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Halden Fuel and Material Experiments Beyond Operational and Safety Limits

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ABSTRACT: One of the main tasks of any research reactor is to investigate the behavior of nuclear fuel and materials prior to their introduction into the market. For commercial NPPs, it is important both to test nuclear fuels at a fuel burn-up exceeding current limits and to investigate reactor materials for higher irradiation dose. For fuel vendors such tests enable verification of fuel reliability or for the safety limits to be found under different operational conditions and accident situations. For the latter, in-pile experiments have to be performed beyond some normal limits. The program of fuel tests performed in the Halden reactor is aimed mainly at determining:

- the thermal FGR threshold, which may limit fuel operational power with burn-up increase
- the "lift-off effect" when rod internal pressure exceeds coolant pressure
- the effects of high burn-up on fuel behavior under power ramps
- fuel relocation under LOCA simulation at higher burn-up
- the effect of dry-out on high burn-up fuel rod integrity

This paper reviews some of the experiments performed in the Halden reactor for understanding some of the limits for standard fuel utilization with the aim of contributing to the development of innovative fuels and cladding materials that could be used beyond these limits.

KEYWORDS: *Halden reactor, fuel, burnup, FGR, LOCA, relocation, fragmentation, lift-off, dry-out, ramp.*

I. INTRODUCTION

The effectiveness of nuclear fuel utilization for energy production is restricted by the operational and safety limits which are…" introduced to avoid fuel failures during normal operation and to mitigate the consequences of accidents in which substantial damage is done to the reactor core" [1]. These limits are usually determined with some margin which may be related to differences between actual conditions and the safety limit as well as fuel failure (Figure 1).

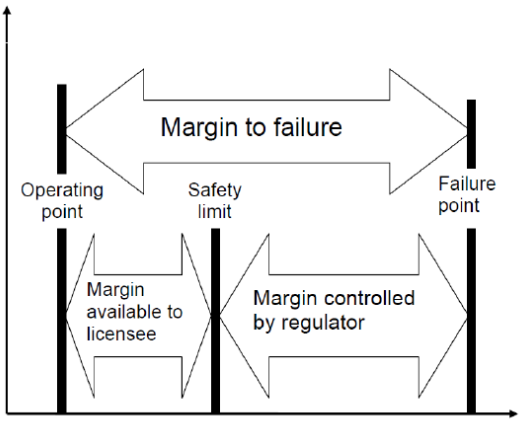

Figure 1 Types of margin [1].

Margins are an expression of incomplete knowledge and uncertainties. The ways for improving of nuclear fuel utilization: the first is to reduce the margins (conservatism) between actual state and failure point, which can be achieved by better knowledge and reduction of uncertainty on the behavior of conventional fuel and materials and the second is to develop a modified or advanced materials which are able to be used beyond safety limits or the failure point adopted for ordinary materials.

Both these approaches require not only theoretical considerations but also experimental confirmation from some special tests performed in research reactors. In this case the tests should be carried out beyond the operational and safety limits, and sometimes up to fuel failure.

The experiments carried out in the Halden reactor both within the framework of the OECD HRP Joint Programme and on a bilateral basis are aimed at increasing fuel safety, reliability as well as increasing utilization of the fuel. Some of the "standard" and unique test capabilities of the Halden reactor as well as results from the tests performed beyond operational or safety limits are presented in this paper.

II. HALDEN EXPERIMENTS BEYOND OPERATIONAL AND SAFETY LIMITS

Power generation and time of fuel operation define the energy production and thereby the level of fuel utilization within NPPs. Fuel vendors, within the licensing process, establish a design limit for the fuel they supply, which restricts power rating and fuel discharge burnup, schematically shown in Figure 2 [1]. The following tests performed in the Halden reactor are aimed at setting the safety criteria shown in Figure 2:

- Tests to determine dimensional stability of the fuel cladding such as mechanical elongation due to applied

stress and irradiation growth;

- Tests for determination of FGR threshold which may restrict the LHGR with burnup related to the design limit;
- Cladding "lift-off" testing to determine the gas overpressure limit to avoid feedback effects due to reopening of cladding-fuel gap;
- Ramp testing to determine PCI failure thresholds;
- Fuel thermal conductivity degradation tests which may affect the fuel melting limit;
- Corrosion tests to detect limits related to hydriding of Zr based fuel claddings.

