XA9846753

BWR FUEL R&D PROGRAMS AT STUDSVIK

M. GROUNES, C. GRÄSLUND, G. LYSELL,
A. LASSING, T. UNGER, M. CARLSSON
Studsvik Nuclear AB, Nyköping,
Sweden

Abstract

The different types of BWR fuel R&D programs at Studsvik are discussed. The R2 test reactor, its facilities for irradiation experiments and post-irradiation examinations are described.Reviews are given of some of STUDSVIK's earlier international fuel R&D programs, of STUDSVIK's defect fuel degradation experiments and of the following new upcoming international fuel R&D projects: the DEFEX II, ULTRA-RAMP, SUPER-RAMP III/10x10, STEED, Post-DO/DNB and TRANS-RAMP III Projects.

1 Fuel R&D Programs at Studsvik – Introduction

1.1 Background

Studsvik AB, the parent company in the STUDSVIK group, is partly owned by Vattenfall AB, and is performing R&D work and associated activities, primarily in the nuclear energy field. Studsvik AB is a commercial company, active in the areas of services, supply of special equipment and systems and also consulting. 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 580 employees and a turnover of about 425 MSEK/year.

The facilities of interest in this connection are the R2 test reactor, the Hot Cell Laboratory and various other laboratories, all located at Studsvik, 100 kilometers south of Stockholm, Sweden.

1.2 Fuel Testing

STUDSVIK NUCLEAR's R&D work in the area of fuel testing was started in the early 1960's. In a very general sense the purpose of fuel testing can be described as follows:

- Increasing of reactor availability by decreasing fuel-related operational power restrictions, defining the operational power limits.

- Acquisition of experimental data for fuel-related safety considerations.

- Decrease of fuel costs by making increases in fuel burnup possible.

The fuel testing activities can be divided into a number of well-defined steps as follows:

- Base irradiation, performed
  - in a power reactor, or
  - in STUDSVIK's R2 test reactor.

- Power ramping and/or other in-pile measurements, performed
  - in STUDSVIK's R2 test reactor.

- Non-destructive testing between different phases of an experiment, performed
  - in STUDSVIK's R2 test reactor pool, or
  - in STUDSVIK's Hot Cell Laboratory.

- Destructive post-irradiation examinations, performed
    - in STUDSVIK's Hot Cell Laboratory, or
    - in the sponsor's hot cell laboratory.

Fuel examination can be performed on standard (full-size) fuel rods from power reactors, which can be investigated in the Hot Cell Laboratory. If required, some types of tests could also be performed on such fuel rods in the R2 test reactor. However, due to the rather large initiation costs, such tests have not yet been performed. It should be noted, however, that short fuel rodlets, suitable for ramp testing and other on-line measurements in the R2 test reactor, are routinely fabricated from irradiated full-size power reactor fuel rods by the STUDFAB refabrication process.

Fuel testing in the R2 test reactor is usually performed on fuel segments (rodlets) of 300-1000 mm length. However, tests have also been performed on full-size demonstration reactor fuel rods with up to 2.5 m length. In those cases only the lower 0.6 meters were irradiated. Irradiation at constant power is performed in boiling capsules (BOCA rigs) in fuel element positions, or in pressurized water in-pile loops operating under BWR (or PWR) pressure/temperature conditions, as described in Section 2.

Ramp tests, incorporating a very fast-responding test rod power measuring system and associated on-line measurements, such as rod elongation and noise measurements for studies of the rod thermal performance, are performed in the in-pile loops. The ramp tests are a form of integral performance tests where the complex interplay between the pellets and the cladding of a power reactor fuel rod is reproduced. The primary test objectives are:

- Determination of the failure boundary and the failure threshold, see Figures 1 [1] and 2.

- Establishing of the highest "conditioning" ramp rate that safely avoids failure occurrence.

- Study of the failure initiation and progression under short time over-power transient operation beyond the failure threshold.

- Proof testing of potential pellet-clad interaction (PCI) remedies.

Other, more specific test objectives have also been pursued in some ramp projects.

The rod overpressure experiments utilize 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 overpressure might lead to clad outward creep and an increased pellet-clad gap with consequent feedback in the form of increased fuel temperature, further fission gas release, further increases in overpressure etc. On the other hand increased fuel swelling might offset this mechanism. In connection with such considerations the Rod OverPressure Experiments were initiated. They will be discussed in Section 4.4.

The defect fuel degradation experiments also utilize the on-line measurements associated with the ramp tests and combine these with non-destructive and destructive examinations after the irradiation. Fretting type failures are predominant causes of the 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. In these tests an irradiation test scheme, adapted to the experimental conditions in the R2 test reactor was introduced. This scheme offers the possibility of executing comparative investigations of the process of degradation of commercial types of LWR fuel under simulated primary defect conditions as well as of the mechanisms involved. These experiments will be discussed more in detail in Section 5.

The fuel testing projects executed at Studsvik have been organized under three different types of sponsorship:

International (multilateral) fuel projects

- Jointly sponsored internationally on a world-wide basis.
- Project information remains restricted to the project participants throughout the project's duration and some predetermined time after project completion.

Bilateral fuel projects

- Sponsored by one single organization, or a few co-operating organizations.
- Project information remains restricted to the sponsor, sometimes published later.

In-house R&D work

- Sponsored by STUDSVIK NUCLEAR.

Several new hot-lab techniques have also been introduced in recent years [2]. The STUDFAB process for refabrication of rodlets from full-size irradiated fuel was mentioned above. Fuel ceramography can include scanning electron microscopy (SEM) and electron probe microanalysis (EPMA).

Descriptions of the fuel testing facilities and the associated techniques will be given in Section 2. The noise measurements introduced for studies of the rod thermal performance have been described elsewhere [3-5]. Several other novel testing techniques have also been introduced [6]. A very fast ramp rate, up to 3 000 W/(cm·min) can be used to obtain fast power transients and to determine the pellet-clad interaction/stress-corrosion cracking (PCI/SCC) failure boundary. Still faster, "ultra-fast" ramps, are discussed in Section 6.2. On-line elongation measurements can be performed during ramp tests, Figure 3. Test fuel rodlets can be fitted with on-line pressure transducers through a refabrication process.

Since the early 1970's, a long series of bilateral and international fuel R&D projects, primarily addressing the PCI/SCC failure phenomenon have been conducted under the management of STUDSVIK NUCLEAR [1, 6-8]. These projects are pursued under the sponsorship of different organizations; the bilateral projects mainly different fuel vendors and the international projects 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 reactor. As mentioned the current projects are not limited to PCI/SCC studies but also include other aspects of fuel performance, such as end-of-life rod overpressure [9-12] and defect fuel degradation [13-16]. An overview of the projects that have been completed and those that are currently in progress or planned has been published [8], recent and upcoming projects will be discussed in this paper. In most cases, the test fuel was base irradiated in commercially operating light water power reactors. In some instances, however, the base irradiation took place in BOCA rigs in the R2 reactor.

In general, the international fuel R&D projects can be divided into two main categories:

- Projects aimed at decreasing the fuel costs by increasing fuel utilization and reactor availability.
- Projects providing data for fuel-related safety considerations.

