Agenzia nazionale per le nuove tecnologie, l'energia
e lo sviluppo economico sostenibile

Ministero dello Sviluppo Economico

RICERCA DI SISTEMA ELETTRICO

Rapporto Test su combustibili e materiali avanzati presso HRP

F. Cantini, M. Cherubini, D. Lazzerini, F. D'Auria

RdS/2012/166

RAPPORTO TEST SU COMBUSTIBILI E MATERIALI AVANZATI PRESSO HRP

F. Cantini, M. Cherubini, D. Lazzerini, F. D'Auria CIRTEN
Settembre 2012

Report Ricerca di Sistema Elettrico
Accordo di Programma Ministero dello Sviluppo Economico - ENEA
Area: Governo, Gestione e Sviluppo**,** del Sistema Elettrico Nazionale
Progetto: Nuovo Nucleare da Fissione: Collaborazioni Internazionali e sviluppo Competenze in Materia Nucleare
Responsabile del Progetto: Massimo Sepielli,ENEA

Agenzia nazionale per le nuove tecnologie, l'energia
e lo sviluppo economico sostenibile

Ministero dello Sviluppo Economico

RICERCA DI SISTEMA ELETTRICO

Rapporto Test su Combustibili e Materiali Avanzati presso HRP

F. Cantini, M. Cherubini, D. Lazzerini, F. D'Auria

CIRTEN

Consorzio **I**nteruniversitario per la **R**icerca **TE**cnologica **N**ucleare

Report RdS/2012/1506

RAPPORTO TEST SU COMBUSTIBILI E MATERIALI AVANZATI PRESSO HRP

F. Cantini, M. Cherubini, D. Lazzerini, F. D'Auria
Gruppo di Ricerca Nucleare S. Piero a Grado – Università di Pisa

Luglio 2012

Report Ricerca di Sistema Elettrico
Accordo di Programma Ministero dello Sviluppo Economico - ENEA

CIRTEN

Consorzio **I**nteruniversitario per la **R**icerca **TE**cnologica **N**ucleare

UNIVERSITY OF PISA

S. PIERO A GRADO NUCLEAR RESEARCH GROUP

Rapporto Test su Combustibili e Materiali Avanzati presso HRP

Autori

F. Cantini

M. Cherubini

D. Lazzerini

F. D'Auria

CERSE-UNIPI RL 1506/2011

PISA, July 2012

Lavoro svolto in esecuzione dell'Attività LP1.C3.b
AdP MSE-ENEA sulla Ricerca di Sistema Elettrico - Piano Annuale di Realizzazione 2011
Progetto 1.3.1 "Nuovo Nucleare da Fissione:
collaborazioni internazionali e sviluppo competenze in materia nucleare"

Contents

List of Figures

List of Tables

Summary

The Halden Reactor Project has been in operation since 1958 and is the largest NEA joint project. The programme is primarily based on experiments, product developments and analyses carried out at the Halden establishment in Norway, and is supported by more than 130 organizations in 19 countries. The Fuel and Materials programme includes Fuels Safety and Operational Margins, including loss-of-coolant accidents, Plant Ageing and Degradation, International Gen IV Research. The codes are being used for R&D purposes, for the design of fuel rods, new products or modified fuel cycles and to support loading of fuel into a power reactor, i.e. to verify compliance with safety criteria in safety case submissions. An overview of the Halden ractor Project experiments planned in the next years has been presented. The objectives are quite wide, because various issues are considered, covering both actual and innovative materials; different plant design (PWR, BWR and VVER); fuel behavior under accident conditions.

The gathering of experimental data will surely improve the understanding of fuel mechanical behaviour, explicitly considering the actual trend of increasing the burn up. A parameter already demonstrated to have a major influence on observed fuel phenomena, such as fission gas generation and release, hydrogen generation, PCI, etc.

Experimental data may serve also to verify if correlations embedded in fuel mechanic codes are still valid to simulate recent fuel operation and innovative material behaviour. Hence databases coming from Halden Project have to considered of high relevance because they address phenomena still not completely understood and consequently not fully replicable by computational tools.

On this aspects an overview of TRANSURANUS code is provided, discussing its adequacy and including possible improvements.

Introduction

This report is composed by two main parts: the first introduce the Halden reactor and the experimental campaign here conducted and the second deals with computational codes used to predict fuel behavior in a nuclear power plant.
The Halden Reactor Project has been in operation since 1958 and is the largest NEA joint project. It brings together an important international technical network in the areas of nuclear fuel reliability, integrity of reactor internals, plant control/monitoring and human factors. The programme is primarily based on experiments, product developments and analyses carried out at the Halden establishment in Norway, and is supported by more than 130 organizations in 19 countries.
The Fuel and Materials programme includes Fuels Safety and Operational Margins, including loss-of-coolant accidents, Plant Ageing and Degradation, International Gen IV Research.
The 2012-2014 work programme in the nuclear fuel area emphases on fuel behavior and properties after prolonged in-core service and at burn-ups in excess of current discharge levels, as extended fuel utilization remains an industry priority. The programme also addresses LOCA issues and the response of high burnup fuel to fast power and temperature transients, focusing on in-reactor effects that are different from those obtained in out-of-reactor tests [1].
The prediction of the behavior and the life-time of the fuel rods is a key requirement in order to ensure the safe and economics operation of a NPP. The accurate description of the fuel rod's behavior involves various disciplines such as nuclear and solid state physics, metallurgy, ceramics, applied mechanics and the thermal heat transfer. To deal with the complexity of the subject, fuel designers and safety authorities rely heavily on computer codes describing the general fuel behavior, since they require minimal costs in comparison with the costs of an experiment or an unexpected fuel rod failure. The codes are being used for R&D purposes, for the design of fuel rods, new products or modified fuel cycles and to support loading of fuel into a power reactor, i.e. to verify compliance with safety criteria in safety case submissions. In addition to steady-state irradiation, the fuel rod behavior is also being simulated under transient and accident conditions.
The validation of the fuel computer codes against experiments is a fundamental activity in order to ensure the reliability of the codes themselves and the ability to properly simulate normal and off normal operating conditions, material properties, material degradation. The computer code used at the GRNSPG-UNIPI for the thermo-mechanical analysis of the fuel rods behavior is TRANSURANUS.