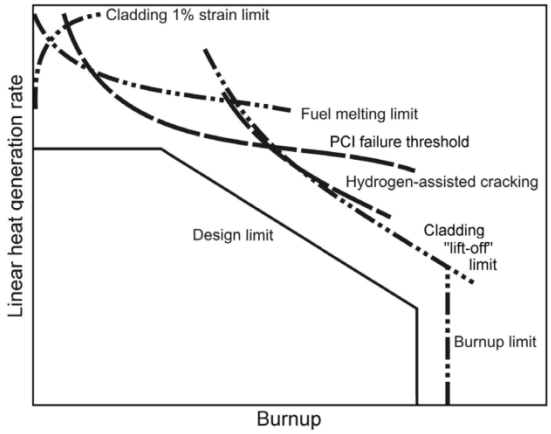

Figure 2 Various operational fuel safety criteria vs. design limit of LHR and burnup [1]

Most of these tests are performed to find power- burnup limits with real safety margins.
In addition:

- The series of LOCA simulation tests are one of the main topics of the HRP programme, aimed at determining fuel fragmentation and relocation as well as secondary hydriding of cladding at high burnup which may influence the safety limit for Design Basis LOCA;
- A test beyond DB LOCA is also considered feasible with the aim of studying fuel melting and hydrogen release;
- Dry-out tests have also been performed in Halden to study cladding corrosion and hydriding under overheating, and other effects related to high burnup can also be studied e.g. FGR from "rim" fuel or cladding "lift-off" at overpressure.

1. Fission Gas Release threshold

The test to study integral FGR and PCMI fuel behavior at high burnup is one of the "standard" in-pile tests in Halden. The rig contains two refabricated rods pre-irradiated in a commercial NPP, and is shown schematically in Figure 3. The rods are instrumented with a pressure transducer (PF) or a cladding elongation detector (EC) in one end and with fuel thermocouples in other end. The rod instrumentation allows FGR to be detected at the moment, and at the fuel center temperature, at which it occurs. The rig is designed and produced with He-3 coil surrounding the test channel in order to control the power of the test fuel assembly independent of the reactor power.
The test is usually performed with a step-wise increase in fuel temperature, with steps of around 50°C held for 12-24 hours to allow the fission gases to be thermally activated and released if the thermal threshold is exceeded.

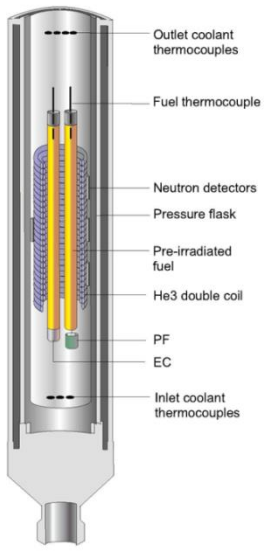

Figure 3 Test rig for determination of FGR threshold and study PCMI for high burnup fuel.

The typical results of the in-pile measurements during the test are shown in Figure 4.

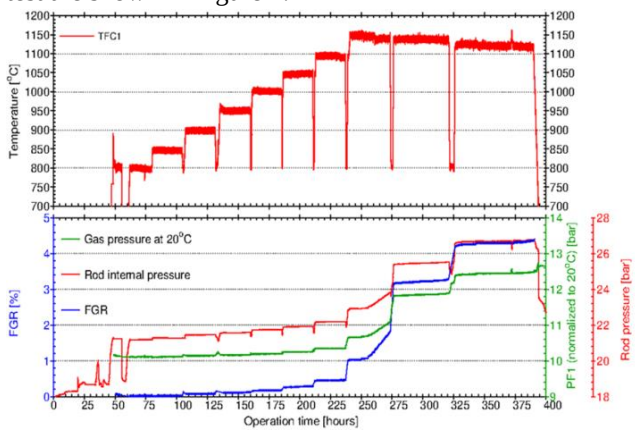

Figure 4 Typical in-pile measurements for FGR threshold determination

In addition, power / fuel temperature reductions are usually performed at the end of each hold period in order to ensure that all gas released during the preceding power/temperature period, but trapped due to tight fuel-clad contact or within fuel cracks, actually reaches the pressure detector that is positioned in the fuel rod plenum.

