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

Ministero dello Sviluppo Economico

RICERCA DI SISTEMA ELETTRICO

Rapporto "Programma di collaborazione ENEA-HRP"

M. Cappelli, A. Del Nevo, D. Rozzia, G. Gola, M. Lind

Report RdS/2012/147

RAPPORTO “PROGRAMMA DI COLLABORAZIONE ENEA-HRP”

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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 Progetto: Massimo Sepielli, ENEA

Titolo

Rapporto "Programma di collaborazione ENEA-HRP"

Descrittori

Tipologia del documento: Rapporto Tecnico

Collocazione contrattuale: ACCORDO DI PROGRAMMA Ministero dello Sviluppo Economico – ENEA sulla Ricerca di Sistema Elettrico PIANO ANNUALE DI REALIZZAZIONE 2011 Progetto 1.3.1: Energia nucleare: NUOVO NUCLEARE DA FISSIONE: COLLABORAZIONI INTERNAZIONALI E SVILUPPO COMPETENZE IN MATERIA NUCLEARE, PAR 2011.

Argomenti trattati: Interfaccia Uomo-Macchina, Fattori Umani, Comportamento dei materiali sotto irraggiamento, Comportamento del combustibile nucleare

Sommario

Il presente documento descrive l'attività svolta dall'ENEA nel contesto della sua partecipazione al Progetto internazionale dell'OECD/NEA Halden Reactor Project (HRP).

L'ENEA è entrata nuovamente a far parte del Progetto nell'annualità 2011, dopo una breve pausa di qualche anno, come Membro Associato. Anche nella presente annualità, l'ENEA ha partecipato come Membro Associato.

In questo documento sono riportati due studi relativi rispettivamente al settore della Strumentazione e Controllo (svolte con il contributo di Giulio Gola dell'Halden Reactor Project e del Prof. Morten Lind della Danish Technical University) e al settore del Fuel and Materials (svolto in collaborazione con Davide Rozzia dell'Università di Pisa).

Il primo studio descrive una tecnica di modellazione, il Multilevel Flow Modeling (MFM) utilizzata per monitoraggio, diagnostica e prognostica ai fini della sicurezza per impianti attuali e futuri. I risultati generali sin qui ottenuti serviranno come base per applicazioni future a impianti reali.

Il secondo studio riguarda il problema della chiusura del gap nella camiciatura di una barra di combustibile. Per analizzare tale fenomeno è stato condotto un esperimento con database Studvisk sulla risposta del cladding a rampe di potenza. Lo studio serve come base di riferimento per esperimenti dello stesso tipo in corso di svolgimento ad Halden ai quali si prenderà parte in futuro.

Viene infine riportato un documento a cura dell'Università di Pisa in cui si analizzano in dettaglio le varie attività su combustibile e materiali previste dal Programma HRP 2012-14.

Note

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Sommario

DESCRIZIONE GENERALE DELLA PARTECIPAZIONE ENEA AL PROGETTO HALDEN REACTOR PROJECT

L'anno 2012 ha rappresentato il secondo anno della rinnovata partecipazione dell'Italia al progetto internazionale OECD/NEA Halden Reactor Project (HPR).

Attualmente, ENEA è in grado di accogliere la richiesta economica fatta da HRP per la partecipazione come Membro Associato per l'anno 2012, con una quota pari ad 80.000 € e rinnovabile di anno in anno per tutta la durata del programma triennale (2012-2014). Tale quota sarà finanziata attraverso i fondi del PAR 2011. Anche altre forme di contribuzione *in kind* (per es., con accordi di *secondments* di ricercatori ed esperti su tematiche di comune interesse) saranno possibili per gli anni successivi, attraverso appositi accordi con IFE.

La necessità di un approfondimento generale degli aspetti di sicurezza e delle prestazioni nel funzionamento degli impianti, conferma le condizioni per lo sviluppo delle competenze in settori fondamentali come il *Fuel&Materials* e *Instrumentation&Control*. In accordo a quanto riportato nell'AdP ENEA-MSE (PAR 2011), "il rilancio della partecipazione italiana all'Halden Reactor Project permetterà di avvalersi dell'esperienza di un Centro internazionale di eccellenza nel dominio *Fuel&Materials* e *Instrumentation&Control*, a supporto dell'esercizio sicuro ed efficace dei reattori attuali e futuri. Temi di particolare attenzione sono il comportamento del combustibile e materiali strutturali in condizioni di alto *burn-up* e lunghi tempi di residenza in reattore, insieme alle tecnologie digitalizzate *Man Machine Interface* (MMI) per il controllo reattore. Tali attività sono consistenti con i progetti attualmente in corso finanziate dal MISE e dalla comunità europea. La partecipazione offre anche l'opportunità di scuola/training per giovani ricercatori, insieme a quella di agire quale riferimento di base per lo sviluppo di un simulatore ingegneristico in ENEA". Inoltre, gli studi in entrambi i settori potranno essere di fondamentale importanza anche per le ricadute in settori paralleli a quelli della produzione elettrica, come la gestione dei rifiuti radioattivi, la chiusura del ciclo del combustibile e il *decommissioning*.

La partecipazione ad HRP è attesa avere, pertanto, ricadute benefiche sia sul processo di formazione delle nuove risorse umane, sia sul ruolo di possibile TSO (Technical Support Organization) nei confronti dell'Autorità di Sicurezza Nucleare, a cui l'ENEA intende candidarsi.