1 Description of Halden Reactor

The Halden Boiling Heavy Water Reactor (HBWR) is in operation since 1959 in Halden, a coastal town in south-east Norway near to the border to Sweden. The reactor vessel primary circuit system is inside a rock cavern. Heat removal circuits are either placed inside the reactor hall or in the reactor entrance tunnel. Control room and service facilities are placed outside the excavation. The utilization of the reactor is 24 hours per day, 7 days per week, 28 weeks per year producing 4000 MWdays/year [2].
The Halden Boiling Heavy Water Reactor (HBWR) is a natural circulation boiling heavy water reactor, Fig.1 The maximum power is 25 MW (thermal), and the water temperature is 240°C, corresponding to an operating pressure of 33.3 bar.
The reactor pressure vessel (Figure 1) is cylindrical with a rounded bottom. It is made of carbon steel and the bottom and the cylindrical portion are clad with stainless steel. The flat reactor lid has individual penetrations for fuel assemblies, control stations and experimental equipment. 14 tons of heavy water act as coolant and moderator. A mixture of steam and water flows upwards by natural circulation inside the shroud tubes which surround the fuel rods. Steam is collected in the space above the water while water flows downwards through the moderator and enters the fuel assemblies through the holes in the lower ends of the shroud. The steam flows to two steam transformers where heat is transferred to the light water secondary circuit.
In the secondary circuit (Figure 2), two circulation pumps pass the water through the steam transformers, a steam drum and a steam generator where steam is produced in the tertiary circuit. The tertiary steam is normally delivered as process steam to the nearby paper mill, but may also be dumped to the river.
Table 1 shows the most relevant data for nominal operating conditions.

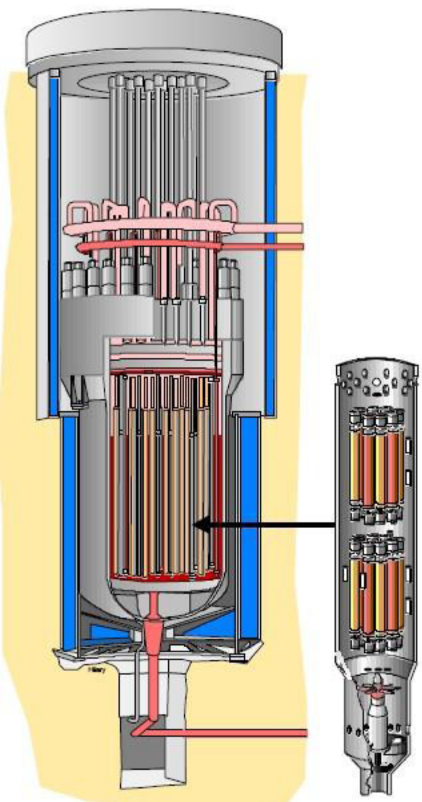

Figure 1 – Simplified flow sheet of the reactor system

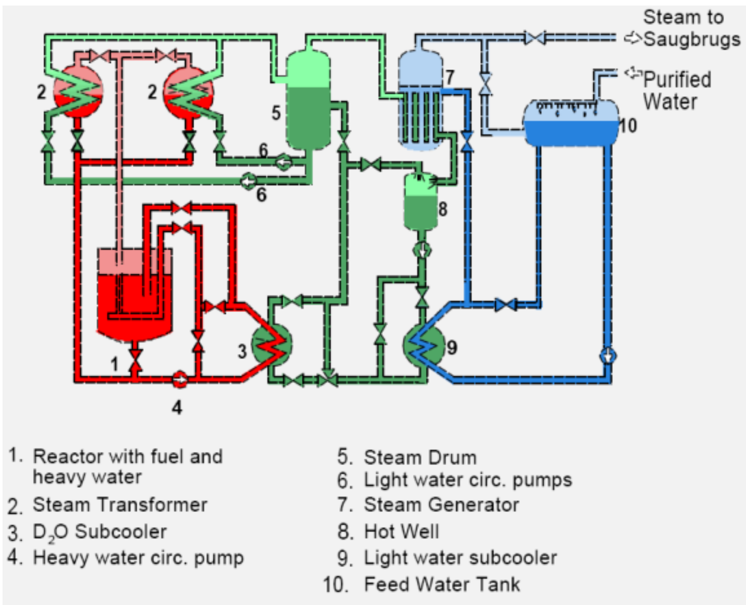

Figure 2 – Simplified scheme of the reactor

Nominal Reactor Operating Data	
Power Level	up to 20 MW (th)
Reactor Pressure	33.3 bar
Heavy Water Saturation Temperature	240°C
Maximum Subcooling	3.0 MW
Primary Steam Flow (both circuits)	160 ton/h
Return Condensate Temperature	238°C
Subcooler Flow	160 ton/h
Plenum Inlet Temperature	237°C

Table 1 – Nominal reactor operating data

1.1 Description of Halden Reactor Project (fuel part)

Competitive electricity generation from nuclear power requires the availability of high performance fuel with high burn-up capabilities and reliability in accord with zero failure. While the operational performance of nuclear power plants has improved considerably in recent years, with the introduction of extended operational cycles, increases in discharge burn-up and power up-rates, operational experience shows that unforeseen deviations from normal fuel performance do occasionally occur in some conditions. Such deviations can limit plant operation or even lead to premature shutdown to discharge the affected fuel. Remedies have involved fuel design modifications, new materials or changes to water chemistry aimed at improving fuel reliability in a variety of demanding service conditions.
Increased utilization of fuel provides challenges not only for normal operation, but also in safety transients. It is therefore essential to establish a knowledge base for safety assessments and to demonstrate the capabilities of high exposure fuel in off-normal situations. Since such situations are undesired and extremely rare by definition, the related database must be obtained by dedicated investigations conducted in test reactors in light of envisaged technical solutions and the complementary work planned.
The HRP fuels programme aims to determine fuel safety and operational margins for use in design and licensing by studying:

- Gas release under irradiation – fission gas release behaviour, gas inventory increase, tolerable rod overpressure
- Fuel thermal and mechanical performance - conductivity degradation, densification, swelling, fuel creep, pellet-clad-mechanical-interaction (PCMI)
- Fuel behaviour under accident scenarios - loss of coolant accident (LOCA)
- Demanding operation conditions - power transients, PCMI, cladding transient creep, cladding corrosion and hydriding

1.2 HRP 2012-2014 program: fuel related activities

Increased utilization of fuel provides challenges not only for normal operation, but also in safety transients. It is therefore essential to establish a knowledge base for safety assessments and to demonstrate the capabilities of high exposure fuel in off-normal situations. There is therefore

consensus among HRP participants to continue experimental activities with the objective of generating new and improved data on fuel properties and fuel behavior.
The experiments related to gas release under irradiation considered for the next programme period are indicated in the following list:

- Integral fuel performance studies
- A MOX He release
- An ultra-high burn-up irradiation experiment
- The cladding lift-off experiments
- Fission gas release from standard, large grain, gadolinia and chromia fuels as well as BeO bearing fuels
- Fuel thermal conductivity degradation and recovery mechanisms
- Fission induced fuel creep
- Gd-fuel behaviour
- VVER fuel behaviour
- The thermal behaviour of modified fuels
- Integral fuel performance studies
- The cladding lift-off experiments

1.3 Overview of experimental activities

The experiments related to gas release under irradiation (*Integral fuel performance studies*), have the objective to produce experimental data for understanding and modelling of high burn-up fuel behaviour by the concurrent measurement of temperature, fission gas release and PCMI in controlled power transients and steady state conditions. The following list reported the tests:

A *MOX He release* test has the objective to understand the helium release and retention capability of MOX fuel under irradiation conditions in terms of influential variables such as temperature, microstructure, rating and burn-up. The experiment employs disk fuel which has the advantage that larger quantities of fuel are exposed to the same conditions.