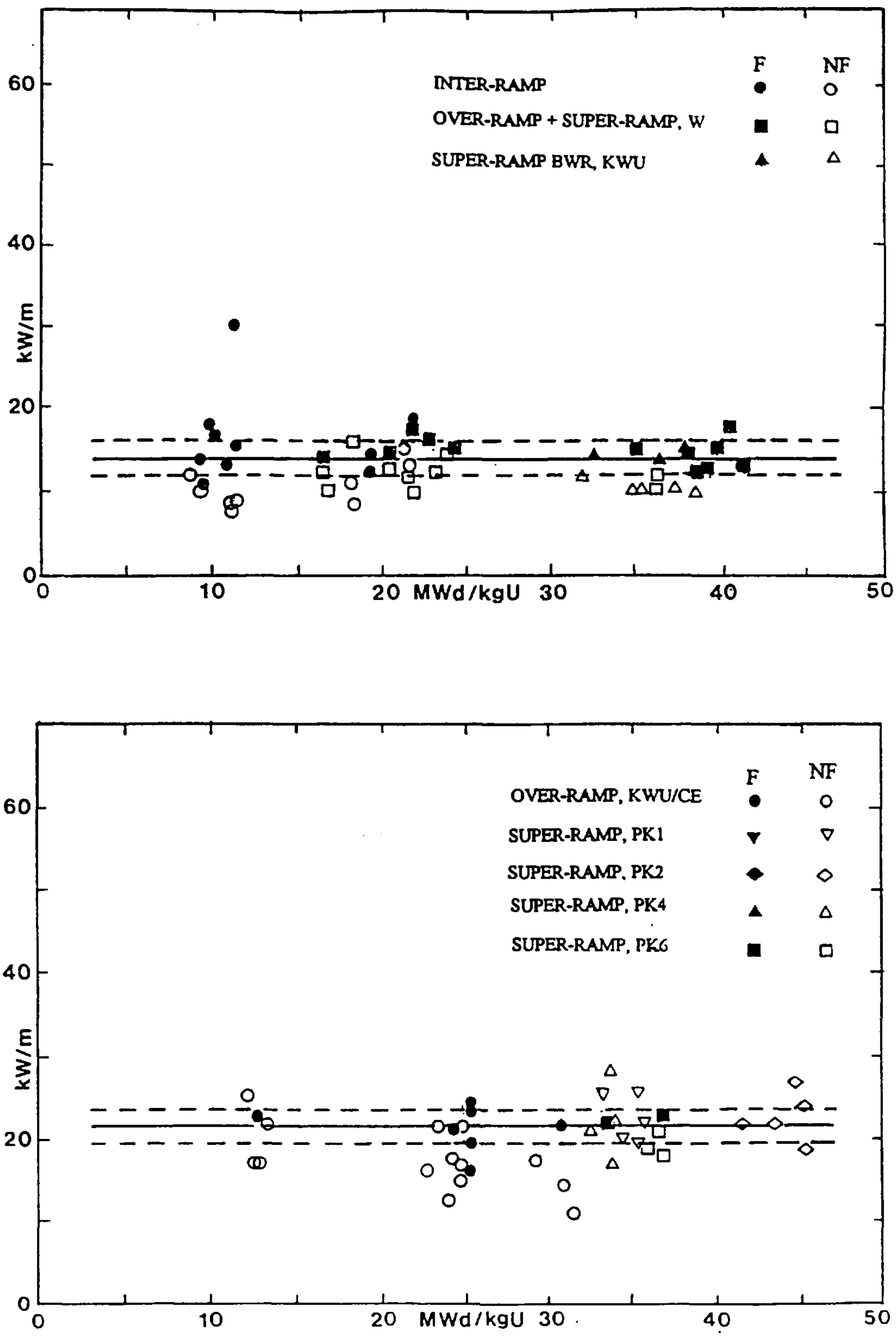

Figure 1
Summary of Some Data From the INTER-RAMP, OVER-RAMP and SUPER-RAMP Projects. The Incremental Failure Threshold as a Function of Burn-Up for Different Groups of Fuel Rods.

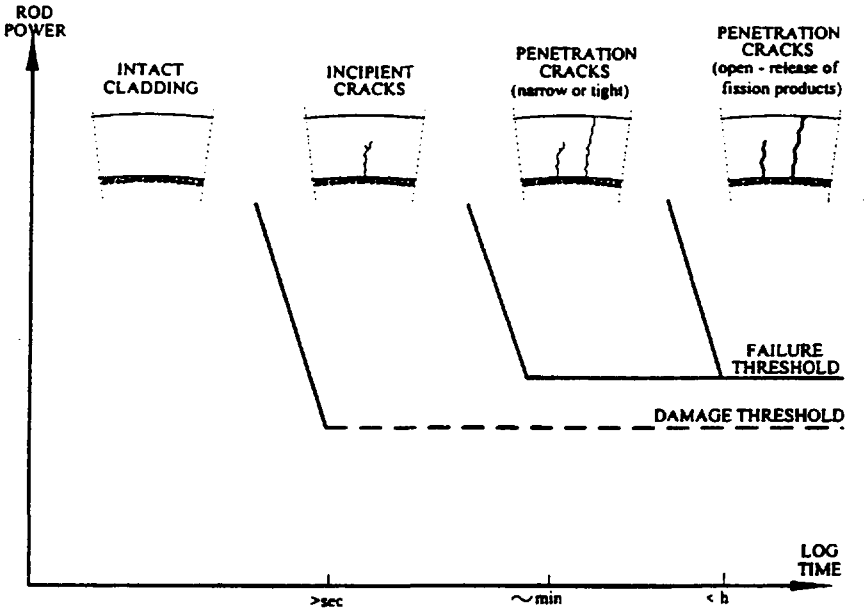

Figure 2
Schematic PCI Failure Progression Diagram.

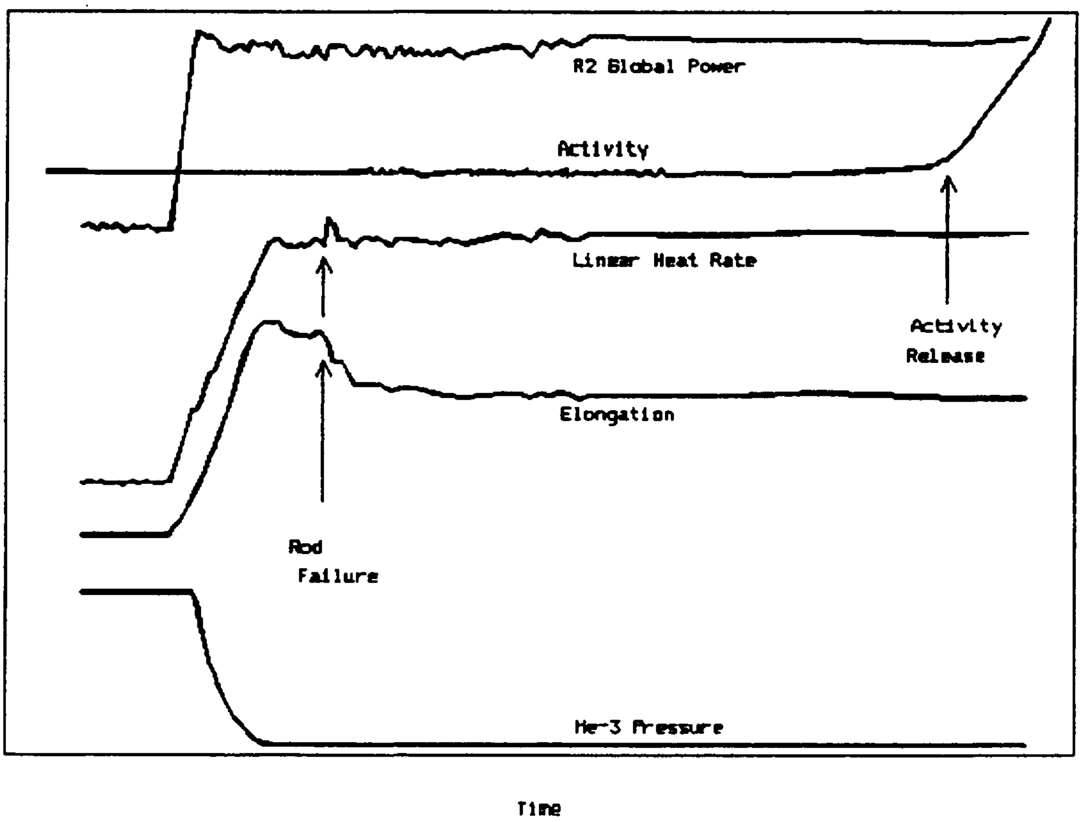

Figure 3
On-Line Measurements During a Ramp Test Showing a PCI Failure Event.

A typical example of the former category is the SUPER-RAMP project [17], where several groups of fuel from different fuel vendors and with different "PCI remedies" were tested. A summary of some of the data from this category of projects is shown in Figure 1 [1]. Reviews of the projects in the latter category have also been published [18-19]. Presently a new international project, combining the features of these two series, is under discussion: the ULTRA-RAMP project, to be discussed in Section 6.2.

The test data are often used as "benchmarking" data in the project participants' own fuel modeling work. In recent years many ramp test data have also been analyzed with the INTERPIN code, developed by STUDSVIK [20-22]. INTERPIN is a fuel performance code which satisfies real-time simulation requirements when implemented on a minicomputer.

European, Japanese and U.S. fuel manufacturers and research organizations have also for many years been utilizing the R2 test reactor and the associated hot-cell laboratories for bilaterally sponsored research[1)]. ABB Atom AB has made many series of ramp tests. General Electric Co. has executed several series of ramp tests at R2, as part of the efforts to develop the zirconium barrier fuel concept. Some of the ramp techniques requested were innovative, for example the "double ramping" of the test rods. Other major customers are Exxon Nuclear Co. (later Advanced Nuclear Fuels Corporation, later Siemens Nuclear Power Corporation, now Siemens Power Corporation), B&W Fuel Company (now Framatome Cogema Fuels), Hitachi Ltd., Mitsubishi Heavy Industries, Ltd., and Toshiba Corporation. Tests have also been performed on behalf of other organizations but the results have not always been published.

STUDSVIK NUCLEAR's in-house R&D work is mainly associated with improvements of test irradiation techniques, instrumentation and post-irradiation examination, all in support of ongoing or upcoming irradiation projects. Progress in these areas has made it possible to achieve important progress in fuel research. For example, the characterization of the PCI failure progression in some recent projects was only made feasible through a combination of several new techniques. These included a very fast ramp execution using a ramp rate of up to 3 000 W/(cm·min), compared to the previous maximum of 200 W/(cm·min), a prompt detection of the through-failure event using the on-line elongation detector and a subsequent special clad bore inspection technique. Another result of the in-house R&D work is the noise measurement technique mentioned above.

STUDSVIK NUCLEAR has also been carrying out an in-house R&D program aimed at improving the performance of LWR fuel by the utilization of a design concept with cladding tubes which have been "rifled" on a micro-scale [23]. Results of R2 irradiations of such fuel and the associated modeling work have been published [24-31].

2 The R2 Test Reactor - a Versatile Tool for Fuel R&D

The R2 reactor is a 50 MW(th) materials testing and research reactor developed in the USA and in operation since 1960, Figure 4. The R2 reactor has a high neutron flux, and special equipment for performing sophisticated in-pile experiments. An important feature of the R2 test reactor is that it is possible to run fuel experiments up to and beyond failure of the cladding. This is obviously not possible in a commercial power reactor. Detailed descriptions of the R2 reactor have been published elsewhere [32-33]. Fuel tests are performed in the two high pressure loops in the core. These in-pile loops can be operated under BWR pressure and temperature conditions and are used for all irradiations under power changes, some of the base irradiations, some materials testing experiments and the in-pile corrosion experiments, discussed in Section 3. Most base irradiations of test fuel, i.e. irradiations at constant power, where fuel burnup is accumulated under well defined conditions, are performed in Boiling Capsules (BOCA rigs). The R2 core has an active length of 60 cm.