Il programma di ricerca e sperimentazione **Halden Reactor Project (HRP)** costituisce uno dei più importanti progetti di R&S sul Nucleare da Fissione promossi dalla OECD-NEA (Nuclear Energy Agency), avviato nel 1958 nell'ambito di un accordo internazionale di collaborazione tecnologica incentrato sulla sicurezza dei reattori nucleari ed in particolare sugli studi e ricerche sul combustibile, i materiali di nocciolo e circuito primario, insieme alla conduzione di attività sperimentali per il miglioramento dei sistemi di controllo reattore.

I programmi-progetti internazionali promossi dalla OECD-NEA sono mirati a dare risposta a questioni rilevanti per la comunità nucleare attraverso un'attività di ricerca comune svolta da tutti i Paesi aderenti. Essi permettono di migliorare l'interscambio tecnico-scientifico e la cooperazione internazionale, sostengono la continuità operativa di impianti sperimentali ritenuti di interesse primario per la comunità nucleare dei Paesi membri, e supportano il mantenimento dell'esperienza e delle infrastrutture strategiche per l'energia nucleare, facilitando il raggiungimento di risultati comuni attraverso la condivisione dei costi fra i partecipanti. Essi forniscono, inoltre, l'opportunità di accesso a tutte le conoscenze sul nucleare oggetto del Programma Triennale comune (Joint Programme), in un contesto internazionale mirato alla promozione di ulteriori collaborazioni, acquisizioni e miglioramento di conoscenze.

Il Programma è strutturato su due aree principali, *Fuel and Materials* e *MTO (Man Technology Organization)*, le cui attività possono avvalersi di facilities fondamentali quali il reattore HBWR (Halden Boiling Water Reactor) e i laboratori di simulazione e modellistica per l'interfaccia uomo-macchina HAMMLAB (Halden Man Machine Laboratory) e HVRC (Halden Virtual Reality Center), del centro di ricerca di Halden in Norvegia, situato circa 100 chilometri a sud di Oslo.

HRP è cresciuto storicamente intorno al reattore internazionale di ricerca HBWR e utilizzato sin dagli anni '60 per la sperimentazione di combustibili nucleari e di materiali destinati alle diverse tipologie di reattori esistenti. Il reattore, della potenza massima di 25 MWt, è moderato ad acqua pesante e refrigerato ad acqua bollente, alla pressione di 33,3 bar e a una temperatura di uscita di 240 °C; è collocato all'interno di un tunnel scavato in una formazione rocciosa di granito tipico della Scandinavia, non lontano da una grande cartiera alla quale fornisce una parte rilevante del calore di processo sotto forma di vapore.

La riapertura della partecipazione all'Halden Reactor Project permetterà di avvalersi dell'esperienza di un Centro internazionale di eccellenza nel dominio *Fuel&Materials* e *Instrumentation and Control*.

L'Italia ha partecipato stabilmente a HRP dalle origini fino all'inizio degli anni '90 quando la partecipazione fu interrotta in conseguenza degli effetti del referendum del novembre del 1987. La partecipazione fu ripresa verso la metà degli anni '90 e terminata nuovamente a fine 2002.

L'obiettivo primario dell'*Halden Reactor Project* è la produzione e il trasferimento di conoscenze, dati, risultati e prodotti rilevanti per l'esercizio sicuro ed efficiente degli impianti nucleari, ivi inclusi gli aspetti di *maintenance* e *decommisssioning*, attraverso la collaborazione di un'ampia comunità internazionale, condotta e sviluppata sulla base della ripartizione dei costi fra partecipanti.

Il progetto opera sulla base di un Programma Comune triennale (*Joint Programme*), sottoscritto dai partecipanti, sotto gli auspici della OECD/NEA. Il programma tecnico-scientifico è definito in accordo a criteri prioritari concordati dalle stesse organizzazioni partecipanti. L'esecuzione del Programma viene monitorato dall'*Halden Programme Group* (HPG) e da un *Board* internazionale, *Halden Board of Management* (HBM), che si riuniscono entrambi due volte l'anno. I risultati sono trasferiti ai partecipanti sotto forma di *reports*, data *files* e computer *programs*.

La partecipazione al HRP avviene come Paese Membro (*Member Party*) o Associato (*Associated Party*). E' prevista, tuttavia, anche la possibilità di partecipazioni bilaterali con quei Paesi o organizzazioni che possono avere esigenze specifiche da perseguire al di fuori del *Joint Programme*. L'invio di personale tecnico-scientifico dei paesi/organizzazioni partecipanti, a lavorare direttamente presso il Progetto, attraverso la formula del *secondement*, favorisce la formazione e il trasferimento delle conoscenze sviluppate.

