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Present Status and Future Plans of the Halden Reactor

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1 Introduction

The Halden Reactor (HBWR) provides a means for testing fuels and reactor component materials in a highly controlled manner that also allows thorough monitoring of in-reactor behavioural parameters. The reactor continues to perform well with authority requirements fulfilled with ample margin, and each application to the Norwegian Government for an operating license has resulted in one being granted. The current license is valid until 2015 and work is now being started to prepare the basis for the next operating license application, which is also fully expected to be granted. Maintenance and upgrading of the plant is carried out on a continuous basis, such that there is no interruption to the fuel and materials research and development programs, and since its initial start-up on June 26, 1959, most of the original reactor plant components have now been replaced, including much of the primary system. The reactor pressure vessel is essentially the only remaining original component and the results from independent inspections, testing and data analysis, carried out most recently in 2006 and 2007, constitute a solid technical basis for saying that the reactor can be operated safely well beyond the year 2030.

The HBWR is the main facility of the OECD Halden Reactor Project (HRP), which has the main goal of contributing to safe and reliable operation of nuclear power plants. The HRP is strongly results oriented, with much of its success coming from the ability to satisfy many participants in an efficient and effective manner: the HRP has in excess of 100 participating organisations from eighteen countries. The vehicle by which the HRP delivers to its participants is the Halden Joint Programme - a central experimental programme comprising approximately 10-12 in-reactor tests at any given time, designed to generate key information for safety and licensing assessments. The content of this programme is discussed with the Halden participants every 3 years and renewed and adjusted accordingly, making the HBWR an asset for the nuclear community at a time in which maintaining centres of expertise at an accessible cost becomes increasingly important.

As the global nuclear industry enters a period of change generally considered to be a ”nuclear renaissance”, the HRP is thus well placed to contribute to the safety-related research activities associated with this nuclear renaissance, which can be divided into three main areas: current LWR sustainability programs, fuels research including for Generation 3+, and instrumentation development including R&D for GEN-IV.

2 Current LWR Sustainability Programs

The nuclear power industry in countries with ageing fleets of existing reactors is heavily involved with plant lifetime management programs. Such programs work towards license extensions past original reactor design lifetimes, for example from 40 years to 60 or even 80 years. Data on the behaviour of high radiation exposure material will be in great demand to support such programs. One of the main topics in the Halden Joint Programme is related to nuclear reactor materials – how reactor plant materials behave under the deteriorating effects of water chemistry and nuclear radiation. Two main test programmes within this topic are irradiation assisted stress corrosion cracking (IASCC) and stress relaxation.

2.1 Irradiation Assisted Stress Corrosion Cracking Test Programme

The Halden experimental programme on IASCC is aimed at: evaluating the effects of fluence, temperature, corrosion potential and stress intensity on cracking behaviour; generating long-term crack growth rate data; and determining 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. The tests are conducted in representative BWR and PWR environments. Use is made of irradiated materials which have been retrieved from commercial reactors for in-core measurements of crack growth rates at various stress intensity levels and test temperatures.

The Halden Project relies on recommendations and guidance from experts in the field in directing the course of the IASCC research activities. Regular review meetings are convened where, together with IASCC experts, the programme objectives are defined and reviewed on the basis of data that are generated both at Halden and within other national and international test programmes on materials ageing issues.

Long-term crack growth rate data from compact tension (CT) specimens prepared from irradiated core component materials have been generated in earlier BWR crack growth rate tests. Data were obtained for 347 SS, 316NG SS and 304L SS with neutron doses in the range ~0.1 - 3 dpa and ~13 - 35 dpa in both oxidizing (5 ppm O_2) and reducing (2 ppm H_2) environments. The benefits of low potential in reducing rates of crack growth were clearly evident for the lower dose materials, while for the higher dose the beneficial effect of H_2 addition was lost at higher K levels. A more recent BWR CGR study has now been completed with 316L and 304L stainless steels removed from Oskarshamn and Barseback NPPs respectively. These recently investigated materials have dose levels in the range 7 – 11 dpa, i.e. in between those of the earlier investigated materials.