An *ultra-high burn-up irradiation* experiment, utilizing rods of fuel disks previously irradiated, has the objective to study fission gas release of high burn-up structured fuel subjected to a relatively fast power up-rating. Two UO2 rods and two MOX fuel rods, irradiated to burn-ups of ~120 and 110 MWd/kg respectively, are planned for the first set of tests, which are due to start already in the current program period.

The *cladding lift-off* experiments at Halden have been carried out for many years and provide direct and convincing data on maximum tolerable rod overpressure. The objectives are to determine the maximum ΔP above system pressure to which fuel rods of different designs (PWR, BWR, VVER) and types of fuel (UO_2, MOX) can be operated without causing lasting/continuous fuel temperature increase and thus a potential threat to rod integrity. Cladding elongation is also monitored allowing the state of PCMI during the test to be determined. In addition to lift-off data, the tests are designed to produce data on axial gas communication within high burn-up fuels. The influence of filler gas (Ar/He) and gas pressure on steady state and dynamic fuel thermal response can also be studied.

Fission gas release from standard, large grain, gadolinia and chromia fuels as well as BeO bearing fuels will be studied through a range of burn-up in several experiments started from fresh fuel.

Fuel thermal conductivity degradation and recovery mechanisms are proposed to be studied in a separate effects test including innovative fuel types as available. Fuel thermal conductivity is an essential materials property required for modeling of fuel behavior.

Fission induced fuel creep: The creep of UO_2 and MOX fuels under irradiation has been shown to be a function of the applied stress and fission rate per unit volume, but independent of temperature below about 1000°C.

Gd-fuel behaviour will continue to be investigated in experiments dedicated to this fuel type. A comparative irradiation test already started in a previous programme period will continue into the next in order to reach the target burn-up of about 50 MWd/kg (UO^2 rods) with PIE and higher power operation planned in order to study PCMI. The proposed test set-up is a fuel disk type of irradiation, with a test matrix to study fuels of varying Gd content at varying irradiation temperature and power / fission rate.

VVER fuel behaviour will continue to be studied in an experiment dedicated to this fuel type. A comparative irradiation test already started in a previous programme period ,containing standard VVER fuel and fuel with additives, will continue into the next period in order to reach the target burn-up of about 60 MWd/kg (UO_2 rods).

The thermal behaviour of modified fuels is investigated in an experiment that commenced in 2010 that will continue throughout the next programme period. Of special interest is the thermal performance of beryllium oxide in a UO2 matrix resulting in improved (higher) thermal conductivity and thus lower fuel temperatures.

Integral fuel performance studies will also yield data on fuel temperatures and PCMI. Candidate fuel for testing in this series is Gd-fuel with a burn-up of about 50 MWd/kg.

The *cladding lift-off experiments* are also designed to produce data on fuel temperature, fuel swelling and axial gas communication within high burn-up UO_2 and MOX fuels.
In most tests, the fuel segments will be equipped with multiple instruments to study the interrelation between the various performance parameters. The planned investigations and analyses will be:

- Expand the database of fuel thermal conductivity and its degradation with burn-up
- Expand the database of PCMI behavior at different exposures
- Provide new performance data on modified and innovative fuel
- Generate more data on the behavior of production line gadolinia fuel
- Generate more data on the behavior of VVER fuels
- Provide long term measurements on PCMI behavior and rod growth rate due to fuel swelling and fuel-clad bonding

- Produce direct measurements of in-pile creep of UO2, MOX and Gd-fuel

This will be achieved by using fresh fuels as well as irradiated and re-fabricated fuel segments from PWR, BWR and VVERs.
Of special interest is the Fuel Behaviour under Accident Scenario. The introduction of new cladding materials and, in particular, the move to higher burn-up have generated a need to re-examine the safety criteria for loss-of-coolant accidents and to verify their continued validity. The Halden Project has implemented a LOCA test series to study the integral in-reactor fuel behaviour under expected and bounding conditions. The Halden reactor is suited for integral in-pile testing of fuel behaviour under LOCA conditions using single fuel rods. The decay heat is simulated by a low level of nuclear heating which produces a temperature distribution in the fuel rod similar to the real case. The objectives of the HRP LOCA test series and the test execution conditions were defined as:

- Measure the extent of fuel (fragment) relocation in to the ballooned region and evaluate its possible effect on cladding temperature and oxidation.
- Investigate the extent (if any) of “secondary transient hydriding” on the inner side of the cladding above and below the burst region.

At the time of writing the draft programme proposal, seven tests with irradiated fuel segments (burn-up 40 – 92 MWd/kg) from commercial NPPs have been carried out.

1.4 Fuel Behaviour under Accident Scenarios

The introduction of new cladding materials and, in particular, the move to higher burn-up have generated a need to re-examine the safety criteria for loss-of-coolant accidents and to verify their continued validity. As part of international efforts to this end, the Halden Project has implemented a LOCA test series to study the integral in-reactor fuel behaviour under expected and bounding conditions.
The Halden reactor is suited for integral in-pile testing of fuel behaviour under LOCA conditions using single fuel rods. The decay heat is simulated by a low level of nuclear heating which produces a temperature distribution in the fuel rod similar to the real case. Thus a more correct differential fuel-cladding thermal expansion is obtained compared to out-of-reactor tests where the cladding is heated from outside and more than the fuel.
Different degrees of contamination of the loop system employed in the series were observe
test to test. A procedure has been implemented to quantify the amount of iodine released after ballooning and burst since the source term is important for evaluating the consequences of a LOCA.
A continuation of the HRP LOCA test series will aim to provide answers to the original objectives
as well as new questions arisen from the tests carried out so far:

- When do fuel relocation and fuel dispersal occur and when can they excluded?
- Effects of burn-up, rod pressure, and corrosion (hydrogen) on integral fuel behavior during LOCA
- Quantification of the source term (iodine release)

