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RESEARCH REACTORS FOR POWER REACTOR FUEL AND MATERIALS TESTING - STUDSVIK's EXPERIENCE

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ABSTRACT

Presently STUDSVIK's R2 test reactor is used for BWR and PWR fuel irradiations at constant power and under transient power conditions. Furthermore tests are performed with defective LWR fuel rods. Tests are also performed on different types of LWR cladding materials and structural materials including post-irradiation testing of materials irradiated at different temperatures and, in some cases, in different water chemistries and on fusion reactor materials. In the past, tests have also been performed on HTGR fuel and FBR fuel and materials under appropriate coolant, temperature and pressure conditions.

Fuel tests under development include extremely fast power ramps simulating some reactivity initiated accidents and stored energy (enthalpy) measurements. Materials tests under development include different types of in-pile tests including tests in the INCA (In-Core Autoclave) facility. The present and future demands on the test reactor fuel in all these cases are discussed.

1. Introduction

The R2 test reactor is owned by STUDSVIK AB, a commercial company, active in the areas of services, supply of special equipment and systems and also consulting. The company is performing R&D work and associated activities, primarily in the nuclear energy field. STUDSVIK NUCLEAR AB, which is the largest subsidiary within the STUDSVIK group, is one of the direct offsprings of AB Atomenergi, the origin of the STUDSVIK group, which was formed in 1947. The STUDSVIK group has about 700 employees and a turnover of about 500 MSEK/year.

During the 1950's and 60's, an ambitious nuclear program was launched in Sweden. The experience and competence gained from a large number of advanced projects constitutes the basis upon which the present activities of STUDSVIK NUCLEAR are based. Since the 1970's, the efforts have been concentrated on light water reactor fuel and materials, and the originally domestic R&D programs have been expanded so that a large fraction is now financed by non-Swedish sponsors.

2. The R2 Test Reactor

The R2 reactor is a tank-in-pool reactor in operation since 1960 and originally similar to the Oak Ridge Research Reactor, ORR [1]. The reactor core is contained within an aluminum vessel at one end of a large open pool, which also serves as a storage for spent fuel. Light water is used as core coolant and moderator. The reactor power was increased to 50 MW(th) in 1969. In 1984-85 a new reactor vessel was installed.

The R2 reactor has a high neutron flux, see Table 1, and special equipment for performing sophisticated in-pile experiments. An important feature of the reactor is that it is possible to run fuel experiments up to and beyond failure of the cladding, which is not possible in a commercial power reactor.

Table 1
Technical Data for the R2 Test Reactor

Power	50 MW(th)
Moderator/coolant	H_2O
Reflector	D_2O, Be
Fuel length	600 mm
Fuel assembly length	924 mm
Fuel assembly cross section	79x82 mm^2
Number of fuel plates per assembly	18
Neutron flux in experimental positions	
Thermal	(0.3-2.5) x 10^{14}n/(cm^2·sec)
Fast (>1 MeV)	(0.5-2.5) x 10^{14}n/(cm^2·sec)

The components of the core are arranged in an 8x10 lattice, typically comprising 46 fuel elements, 6 control rods, about 12 beryllium reflector assemblies and a number of in-pile loops, irradiation rigs and aluminum fillers.

The R2 driver fuel assemblies are, since the beginning of 1993, of the LEU type. They have 18 curved fuel plates containing an aluminum-clad aluminum/uranium-silicide matrix. The initial fuel content is 400g ^{235}U per fuel assembly, enriched to less than 20 %. The burnup of the spent fuel of this type reaches about 65 %.

The R2 reactor and some of its irradiation facilities have been described in the literature [2,3]. Most base irradiations of test fuel (irradiations at constant power, where fuel burnup is accumulated under well-defined conditions) are performed in boiling capsules (BOCA rigs). Some base irradiations and all ramp tests (irradiations under power changes) are performed in one of the two in-pile loops, which can be operated under either BWR or PWR pressure and temperature conditions. The ramp tests, simulating power transients in power reactor fuel, are achieved by the use of ^{3}He as a variable neutron absorber. Structural materials, such as samples of Zircaloy cladding, steels for pressure vessels and vessel internals and candidate materials for advanced reactors can also be irradiated in special rigs either in the loops or in special NaK-filled irradiation rigs in fuel element positions with a well-controlled irradiation temperature. Special equipment for in-pile corrosion experiments in the loops has recently been developed.