In the recent BWR tests, crack growth data were obtained from CTs at 325°C in both oxidizing (5 ppm O_2) and reducing (2 ppm H_2) conditions. In 5 ppm O_2, the growth rates ranged from ~10^{-7} to 10^{-5} mm/s at estimated K levels between 11 - 20 MPa$\sqrt{m}$. At low ECP (2 ppm H_2) the crack growth rates typically decreased by an order of magnitude, indicating the effectiveness of lowering the ECP in decreasing the crack growth rate of medium dose material.

Two crack growth rate tests on irradiated materials have been performed earlier in PWR conditions. In the first of these, crack growth data were obtained from CT specimens prepared from 304 SS materials with doses of 13, 17 and 35 dpa. In the second test, crack growth rate data were generated for CTs prepared from ~12 dpa Barseback 304L SS and CTs prepared from the same 304L SS with a dose of ~3 dpa. The growth rates for the higher dose material were an order of magnitude (~10^{-6} mm/s) higher than those of the lower dose samples.

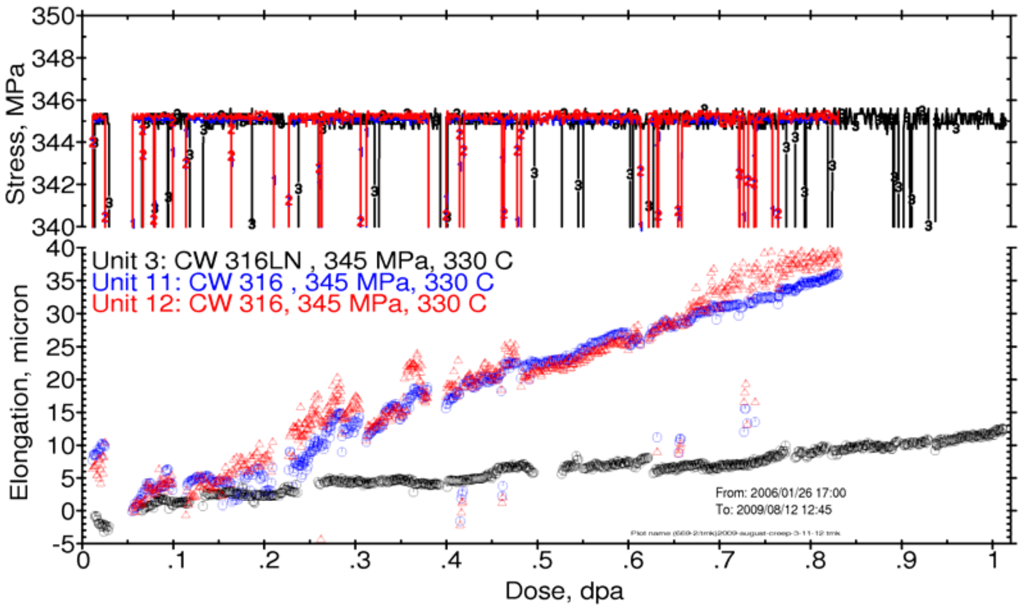
A third PWR crack growth test is now under preparation with CTs made from 304L SS, 304 SS and CW 316. Material doses range from 2 to 30 dpa and some have been given a post irradiation annealing (PIA) treatment (500°C for 6 hours). During irradiation the specimens will be exposed to PWR primary water chemistry conditions, with the addition of Li, B and H_2. The effects of temperature on cracking response will be studied by operating the specimens in coolant at ~280 and at 325 - 335 ºC. The use of PIA as a means for improving / restoring resistance to SCC in irradiated materials will also be evaluated.

In preparation for future crack growth tests, the possibility of securing weld material (308 SS) from the Barseback NPP as well as identifying and securing IASCC resistant materials from other test programmes will also be explored.