2.2 Ramp Test Facility

Ramp testing in the R2 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

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1) A list of available publications can be obtained upon request from the authors.

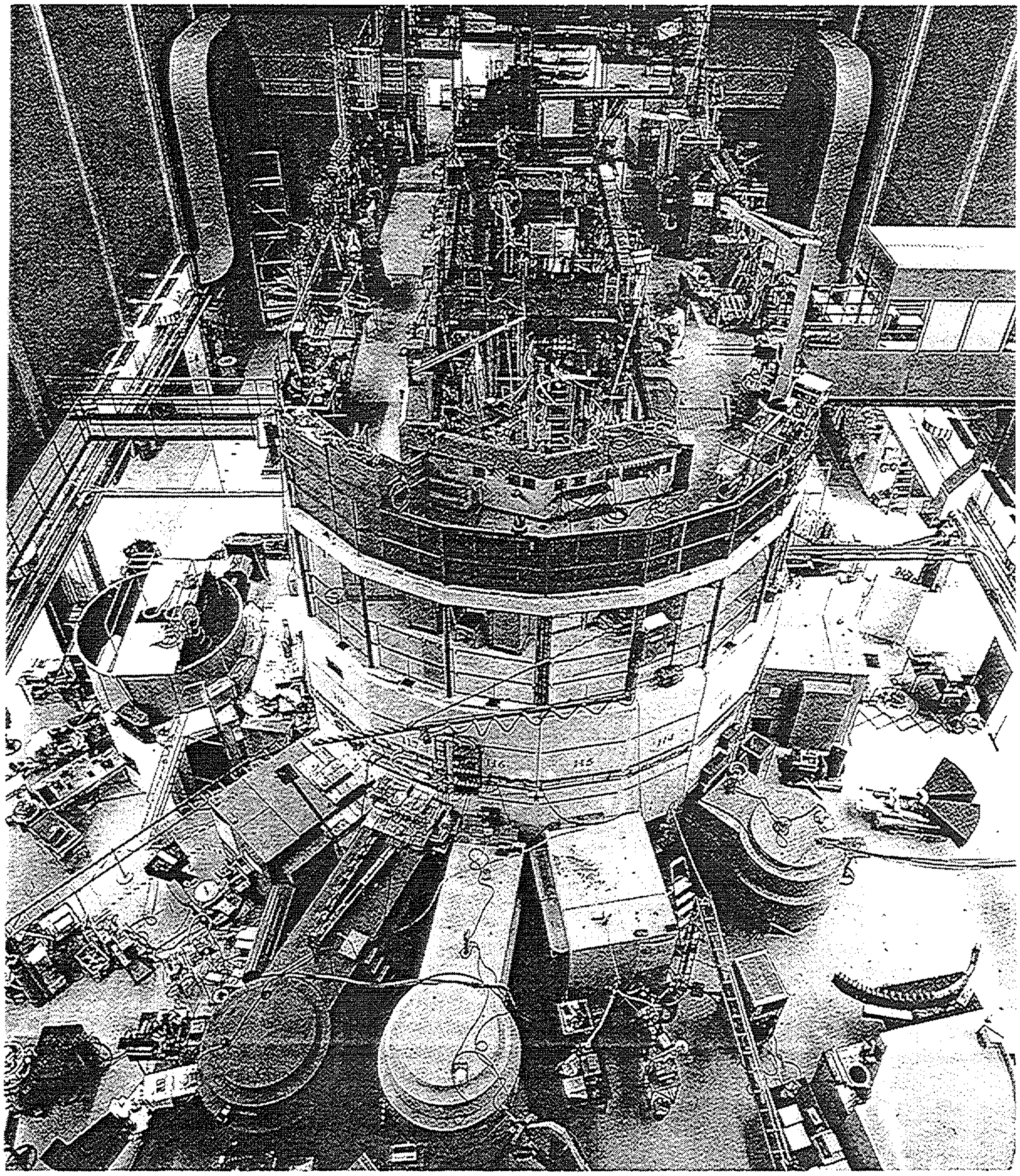

Figure 4
A General View of the R2 Test Reactor.

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). The ^3He absorber system is designed to achieve a 100 % power increase within 90 seconds, when operating with the normal pressure variation (bellows system).

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 "^{3}He ramping". This technique with combined ramp systems is called "double step up-ramping". The technique 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).

2.3 Boiling Capsules - BOCA Rigs

The Boiling Capsule (BOCA) facility is used for irradiations at constant power where fuel burnup is accumulated under well-defined conditions. It was introduced in 1973.

Up to five BOCA rigs can be operated simultaneously in the reactor. Two independent pressurization systems are available, each capable of supplying 3 to 5 BOCA rigs with water. Each BOCA rig is connected to a separate outlet circuit.

3 INCA - A New Facility for Waterside Corrosion Studies in the R2 Test Reactor

The experimental program at Studsvik has recently been extended to include investigations on clad and structural materials in an in-pile corrosion rig, INCA (In-Core Autoclave), designed to make it possible to study the in-core corrosion behavior of stainless steel and zirconium alloys.

A reference electrode for long term in-pile corrosion potential measurements has been developed. Such an electrode together with a water supply and analysis system will make it possible to perform long-term irradiations of materials under well-defined water chemistry conditions in a BWR environment. Possibilities to move the electrode and the test specimens up and down in the core could easily be introduced in order to monitor the corrosion potential changes due to radiolysis.

Waterside corrosion becomes a more and more important phenomenon with increasing power plant efficiency. The more demanding operating conditions, such as higher fuel burnup and longer in-reactor fuel cycles, make the cladding corrosion performance a limiting factor.

A number of different conditions influence the corrosion and hydriding behavior of cladding materials:

- Water chemistry
- Crud deposits
- Material characteristics, e.g. annealing treatments
- Neutron dose
- Hydrogen pick-up
- Boiling conditions

The new INCA facility has been developed in collaboration with our sister company Studsvik Material AB. The main feature of the new facility is the ability to control and monitor the water chemistry. Therefore the facility is of the once-through type, which means that the rig is supported by a water supply system of its own and that the water passes the rig only once. The desired water chemistry is created by adding impurities and additives to a purified water flow, close to the test area. This technique has successfully been used out-of-pile by Studsvik Material AB to obtain well characterized conditions. The facility has a flexible design and can easily be modified to suit different types of corrosion and water chemistry experiments.

The corrosion test rig, which can be seen in Figure 5, is the in-pile part of the INCA facility and is installed in one of the main in-pile loops in the R2 test reactor. It consists of two major parts, the rig tube and the electrode rod.

The rig tube separates the water system of the test rig from the in-pile loop main flow. The inlet water to the rig, which is degassed high purity water, is fed from a separate water supply system. It is heated by the loop water in a preheater coil and subsequently led into the rig tube. Additives and impurities (oxygen, hydrogen peroxide, zinc etc) can be added both before the rig and inside the rig just before the in-core test

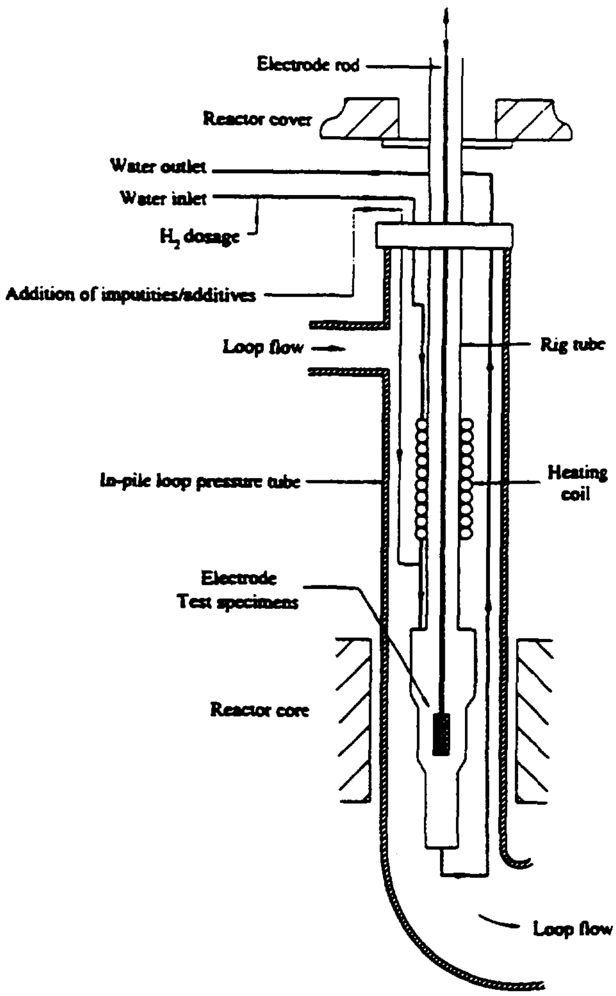

Figure 5
INCA – Conceptual Test Rig Arrangement.

area in order to establish the desired water chemistry. The presently available inner diameter of the rig tube is 21 mm, but in the future it will be possible to vary the diameter or to have different diameters in different axial sections of the core.

The electrode rod is installed in the rig tube and is a carrier for the test specimens, the reference electrodes etc. The tube for the injection flow is also assembled on the rod. This arrangement makes it possible to change the electrode rod from one reactor cycle to another. For the moment the rod is bolted on top of the rig tube, but it could easily be rebuilt to be movable up and down in the core section and below the core during operation.