It is mandatory to utilize fuel rod irradiated in commercial reactors to relevant burn-ups with a thorough characterization regarding the state of the cladding and the bonding with fuel. Participating organizations have made available both PWR, BWR and VVER fuel with the

desired characteristics of which all four segments with burn-up >80 MWd/kg have been used. Experience shows that about four LOCA experiments can be execute in a three years period, including the necessary refabrication work before and PIE (Postulated Initiating Event) after the in-reactor part.
The Halden LOCA series is expected to include both bounding conditions and industry representative conditions. The latter categories includes the break of a recirculation pump as the limiting pump as the limiting design base accident for a BWR 6. Calculations according to Appendix K of 10CFR50 show low values for cladding temperatures and low oxide thickness on the assumption of no fuel relocation.
Halden LOCA tests have also been discussed extensively in the NEA-CSNI context by the Working Group on Fuel Safety (WGFS). Among others, their recommendations include:

- Determine the impact of axial gas transport on ballooning, e.g. by including a spacer grid between the upper plenum and the balloon area that would act as a prototypical distension restriction and cooling improvement similar to what can be expected in the real situation.
- Investigate fuel relocation as influenced by the driving force provided by the amount of gas available in the experiments.

1.5 Innovative fuel (research)

In addition to the improved fuel and claddings being developed to withstand the challenges to fuel integrity described in previously sections, innovative materials are also being looked into as an option for fuels and claddings, especially for Generation 3+ or Generation IV reactor types.
For fuels, it is desirable to be able to operate at a lower temperature for a given power output in order to reduce fission gas release and other deleterious effects of high temperature operation.
Higher fuel thermal conductivity is a prerequisite for this, and candidate materials include uranium nitride, fuel containing beryllium oxide (possibly in a whisker form), and fuel rods with a liquid metal in the fuel-clad gap. In addition, improved fuel performance may be achieved through careful control of fuel microstructure by producing pellets with a microstructure that varies radially through the pellet such as grain size, Gd-content or enrichment.
For cladding materials, it is desirable to avoid the deleterious effects induced by high residence time in a corrosive and radiation environment such as accelerated creep and growth or hydride embrittlement. Sustained dimensional stability is a prerequisite for this, and SiC could be a suitable cladding material in this respect for ultra-high utilization of uranium fuel. An obvious added advantage of this material is that it is also relatively chemically inert.

2 Critical review of experiment on nuclear fuel

A consensus exists among HRP participants to continue experimental activities with the objective of generating new and improved data on fuel properties and fuel behavior.
In many cases it is important to have data from experiments like above mentioned, for what concerns the thermal-mechanical and chemical behavior of the fuel and the cladding. For example, the fission induced fuel creep under irradiation has been shown to be a function of the applied stress and fission rate per unit volume.
The experiments planned at Halden reactor foresee high burn up. This because actual trend is to push as much as possible the use of the fuel inside the reactor. Thus specific data have to be

acquired both to verify if actual model can be extended to predict high BU fuel behavior, and eventually to define new correlations.
It is important to expand the database of PCMI behavior at different exposures providing long term measurements on PCMI behavior and rod growth rate due to fuel swelling and fuel-clad bonding.
Fuel behaviour under accident scenarios - loss of coolant accident (LOCA), has a direct implication in the reactor safety, as recognized also by the WGFS. Dedicated researches are planned at Halden reactor, to increase the understanding of relocation occurrence, effect of BU, rod pressure, hydrogen under accident conditions. Experiments will be conducted to fuel irradiated in reference plant (PWR, BWR, VVER). This part of activities conducted within the Halden Project should be highly considered for its direct impact in the safety and licensing of nuclear power plants.
Specific campaigns devoted to increase the thermo-mechanical behavior of current and innovative materials should be followed with suitable attention. From these tests valuable data surely will come which will constitute solid bases especially for the modeling of new materials. Among new materials attention is put to clad materials, searching for suitable solution form the neutronics and chemistry point of view. Namely low neutron absorption and low chemical reactivity are two parameters which can be used to judge the real benefit of innovative materials.

3 Adequacy of computational tools for fuel analysis: TRANSURANUS

The following sections deal with a review of fuel pin mechanic capabilities taking TRANSURANUS code as an example.
TRANSURANUS is a computer program for the thermal and mechanical analysis of fuel rods in nuclear reactors. It was developed at the Institute for Transuranium Elements (ITU). The TRANSURANUS code consists of a clearly defined mechanical–mathematical framework into which physical models can easily be incorporated. Besides its flexibility for fuel rod design, the TRANSURANUS code can deal with a wide range of different situations, as given in experiments, under normal, off-normal and accident conditions. The time scale of the problems to be treated may range from milliseconds to years. The code has a comprehensive material data bank for oxide, mixed oxide, carbide and nitride fuels, Zircaloy and steel claddings and several different coolants. It can be employed in two different versions: as a deterministic and as a statistical code [3],[4].
During its development great effort was spent on obtaining an extremely flexible tool, which is easy to handle and exhibits very fast running times. The total development effort is approximately 50 man-years.

3.1 Geometrical idealization

In principle, our spatial problem is three-dimensional (3D). However, the geometry of a cylindrical fuel rod (a very long, very thin rod) suggests that any section of a fuel rod may be considered as part of an infinite body, i.e. neglecting axial variations. By further assuming axial-symmetric conditions because of the cylindrical geometry, the original 3D problem is reduced to a one-dimensional one. Analyzing the fuel rod at several axial sections with a (radially) one-dimensional description is sometimes referred to as quasi 2D or 1 1/2D. TRANSURANUS code fall into this category. Generally, real 2D or even 3D codes are used for the analysis of local

effects, whereas the other codes have the capability to analyze the whole fuel rod during a complicated, long power history. Even with the computer power of today, a full 3D analysis of some transient can require long simulation runs or even be practically impossible. In addition, such an analysis would also not be very meaningful, as the fuel fragments shape and position are determined by a stochastic process, or as the differences in the azimuth direction for physical quantities are unknown.
The geometrical representation of the fuel rod used by TRANSURANUS is shown in Figure 3. The fuel is divided into axial slices. Each slice has a different axial coordinate and height. In the reference state the coordinates of the fuel and the cladding of the same slice are identical. During irradiation, however, they may differ due to different axial deformations.
TRANSURANUS offers two different options on how to treat the fuel rod:

- the "slice" option and
- the "sectional" option.

