techniques for real-time process tracking in normal and abnormal conditions, and CAMS integrated with the BWR simulator in HAMMLAB and results from operator's use of the CAMS system in accident scenarios.

### 8.7 Decommissioning Planning

As more nuclear installations begin to reach the end of their useful lives, decommissioning projects have become more common, and the technical aspects of the decommissioning process have become better understood. Decommissioning is, however, a challenging task, partly because of the complexity of the structure and partly because parts of the plant are contaminated. Major issues of the decommissioning task include maintaining a high level of security during the dismantling process and to maintain the project schedules.

The Project proposes to develop a system that integrates an existing CAD environment by application of Virtual Reality technology to be applied as a decommissioning planning and training tool. The general idea is that this tool will make it possible to import or create a VR model of an existing process environment. All pumps, valves, and pipes will be connected to a database with information about each component. Then it will be possible to dismantle the different components in the virtual environment. This will help the planner to decide on safe procedures, minimise radiation exposure of the staff according to the "A.L.A.R.A" (As Low As Reasonable Achievable) principle, minimise radiation, and estimate the duration and cost of the separate tasks, as well as plan the waste recycling procedures.

To efficiently use VR in the planning and execution phases, a detailed 3D model of the plant including all objects is mandatory. Some plants have 3D CAD models that cover large parts of the plant, and in these cases it is possible to convert the CAD data to a format accepted by a VR visualisation tool, which is better suited for dynamic interaction than the CAD tools themselves. Sometimes only a limited amount of data exists, which implies that a 3D model must be created from scratch. 3D laser scanning of the plant simplifies this process and should be considered.

The 3D model can be used for several purposes:

- The order of dismantling can be verified beforehand in the VR model
- Using collision detection, the optimal route for removing large components can be verified
- The operation and position of specialised machinery to aid the dismantling process can be planned
- The VR model can be used to brief the crew which is responsible for the dismantling
    - Walk routes to minimise radiation exposure
    - Dismantling procedures
- Make crew acquainted with inaccessible areas
- The VR model can be dismantled simultaneously with the plant - keeping track of the progress, enabling current status view and progress reports
- Decide where to put radiation protection shields
- A component database can be connected to the VR-model to be used for:
    - Retrieving information about component attributes, e.g. who is responsible for the dismantling, when is it scheduled for dismantling, what is the radiation level, where should it be transported, tools required, etc.
    - Print work orders to the crew
    - Calculate the radiation dose that crew is exposed to
    - Planning use of crew members depending on radiation exposure

It will also be considered to introduce sophisticated radiation dosimeter functionality. If each component is associated with radiation data it can be imported into a 3D-calculation model, which again is connected to the 3D model supplying the topology data of all the components. Using a VR mannequin a simulated walkthrough can be performed in the model to provide time and positional data to be used in the calculation module.

In some cases a detailed 3D model can be useful for training in the different stages of dismantling a component, i.e. find the right tools and train the order of operations.

# 9 SYSTEM SAFETY AND RELIABILITY

## 9.1 Introduction

With the increasing use of computer-based systems for control and supervision of NPPs it is necessary to establish methods to develop reliable systems, and to show their safe application.

The use of formal methods for the development of computer-based systems has been a research area at the Project for many years. The objective has been to investigate the benefits of using such methods to develop safety-critical systems, i.e. small systems with very high reliability requirements. However, with the increasing maturity of formal development methods, which also includes improved availability of support tools, there is a tendency to apply formal methods also to larger and more complex systems. Accordingly, plant control and supervision systems are becoming more relevant as possible target systems for the application of formal methods.

The use of computer-based systems in nuclear power plants makes it necessary to incorporate an evaluation of these into the total safety assessment and in the general acceptance criteria for equipment in NPPs. There is a variety of analysis methods and information sources relevant for such an assessment. There is, however, a need for investigations on how to combine these within a framework of regulation criteria. Regulation criteria based on Probabilistic Safety Assessment (PSA) and Probabilistic Risk Assessment (PRA) have been recommended as a means of evaluating the safety of nuclear facilities. However, the application of software based systems for plant control and supervision introduces new aspects concerning probabilistic methods which are not solved, and which require substantial research activities.

Investigation of formal development methods and incorporation of different methods for safety assessment of programmable plant control and supervision systems will be major research activities at the Project in the period 2000 - 2002.

## 9.2 Formal Development Methods

The planned research activities reflect the importance of formal development methods to software engineering and software quality, but also to other areas where formal description and analysis are essential elements.

- *Formal Specification Methods*

It is generally accepted that the specification phase is the most crucial phase in the formal development of a software system. This is in particular true if the specification is used as a basis for verifiable design steps. Many formal specification methods have now reached a level of maturity where they are used in industrial applications. In order to give guidance in the selection and use of these methods, the Project will on a continuous basis evaluate leading specification languages and tools with respect to their potentials for use in the development of systems of interest to member organisations. The evaluation results are expected to be of great value to companies involved in the development of complex software systems.

- *Formal Specification of Man-Machine Interfaces*

The Project proposes to utilise its experience from formal methods research in developing a formal or semi-formal description language to facilitate the design process of man-machine interfaces (MMI) (see also 7.3). The description language will support a combined approach that incorporates digital systems technology and human factors, and will be directed towards the contents and dynamics of user interfaces. Along with the development of the description language, the research activities will include the development of support tools facilitating the analysis and the verification and validation of various elements of a described MMI.

- *Formal Methods for Procedure Evaluation*

The formal system development methods may also be used in systems analyses in general. Earlier research activities on algebraic specifications and theorem proving have demonstrated that such methods can be utilised in the description and verification of operator procedures, control strategies, etc. The Project proposes to further explore the potentials of formal methods in this connection. As an example, the potentials of using the specification languages SDL (Specification and Description Language) and MSC (Message Sequence Charts), together with the associated case-tool SDT, to describe and validate procedures may be investigated (see also section 8.5). Another possible approach is to use so-called assumption/ commitment techniques to describe the effect of procedures with respect to assumptions about their environments. Since the environment assumptions are stated explicitly in the specifications, such specifications make it easier to decide if one procedure can be safely replaced with another.

- *Formal Methods Development Tools*

The results from the various formal methods research activities are utilised in the development of support tools. Particular emphasis will be given to tools facilitating the integrated use of graphical and textual notations in formal software specification. In particular, a suite of tools will be developed on the basis of the Projects research activities on algebraic specifications and Petri nets.

In order to facilitate more wide-spread use of the HRP Prover developed at the Project, some work will concentrate on support, maintenance, and dissemination activities. These include regular software maintenance of the new HRP Prover with its various extensions and support tools, application in software development projects, and based on these experiences, continuous improvement of the instruction material to the tutorial programme. The Project will be able to provide assistance to member organisations that want to apply this material and the new HRP Prover in relevant projects.

- *Formal Methods and Software Quality*

In the previous three years period, a long term research activity was initiated in the area of formal methods and software quality. A particular concern is to reach a consensus between regulators, licensees and the nuclear industry on questions related to the effective, industrial use of formal methods. This research activity will continue, with an emphasis on four main problem areas:

- System aspects, i.e. the relevance of formal methods in achieving a rigorous treatment of system aspects in the design and implementation of digital I&C systems.

- Technology transfer, i.e. the integration and application of experiences, best practice, and new formal methods technology to nuclear power plants.
- Integrated quality assurance, i.e. the contribution of formal methods in integrating process-oriented and product-oriented software quality assurance.
- The effectiveness of formal methods, i.e. effectiveness compared to conventional approaches to software development.

Particular emphasis will be given to the establishment of evaluation criteria, metrics, and guidelines. In order to reflect both short-term and long-terms needs of the nuclear power community, as well as contributing to reaching a consensus in the problem areas addressed, the research activities will benefit from close co-operation with member organisations of the Project.

### 9.3 Verification Methods

No method can guarantee correctness with 100% confidence. One can actually only show the presence of faults, it cannot prove the complete absence of faults. However, the more one search for faults, the higher is the probability to find all residual faults. The objective of the verification methods is to effectively reveal (and remove) potential faults, to obtain a high confidence in fault freeness. There are several different types of methods which are complementary to each other, and the use of a combination of these methods will increase the confidence in the target system. These can be divided into some main classes: static program analysis, formal program verification, and execution testing.

- *Program Analysis*

Program analysis is defined as the process of evaluating a computer program without executing it. Program analysis can be performed at various stages in the development process, and can therefore be applied during inspections and walkthroughs. It can, however, also be applied to the final program system. Program analysis can be used to reveal structural weaknesses in the program and violations of rules for good programming. A further use of program analysis is to compute software measurements which can be very useful to assess the complexity of the program, the vulnerability to errors, and the amount of work needed to perform the safety assessment.

With the increased acceptance of object-oriented programming languages, especially in graphical programming and the design of graphical user interfaces, the need for models, methods and metrics for object-oriented programming arises. The object-oriented paradigm is very different from the sequential, and introduces new elements, such as inheritance, encapsulation and polymorphism, which again introduce potentially new safety risks.

The proposal for the next three years period is to develop program analysis tools for quality assessment of programs developed using object-oriented techniques.

- *Formal Verification*

Formal verification means to show, by formal (i.e. mathematical/logical) methods and corresponding tools (as e.g. the HRP prover), that certain properties of a program fulfil their requirement specifications. The plan is to investigate the possibility to apply Model Checking for verification purposes. This is a technique that relies on building a finite model of a system and checking that a desired property holds in that model. Model checking is typically applicable to systems whose intricacy resides more in the control than in data, i.e. systems whose role is described by their possible interaction sequences with their environment. Model checking has been conceived as an automatic machine-implemented approach. It requires no intervention of the software developer and provides explicit information about what could be wrong when it fails to check a property.

One aspect which will be investigated is the possibility to use model checking in different application areas in the control and supervision of NPPs. A requirement for this is that the application is modelled in a formal language. This activity is therefore closely related to the work planned on formal specifications. Another interesting application area is related to the activities on functional modelling and computerised procedures (see section 8.5).

- *Testing*

To test a program is to execute it with selected test data to demonstrate that it performs its task correctly. Testing is an essential part in the assessment of a software product, and complementary to other V&V activities. Ideally, the test data should be selected so that all potentially residual faults should be revealed. Exhaustive testing is in general not possible. The optimal test strategy is the one which maximises the probability to reveal all possible residual program faults. A proposed task is to investigate the possibility to use formal specifications to derive such test strategies, and to measure their cost-effectiveness.

### 9.4 Assessment of Pre-Existing Software (COTS and Legacy software)

In practice, programmable systems for control and supervision are to a large degree based on pre-existing software. One can here distinguish between two types of pre-existing software. One is commercial-of-the-shelf software (COTS), which is software systems which can be purchased "off the shelf" as commercially available products made as standard components for use in a broad application area, and not for a particular safety relevant purpose. Examples of such use are firmware in standard instruments, embedded software in PLCs, modules in software libraries, configurable software systems, etc.

The other type is Legacy software, i.e. software which is inherited from previous use, and which form a basis for an extended or modified system. A typical example is when one uses major parts of the software in an existing system in the development of a control and supervision system of a substantially modified or a new plant.

Even if pre-existing software in principle shall fulfil the same requirements as tailor made software, it has some particular characteristics which necessitate special attention. Some are common to COTS and Legacy software, others are particular to each of them. The proposed task for the three years period is to investigate how these characteristics should be taken into account in the dependability assessment of such systems.

A particular characteristic of pre-existing software is that there exists a certain amount of experience data with respect to this software, and a topic of research is how this experience can be utilised in the dependability assessment. Another characteristic is that pre-existing software may make some assumptions about the environment in which it is used which are not always understood, and which may have serious consequences in new applications (as e.g. the explosion of the ARIANE 5 space rocket). The assessment of pre-existing software should take into account this possibility, and the research task will be to establish a method to reveal such assumptions and thus prevent this type of failure to occur.

In principle, the COTS should have been developed and maintained according to an accepted standard for software engineering and quality assurance (e.g. IEC-880 or IEC-61508) if it shall be used in safety related applications. However, COTS is often delivered without appropriate information about conformance to these standards, and it may be necessary to show this conformance through a retrospective analysis. This problem will also be addressed.

### 9.5 Risk Based Program Safety Assessment

For final acceptance of a safety relevant system, a thorough safety assessment is necessary. An acceptance process is based on a set of regulation principles, which may be based on analysis, probabilistic methods, or engineering judgement. Analysis methods are based on written documentation such as design documents and code listings, and the analyses can be made with or without support tools. Probabilistic methods are based on collection of data, in particular failure data, and probability theory. Judgement methods are systematic ways of evaluating aspects of qualitative nature of relevance to safety, as e.g. quality of production methods, comprehensibility of man-machine interfaces etc. All these methods will be targets of research activities at the Project during the upcoming period and more details are outlined in the following.

- *Specification Based Risk Analysis*

Conventional risk analysis methods have mainly been developed for hardware based systems, and are not directly applicable when programmable digital systems are integrated into plant control and supervision. There is therefore a need to modify these methods to also comprise analysis of integrated digital systems. A main objective of such analysis is to identify possible risks caused by deviation from the intended behaviour of a system. To obtain this objective it is necessary to have a detailed and unambiguous description of this behaviour. The formal specification languages discussed above are good candidates for such descriptions. The proposed research activity in this area is therefore to investigate how formal descriptions of computer based systems and their interactions with plant and operators can be utilised in conventional risk analysis methods.

- *Probabilistic Methods*

Probabilistic assessment based on statistics is best suited to measure properties of mass-produced components, of parameters where one has large statistical material, or with results from controlled experiments. This interpretation can be applied on the hardware components of a system, and basic rules for probability computation can be used to compute the prcbability of a hazard on the system as a whole. There exists reliability models for software based on operation and testing data. However, a problem with these measures is that they do not take into account that there are several factors which are important to software reliability, even if they cannot be put directly into a reliability formula. Some of these are of qualitative nature, like the producer's reputation, the development quality etc. Others are measurable, but not directly connected to reliability estimation, like program size, program complexity etc. So the connection between these quantities and software reliability is also of qualitative nature.

A more qualitative type of reliability measure is expressed as a subjective judgement, as a "belief" in fault freeness. A methodology which is studied is the application of Bayesian Belief Networks (BBN) to combine evidences from different information sources for a quantitative assessment of this belief. The objective of using BBNs in software safety assessment is to show the link between basic information and the confidence one can have in a system. This measure of confidence is of probabilistic nature, and an objective of the research programme is to study how this measure can be incorporated in PSA/PRA. Investigations of this methodology will continue in the next period, emphasising development of a practical method for its application to real systems. Application on realistic examples will provide a basis for formulation of guidelines for practical use.

A particular activity will be to study how recent achievements in qualitative reasoning, simulation, and analysis can be combined with algebraic, symbolic, and numeric techniques with the aim of producing the necessary quantitative data. The investigations will also include the applicability of continuous, hybrid, and stochastic Petri nets, in particular in combination with the other techniques. The research activity aims at evaluating how well the different techniques contribute to facilitating PSA/PRA, and providing guidelines based on these evaluations. An important consideration in this respect is the applicability of the methods to the assessment of embedded proprietary software.

## 10. PROGRAMME BASIS, FUEL AND MATERIALS

The execution of the proposed fuel and materials testing programme is based on expertise and facilities established at the Halden Reactor on the background of more than thirty years of operation and experience. This comprises all steps from design and fabrication of test rigs to irradiation, data acquisition and evaluation, ensuring an expedient and flexible execution of plans with in-house control of quality and timing. The major factors and requirements in this process are highlighted in the following sections. Additional details can be found in Annex I and Annex II.

### 10.1 Design and Fabrication Capabilities

The design of an irradiation rig is one of the most crucial stages of an experiment since it almost irrevocably determines the capabilities of a test and most of the conditions under which it will be executed. The Project has therefore in recent years emphasised the utilisation of modern tools for design and fabrication.

Computer aided design (AutoCad) is employed for fast and versatile production of drawings and parts lists. This allows re-use and easy modification of existing layouts as well as the application of basic constructs. Computer programmes are also applied for the evaluation of hydraulic and cooling conditions, and for structural analysis.

The fabrication workshop contains all necessary tools and machinery for precision machining and welding. Numerically controlled turning lathes ensure the exact and repeatable manufacturing of parts with small tolerances.

Refabrication and re-instrumentation equipment have been developed and constructed in a joint effort between Halden and the Kjeller laboratory. This technique has been proven and applied on high burn-up rods and will be suitable for assessing the performance of LWR fuel segments. Re-instrumentation can be performed for fuel centre thermocouples, pressure transducers and cladding elongation detectors.

Two fuel inspection compartments are used for interim inspections and non-destructive measurements. One is located within the containment and near the reactor. This is also used for fuel unloading/reloading from test rigs and standard assemblies. The other compartment was built in 1995 - 96 and is used primarily for materials inspections and special operations on active materials (e.g., instrument attachment to irradiated specimens).

The design and fabrication work is carried out under strict quality control routines, which involve function tests and calibration of individual components/instruments as well as testing of the complete test rigs in a loop, operated at full temperature and pressure conditions, prior to loading in the reactor. An efficient formal documentation system for quality control purposes has been implemented.

### 10.2 In-Core Instrumentation

The in-core instrumentation in use at the Halden Reactor is subject to a continuous upgrading in response to the changing and ever more exacting needs for the experimental programmes. New types of sensors have therefore been developed in recent years, and commercially available sensors have been evaluated and applied to meet requirements, especially in the areas of materials testing and high burn-up investigations.

The conversion of HBWR instrumentation technology for use in other reactor systems is pursued mainly through co-operative arrangements with participating organisations. Deliveries of tailor-made products for special use have included coolant instruments (turbines, void gauges) and rod instruments (pressure gauges) for test reactor application, fuel pond inspection instruments like diameter gauges, flux mapping instruments ($\gamma$-thermometers, cobalt detectors), assembly length monitors for power reactor use, and instruments (pressure sensors, fuel and rod length monitors) for fuel rod segments in power reactors.

- *Fuel Testing Instrumentation*

Rod power is an essential parameter in relation to many aspects of fuel performance. A considerable part of the instrumentation is therefore directed towards maintaining an adequate power monitoring. High quality *turbine flow meters and coolant thermocouples,* together with advanced experimental techniques, make it possible to determine the channel power with an accuracy of ± 3% both in the HBWR core and in the LWR simulation loops. Careful selection and proper placement of *self-powered neutron detectors,* together with the channel power calibration and physics calculations, allow the continuous evaluation of power and burn-up for a test assembly as a whole, for each individual rod, and at different axial nodes. Techniques like noise analysis for flow velocity determination and advanced core physics codes are applied to further improve the quality of Halden experimental data, especially at high burn-up and after long in-reactor residence times.

Fuel parameters are measured with a variety of proven instruments including high temperature *thermocouples* for fuel temperature measurements, fuel *stack elongation detectors* to evaluate densification and swelling behaviour, and *bellows pressure transducers* for continuous monitoring of fission gas release. Recent developments encompass *expansion thermometers* for reliable fuel temperature measurements with high data quality at extended burn-up as well as *$\gamma$-thermometers* for power monitoring.

Cladding parameters and overall rod behaviour can be assessed with the help of *cladding elongation detectors* for PCMI assessment, *cladding thermocouples,* and *diameter gauge* systems to measure cladding dimensional changes during in-pile operation with an accuracy of about ±2 μm.

- *Materials Testing Instrumentation*

Water chemistry parameters are currently measured out-of-vessel and in cold conditions by means of sampling lines connecting the primary circuit to the measuring equipment. This, however, does not give a correct picture of the electro-chemical conditions prevailing in the core. For corrosion studies of in-core materials, it has become increasingly important to understand how the corrosion potential varies in-core as function of varied feedwater chemistry parameters. The Project has addressed this item primarily in terms of developing *electro-chemical potential sensors* as well as validating and applying sensors produced by participants. It is foreseen that demonstration installations will be realised at Halden as well as at commercial plants.

Wedge loaded Double Cantilever Beam specimens (DCB) with on-line crack measurements based on the potential drop method have been used with good results. A modified design with active bellows constant force loading is available for future IASCC studies aiming at investigating crack growth as a function of stress intensity. Similarly, the use of Compact Tension (CT) specimens has been demonstrated in-reactor. Another specimen type is the pressurised tube variant for constant stress IASCC tests. Pressures of several hundred bars can be obtained in small sealed tubes made of the materials to be investigated.

Cladding elongation detectors can be used for an assessment of zirconium oxide conductivity under reactor operating conditions. The sensor will detect the additional thermal expansion of the cladding due to the increased temperature which is caused by the resistance of the oxide layer to the heat flux. The comparison with reference material will allow the deduction of the conductivity.

### 10.3 Irradiation Facilities

The Project can make available a wide range of versatile test rig designs to be irradiated either under HBWR conditions or in light water loop systems with prototypic LWR temperature and pressure conditions. The HBWR core provides the thermal flux levels required for obtaining adequate power ratings with prototypic LWR enrichment fuel.

The experimental activities are supported by a highly experienced reactor operation and maintenance staff, as well as by experimentalists and data handling personnel. A chemical engineering group supervises the monitoring and control of the water chemistry related testing activities in the reactor and in the light water loops.

- *The Halden Reactor*

Several features make the Halden Boiling Heavy Water Reactor (HBWR) suited for studies of reactor fuels and materials. Being a heavy water moderated and cooled reactor, it has a considerably wider core than a light water reactor of the same power. This gives better room for instruments and experimental equipment.

The top lid of the reactor is designed with a penetration for each test rig, thus making handling and loading relatively simple. Above all, it allows making the instrument cable seals an integral part of an irradiation rig.

The reactor is operated with a core of about 110 fuel assemblies of which up to 40 are instrumented test rigs. In addition, loop systems will be utilised for tests requiring water chemistry and thermal/hydraulic conditions representative of modern light water power reactors. The neutron and gamma flux conditions in the various test positions in the core can be varied according to experimental needs.

- *Test Rigs*

Instrumented test *carriers* are designed both for fuel rods which shall not be replaced as well as for replaceable fuel rods. In the former case the design is more flexible and instrumentation of high complexity and density can be accommodated, such that a large variety of experimental objectives are met. An example of this design is the Project's gas flow and fission product release assembly, where highly instrumented fuel rods are connected to an out-of-reactor gas supply, and the gas composition in each rod can be individually controlled by using in-core valves for either gas flow/gas exchange experiments or fission gas release experiments.

A number of test rigs are designed such that fuel rods can be removed and replaced. The advantage of this concept is that fuel rods can successively be exposed to different operating and measuring sequences in any one of a number of specialised rigs. *Base irradiation rigs* allow efficient accumulation of burn-up with varying degree of instrumentation, whereas *ramp rigs* allow up to four rods to be ramped through use of a helium-3 system. The He-3 system is also applied for load follow and frequency control type tests. Rigs with *movable absorber shields* are suitable for a variety of power cycling experiments, and *diameter rigs* allow diameter measurements on up to three fuel rods simultaneously. In the event experimental requirements demand it, the special features described above can be combined.

The complex mechanisms of the rigs are operated by *external control systems*. The reactor's heavy water is used as hydraulic fluid for the *fuel rod drive system*, the *absorber shield system* and diameter *gauge drive systems*. The *helium-3 system* controls the pressure in the absorber coils, and the *fuel rod gas flow system* permits flushing of helium and other gases through fuel rods, exchange of different gases, analysis of fission products, and operation of in-core valves.

- *High Pressure Light Water Loops*

Fuel testing under pressure and temperature conditions typical of pressurised and boiling water reactors has been performed in the HBWR for more than ten years.

Each loop system can serve one or more test sections and have external systems for controlling the coolant thermal hydraulic and chemistry conditions. For example, waterside corrosion investigations can be performed under typical PWR and BWR water chemistry conditions, utilising segments pre-irradiated in power plants. The loop systems are also used for material testing (IASCC investigations).

Within the dimensional constraints of the pressure flask, the test rigs in use for loop testing can incorporate all the essential rod and rig instruments in use under HBWR conditions as described above.

### 10.4 Inspection and PIE Facilities

The irradiation programmes are complemented by two inspection compartments at Halden and a variety of PIE possibilities at the associated hot cell of Institutt for energiteknikk at Kjeller.

- *Inspection Compartment*

A number of measurements complementary to in-core results can be performed in the inspection compartments. These include diameter profilometry and length change determination based on the precise measurement of the distance between v-groo  machined into a rod. Eddy current measurements are being used to detect cladding cracks, and EC proximity probes are applied for oxide thickness determination. Further, it is possible to pick up the positional change (relative to a pre-irradiation calibration) of a core inside the rod using an external LVDT. With this technique, fission gas release and fuel stack length changes can be determined non-destructively even for rods which could not be irradiated in a test rig with an LVDT. Burn-up and power profiles can be determined with a γ-scanning facility. Weight gain of metal coupons for corrosion tests, non-destructive and destructive operations on metal specimens as well as special instrument attachments on irradiated materials are carried out in the compartments.