2.2 Stress Relaxation Test Programme

Irradiation enhanced stress relaxation is a potential degradation mechanism that may affect the long-term performance of core-internals. An in-pile study aimed at assessing the effects of irradiation and applied load on stress relaxation in materials commonly employed in LWRs is being conducted at Halden under dry irradiation conditions.

A total of 12 instrumented tensile specimens and 18 non-instrumented tensile specimens are included in the test matrix. The target dose is 2 dpa. The instrumented specimens are installed in capsules that allow both temperature and stress control. Load is applied to the specimens by means of bellows that are compressed by gas pressure introduced into the chamber housing the bellows. Constant displacement of the specimens (stress relaxation mode) is maintained by monitoring sample elongation with LVDTs and reducing the applied load on-line by decreasing the pressure in the bellows housing units. Otherwise the specimens can also be operated in creep mode by keeping the stress constant. The test units are also equipped with gas lines that enable the specimen temperatures to be controlled at the 290, 330 and 370°C targets by altering the composition of helium-argon gas mixture surrounding the specimens. The 12 non-instrumented samples were preloaded by means of bolted assemblies and installed in units that were pre-filled with helium-argon gas mixtures and sealed.



Summary of elongation data recorded for CW 316 SS and CW 316 LN

The test matrix includes CW 316 SS, primarily from one heat of material that is used for replacement baffle bolts, and CW 316N lot, which is an extensively tested material for irradiation creep. 304 L SS is also included, which has also been used elsewhere in irradiation creep tests, as well as aged Alloy 718, which is presently being tested for stress relaxation in a commercial PWR and thus will allow comparison of the Halden and commercial data. The final material included is 316 LN (low C, high N) with a low Stacking Fault Energy (SFE) and this has been included because other work has indicated that low SFE material has increased creep resistance, so the material may be attractive as a replacement for current baffle bolt material.

Both creep and stress relaxation data have been obtained for CW 316 SS specimens. The creep data are fitted to an equation of the form $e/\sigma=A[1-\exp(-Bf)]+Cf$ and the stress relaxation data are represented by $\sigma/\sigma_0 = \exp\{-E[A(1-\exp(-Bf))+Cf]\}$. The CW 316N lot and aged Alloy 718 specimens exhibit higher stress relaxation than the CW 316 SS specimens and the CW 316LN (low SFE) and 304L SS samples are more creep resistant than the CW 316 SS.

The current dose of the specimens is ~0.9 to ~1.1 dpa and irradiation is scheduled to continue through 2010 and 2011. A decision as to whether to continue beyond the 2 dpa target will be based on results obtained at that dose level. On completion of this first test, a second test may be defined to include Alloy 718 creep specimens and/or 304L SS specimens with a geometry that is more suited to obtaining stress relaxation data than those currently being tested.

3 Fuels Research and Generation 3+

Improvements to existing LWR designs are being undertaken by all the major reactor vendors, such that much of the upcoming new-build units will be Generation 3+ or Next Generation LWRs. Significant changes to fuel or fuel assemblies instigates a need for re-licensing or approval of the modifications by the appropriate safety authorities. Associated with such approvals is a need for supporting data to demonstrate as good as or improved safety of the modified fuel. Another main topic in the Halden Joint Programme is related to nuclear fuels – how fuel and cladding performs under normal, transient and accident conditions, with emphasis on high burn-up and new fuel and cladding types. Two main test programmes within this topic are an experiment studying integral behaviour of new fuel types and a test series studying fuel rod performance during LOCA.

3.1 Integral Behaviour of New Fuel Types

Fission gas release (FGR) and the subsequent increase in rod internal pressure can lead to safety criteria imposed limits in fuel utilisation in LWRs. This issue is studied in a number of experiments carried out in the Halden reactor, among them one planned to start at the end of 2009 will address fission gas release properties of standard, doped and modified fuel.

In order to achieve favorable fission gas release properties for doped fuel, the effect of increasing the grain size (diffusion length) must be balanced against the negative influence arising from increases in the diffusion coefficients as a result of the dopants. The question is whether the positive influence of large grain, observed in previous tests, prevails at increased temperatures since the diffusion coefficients vary with temperature. The objective of this new test at Halden is therefore to study the effect of dopant concentration and grain size.