With this experimental approach, whereby there is a simultaneous measurement of fuel temperature and gas pressure, the thermally activated FGR threshold can be determined. An empirical correlation for the 1% FGR threshold as a function of fuel temperature and dependent on burnup [2] was developed based on many such tests performed in both the Halden reactor and other research

reactors and the formulation is shown below:

$$T_{FGR} = 9800/\ln(1000 \cdot \frac{Bu}{5}) \quad (1)$$

where:

Bu is the max fuel burnup in MWd/kg UO_2;

T_{FGR} is the max fuel center temperature in ^{o}C, at which ≈1% of fission gasses has been evaluated;

The FGR threshold formulation was based on data covering a maximum burnup of about 35 MWd/kg oxide as shown in Figure 5. Data obtained later in Halden for higher burnups show that the temperature of fission gas release onset exhibits a decreasing trend, as also shown in Figure 5.
The so-called Vitanza threshold and the trend obtained with additional data represent a valuable knowledge base when it comes to controlling experiments in such a way that fission gas release is avoided or provoked.

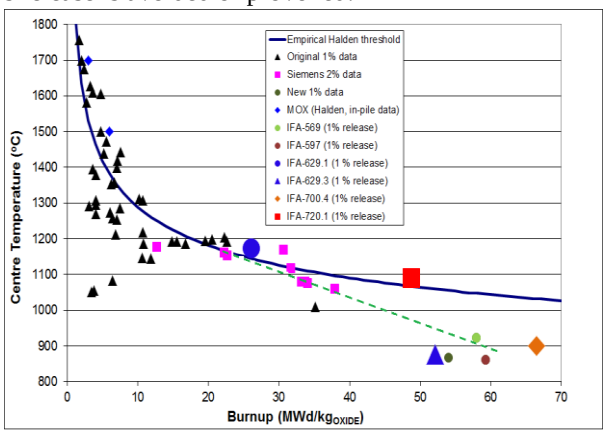

Figure 5 Thermal FGR Vitanza threshold vs fuel burnup [2].

It should be noted that the thermal FGR threshold may be dependent on fuel type (UO_2 or MOX, Gd-doped) but independent of fuel design (PWR, BWR and VVER). The thermal threshold can be recalculated to a LHR threshold for certain fuel designs as a useful approach to limiting fuel rod power as a function of burnup in order to avoid excessive FGR during irradiation.

However, FGR inventory may exceed the operational limit that is imposed to avoid the "lift-off" effect. This limit is usually set at rod internal pressure equaling the coolant system pressure. The study of fuel behavior at rod internal gas pressure beyond the limit for high burnup fuel is also part of the Halden research programme.

2. Lift-off effect and rod pressure tolerances

The series of "lift-off" experiments is related to the issue of rod overpressure which may increase the potential for further fission gas release with increasing burnup. The studies are primarily aimed at:

- evaluating the overpressure leading to onset of fuel temperature increase - indicator of cladding "lift-off";
- investigating the temperature response at different overpressure levels;
- assessing the creep-out of different cladding

For the test, the fuel rod is re-fabricated from a pre-irradiated rod and equipped with a fuel centerline thermocouple, and then subjected to various levels of rod overpressure. The latter is provided by an ultrahigh gas pressure system connected to the top and bottom of the fuel rod through pipe lines. Supplementary measurements such as cladding elongation, hydraulic diameter, gas release measurements by means of gamma-spectrometry and noise analysis are combined with the primary temperature measurement in order to gain a more complete picture of the fuel performance under rod overpressure conditions. The rig with advanced design features including a diameter gauge is under development for future tests to study a combination of lift-off by temperature measurements and cladding creep-out by diameter measurements (Figure 6).

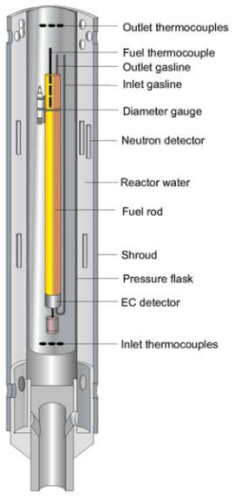

Figure 6 Advance test rig for "lift-off" study.

The test procedure usually consists of the following important steps with some variations:

- Power calibration;
- Hydraulic diameter measurements at different power levels (for solid pellets);
- Temperature measurements with different gas mixtures in the tests rod (Ar or He);
- Pressurization of the rod at stable power with overpressure steps of 50-100 bar and holding times of about 500 hours (to avoid the effect of transient (primary) cladding creep).

In order to detect a clad lift-off effect, fuel temperature measurements are normalized to the same power (Figure 7).