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($cm^2 \cdot s$) respectively, can be achieved.

The facility is suitable for different kinds of experiments, for instance materials irradiations, waterside corrosion studies and in-core materials testing, all under controlled water chemistry. It has been in operation since March 1995. One of the objectives has been to develop reference electrodes for long term in-pile use. A radiolysis study where experimental measurements and computer results were compared has also been performed.

4 STUDSVIK's International Fuel R&D Programs

4.1 Introduction

A long series of international fuel R&D projects, primarily addressing the PCI/SCC (Pellet-Cladding Interaction/Stress Corrosion Cracking) failure phenomenon have been conducted under the management of STUDSVIK NUCLEAR since 1975. 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. The recent projects have not been limited to PCI/SCC studies but some of them also include other aspects of fuel performance, to be discussed below. During the late 1970's and 1980's the series of international PCI ramp projects branched out in two directions. One series was initially concentrated on the PCI phenomena under normal operational conditions in different types of BWR fuel rods subjected to increased burnup. These projects were in a broad sense aimed at decreasing fuel costs by increasing fuel utilization and reactor availability. Most of the bilateral fuel projects fall into this category. The other series of international projects was concentrated on more safety-oriented issues, aimed at providing data for fuel-related safety considerations. The first-mentioned series of international projects includes the INTER-RAMP, OVER-RAMP, DEMO-RAMP I, SUPER-RAMP, SUPER-RAMP EXTENSION, SUPER-RAMP II/9x9 and SUPER-RAMP III/10x10[1] projects. The second series includes the INTER-RAMP, DEMO-RAMP II, TRANS-RAMP I and TRANS-RAMP III[1] projects. Presently a new international project, combining the features of these two series, is under discussion: the ULTRA-RAMP project[1].

In later years two other types of fuels R&D project have been introduced: end-of-life rod overpressure studies and defect fuel degradation experiments. The BWR end-of-life overpressure project was designated ROPE I (Rod OverPressure Experiment). The defect fuel degradation projects are designated DEFEX and DEFEX II[1].

Furthermore other types of projects are also under discussion, they will be described in Section 6. An overview of the 9 international BWR projects that have been completed is given in Table 1.

4.2 Ramp Resistance - BWR Fuel

The first of STUDSVIK NUCLEAR's international fuel projects, INTER-RAMP, was executed in 1975-79 under the sponsorship of 14 organizations from 9 countries. The results have been described [34-35]. The main objectives of the program were to investigate systematically the failure propensity and associated phenomena of well-characterized BWR fuel rodlets when subjected to fast overpower ramps under relevant and well-controlled experimental conditions.

The DEMO-RAMP I project was executed in 1979-82. The results have been described [36]. The main objective of this program was to investigate the effects of two PCI remedies, annular pellets and niobia doping of the UO_2, on the ramp behavior, especially the fission product release and the pellet-cladding mechanical interaction (PCMI), of 8x8 type fuel rodlets.

The SUPER-RAMP project was executed in 1980-83. The results have been described [17, 37]. The main objective of this project, which was co-sponsored by 20 organizations from 11 countries, was to make a valid contribution to the general understanding of the PCI phenomenon for commercial type LWR fuel rods at (then) high burnup levels under power ramp conditions. For the BWR subprogram the more specific objectives were to

- Establish the PCI failure threshold for standard type test fuel rods on fast power ramping at burnup levels exceeding about 30 MWd/kgU.

- Identify any change in failure propensity or failure mode as compared to the failure behavior at lower burnup levels.

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[1] These projects will be discussed in Section 6.

- Establish a failure-safe reduced power ramp rate for passing through the PCI failure region.

In the SUPER-RAMP EXTENSION project, which was executed in 1984-86, further ramp tests were performed.

In the SUPER-RAMP II/9x9 project, executed in 1987-90, the failure boundary of 9x9 type fuel rodlets, irradiated in a commercial power reactor to a burnup of 25 MWd/kgU, was determinad [38], see Figure 6.

4.3 Safety-Oriented Ramp Resistance Studies

In a series of international fuel research projects (the INTER-RAMP, DEMO-RAMP II, and TRANS-RAMP I projects) it was demonstrated by means of power transient tests (intentionally interrupted power ramp tests) that when BWR test fuel rods were exposed to overpower ramps of increased severity, they exhibited a regular Pellet-Clad Interaction (PCI) failure progression (Figure 2). A higher transient peak power level resulted in an earlier fission product leakage from the fuel rods. Stress corrosion cracks (SCC) initiated promptly on fast up-ramping, i.e. within the order of seconds, and penetrated the cladding wall within about a minute. Depending on the actual power "over-shoot" and the time spent beyond the failure threshold, the transient passed consecutively through a number of power-time regions defining the progressive steps of the failure process.

Some of the BWR fault transients of the types that might be expected to occur once in a reactor year or once in a reactor lifetime carry a potential for causing PCI fuel clad damage or failure (i.e. through-wall crack penetration) on surpassing the PCI failure threshold. A question of prime concern is then whether a fast single transient of the type mentioned will result in fuel failure due to PCI, eventually followed by a release of radioactivity to the coolant.

In the INTER-RAMP (IR) Project, executed during 1975-79 [34, 35], BWR fuel rods were subjected to power transients of varying "over-power" levels beyond the PCI failure threshold, where cladding failure and fission product release occur after a sufficient time. The results demonstrated a systematic time dependence of the fission product release to the coolant from the failed fuel rods. An increase in the power "over-shoot" of 5 kW/m caused a decrease of the time to fission product release by a factor of about 10.

In the DEMO-RAMP II (DRII) Project, 1980-1982 [39], BWR fuel rods of intermediate burnup levels were subjected to intentionally interrupted short-time power transients at linear heat ratings a few kW/m above the PCI failure threshold. No cladding failures were detected after the transients but a large number of non-penetrating (incipient) cracks were observed. They had been formed very rapidly, within a minute. These cracks could be observed by destructive post-irradiation examinations only. The crack depths ranged from 10 to 60 percent of the cladding wall thickness.

In the TRANS-RAMP I (TRI) Project, 1982-1984 [40], BWR fuel rods of intermediate burnup levels were subjected to simulated short time power reactor transients of a wide range of "over-powers" but at characteristic very fast ramp rates, in the range of 10 000 W/(cm·min). The test results were similar to the DR II results and permitted a tentative interpretation of the PCI failure progression in terms of well-separated power/time boundaries defining 1) crack initiation at the inside surface of the cladding, 2) through-wall crack penetration and 3) leakage of fission products to the coolant water.

The crack initiation and penetration processes can only be detected by special hot cell laboratory or test reactor techniques. The delay of the fission product release indicates that in power reactors cladding failures that occur during fast transients and terminate before any outleakage of fission products may go undetected until manifested in later operational manoeuvers.

4.4 Studies of "Lift-Off" Phenomena

When LWR fuel is used at increasingly 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.

Table 1
Overview of STUDSVIK NUCLEAR's International BWR Fuel R&D Projects 1975-1991

Project (duration)	Fuel Type (No of rods)	Base Irradiation (MWd/kgU)	Research Objectives	Data published
INTER-RAMP (1975-79)	BWR (20)	R2 (10-20)	Failure threshold Failure mechanism Clad heat treatment Modeling data	Yes (Ref 1,2)*)
DEMO-RAMP I (1979-82)	BWR (5)	Ringhals I (15)	PCI remedies (Annular, niobia doped pellets)	Yes (Ref 3)
DEMO-RAMP II (1980-82)	BWR (8)	Würgassen (25-29)	Failure threshold PCI damage by overpower transients	Yes (Ref 4,5)
SUPER-RAMP (1980-83)	BWR (16)	Würgassen (30-35) Monticello (30)	Failure threshold High burn-up effects PCI remedies Safe ramp rate Gd fuel	Yes (Ref 6)
SUPER-RAMP EXTENSION (1984-86)	BWR (9)	Oskarshamn 2 (27-31)	Safe ramp rate	No
TRANS-RAMP I (1982-84)	BWR (5)	Würgassen (18)	Failure boundary Crack init. and prop. Structural changes Fission gas release Modeling data	Yes (Ref 4,7)
ROPE I (1986-89)	BWR (4)	Ringhals (36)	Investigate clad creep-out as a function of rod overpressure	Yes (Ref 8)
SUPER-RAMP II/9x9 (1987-91)	BWR (4)	Dresden (30)	PCI performance	Yes (Ref 9)
DEFEX (1993-95)	BWR (6)	Initially unirradiated rodlets	Study secondary damage formation in fuel rods with simulated fretting defects	No

*) References, see next page.

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On one hand end-of-life overpressure might lead to clad outward creep and an increased pellet-clad gap with consequent feedback in the form of increased fuel temperature, further fission gas release, further increases in overpressure etc. On the other hand increased fuel swelling might offset this mechanism.

In connection with such considerations STUDSVIK NUCLEAR initiated a limited in-house R&D activity in order to investigate the ability to detect the phenomena of interest: the ROPE Pre-project [9]. This was followed by an international Rod OverPressure Experiment (ROPE I project).