Aree di attività

Il Progetto opera su due aree di attività, *Fuel&Materials e MTO*, i cui scopi principali sono:

Fuel and Materials

▪ *Fuel safety and operational margins*

○ *Thermo-mechanical studies under irradiation*

○ *Fuel behavior under demanding operation conditions*

○ *Fuel behavior under accident scenarios*

○ *Innovative fuels and claddings*

▪ *Plant aging issues and degradation*

○ *Internals irradiation assisted stress corrosion cracking*

○ *Creep and stress relaxation of in-vessel materials*

○ *Pressure vessel integrity studies*

▪ *Contribution to international GEN-IV research*

○ *Instrumentation development*

- o *Material testing*

Man-Technology-Organization

- ▪ *Human factors research for existing and new reactors*
- o *Human reliability*
- o *Human and organizational factors (includes decommissioning)*
- o *Human-System Interfaces*
- o *Control Centre Design and Evaluation*
- o *Outage and field work, including decommissioning*
- o *Future Operational Concepts*

- ▪ *Digital systems research for existing and new reactors*
- o *Software systems dependability*
- o *Condition Monitoring and Maintenance Support*
- o *Operational support of digital system*

Partecipanti e finanziamento

E' attualmente in corso il Programma triennale 2012-14 che vede la partecipazione di 18 Paesi, con in testa quelli più industrializzati nei quali l'energia nucleare rappresenta una realtà di dimensioni rilevanti (Tab.1). I Paesi Membri *(Member Parties)* sono, oltre alla Norvegia, Stati Uniti, Giappone, Francia, Germania, Regno Unito, Spagna, Belgio, Danimarca, Finlandia, Svezia, Svizzera, Corea del Sud, Russia, Ungheria, Rep. Ceca, Slovacchia e Kazakstan. Ai Paesi Membri si aggiunge un numero di istituzioni pubbliche e private, dell'industria e della ricerca, di vari paesi *(Associated Parties)*.

Il budget totale previsto, per il triennio 2012-14 (v. Tab. 2), ammonta a 415.4 milioni di NOK (circa 53 milioni di Euro) di cui 330.1 milioni di NOK provengono dai Paesi membri e 85.3 milioni di NOK dagli Associated Parties. La Norvegia, come Paese Membro, contribuisce con 145 Milioni NOK (35% del totale).

Interessi nazionali e obiettivi per il triennio 2012-2014

In accordo a quanto sopra riportato, l'interesse nazionale ad Halden Project, è rivolto ai seguenti obiettivi prioritari:

Area Fuel&Materials

- Stato su progettazione, comportamento e prestazioni del combustibile nucleare e materiali strutturali in condizioni operative rilevanti per i reattori LWR (Gen. III and III+).
- Definizione di linee di analisi, ricerca e sviluppo di più specifico interesse nazionale a supporto dello sviluppo dei reattori di **IV Generazione** (LFR, ESFR), e in particolare.

o combustibili UOX ad alto arricchimento e MOX per l'alto burn-up (>60 MWd/kg), combustibili e materilai di guaina innovativi, risposta del combustibile in condizioni di alto burnup, in regime di *funzionamento normale e incidentale*

o aspetti di back-end del ciclo combustibile (tecniche innovative di gestione e smaltimento del combustibile irraggiato e dei rifiuti nucleari, caratterizzazione, processi di trattamento e condizionamento, trasporto di materiali nucleari, progetto concettuale del deposito e sitologia, analisi di sicurezza, addestramento degli operatori con tecniche avanzate di simulazione)

o studio del comportamento di materiali di camicia innovativi sotto irraggiamento (corrosione e creep) per applicazione a reattori di IV generazione

Area MTO (Instrumentation&Control)

- Review generale sullo stato dell'arte dell'*Instrumentation and Control*
- Definizione di linee di analisi, ricerca e sviluppo di più specifico interesse nazionale, tra le quali:
  - Nuovi requisiti per la sensoristica, la strumentazione e il controllo per la modernizzazione di impianti di ricerca e facilities sperimentali
  - Sistemi di acquisizione, controllo e protezione avanzati (remotizzazione da sale manovra digitali e multi-unità)
  - Utilizzo delle tecnologie di Realtà Virtuale per la progettazione, pianificazione delle attività di manutenzione, visualizzazione del processo, simulazione e addestramento degli operatori, soprattutto per facilities adibite al trattamento dei rifiuti radioattivi.
  - Analisi di affidabilità del software in condizioni normali e incidentali
  - Studio e sviluppo di modelli supporto alle decisioni avanzati per la diagnostica e il controllo mediante metodi e modelli di ottimizzazione della sicurezza, come per esempio il Multilevel Flow Modeling (MFM), con applicazione alla diagnosi dei guasti in impianti e facilities reali.

Tab. 1 - Halden Project Participants, who have signed/confirmed their participation in the programme 2012- 2014

Norway	**IFE** - Institutt for energiteknikk
Belgium	**SCK/CEN** - Belgian Nuclear Research Centre
Denmark	**Risø DTU**
Finland	**TYÖ** – JA ELINKEINO-MINISTERIÖ - Ministry of Employment and Economy, operated by **VTT**
	- *Includes Finnish utilities*
France	**EDF** - Electricité de France
	IRSN is associated party
	CEA membership under discussion
Germany	**GRS** - Gesellschaft für Anlagen- und Reaktorsicherheit mbH
	Includes German utilities and AREVA
Japan	**JNES** - Japan Nuclear Energy Safety Organization
	Japan Associated Parties
	JAEA, Japan Atomic Energy Agency
	CRIEPI (incl. Toshiba and 11 utilities)
	Mitsubishi Nuclear Fuel (incl. MHI)
Korea	**KAERI** - Korean Atomic Energy Research Institute
	Includes the Korean utility (KEPCO) and the fuel vendor (KNF)
Spain	**CIEMAT** - Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas
	Includes the Spanish safety authority (CSN) fuel vendor (ENUSA)
Sweden	**SSM** - Swedish Radiation Safety Authority
	Includes the Swedish fuel vendor (W) and utilities
Switzerland	**ENSI** - Swiss Federal Nuclear Safety Inspectorate
	Includes the Swiss utilities and nuclear research centre (PSI)
UK	**NNL** – National Nuclear Laboratory
	Includes UK utilities and safety authority (Rolls-Royce under discussion)
USA	**US NRC** - United States Nuclear Regulatory Commission
	USA Associated parties:
	GE/ GNF - Global Nuclear Fuel
	Westinghouse Electric Power Company
	EPRI - Electric Power Research Institute
	DOE – US Department of Energy
Czech Rep.	**NRI** - Nuclear Research Institute, includes Czech utilities and safety authority
Slovak Rep.	**VUJE** Slovak Nuclear Power Plant Research Institute, includes Slovak utilities
Hungary	**KFKI** Hungarian Academy of Sciences, utilities and safety authority
Russia	**“TVEL” JSC**
	Russian associated parties
	Research Centre “Kurchatov Institute”,
	Research Institute VNIIAES
Kazakhstan	**UMP,** Fuel manufacturing company