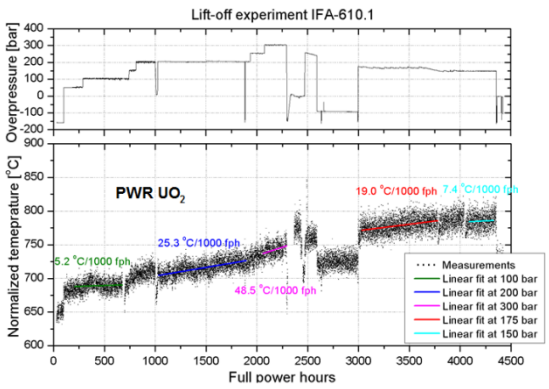

Figure 7 In-pile measurements in PWR "lift-off" study [3].

The normalized fuel temperature increased rates due to "lift-off" are determined at different overpressure steps. The detailed analysis for standard PWR fuel with Zr-4 cladding was presented in [3]. It was concluded that the lift-off effect for PWR fuel irradiated to a burnup of 52 MWd/kg UO_2 with standard Zr-4 cladding is 138±9 bar.

Further tests with PWR fuel confirmed the conclusion that the lift-off effect commences at the overpressure level of about 150 bar with a total pressure limit of about 300 bar.

A similar "lift-off" overpressure effect was obtained for the standard BWR fuel with Zr-2 cladding with a total pressure limit of about 220 bar.

The PWR (IFA-610.1 and IFA-610.5) and BWR data (IFA-610.10) are compiled in Figure 8 together with the data for the pre-irradiated VVER-440 fuel (IFA-610.11).

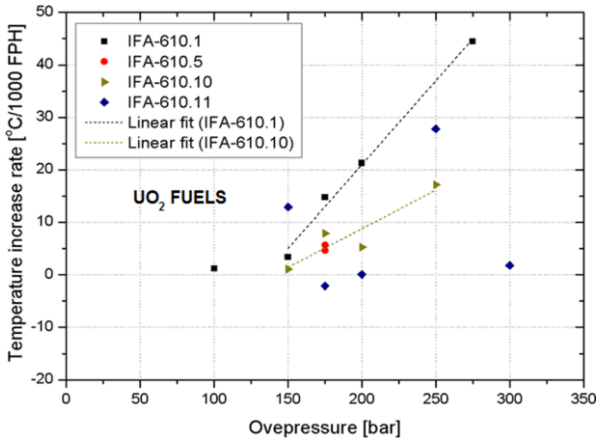

Figure 8 "Lift-off" criteria for different fuels tested in the Halden reactor.

The complete analysis of the "lift-off" tests shows that all of the following may influence the "lift-off" effect: cladding temperature; power rating before and during the tests; fuel burnup and fuel-clad bonding; the type of cladding including its texture as well as its hardening under irradiation.

The test series is being continued to determine the "lift-off" limits for different types of claddings at high burnup.

It should be noted that the "lift-off" testing programme is quite unique because it is performed at a total rod pressure of up to 450 bar which is well beyond the normal rod pressure estimated for the current accepted discharge burnup level in commercial NPPs.

3. Power ramp tests and determination of limits

The Halden ramp testing program was re-initiated after the Sudsvik reactor R2 was shut down in 2005. In order to cover all the test requirements a new test rig was developed which allows testing of up to 4 rods – one by one – within one reactor cycle. This is achieved by a specially designed hydraulic system which remotely controls the test rod's movement from a so-called "parking" position to the "ramp" position and back after the test. The "ramp" position is surrounded by a He-3 system which is effectively used for rapid power variation up to 100 W/cm/min during the ramp tests. This particular rig design has been developed to taking account of the typical Halden reactor experimental cycles and other tests that are being performed in the same cycle. The schematic view of the ramp rig is shown in Figure 9 together with an example of a fuel failure due to PCMI/PCI after one of the tests.

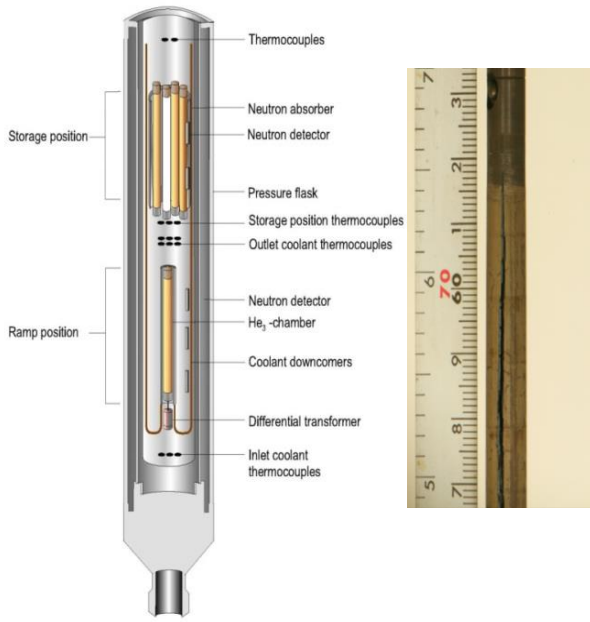

Figure 9 Power ramp test rig with He-3 system and possible PCI fuel failure under fast power ramps