3. Fuel R&D - earlier work

Since the early 1970's, a long series of bilateral and international fuel R&D projects have been conducted under the management of STUDSVIK NUCLEAR [4-6]. These projects have been pursued under the sponsorship of different organizations: the bilateral projects mainly by fuel vendors and the international projects by different groups of fuel vendors, nuclear power utilities, national R&D organizations and, in some cases, licensing authorities in Europe, Japan and the U.S. In most of the projects, the clad failure occurrence was studied under power ramp conditions utilizing the special ramp test facilities of the R2 test reactor. In recent years the projects have not been limited to PCI/SCC (Pellet-Cladding Interaction/Stress Corrosion Cracking) studies. Some of them also included other aspects of fuel performance: end-of-life rod overpressure studies and defect fuel degradation experiments.

Ramp testing in the R2 test reactor began in 1969. In the present Ramp Test Facility, introduced in 1973, the fuel rod power during a ramp test in a loop is controlled by variation of the ^{3}He gas pressure in a stainless steel double minitube coil screen which surrounds the fuel rod test section. The principle of operation of this system is based on the fact that ^3He absorbs neutrons in proportion to its density, which can be varied as required by proper application of pressure. The efficiency of the ^3He neutron absorber system makes it possible to increase test rod power by a factor of 1.8 to 2.2 (depending on the fissile content of the fuel). In order to achieve a higher power increase than a factor of about 2, the reactor power must be increased before or simultaneously with the "^3He ramping". This technique with combined ramp systems is called "double step up-ramping" and makes it possible to increase the test fuel rod power by a factor of about 3. In the Ramp Test Facility ramp rates can be achieved in the range of 0.01 W/(cm·min) to about 3 000 W/(cm·min). The maximum achievable ramp terminal level depends on the neutron flux in the experimental position and on the fissile content in the test rod.

The rod overpressure experiments utilized the on-line measurements associated with the ramp tests combined with non-destructive examinations between reactor cycles and destructive examinations after the irradiation. When LWR fuel is used at higher and higher burnups the question of how the fuel might behave when the end-of-life rod internal pressure becomes greater than the system pressure attracts a considerable interest. On one hand end-of-life overpresure might lead to clad outward creep and an increased pellet-clad gap with consequent feedback in the form of increased fuel temperature, further fision gas release, further increases in overpressure etc. On the other hand increased fuel swelling might offset this mechanism.

The defect fuel degradation experiments also utilized the on-line measurements associated with the ramp tests and combined these with non-destructive and destructive examinations after the irradiation. Fretting type failures are predominant causes of the very few fuel failures that have occurred in recent years in LWRs. These primary failures are sometimes followed by secondary failures which frequently cause considerably larger activity releases. In such cases the subsequent degradation of the defect fuel rods by internal hydriding of the cladding and by oxidation of the fuel are the common destructive mechanisms.

During the 1970's extensive series of HTR fuel irradiations were performed in a special HTR gas loop system operating with on-line measurements and analyses of the released fission gas and of the fuel temperature.

4. Fuel R&D - upcoming work

The question of the ramp behavior in LWRs at "ultra-high" burnup (above 50 MWd/kgU) has been widely discussed in recent years as regards both normal and off-normal ramp rate conditions. However, only limited experimental information seems to be available at burnup levels beyond 30 MWd/kgU.

The concerns relate to the impact of changes in the physical properties of the fuel pellets at high burnup and their effects on the ramp behavior of the fuel rods. The fuel pellets tend to crack up in minor fragments and may no longer behave as solid bodies. The fission gases will be entrapped in a magnitude of small bubbles and might cause unacceptable fuel rod swelling on up-ramping. Other concerns relate to the loss of thermal conductivity and the impact of the rim zone on fuel ramp behavior.

The prospective ULTRA-RAMP project will constitute a combination of three groups of ramp projects. The ramp behavior and ramp resistance of current fuel types would be studied both under normal operating conditions ("slow" ramps), under off-normal operating conditions ("fast" ramps or transients simulating ramps corresponding to ANSI Class II and III events), and under "ultra-fast" conditions related to some ANSI Class IV events.

A few recent simulated RIA experiments (Reactivity Insertion Accidents) with high burnup fuel (55 and 65 MWd/t) have focussed interest on Class IV events. STUDSVIK is proposing a new type of "ultra-fast" ramps, faster than the fast ramps performed in earlier safety-related ramp projects but slower than the simulated RIA experiments. These new "ultra-fast" ramps could reach e.g. 100 kW/m during an 1 sec effective ramp time, corresponding to an enthalpy increase of 45 cal/g. Preparations for a demonstration experiment of this type are in progress.