Photo and video equipment is available for close-up inspection of rod surfaces, documentation of changes such as crud build-up, corrosion and nodule formation. The information can be stored in analogue and digital form for further evaluation and processing.

- *Post Irradiation Examination*

Comprehensive PIE capabilities are provided by the hot cells of Institutt for energiteknikk (IFE). They comprise non-destructive (profilometry, eddy current testing, gamma scanning, neutron radiography) and a variety of destructive examinations (puncturing and fission gas analysis, metallography/ceramography, autoradiography, burn-up analysis, transversal γ-scanning). Emphasis is put on micro analyses in conjunction with the determination of parameters associated with extended and high burn-up such as retained fission gas, burn-up analysis (both from micro samples) and SEM examination.

## 10.5 Data Acquisition and Analysis

In parallel with the Project's experimental efforts, work is conducted in the areas of data qualification, data analysis, model development and verification, and knowledge transfer to Project participants. The knowledge transfer between the Project and its participants is in particular dependent upon a proper understanding of the data and upon having available effective means of transferring that knowledge. Computerised information packages allow an improved and fast overview of available experiments suitable for particular purposes. The generation of comprehensive, qualified data packages for modelling purposes will continue to be important also in the new programme period. Further, the Project is exploring the feasibility of transferring the inventory of declassified reports on computer files.

The *data acquisition* and *reactor supervision* system is running on a G2/RTP process control computer. More than one thousand signals from the instrumented test rigs and from the plant are logged at 0.5 sec intervals. As a routine, raw data are saved permanently every minute for future use, and converted data are entered into the Test-Fuel-Data-Bank (TFDB) at least every 15 minutes. Logging faster than 0.5 sec is covered by a special program for sampling intervals from 10 milliseconds and upwards. Typical applications are diameter trace recording and sampling for noise analysis.

Efficient data treatment and reduction are indispensable requirements to cope with the ever-growing data base of the Halden Project and its participants. Long-ranging experiments, comprising several hundred thousand measurement records, are not uncommon and increase in number when e.g., high burn-up tests are performed. The TFDB, in particular with its graphics display module VISION, provides versatile functions for data screening and manipulation during various stages of data retrieval and display. Further improvements of these capabilities are suggested by the everyday use of this tool and are gradually introduced. The TFDB thus evolves into a high-level data processing language with data analysis functions developed on the basis of the many years of experience of the Halden Project, and designed to maximise the information which can be gained from the experiments.

Widely accepted standards have emerged during the past years which facilitate data treatment and information exchange. The Project intends to keep track of such developments and to introduce hardware and software in order to follow generally accepted trends.

### 10.6 Experimental Control Room Systems

The Project will continue the upgrading of the environment used by experimentalists and reactor operators. Hardware and software are being installed with the aim of providing intimate supervision and control of experiments and of the reactor as a whole. This secures the successful realisation of demanding experiments and helps to consolidate otherwise conflicting requirements of reactor operation, thus making the HBWR an attractive tool also in the future.

Software systems developed at the Halden Project for plant monitoring and control can advantageously be applied to many tasks emerging in conjunction with experiment supervision and execution. Status and alarm displays of the reactor as a whole, the experimental loop systems and the individual experiments have been defined using PICASSO together with the real time data conversion system of the TFDB. At the same time, the real-life application of a system like PICASSO close to its developers constitutes an ideal opportunity for testing and further improvements. The daily usage by experimentalists and reactor operators provides valuable feedback and spawns the implementation of new features and improvements.

All these efforts as well as improvements of the Test-Fuel-Data-Bank system aim at creating an integrated environment for data handling, process information and control, maximising the obtainable knowledge and enhancing the process safety. The benefits of this approach are manifold: reliability of software and hardware systems are demonstrated in real-life testing, undelayed and direct inclusion of daily operating experience and requirements can lead to further developments. Further, improved support of reactor operation and experiments, data handling and evaluation is ensured with unified access to both real-time and off-line data.

# 11 PROGRAMME BASIS, MAN-MACHINE

The proposed research programme is to a significant degree based on the experience and know-how gained through the work performed during previous programme periods on applying computers for plants surveillance and control. An active co-operation with member organisations has been established over the years. This interaction represents an important factor for the future research and development work by keeping the programme focused on user needs and by co-ordinating the efforts at the Project and in the member organisations.

The Project has been engaged in establishing a consistent basis for the design, development, and analysis of *man-machine applications* in process control domain. This requires the combination of a multitude of technical and behavioural expertise with a thorough process knowledge and appreciation of practical problem solving. In addition, the necessary resources for development and experimentation with new technology and systems must be available.

Through the extension of the HAlden Man-Machine LABoratory (HAMMLAB), now including three full-scope nuclear simulators coupled to an advanced experimental control room facility, and the associated research methodology and staff, a unique basis for development and testing of operator support and plant performance monitoring systems and for general studies of operator behaviour exists at the Project.

A list of the human factors related experiments performed in HAMMLAB since the establishment is summarised in Table 11.1. A list of Applications and system toolboxes developed by the Project during the last 10 - 12 years is given in Table 11.2.

## 11.1 Collaboration with Project Participants

As important and unique as the laboratory resources of the Project itself, is the international character of the Project, permitting information exchange as well as close working contacts with utilities, reactor and system vendors and research organisations in the member countries. In a number of fields, co-operation has been established not only with national research laboratories, but to an increasing degree also with the industry and the regulatory bodies.

Most member organisations are engaged in the field of *computer-based operator support systems*, including both utilities, vendors, research organisations and licensing bodies. In many countries modern digital technology is entering the NPP control rooms through backfitting programmes, for example several on-going or planned control room upgrades in Sweden.

Work on specific operator support systems for disturbance handling, diagnosis, procedure execution, accident management, etc. is pursued by many member organisations and in working groups in international organisations like OECD/NEA and IAEA, and the activities at Halden are co-ordinated with and complement this work.

The Project has over the years been engaged in work on alarm handling, disturbance analysis and plant diagnostics. The activities in these areas have been co-ordinated with on-going activities in member organisations. The work on alarm handling and disturbance analysis has been performed in co-operation with both research institutions (GRS (Germany) and EPRI (USA)) and utility organisations (IVO and TVO (Finland) and the former CEGB (UK)). Development and evaluation of improved alarm systems for nuclear power plants have been co-ordinated with similar efforts at Spanish member organisations Tecnatom, (CIEMAT). The Danish company FLS Automation utilised the principles of the HALO alarm handling system in its development of new supervision systems and the Swedish mining company LKAB has recently applied the alarm handling principles developed in the COAST / CASH projects. KEMA, the Netherlands is using COAST/CASH based principles in development of the alarm system for their NPP simulator. The Korean organisations KAERI and KEPRI utilise the COAST system in their preparatory work for the new generation control room in the KNGR project.

The model-based early fault detection system developed at the Project has been tested in a pilot installation at the Loviisa nuclear power plant in Finland in co-operation with the Finnish utility IVO. The system has been successful in detecting internal leakage's in preheaters in the feedwater circuit. A system for signal validation of flow sensors using the same basic methodology is tested out in another pilot system at Loviisa. Together with General Electric Co. (USA) the possible use of the early fault detection methodology within the control system of the new reactor design PRISM has also been explored. The same principles have also recently been applied in a co-operative exchange programme at the FUGEN reactor in Japan.

During the current period evaluation of the signal validation toolbox, PEANO, based on neural networks/fuzzy logic techniques has been carried out in co-operation with EDF and CEA, France which have provided simulated data from French PWRs for testing the PEANO tool. In co-operation with Tecnatom, Spain PEANO has been tested on-line in their BWR simulator in Madrid.

The use of expert system techniques for fault diagnosis has also been evaluated. The expert system DISKET, originally developed at JAERI (Japan) was implemented at the Project and tested against disturbance transients generated on the NORS simulator.

In a co-operative project between the Swedish Nuclear Inspectorate (SKI), the Swedish utility Forsmarkverket and the Project a post trip analysis and guidance system (SAS-2) for the Forsmark BWR, unit 2 has been developed and tested.

A close co-operation with the Swedish State Power Board has been established in core surveillance. The predictive part of the core surveillance system, SCORPIO, developed at the Project has been installed at the Ringhals nuclear power plant to serve the three PWR units at the site, and the core follow part of SCORPIO has been installed at one unit, Ringhals 2.

The SCORPIO system has also been installed at the British PWR Sizewell B, at all seven PWR's of the American Duke Power utility, and demonstration and training systems have been installed in Belgium (Tihange NPP, AVN Vincotte), and the Czech republic (NRI, Rez). Through combined use of measurements, mathematical models and powerful MMI SCORPIO has proved to be an efficient and valuable tool for both the operators and the reactor physicists. In a recent OECD/NEA project sponsored by STA and JAERI, Japan, and executed by NRI, Skoda and Chemcomex, Czech

Republic and Institutt for energiteknikk (IFE), a SCORPIO system for VVER reactors has successfully been implemented at the Dukovany NPP in the Czech Republic.

The Project on Computerised Procedure MAnual (COPMA) was requested by several member organisations, among which NRC (USA), former CEGB (UK), and ENEA (Italy) have been the most active. The first version of COPMA was installed at ENEA's PWR engineering simulator, and at GRS in Germany. The second version of COPMA, which has been implemented on modern work stations, has in co-operation with NRC (USA) been delivered to North Carolina State University where the system has been evaluated with positive results. NRI, Rez interfaced COPMA and Picasso-2 with their diagnostic expert system RECON. COPMA was also evaluated at the NPP Temelin in the Czech republic.

COPMA has continuously been extended with new features and organisations in UK (Scottish Nuclear), Czech republic (NRI, Rez), Spain (CSN, Tecnatom), USA (GE and ABB-CE) are evaluating COPMA for use in their organisations. The Korean organisations KAERI and KEPRI utilise COPMA in their computerised procedure work in the KNGR project.

Currently, the latest version, COPMA-III, is utilised in an OECD/NEA project for installation of a Plant Safety Monitoring and Assessment System (PLASMA) at the Paks NPP in Hungary. In this project, which is sponsored by STA and JAERI, Japan and executed by IFE and KFKI Atomic Energy Research Institute, Hungary, COPMA is supporting the use of the EOPs at the plant.

The Nordic Council sponsors a co-operative programme on nuclear safety. Through the involvement of Institutt for energiteknikk (IFE) in projects within this programme, the Halden Project has access to the results from these projects. In previous programme periods a project was carried out investigating how advanced information systems can be used to provide a more reliable and consistent data base for decision makers in emergency situations. Experiments with this prototype system were carried out with staff from one of the Swedish county emergency centres in 1989. In the previous and current programme period, IFE and the Halden Project are taking part in a Nordic Council sponsored project investigating the feasibility of a Computerised Accident Management Support (CAMS) system for the control room and technical support centre of a nuclear power plant.

The Swedish licensing authority, SKI, has evaluated the CAMS system during a national emergency drill. Co-operative efforts with national programmes on accident and emergency management have recently been established with Spanish and Italian organisations.

The Picasso-2 graphical display system has been delivered to ENEA (Italy), National Power and Nuclear Electric (UK), Siemens/KWU research laboratory (Germany), Toshiba Nuclear Laboratory (Japan), Kernkraftwerk Leibstadt (Switzerland), and the Technical University and University of Copenhagen (Denmark) for use in man-machine interface design and prototyping.

Presently the Picasso-3 graphical display system has taken over for Picasso-2 and it has been taken into use in most member countries. In particular, Picasso-3 has been applied in a number of industrial projects in Norway, and several member organisations in the Czech Republic, Finland, Germany, Italy, Japan, the Netherlands, Sweden, Switzerland, Slovak Republic, Spain, USA, and UK are applying the system for supervisory control tasks, simulator developments and research projects on man-machine interfaces. NRC (USA) use Picasso-3 at their Chattanooga training centre for the operator communication systems for training simulators, and ABB-CE (USA) is using Picasso-3 in their control room concept for their NPP deliveries in the Republic of Korea.

There is a general interest among member organisations in the design and evaluation of advanced human-machine systems for operator support. Vendors and utility organisations are involved in the design of systems ranging from stand-alone computerised operator support systems to entire advanced control rooms. At the same time, regulatory agencies are faced with the task of licensing this spectrum of advanced systems now or in the near future. Thus, the results of Halden *human-machine interaction research* address the needs of member organisations in a number of ongoing research programmes aimed at providing technical bases for regulation of current control rooms as well as advanced reactor designs.

The experimental programme in HAMMLAB depends on the availability of a qualified group of test subjects. In experiments where detailed process knowledge and experience is required, use is made of operators from the Loviisa plant in Finland.

The work at the Project on experimental validation of computerised operator support systems has over the years been performed in close co-operation with member organisations. In earlier programme periods the Critical Function Monitoring System (CFMS) validation project was carried out together with C-E (USA) and VTT and IVO (Finland). This project was followed by the later development and validation of C-E's Success Path Monitoring System (SPMS) in HAMMLAB. Other operator support systems which have been evaluated in HAMMLAB is the Integrated Process Status Overview (IPSO) in co-operation with C-E and the diagnostic system DISKET from JAERI (Japan).

An assessment of how the CFMS and HALO-alarm handling system developed at the Project would function under a number of disturbance scenarios for a German PWR plant was performed in co-operation with GRS (Germany).

In co-operation with the former CEGB (UK) an investigation of the effects of mixed-instrumentation systems (conventional instrumentation in combination with computer-driven colour screens) on operator performance was carried out in HAMMLAB.

In the 1991-93 programme period the Project carried out an extensive experimental evaluation of the SAS-2 post-trip analysis and guidance system at the Forsmark simulator in co-operation with SKI and Forsmarkverket (Sweden) where shift supervisors from the Forsmark NPP used the system during simulated accident scenarios.

In the 1985-87 period the Project, together with ENEA (Italy), performed a review of the process information system for the control room of a nuclear power plant. The work consisted in establishing design criteria for the lay-out and information content of the process-display formats consistent with the total information array in a hybrid (conventional panels and CRTs) control room. In bilateral projects in Norway the Project's knowledge and experience in design of process-display formats and control room lay-out have been utilised in specification of control rooms for chemical factories and electric power dispatch centres. Similar reviews of control rooms for off-shore oil production platforms and diving operations in the North Sea have been carried out.

In the present period the Project has been heavily involved in the Swedish control room modernisation programme. An approach taken in Sweden on control room upgrading is to consider a complete replacement of the existing control room with a solution utilising modern technology in a most efficient manner. This has opened for practical use of the control room development programme performed within the Halden Project. The Project staff has contributed in several areas such as establishing a control room philosophy, control room design, task analysis and function allocation and advanced alarm system design. The most recent technology applied is Virtual Reality which has proved very useful in the control room design work.

Together with NRC (USA) the Project has initiated collection of data on human error probabilities from the experiments in HAMMLAB for inclusion in the Nuclear Computerised Library for Assessing Reactor Reliability (NUCLARR) data base. In the 1991-93 period data from two evaluation experiments on computerised operator support systems have been contributed to NUCLARR. Analysis of human errors performed in the Norwegian off-shore oil industry has been made.

Human factors staff have been involved in work on evaluating an expert system for BNFL (UK), investigating staffing issues for advanced control rooms for USNRC, task analysis and control room design work for air-traffic control centres and process industries in Norway, and have contributed to increased knowledge on task allocation issues in connection with control room operation. In co-operation with USNRC and Brookhaven National Laboratories the staff carried out a large study on computerised alarm systems in HAMMLAB.

In 1998 the Halden project carried out a human factors experiment for the *Institut de Protection et de Sûreté Nucléaire (IPSN),* France. The purpose of the experiment was to study how two types of automation influenced operator performance in different operating situations. The experiment was carried out in HAMMLAB and involved six crews of licensed nuclear power plant operators from the Loviisa plant in Finland. The experiment contributed to increased knowledge on operator-automation interaction. This knowledge will be used by IPSN to initiate the development of a knowledge base, which eventually shall support evaluation of changes in task allocation between operators and automatic systems in French nuclear power plants.

Activities on investigation and development of methods and tools for development of *safe and reliable programmable systems* are ongoing in all member countries. Work in this area is often conducted in collaborative international projects as e.g. ESPRIT or EUREKA projects, and the Project has co-operated with several such projects. One aim of these activities is to establish an international consensus on requirements for approval of programmable systems for safety critical applications, and development of international standards and guidelines for dependable software. This is made in national licensing authorities as well as international organisations like NEA, IEC, IAEA and EWICS, where the Project is actively working together with other member organisations.

The Halden Project's work in this area has been performed in close co-operation with member organisations like TüV-Nord and GRS/ISTec in Germany, Nuclear Electric, AEA-Technology and Adelard in the UK, SKI and ABB-Atom in Sweden, VTT in Finland, ENEA in Italy, and AVN in Belgium.

Examples of work in previous periods are, in the field of software reliability, the SOSAT (SOftware SAfety Tools) project, which was carried out in close co-operation between TüV-Nord, GRS, and University of Braunschweig (Germany) and the Halden Project. This work aimed at developing tools which can assist the licensing authorities in the analysis of safety relevant software. Another activity at Halden addressing verification and validation of safety critical software was the SAP (Safety Assessment of Programs) project which was performed in co-operation with Nuclear Electric, National Power and SRD in UK. The SAP-project was a continuation of the PODS (Project on Diverse Software) and STEM (Software Testing and Evaluation Methodologies) projects which were carried out in co-operation with the same organisations plus VTT (Finland). Currently, a methodology using algebraic specification for formal specification of safety critical software has been investigated in a real-life application from the Barsebäck plant in Sweden period with positive results. In co-operation with VTT (Finland) and ENEA (Italy) the Project has also been exploring the application of algebraic specification and qualitative physics as a basis for verification of the specifications of a safety critical system.

In the present period there has been a joint project with ISTec to develop a tool to check code against various standards for safe coding. A project on reliability assessment of programmable protection systems is conducted in co-operation with ABB-Atom and VTT. Together with ENEA there has been a joint project on combining Petri-nets and algebraic specification.

Large programmes on *advanced control rooms* are underway in several countries. In the US, the topics of automation and redesign of the man-machine interface are included in projects on advanced concepts both on the BWR and PWR side. (ABWR, Nuplex 80+). In Germany, the PRISCA information presentation system is in use. All Japanese vendors have large programmes, an example being the Toshiba projects on advanced control rooms, APODIA being realised today, IPODIA being developed for a future generation of reactors. In France, the N4 advanced control room is operational at the 1500 MW PWR at Chooz B. In Korea work is going on to design and develop the advanced control room for their next generation PWR, the KNGR.

The activities in Halden are conducted in close contact with member organisations in order to ensure that the Halden efforts are properly focused, and that the results can be effectively transferred.

The implementation of the NORS simulator in the Project's experimental control room facility was made in close co-operation with the simulator manufacturer Nokia of Finland and the Finnish utility company IVO. During modifications and extensions of the simulator both IVO and VTT have been co-operating with the Project. The development and installation of the CP-0 simulator is made in close collaboration with EDF and the simulator manufacturer Thomson Training & Simulation in France during the present period. The development of the F-3 simulator is made in close collaboration with the Swedish and Finnish BWR utilities and VTT in Finland, also during this period.

In co-operation with KEMA (The Netherlands) the Project took part in the development and implementation of the operator communication system of the training simulator for the Doodeward nuclear power plant. An upgrade of this simulator using the Picasso-3 system was also carried out together with KEMA.

Institutt for Energiteknikk (IFE) has, in co-operation with the Norwegian simulator vendor Norcontrol Simulation, delivered full-scale simulators for training of process operators at oil/gas production platforms in the North Sea both for the Norwegian and British sector. Through the participation of IFE in these simulator projects a mutual beneficial exchange of know-how on simulators and computer-based control room systems between the Project and the off-shore industries is taking place.

IFE also co-operates with the Norwegian electric power industry in development of decision support systems for the control rooms of the operations centres.

## 11.2 Technical Tools

The *Halden Reactor* and the extensive experimental programme carried out at this facility have from the start of the process control development efforts at the Project, provided an important practical engineering basis for this work. Through implementation and testing of computer-based data collection and analysis systems for the test fuel programme, the Project has acquired valuable experience with respect to installation and operation of plant computer systems. The experienced operations staff of the Halden Reactor will be a major asset also in carrying out the 2000 - 20002 programme on man-machine systems research through participation in the testing and evaluation of the prototypes of operator support systems and advanced control rooms.

The other major technical basis is the *HAMMLAB experimental control room facility*. The full-scope nuclear power plant simulators coupled with the HAlden Man-Machine LABoratory (HAMMLAB), which includes the experimental control room as well as an established research methodology and staff, constitute a unique basis for the design, development and validation of operator support systems, as well as for more basic operator performance experimentation.

### 11.2.1 HAMMLAB Simulators

After the completion of the HAMMLAB 2000 project, HAMMLAB will consist of three nuclear simulators, the NORS modified VVER simulator, the Fessenheim-1 PWR simulator, and the Forsmark-3 BWR simulator.

- *The NORS Simulator*

The NORS simulator is a "westernised" VVER simulator of the Loviisa nuclear power plant in Finland. NORS was manufactured in 1983 by Nokia Electronics, and has since then been the nucleus of HAMMLAB. Several modifications and additions have taken place of the NORS simulator models, and NORS is now considered to be a good simulator for the purpose of being "the process" when performing experimental studies. However, several shortcomings are identified with NORS limiting the applicability, such as no accident simulation, troublesome to modify, and lack of proper documentation. Although HAMMLAB now consists of more simulators, NORS is foreseen to play a role in the coming years as one of the HAMMLAB pool of simulators.

- *The CP-0 Fessenheim-1 Simulator*

The member organisations of the Halden Project clearly expressed the wish of having a representative western PWR simulator as part of HAMMLAB. Based on this, the Project entered a contract with Electricité de France and Thomson Training and Simulation regarding a delivery of a full-scope PWR simulator of the EDF-operated Fessenheim plant. This plant is a Westinghouse-like 900 MW 3-loop plant built by Framatom. The simulator was delivered to Halden in the middle of 1998, and is based on the full-scale training simulator which was delivered to EDF in 1997. The simulator was originally equipped with a soft-panel interface, but the development of a new mimic-based interface was initiated in the current three-year period.

- *The F-3Forsmark-3 Simulator*

In order to prepare for maximum transferability of results from experimental studies, it was also decided to develop a BWR simulator for HAMMLAB. The development of a simulator of the Swedish Forsmark-3 1160 MW BWR has started in the current three-year period through a co-operation between the Halden Project and VTT Energy in Finland. The simulator will be installed in HAMMLAB, and will have a mimic-based MMI designed according to the unified MMI principles made for the NORS simulator.

- *Oil / Gas Production*

The Norwegian off-shore industry is very interested in performing human factors studies related to control room activities within the oil and gas production. Therefore it was decided to include the Oseberg simulator as one of the HAMMLAB pool of simulators. The Oseberg simulator is a full-scope simulator of the Oseberg A oil production platform located in the North Sea. The simulator was originally manufactured for Norsk Hydro in the late 80-ies by IFE and Norcontrol Simulation in Norway. This simulator will perform the role of a representative process for studies performed for the petroleum industry. Development and operation of this simulator is financed outside the joint programme, but results are expected to be transferable into the nuclear programme.

#### 11.2.1 Experimental Control Room

Figure 7.1 shows the layout of the HAMMLAB experimental control room as of 1998. It consists of 1 large screen (size: 2.00 x 0.75 m), 2 operator stations and 1 supervisor station. All units are on wheels in order to have a flexible and easily re-configurable control room. The overview information for the crew is presented on the large screen and on the 2x4 CRT screens in the upper row of the operator stations. For detailed process surveillance & control, each crew member has available 4 CRT screens, plus 1 CRT screen for computerised procedures, logbook, etc.