The purpose of the ROPE I project was to investigate the behavior of BWR fuel rods. ABB-ATOM 8x8 fuel rods, irradiated in the Ringhals 1 reactor to a burnup of about 35 MWd/kgU were tested [12]. The rods

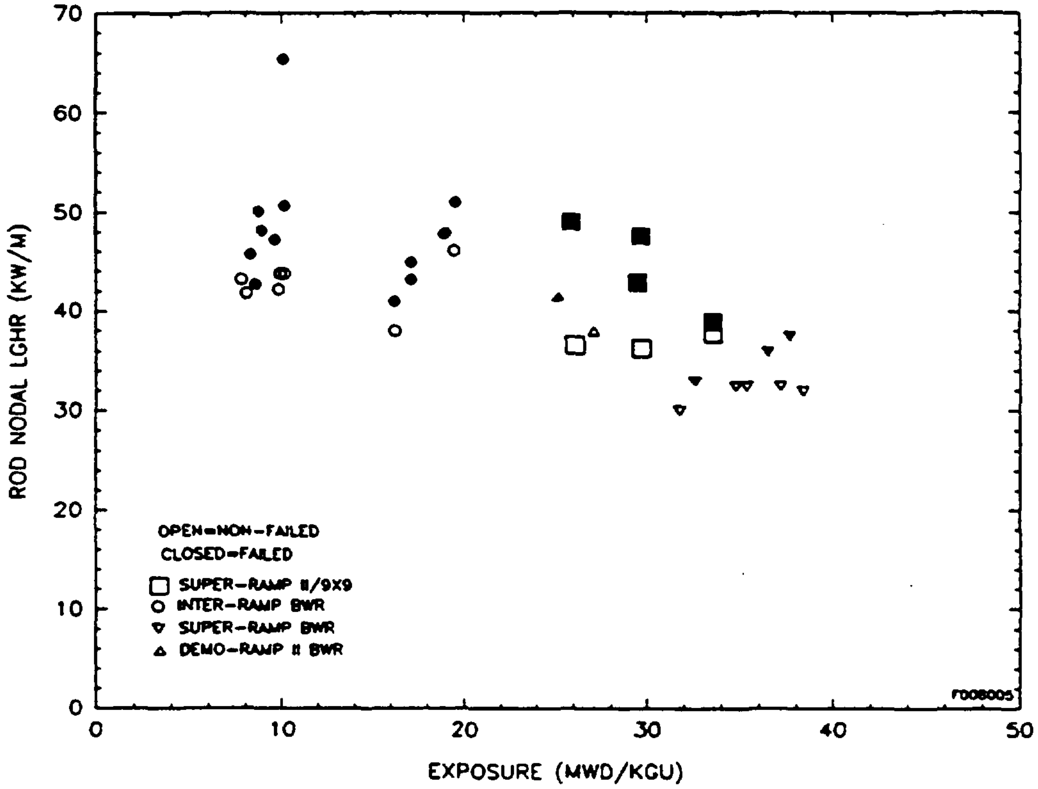

Figure 6
Comparison of 9x9 Ramp Data with Data from Other Ramp Programs.

were refabricated and pressurized to give hot internal overpressures during R2 irradiations of approximately 0.4 and 14 MPa, respectively. The clad creepout and the time dependent changes in fuel rod conductance were investigated as functions of rod overpressure. The fuel rods were irradiated one at a time during three 3-day cycles in an instrumented rig in an in-pile loop in the R2 reactor. During these cycles the fuel rod thermal response was determined on-line by noise analysis [3, 4]. Between the 3-day cycles the rods were irradiated together for a total of six 15-day cycles without the on-line measurements mentioned. In the intermissions between these cycles, profilometry measurements were performed in the R2 reactor pool. After irradiation, the rods underwent non-destructive and destructive examinations. The rod with the highest overpressure had a measured diametral cladding outward creep strain of 11 μm after 1634 hours irradiation, with no apparent primary creep. This exceeded the expected pellet diameter increase attributable to fuel matrix swelling, since the average swelling rate measured in the fuel would only have resulted in a pellet diameter increase of 3.2 μm, after 1634 hours. Thus it was successfully demonstrated that a BWR fuel rod with an internal overpressure in excess of the pressure causing a cladding creepout rate as fast as the fuel solid swelling rate, can be operated at a Linear Heat Rate of up to 22 kW/m for more than 2 months without any apparent detrimental effect.

5 STUDSVIK's Defect Fuel Degradation R&D Projects

5.1 Technical Background

Several instances of fuel failure in nuclear power plants have been reported [41]. In a number of cases, primary defects in the fuel have led to the development of secondary defects caused by internal hydriding. An example was the failure of a liner rod in the Oskarshamn 3 BWR plant in 1988. The rod exhibited five longitudinal cracks, extending along most of its length [42]. Extensive hydriding of the cladding was observed. The primary cause seemed to be a fretting defect at the top of the fuel rod. Similar cases have been reported from other Swedish NPPs, a BWR plant in Switzerland and BWR plants in the USA, among others. Although the failure rate is small, the economic consequences can be unacceptable.

The mechanism for hydride formation has been theoretically evaluated [43, 44]. It is proposed that water penetrating through the primary crack oxidizes the fuel and the inside surface of the cladding, thus producing an excess of hydrogen. An obstruction of the axial gas communication will prevent a steady supply of steam, since the mass flow rate is proportional to the third power of the hydraulic diameter in the fuel [45]. Consequently, the criterion for hydride formation may be fulfilled in locations at some distance from the primary defect, especially if there is a constriction in between, caused, for instance, by gap closure due to a power maximum. Observations at Studsvik support the statement that the production rate of hydrogen is highest at the moment of clad defection; during the next few days, however, it levels off at a considerably lower value [46]. Massive hydride formation in the cladding will give rise to large circumferential stresses, eventually causing the cladding to split.

5.2 Technical Considerations

An experimental test scheme to be used at the R2 reactor is limited by the following non-compromisable constraints [47, 48].

- The length of the test fuel rodlets must not exceed the active core height (600 mm).

- The release of fission products and fuel material from the defected fuel rodlets has to be minimized in order not to cause intolerable contamination of the in-pile test loop.

When planning the experimental scheme it was assumed that the peaked power profile of the R2 reactor would cause a constriction of the pellet-clad gap at the middle section of the test rodlet due to mechanical interaction, if the Linear Heat Rate was sufficiently high. This situation would to a large extent prevent steam from penetrating downwards from the top of the test rodlet. In this way the conditions leading to hydriding in a power reactor fuel rod would be simulated (Figure 7).

In order to avoid contamination of the test loop, the intrusion of water was simulated by applying a water reservoir in an extended stainless steel plenum on top of the rodlet using a process developed by STUDSVIK. Consequently, no leakage of fission products to the test loop is expected to take place unless the cladding fails due to hydriding. This scheme offers the added advantage that it is possible to analyze the released fission gases and the hydrogen that has been formed, if the cladding does not rupture (Figure 8). In the experiments the amount of water in the reservoir is around 0.8 cm^3, out of which 0.2 cm^3 evaporates as the fuel rodlet is inserted into the test loop. At the beginning of an experiment the water vapor is distributed along the fuel stack. When the rodlet is lowered further into the reactor core, the fuel-clad gap, which is initially of the order of 0.15 mm, closes, provided that the Linear Heat Rate is above 30 kW/m. The rodlets are subjected to neutron radiography before and after each test irradiation in order to check the distribution of water. It has been verified that part of the water is left in the reservoir after the test.

5.3 Exploratory Defect Fuel Experiments

The two objectives of the test program were:

1. Demonstration of the feasibility to accomplish internal hydriding using the simulation scheme described above.

2. Investigation of remedies to suppress hydriding, given that the above objective is fulfilled.

The program was made up of two parts:

In Defect Fuel Experiment No. 1 (DEF-1) two fuel rodlets with two different types of cladding available from earlier STUDSVIK projects were tested, mainly for exploratory purposes. For Defect Fuel

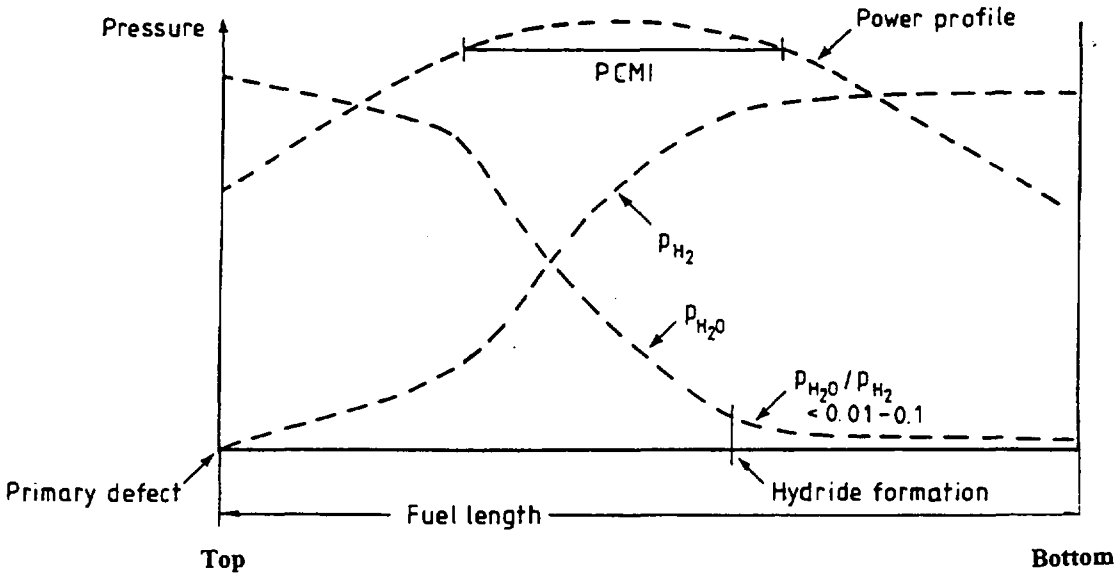

Figure 7
Conceptual Power Profile of the R2 Reactor, Distribution of Partial Pressures and Location of Hydride formation Following Water Intrusion.