The test rods are fitted with a cladding elongation detector (EC) in the bottom used for detection and measurement of PCMI during power ramping. A typical example of the in-pile measurements is shown in Figure 10 where cladding elongations were measured as a function of ALHR (Average Linear Heat Rating).

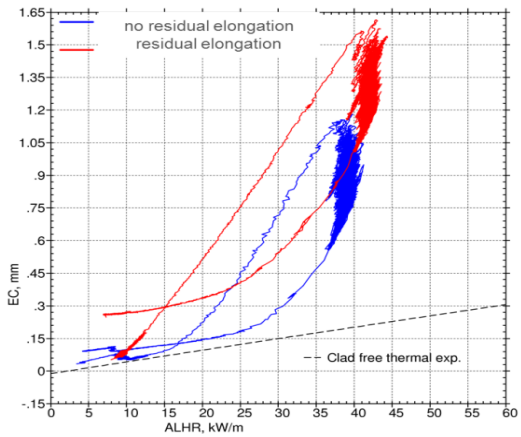

Figure 10 Typical results of the cladding elongation measurements during ramps with or without residual deformation due to PCMI

The EC measurements indicate a stress relaxation in the cladding due to PCMI and fuel pellet creep and also provide information on residual cladding elongation and eventual fuel failure. The latter is also detected by the gamma monitors installed in the loop systems.

Ramp tests are performed in Halden on a bilateral basis with the aim of providing information on the power rating limit due to failure as a function of rod burnup.

In addition, some of customized tests have been performed with new fuel cladding types, designed to mitigate the consequences of power ramps such as has been published in [4] and reproduced in Figure 11.

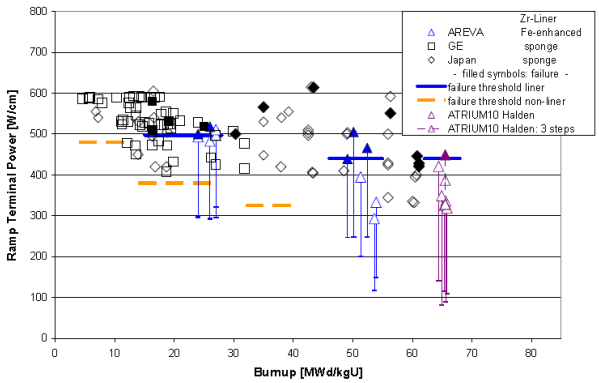

Figure 11 Published data on LHR limits vs. fuel burnup

4. Fuel fragmentation and relocation under LOCA

The Halden Reactor Project LOCA testing started about ten years ago, and since then fourteen tests have been executed. The main objective of the experiment series is to study the effect of fuel fragmentation, relocation and dispersal as one of the major issues of high burnup fuel behavior in a LOCA [5]. Other objectives are secondary hydriding of the cladding and fission product release.

The experiment setup consists of a single fuel rod inserted into a pressure flask connected to a water loop (Figure 12). A low level of nuclear power generation in the fuel rod (10 – 25 W/cm depending on required peak clad temperature to be reached) is used to simulate decay heat. The electrical heater surrounding the rod is simulating the heat from neighbour rods.

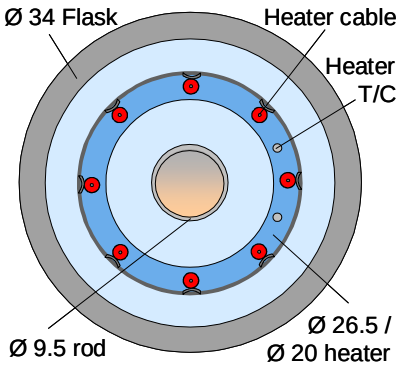

Figure. 12 Cross section of fuel pin, heater and pressure tube for HRP LOCA studies

The instrumentation consists of two cladding thermocouples (TC) at the upper part of the rod, one cladding TC at the lower part, three heater thermocouples at different axial elevations, a cladding extensometer and a rod pressure sensor.