**Table 11.1:** *Human Factors Experiments Performed in HAMMLAB since the establishment.*

UT: User Test; Exp: Experiment; Guid: Guideline Evaluation; Mod: Model-Based Evaluation.

| Year | Title | Type | Systems | Authors | Reports |
|---|---|---|---|---|---|
| 1998 | Operator Performance at different Levels of Automation | Exp. | | Miberg | In preparation |
| 1997 | The effects of alarm processing and display on operator and plant performance. | Exp | | O'Hara, Brown, Hallbert, Skraaning, Persensky | Preliminary report, NRC |
| 1997 | A Questionnaire Comparison of Two Alarm Systems | Guid | | Collier | HWR-508 |
| 1997 | Human Error Analysis Project - The Fourth Pilot Study: Verbal Data for Analysis of Operator Performance | Exp | | Braarud, Drøivoldsmo, Hollnagel | HWR-495 |
| 1997 | The Effects of Advanced Plant Design Features and Control Room Staffing on Operator and Plant Performance | Exp | | Hallbert, Sebok, Morisseau, Persensky | HP-external-147 |
| 1996 | Results of the Study of Control Room Crew Staffing for advanced passive reactor plants. | Exp | | Hallbert, Sebok, Morisseau, Persensky | HPR-348 Vol. 1, subvol.14 |
| 1996 | Human Error Analysis Project (HEAP) - The Fourth Pilot Study: Scoring and Analysis of Raw Data Types | Exp | | Hollnagel, Braarud, Drøivoldsmo, Follesø, Helgar, Kaarstad | HWR-460 |
| 1996 | Practical Insights from Studies Related to Human Error Analysis Project (HEAP) | Exp | | Follesø, Kaarstad, Drøivoldsmo, Hollnagel, Kirwan | HWR-459 |
| 1995 | Human Error - The Third Pilot Study | Exp | | Follesø, Drøivoldsmo, Kaarstad, Collier, Kirwan | HWR-430 |
| 1995 | Human Error - The Second Pilot Study | Exp | | Kaarstad, Follesø, Collier, Hauland, Kirwan | HWR-421 |
| 1994 | Human Error - The First Pilot Study | Exp | | Kaarstad, Kirwan, Follesø, Endestad, Torralba | HWR-417 |
| 1994 | Measurement of the Operator's Situation Awareness of Use within Process Control Research: Four Methodological Studies | Exp | | Hogg, Follesø, Torralba, Strand Volden | HWR-377 |
| 1993 | The ISACS-1 Evaluation: A Simulator-Based User Test of the ISACS-1 Prototype | UT | ISACS | Follesø, Volden | HWR-342 |
| 1993 | The GOMS Evaluation of the Computerised Procedure Manual COPMA-II | Mod | COPMA-II | Meyer | MMSR-1444 |
| 1993 | GOMS Analysis as an Evaluation Tool in Process Control: An Evaluation of the ISACS-1 Prototype and the COPMA System | Mod | ISACS | Endestad, Meyer | HWR-349 |

| | | | | | |
|---|---|---|---|---|---|
| 1993 | Validation of the Post Trip Disturbance Analysis System SAS II at Forsmark | UT (Exp) | SAS-II | Holmström, Jacobsson, Henriksson | HWR-329 |
| 1992 | The Second Experimental Evaluation of DISKET: The Diagnosis System Using Knowledge Engineering Technique | Exp | DISKET HALO | Endestad, Holmström, Volden | HWR-307 |
| 1992 | A Guideline Evaluation of the Human-Machine Interface of the ISACS-1 prototype | Guid | ISACS | Follesø, Volden | HWR-309 |
| 1991 | Voice Output of Alarms Concerning Critical Safety Functions in a Nuclear Power Plant | Exp | CFMS | Holmström, Volden | EHPG '91 - C 5.3 |
| 1990 | The Experimental Evaluation of the Computerised Procedure System COPMA | Exp | COPMA | Nelson, Fødestrømmen, Holmström, Krogsæter | HWR-227 |
| 1989 | The First Experimental Evaluation of DISKET: The Diagnosis System using Knowledge Engineering Technique | Exp | DISKET HALO | Holmström, Nelson, Berg, Kårstad | HWR-242 |
| 1988 | The Experimental Evaluation of the Success Path monitoring System SPMS | Exp | SPMS, CFMS HALO | Baker, Marshall, Reiersen, Smith, Gaudio | HWR-223, HWR-224 |
| 1987 | The Evaluation of a Prototype Human-Machine Interface for the Early Fault Detection System EFD | UT | EFD | Verle, Marshall | HWR-202 |
| 1987 | A Comparison of Operator Performance When Using Either an Advanced Computer-Based Alarm System or a Conventional Annunciator Panel | Exp | HALO | Reiersen, Marshall, Baker | HPR-331 |
| 1987 | Further Evaluation Exercises with the Integrated Process Status Overview IPSO | UT | IPSO | Reiersen, Marshall, Verle, Gertman | HWR-184 |
| 1986 | Proof of Principle Evaluation of the Integrated Process Status Overview IPSO | UT | IPSO | Gertman, Gaudio, Nilsen, Burns | HWR-158 |
| 1986 | Further Comparisons of Operator Performance when using Differing Display and Control Modes | Exp | NORS | Baker, Holmström, Marshall, Reiersen | HWR-178 |
| 1985 | A Comparison of Operator Performance using Three Display Modes | Exp | NORS, HALO | Baker, Holmström, Marshall | HWR-152 |
| 1985 | Experimental Comparison of Three Computer Based Alarm Systems | Exp | NORS, HALO | Baker, Gertman, Hollnagel, Holmström, Marshall, Øwre | HWR-142 |
| 1984 | Pilot Experiment on Multilevel Flow Modelling Displays using the GNP-Simulator | UT (Exp) | MFM | Hollnagel, Hunt, Prætorius, Yoshimura | HWR-114 |
| 1983 | The Experimental Validation of the Critical Function Monitoring System CFMS | Exp | CFMS | Hollnagel, Hunt, Marshall | HWR-111 |

**Table II.2:** *Applications and System Toolboxes developed with basis in the Halden Reactor Project.*

| System | Short description | Status |
|---|---|---|
| Picasso-2 | Graphic user interface and Application interface for the energy and process industry | Concluded, in-use |
| Picasso-3 | Object-oriented User Interface Management System | In-use, Further development |
| Software Bus | Communication Software for distributed process control | In-use, Further development |
| COAST | Object-oriented alarm system toolbox | In-use, Further development |
| PEANO | Signal validation based on neural nets and fuzzy logic | In-use, Further development |
| ALADDIN | Event identification system based on neural nets and fuzzy logic | In-use, Further development |
| EFD | Model based, dynamic alarm system and Signal validation | Concluded, In use |
| MOCOM | Model based condition monitoring system based on EFD principles | Concluded |
| DD | Detailed diagnosis system based on EFD principles | Concluded |
| SCORPIO | Core Surveillance System<br>- PWR version<br>- VVER version | In-use |
| COPMA | Computerbased procedure system | In-use, Further development |
| SAS-II | Expert system for monitoring critical safety functions in BWRs | Concluded |
| ASA | Expert system for analysing the status of control system | Concluded |
| CAMS | Accident Management System | Further development |
| MEMBRAIN | Emergency management system | Concluded, Redesign to be considered |
| ASSIST | Virtual Reality toolkit for building training systems | Further development |
| ACCESS | Toolkit for knowledge management | Concluded, In-use |
| HRP-Prover | Toolkit for formal specification of software | In-use, Further development |
| SOSAT | Toolkit for safety analysis of programmes | Concluded, In-use |
| SAP | Safety assessment of programmes | Concluded |
| STEM | Software testing method | Concluded |

Attachment 2

# Executive Summary

## Achievements of the Halden Project Programme in the 1997-1999 Period: Man-Machine Systems Research

October 1999

# FORMAL SOFTWARE DEVELOPMENT METHODS

***Objective***

Formal software development methods are methods which provide a mathematically based framework within which specification, development and verification of software systems can be done in a systematic and precise way. The main idea is to use mathematical techniques to describe properties of the desired system and proofs to verify the design steps from this description. Formal methods provide an efficient way of developing a system and its proof hand-in-hand. This gives more reliable systems giving their designers and customers more confidence than they otherwise would have.

***Results***

A formal software development methodology based on algebraic specifications was established. This methodology has been applied in the EvalFM project (see separate work item). Another application has been the redevelopment of the HRP Prover, a tool applied in formal software development.

A survey has been made to study methodologies assumed well suited for describing behaviour of real- time systems, i.e. those being able to describe properties related to concurrency, communication, time constraints etc. in an explicit way. The activity in this area has been to apply two formal methods, LOTOS and CSP. The chosen system is a control system for a steam boiler in a NPP.

***Main ref.***

T. Sivertsen, HWR-455 (1996). A.-K. Groven, HWR-453 (1996).

***Next step***

To apply formal development methods on a realistic project. One such project is the development of new software for HAMMLAB. Particular emphasis will be to make formal methods understandable, and thereby usable, to people of different background and expertise.

To complete the formal development of a new version of the HRP Prover. The new version will have a functionality almost identical to the existing version, but is developed in accordance to the established formal development method.

To facilitate the introduction and use of the HRP Prover by developing instruction material to a tutorial programme. This programme will be applied internally at the Halden Project, and the experiences thus obtained will be used to update the material.

# GRAPHICAL TECHNIQUES IN FORMAL SPECIFICATIONS

***Objective***

Contribute to a clarification of the relationship between graphical descriptions and formal specifications, and provide guidelines for how they can be combined in order to utilize the strengths of each approach. In particular, the research aims at investigating how graphical descriptions can be supported by the algebraic specification language and associated tool (the HRP Prover) developed at the Halden Project. A related objective is to generalize the research results to other non-graphical formal specification languages.

***Results***

The relationship between graphical descriptions and formal specifications are studied in a co-operative project with ENEA. The work utilises results achieved in research projects like IPTES (ENEA) and EvalFM (Halden Project). In this period, ENEA has been applying the IPTES technology on the PRM example from the EvalFM Project.

Since many graphical methods can be translated to Petri nets, a main contribution from the Halden Project on this subject has been focused on the relationship between Petri nets and algebraic specifications. One of the achievements has been the establishment of a uniform approach to the translation of autonomous Petri nets into algebraic specification, with subsequent safety analysis using the HRP Prover. The established approach is sufficiently general to form a basis for further research on extended Petri net models and on related graphical description languages.

***Main ref.***

T. Sivertsen, HWR-454 (1996). R. Bove et al., HWR-457 (1996).

***Next step***

To utilise the research results in the translation of various graphical languages into algebraic specification. The research will focus on industrially accepted graphical languages, and investigate how these can be combined with non-graphical formal specification languages. A related research issue is to which extent such a combination would benefit software development, verification and validation, and licensing. Priority will be given to languages that are found useful in the nuclear industry.

The research on graphical methods will constitute a basis for the development of graphical front-ends to the HRP Prover supporting the combined use of these methods and algebraic specification. The first front-end to be developed is a tool that supports hypertext-based editing of algebraic specifications, version-control of edited specifications, and graphical representation of the hierarchical relationship between specifications.

## APPLICABILITY OF FORMAL METHODS - THE EVALFM PROJECT

| | |
|---|---|
| ***Objectives*** | Investigate the applicability of formal methods in the development of safety-critical software systems. The EvalFM project has in particular focused on the evaluation of the formal software development method based on algebraic specification and the HRP Prover, by applying them in a realistic case study. |
| ***Results*** | The method was applied on a case example based on the computer-based power range monitoring (PRM) system installed at Barsebäck NPP in Sweden. The method was used to formally specify and design one out of four similar subsystems of the PRM system. Based on the requirements document, a formal algebraic specification has been written, utilising the mathematical tool-kit defined in /HWR-331/. Using the design and implementation techniques discussed in /HWR-363/, the subsystem has been designed and implemented in a safe subset of Pascal. The project has also investigated how the design can be varied to allow implementation in other languages and to put stronger emphasis on efficiency, as well as the applicability of algebraic specification for aspects relating to timing, communication and diversity. The overall purpose of these investigations has been to contribute to an understanding of the role and applicability of formal methods for these and other aspects of relevance to software V&V. |
| ***Main ref.*** | A.K. Groven and T. Sivertsen, HWR-397 (1994). T. Sivertsen, HP-External No. 114 (1996). |
| ***Next step*** | The project is completed. |

## SOFTWARE TESTING AND RELIABILITY ASSESSMENT - THE STRAM PROJECT

***Objectives***

Investigate software testing and reliability assessment methods, and apply some of them to various programs.

***Results***

A literature review (*RESTRAM*) of the last decade of research in the field of software testing and reliability assessment methods has been made. The review presents the most commonly used methods/techniques within the fields of software reliability models, fault-tolerant computing, and software testing methodologies. Emphasis is put on the practical application of the methods, in particular for the use in safety assessment of safety critical software. In addition, particularly interesting methods were identified for further investigations at the Halden Project.

An experimental investigation (*EISTRAM*) is made of the applicability and feasibility of one of the reviewed methods, viz. the PIE-method proposed by dr. Jeffrey Voas. The method, which combines random testing with sensitivity analysis, may provide an estimate of the minimum fault-size in a program over a class of possible faults. This is accomplished by seeding a set of artificial faults (called mutants) belonging to this class into the program. A number of different mutants have been identified and classified according to structural and syntactical differences. The method was manually applied to two small programs.

A semi-automatic test harness is currently being developed, including an automatic mutant generator and report generator. As more mutant classes are being implemented, the fraction of manual labour necessary is greatly decreased. The PIE-method is being applied to the APRM program developed in the EvalFM project.

***Main ref.***

H. Thunem, HWR-425 (1995). B.A. Gran and H. Thunem, HWR-456 (1996).

***Next step***

To finish the tests of the APRM-program.

Develop an automatic perturbation code generator for the propagation part of the PIE-method.

To apply the PIE-method and various software testing and reliability assessment methods to proprietary software.

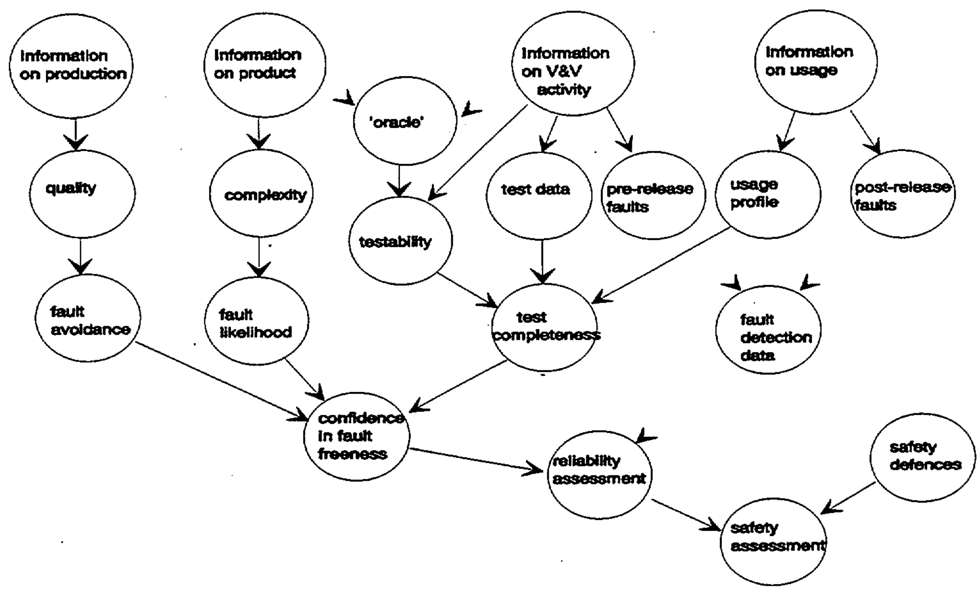
## SAFETY ASSESSMENT OF SYSTEMS CONTAINING CONFIGURABLE SOFTWARE

***Objectives***

To establish a method to assess the safety of configurable software systems.

***Results***

The figure gives an illustration of the acceptance process for a safety critical system which contains configurable software, i.e. a program which is built up by an application program and a set of standard software modules. The safety assessment is influenced by an assessment of the reliability of the system, and also by the degree of additional safety defenses which exists. The top nodes in the graph represent the basic information sources which are used in the acceptance process. These include an assessment of the producers products in general, the quality assurance policy and the conformance to standards for safety critical systems. Information about the product can be obtained from the specification, code listing etc.




To assess complexity, which is of relevance for the vulnerability to errors, one should make a program analysis. Several methods have been investigated, including control and data- flow analysis, Failure Mode And Effect Analysis and Fault Tree analysis. Further information is data on previous experience, as field of usage, number of users, total time of usage, reported faults etc. In addition, safety defenses, like e.g. functional diversity, should be made to protect against undesirable consequences of failures and thereby enhance the safety.

***Main ref.***

G. Dahll and U. Pulkkinen, HWR-458 (1996). G. Dahll, HP-External No. 135 (1996).

***Next step***

To investigate the use of Bayesian networks to describe the influence from the different information sources on the final safety assessment.
To apply the methodology on a realistic case example.

## LESSONS LEARNED ON SOFTWARE DEPENDABILITY

***Objectives***

A review of all Halden Project activities has made in the field of software reliability and verification and validation from 1977 to 1994, with special emphasis on the observations and conclusions made from these activities, in particular how they can be utilised by different types of organisations, as e.g. licensing authorities, to assure adequate integrity of software for safety critical applications, safety assessors, power companies and software developers.

***Results***

A lessons learned report (in two volumes) was made, containing descriptions of the actual work which has been made. Part I (HWR-374) is intended to be a guide through the Halden activities and corresponding reports. It contains short descriptions of ten projects on software dependability, each containing a short abstract, reference to relevant reports, the outcome (i.e. products and type of results) of each project, and conclusions and recommendations. In addition, this volume contains a glossary of terms, a subject index with reference to the reports and a summary of conclusions and recommendations

Part II (HWR-375) describes the different projects in more detail. For each project there is a separate chapter, with sections containing a fairly short description of each project, reference to written material and observations made during the project. The observations, which were based on direct results as well as on the experience gained by working on the projects, are not necessarily of general nature, but they can support more general conclusions and recommendations.

***Main ref.***

G. Dahll and T. Sivertsen, HWR-374 (1994). G. Dahll and T. Sivertsen, HWR-375 (1994).

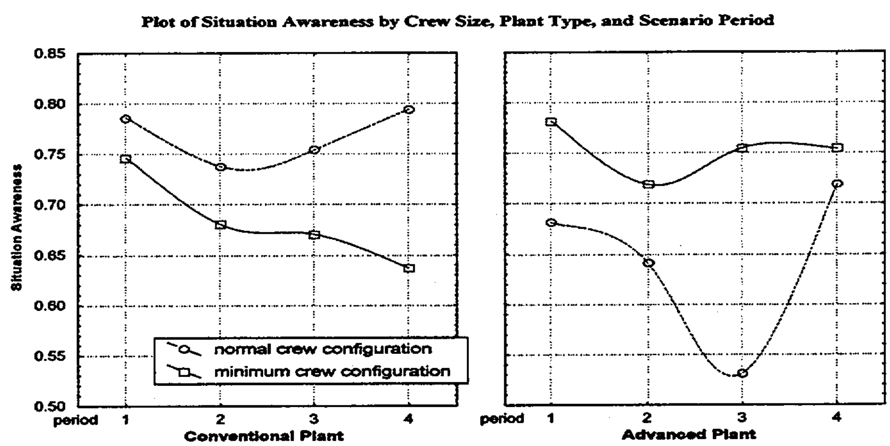
***Next step***

The project is completed

# A STUDY OF STAFFING LEVELS FOR ADVANCED REACTORS

***Objective***

A study was conducted to evaluate issues affecting staffing requirements in advanced plants and to provide data supporting design review guidance. Two factors were evaluated across a range of plant operating conditions: 1) control room crew staffing configuration; and, 2) characteristics of the operating facility itself, whether employing conventional or advanced features. This work was done as part of a bi-lateral agreement with the U.S. Nuclear Regulatory Commission.




***Results***

Crews in the advanced plant setting achieved higher situation awareness than crews in the conventional plant. Minimum-sized crews performed better than normal-sized crews in the advanced plant; conversely, normal-sized crews performed better than minimum-sized crews in the conventional plant. This interaction may be explained by design aspects of the control rooms which better support different crew staffing complements. Crews in the advanced plant setting also exhibited better and more stable team interaction than crews in the conventional plant.

Minimum-sized crews experienced more workload than normal-sized crews, with the control room supervisor experiencing most of the additional workload. Advanced-plant crews experienced significantly higher workload than their conventional plant counterparts.

Regarding objective performance, normal-size crews performed better than minimal-size crews in cooldown and stabilization of the plant. However, both crew sizes performed similarly on transient mitigation.

***Main ref.***

B. P. Hallbert et al., NUREG/IA-0137.

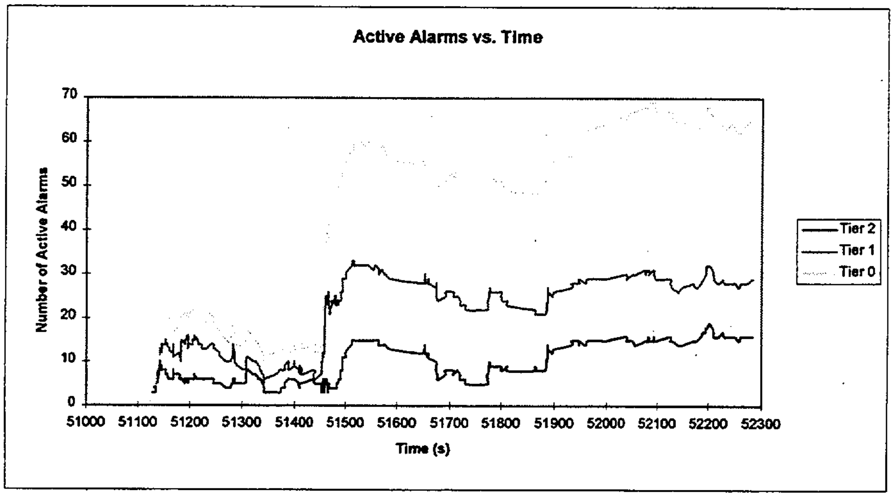
***Next Step***

Future work could be performed to determine whether relationships exist between the metrics. This could help improve our understanding of, for example, how workload transition affects operator situation awareness and team interaction, or the role team interaction plays in the development of crew situation awareness.

# EFFECTS OF ALARM PROCESSING AND DISPLAY ON OPERATOR AND PLANT PERFORMANCE

***Objective***

The objective of this project is to provide a technical evaluation the impact of alarm system display and processing on operator performance relative to plant system performance. This study is intended contribute to the understanding of potential safety issues, and to provide data to support the development of design review guidance for alarm guidance and display. This project is part of a bi-lateral agreement with U.S. Nuclear Regulatory Commission.




***Results***

Sixteen scenarios were designed to evaluate the effects of alarm processing and presentation on operator and plant performance. To evaluate the effects of alarm processing techniques, three alarm processing "tiers" or conditions were developed. These were based on techniques or algorithms to reduce the number of incoming alarms by suppressing nuisance or redundant alarms.

In addition to the alarm processing techniques, three different types of alarm displays were developed for presenting alarms to operators. These included: a tile-based system, similar in nature to conventional alarm systems; alarm lists providing textual information, and; alarm lists supplemented with graphical integration of the alarms on the process control and information displays.

Six crews of operators came to Halden from the Loviisa nuclear power generating station in Loviisa, Finland. During the experiments that followed, data were collected on situation awareness, operator workload, objective performance, complexity, and plant performance.

***Main ref.***

Work in Progress.

***Next Step***

Analyses of data were begun in 1996 and will be completed in early 1997. A final report will be issued, and a prioritized list of unresolved issues associated with the alarm systems will be produced. The unresolved issues will be used to identify future research issues.

# OPERATOR TRAINING PROGRAM

***Objective***

The objective of the training program is to improve operator competency with the instrumentation and control (I&C) systems and model of the PWR process used in HAMMLAB for conducting experimental studies. Operators from the Halden HBWR are accustomed to operating a nuclear process different to the simulated PWR modeled in HAMMLAB. In addition, the I&C systems at the HBWR are different than those used in HAMMLAB. Thus, a training program based on enabling and terminal learning objectives similar to those used at commercial nuclear power plants is needed. This will provide our own operators with sufficient background and experience with HAMMLAB systems to achieve a level of standard proficiency for participation in experimental studies of new systems and concepts in control room I&C.

***Results***

A number of activities have been accomplished on the new training program for HBWR operators in HAMMLAB. Operators from the HBWR came to HAMMLAB to learn about the research activities we plan to carry out, and to observe and participate in operations in the simulator. Sixteen operators to date have volunteered to take part in the new training program.

A more systematic approach to training of HBWR operators is being adopted, involving more formal lesson plan preparation and assessment of operator performance. A training needs assessment was developed and delivered to the HBWR operators, and their answers were obtained. Subsequently, a more detailed training program was developed to adapt their skills as HBWR operators for operation of the simulated PWR process in HAMMLAB, and the different types of automation employed. The results obtained from the needs assessment will also serve as a baseline of PWR process knowledge that can be used to evaluate training effects.

To date, all volunteers have participated in a number of training sessions in HAMMLAB. These training sessions have focused on process familiarization, the NORS interface, and operating situations. Operators also traveled to the Loviisa nuclear power station which is the process that is modeled in HAMMLAB. While at Loviisa, operators became more familiarized with the systems, plant layout, and operation of the plant. It is expected that these and future training activities will provide greater control over the prior experience of participants in HAMMLAB studies, and even for the use of training itself as a subject of study at Halden.