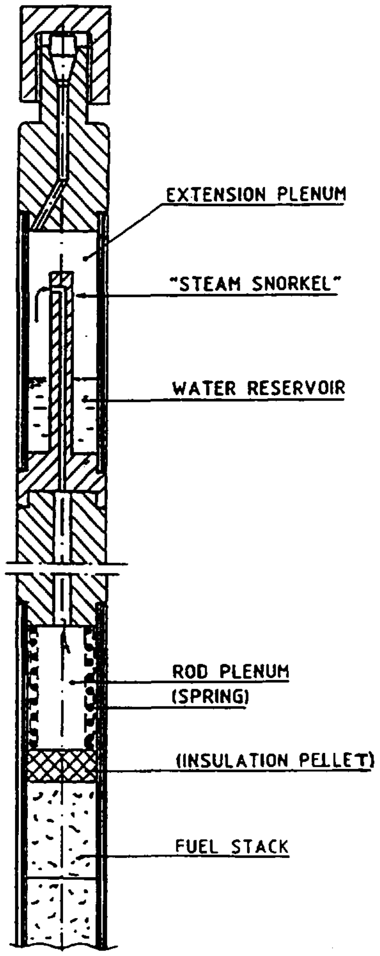

Figure 8
Array Simulating Primary Defects in Test Fuel Rodlets

Experiment No. 2 (DEF-2) five rodlets with three kinds of cladding were manufactured for the project. The rodlets were designed to be similar to 8x8 BWR fuel with Zircaloy-2 cladding. In order to simplify the manufacture, only unirradiated fuel was used [47, 16].

In DEF-1 standard cladding was compared to so-called rifled cladding, which is a proposed remedy for avoiding hydriding [30]. In the second series, standard design cladding, cladding with sponge zirconium liner and rifled cladding were irradiated. Table 2 gives data for the test rodlets used in DEF-1 and DEF-2.

Mainly two parameters were varied in the experiments:

1 The Linear Heat Rate was kept at 45 kW/m in all tests with the exception of DEF-2.4. For this test 50 kW/m was chosen.

2 The rodlets were generally irradiated for a full reactor cycle, amounting to between 15 and 17 days. Exceptions were test DEF-2.4, which lasted 1 day, and test DEF-2.5, which went on for 5 days.

In addition, a hafnium shield was used in test DEF-2.5, reaching a level of 160 mm from the bottom of the fuel stack. In this way, the local power at the lower end of the test rodlet decreased. The closure of the gap was then expected to be less rigid than in the other tests, giving more room to the mixture of hydrogen and water vapor.

The test rodlets were investigated in the following ways:

- Elongation measurements were done on-line during irradiation.

- Profilometry was carried out on the cladding before and after irradiation.

- Eddy current testing in search of cladding defects after irradiation was done before and after irradiation for most tests.

- Neutron radiography was done after irradiation in order to find out if hydriding had taken place. A check was also done before irradiation.

Destructive examination comprised determination of gas composition and metallography and ceramography investigation of the cladding and the fuel. Some results have been published [47]. The observations are summarized in Table 2.

The <u>on-line measurements of the elongation</u> of the test rodlets yielded seven curves which are shown in Figure 9. The elongation curves exhibit a similar shape. Initially, there is a sharp increase, when the reactor power is increased, until the Linear Heat Rate reaches its predetermined value. Then follows a relatively fast relaxation phase until a minimum is arrived at. As the irradiation progresses, a monotonous increase is obtained. In tests DEF-1.1, DEF-2.1, DEF-2.3 and DEF-2.4 a distinct peak is superimposed on the elongation curve during or immediately after the relaxation period.

In general, the <u>profilometry scans</u> look quite similar before and after irradiation. The curves from tests DEF-2.3 and DEF-2.4, however, show pronounced ridging after irradiation.

The <u>eddy current scanning</u> of the rodlet in test DEF-1.1 shows ridging as a result of the irradiation. Moreover, there is a clear sign of a secondary defect at the bottom of the rodlet (Figure 10). Ridging is clearly shown for the rodlets used in tests DEF-2.3 and 2.4 too. There is also an indication of a secondary defect at the bottom of the rodlet in test DEF-2.4.

The <u>neutron radiograph</u> of the rodlet from test DEF-1.1 shows that heavy hydriding has taken place at the 6th pellet level from the bottom. There is also minor hydriding at the 4th and 2nd pellet levels.

Table 2a
Defect Fuel Experiments 1 and 2 – Data for Test Fuel Rodlets

Test No.	Rodlet No.	Type of Cladding[1]	Fuel Parameters
DEF-1.1	1117	Zr-2 std	L/D: 1.2; dished, no chamfer; density: 10.17 g/cm³; gap: 0.15 mm
DEF-1.2	2050	Zr-2 rif	L/D; 1.05; no dish, slight chamfer; density: 10.41 g/cm³ (ABB standard); gap: 0.20 mm
DEF-2.1	2332	Zr-2 std	same standard as above; gap: 0.15 mm
DEF-2.2	2330	Zr-2 lin	same standard as above; gap: 0.15 mm
DEF-2.3	2328	Zr-2 rif	same standard as above; gap: 0.15 mm
DEF-2.4	2333	Zr-2 std	same standard as above; gap: 0.15 mm
DEF-2.5	2329	Zr-2 rif	same standard as above; gap: 0.15 mm

Table 2b
Defect Fuel Experiments 1 and 2 – Test Parameters and Observation from Elongation Measurements, Eddy Current Scans and Profilometry

Test No.	LHR (kW/m) Peak	Irradiation time (days) after start	Elongation peak, hrs	Profilometry	Eddy Current [1]	Neutron Radiography [1]
DEF-1.1	45	16.5	5	-	R + DF	H
DEF-1.2	45	17	None	No R	(R)	No H
DEF-2.1	45	17	6	No R	(R)	No H
DEF-2.2	45	14.5	None	No R	(R)	No H
DEF-2.3	45	17.5	19	R	R	No H
DEF-2.4	50	1	18	R	R+D	H
DEF-2.5	46	4.5	None	No R	(R)	No H

[1] Std = Standard
Rif = Rifled
Lin = Liner
R = Ridgning
DF = Defect
D = Small Defect
H = Massive Hydride

In comparison, the rodlet irradiated in test DEF-1.2 did not exhibit any hydriding.

The neutron radiograph from test DEF-2.4 shows hydriding at the level of the bottom pellet.

The radiographs of the other rodlets from Defect Fuel Experiment No. 2 gave no indication of hydriding.

The metallography data published so far reveal a large hydride precipitate in the cladding at the 6th pellet level in the rodlet from test DEF-1.1. This precipitate was investigated as shown in the micrograph in Figure 11.
Phenomena influencing the growth of the cladding are in summary:

- The oxidation of UO_2 to UO_{2+x} should lead to a contraction because of the decrease of the lattice parameter.

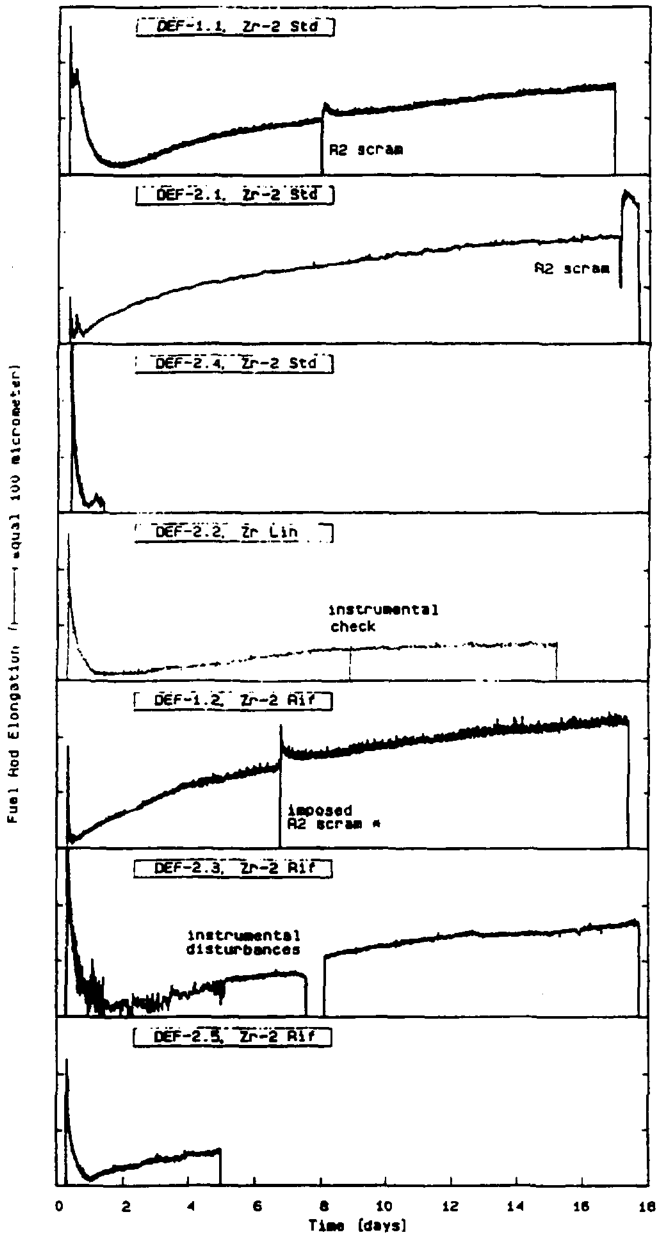

Figure 9
Compilation of On-Line Recordings for the Seven Fuel Rodlets Listed in Table 2.