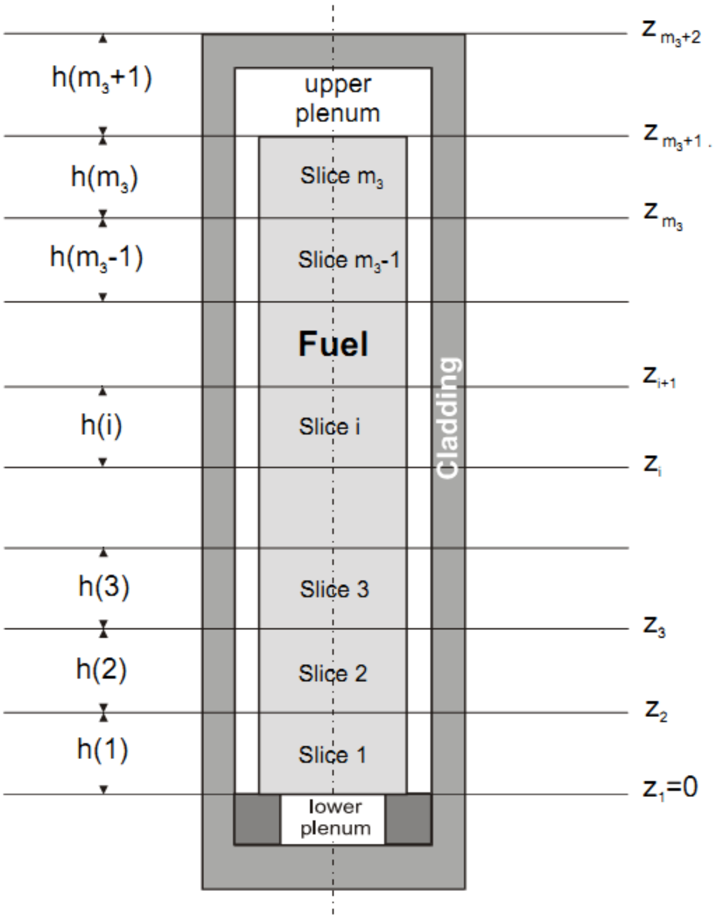

Figure 3 – Axial discretization of the fuel rod

In both cases the fuel is analysed slice per slice, starting from first slice up to last one. The difference is that with the slice option, a slice is analysed at the middle whereas with the sectional option a slice is analysed at the bottom and the top. In addition, there is another difference: in the slice option it is assumed that all axial quantities, e.g. the linear rating, are constant along the slice, whereas in the sectional option these quantities may vary linearly along the slice.

The mechanical analysis performed by TRANSURANUS is based on the following assumptions:

- the fuel rod is axially symmetric,
- a generalised plane-strain condition applies for the axial direction and
- the complex structure (fuel and cladding) can be described piecewise by isotropic spatially invariant elastic constants.

A quasi-analytical solution results, in which only some integrals need to be determined numerically. Consequently, the fuel and the cladding are divided into a number of rings, called coarse zones (Figure 4), in which the elasticity modulus E and Poisson's number are isotropic and constant. Each coarse zone is divided further into finer zones in order to perform the numerical integrations.

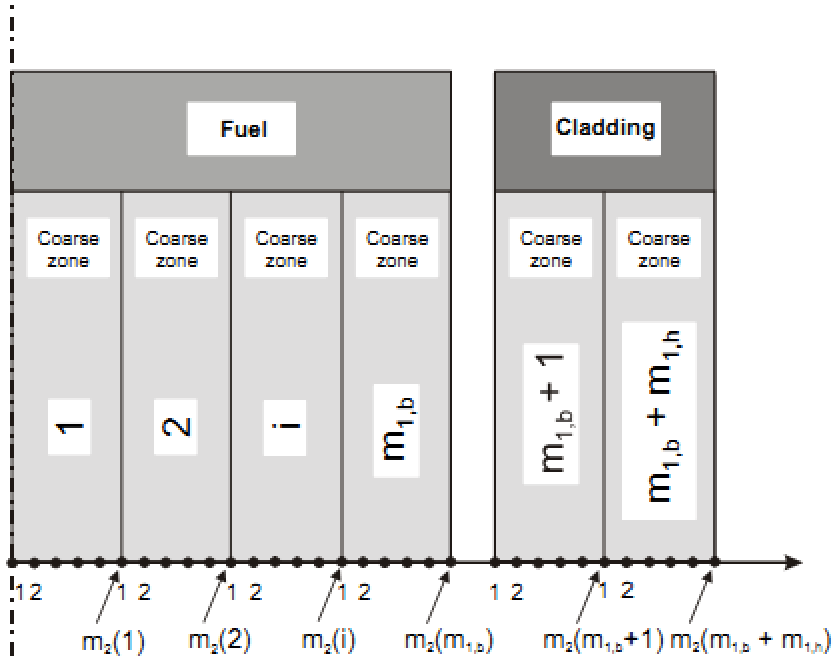

Figure 4 – Radial discretization of the fuel rod

3.2 Uncertainties and limitations

In general, the uncertainties to be considered may be grouped into three categories:

1. Prescribed quantities. The fuel rod performance code requires on input the fuel fabrication parameters (rod geometry, composition, etc.) and irradiation parameters (reactor type, coolant conditions, irradiation history, etc.).
2. Material properties, such as the fuel thermal conductivity or the fission gas diffusion coefficients.
3. Model uncertainties.

A good example of such an uncertainty is the plain strain assumption in the axial direction as illustrated in Figure 5 representing the interaction of the deformed and cracked fuel with the cladding. Intuitively, it is clear that for a detailed analysis of such problems 2D or even 3D models are indispensable. One of the most important consequences of all uncertainties is that one must implement models of “adequate” complexity.

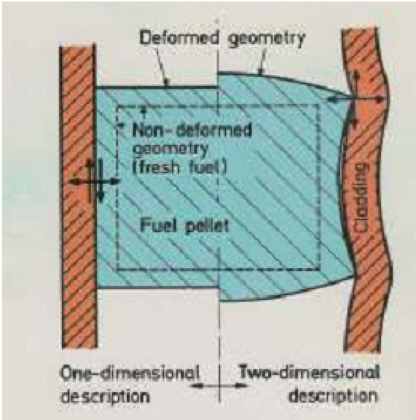

Figure 5 – Schematic view of a deformed fuel pellet; comparison between a one-dimensional and a two-dimensional description

3.3 Basic TRANSURANUS structure

The TRANSURANUS code is designed using the following main levels:

- Thermal analysis
- Mechanical analysis
- Iteration in sections or slices
- Axial iteration

3.3.1 Thermal analysis

The temperature distribution in a fuel rod is of primary importance for several reasons. First of all, the commercial oxide fuels have poor thermal conductivities, resulting in high temperatures even at modest power ratings. Secondly, the codes are used for safety cases where one has to show that no fuel melting will occur, or that the rod internal pressure will remain below a certain limit. Finally, many other properties and mechanisms are exponentially dependent on temperature.