The test is designed based on previous successful long-term fuel irradiation experiments containing 6 fuel rods in a circular arrangement. The six rods are as follows:

1. Standard UO_2, grain size about 10 μm.

2. Large grain size Cr doped fuel (1055 ppm Cr_2O_3)

3. Large grain size Cr doped fuel (1577 ppm Cr_2O_3) at the Cr solubility limit.

4. Large grain size UO_2 without significant dopant (10 ppm Cr_2O_3).

5. Fuel made from sintering active UO_2 powder achieving large grain, high density fuel. Grain size is ~50 μm making it possible to compare with rod 4.

6. Composite uranium oxide-beryllium oxide fuel with improved thermal conductivity to obtain lower temperatures and thus lower fission gas release.

Each rod is equipped with fuel thermocouples, fuel stack elongation detector and rod pressure transducer. The instruments provide essential data for the phenomena to be addressed. The fuels with deliberately larger grains have a grain size of 55 – 70 µm and all fuels have an enrichment close to the 5% limit for commercial fuels. A comparison of rods 2, 3, 4 and 5, which have similar grain sizes and will operate at the same power and temperature, should reveal the effect of Cr content (0, ≈900 and ≈1600 ppm). All fuels were provided to Halden by commercial fuel manufacturers.

The fuel will be irradiated in Halden reactor conditions at about 235°C coolant temperature and 34 bar pressure. In accordance with discussions in the Halden Programme Group, the envisaged power during the first two or three operation periods will be 30 – 35 kW/m corresponding to 1200 – 1300°C fuel centre temperature. This will keep the fuel below the FGR threshold. Then, depending on fuel type, FGR is expected to start and differences in both onset and amount of FGR will become apparent. Maintaining similar peak fuel operating temperatures is important for comparative purposes, but a certain neutron flux tilt across the test rig is unavoidable. The rods are grouped in pairs exposed to about the same neutron flux, e.g. doped fuels, non-doped fuels, and standard and high conductivity fuel.

3.2 Loss of Coolant Studies

The Halden in-core LOCA experiments focus on effects that are different from those obtained in out-of-reactor tests. Heating is provided with a low level of fissions simulating the initial decay heat so the relative thermal expansion of fuel and cladding is closer to the real situation and hence phenomena such as axial gas flow and maintaining/breaking fuel-clad bonding are realistically reproduced. By mid 2009, six tests have been conducted on pre-irradiated segments. Some characterisation data of the segments are shown in the table below:

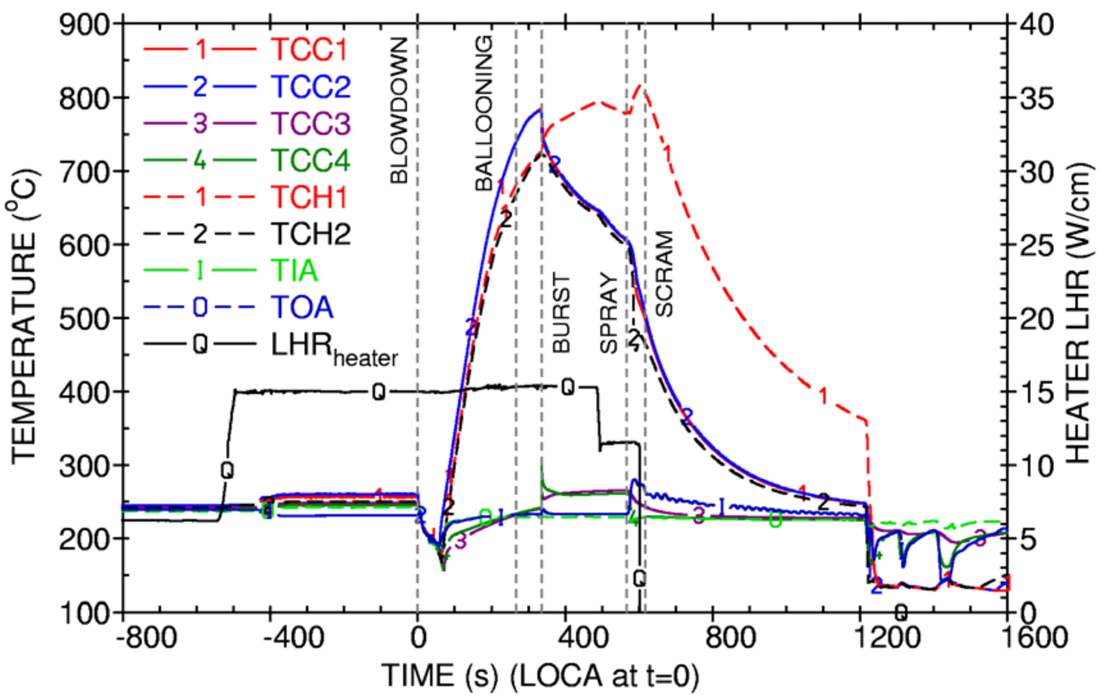
Items	**650.3**	**650.4**	**650.5**	**650.6**	**650.7**	**650.9**
Target PCT (°C)	800°C	800°C	1100°C	850°C	1100°C	1100°C
Fuel type	PWR	PWR	PWR	VVER	BWR	PWR
Segment length (mm)	50	50	50	45	50	50
Base irradiation	6 cycles	7 cycles	6 cycles	4 cycles	3 cycles	7 cycles
Burn-up (MWd/kgU)	82	92	83	56	35	90
Cladding type	Zr-4/1.47Sn	Zr-4/1.47Sn	Zr-4/1.47Sn	E110	LK3/L	Zr-4 / 1.47Sn
Oxide thickness (µm)	25-28	10	70-80	~5	<10	7-8
Hydrogen content (ppm)	250	50	650	-	-	30
O.D. / thickness (mm)	10.75/0.725	10.75/0.725	10.75/0.725	9.13/0.68	9.62/0.63	10.75/0.725
Liner (µm)	150	100	150	No	Yes	100
Heat treatment	SRA	SRA	SRA	Standard	Standard	SRA

Due to its special importance, the test series is discussed regularly in the Halden Programme Group at workshops and programme review meetings. The primary objective is to observe the overall fuel behaviour under expected and bounding conditions in an in-reactor test. An essential part is to assess the extent and effect of fuel relocation into the balloon region. The experiments executed so far have indicated that relocation occurs as a consequence of ballooning and at the time of rupture. Other aspects to be assessed are the filling ratio of the balloon region with fuel fragments and the effect of locally increased temperature on cladding behaviour and properties. The secondary objective is to investigate local oxidation and transient hydriding.

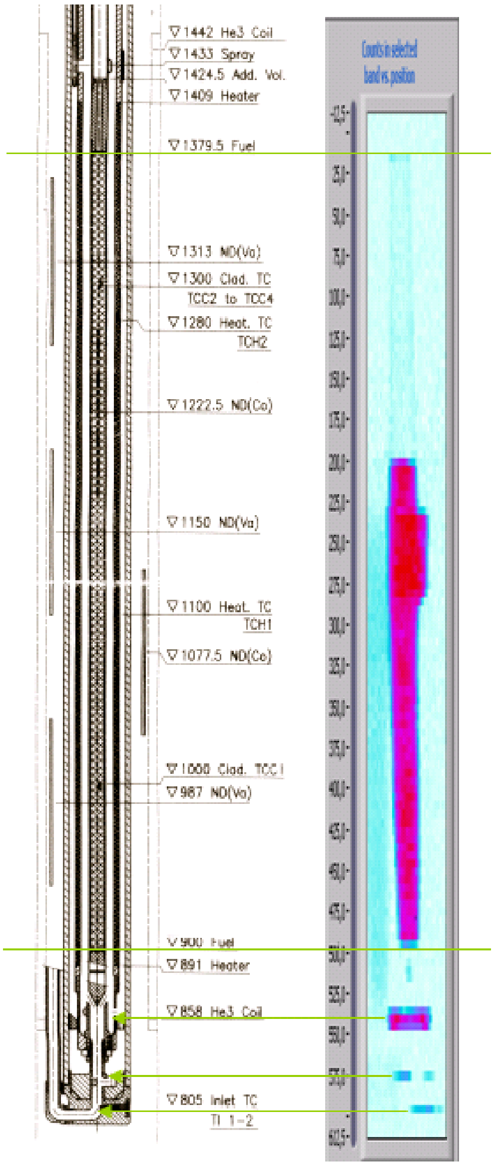
The Halden LOCA test set-up includes the following features:

- Single rod experiment using high burn-up fuel
- Heating provided from within the rod by low nuclear power generation simulating decay heat
- Simulation of thermal boundary conditions and constraints is provided by an insulating channel, the shroud of which can be heated electrically to give an even temperature distribution
- Provision of a spray system to ensure sufficient availability of steam for cladding oxidation
- Rod instrumentation includes cladding surface thermocouples, cladding extensometer and a rod internal pressure sensor

The previous LOCA experiments have shown that fuel relocation is best detected by the response of the cladding and heater thermocouples. The built-in rod free volume is 15-18 cm^2 for maintaining an absolute amount of gas similar to that in a full-scale fuel rod. The LOCA test itself encompasses the phases of blow-down, heat-up and a hold period of 5-6 minutes at peak clad temperature (PCT). The tests are terminated by a reactor scram and normal cool-down. The pressure flask and fuel segment inside is stored in helium prior to gamma scanning at Halden and then transported to the IFE-Kjeller hot cell facilities for PIE.



Temperatures recorded during the LOCA test 650.4. TCC1-TCC4 are cladding temperatures; TCH1 and TCH2 are lower and upper position heater temperatures. The increase in TCH1 relative to TCH2 after the burst indicates increased local temperature due to fuel relocation.



Important observations so far include substantial fuel fragmentation in tests number 650.4 and 650.5, with significant fuel relocation and ejection from the rod in test 650.4. In test 650.5, due to a much smaller balloon and burst opening, only a small amount of fuel was ejected from the rod, but the fine fragmentation of the pellets (seen in PIE) indicates the potential was also there for this rod had the ballooning be larger. Since the burn-up of the fuel rods used in 650.4 and 650.5 exceed the current highest discharge burn- limits, Halden participants have expressed an interest in exploring the burn-up limit for substantial fuel fragmentation.

Both these tests were used by participants of the second NEA/CSNI-WGFS benchmark for testing the predictive capabilities of codes in assessing LOCA transients. The results showed varying agreement with the measurements and observations, indicating a need for further improvement of such codes.

The sixth test with a pre-irradiated segment (650.9), executed in April 2009, also used a high burn-up duplex fuel segment similar to those used in 650.4 and 650.5 and the experimental conditions and results were similar to those of 650.4 e.g. the measured cladding and heater temperatures showed the effect of fuel relocation which was confirmed by gamma scanning later on. This test rod is now undergoing PIE.

Gamma scanning results from test 650.4 indicating ballooning of the cladding at the central axial location and loss of fuel from the upper part of the test segment (ejected through the burst opening in the cladding).

The further planning of experiments in the LOCA test series includes:

- PWR Zry-4 fuel with a burn-up of 60 MWd/kg to confirm high temperature performance and fragmentation/relocation behaviour at that burn-up level

- VVER fuel with high burn-up

- BWR fuel with a burn-up of 70 MWd/kgU to explore the fragmentation limit

The exact test programme on LOCA behaviour of LWR fuel will not only depend on the outcome of the in-core phase, but also on the results of quite large PIE campaigns. Each of the future tests will therefore also carefully be discussed within the Halden Programme Group and accompanied by pre-test evaluations provided by participants.