***Main ref.***

Training documentation and needs assessments.

***Next Step***

A training schedule and curriculum have been developed for 1997. System operation, transient and procedure familiarization will continue in 1997. Use of operators in experimental studies will be assessed this year as part of training assessment activities.

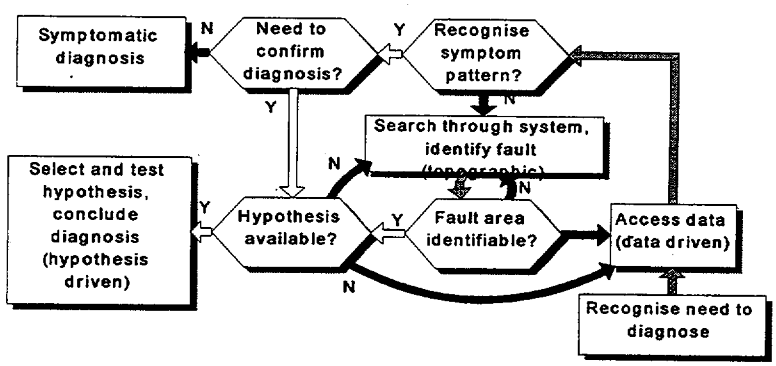
# THE HUMAN ERROR ANALYSIS PROJECT (HEAP): IMPROVED UNDERSTANDING AND MODELLING OF OPERATOR DIAGNOSIS

***Objective***

Enhance the understanding of performance failures due to operator cognition and characteristics of diagnostic performance in accident situations, both to provide design guidance to support operators in conventional and advanced control rooms, and to improve the modeling and assessment of such performance failures in risk and reliability assessment.

***Results***

A plan for an experimental program was developed for HEAP to investigate the impact of human cognition on performance failures and in particular on diagnosis. A series of four methodological studies have been carried out which have provided guidance on efficient ways of collecting and analyzing data pertinent to the purpose.




*A model of the operator's diagnostic process developed through research conducted in the Human Error Analysis Project*

The results address the use of data sources such as concurrent, interrupted, and post-hoc auto-confrontation verbal protocols, eye movement tracking, expert commentators, audio, video, and various forms of questionnaires and direct performance measures. The last of the methodological studies looked specifically at the relative contribution of the various data sources.

As a basis for analyzing diagnostic strategies, an *a priori* model had been proposed which assumed that operators would select one of a set of "pure" strategies (topographic, symptomatic, hypothesis-driven, and data-driven). The results showed that operators use a mixture of strategies, conforming with the demands of the situation and the working conditions (mainly level of experience and team vs. single operators).

***Main ref.***

B. Kirwan, HWR-378 (1994). D. Meister et al., HWR-379 (1995). E. Hollnagel et al., HWR 459 (1996).

***Next Step***

Development and test of a set of methods for predicting the likely error modes for performance under given conditions, and use the results of this to improve the modeling of operator performance.

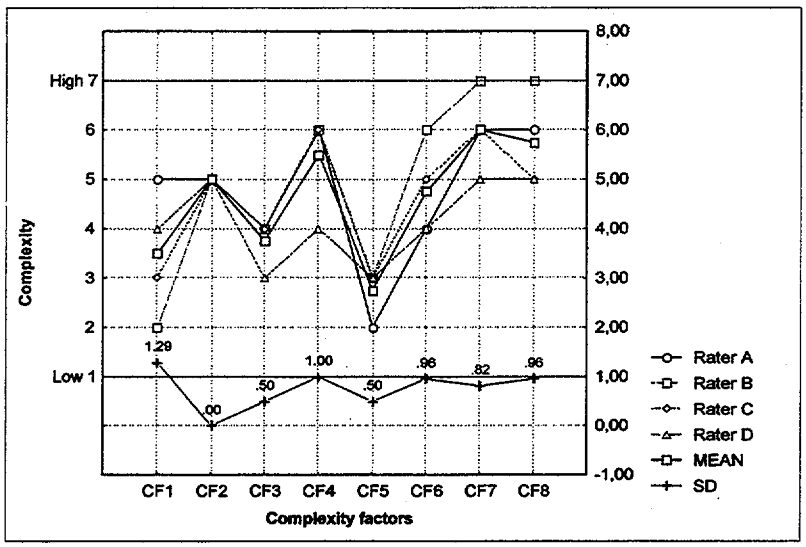
# THE HUMAN ERROR ANALYSIS PROJECT (HEAP): OBTAINING INSIGHTS INTO COMPLEXITY

***Objective***

Complexity is recognized as a main factor contributing to human performance and error in complex technological environments. As part of achieving the objectives of the Human Error Analysis Project, efforts have been focused on identifying the components of complexity.

***Results***

A separate effort has been dedicated to the development of a complexity profiling system. During some of the methodological studies, as well as in the Study of Staffing Levels for Advanced Reactors, a 32-point questionnaire addressing the concept of complexity was filled out by the subjects.




*Example complexity profile of a scenario used in the Alarm Project study, 1996*

Based on these data, a set of eight complexity factors were identified by statistical techniques. The eight factors can be used to establish a complexity profile of a situation. This technique has been applied in the selection of appropriate scenarios for the alarm study this year, and will be further developed into a method to aid performance prediction. The potential for using the complexity profiling system with human error prediction in risk assessment will also be evaluated. Reliability and validity issues of the complexity profiling system is currently being investigated, using data from the recent HEAP experiment.

***Main ref.***

B. Kirwan, HWR-378 (1994). K. Follesø et al., HWR-430 (1995). E. Hollnagel et al., HWR-459 (1996).

***Next Step***

Continued refinement of an approach to develop a complexity profile for a situation, and investigation of its applicability to performance prediction.

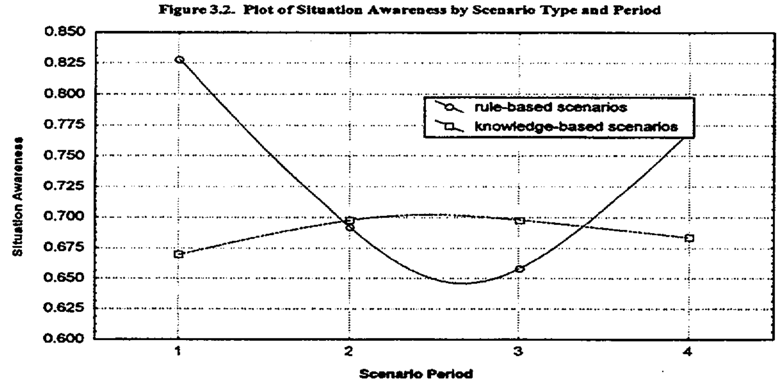
# DEVELOPMENT AND DEMONSTRATION OF A METHOD FOR MEASURING THE OPERATOR'S SITUATION AWARENESS

***Objective***

To develop and validate a measurement technique which is sensitive to the operator's awareness of the plant and process states. The measurement technique should provide researchers with a means of assessing the operator's situation awareness in response to dynamic changes in the processes, thus providing insight into how to design systems to best support the operator.

***Results***

A series of four methodological studies were performed in which a technique to measure the operator's situation awareness was developed, it's sensitivity to differences in operator competence and changes in the process state were investigated, and measures of validity were studied.




*An example of situation awareness data gathered using the SACRI measurement technique (Source: The Study of Staffing Levels for Advanced Reactors, B. Hallbert et al.)*

The Situation Awareness Global Assessment Technique (SAGAT) developed by Mica Endsley was adapted for use in measuring situation awareness in a process control environment.

The Situation Awareness Control Room Inventory (SACRI) was shown to be sensitive to differences in operator competence, changes in the process state, and system interface differences. In addition, SACRI appears to be reliable, and possess important metric properties such as content and construct validity which are important to it's acceptance as a valid means of measuring operator performance.

***Main ref.***

D.N. Hogg et al., HWR-377 (1994). D.N. Hogg et al., "Development of a situation awareness measure to evaluate advanced alarm systems in nuclear power plant control rooms." *Ergonomics*. 38(11), pp. 2394-2413, 1995.

***Next Step***

Continued application of SACRI in studies to evaluate it's utility in evaluating different methods for presenting information to control room operators, and study the situation awareness of individual control room crew members.

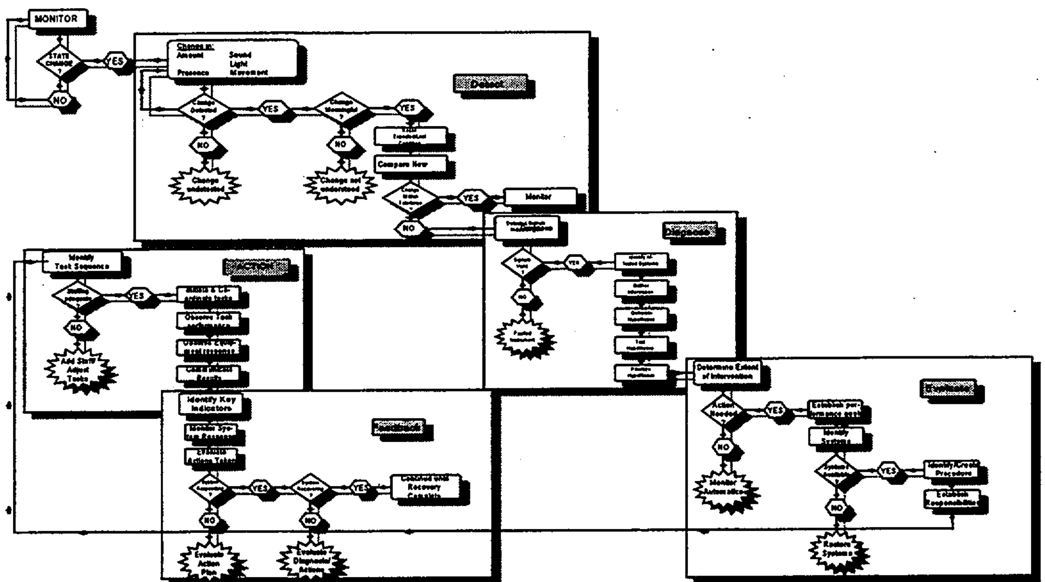
# LESSONS LEARNED RELATED TO DESIGN AND EVALUATION OF HUMAN-MACHINE SYSTEMS

***Objective***

To provide a description and summary of the lessons learned from the tests and evaluations related to the design of and the evaluation of new control room instrumentation and control technologies. A further objective is to provide information relevant to understanding human-machine systems which, together with other research, can serve as technical bases in the formulation of guidelines for design and evaluation of human-machine systems.

***Results:***

Human factors studies performed by the Halden Project in the 1983-93 period were analyzed.



*Process Control Human Performance Model developed in the lessons learned report*

Data from these studies were extracted and presented to show how the various systems evaluated supported the different activities of fault detection, diagnosis, evaluation and formulation of a response plan, carrying out control room tasks, and obtaining feedback from the control room information systems about the state of the process.

In addition to findings related to these activities, the report discusses and provides illustrations of the potential of new control room systems and technologies to:

- affect operational issues including the role of the control room
- operator, whether intentional or not, and
- moderate or increase operator workload.

The report further discusses the need to integrate new systems with the goals and objectives of the control room crew.

***Main ref.***

B.P. Hallbert & P. Meyer, HWR-376 (September, 1995).

***Next Step***

The project is completed.

## LESSONS LEARNED ON AUTOMATION AND ALLOCATION OF FUNCTIONS

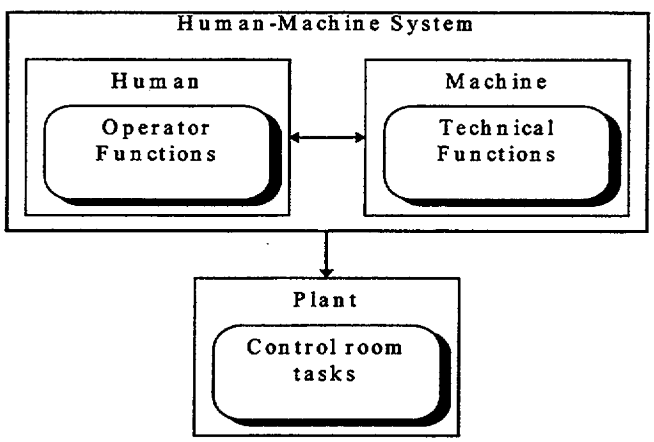
***Objective***

To derive lessons from Halden human factors research work on allocation of functions to people and machines and on joint human-machine system performance.

***Results***

The degree of automation and how machines and humans work together continues to be important. Several Halden studies of operator support systems offered incidental experience in this area. Lessons were drawn from these experiences concerning:

- The combinations of operator functions and machine functions (the human-machine system) and how these can be better optimised so as to support the required tasks of the central control room.
- The ways that allocations of function between human and machine can be decided.




*A figure of how humans and machines work together, taken from the lessons learned report*

Human factors studies performed by the Halden Reactor Project were analyzed, covering the period between 1983 and 1995. These studies concerned various operator support systems. For each system, lessons were derived on the effects of automation on the joint system performance (human and machine).

The main lesson learned was that the lack of integration of operator support systems can create problems. This is discussed in terms of the common problem of a 'leftover' strategy for automation, in which functions defined without regard to human factors criteria as being outside a system's boundaries are left to the human to deal with. This strategy is compared with a 'complementarity' strategy, in which the performance of human and machine is considered as a joint system.

***Main ref.***

S. Collier, HWR-461 (1996).

***Next Step***

The issues of automation and allocation of function have been introduced to the 1997-1999 research program.

## WORKSHOP ON STUDIES OF OPERATOR PERFORMANCE DURING NIGHT SHIFTS

***Objective***

The purpose of this workshop was to discuss and make recommendations on specific needs for the study of operator cognitive performance at night and identify the relevant research issues for which Halden could provide resolution.

***Results***

Six invited speakers with expertise in studies of shift work gave presentations, and three working groups discussed the following issues in parallel:

- Lines of Research to Be Pursued
- Methods and Measures to Be Used in Research on Cognitive
- Performance at Nights
- Products of the Research on Operator Performance at Night.

Many issues were discussed and presented. A summary of the overall conclusions, is:

The Halden Project's research program on operator performance at night should be both a short time and a long-term program. The overall emphasis should be on higher level cognitive functions and possible circadian effects on those functions because past research has not addressed such issues to any discernible extent.
International co-operation should be sought, especially with respect to member countries providing data from their own research.
The program should consider both passive or routine monitoring and disturbance conditions.
Interviewing operators at the HBWR about the problems they have experienced with working night shifts would be a good potential starting point for this research.
Design of computerised operator support systems were not discussed in any detail. These should, however, be included in studies that address higher order cognitive functions during night operations because of their potential for improving operator performance.
While some complementary research on shift schedules might be useful, it should not be the focus of this proposed research program; there is already a great deal of data in that area.
The Halden Reactor Project should continue to issue lessons-learned reports especially on issues such as the proposed research on operator performance at night; these reports have proven useful in the past.
In the long-term program, studies on countermeasures such as increased lighting levels and naps should be considered; data from such studies would be valuable for all member countries.

***Main ref.***

D.S. Morisseau et al., HWR-462 (1996).

***Next Step***

Initiation of a simulator-based program of research of operator performance during night shifts

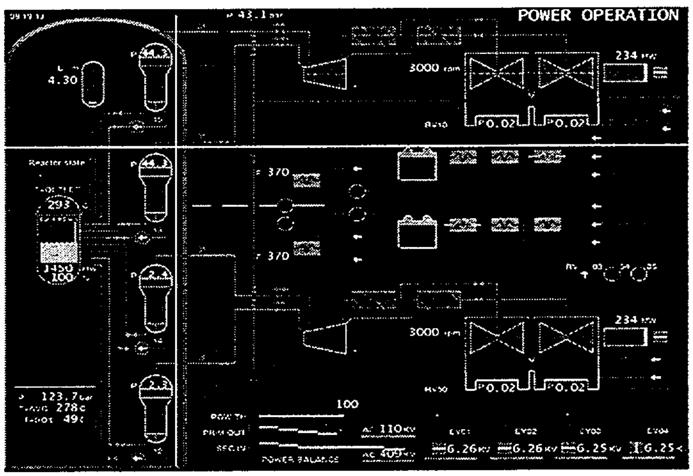
# EYE-MOVEMENT TRACKING ANALYSIS - ACHIEVEMENTS AND WAYS OF USING THIS METHOD

***Objective***

To develop techniques to study the information gathering activities of operators by use of eye-movement tracing equipment and to develop methods from this data to infer mental activities.

***Results***

During the 1994-1996 period, eye-movement tracking analysis (EMTA) has been studied to determine how this data collection method complements other data collection techniques when studying how operators gather information to support diagnosis.




*A figure showing eye-movement tracking data. The intersection of the cross hairs superimposed onto the process format shows the position of the operator's visual focus in real time*

Further studies showed how this technique could be combined with verbal protocol data to solve a persistent problem: the gaps in verbal protocols that occur during cognitively demanding problem solving situations.

The methodology and software developed for scoring eye data, Eyecon, has been improved, and a new version of this software was developed.

The equipment has also been made portable, enabling testing and evaluations of systems in the field.

***Main ref.***

P.Ø. Braarud et al., HWR-460 (1996). G. Hauland: Building a Methodology for Studying Cognition in Process Control: A Semantic Analysis of Visual and Verbal Behaviour. Post-Graduate Thesis in Psychology, NTNU (1996). M. Kaarstad et al., HWR-417 (1994), HWR-421 (1995).

***Next Step***

Further evaluate system designs, do further testing of what information eye movement data can give about operators' cognition.

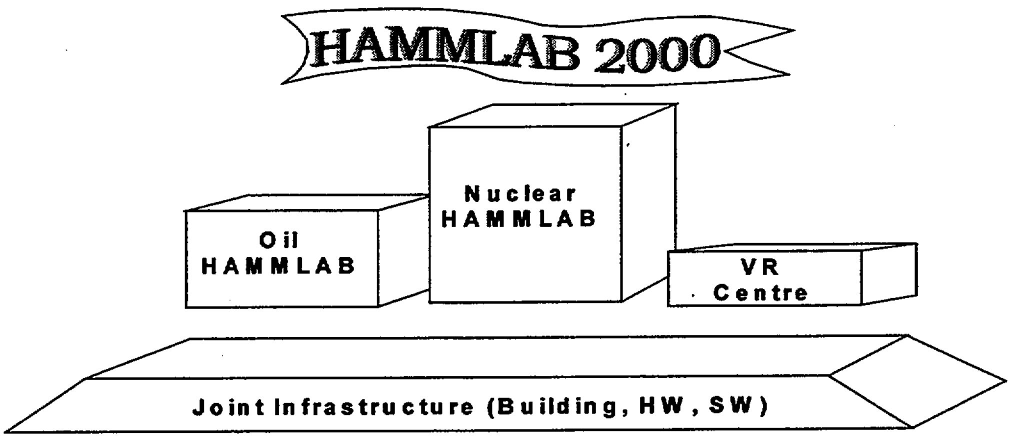
# HAMMLAB 2000 - THE FUTURE HALDEN MAN-MACHINE LABORATORY

***Objective***

The goal is to make the HAlden Man-Machine LABoratory (HAMMLAB) a global centre of excellence for the study of human-technology interaction in the management and control of industrial processes.

***Results***

In order to make sure that HAMMLAB also in the future can meet the requirements to an experimental facility from the research programme, a pre-project was initiated with the goal of identifying whether today's HAMMLAB could meet these requirements.




*Main elements of the HAMMLAB 2000 projects*

Based upon a prediction of what type of studies and experiments would be interesting within the human-machine systems research, requirements to a future experimental facility were identified. The conclusion was that today's HAMMLAB had several weaknesses, and that a new experimental facility was required to meet the research demands for the next decades.

As a result of the above conclusion, the HAMMLAB 2000 project was initiated in the fall 1996. An organisation has been established for this project including advisory groups, acting as a link between the HAMMLAB 2000 project and HPG as well as interested member organisations. The overall HAMMLAB 2000 concept will focus on an experimental facility for the Halden Project member organisations within the nuclear field, but will also include an experimental facility based on the needs from the oil and gas industry, and a centre for VR applications.

***Main ref.***

C.O. Fält et al., HWR-428 (1995). N.T. Førdestrømmen et al., HWR-476 (1996). J. Kvalem et al., HWR-452 (1996).

***Next Step***

The work in 1997 will focus on several main areas. One is the identification of simulation capabilities required for the new laboratory. Discussions were initiated in 1996 regarding both PWR and BWR simulators, and decisions will be made early 1997. Based upon these decisions work will be initiated regarding the installation/development of these simulators.

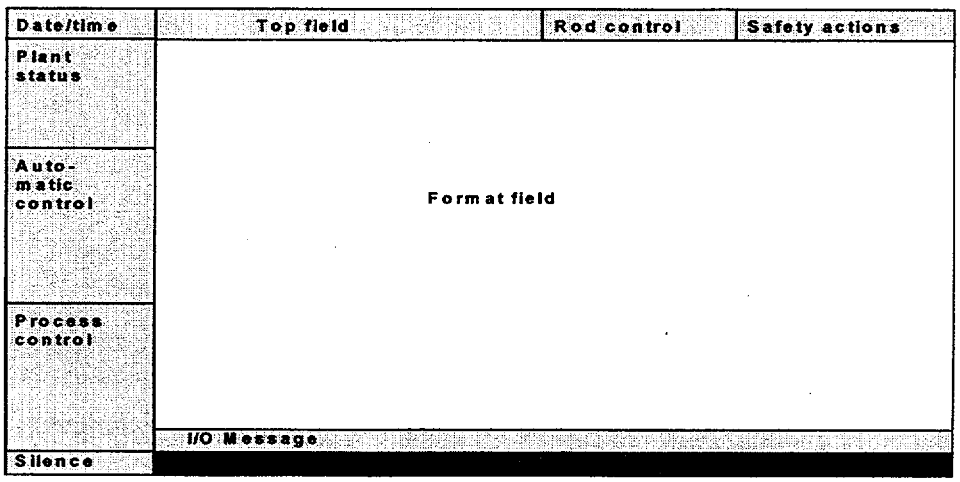
## UMMI: UNIFIED MAN-MACHINE INTERFACE FOR HAMMLAB

***Objectives***

Develop an unified, flexible and integrated interface for the NORS process and the set of operator support systems for use in a wide range of experimental control room setups in HAMMLAB. The design should demonstrate new advanced features, and be applicable for retrofitted as well as advanced control rooms.

***Results***

The Unified Man-Machine Interface for NORS/ISACS-1, UMMI, has been designed, implemented, tested out, and made operational in HAMMLAB. The design of the realized displays as well as the future planned ones are all based on a generic display layout. This layout is divided into a set of fields and each field has well-defined functions, see figure below.




*Generic display layout for UMMI & NORS/ISACS-1*

The functions of the old NORS touchpanel keyboard have been integrated into the displays. This includes display retrieval and all process control functions. Thus, only a standard alphanumeric keyboard is needed on the operator`s desk. The NORS process displays have also socalled "smart" functions included: For process control, the system checks whether the different control buttons (start/stop, on/off) are allowed to be operated or not in a given process situation. Currently, the display system consists of 46 process and control formats for the NORS simulator. In addition, about 100 trend displays for predefined combinations of process parameters are available, together with CASH selective displays and event log display. The UMMI is implemented by the graphical interface system Picasso-3.

***Main ref.***

J.Ø. Hol et al., HWR-399 (1994). N.T. Førdestrømmen et al., HWR-476 (1996).

***Next Step***

The UMMI will be further tested out and corrected for errors and shortcomings. In addition, it will gradually be upgraded by including new advanced design features. The set of trend diagrams, each with 6 parameters shown, will be expanded to about 200 displays.

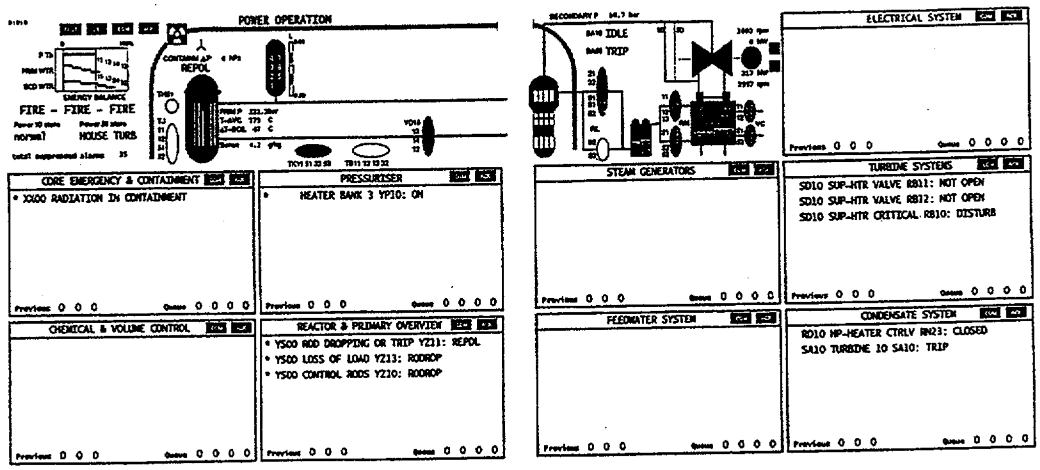
# CASH - NEW ALARM SYSTEM IN HAMMLAB

***Objective***

Develop an alarm system where new and innovative alarm system concepts are demonstrated. Put main emphasis on user interface design and efficient alarm generation and structuring/suppression methods.