3.3.2 Axial heat transfer in the coolant

In general three regimes must be covered in a LWR:

1. The sub-cooled regime, where only surface boiling occurs. This regime is typical for PWR's under normal operating conditions.
2. The saturated, two phase regime. This regime is typical for BWR's under normal operating conditions.
3. The saturated or overheated regime. This regime may be reached in all off-normal situations. A typical example is a LOCA.

TRANSURANUS uses one-dimensional (axial) fluid dynamic equations that can only cope with the first two regimes. For simulating the third type of regime, the whole reactor coolant system needs to be analysed by means of thermo-hydraulic system codes.
The temperature calculation in the coolant serves two purposes. First of all, the axial coolant temperature in the basic (fictional) channel provides the (Dirichlet) boundary condition for the radial temperature distribution in the fuel rod. It results from the combined solution of the mass, momentum and energy balance equations. The second objective of the heat flow calculation in the coolant is the derivation of the radial temperature drop between the coolant and the cladding resulting from convection.

3.3.3 Heat transport through the cladding

The heat transport through the cladding occurs through conduction and the heat generation in the cladding is neglected (the gamma-heating, as well as the exothermic clad oxidation process are disregarded). In order to account for the presence of an outside oxide layer with a thermal conductivity, the total equivalent cladding conductivity can be obtained by applying the formula for serial thermal resistances.

3.3.4 Heat transport from cladding to the fuel pellet

The temperature difference in the pellet-cladding gap is calculated neglecting the heat transfer by convection. The heat transfer coefficient between fuel and cladding depends on

1. gap width or contact pressure between fuel and cladding;
2. gas pressure and composition;
3. surface characteristics of cladding and fuel.

It is important to note that, despite very detailed formulations for each contribution of the gap conductance, there is an unavoidable uncertainty in the gap size due to input uncertainties, but also because of uncertainties in the mechanical computation.

Heat transport in the fuel pellets
The heat produced by the slowing down of the fission fragments in the fuel pellets is removed through conduction in the pellets. The temperature distribution in the pellets is therefore affected by two terms: the heat source and the fuel thermal conductivity, which depends on several parameters.

Mechanical analysis
The first barrier against release of radioactive fission products to the environment is the cladding of the nuclear fuel rod. The stress and associated deformation assessment of the cladding are therefore essential in fuel performance calculations. Furthermore, the deformation of both the pellets and the cladding affect the gap width, which in turn affects the conductance of the gap, hence the temperature distribution in the pellets. The thermal and mechanical analyses are therefore equally important and closely coupled.
The main assumptions made in TRANSURANUS performance codes are:

1. The system is axisymmetric, i.e. variables don't vary tangentially.

2. Although the fuel and cladding move axially (not necessarily at the same rate), planes perpendicular to the z-axis remain plane during deformation (plain strain condition), i.e. the rod remains cylindrical.
3. Dynamic forces are in general not treated, and the time dependence inherent in the analysis (creep) is handled incrementally.
4. Elastic constants are isotropic and constant within a cylindrical ring.
5. The total strain can be written as the sum of elastic and non-elastic components.

The fuel stack and cladding are treated as a continuous, uncracked medium, no discontinuities are allowed in their displacements.
The non-elastic component of the strain includes several contributions: thermal strain, swelling. Plasticity and creep, a contribution taking into account for the pellet cracking.

3.4 Code verification

Since its inception, the development as well as the verification of the code is carried out following rigorous quality procedures, and is organised in three steps. The first step consists of verifying the mechanical–mathematical framework. To this end, the models in the code are compared with exact solutions, which are available in many special cases (analytical verification), and several solution techniques are tested, which are applied in order to optimise the numerical analysis. During the second step, extensive verification of separate models incorporated in the fuel performance code is performed on the basis of separate-effect data. Finally, in the third and last step the verification is completed by code-to-code evaluations as well as comparison with experiments in the frame of international benchmarks organised by the IAEA. An overview of the main experimental data used for TRANSURANUS verification is shown in
Figure 6.

Experiment	Fuel type	Nr. of rods	Reactor	Burn-up
Contact	UO_2	3	PWR and Siloe	23 MWd/kgHM
Osiris	UO_2	4	PWR, Osiris	23-50 MWd/kgHM
HBEP	UO_2	28	BR3	60-65 MWd/kgHM
IFA 429	UO_2	3	HBWR (PWR)	60 MWd/kgHM
IFA 432	UO_2	5	HBWR (BWR)	30-34 MWd/kgHM
IFA 503.1,2	UO_2	15	HBWR (WWER,PWR)	15-26 MWd/kgHM
IFA 504	UO_2	4	HBWR	50 MWd/kgHM
IFA 508	UO_2	1	HBWR	17 MWd/kgHM
IFA 515	UO_2, $(U,Gd)O_2$	6	HBWR	96 MWd/kgHM
IFA 533	UO_2	1	HBWR	
IFA 534		2	PWR, HBWR	52 MWd/kgHM
IFA 535.5,6	UO_2	4	HBWR	43 MWd/kgHM
IFA 597.3	UO_2	3	BWR, HBWR	52 MWd/kgHM
IFA 633	MOX, UO_2	6	HBWR	43 MWd/kgHM
IFA 636.1,2	UO_2, $(U,Gd)O_2$	4	HBWR	30 MWd/kgHM
IFA 650.2,3,4	UO_2	3	PWR, BWR	0, 82, 92 MWd/kgHM
IFA 651.1,2	MOX	2	HBWR	32 MWd/kgHM
IFA 663	Zry, M5, E110, Zirlo		HBWR	9000 hours
IFA 681	UO_2, $(U,Gd)O_2$	6	HBWR	20-30 MWd/kgHM
IFA 597.4,5,6	MOX	2	HBWR	32 MWd/kgHM
Kola3	UO_2	32	WWER-440 and MIR	46-48 MWd/kgHM
Risoe-1	UO_2	11	HBWR,DR3	32 MWd/kgHM
Risoe-2	UO_2	15	HBWR,DR3 (BWR)	27-42 MWd/kgHM
Risoe-3	UO_2	16	HBWR,DR3	13-46 MWd/kgHM
REGATE	UO_2	1	PWR and Siloe	47 MWd/kgHM
SOFIT-1	UO_2	12	WWER-440 and MIR	10 MWd/kgHM
SUPER RAMP	UO_2	28	PWR, R2	33-45 MWd/kgHM
TRIBULATION	UO_2	19	BR3,BR2	20-51 MWd/kgHM
DOE WG-MOX	WG-MOX	9	ATR	20-50 MWd/kgHM
PRIMO	MOX	1	BR3 and Osiris	30 MWd/kgHM
OMICO	UO_2, MOX	4	BR2	10-15 MWd/kgHM
M501, M502	SBR-MOX	8	PWR	35-55 MWd/kgHM
Zaporoshye	UO_2	22	WWER-1000	42-51 MWd/kgHM
Novovoronezh	UO_2	15	WWER-1000	47 MWd/kgHM
FUMEX-I	UO_2	6	PWR, HBWR	16-55 MWd/kgHM

Figure 6 – Overview of the main integral experimental data used for the verification of TRANSURANUS

4 Using TRANSURANUS in experiment modelling

The wide range of situation TRANSURANUS code is able to deal with includes the simulation of experiments. Anyway, the modelling of experiments (as well as off-normal and incident condition modelling) usually requires extra care in fuel pin settings, correlation and boundary conditions selection since in this situations fuel and cladding are often overstressed. In addition, the simulation of experiments often requires the use of the so called restart option needed to cope with cladding refabrication.