4 Instrumentation Development and R&D for GEN-IV

The development and optimisation of instrumentation for in-core monitoring form an integral part of the materials programme and will be continued. Developments include instrumentation for monitoring in-core environmental parameters such as high temperature conductivity and corrosion potential and instrumentation for monitoring corrosion processes on-line.

One of the most important parameters that affects stress corrosion cracking (SCC) in reactor constructional materials is the electrochemical corrosion potential (ECP) of the material. The ECP should be measured at the same location in which the SCC samples are positioned; hence in-core measurements are required for Halden tests, as well as for in-plant studies. The in-core measurements require ECP reference electrodes that can withstand the aggressive in-core conditions, including exposure to gamma and neutron fluxes. Developments in this field have been ongoing in Halden since 1996 and different types of in-core ECP electrodes have been developed since then. The main technical difficulty in designing ECP electrodes is related to the required sealing between metal and ceramic parts of the electrode.

The Halden Project, in conjunction with VTT, Finland, has developed (in 1997) and in-core palladium (Pd) reference electrode that can be used under PWR as well as BWR conditions (using hydrogen water chemistry). The operation of this electrode, which works reliably in-core, requires however a complicated polarization sequence. Therefore a platinum reference electrode has been developed. This can be used as a reference electrode under so-called reducing conditions, normally defined as when the molar ratio of hydrogen to oxygen is greater than 2. In this case, the platinum functions as a hydrogen electrode. The standard Halden Pt electrode is based on a mechanical seal between the ceramic tube (Mg stabilized zirconium oxide or Mg PSZ) and the metal parts and is therefore relatively bulky. The outer diameter of the Pt tip is 13 mm. The diameter of the signal cable is 1 mm.

Recently, a new miniaturised platinum electrode has been developed. In this design, the seal is made by brazing, resulting in a significant reduction in the diameter of the electrode – down to 7 mm. The Pt tip that is exposed to water has a diameter of 1 mm and is 10 mm long. This newly developed electrode has been extensively tested in autoclave up to 150 bar and 350^{o}C and works well. This electrode can of course also be used as a simple and compact electrical feed-through for performing in-core electrochemical noise or electrochemical impedance measurements, where one has to connect the inner wire of the signal cable in a leak-tight way to the sample under investigation.

In addition, Halden is currently developing instruments that will be able to withstand demanding supercritical water conditions, with so far very promising results. This is very important in relation to the worldwide research into Generation IV reactors and in particular to the High Performance Light Water Reactor (HPLWR) concept, presently being pursued within the European Framework programs, in cooperation with the Generation IV International Forum (GIF). The HPLWR will be operated with a pressure of at least 250 bar and a temperature of at least 500°C, which will make it a highly efficient system and thus give substantial economic and environmental advantages over today's NPPs.

The heart of many of Halden's in-core instruments is the Linear Variable Displacement Transducer (LVDT). The LVDT is a versatile instrument used to transform a mechanical movement into an electrical signal. The primary coil is activated by a 400 Hz constant-current generator and the position of the magnetic core in relation to the coils affects the balance of the signal from the secondary coils. Thus any mechanical movement that changes the position of the magnetic core generates a corresponding signal that can be measured. The LVDTs in general use are designed to operate under PWR conditions (350^{o}C and 150 bar) but they can also be operated for shorter periods up to 500^{o}C. Presently, Halden is developing LVDTs that can have a working range up to 600°C. In these LVDTs another type of wire is used. The signal cables used are 2-wire mineral insulated (Al_2O_3) cables with Inconel 600 sheath. The outer diameter of the cables is 1.0 mm. LVDTs for higher temperature operation are presently under development. In these LVDTs the wires are either made of anodized aluminium or of ceramic insulated silver alloy wire. Some of these high temperature LVDTs will soon be tested in a super critical water loop.

Halden has also joined the Nordic Network for Generation IV Nuclear Power Reactors that organises information exchange and cooperation between research groups. In addition, a successful feasibility study has been completed with a view to building a very high temperature and pressure coolant loop system in the Halden reactor, which would enable fuel and materials to be tested under SCWR conditions in the future.

5 Acknowledgements

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