***Results***

CASH has been designed, implemented, tested out, and made operational. The system consists of 3 main modules; alarm generation, structuring and presentation.




*CASH Overview no.1; group-list based*

Three alternatives for overview displays have been made; one group-list based, one process mimics based, and one icon/group list based, refer figure above. A set of selective displays are also available for the operator through a menu. They provide supplementary information to the overview. The system possesses a high degree of alarm suppression capability. The design is very flexible, making it relatively easy to change the man-machine interface and the degree of alarm suppression.

In 1996, CASH has been used in a human factor experiment on alarm systems, where three different interface designs; tile-, process mimics-, and icon/group-list based was tested together with three levels of alarm reduction.

The man-machine interface is implemented using the graphical user interface system Picasso-3, while the alarm generation and structuring modules are implemented using the alarm system toolbox COAST.

***Main ref.***

N.T. Førdestrømmen et al., HWR-398 (1994). P. Miazza et al., HWR-362 (1993). B. Moum et al., HWR-480 (1996).

***Next Step***

Future work related to CASH will be based upon the results of the alarm experiment being conducted in the fall of 1996. These results are anticipated to be made available in the spring of 1997.

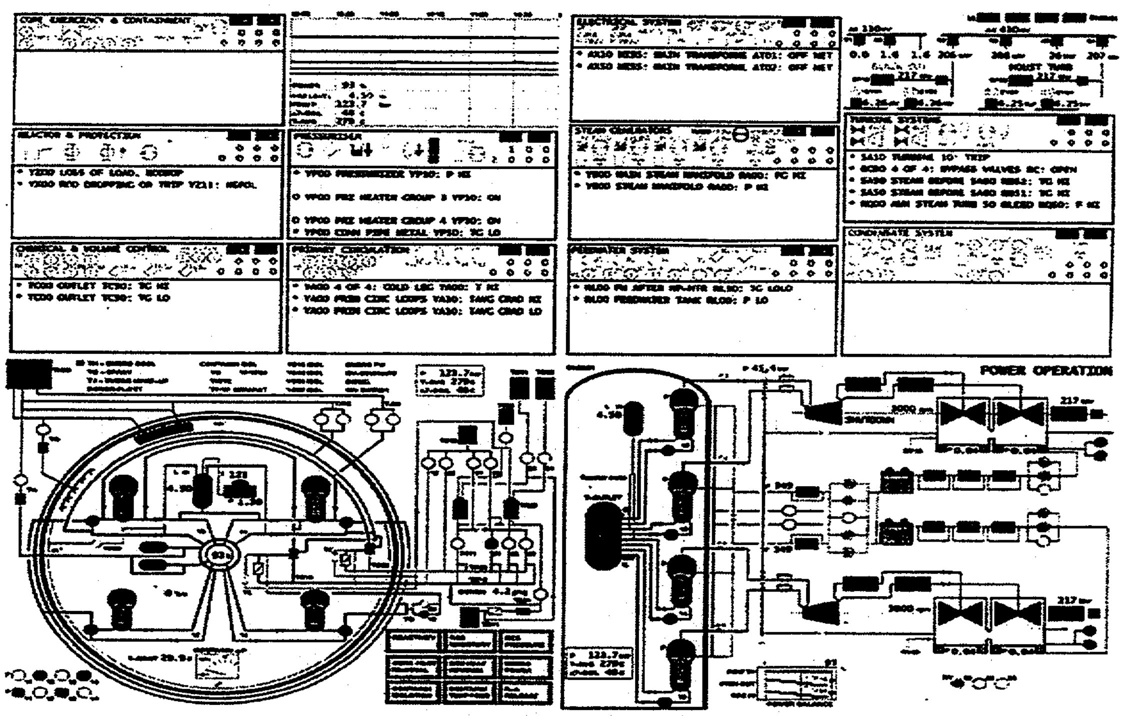
## INFORMATION OVERVIEW DISPLAYS

***Objective***

Investigate how computerised overview displays can be designed and presented to the operator in an integrated fashion.

***Results***

An overview display integrating information from different sources in the control room is supposed to cope with large amounts of data and existence of numerous computerised support systems. Such an integrated overview display combines different information and different ways of presenting this information, utilising modern graphical technology. A prototype of such an integrated information overview display has been made and integrated into the Halden Man-Machine Laboratory. This prototype consists of process information, alarm information and critical safety functions presented as an integrated display (see figure)




*Integrated information overview in HAMMLAB today*

***Main ref.***

C. Decurnex et al., HWR-451 (1996).

***Next Step***

Future work will be to integrate other sources of information in the display, e.g. information from a computerised procedure system, but also to investigate further the usability of such an integrated approach.

Attachment 3

# Executive Summary

## Achievements of the Halden Project Programme in the 1997-1999 Period: Fuels and Materials Research

October 1999

# OVERVIEW OF FUELS AND MATERIALS EXPERIMENTS IN THE 1997 - 1999 PROGRAM PERIOD

| IFA | fuel (f) clad (c) or material type and origin | # of rods / speci-mens | burnup MWd/kg oxide / fluence n/cm$^2$ ($x10^{20}$) | MEASUREMENTS | | | | | | | | | APPLICATIONS | | | | | | | | recent reports (HWR) | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | temperature | pressure | clad elongation | fuel elongation | gas flow | oxide thickness | crack length | clad diameter | ECP | thermal conductivity | fission gas release | densification, swelling | PCMI | clad creep | corrosion | IASCC | irr. ind. mat. changes | | |
| 503.1 | Electrostal (c,f), VVER | 6+6 | 19 / 25 | x | x | | x | | | | | | x | x | x | | | | | | 467,541 | VVER fuel performance |
| 503.2 | Electrostal (c,f), VVER | 6+6 | 0 | x | x | | x | | | | | | x | x | x | | | | | | | VVER fuel performance |
| 504 | KWU, BNFL | 4 | 75 | x | | | | x | | | | | x | x | | | | | | | 545 | interlinkage, FGR |
| 509 | additives, ENEA | | 33 | | | x | x | | | | | | | | x | x | | | | | 305 | additive fuel perform. |
| 515 | $UO_2$ / Gd, NFD | 4 | 60 | x | | | | | | | | | x | | | | | | | | 470,547 | non-abs. GD |
| 519 | $UO_2$, 7 / 17 µm grain size | 3 | 90 | | x | | | | | | | | | x | | | | | | | 548 | load follow, FGR |
| 530/D04 | SS / Argonne Nat. Lab. | 60 | 18 - 25 | | | | | | | | | | | | | | | | | x | Loen 99 | dry irradiation |
| 533 | HRP | 2 | 75 | x | | | | | | | | | x | x | | | | | | | 531 | TF re-inst. test |
| 534.13 | KWU(c)/GE(f), 8/22 µm gr. | 2/2 | 29 ... 38 | | x | x | | | | | | | | x | () | x | x | | | | 546 | grain size eff., FGR,PCI |
| 534.15 | KWU(c)/GE(f), 8/22 µm gr. | 2/2 | 52 ... 55 | | x | x | | | | | | | | x | () | x | x | | | | Loen 99 | grain size eff., FGR,PCI |
| 550 | sphere pack, PSI | 2 | 36 | | x | | | | | | | | | x | | | | | | | | Sphere-pack perform. |
| 562.6 | HRP | 3 | 90 | x | x | | | | | | | | x | x | | | | | | | 469 | conductivity degrad. |
| 578 | VTT, VVER 440 weld metal | 42 | 0 ... 0.2 | | | | | | | | | | | | | | | | | x | | embittlement/annealing |
| 585.4 | Zry-4, KWU | 1 | > 100 | | | | | | | | x | | | | | | | | | | 532 | cladding creep |
| 593.4 | Zry-4, low tin | | 245 fpd | | | | | | x | | | | | | | | | x | | | 533 | PWR clad corrosion |

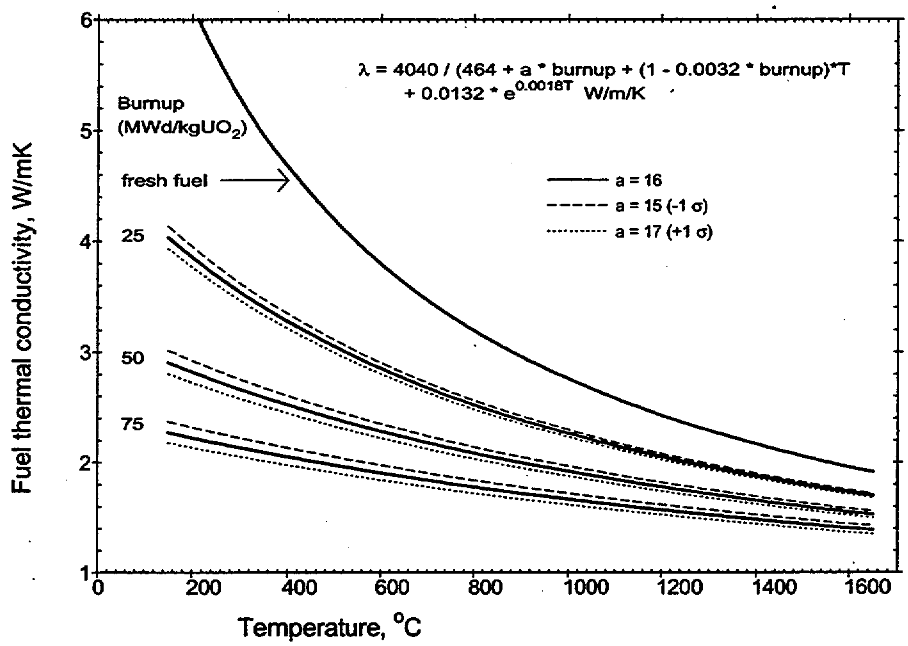
# OVERVIEW OF FUELS AND MATERIALS EXPERIMENTS IN THE 1997 - 1999 PROGRAM PERIOD

| IFA | fuel (f) clad (c) or material type and origin | # of rods / speci-mens | burnup MWd/ kg oxide / fluence n/cm$^2$ (x10$^{20}$) | MEASUREMENTS | | | | | | | | | APPLICATIONS | | | | | | | | recent reports (HWR) | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | temperature | pressure | clad elongation | fuel elongation | gas flow | oxide thickness | crack length | clad diameter | ECP | thermal conductivity | fission gas release | densification, swelling | PCMI | clad creep | corrosion | IASCC | irr. ind. mat. changes | | |
| 597.2 | BWR fuel / ABB | 2 | 59 | x | x | | | | | | | | x | x | | | | | | | 442,466 | re-inst. BWR |
| 597.3 | BWR fuel / ABB | 2 | 59 | x | x | | | | | | | | x | x | | | | | | | 543 | re-inst. BWR |
| 597.4 | MOX | 2 | 0 ... 10 | x | x | | | | | | | | x | x | () | | | | | | 551 | solid and hollow fuel |
| 610.1 | PWR Gösgen ( see 534) | 1 | 50 | x | () | x | | x | | | | | x | x | | x | x | | | | 544,545 | overpressure/lift-off |
| 610.2 | MOX / EDF-BN | 1 | 50 | x | () | x | | x | | | | | x | x | | x | x | | | | Loen 99 | overpresssure/lift-off |
| 611 | SS / GE | 4 | 0 | | | | | | | x | | | | | | | | | x | | | test active loading |
| 612 | SS (Oskarsh.) / GE, KWU | 4 | 8, 11, 90 | | | | | | | x | | x | | | | | | | x | | 556 | IASCC pre-irr. materials |
| 613.1-3 | HRP | 3 | | | | x | | | x | | | | | | | | | | | | 493,494 | dry-out, BWR cond. |
| 617 | see 638 / ABB, Framatome | 4 | | | | | | | | | x | | | | | | x | | | | | creep of modern clad |
| 618 | 304, 316, 347 / Criepi,KWU | 12-34 | 0 ... 16 | | | x | | | | | | | | | | | | | x | | Loen 99 | IASCC, time to failure |
| 629.1 | MOX / EDF-BN | 2 | 23 ... 27 | x | x | x | | | | | | | x | x | () | x | | | | | Loen 99 | re-instr. PWR fuel |
| 634.1-4 | HRP | 2 | | | x | | | | | | | | | | | | | | | | 534 | steam reaction kinetics |
| 636 | 8% Gd/$UO_2$, ENUSA | 6+3 | 0 ... 5 | x | x | x | x | | | | | | x | x | x | x | | | | | Loen 99 | natural Gd |
| 638 | Zry-4, Zirlo, M4, M5, low tin | 3*4 | 20 / 50 | | | | | | x | | | | | | | | | x | | | 566 | corrosion, modern clad |
| 639 | SS 347, 316 NG, 304 | 4 | 9,15,90 | | | | | | | x | | x | | | | | | | x | | | IASCC pre-irr. materials |
| 648 | MOX / EDF | 2 | 50 | x | x | | | | | | | | x | x | | | | | | | | burnup extension |

# THERMAL CONDUCTIVITY OF $UO_2$

***Objective*** To determine the thermal conductivity of $UO_2$ fuel and the rate of thermal conductivity degradation with accumulating burnup.

***Results*** Analysis of several in-pile experiments in the Halden database up to burnups of 75 MWd/kg$UO_2$. Results converge to suggest an update of the correlation for fuel thermal conductivity degradation with burnup.




*Thermal conductivity of $UO_2$ fuel at different burnups as derived from measured Halden in-pile data, together with 1σ uncertainties. Fuel density 95% of the theoretical density.*

***Experimental basis*** The test sample analysed includes fuel rods pre-irradiated to high burnups in commercial power reactors (IFA-597.2, IFA-597.3, IFA-610.1), re-instrumented rods pre-irradiated in the HBWR (IFA-533.2), as well as tests reaching intermediate burnups (IFA-552.1, IFA-432). Fuel rod instrumentation included both expansion thermometers and thermocouples.

***Main ref.*** Thermal properties of high burnup UO2 fuel rods. HWR-599, M. Pihlatie, April 1999.

***Remaining work*** Some further assessments on existing measurement data. Irradiation of re-instrumented fuel rods will continue e.g. in IFA-629 and IFA-610.

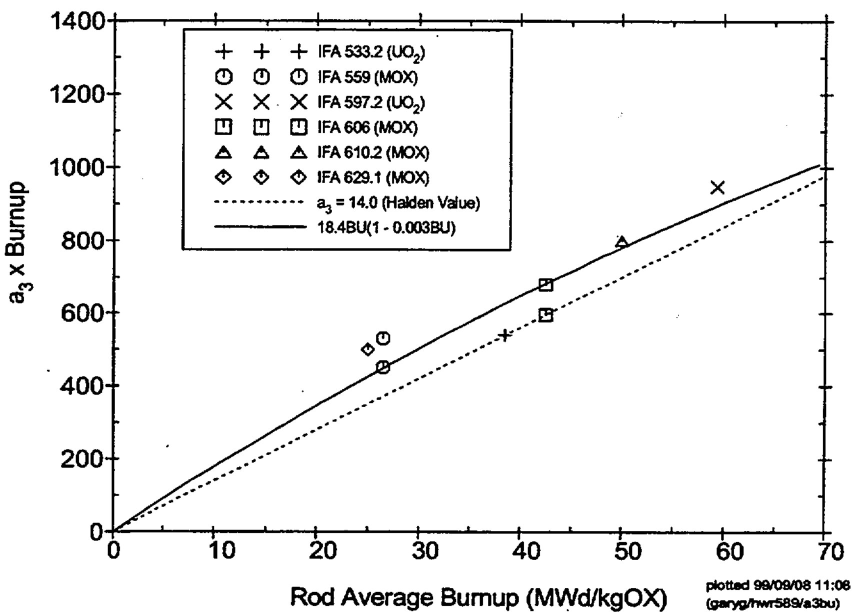
## THERMAL PROPERTIES OF MOX FUEL

***Objective***

Review currently available MOX thermal performance data.

***Results***

Start-of-life thermal data indicates that the standard $UO_2$ fuel thermal conductivity relationship is an adequate representation for the thermal conductivity of MOX fuel with the addition of a conductivity scaling parameter, $K_{mox}$. A value of $K_{mox}$= 0.92 provides a good fit to the data. The thermal conductivity degradation in MOX and $UO_2$ is qualitatively and quantitatively comparable. Evidence points to a possible saturation effect in the fuel thermal conductivity degradation at high burnup.




*Principal conductivity degradation parameter, $a_3$, multiplied by fuel burnup.*

***Experimental basis***

Data from IFAs-226, 559, 597.4, 533.2, 597.2, 606, 610.2 & 629.1.

***Main ref.***

Gary A Gates et al., Thermal Performance of MOX Fuel, HWR-589. Paper presented at the Leon meeting, May 1999.

***Remaining work***

The analysis will be continued as more MOX data become available.

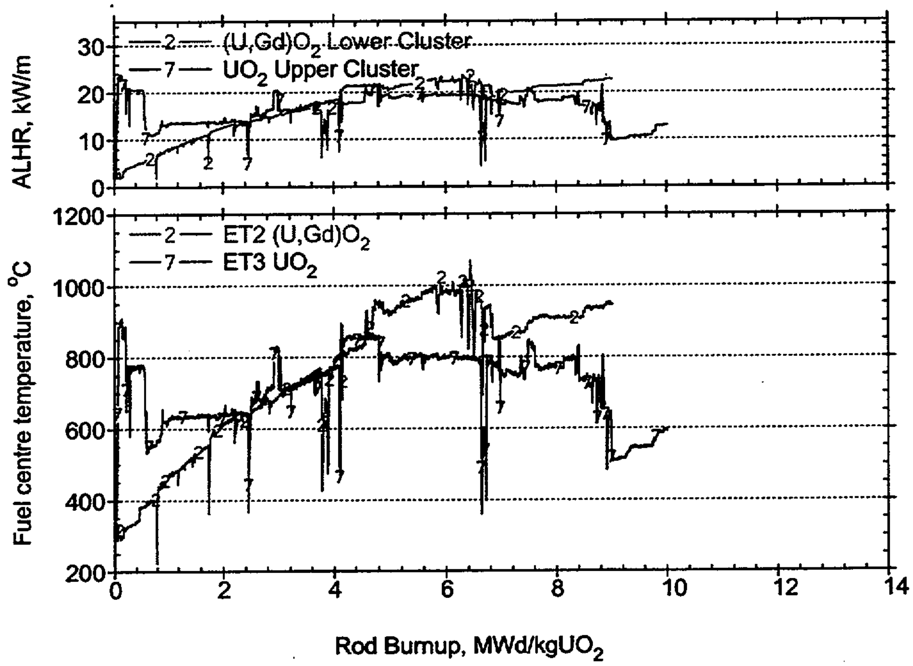
## GD FUEL BEHAVIOUR

***Objective***

Increase the current data base on thermo-mechanical behaviour of 8 wt% gadolinia doped fuel with respect to standard $UO_2$, with focus on comparative fuel thermal performance, fission gas release, fuel densification and swelling, fission gas release and PCMI.

***Results***

Fuel centre temperature and power in gadolinia doped fuel increase monotonously as a consequence of the absorbing gadolinia burn out. Fuel densification in gadolinia doped fuel is lower than in $UO_2$.




*Fuel centre temperature of $UO_2$ (ET3) and $(U,Gd)O_2$ fuel (ET2), together with the rod power experienced*

***Experimental basis***

Nine fuel rods placed in two clusters. The lower cluster consists of 6 matched $UO_2$/$(U,Gd)O_2$ rods, and in the upper cluster, two are $UO_2$ and one $(U,Gd)O_2$; IFA-636.1.
The doped fuel used is the same as used currently in commercial NNP, including the absorbing Gd isotopes, $^{155}Gd$ and $^{157}Gd$. Both fresh fuel and cladding were delivered by ENUSA, Spain.

***Main ref.***

M.T. del Barrio, HWR-600. Paper presented at the Loen Meeting, 1999. Halden Status report, 1058.

***Remaining work***

New power calibration to determine whether the power distribution between lower and upper cluster is correct. Increase the rig power in order to be able to study fission gas release.

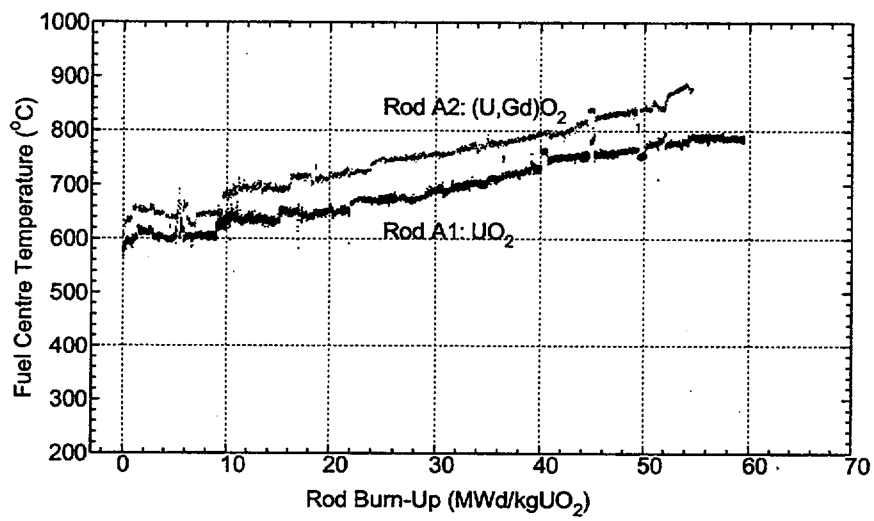
## THERMAL PROPERTIES OF GD FUEL

***Objective***

Compare thermal behaviour of $UO_2$ and $(U,Gd)O_2$ fuel at high burn-up by means of in-pile temperature measurements.

***Results***

$UO_2$ and $(U,Gd)O_2$ fuel show a parallel increase of fuel centre temperature with burnup, giving evidence of thermal conductivity degradation. The $(U,Gd)O_2$ fuel has higher temperatures corresponding to a 15 - 20% difference in thermal conductivity. Scram data confirm the steady-state measurements.




*Fuel centre temperatures normalised to 18 kW/m. Gd-doped fuel shows ~80 °C higher temperature at this heat-rate. The development with burnup indicates a thermal conductivity degradation which is about the same in both kinds of fuels.*

***Experimental basis***

Small diameter fuel rods for rapid burnup accumulation equipped with temperature sensors. Pairs of rods with 11.5% w/o U-235 enriched fuel and 13% U-235 and 8% non-burnable $Gd_2O_3$ are being irradiated in IFA-515 loading 10.

***Main ref.***

T. Tverberg et al. HWR-547. Paper presented at the Lillehammer meeting, March 1998.

***Remaining work***

Irradiation will continue to reach a burnup of > 80 $MWd/kgUO_2$.

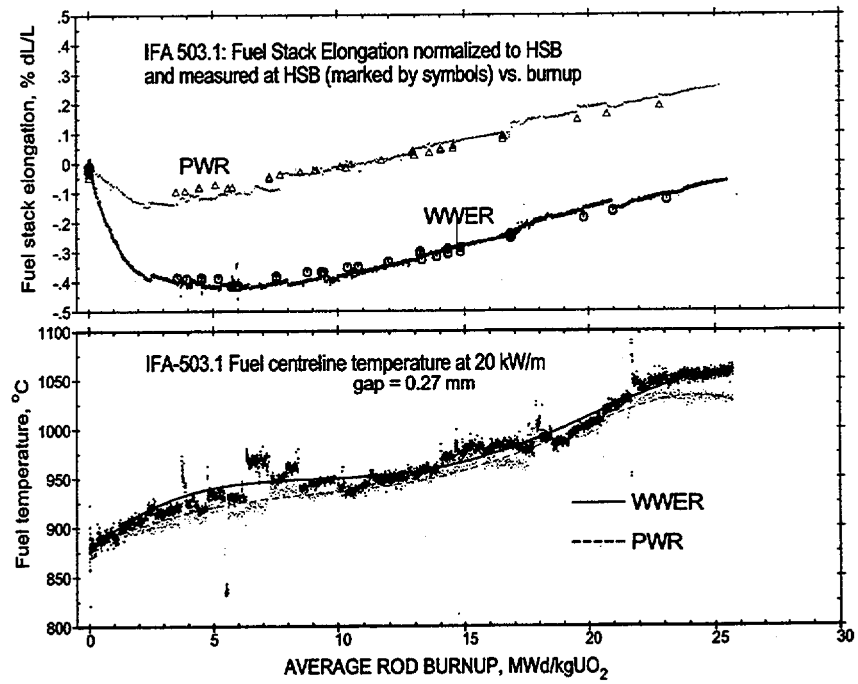
## COMPARISON BETWEEN WWER AND PWR FUEL BEHAVIOUR

***Objective***

Generate a representative and comparative data base of different WWER-440 fuel types produced by MSZ Electrostal (Russia) and typical PWR fuel.