Tab. 2 - Ripartizione delle spese totali del HRP-Joint Programme per il periodo 2012-2014

Institutt for energiteknikk
OECD HALDEN REACTOR PROJECT

HP-1343
Final –May 15, 2012

PROJECT BUDGET

Covering the Period 1st January to 31st December, 2012

I. EXPENDITURES

PROGRAMME ITEMS	*2012 (In parenthesis: staff/operating costs) In mill. Norw.kr.*	*TOTAL 2012/14*
Reactor Operations	35.9 (26.8/9.1)	108.3
High Burn-up Fuel Performance	30.9 (21.9/9.0)	93.7
In-Core Materials & Water Chemistry Effects	13.6 (10.0/3.6)	41.2
Man Technology Organisation	53.7 (42.5/11.2)	162.2
Reserves	3.1 (1.7/1.4)	10.0
Total Expenditures	137.2 (102.9/34.3)	415.4

II. INCOME

ITEMS	*2012*	TOTAL 2012/14
Signatory Contributions	108,8	330,1
Associated Parties	28,4	85,3
Total Income	137,2	415,4

Tab. 3 - Principali risultati delle performance ottenute dll'HPR nel periodo 2006-2011

Achievements – Reactor operation 2006-11

Reactor operations 2006-2008	2006-I	2006-II	2007-I	2007-II	2008-I	2008-II	Total	Avg/year
Integrated power, MW	1292	1239	1594	1173	1179	1887	8364	2788
Availability, %	44,8	40,2	50,4	38,5	49,1	53,14		46,0
Number of scrams	3	2	4	4	5	0	18	3
Number of slow scrams	3	2	3	3	4	1	16	2,7
Emission to air, % of half of year limit	6,8	6,4	8	4,8	5,4	9,4		6,8
Discharge to water, % of half the annual limit	1,4	2,4	1,8	1,6	1,4	1,2		1,6

Reactor operations 2009-2011	2009-I	2009-II	2010-I	2010-II	2011-I	2011-II	Total	Avg/Year
Integrated power, MWd	1485	1986	1597	1966	1114	2014	10162	3387,3333
Availability, %	49,2	53,1	49,9	62	39,1	70		53,9
Number of scrams	0	0	0	2	1	0	3	0,5
Number of slow scrams	7	2	2	5	5	1	22	3,7
Emission to air, % of half of year limit	7,2	7,2	5	6	3,6			4,8
Discharge to water, % of half the annual limit	1,0	1,0	1,2	1,0	1,0			0,9

FUNCTIONAL FLOW MODELS FOR REPRESENTING AND REASONING ABOUT FAULT CAUSES AND CONSEQUENCES IN NUCLEAR POWER PLANTS

Introduction

Innovative modelling approaches, techniques, and solutions are needed to support the monitoring, diagnostic and prognostic requirements of current and future nuclear power plant designs. Longer fuel cycles, reduced staffing, higher intrinsic safety and other related factors are all likely to play an important role in shaping these requirements in the direction of additional flexibility, robustness and efficiency when compared to the systems and techniques that are currently used or being developed today. Detecting malfunctions, identifying their causes and possibly predicting their consequences are major challenges, especially if extended to the whole plant.

Condition monitoring techniques are available to supervise plant operation and performances with the aim of detecting anomalies during operation, providing early warnings and eventually scheduling maintenance. Nevertheless, relevant challenges arise when operators are asked to handle the large amount of information provided by such systems and to relate it with the complex design of process and control logics.

For example, online monitoring techniques based on data reconciliation are currently available for early fault detection, i.e. for identifying abnormal residuals between measured and estimated parameters. Nevertheless, the actual analysis and interpretation of these results is typically a manual process. If one envisions the likely centralization of condition monitoring functions in fleet-wide monitoring centres, then it becomes evident that supporting functions such as automated diagnosis would become an all-important requisite.

Functional models offer a viable framework to handle the complexity of such tasks. A function-oriented modelling approach, called Multilevel Flow Modelling (MFM) [1], is illustrated in this report. MFM-based approaches have been successfully applied for early/hidden fault detection [2,3], diagnostics [4-10], on-line alarm analysis and filtering [11,12], to analyse faulty scenarios in HAZOP studies [13], modelling control purposes in power systems [14-19] and fault tree generation and risk analysis [20].