An example of the in-pile measurements made during the test, simulating LOCA conditions is shown in Figure 13.

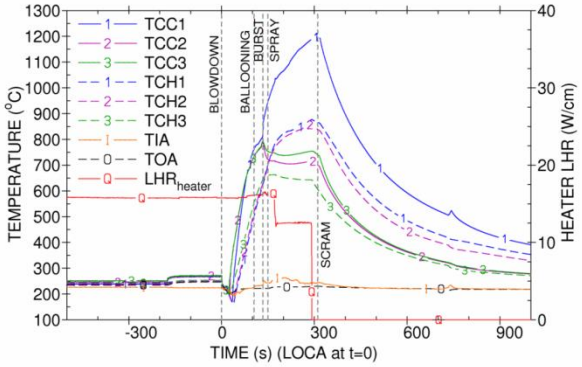

Figure 13 Example of the in-pile measurements for LOCA simulation test in the Halden reactor.

After the test, the main focus on detecting fuel fragmentation and relocation, both of which are examined first using non-destructive gamma scanning and neutron radiography followed by PIE - both illustrated in Figure 14.

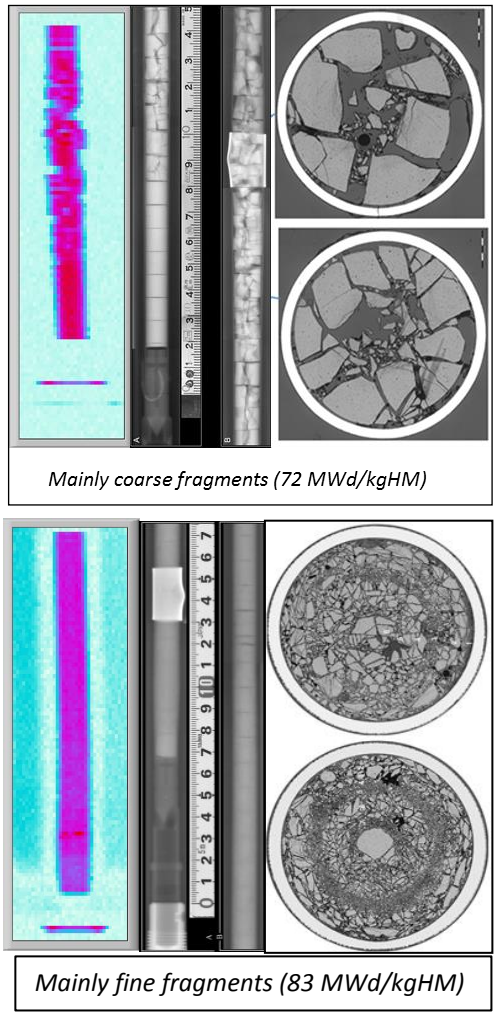

Fig. 14 Appearance of fuel fragmentation after LOCA testing in Halden

Gamma scanning is carried out a few days after test execution. The fuel segment is kept vertical in order to not change the fuel distribution in the cladding tube. The ragged

appearance of the fuel column in the upper gamma scan indicates coarse fragments which have moved laterally where ballooning has created some extra space. A small axial gap at the upper end indicates a little downward movement of the fuel column. The scan in the lower figure shows no discernible structure and is an example of fine fragmentation and no lateral movement due to little ballooning. In both cases, the horizontal gamma signal line at the bottom of the pressure flask indicates fuel fragment dispersal.

Neutron radiography is done a few months after test execution when the fuel has cooled sufficiently. The pictures show the same fuel as the gamma scanning. Neutron radiography reveals more details. Intact pellets and coarse fragments can be seen in the upper picture. The lower picture reveals that the lower part of the segment must be strongly fragmented since pellet-pellet interfaces and dishing are not visible, while they are discernible in the upper part.

Ceramography is the final examination. The ceramographies confirm the fragmentation deduced from gamma scanning and neutron radiography, but with much more detail at selected locations.

Fuel fragmentation is not an obvious function of a single parameter. The qualitative impression from eleven HRP LOCA tests with burnup from 44 to 92 MWd/kgHM is that up to about 60 MWd/kg, fragmentation is more or less caused by the cracking during normal operation. The test segments with burnup of 72 MWd/kg and higher show additional fragmentation which progressively affects more and more of the pellet as burnup increases. The formation of high burnup structure is playing an important role in this context. Another important influence on fuel fragmentation seems to be cladding distension. Where the pellets stay in contact with the cladding, they can remain quite unaffected by the LOCA transient and even form a plug that hinders gas flow and limits the amount of fuel available for relocation and dispersal. Pellet plugs and slow pressure drop were observed in three experiments. Since cladding distension is a sensitive function of the local temperature and since spacers improve the local heat transfer, it can be surmised that such plugs will form underneath the spacers. The effect will be studied in a future HRP LOCA test.