***Results***

Densification, swelling, FGR and thermal behaviour of the standard WWER fuel and PWR fuel type were compared together to a burnup of 25 MWd/kg $UO_2$. The data analysis showed insignificant differences in the behaviour of the two types of fuel with the exception of densification. Modified WWER fuel types being tested in the second loading have shown improved densification properties.




*Comparison of fuel stack elongation and fuel temperatures measured in the WWER and PWR fuel types as a function of burnup.*

***Experimental basis***

Six pairs of the two types of the rods were incorporated into the lower and upper cluster in IFA-503.1. The rods were equipped with extension thermometers, pressure transducers and fuel stack elongation detectors.

***Main ref.***

B.Volkov et al., HWR-590, HWR-610. Paper presented at the Loen meeting, May 1999.

***Remaining work***

Post irradiation examinations of selected rods from IFA-503.1 with aim to determine fission gas release. Irradiation of the modified WWER fuel types in IFA-503.2 for the two or three years in order to study fuel swelling, thermal behaviour and FGR. The second loading in IFA-503.2 of the test fuel rods with three modified WWER fuel types and PWR fuel is designed and instrumented similar to the first loading.

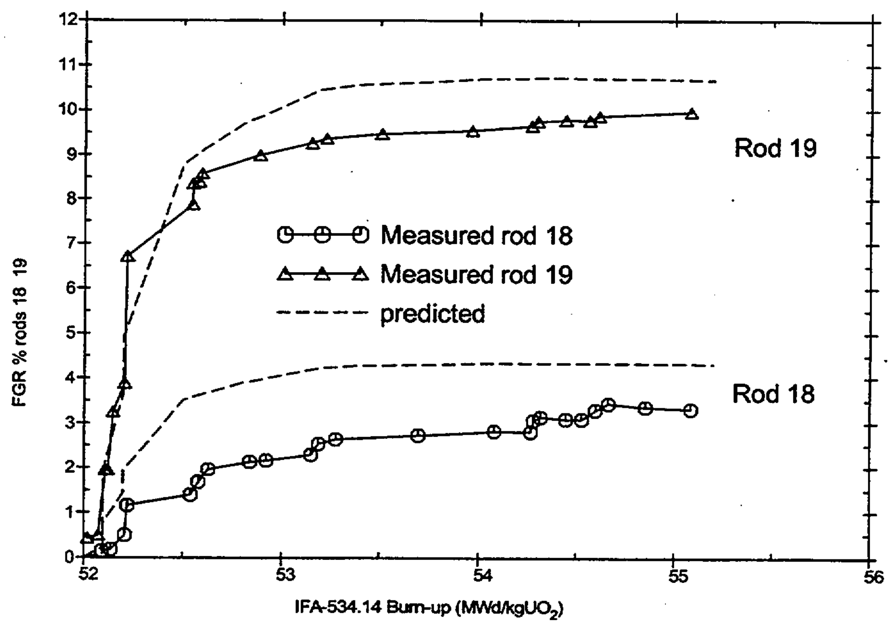
# THE INFLUENCE OF GRAIN SIZE ON FISSION GAS RELEASE FROM UO2

***Objective***

To assess the data demonstrating the effect of fuel grain size on the amount of fission gas release (FGR) from $UO_2$ fuel.

***Results***

Experimental data from Halden and other experiments were analysed using a simple FGR code benchmarked against the Halden 1% threshold including the new high burn-up data. It was demonstrated that in most cases the effect of increasing grain size reduced the FGR according to the theoretical diffusion model. With the help of the model, it is shown that this is most effective at modest powers and FGR ≤10%, and at higher powers and high values of release, an increase in grain size is less effective. Analysis of experiments where FGR from different grain size material is compared shows that particularly at high power, FGR may be affected by grain growth of the smaller grain material, thus influencing the conclusions regarding the benefits or otherwise of advanced fuels.




*Measured and predicted FGR from 8.5 and 22 μm $UO_2$ in IFA-534.14.*

***Experimental basis***

Data from IFA-519.9, IFA-534.14, Studsvik OVER RAMP and IFA-566.

***Main ref.***

J. A. Turnbull: "An Assessment of Fission Gas Release and Effect of Microstructure at High Burn-up", HWR-604. Paper presented at the EHPG, Loen 1999.

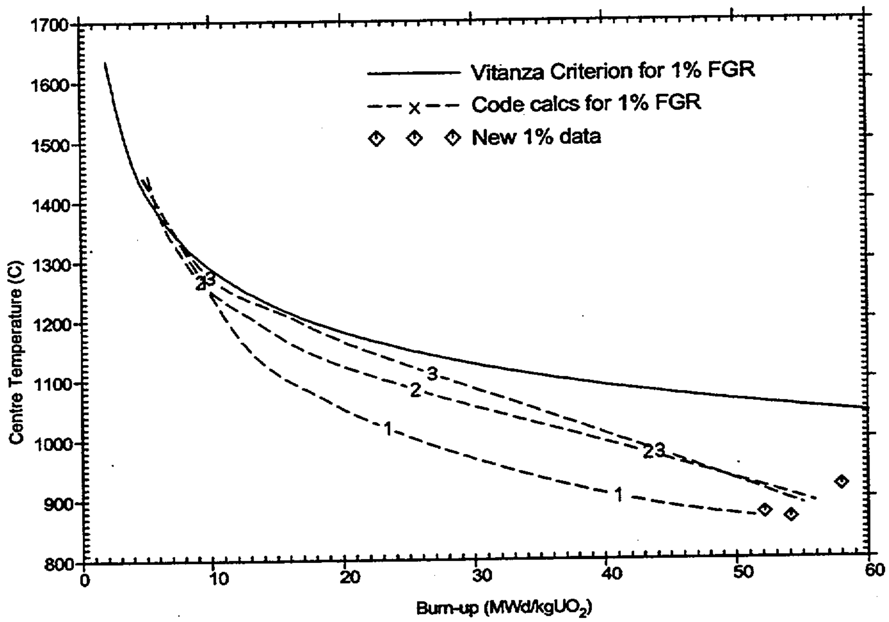
***Remaining work***

Although the effect of grain size on FGR in pure $UO_2$ is conclusive, the effect of large grain material manufactured with the help of additives requires further attention.

# THE ONSET OF FISSION GAS RELEASE FROM UO2 AT HIGH BURN-UP

***Objective*** To modify the Halden release threshold to accommodate new high burn-up data.

***Results*** The experimentally derived Halden threshold for 1% fission gas release (FGR) expressed in terms of fuel centre temperature and burn-up provides an excellent criterion for the onset of FGR for fuel rods with a burn-up ≤30 MWd/kg. However, it significantly overestimates the temperature for new data obtained from rods of 52-58 MWd/kg, where release starts at around 900°C compared to 1050°C predicted. The original criterion can be reproduced theoretically in terms of diffusion and grain boundary re-solution of single gas atoms. By introducing empirical modifications such that the diffusion coefficient increased and the re-solution parameter decreased with burn-up, a new criterion was derived which accommodated the new high burn-up data whilst leaving the low burn-up agreement unchanged.




*Comparison ofcriteria and new high burn-up data*

***Experimental basis*** Data from IFA-569 rod 8, IFA-534.14 and IFA-610 rod 1.

***Main ref.*** J. A. Turnbull: "An Assessment of Fission Gas Release and Effect of Microstructure at High Burn-up", HWR-604. Paper presented at the EHPG, Loen 1999.

***Remaining work*** Refinement of the criterion is required for medium to high burn-up, i.e. in the range 30-60 MWd/kg, hence further experimental data are required at these and higher burn-up.

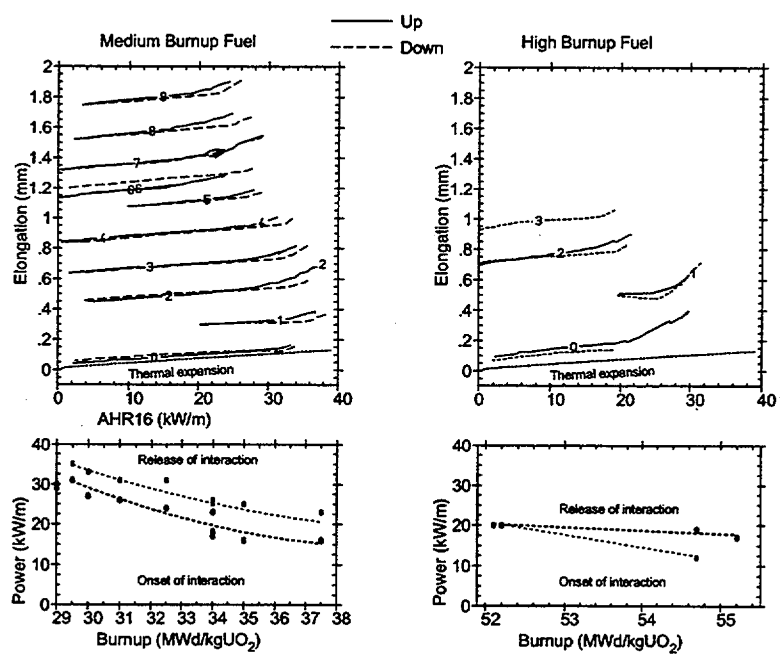
# PCMI BEHAVIOUR OF HIGH BURNUP FUEL

***Objective***

Investigate the development of the onset of PCMI with burnup, axial ratchetting as consequence of shut-down/start-up operation, and relaxation behaviour during hold times at high powers for medium to high-burnup $UO_2$ fuel.

***Results***

During early start-ups there is stronger PCMI than for later ramps, followed by relaxation.
Later shut-down/start-up sequences show ratchetting effects accompanied by relaxation.
The contact point for axial strain is at the pellet periphery, even for high burnup fuel.
Burnup dependent observations:
The high burnup fuel elongation curves (normalised to a constant power) show an increase with burnup that reflects fuel swelling.




*Development of point of onset/release of pellet-cladding interaction for medium (left) and high burnup (right) fuel.*

***Experimental basis***

5 $UO_2$-rods equipped with cladding elongation gauges (EC), in IFAs 534.13, 534.14 and 597.3, with burnup ranging from ~30 to ~60 $MWd/kgUO_2$ were analysed. The development with burnup of the point of onset/release of pellet cladding interaction was analysed and compared with typical solid fission product fuel swelling rates.

***Main ref.***

T. Tverberg, HWR-619. Paper presented at the Loen meeting in May, 1999.

***Remaining work***

Extend the investigation to higher burnups and also to include MOX fuel.

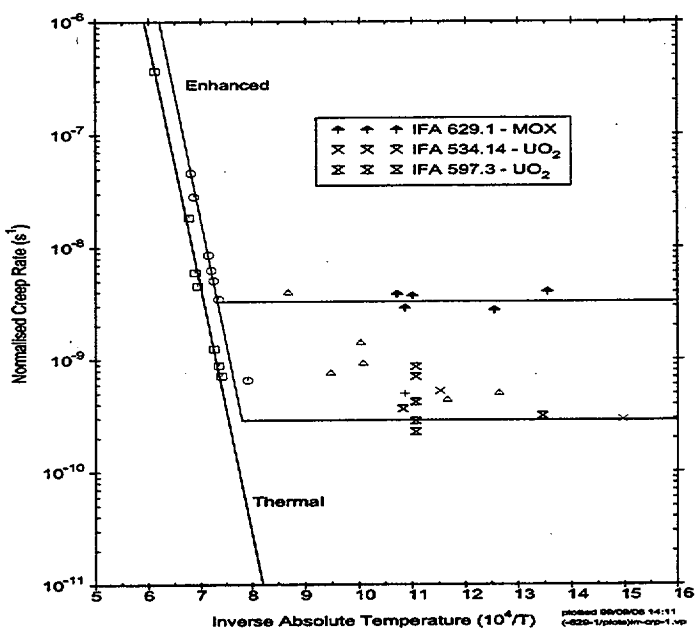
# PCMI BEHAVIOUR OF MOX FUEL

***Objective***

Determine the PCMI behaviour of MOX fuel from in-pile clad elongation data.

***Results***

Analysis of clad elongation data from a commercially irradiated MIMAS-MOX fuel rod reinstrumented with a centre-line thermocouple and clad extensometer reveals that the clad elongations during up-ramps are driven by fuel-clad contact and controlled by the thermal expansion of the fuel in the pellet land region. Subsequent relaxation of the cladding strains are possibly associated with enhanced irradiation creep in the MOX fuel and comparisons indicate an enhancement factor of eight over $UO_2$. This is possibly the origin of the good PCI resistance of MOX fuel.




*Creep data normalised to a stress of 24 MPa and a fission rate of $1.2x10^{19}f/m^3/s$.*

***Experimental basis***

Data from the re-irradiation of commercially irradiated MIMAS-MOX in IFA-629.1, fuel burnup ~25 MWd/kgMOX.

***Main ref.***

Rodney J White, The Re-irradiation of MIMAS-MOX Fuel in IFA-629.1, HWR-586, Paper presented at the Leon meeting, May 1999.

***Remaining work***

Continue PCMI investigation with further loadings of higher burnup MOX fuel in IFA-629.

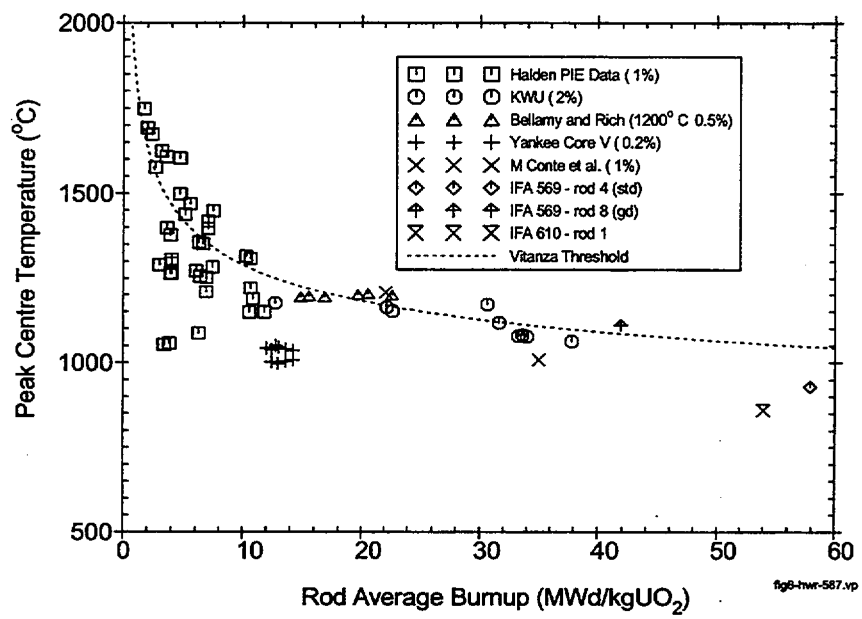
## FISSION PRODUCT RELEASE EVALUATION BY γ-SPECTROMETRY

***Objective***

Investigate the release of radioactive fission gases from the Halden gas-flow rigs, IFAs-504 and 610.1

***Results***

The release to birth rate ratios from IFA-504 measurements indicate that the majority of fission products are produced by $^{235}U$ fission. This is in contrast to the data obtained from the reinstrumented rods in IFA-610.1 which show that the majority of fission products are produced by $^{239}Pu$ fissions. Analysis of the results for IFA-610.1 indicates that the empirical fission gas release threshold may be lower than expected at high burnups.




*Halden empirical fission gas release threshold with the addition of IFA-610.1 data point.*

***Experimental basis***

Fuel rods equipped with gas-flow lines in IFAs-504 and 610.1.

***Main ref.***

Gary A Gates et al., Recent Radioactive Fission Gas Release Measurements From IFA-504 and IFA-610.1, HWR-587, Paper presented at the Loen Meeting, May 1999.

***Remaining work***

Radioactive fission gas release measurements will continue on IFA-504 and the new loadings of IFA-610 to obtain data at higher burnups.

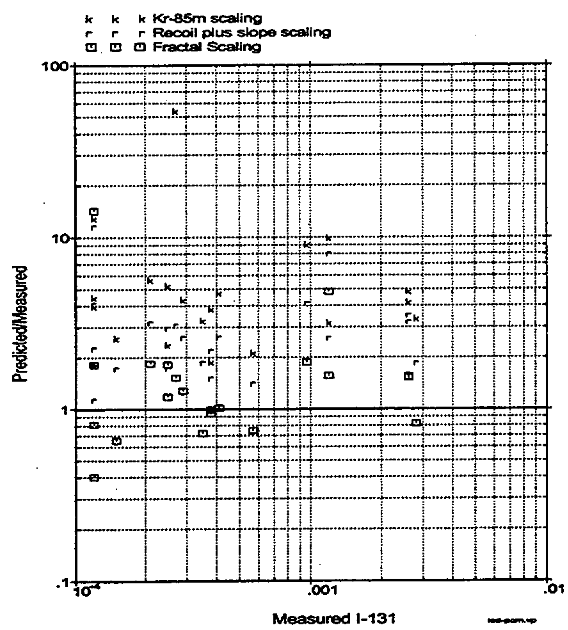
## IODINE RELEASE ASSESSMENTS

***Objective***

A re-examination of the unstable fission gas release data from the gas flow rigs using fractal analysis.

***Results***

The surface of $UO_2$ shows a strong degree of self-similarity defined by fractal coefficients between -0.5 and -1.0. Using fractal analysis it has been shown that the apparent anomalous behaviour of short-lived and long-lived fission products can be described by a single two-term diffusion coefficient. The third term, previously used in the analysis for the long-lived fission products, does not exist and is an artifact of an earlier analysis. The fractal model also provides an explanation for the anomalously low values of $^{131}I$ release measured in the gas flow rigs.




*Comparison of predicted $^{131}I$ releases compared to measured values for the constant surface to volume methods and the fractal scaling approach. The fractal scaling method avoids the large overprediction inherent in alternative methods.*

***Experimental basis***

Data from the gas-flow rigs IFA-504, 558 and 563.

***Main ref.***

Rodney J White, The Fractal Nature of the Surface of Uranium Dioxide: A Resolution of the Short-Lived/Stable Gas Release Dichotomy, HWR-550, Paper presented at the Lillehammer meeting, March 1998.

***Remaining work***

Completed.

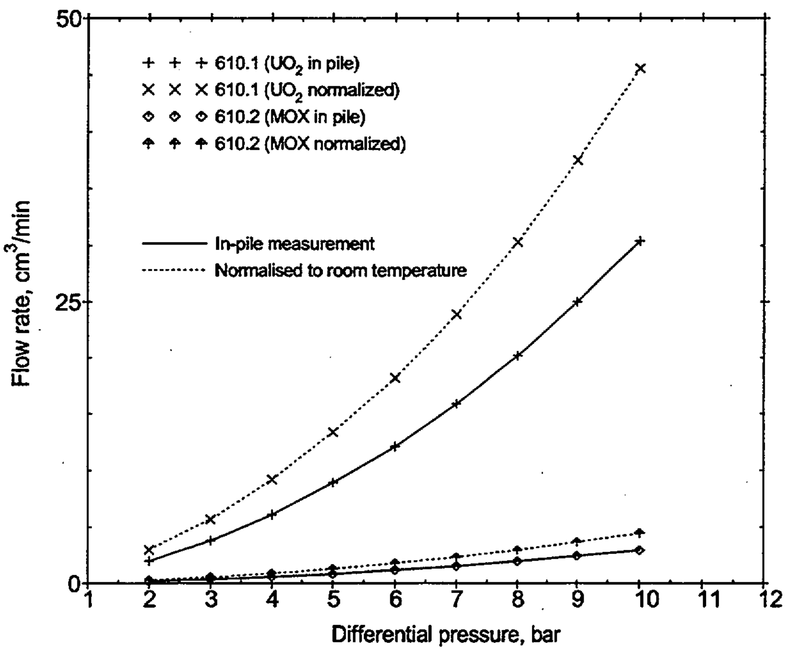
## AXIAL GAS TRANSPORT IN OPERATING FUEL RODS

***Objective***

Quantify axial gas transport in fuel rods as function of fuel type, operating conditions and burn-up. Based on these measurements, axial flow resistance, expressed as hydraulic diameter is calculated.

***Results***

Data on $UO_2$ rods with burn-up of about 72MWd/kg$UO_2$, and MOX rods with burn-up of 50MWd/kg. The results show an impeded axial gas communication, especially for MOX fuel at high burnup.




*Axial gas flow at hot standby in a $UO_2$ rod and a MOX rod of similar geometry and burn-up (about 50MWd/kg).*

***Experimental basis***

Fuel rods equipped with gas lines in both ends, connected to an external gas flow control system. Measurements of flow rates and pressures.

***Main ref.***

K. Bråten, Y. Minagawa: HWR-617. Paper presented at the Loen meeting, May 1999.

***Remaining work***

Continue irradiation to higher burn-ups. New experiments with various fuel types.

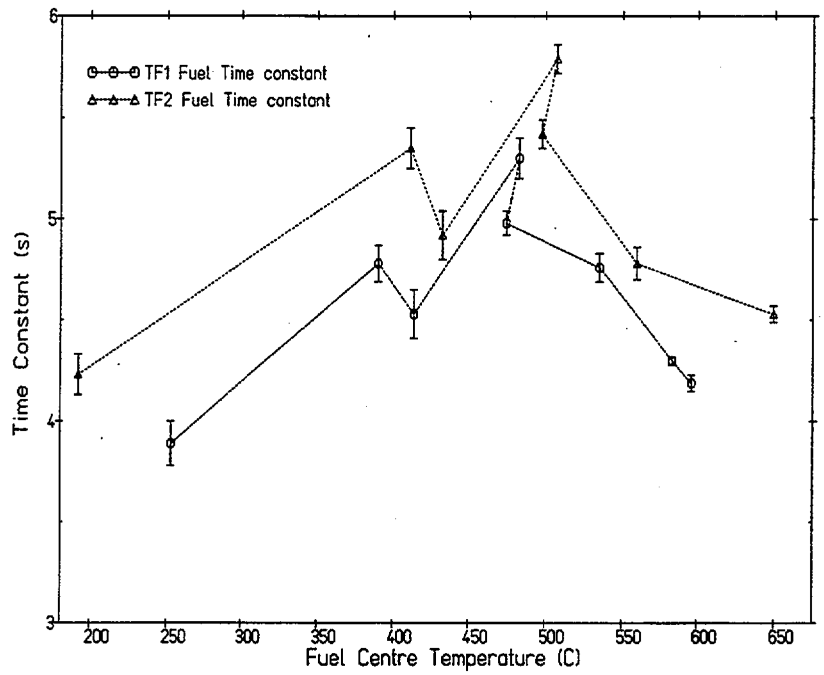
## NOISE ANALYSIS

***Objective***

Study of fuel thermal performance, fission gas release and PCMI by analysis of noise data.