The means-end orientation of MFM exploits the principles of qualitative reasoning. MFM presents the plant processes and control logics at different levels of abstraction by defining the functions performed by the components toward the achievement of specific goals. Functions and goals are connected via causal relations. The propagation (backwards for fault diagnosis, forwards for prognostic purposes) of the information (e.g. related to system or sensor faults) is carried out by resorting to a rule-based reasoning approach which combines a number of generic rules with the actual casual relationships between functions specified in the MFM model.

MFM derive its power from the representation of process knowledge on several levels of specification within a means-end perspective. The detailed specification is at the basis for the implementation of automated model-based reasoning functions, whereas the more abstract specifications provide generic process representation and knowledge for the formulation of reasoning strategies and the understanding of the underlying operation and control logics. Reasoning strategies, together with their representation and explanation, can be directly visualized in terms of the means-end topology of the multilevel flow models and may be used for design of human machine interfaces supporting diagrammatic reasoning about spatial-temporal aspects of dynamic situations. The principles described in the report have been used within the implementation of a model-based reasoning system.

In the followings, the concepts, the overall principles of reasoning and representation of process knowledge and control logics in MFM are described with the support of some examples.

Multilevel flow modelling

Multilevel flow modelling is a modelling methodology for process and automation design and for reasoning about fault management and control of complex plants. MFM has been developed for more than two decades and a recent up-to-date introduction to MFM can be found in [1]. MFM belongs to a branch of Artificial Intelligence (AI) research called qualitative reasoning aiming at representation and reasoning about qualitative knowledge of physical phenomena and systems. MFM realizes these aims within the general domain of industrial processes and their automation systems. A particular challenge in MFM research is to develop qualitative modelling and reasoning techniques that can handle the complexity of large-scale dynamic processes.

The conceptual foundations, the development of MFM modelling languages, tools and applications have been on-going for more than two decades. The basic ideas of MFM were conceived by Prof. Morten Lind [5] and have been developed over the years by his research group and by research groups in several other countries, including the OECD Halden Reactor Project in Norway. The research has originated in problems of representing complex systems in Human Machine Interfaces (HMI) for supervisory control, but has developed into a broader research field dealing with modelling for design and operation of automation systems for safety critical complex plants. MFM concepts and symbols are shown in Figure 1.

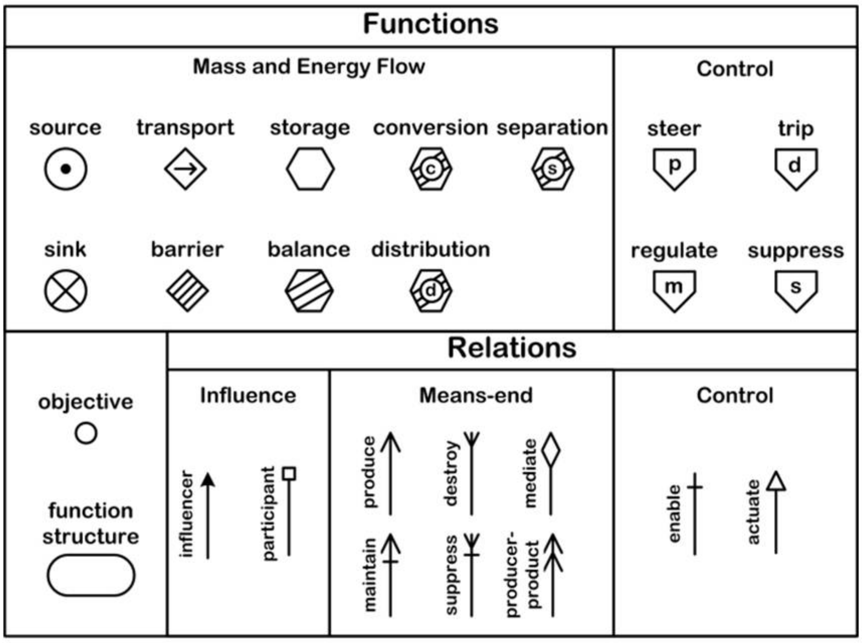

Figure 1. MFM concepts and symbols

MFM concepts

Means-end and whole-part decomposition and aggregation play a foundational role in MFM. These concepts enable human-like systems, engineers and plant operators to cope with complexity by allowing reasoning about situations on different levels of abstraction. The power of means-end and part-whole concepts in dealing with complexity has roots in natural language [5]. But natural language is not efficient for representing and reasoning about complex physical artefacts. MFM development draws on insights from natural language semantics, but is designed as an artificial language which can serve modelling needs of complex engineering domains which cannot be handled within the common sense limitations of natural language.

Representation of process knowledge in MFM

Efficient representation of process knowledge is a prerequisite for all knowledge-based systems reasoning about complex industrial processes. MFM models are efficient for this purpose because they combine process knowledge on four interdependent levels of specification as shown in Table 1.

Table 1. Reasoning about causes and consequences in MFM is based on process knowledge on four levels of specification

<table>
<tr><th>Level</th><th colspan="2">Knowledge categories</th></tr>
<tr><td>4</td><td colspan="2">Event propagation paths</td></tr>
<tr><td rowspan="2">3</td><td colspan="2">MFM patterns</td></tr>
<tr><td>Influence patterns</td><td>Means-end and control patterns</td></tr>
<tr><td>2</td><td>Influence relations</td><td>Means-end and control relations</td></tr>
<tr><td>1</td><td colspan="2">State dependency relations</td></tr>
</table>

On the most fundamental level 1, state dependency relations represent cause-effect and logic relations between states of process objectives and functions. The state dependency relations are generic and are instantiated by matching them with influence relations and means-end and control relations on level 2 and with influence, means-end and control patterns involving goals and functions in the MFM model on level 3.