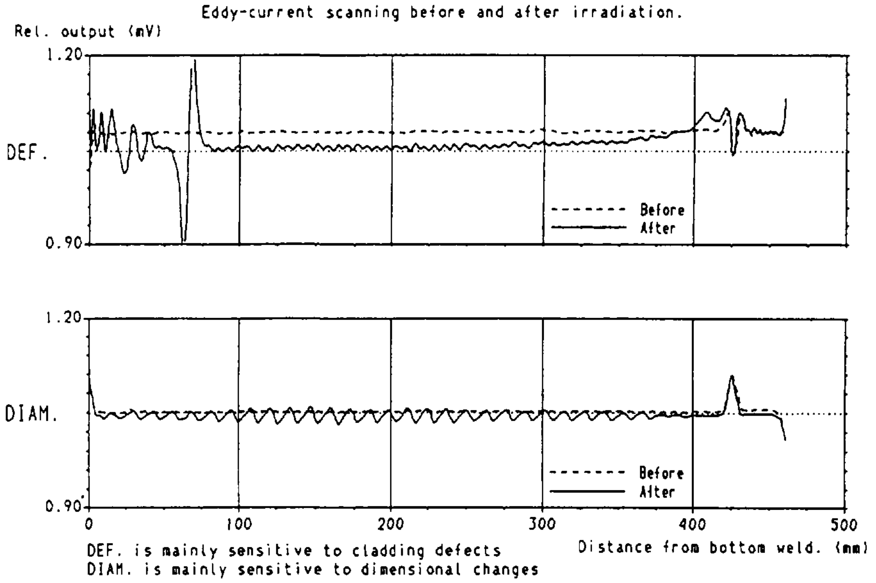

Figure 10
Eddy Current Measurement, before and after Irradiation of Test Rodlet 1117 with Standard Cladding Used in DEF-1.1.

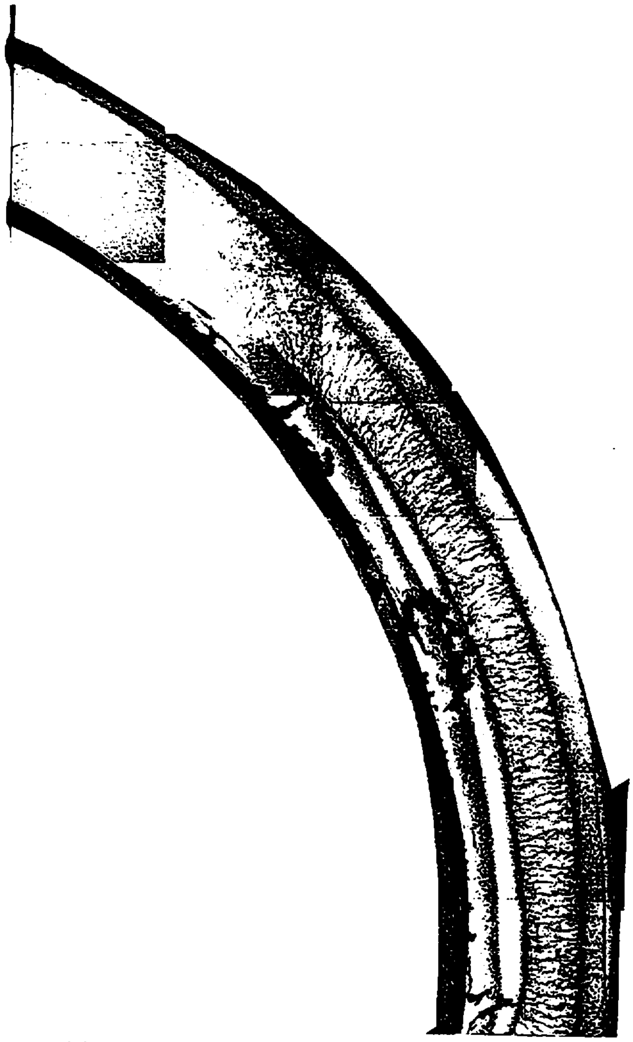

Figure 11
Micrograph of the Hydride Precipitate at the 6th Pellet Level from the Bottom of the Test Rodlet in DEF-1.1.

- The decrease of the thermal conductivity upon oxidation should lead to elongation of the rodlet.

- When helium is replaced by steam, a decrease of the thermal conductivity in the gap is expected, causing an elongation.

- When hydrogen is produced and more steam is prevented from entering, the thermal conductivity is increased, causing a contraction.

- Irradiation growth of the cladding leads to an elongation.

The elongation pattern in DEF-1 and DEF-2 differs from the behavior of a normal test rodlet without water. The rapid initial relaxation is ascribed to the production of hydrogen upon oxidation, apart from the stress relaxation.

It is suggested that the small peak appearing in some tests during or after relaxation can be explained in the following way. As the concentration of hydrogen increases, steam is reduced as it is diffusing downwards and the hydriding condition is fulfilled. When the hydrogen is consumed by the cladding upon hydriding, the rodlet starts to elongate until more hydrogen can barely be absorbed. After some time, the hydrogen content of the gap gas again rises slowly when it is replenished through the reduction of steam from above, which causes the rodlet to contract again. Stress relaxation will tend to assist in the process.

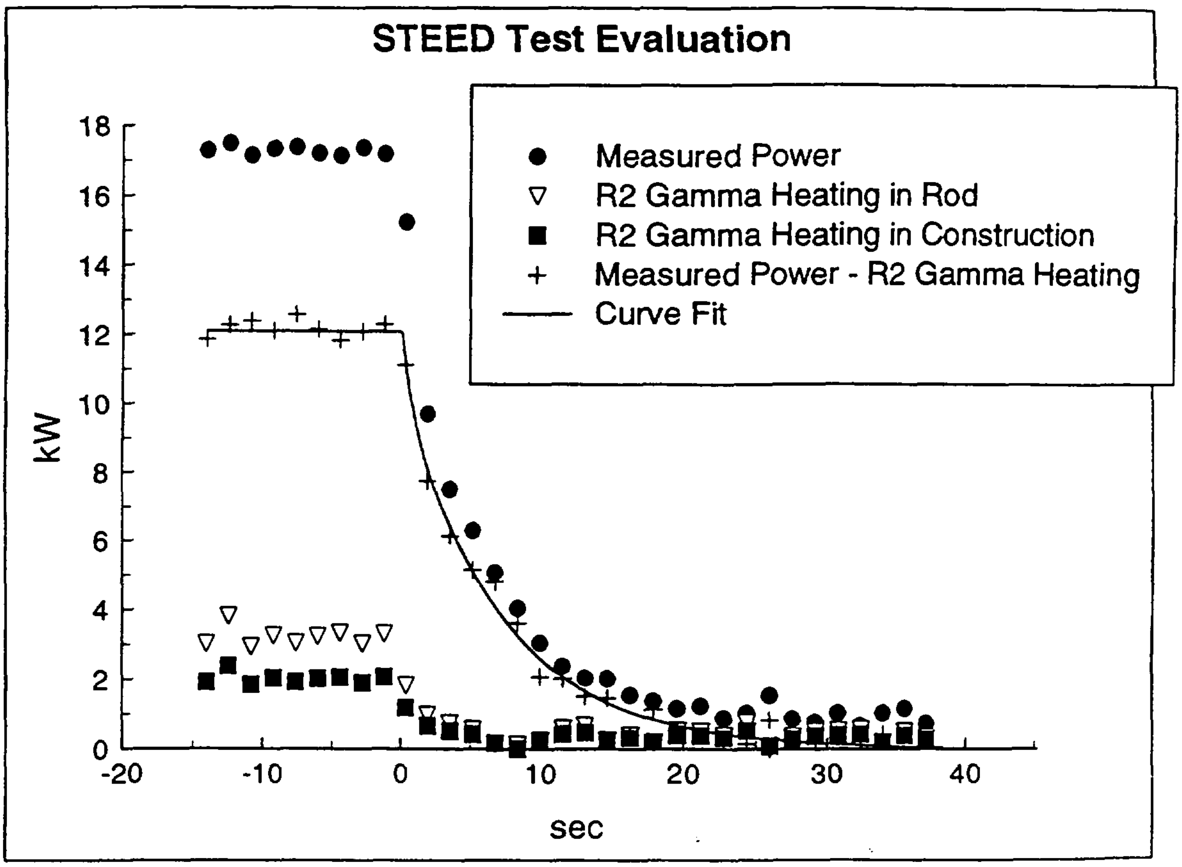

Figure 12
STEED Project – Example of Stored Energy Evaluation.
(The top curve and the two bottom curves are measured, the second one from the top is evaluated from the others. The stored energy is obtained from the area under the derived curve).

The long-time behavior of the elongation curve is probably due to the fact that the continuing but decreasing production of hydrogen is offset by the diffusional mixing with steam, which causes the thermal conductivity to decrease.

The pronounced ridging found for the test rodlets used in DEF-1.1, DEF-2.3 and DEF-2.4 indicates that the Pellet-Clad Mechanical Interaction has been strong. In consequence, one of the pre-requisites for hydriding in the bottom part of the cladding is fulfilled.