***Results***

Measurements on IFA 610.1 with He and Ar fill gases have shown the effect of fill gas on fuel time constant.
Measurements on IFA 629.1 have demonstrated the use of noise analysis in detecting gas release.
Measurements made in the March 1999 start-up show effects of pellet cracking/relocation and PCMI in IFAs 503.2 and 648.1




*Noise results from IFA 648.1 showing the effect of fuel-clad gap closure at temperatures of about 500 °C*

***Theoretical basis***

Tools have been developed to analyse noise data by a range of methods:
1) Calculation of fuel and thermocouple time constants.
2) Lag time calculation from cross-correlations.
3) Coherency spectra for PCMI assessment.
4) Gain and Phase spectra for frequency space analysis

***Main refs.***

P. Brohan, HWR-601 & HWR-607, Papers presented at the Loen meeting, May 1999.

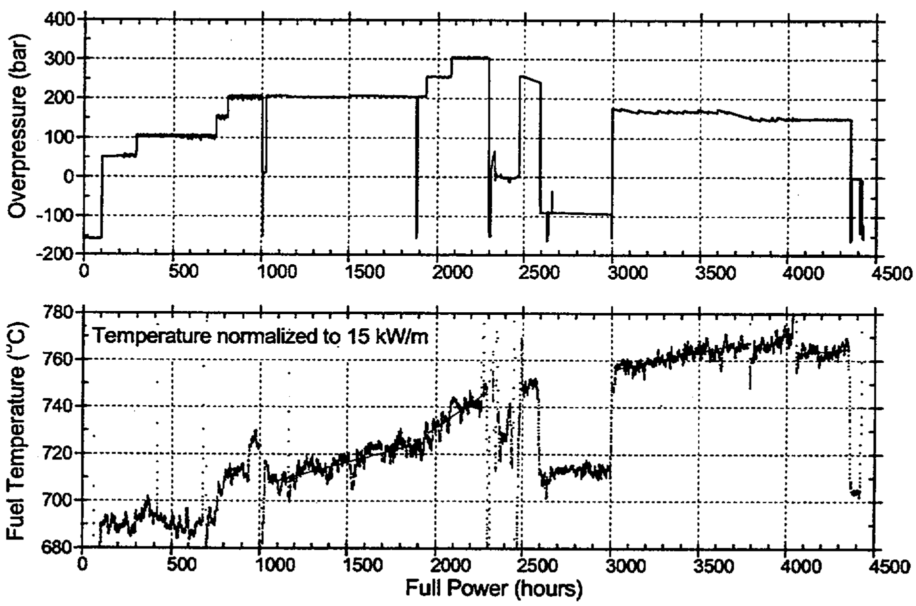
***Remaining work***

Further measurements on IFAs 648.1 and 503.2 will provide data on burn-up effects.
Possibilities for noise measurements on future experiments will be assessed while the experiments are being planned and executed.

# THE LIFT-OFF EXPERIMENT WITH $UO_2$ FUEL ROD IN IFA-610.1

***Objective*** To investigate the integral rod behviour during overpressure operation and to assess the overpressure limit for lift-off occurrence.

***Results*** Each increase in pressure was accompanied by an increase in the measured temperature, as the fuel clad gap progessively enlarged by clad distension. The onset of clad lift-off appeared for an overpressure of +150 bar.
The rate of fission product swelling is estimated to be around 0.8% / (10 MWd/kgU).




*Overpressure history and temperature response, IFA-610.1*

***Experimental basis*** High burnup PWR $UO_2$ fuel rod base irradiated in the Gösgen reactor up to 52 MWd/kg$UO_2$.
Refabricated and reinstrumented with a fuel centreline thermocouple and a cladding extensometer.

***Main ref.*** S. Béguin, HWR-544, January 1998;
K. Bråten, R. V. Sandberg, HWR-545, December 1997;
S. Béguin, HWR-572, February 1998;
G. Gates, R. J. White, K. Bråten, HWR-587, March 1999.

***Remaining work*** Definitively unloaded in May 1998.
Post irradiation examinations in 1999: clad diameter, corrosion layer, fuel microstructure, diameter measurements on the spare Gösgen rods for clad creep assessment.

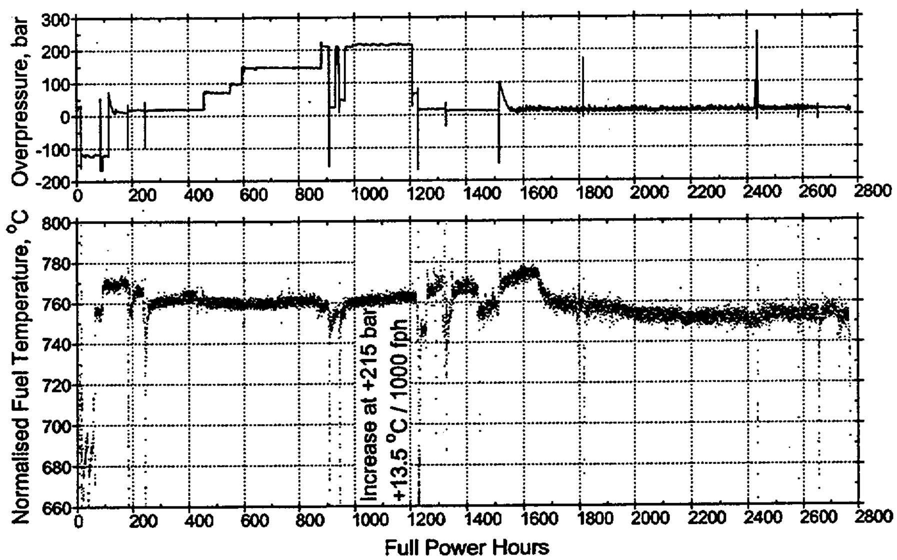
## THE LIFT-OFF EXPERIMENT WITH MOX FUEL ROD IN IFA-610.2

***Objective*** To investigate the integral rod behviour during overpressure operation and to assess the overpressure limit for lift-off occurrence.

***Results*** The normalized temperature remained constant during the main part of the overpressure experiment. It does not imply there are no elastic or creep deformations of the cladding. It would rather suggest that the fuel stack and the cladding were bounded together (the fuel fragments follow the cladding).

However, a slight and regular increase in the normalised temperature can be pointed out during the last stage of the experiment at a constant overpressure of +215 bar.

The rate of fission product swelling is estimated to be around 0.6% / (10 MWd/kgU).




*Overpressure history and temperature response, IFA-610.2*

***Experimental basis*** High burnup PWR MOX fuel rod base irradiated in the Gravelines 4 reactor up to 50 MWd/kgMOX.

Refabricated and reinstrumented with a fuel centreline thermocouple and a cladding extensometer.

***Main ref.*** S. Béguin, HWR-603, April 1999 ;

C. Jean, E. Muller, HWR-???, October 1999.

***Remaining work*** Definitively unloaded in May 1999.

Post irradiation examinations in 2000.

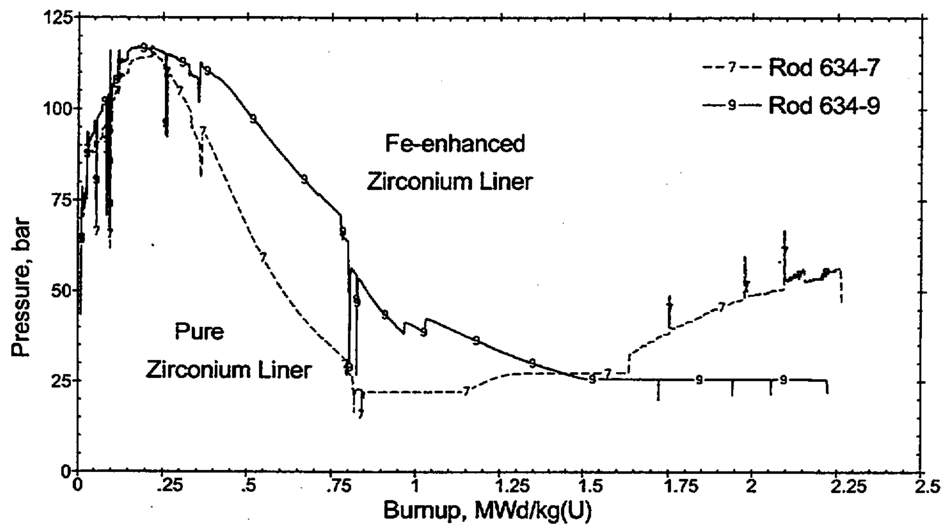
## STEAM REACTION KINETICS IN OPERATING RODS

***Objective***

To study the reaction kinetics of oxidation and hydriding of fuel and cladding in fuel rods after a rod failure.

***Results***

The rod pressure increased rapidly during the first few power days and reached a peak whose value depended on the linear heat rate. Since the pressure behaviour reflects the reaction kinetics in the rods a clear difference can be expected for different cladding materials and a dependence on linear heat rate is seen.




*Pressure development in fuel rods with various inner liner materials*

***Experimental basis***

Four loadings have been carried out for IFA-634.
Each test contained two rods of similar design fabricated from unirradiated Zircaloy-2 material with various inner liners and filled with a known amount of water to simulate rod failures. The tests were operated in a fully insulated pressure flask under BWR conditions. The rod internal pressure with time was measured during operation.

***Main ref.***

Y. Broy, HWR-602. Paper presented at the Loen Meeting, 1999.

***Remaining work***

Non-destructive and destructive Post Irradiation Examinations will be done to clarify the different influences on the reaction kinetics of cladding, liner and fuel as well as the reason for the pressure increase in rod 7 (failure?).

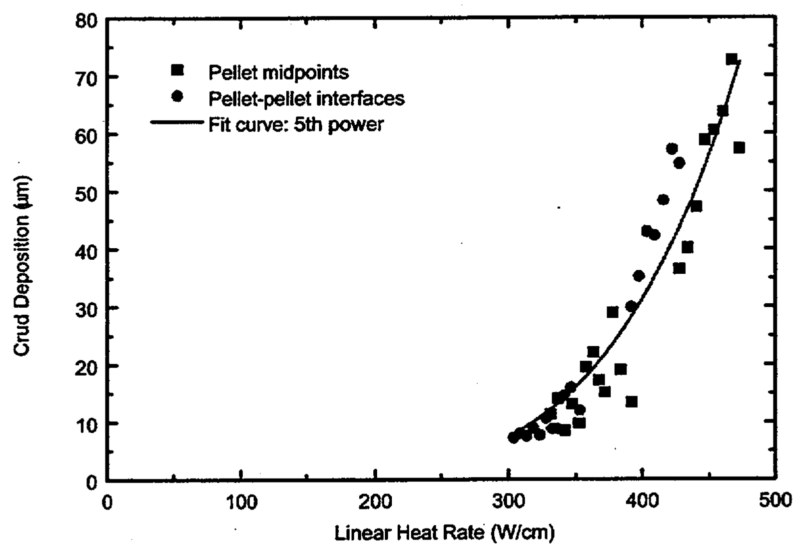
## FUEL ROD CRUD DEPOSITION, EFFECTS OF HEAT LOAD

***Objective***

Estimate the effects of heat load on the crud deposition onto fuel rods under zinc water chemistry, for heat load variations both global and local within a pellet length.

***Results***

The dependence of crud deposition rate with heat flux becomes very strong after the onset of nucleate boiling. Locally, crud deposition is smaller in the regions around the pellet-pellet interfaces, due to the depressions in heat flux.




*Crud deposition vs. linear heat rate, showing local variations within pellet lengths.*

***Experimental basis***

Fuel rods equipped with diameter-gauges to allow in-core measurements, in test rig IFA-585.1, under water chemistry with zinc injected in the form of spikes.

***Main ref.***

F. Barrera, HWR-618. Paper presented at the Loen meeting, May 1999.

***Remaining work***

Study of possible crud deposition in different coolant flow and chemistry conditions.

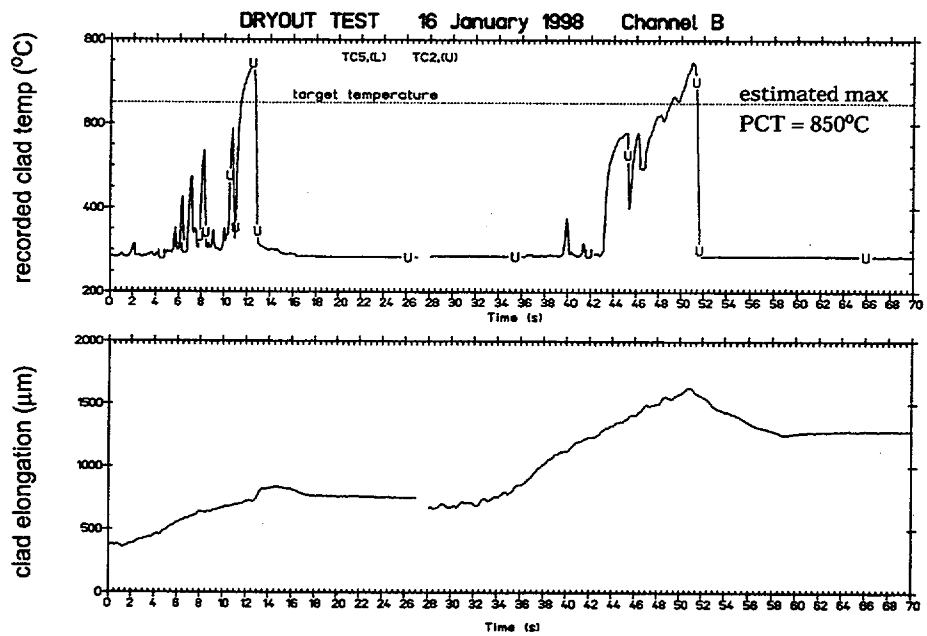
## SHORT-TERM DRY-OUT (IN-PILE)

***Objective***

To assess the consequences of exposing fuel rods to short-term dry-out in terms of the integrity of the fuel under continued operation.

***Results***

Fuel rod integrity was shown to be maintained for up to a month of operation following severe dry-out exposure which induced clad surface temperatures of 950-1200°C in the upper regions of the rods. More benign dry-out events were also achieved inducing clad surface temperatures in the range 750-850°C.




*Dry-out exposure conditions for a pre-irradiated PWR fuel segment.*

***Experimental basis***

Eight rods, including fresh and pre-irradiated PWR and BWR segments equipped with surface thermocouples and clad elongation detectors, were exposed individually to dry-out events induced by reducing coolant flow in one of the three test channels in IFA-613.

***Main ref.***

M. A. McGrath, B. C. Oberländer Myklebust et al., paper presented at the Loen meeting, May 1999.

***Remaining work***

All in-pile testing is complete.

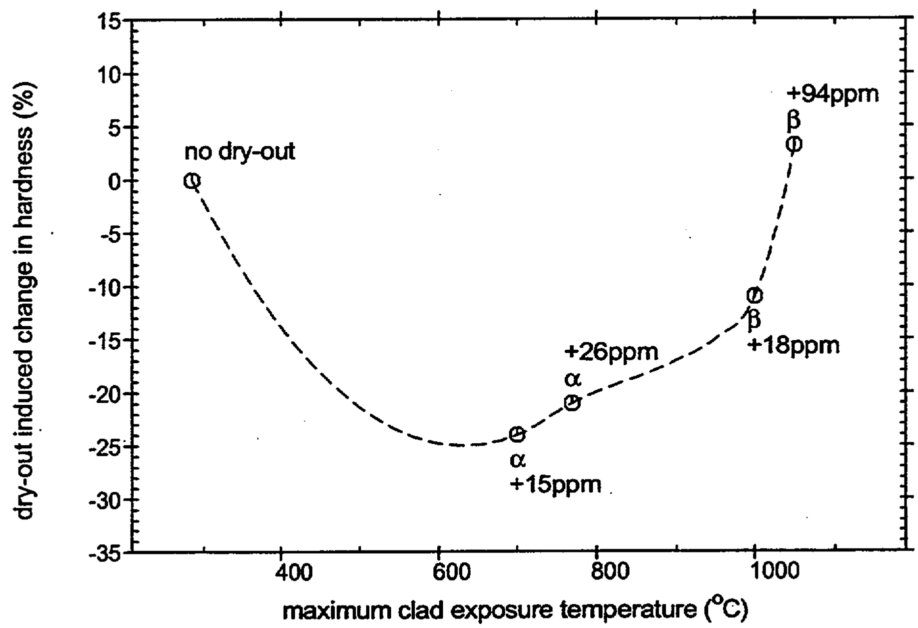
## SHORT-TERM DRY-OUT (PIE RESULTS)

***Objective***

To assess changes in the microstructural and mechanical properties of both fuel and clad of rods exposed to short-term in-pile dry-out.

***Results***

Significant improvements in clad ductility resulted from in-pile dry-out which raised the clad temperature to between 550 and 800°C, resulting in annealing with minor $H_2$ pick-up. Exposure >1000°C, resulting in β-phase formation and significant $H_2$ pick-up, was deleterious to clad properties. The fuel was unaffected.




*The effect of in-pile dry-out on pre-irradiated clad hardness as a function of exposure temperature, $H_2$ pick-up and clad structure.*

***Experimental basis***

Rods exposed to in-pile dry-out in IFA-613 were subjected to PIE consisting of: visual inspection, profilometry, neutron radiography, rod puncturing with fission gas release analysis, ceramography, metallography, hydrogen content determination, hardness and tensile testing.

***Main ref.***

M. A. McGrath, B. C. Oberländer Myklebust et al., paper presented at the Loen meeting, May 1999.

***Remaining work***

Additional PIE of the transition regions of the severely exposed rods, where the clad surface temperature during dry-out would have been in the range 500-750°C.

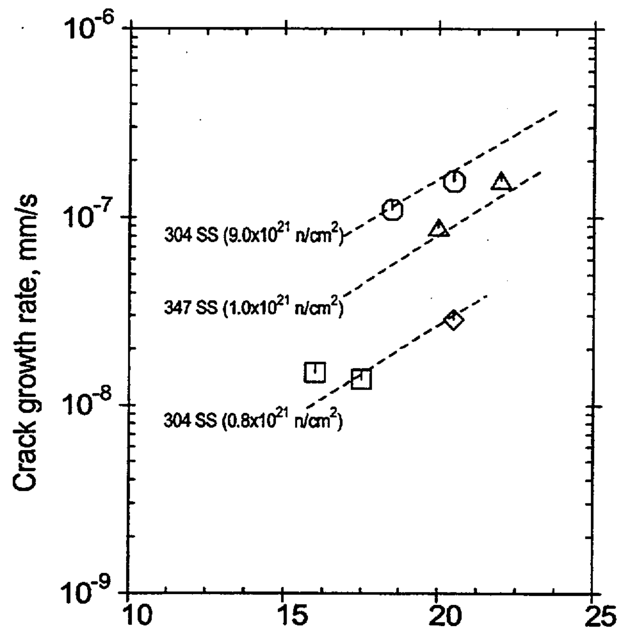
## FLUENCE EFFECT ON IRRADIATED CT SPECIMENS IN IASCC STUDIES

***Objective***

Demonstrate the feasibility of using subsize Compact Tension (CT) specimens with Electron Beam (EB) welded inserts of irradiated material in crack growth studies and evaluate the effect of fluence on crack growth rates.

***Results***

Crack growth rates have been measured in CTs prepared with inserts of pre-irradiated 347 SS (with a fluence of $1.0 \times 10^{21}$ n/cm$^2$) and 304 SS with fluences of $0.8 \times 10^{21}$ and $9.0 \times 10^{21}$ n/cm$^2$. Highest rates of crack growth were measured for the highest fluence specimen, followed by the lower fluence specimens.




*Effect of fluence on crack growth rate in irradiated CT specimens*

***Experimental basis***

Four CTs were prepared with EB welded inserts of pre-irradiated material with a disc or a square geometry. The CTs were instrumented for crack growth monitoring with the DC potential drop method and fitted with bellows for on-line variation of applied stress intensity in IFA-612. Crack growth rates were measured for each specimen in NWC (400 ppb inlet $O_2$) and in HWC (400 ppb inlet $H_2$).

***Main ref.***

T.M. Karlsen, HWR-556. Paper presented at the Lillehammer meeting, March 1998

***Remaining work***

Completion of CT specimen post irradiadiation examination. Preparation of final HWR.

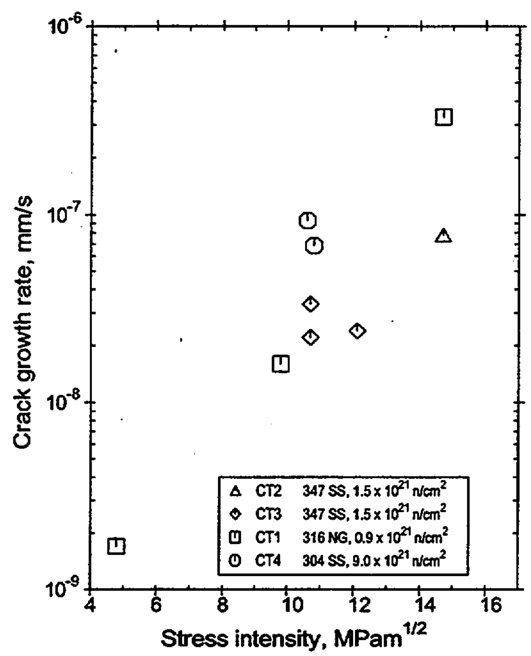
## INFLUENCE OF STRESS INTENSITY ON IASCC

***Objective***

Study Irradiation Assisted Stress Corrosion Cracking (IASCC) behaviour in Compact Tension (CT) specimens prepared from irradiated austenitic Stainless Steels in a simulated BWR environment.

***Results***

Crack growth rates have been measured in the CTs subjected to stress intensity levels ranging from 5 to 15 MPa$\sqrt{m}$ in NWC conditions (6-7 ppm $O_2$). The rates are considered reasonable showing a trend of increased crack growth rates determined at increased stress intensity levels, as illustrated in the figure below.




*Crack growth rates measured in 304, 347 and 316NG SS as a function of various stress intensity levels*

***Experimental basis***

Four CT specimens prepared from irradiated 316NG, 347 and 304 SS (fluence: 0.9 - 9.0 x $10^{21}$ n/cm$^2$) and instrumented for crack propagation monitoring using the DC potential drop technique. External bellows loading system enables on-line adjustment of applied stress intensity in IFA-639.

***Main ref.***

E. Hauso, T. M. Karlsen, HWR-614 (paper presented at the Loen meeting, May 1999), Status Reports

***Remaining work***

The study is being continued by measuring long term crack growth rates in NWC (and HWC, matter to be discussed)

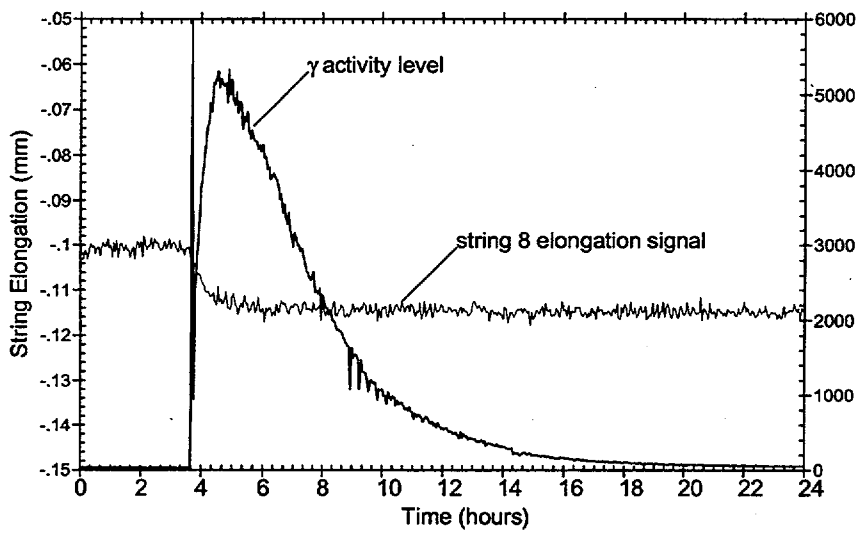
## CRACK INITIATION STUDIES

***Objective***

To study the effects of material, stress level and fluence on the initiation of cracking in pressurised stainless steel tube specimens in a representative BWR environment.

***Results***

Fluence accumulated for high fast flux and low fast flux specimens was $0.4 \times 10^{21}$ - $2.0 \times 10^{21}$ n/cm$^2$ and $5.6 \times 10^{15}$ - $7.5 \times 10^{18}$ n/cm$^2$. Three ruptured specimens were detected on-line and confirmed by weight measurements in interim inspections (January 1999).
No.3-5 (347 SS CW/welded/SRA, 1.2 x YS, $1.08 \times 10^{21}$ n/cm$^2$).
No.8-14 (304 SS sensitised, 2.5 x YS, $3.46 \times 10^{16}$ n/cm$^2$).
No.8-16 (304 SS sensitised, 2.5 x YS, $1.16 \times 10^{18}$ n/cm$^2$).




*Example of pressurised tube rupture detected on-line with string elongation and gamma activity measurements*

***Experimental basis***

The 34 tube specimens, which are prepared from sensitised 304SS, cold worked 347SS and solution annealed 304SS, 316LSS and 347SS have been pressurised to stress levels ranging from 0.8 to 2.75 times the yield stress of the unirradiated material and have been arranged in strings in high fast flux and low fast flux positions. The strings are fitted with elongation detectors. Loop has been operated with an inlet temperature of 290°C, an oxygen content of 3000 ppb and a solution conductivity of 0.5 µS/cm.

***Main ref.***

M.Nakata, HWR-613. Paper presented at the Loen meeting, May 1999.

***Remaining work***

Interim inspection will be performed to confirm further specimen ruptures. Ruptured specimens will be inspected in more detail by PIE. Irradiation of IFA-618 will be continued to higher fluence levels.