Influence, means-end and control relations represent potentials for interaction between the state of goals and functions. They comprise a separate level of representation which is more abstract than level 1 by only indicating the existence of a state dependency relation between two flow functions and not the specific states dependency relations which are represented on level 1. Influence and means-end relations combine functions and goals into MFM patterns on level 3. MFM patterns are generic combinations of goals and functions and are the building blocks of event propagation paths on level 4. Event propagation paths are important for representing the temporal unfolding of a disturbance in MFM models and for formulation of strategies for reasoning about causes and consequences of the disturbance. The paths shown in Figure 3 are examples of using event propagation paths for reasoning about causes and consequences and of visualizing the spatial-temporal development of the disturbance in the means-end topology. Note that the explanation of the reasoning strategy is based purely on process knowledge on level 4. It does not require explicit reference to the mechanisms of the underlying detailed inference processes which combine knowledge on levels 1, 2 and 3.

MFM patterns, influence relations and means-end and control relations provide efficient encodings of process knowledge for diagnosis and control. The ability to distinguish and combine knowledge on these three levels in reasoning about dynamic situations makes MFM a powerful representation for model based reasoning.

Knowledge on all four levels of specification is integrated in the fact and rule bases used for reasoning about causes and consequences in the MFM reasoning system currently under development at the Technical University of Denmark and the OECD Halden Reactor Project. The reasoning system is part of the MFM workbench which also includes tools for model building and verification.

The water mill example

The overall principle for reasoning about causes and consequences in a Multilevel Flow Model is illustrated in Figure 2 by a MFM model of a water mill.

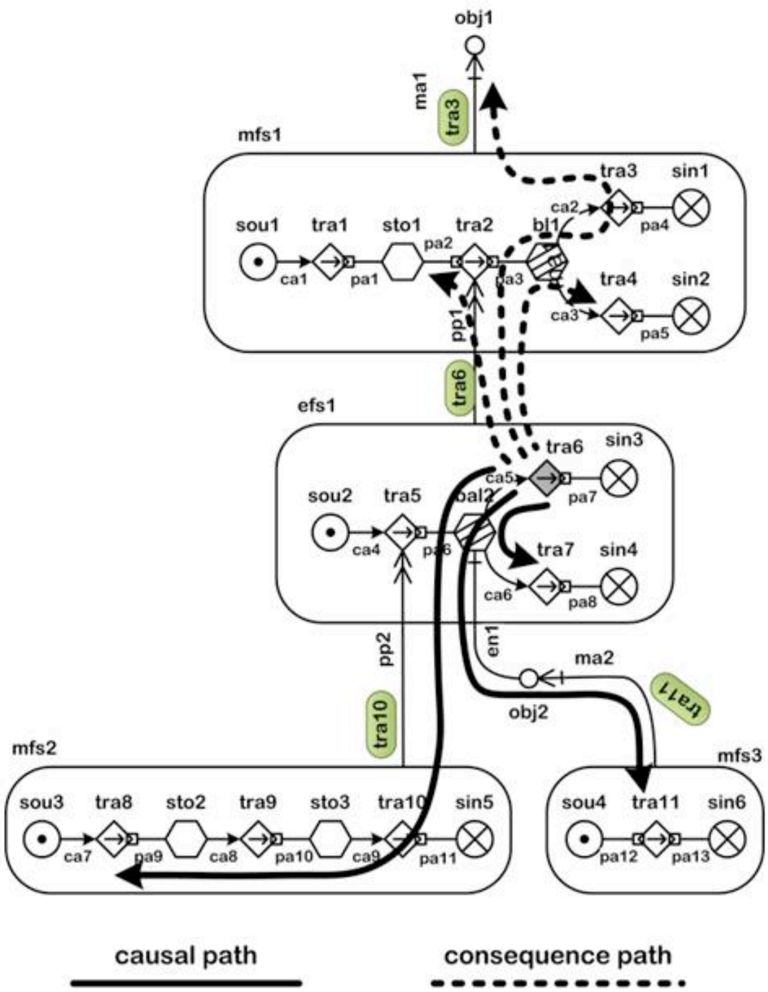

Figure 2. Reasoning about causes and consequences of a disturbed transport function tra6 in an MFM model of a water mill [1]

The overshoot water mill shown in Figure 2 is a hydraulically powered flour grinding process and will used to illustrate the basic ideas of MFM.

A water mill is machine that uses running water to drive a mechanical process such as grinding. Water is diverted from a river along a channel known as the flume. On the flume, a sluice gate controls the water flow rate. A wheel is rotated by the momentum of the falling water striking and filling the buckets of the wheel. The weight difference between the side with water filled buckets and the side with empty buckets turns the wheel, which in turn rotates a drive shaft with a toothed wheel. By means of the horizontal toothed wheel, the angle of rotation changes and rotates a spindle on which a runner stone is mounted. The runner stone spins above a stationary bed stone to create the grinding action. The runner stone has a hole at the centre into which the grain is fed. As the grain is ground between the two stones, it moves towards the outer edge, and passes as flour into the casing. The grinding process also separates the grain shells from the flour through a sieve. This hydraulic process only has two control possibilities available to the miller: 1) to adjust the water flow rate by means of the sluice gate, and 2) to adjust the grain feeding rate to the runner stone.