4.1 Boundary conditions

TRANSURANUS is a computer code for the analysis of thermal-mechanical behaviour of a single fuel pin. Even if the code is able, in stationary conditions, to simulate the cooling channel surrounding the fuel pin, the thermo-hydraulic simulation it is not its primary goal. TRANSURANUS it is not even a neutron code. In non-stationary conditions (but also in stationary conditions it is recommended), it is necessary to provide to the TRANSURANUS code time dependent boundary conditions values for relevant quantities in order to allow the thermal-mechanical simulation. The physical quantities the boundary conditions can be provided for include:

- Linear rod power
- Fast neutron flux
- Coolant flow rate
- Mean neutron energy
- Heat coefficient transfer in the gap
- Coolant temperature
- Coolant pressure
- Gamma heating in the cladding, coolant and structure

Not all the listed quantities are always necessary.
The boundary conditions (linear power, coolant temperature and pressure, neutron flux etc.) can be provided as measured experimental data or calculated using other codes (i.e. thermo-hydraulic codes).
According to the TRANSURANUS manual, in the LOCA simulation coolant temperature and heat transfer coefficient must be provided since a complete thermo-hydraulic analysis of the sub-channel is not feasible by the code.

4.2 Restart option

The fuel rods irradiated in experimental setups are often refabricated segments coming from a bigger father rod irradiated in a commercial nuclear reactor. The refabrication procedure usually implies the loss of the fission gas released by the fuel and stored in the cladding and the refilling with other gases as well as the change of some geometric parameter (i.e. the upper plenum volume). Moreover experimental reactor characteristics will be in general different from the commercial one the rod is coming from. In order to allow the proper simulation of the whole

irradiation history of the fuel rod behaviour it is necessary to cope with the changes of this quantities.
TRANSURANUS code includes a restart option manageable by the input file. The option allows stopping the simulation, storing the simulation variables in a file and continues the simulation from the stopping point. Unfortunately, no way to change simulation parameters and variables is available using only the TRANSURANUS standard input file.
Together with TRANSURANUS, a restart package, allowing the ability to change parameters and variables values, is provided [1]. The use of the restart package is quite uncomfortable compared to the normal use of the code. The procedure requires the modification of a FORTRAN source code, the compilation of the package and the execution of the generated executable. The procedure is sketched in Figure 7.

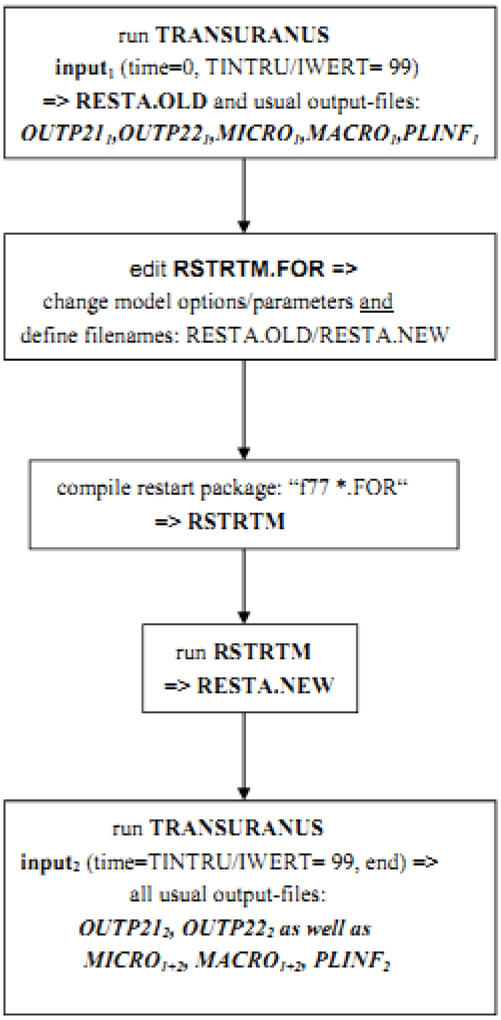

Figure 7 – Restart option flowchart

One of the main complications of the TRANSURANUS restart option is that the names of the variables one can change in the restart package source code are different from the names of the variables set in the standard input file. This is because the variables in the standard input file are used to calculate the values of other variables actually used in the simulation. As an example, in the input file, the initial content of gas in the pin is specified by the filling pressure, the filling temperature and the fractional content of ten gas species. This data, together with the pin geometric parameters allow the calculation of the amount (in micromol) of gas in the rod by the input file reading procedure. Instead, the change of the gas content during the restart, requires the value of the amount of gas (in micromol) to be directly set in the restart source code. The mentioned restart procedure is quite powerful, since virtually any parameter set in the input file can be changed. Anyway, because of its unfriendliness it is useful if not necessary a good knowledge of the TRANSURANUS code structure to cope with it.

4.3 Sensitive input quantities

The main goal in modelling experiments is to verify the ability of the computer code to reproduce the experimental results. In order to use experimental data to validate models and material properties, the errors on the prescribed quantities (see devoted section above) have to be minimized.
It is important that experiment specifications include data about prescribed quantities sensible to the experiment. A non- comprehensive list of possible sensitive input quantities is reported in Table 2.