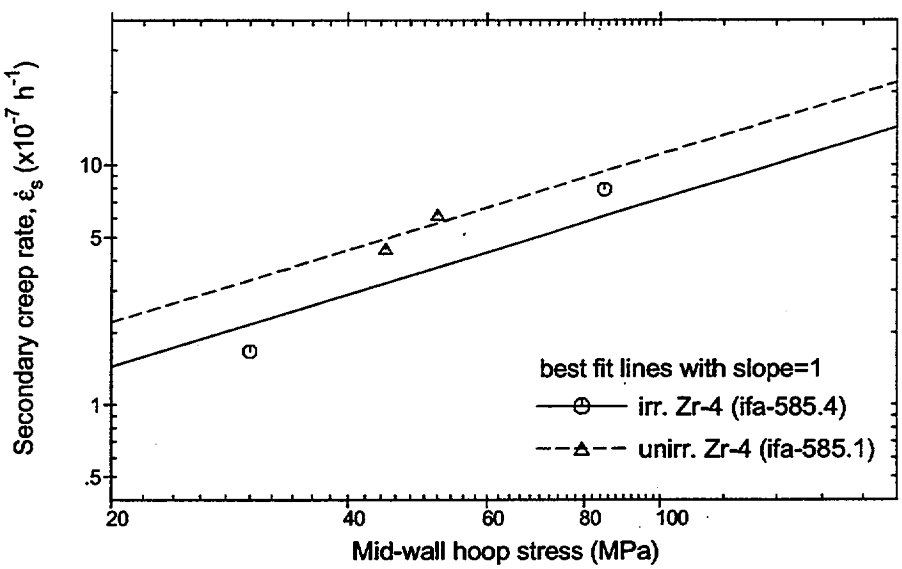
## CREEP OF PWR CLADDING MATERIAL (I)

***Objective***

To study the in-reactor creep behaviour of pre-irradiated Zr-4 cladding and to make a comparison to the fresh Zr-4 tested in-pile in an earlier loading.

***Results***

The in-pile primary creep strain and secondary creep rate of Zr-4 clad material were reduced as a result of pre-irradiation to a fast neutron fluence of $1.2 \times 10^{22}$ $ncm^{-2}$.




*Comparison of the in-pile stress dependence of secondary creep rate for fresh and pre-irradiated Zr-4 cladding.*

***Experimental basis***

A test rod consisting of two segments of clad, independently sealed at room temperature and joined by a mid-plug was loaded in a light water loop pressure flask assembly, IFA-585.4, surrounded by high enrichment booster rods. A 3-point contact scanning diameter gauge was used to determine the extent of clad creep, which resulted from the fixed hoop stress of 85 or 30 MPa.

***Main ref.***

M. A. McGrath, HWR-532. Paper presented at the Lillehammer meeting, March 1998.

***Remaining work***

To include the results from this experiment in a combined HPR covering all creep testing carried out in the IFA-585 test rig.

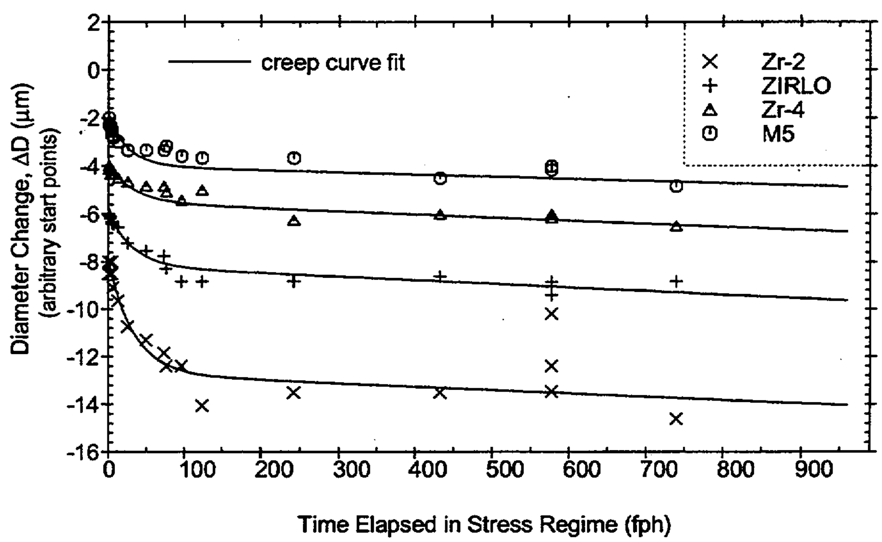
## CREEP OF PWR CLADDING MATERIAL (II)

***Objective***

To obtain in-pile creep data from modern, pre-irradiated PWR cladding materials, tested under variable loading conditions.

***Results***

The four different pre-irradiated clad segments (Zr-2, Zr-4, M5 and ZIRLO) exhibit similar primary creep behaviour under a compressive hoop stress of 75 MPa. Steady state conditions having been established show that each segment also exhibits similar secondary creep characteristics.




*Creep curves from each of the test segments for the first stress regime of 75 MPa in compression.*

***Experimental basis***

Two test rods, each consisting of two clad segments welded together, are loaded in a light water loop pressure flask, IFA-617.2, surrounded by high enrichment booster rods. Hoop stress variations are made by internal gas pressure control and 3-point contact scanning diameter gauges are used to determine the extent of clad creep.

***Main ref.***

M. A. McGrath, paper presented at the Loen meeting, May 1999.

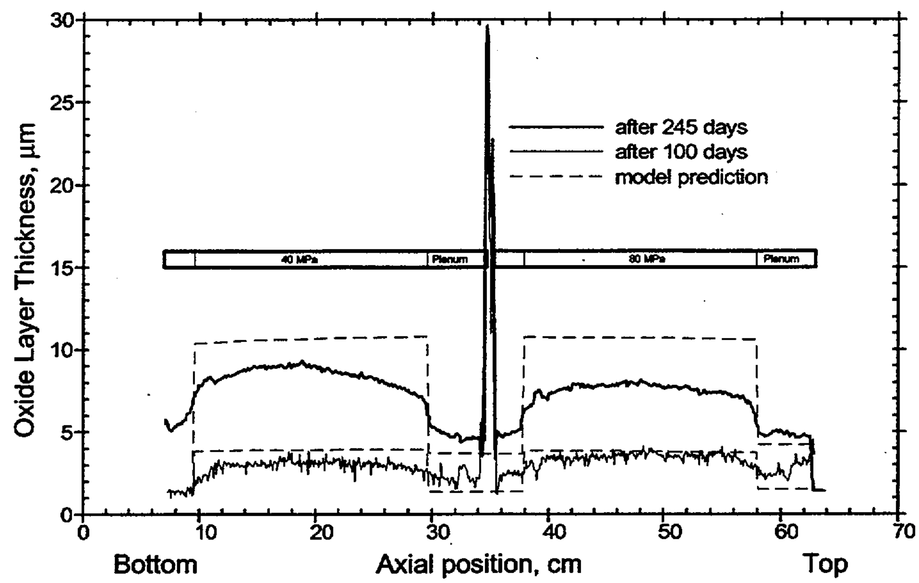
***Remaining work***

To subject each segment to a period of zero stress, followed by a period in tension, another period in recovery, several stress increments and a stress reversal.

## CORROSION OF PWR CLADDING (I)

***Objective*** To investigate the possible causes of accelerated corrosion of Zr-4 cladding under typical PWR operating conditions.

***Results*** In comparison to standard fresh Zr-4, a reduction in clad tin content improved corrosion resistance and an accumulation of radiation damage had a deleterious effect. Zero heat flux reduced corrosion rate, but no difference could be distinguished for different levels of heat flux. Corrosion rate was unaffected by varying clad hoop stress.




*Comparison between measured and predicted oxide layer thickness for low-tin Zr-4 cladding, indicating improved corrosion resistance.*

***Experimental basis*** Fuel segments were exposed in-pile for up to 245 fpd to PWR water chemistry in a series of loadings of IFA-593. Oxide layer thickness was measured with the eddy current technique at three interim inspections. The results were referenced to predictions from the EPRI/C-E/KWU corrosion model.

***Main ref.*** M. A. McGrath and D. Deuble, HWR-533. Paper presented at the Lillehammer meeting, March 1998.

***Remaining work*** All work relating to this IFA has been completed.

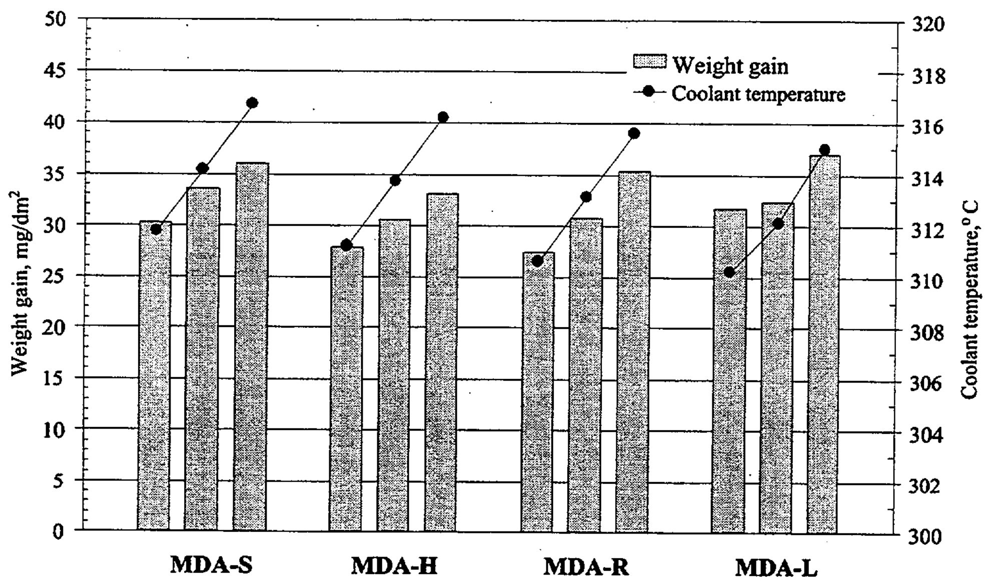
## CORROSION OF PWR CLADDING MATERIAL (II)

***Objective***

To study the corrosion behaviour of modern cladding materials at increasing burnup/neutron fluence under PWR conditions.

***Results***

The assembly has accumulated a burnup of ~12 MWd/kg$UO_2$ and 260 FPD (>2MW) since the start of irradiation (June 1998).
First interim inspection performed in January 1999, included visual examination, weighing of the coupons and oxide thickness measurements using the eddy current technique. The oxide thickness of the coupons is still thin (about 2μm) after 118 FPD.




*Weight gain of MHI coupons after 118 FPD of irradiation.*

*MDA types: S=standard, H=high pilgering temperature, L=low Nb (0.1 w/o), R=recrystallisation annealed*

***Experimental basis***

Three fuel rods which constitute both pre-irradiated and unirradiated cladding materials (Zr-2, Zr-4, ZIRLO, E635, Alloy A, M4 and M5), and 38 unfuelled coupons located both above the test fuel rods and axially in the centre of the rig, IFA-638.

***Main ref.***

M.Nakata and E.Hauso, HWR-566 (Characterisation data on cladding materials used in the corrosion test IFA-638).
HWR-606 report will be issued after the second interim inspection.

***Remaining work***

The second interim inspection will be performed to assess the oxide growth rate of PWR cladding materials in November 1999. Irradiation of IFA-638 will be continued to a burnup of about 50 MWd/kg.

## IFA-578: IRRADIATION, ANNEALING AND RE-IRRADIATION OF WWER-440 RPV WELD MATERIAL

***Objective***

The work is performed within an IAEA co-ordinated research programme on embrittlement of WWER-440 reactor pressure vessel weld material and effects of annealing and subsequent re-irradiation.

***Results***

The first irradiation cycle is completed. The target temperature and fluence was 268±5 °C and 1.9 x $10^{19}$ n/cm$^2$ (E>1 MeV, flux >2.3 x $10^{12}$ n/cm$^2$s), respectively. Table 1 summarises the results of the analysis of the melting alloy temperature monitors which were mounted in specimens in two of the six irradiation contatiners. The melting alloys indicated an irradiation temperature above 264 °C and below 272 °C, i.e. within target. Preliminary calculations of fluence gave a value of 1.9 x $10^{19}$ n/cm$^2$ (E>1 MeV). Two thirds of the specimens have been send to VTT for testing in the irradiated and irradiated+annealed condition.

*Table 1: Evaluation of melting alloy temperature monitors, PbPd5 ($T_m$=264 °C), PbAg1.9Sb5 ($T_m$=272 °C), PbAg1.9Sb4.5 ($T_m$=279 °C) and PbAg2Sb3.5 ($T_m$=287 °C).*

| Specimen No. | Rod No. (Pos.) | |
|---|---|---|
| | 529-27 (1) | 529-28 (2) |
| 1 | - | - |
| 2 | 287 °C, Not Melted | - |
| 3 | 272 °C, Not Melted | 279 °C, Not Melted |
| 4 | 264 °C, Melted | 264 °C, Melted |
| 5 | 279 °C, Not Melted | 272 °C, Not Melted |
| 6 | - | 287 °C, Not Melted |
| 7 | - | |

***Experimental basis***

Specimens of various geometries (Charpy-V, fracture mechanics and tensile) of standard and subzise dimensions will be tested in the irradiated (I), annealed (IA) and re-irradiated (IAI) conditions.

***Main ref.***

G. Ø. Lauvstad, EP-1578.4. "Experimental Procedure for IFA-578.4: Effects of Irradiation, Annealing and Re-irradiation on WWER-440 Pressure Vessel Material", 99-01-13.

***Remaining work***

The remaining one third of the specimens will be annealed at 475 °C for 100 hrs prior to the second irradiation cycle which will commence in November/December 1999.

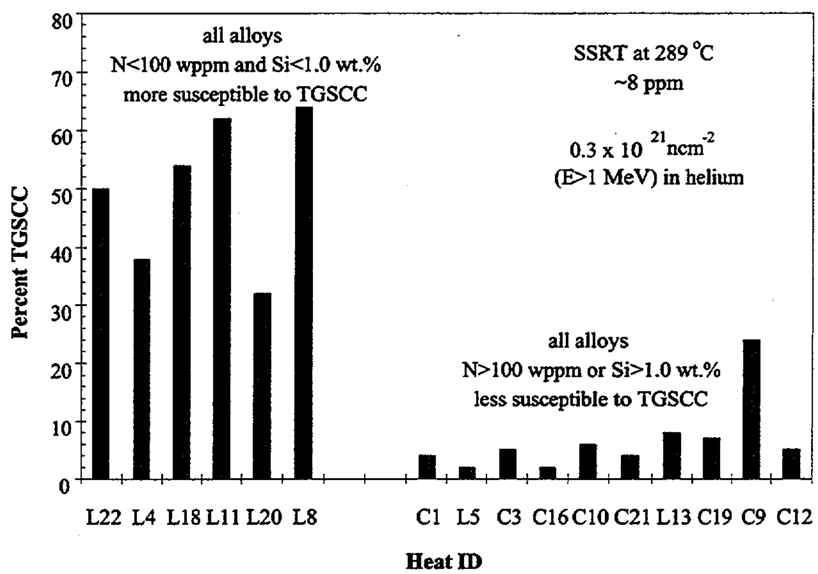
## DRY IRRADIATION

***Objective***

Evaluate the effects of fluence on the microstructural and mechanical properties of model austenitic stainless steels (Types 304, 316 and 348 SS).

***Results***

The experiment is conducted in co-operation with the USNRC and ANL. Slow Strain Rate Tensile (SSRT) tests and fractographic analyses have been performed on specimens irradiated to low ($0.3 \times 10^{21}$ n/cm$^2$) and medium ($0.9 \times 10^{21}$ n/cm$^2$) fluence levels. The percentage TGSCC on the fracture surfaces could be correlated well with N and Si concentrations.




*Percentage TGSCC on the fracture surfaces, classified as a function of N and Si content of the low fluence alloys.*

***Experimental basis***

Dry irradiation (in an inert helium environment at 288 °C) of 96 Slow Strain Rate Tensile (SSRT) and 24 Compact Tension Fracture Toughness (CTFT) specimens to low (0.3 x 1021n/cm2), medium (0.9 x 1021 n/cm2) and high (2.00 and 2.5 x 1021 n/cm2) fluence levels, followed by testing in BWR simulated conditions at Argonne National Laboratory, USA.

***Main ref.***

H.M. Chung et al., paper presented at the Loen meeting, May 1999

***Remaining work***

Continue post irradiation characterisation of medium fluence specimens and high fluence specimens irradiated to $2.0 \times 10^{21}$ n/cm$^2$. Continued irradiation of the remaining 32 SSRT and 8 CTFT specimens to a fluence of $2.5 \times 10^{21}$ n/cm$^2$.

# APPLICATIONS OF THE PHYSICS CODE HELIOS AT HALDEN

***Objective***

To provide reactor physics calculations for the support of experiments at the Halden Reactor. With main applications being; core planning, power determination, transport assessments and test rig design.

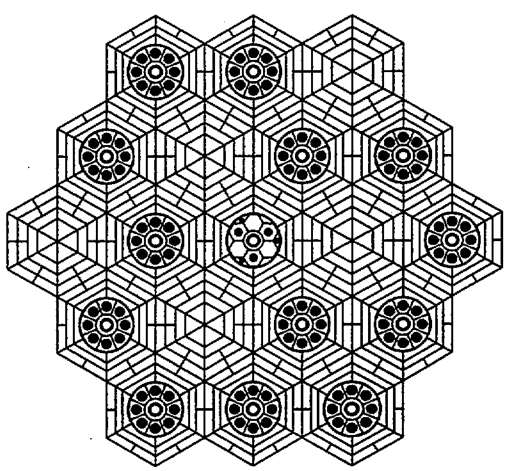
***Results***

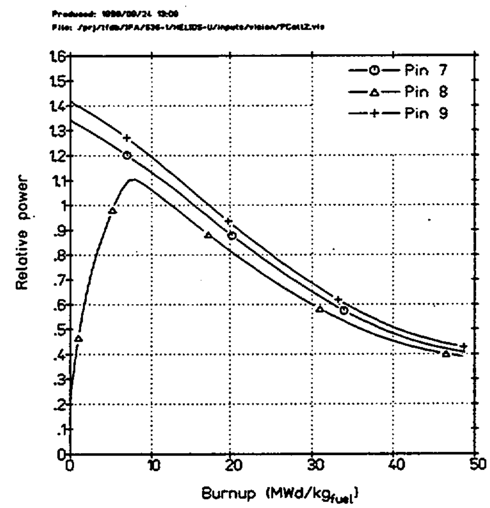
Since the introduction of the HELIOS code system, validation work has been carried out, see main reference.

The advantage of HELIOS lies in the ability to easily model the surrounding environment of a test IFA, see below left. This has improved fuel depletion calculations, especially in the presence of strong absorbers such as gadolinia. The results of the depletion calculations for IFA-636.1 can be seen below right.

The HELIOS calculations have also been configured to give values of the fast flux, either in relation to assembly linear heat rate, or neutron detector signal.

An estimate of the KG factor can now also be obtained from the HELIOS calculations. This greatly helps when deciding the position of a new test rig.






*Local environment and the pin power relative to average power at BOL; modelled by HELIOS for the upper cluster of IFA-636.1.*

***Main ref.***

U. Kasemeyer and W. Beere, HWR-609. Paper presented at the Loen meeting, May 1999.

***Remaining work***

The complete HELIOS system at Halden is still under construction, but the majority of the work required has already been achieved.

The placement of the input files for each HELIOS case has been standardized, and the compilation and execution of the HELIOS code has been automated.

To integrate the Revision Control System (RCS) for the generation of HELIOS input is currently being implemented. This work is well under way and should ensure that any results previously generated can be recreated, thus allowing cross referencing to previous results.

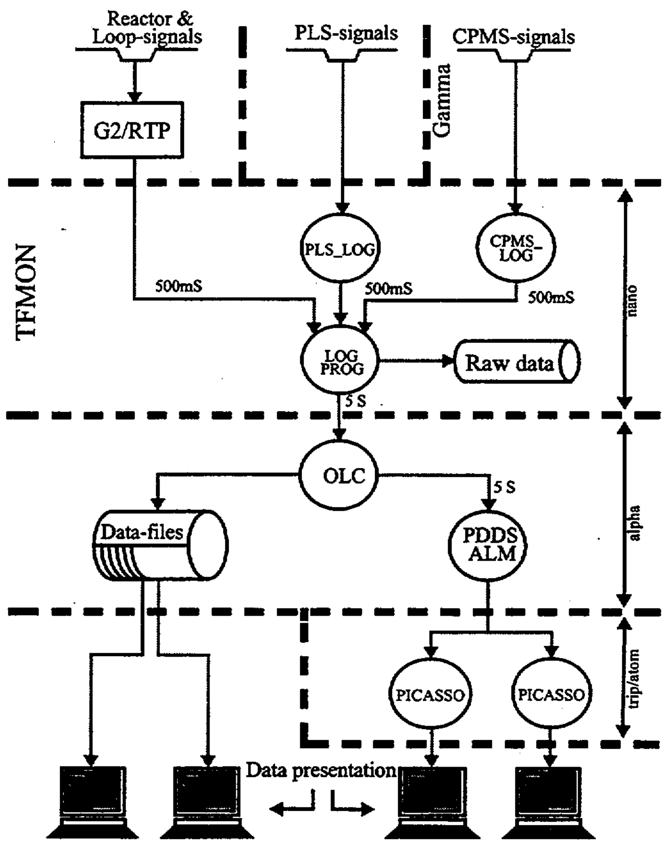
## DATA AQUISITION SYSTEM

***Objective***

- Integration of PLC- and DCB-data into TFD.
- Control room panel status through Picasso Alarm System.
- Storing remaining data from magnetic tapes into disk-resident systems.
- Replacement of G2 processor.

***Results***

Except for the last objective, all tasks have been successfully completed. More servers, X-terminals and net resources have been added. Additionally an UPS unit has been installed to ensure unbroken data acquisition and presentation at electrical power drop




*Data Acquisition system*

***Experimental basis*** Existing data acquisition and computer systems for on-line data conversion. RT-computer running HP-RT operating system.

***Main ref.*** Halden Status Reports.

***Remaining work*** Replacement of G2 processor with an IFE written data logging program. First system test September 1999.

## MEMBERS OF THE HALDEN REACTOR PROJECT

The members of the OECD Halden Reactor Project consist of signatory members and associated party members. Representatives from signatory members of the Halden Reactor Project may vote on issues brought before the Halden Board of Management and the Halden Program Group. Representatives from associated members of the Halden Reactor Project may attend meetings of the Halden Board of Management and the Halden Program Group, but have no vote on issues addressed by these bodies. The signatory members and associates are:

The Norwegian Institutt for Energiteknikk

The Belgium Nuclear Research Center

RISO National Laboratory, Denmark

The Finnish Ministry of Trade and Industry (VTT)

Electricite de France

Gesellschaft fur Anlagen-und Reaktorsicherheit, Germany

The Italian Ente per le Nouve Tecnologie, l'Energia e l'Ambiente (ENEA)

The Japan Atomic Energy Research Institute (JAERI)

Korean Atomic Energy Research Institute (KAERI)

The Spanish Centro de Investigaciones Energeticas, Medioambientales y Tecnologias

The Swedish Nuclear Power Inspectorate (SKI)

The Swiss Federal Nuclear Safety Inspectorate

British Energy, United Kingdom

United States Nuclear Regulatory Commission

The associated party members of the Halden Project are:

Comissao Nacional de Energia Nuclear, Brazil

Nuclear Research Institute, Czech Republic

Atomic Energy Research Institute, Hungary

N.V. Tot Keuring van Elektrotechnische Materialen (KEMA), the Netherlands

Russian National Research Center, Kurchatov Institute

Slovakian Nuclear Power Plant Research Institute, Slovak Republic

Institute for Protection and Nuclear Safety (IPSN), France

Argentinian National Nuclear Commission , Argentina

Associated Parties In The USA:

ABB Combustion Engineering N.P
Electric Power Research Institute (EPRI)
General Electric Company (GE)

Steering Bodies:

Under the Halden Agreement, an international committee, known as the Halden Board of Management, reviews and approves the research and experimental program and budgets on a yearly basis. The Halden Board of Management meets twice a year to conduct its business. Each signatory member of the Project has a representative on the Halden Board of Management. The U.S. Nuclear Regulatory Commission's representative to the Halden Board of Management is Dr. Margaret Federline, Deputy Director, Office of Nuclear Regulatory Research.

An international technical group, known as the Halden Program Group, provides input to the research program and reviews and evaluates the products from the research. The Halden Program group meets two to three times a year to conduct its business. Each signatory member and associate member of the Project has a representative on the Halden Program Group. The U.S. Nuclear Regulatory Commission's representative to the Halden Program Group is Dr. J. J. Persensky, Senior Human Factors Analyst and Team Leader, Regulatory Effectiveness Assessment and Human Factors Branch, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research.