The massive hydride in the test rodlet from DEF-1.1 can be seen as split up in an inner and an outer massive precipitate, separated by metallic Zircaloy (Figure 11). The periferal hydride, extending over a quarter of the circumference, is evidently growing at the expense of the inner hydride, which dissolves as the hydrogen diffuses radially outwards down the temperature gradient. The fact that the massive inner hydride has been covered by metallic Zircaloy along its inner surface, indicates that at some point in time it has ceased to absorb hydrogen gas and now redistributes itself radially outwards.

The following conclusions have been drawn from the seven tests in DEF-1 and DEF-2:

- Secondary failure can develop rapidly by massive hydriding, already within a few hours after the occurrence of a primary defect.

- Massive hydriding seems to cease after a time period of the order of 24 hours, probably because the hydriding condition is no longer fulfilled.

- The test scheme applied is able to fulfill the objectives and can be used for further experiments.

5.4 The DEFEX Project

The conclusions that could be drawn from the Defect Fuel Experiments enabled STUDSVIK to propose and establish a larger project with international funding. The objectives of the DEFEX Project were:

* Investigation of phenomena and design parameters relevant to the degradation process in defected fuel rods, in particular:

    - Initiation of clad internal hydriding,

    - Progression of pellet oxidation, hydrogen release and clad hydrogen pick-up,

    - Clad failure by hydriding,

    - Initiation of clad fracturing.

* Providing a theoretical and experimental basis for further testing of potential remedies against secondary failure by hydriding.

* Data gathering for modeling of defect failure behavior.

* Comparison to Locke type data.

The experimental scheme involves:

* Investigation of the degradation process in BWR 8x8 type fuel rodlets at near zero burnup.

* Investigations under similar test conditions of:

  - Standard Zircaloy-2 cladding,
  - Standard zirconium liner cladding,
  - Potential remedy type cladding.

According to the original program, 6 BWR rodlets were to be tested in a total of 7 irradiation cycles. Non-destructive examination of the rodlets was included in the program as well as post-irradiation examination of the DEF-2 and DEFEX rodlets. At present, a continuation of the DEFEX Project is being planned, which is described in Section 6.

6 Studsvik's Upcoming International Fuel R&D Projects

Discussions are currently in progress on 6 different new international fuel R&D projects: two of them follow-ons to ongoing programs, two similar in principle to earlier projects and two completely new types of projects.

6.1 DEFEX II

The on-going DEFEX (I) project was presented in Section 5. The present plans for the DEFEX II project are tentative. The main program will include:

- Studies, analogous with the DEFEX Project, of the degradation process in irradiated fuel rods with medium (around 20 MWd/kg U) burnup with simulated primary defects.
- Testing of potential remedies against secondary failure by hydriding in order to supplement the experimental data base.
- Modeling of defect fuel behavior.
- Studies of cracking behavior of the hydrided cladding.

The primary defect would be simulated by a technique entailing delayed intrusion of water into the rodlet, thus simulating a case when a power rod is defected during operation.

A two-year BWR fuel program is envisaged, starting in 1997. The cost of the program is estimated at 30 MSEK, seeing that the test rodlets would have to be refabricated from full-length power reactor fuel. It is assumed that the scope of DEFEX II would be approximately the same as the DEFEX Project, i.e. 6 - 8 irradiation tests would be run, and destructive and non-destructive examination would include: neutron radiography, eddy current testing, dimensional measurement, visual inspection, gap squeeze measurement, gamma scanning, SEM investigation, internal gas analysis, metallography/ceramography and stoichiometry determination.

6.2 ULTRA-RAMP

The question of the ramp behavior at 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 so far [48].

The concern addressed appears to 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 seem 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 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.

In the SUPER-RAMP project a substantial fuel rod diameter expansion was observed (> 1 %) in a particular set of fuel rods at burnup levels of 45-50 MWd/kgU on up-ramping by 100 W/(cm·min) to close to 50 kW/m. The transient tests performed in the TRANS-RAMP I and II projects indicated very short times to failure at ramp rates of approximatly 1000 W/(cm·min) up to above
40 kW/m.

The prospective ULTRA-RAMP project would constitute a combination of three groups of ramp projects: one group concentrated on the PCI phenomena under normal operating conditions in different types of fuel and the other two groups concentrated on more safety-oriented issues. Thus the ramp resistance in current fuel types would be studied both under normal operating conditions ("slow" ramps) and under off-normal operating conditions ("fast" ramps corresponding to ANSI Class II and III events and "ultra-fast" ramps corresponding to some ANSI Class IV events).

A few recent simulated RIA experiments (Reactivity Initiated Accidents) with high-burnup fuel (55 and 65 MWd/kgU) 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 projects (such as TRI) but slower than the simulated RIA experiments. These new "ultra-fast" ramps could reach e.g. 120 kW/m during an 1 sec effective ramp time, corresponding to an enthalpy increase of 30 cal/g.

According to our present plans a number of high burnup fuel rodlets would be exposed to fast and slow ramp rates to preselected terminal power levels and to "ultra-fast" ramp rates to preselected enthalpy increases, all in the R2 test reactor [48]. The main objective is to identify any adverse or inadequate fuel rod behavior as for example abnormal fuel rod swelling, fast failure of the cladding, loss of fuel integrity on clad fracturing causing dispersion of fuel particles in the coolant water. Detailed non-destructive and destructive examinations (including advanced types of ceramography) would follow. These plans are quite flexible and depend on the feed-back we receive from discussions with prospective participants.

6.3 SUPER-RAMP III/10x10

In the new types of 10x10 BWR fuel the Linear Heat Rate is lower than in earlier types of fuel and the PCI resistance is presumed to be correspondingly improved. However, some utilities using zirconium liner with earlier types of fuel have raised the question whether the lower linear heat rates in 10x10 fuel really make the added resistance against PCI failure, achieved with zirconium liner fuel, unneccessary. As far as is known no ramp tests have ever been performed on 10x10 fuel.

Thus the proposed SUPER-RAMP III/10x10 project would be similar to the earlier SUPER-RAMP II/9x9 project. It is planned to start and be completed in 1997.

6.4 STEED - Enthalpy Determinations

The stored energy in fuel rods during operation, the enthalpy, depends on the fuel design (dimensions, materials), operating conditions and burnup. Since the heat release from the fuel after a scram is dominated by stored energy during the first minute this quantity is an important parameter in connection with safety considerations such as LOCA evaluations. A new type of tests are now in progress, the STEED project (Stored Energy, Enthalpy Determination).

Experimental determinations of the stored energy provide a valuable alternative to fuel code calculations. Depending on the test technique used the accuracy can be better than the corresponding code calculations, especially with increasing burnup. Apart from producing valuable data for safety related analysis, experimental determinations of the stored energy provide a possibility for interesting comparisons with fuel code calculations. The experimental results can also serve as excellent bench-marking opportunities for fuel modellers for unirradiated fuel.

STUDSVIK's R2 test reactor is well suited for scram experiments where the thermal response of different types of fuel can be compared. The measurements for the STEED project will be performed by analyzing the heat release from a test rod after scram, thereby using the R2 test reactor's calorimetric rod power measurement system. This system is the very same as that used in ordinary ramp experiments.

A demonstration experiment, STEED-I, is in progress on unirradiated fuel rodlets. The objective is to verify the test technique and to evaluate the accuracy. An international project, STEED-II, based on tests of irradiated fuel rodlets, will be discussed in 1997. An example of a "pre-project" demonstration test is shown in Figure 12.

6.5 Post-DO/DNB Ramp Resistance

Early experimental data in power reactors and test reactors have shown that short periods (less than about a minute) of Post-DO/DNB operation have not caused fuel failures (defined as leaking fuel rods). The question whether such operation might cause incipient (non-penetrating) fuel cracks and a decrease in ramp resistance has never been investigated. A parametric fuel modeling study performed by STUDSVIK NUCLEAR indicates that the ramp resistance <u>might</u> deteriorate due to such an event.

STUDSVIK has planned an experiment where fuel rodlets would be exposed to short (20-30 sec) periods of Post-DO/DNB operation along a short part of their length (at the location of the highest linear heat generation rate) and then later ramp tested in the same in-pile loop.

6.6 TRANS-RAMP III

Experience from the earlier safety-oriented STUDSVIK projects (discussed in Section 1), as well as from power reactor operation, shows that non-penetrating cladding cracks form readily during certain short-time power transients and cracks initiate already within 5-10 seconds. In the TRANS-RAMP IV (TRIV) project, 1989-1993 [49], the influence of non-penetrating (incipient) cladding cracks on the PCI failure resistance during an anticipated subsequent transient occuring later in life was studied. Seven short fuel rodlets, refabricated from full-size PWR power reactor fuel with a burnup of 20-25 MWd/kgU, were tested in the program. Three of these rodlets were further irradiated in the R2 reactor after having been subjected to a first power ramp. The effects of these ramps were somewhat different. One of the rodlets was the only one that clearly exhibited detectable non-penetrating (incipient) cracks during the first ramp. After continued base irradiation, giving an additional 4 MWd/kgU, the rodlets were subjected to a second power ramp in order to determine the residual time to PCI failure. The rodlet with the incipient cracks from the first ramp showed considerably shorter time to failure than the other two rodlets.

In the TRANS-RAMP III (TR III) project the influence of non-penetrating (incipient) cladding cracks in BWR fuel rodlets on the PCI failure resistance during an anticipated subsequent transient occuring later in life will be studied.

ACKNOWLEDGEMENTS

The authors are much indebted to all their colleagues at STUDSVIK's R2 test reactor and Hot Cell Laboratory and especially to Mr Hilding Mogard, who initiated the work on fuel R&D projects at Studsvik. The authors are also much indebted to all the sponsors of the bilateral and international projects mentioned.

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