The stored energy in fuel rods, the enthalpy, depends on the fuel design (dimensions, materials) and also burnup. This quantity is an important parameter in connection with safety considerations such as LOCA evaluations. The R2 test reactor is well suited for scram experiments where the thermal response of different types of fuel rods can be compared. The measurements can be performed by use of the R2 reactor's calorimetric rod power measurement system. Evaluation of an already performed demonstration experiment, STEED-I, on unirradiated fuel rodlets is in progress. An upcoming international project, STEED-II, will be based on tests of irradiated fuel rodlets.

5. Structural materials R&D

Specimens of structural materials are now irradiated either in rigs that only allow irradiations during whole 400-hr reactor cycles or in rigs where shorter irradiations, down to less than an hour, are possible. The specimens are either in direct contact with the loop water (temperature selected in the range 230 - 350 °C) or in some cases specimens of pressure vessel steels have been nickel plated in order to avoid corrosion problems during longer irradiations. In-pile rigs for fuel element positions are also used where the specimens are heated by gamma heating. In these rigs close temperature control (about ± 10 °C) has been maintained by placing the specimens in specimen holders filled with a NaK alloy. Varieties of these rigs are also used up to a temperature of 550 °C.

Recent work on stainless steels has to a large extent been concentrated on investigations of fusion reactor materials and on prospective FBR vessel materials. In this work tensile tests, fatigue tests and stress corrosion tests have been performed after irradiations to displacement doses of up to 10 dpa. Other types of post-irradiation tests, such as creep tests, CT tests and corrosion tests have also been performed. In-pile stress relaxation tests have also been performed. Presently in-pile creep tests are under development.

Experiments in the field of water chemistry and corrosion have been performed at Studsvik since the 1960's. A new facility, INCA (In-Core Autoclave) has been developed and put in operation [3]. The design of the facility is flexible in order to make it possible to rebuild it for different types of experiments, and has focused on the ability to control and monitor the water chemistry. The INCA is installed in one of the main in-pile loops. Test specimens and reference electrodes are installed in the facility. Degassed and deionized high purity water is fed into the facility. In order to establish a certain water chemistry different additives and impurities (H_2O_2, H_2, O_2, Li, B, Zn etc.) can be added to the system. The INCA facility can operate under both BWR and PWR conditions. Fast (>1 MeV) and thermal neutron fluxes up to 1.9 and 2.0×10^{14} n/cm^2, respectively can be achieved. The INCA facility is suitable for different kinds of experiments, for instance materials irradiations, waterside corrosion studies and in-core material testing, all under controlled water chemistry condition.

6. Future requirements on the R2 fuel

Potential requirements can be divided into different categories as follows:

* Future ramp tests on LWR fuel as regards
    ** Higher burnups
    ** Higher ramp terminals levels
    ** Higher ramp steps
    ** Higher ramp rates

* Other in-pile tests on fuel and structural materials
    ** Longer fuel cycles

Future ramp tests. So far ramp tests have mainly been performed on fuel with a burnup of up to about 50 MWd/kgU. In connection with future ramp tests burnups in the range of 60 to 100 MWd/kgU are now being discussed. In order to achieve the present ramp terminal levels on fuel with higher burnups and higher ramp terminal levels on fuel with the present burnups and other combinations of these parameters a higher thermal neutron flux is required. Thus a higher power density will require better heat transfer between fuel and coolant in the test reactor. In order to achieve higher ramp steps a shorter time constant is desirable.

Other in-pile tests on fuel. The Defect Fuel Degradation Experiments were discussed in Section 3. In these tests the special gas concentration gradients in the fuel-cladding gap, which act as a driving force for the phenomena under investigation, are eliminated when the reactor is shut down. Thus the present maximum test time is 400 hrs. Tests during longer times, e.g. 1000 hrs, would be desirable. Longer cycles would require a higher uranium contant in the fuel and possibly fuel with burnable absorbers.

Other in-pile tests on structural materials. In-pile tests like creep tests, relaxation tests and corrosion tests would also benefit from longer reactor cycles, e.g. 1000 hrs and in some cases from higher fast neutron fluxes.

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7. References

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