The MFM model can be subdivided into three functional levels indicated in Figure 2 by the flow structures *mfs1*, *efs1* and *mfs2* which represent functions of the water mill related to

- grinding of the grain (mass flow structure *mfs1*)
- converting the potential energy of the water into kinetic energy (energy flow structure *efs1*)
- delivering and transporting water (mass flow structure *mfs2*)

The MFM model shows how these functions of the water mill can be organized in levels as means to achieve ends. The functional levels are connected by two means end relations of the *producer-product* type representing the fact that the transportation of water to the wheel (*mfs2*) is a means of converting energy (*efs2*) and that the conversion and consumption of energy is a means of transporting the grains through the grinding stone and the bed and the grinding of the grains into flour (*mfs1*). The miller can control the operation of the water mill by changing the feed of grain to the grinding stone or by the manipulation of the sluice gate.

The grinding

The flow functions belonging to flow structure *mfs1* represent the functions involved in grinding the grains. The grinding process supports the achievement of the objective *obj1* which is to deliver flour to the miller. The provision of grain is represented by the source function *sou1* and the transportation of the grains from the source to the storage *sto1* in the funnel above the grinding stone is represented the transport function *tra1*. From the funnel the grain is transported (*tra2*) to the grinding stones. The balance function *bal1* represents the conversion of grains into flour and shells and the transportation of flour (*tra3*) and shells (*tra4*) to two sinks (*sin1* and *sin2*).

The means-end relation from flow structure *efs1* connects the transport function *tra2* with the means of transportation represented by the flow structure *efs1* discussed below. The grinding can be controlled by adding more grains to the funnel i.e. by increasing the transportation of grains represented by *tra1*.

The energy conversion

The flow functions belonging to energy flow structure *efs2* represent the functions of the water mill involved in the conversion of the kinetic energy of the running water (*sou2*) into rotational energy of the wheel, the connecting mechanical linkages and the stone (*bal2*) and subsequent consumption of the energy in the grinding (*tra6*) and the production of heat by friction (*tra7*).

The means-end relation connects the energy source *sou2* with the means used for energy conversion i.e. the functions involved in leading the water from the flume to the wheel. These functions are represented in flow structure *mfs2*.

The water transport

The flow functions belonging to flow structure *mfs2* represent the functions of the water mill involved in the delivery and transport of water from the flume to the wheel and away from the wheel. The water source located upstream the flume is represented by the source *sou3* and the transportation and storage of the water in the flume is represented by *tra8* and *sto2*. It is accordingly assumed that the river (or lake) supplying the water has a sufficient capacity for the present purpose so that it can be represented as a source. Flow functions *tra9* and *sto3* represent the functions involved in transporting the water to the wheel and the water in the buckets of the wheel and *tra10* represents the transportation of water away from the wheel to its downstream destination represented here as a sink (*sin5*). The flow of water through the water mill can be controlled by the sluice gate (*tra8*).

Lubrication functions

The MFM representation of the lubrication system illustrates the use of the *enable* relation which is important for representing control functions.

The purpose of the lubrication system is to ensure that the mechanical linkages of the mill can rotate and thereby support the transfer the rotational energy. This support function of the lubrication system is represented by an enable relation connecting an objective *obj2* with the balance function *bal2*. Objective *obj2* is a representation of the lubrication requirement which is related to the lubrication functions shown in the flow structure *mfs3* by the means-end relation called maintain. The node *sou4* in *mfs3* represents the source of lubrication oil and the transport function *tra11* is the transport function performed by e.g. a lubrication pump.

Reasoning paths

Six reasoning paths are shown. Three paths follow the propagation upwards of consequences of a disturbed transport function *tra6*. One path goes directly to the objective *obj1* and the other paths show the possible propagation of the disturbance to storage *sto1* or transport *tra4*. Three paths trace the disturbance downwards in the model to its possible three root causes (disturbances of transport functions *tra8*, *tra7* or *tra11*). The reasoning paths in MFM models follow influence, means-end and control relations (Fig.1).

The paths traverse the model in both a horizontal and vertical direction. The horizontal direction follows the functions and their interconnection within a function (mass flow) structure (e.g. *tra2* to *tra3* and *tra2* to *sto1* in *mfs1*). The vertical direction follows the means-end relations connecting function structures with functions (e.g. *efs1* - energy flow - and *tra2* in *mfs1* are connected by a *producer-product* relation *pp1*).

The reasoning principles described above have been developed in parallel with the MFM modelling language for more than two decades. The core concepts which still are foundational for MFM reasoning were developed in [4-6].

Representation of control logics in MFM

A representation of control systems based on action theory has been added more recently to MFM [16-18]. The four elementary control functions, which are based on elementary action types, are found in Figure 1.

In contrast to the classical signals and systems perspective, control functions have a special role in the perspective of mean-ends modelling: whereas a 'flow-structure' is a functional abstraction of a process, the 'control-structure' is a representation of the intentional structure realized by a control system. This distinction becomes essential when reasoning about control systems. An example of a control structure and a related flow structure is given in Figure 3.