Prescribed quantities	Simulated quantities affected
Neutron flux	He production, materials embrittlement
Dishing volume	Free volume, FGR, inner pin pressure
Porosity	
Upper and lower plena volumes	

Table 2 – Typical TU prescribed quantities and affected simulated quantities

4.4 LOCA simulation

IN TRANSURANUS, the LOCA analysis is carried out in an automatic restart run, which allows the initiation of the above LOCA-specific models either for fresh fuel rods (only LOCA simulation) or fuel rods with specific burnup (simulation of normal operation and LOCA). This latter method provides the unique possibility of the consistent simulation of fuel rod performance under normal operation and accident conditions. The model and material property options for the LOCA analysis are defined separately. The transition from normal to LOCA-specific models occurs at a time point defined by the user. Generally this time point corresponds to the time of the LOCA initiation. The appropriate boundary conditions for the accident analysis (decay power, coolant pressure, coolant temperature and heat transfer coefficient) must be specified in the input.
On the basis of the defined boundary conditions the code calculates the temperature distribution and the fission gas release inside the fuel rod, the corresponding inner pressure, the ZrO2 thickness growth, the equivalent oxidation (ECR) and the plastic deformation (ballooning) of the cladding. Fuel rod burst is checked through appropriate failure criteria.
Owing to the above features the TRANSURANUS code can also be applied in design basis accident (DBA) analyses to complement system level simulations and to verify the fuel-specific safety acceptance criteria on the basis of detailed thermo-mechanical computations.

However, some limitations of the code have to be considered:

- A complete thermo-hydraulic analysis of the sub-channel is not feasible and therefore the coolant temperature and the heat transfer coefficient have to be prescribed as boundary conditions on the basis of detailed thermo-hydraulic analyses.
- Simulation of post-LOCA events is limited by the validity range of the applied correlations and material property functions. In general, the present LOCA-specific models are validated up to 1200 °C.

4.5 High burn up

TRANSURANUS code includes several parameters to allow the code to cope with the High Burnup Structure (HBS). The main quantities affected by the HBS are the fission gas production and release.

5 Possible improvements

5.1 Restart option

The ability to change parameters as filling gas composition and pressure, reactor type, plenum volume, is mandatory in order to proper simulate fuel pin behaviour. This is possible using the restart package available in TRANSURANUS (see devoted section above). Anyway a more user friendly approach would improve the use efficiency of the code. An in-depth approach, where the parameters and variable change at the restart would be possible by using the standard input file would require the involvement of the code developers. A procedure to improve the automation of the current restart procedure can be written using a scripting language (Perl, Python, etc.).

5.2 Visualization tools

TRANSURANUS code comes with a post processing package allowing the analysis of the performed simulations. The tool has advantages and drawbacks. Among the advantages we can mention the ability to plot virtually any variable of the simulation, the possibility to run in batch mode to perform the same post processing analysis on different simulations, a steep learning curve, the possibility to export data in ASCII format. The main drawbacks are the lack of 3D plot, the limit on the maximum number of curves plotted at the same time, the amendable of the general usability of the GUI. Even if this tool probably cannot be completely replaced, the possibility to use a more sophisticated visualization tool would be appreciable. A possible tool we can cite is “Paraview”, developed by a private company in collaboration, among the others, with Sandia National Laboratories and Los Alamos National Laboratories. It allows 3D plot and sophisticated data manipulation. It is open source software released under the BSD license. To use it to analyse TRANSURANUS data, the development of a proper module (a sort of reading plug.in) would be necessary.

5.3 Code parallelization

In contemporary computers, the performance improvement is achieved increasing the number of computational nodes (or the numbers of the processors in a computer, or the number of the cores in a processor) instead of the power of the single processor. The exploitation of the computational power requires codes able to run in parallel. TRANSURANUS is a serial code and parallel execution of any kind is not possible out of the box. User level parallelization can be implemented by running in parallel several instances of TRANSURANUS, to perform simulation of different pins or simulation of the same pin with different parameters in a statistic or a sensitivity analysis. In-depth parallelization of the simulation of a single fuel pin requires (probably a lot of) source code modification.

6 Conclusions

An overview of the Halden ractor Project experiments planned in the next years has been presented. The objectives are quite wide, because various issues are considered, covering both actual and innovative materials; different plant design (PWR, BWR and VVER); fuel behavior under accident conditions. The gathering of experimental data will surely improve the understanding of fuel mechanical behaviour, explicitly considering the actual trend of increasing the burn up. A parameter already demonstrated to have a major influence on observed fuel phenomena, such as fission gas generation and release, hydrogen generation, PCI, etc.

Experimental data may serve also to verify if correlations embedded in fuel mechanic codes are still valid to simulate recent fuel operation and innovative material behaviour. Hence databases coming from Halden Project have to considered of high relevance because they address phenomena still not completely understood and consequently not fully replicable by computational tools.

On this aspects an overview of TRANSURANUS code is provided, discussing its adequacy and including possible improvements.

Reference

[1] http://www.oecd-nea.org/jointproj/halden.html

[2] M.Adorni "METHODOLOGY FOR THE ANALYSIS OF FUEL BEHAVIOR DURING LOCA AND RIA" ,Phd thesis (2011).

[3] Van Uffelen P., Modeling of Nuclear Fuel Behavior, Publications Office, JRC Publications, Report EUR 22321 EN, European Commission, 2006.

[4] Lassmann K., A. Schubert, P. Van Uffelen, Cs. Gyory, J. van de Laar, Transuranus Handbook Version "v1m1j11", EC, JRC, ITU, July 2011.

[5] *Nordström L. Å. TRANSURANUS Restart Option applied to a Fuel Rod Test (IFA-550.11) TRANSURANUS user meeting, ITU, Karlsruhe, 22 - 23 September, 2003.*

Curriculum Scientifico del Gruppo di Lavoro

F. D'auria
Professore Ordinario di Ingegneria del Nocciolo (Modulo dell'insegnamento Termoidraulica e Ingegneria del Nocciolo Cod. 424II) per il Corso di Laurea Magistrale in Ingegneria Nucleare - Università di Pisa.
Professore Ordinario di Termoidraulica (Modulo dell'insegnamento Termoidraulica e Ingegneria del Nocciolo Cod. 424II) per il Corso di Laurea Magistrale in Ingegneria Nucleare - Università di Pisa.
Autore di oltre 100 articoli su rivista:
http://arp.unipi.it/listedoc.php?ide=5808

F. Cantini
Collaboratore dal 2011 presso il Gruppo di Ricerca Nucleare S. Piero a Grado - Università di Pisa.
Autore di articoli scientifici facilmente reperibili sui principali motori di ricerca specializzati.

M. Cherubini
Dottore di Ricerca in Sicurezza Nucleare.
Borsista/Assegnista dal 2003 presso il Gruppo di Ricerca Nucleare S. Piero a Grado - Università di Pisa.
Autore di articoli scientifici facilmente reperibili sui principali motori di ricerca specializzati.

D. Lazzerini
Borsista/Assegnista dal 2007 presso il Gruppo di Ricerca Nucleare S. Piero a Grado - Università di Pisa.
Autore di articoli scientifici facilmente reperibili sui principali motori di ricerca specializzati.