Axial fuel relocation as seen by PIE was limited in most cases except two with extreme burnups (90-92 MWd/kg). For the case with coarse fragments shown in Fig. 3, a local mass increase of about 10% in the balloon area can be deduced. However, this result is probably affected by handling and transport which cause fuel relocation in addition to the situation that developed in-core during the test.
The behaviour between 60 and 72 MWd/kg will be studied with the next test in the Halden reactor LOCA program.

5. Effect of dry-out on fuel integrity

Light water reactor cores may be subjected to thermal-hydraulic transients resulting in inadequate core cooling for short periods of time. The transient is terminated when adequate cooling is resumed, which results in the overheated fuel rods being quenched. It is a safety requirement that after such an event reasonable fuel performance be maintained up to the subsequent shutdown. In order to assess post dry-out and quench fuel performance, it is necessary to know what effect such transients have on the microstructural and mechanical properties of irradiated Zircaloy cladding. To this end a series of dry-out experiments were carried out at the OECD Halden Reactor Project [6]. An instrumented fuel assembly (IFA), connected to a light water BWR loop within the Halden reactor, was designed for in-pile dry-out testing. The main feature of the rig (Figure 15) was that it comprised three individual flow channels, each able to contain one instrumented test rod, allowing for individually controlled dry-outs to be performed.

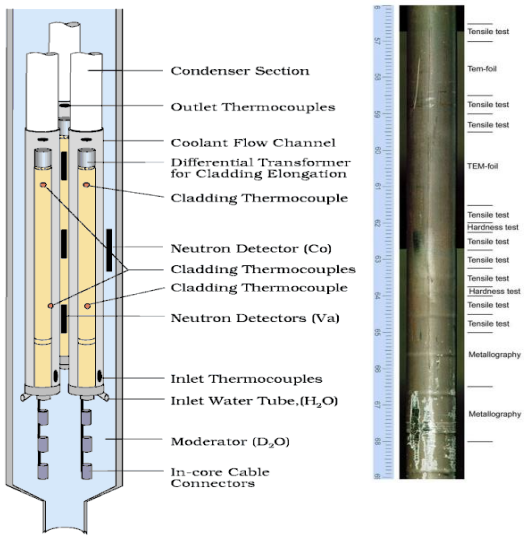

Figure 15 Test rig for "dry-out" testing and image showing the consequences for the fuel rod

Each rod was fitted with 2 or 3 Cr/Alumel thermocouples to monitor clad surface temperature during the dry-out events together with cladding elongation detectors.
Two fresh and six pre-irradiated (22-40 MWd/kgU) fuel rod segments were individually exposed to reduced or no-flow conditions in a heated light water loop within the Halden reactor. Dry-out occurred over the upper region of the six pre-irradiated rods, where peak clad temperatures (PCTs) in the range 950-1200°C were reached whilst PCTs of 750-850°C occurred in the other two fresh rods. All rods were quenched post dry-out and in some cases several dry-out events were initiated until sufficient accumulated time above the target temperature was reached as shown in Figure 16.

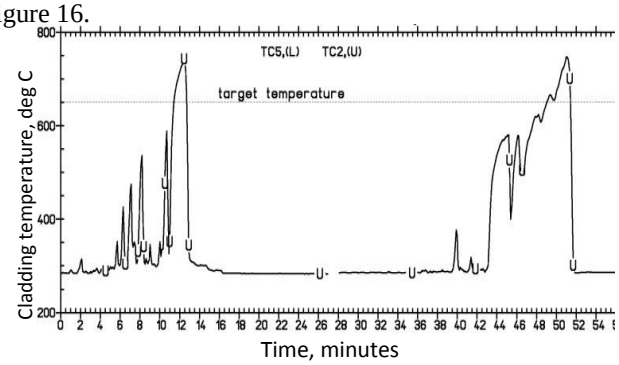

Figure 16 Example of the cladding temperature measurements during "Dry-out" test.

All rods were subjected to PIE in order to extend the data base of post in-pile transient clad properties. It was observed that significant improvement in clad ductility occurs after in-pile dry-out that results in an exclusively small α-phase grain structure being retained i.e. clad exposure temperature of 550-850°C. Multiple exposures taking the clad to above 1000°C (peak dry-out zone) with subsequent quenching, resulting in both the formation (at temperature) of an exclusively large grained β-phase microstructure, hydrogen pick-up with hydrides oriented along former β-grain boundaries and oxide spalling have a detrimental effect on the mechanical properties of fuel cladding. However, despite the induced decrease in clad ductility, in-pile fuel rod integrity was still maintained as evidenced by these rods being operated without failure for 23-30 days post-quench.

The observations during the series of dry-out tests also gave some ideas to use this kind of test not only for a cladding behavior study but also for studying the fuel, for example, to study transient FGR from "rim" fuel under high temperature exposure for commercially irradiated fuel rods.

III. INNOVATIVE DEVELOPMENTS

In addition to already established tests and methods for fuels and materials investigations beyond operational design limits, some innovative ideas for tests beyond safety limits have been proposed to be performed in Halden on a bilateral basis. The innovative experimental ideas were initiated by the events that occurred in the Fukushima Daiichi NPP, which was classified as a beyond design basis accident.

A test rig design with a single fuel rod similar to that designed for LOCA testing is considered feasible where advanced systems would be used allowing the study of fuel behavior in beyond design LOCA conditions (Figure 17).

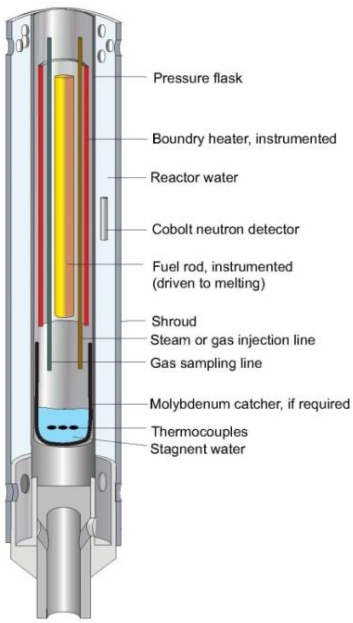

Figure17 Principal design of test rig proposed for studying LWR fuel behavior in beyond DB LOCA at Halden.

The main goal of the test is to expose the fuel rod to temperatures exceeding 1200^{0}C which is limited due to self-sustained steam-Zr oxidation at higher temperature. The test fuel rod may feasibly be instrumented to control the temperature until eventual cladding wreckage or fuel melting. The rig systems would be able to regulate the temperature by injection of different gases or steam providing required cooling of the rig and test rod. The gas sampling line would be used for analyzing gas release including hydrogen generation due to cladding high temperature oxidation.
The tests would be capable of answering the following questions:

- What finally may happen with the fuel rod?
- Until what stage may the rod geometry be maintained?
- What does the rod debris look like?
- How much H_2 or fission products including short-lived isotopes can be released?
- Are there other issues to be concerned about for such an accident scenario?

It is understandable that such a test should be well predefined and prepared to avoid any unsafe occurrences. Nevertheless such experimental principles may also be used for testing of Accident Tolerant Fuel which will be under development in forthcoming decades [7].

IV. CONCLUSIONS

The results from the many different types of tests performed in the Halden reactor are used for integral investigations of fuel behavior within design limits. These tests are important for understanding fuel behavior and can provide information enabling operational margins to be safely reduced and the utilization of fuel to be increased.

Other types of the tests, performed beyond operational and safety limits, are done in order to establish and possibly extend operational and safety limits. These tests may well be used for confirmation of the effectiveness of some innovative fuel developments aimed at extending safe utilization.

In addition, some innovative tests are proposed to extend the knowledge of fuel behavior beyond design limits like DB LOCA for example. The results from these tests may also promote the new Accident Tolerant Fuel developments.

ACKNOWLEDGMENTS

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The authors would like to thank personally C. Vitanza for his invaluable contribution to the tests performed in Halden, particularly for the thermal FGR threshold development and also for the principal proposals to perform some innovative tests including beyond DB LOCA simulation testing in the Halden reactor.

NOMENCLATURE

HRP – Halden Reactor Project
IFE-Institute for Energy Technology, Norway
FGR – Fission Gas Release;
PCMI – Pellet-Cladding Mechanical Interaction;
LHR – Linear Heat Rate;
ALHR- Average Linear Heat Rate
DB LOCA – Design Basis Loss Of Coolant Accident;
MOX – Mixed Oxide Fuel (UO_2 – PuO_2);
PCT – Peak Cladding Temperature.

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