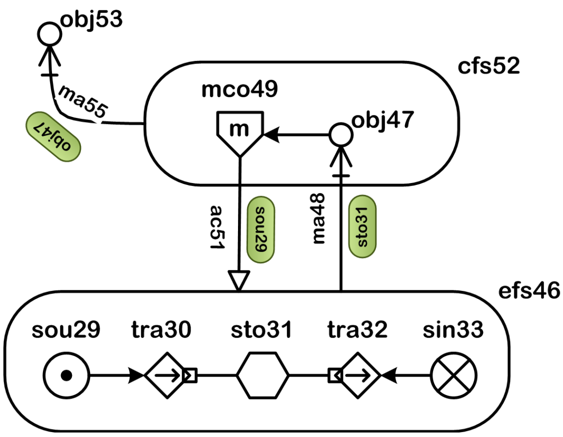

Figure 3. Example MFM Model with energy flow structure and control structure. The energy-flow structure efs46 models a stereotypical balancing process, where both the energy-source on the left and the energy sink on the right influence the storage-level. In this example, the process is balanced by means of a control which aims at maintaining the storage-level by means of actuating the energy source.

Control-objective *obj47* and control function *mco49* are encapsulated in a control structure *cfs52*. Requirements to the performance of the control are formulated as an objective associated with the control structure (performance objective, *obj53*). The control objective is associated via a means-objective relation with the main function (here *sto31*); the state of the main function is subject of control. The control function is connected to the flow-structure via an actuation-relation, *ac51*, targeting *sou29*. In [14, 15] it is shown how this modelling of control can be applied to power systems.

Modelling with MFM provides large perspectives especially for the modelling of controlled processes. MFM facilitates the definition of the roles that a control system may take with respect to a process as well as the different types of requirements that need to be formulated for a process.

Reasoning about control functions in MFM is not presented here, even though it is both depending on and interacting with reasoning about causes and consequences. It is dependent because a control action for remediation of a disturbance must be decided on the basis of the causes and possible consequences of the disturbance and the potential consequence paths which can be used for control. Reasoning about control is interacting with reasoning about causes and consequences because MFM models include control functions whose states are dependent on the states of goals, flow functions and other control functions.

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MODELLING AND ASSESSMENT OF THE PWR-SUPER-RAMP EXTENSION EXPERIMENT

The PWR Super-Ramp Extension experiment was conducted to explain the issues left opened in the PWR-Super-Ramp Project related to the cladding failure mechanism by pellet cladding interaction during ramp tests. This last project is part of the OECD/NEA International Fuel Performance Experiments database. It provides the experimental data of 28 PWR fuel rods irradiated at medium-high burn-up and then power ramped in the R-2 Research reactor (Studsvik, Sweden). Its objective was to establish the failure-safe operating limits of representative PWR fuel rods when subjected to power ramps. The burn-up ranges from 28 to 45 MWd/kgU. At the end of these experiments, there was an unexpected resistance of 10 fuel rods of standard KWU design that survived all the tests (up to a ramp terminal level of 49 kW/m).

The Super-Ramp Extension power ramped 5 additional KWU rods of same design irradiated up to 30-35 MWd/kgU with the objective to understand the anomalous behavior of these kind of rods observed in the PWR-Super-Ramp Project. The Super-Ramp Extension has never been released by Studsvik. Nevertheless, on the basis of a paper available in the literature, the main boundary conditions of the experiment have been reconstructed. The methodology adopted will be presented in the current paper. The aim of the activity is to compare, investigate and summarize the main outcomes achieved after the simulations of 5 PWR KWU standard rods by TRANSURANUS code. The connection with the PWR Super-Ramp Project is also discussed (10 KWU rods).

Introduction

During the normal operation of a Light Water Reactor (LWR), the fuel–cladding gap may close, as a result of several phenomena and processes. In this equilibrium state, a significant increase of local power (like a transient power ramp in the order of 100 kW/m-h), induces circumferential stresses in the cladding. In presence of corrosive fission products (i.e. iodine) and beyond specific stress threshold depending on the material, cracks typical of stress corrosion may appear and grow-up. These cracks may spread out from the cladding internal surface, causing the fuel failure. The relevance of PCI in nuclear technology is connected with the prevention of fuel failures due to stress corrosion cracking (SCC), involving the lost of integrity of the first and the second barriers.

The present activity is focused on the behavior of the fuel component. The objective is the assessment of TRANSURANUS [1] code in predicting fuel and cladding behavior under pellet cladding interaction using two experimental databases based on PWR rods at burn-up ranging from 28 to 45 MWd/kgU. The PWR Super-Ramp Extension (SRX) [2] experiment has been conducted to explain the unexpected resistance of 10 fuel rods of standard KWU design ramp tested in the PWR-Super-Ramp (SR) Project [3]. This last database is part of the International Fuel Performance Experiments (IFPE) [4] and of the IAEA-CRP FUMEX-III [5].

The PWR-Super-Ramp and the PWR-Super-Ramp extension

The PWR-SR investigated 28 individual PWR test fuel rods subjected power ramps. Kraftwert Union AG/Combustion Engineering (KWU/CE) provided 19 fuel rods. Among them, 10 were of standard type and formed two groups: PK1 and PK2. They were base irradiated in the Obrigheim power reactor up to a burn-up range of 35-45 MWd/kgU and then they were ramp tested in the R2 Research reactor (Studsvik). The ramps consisted of:

- Conditioning, with a rather slow increase of linear heat rating from an initial value to 25 kW/m (conditioning level) and 24 hours holding time at this value. The coolant inlet temperature was kept to 314 °C.
- Power ramp at a constant rate in the range 150-600 kW/(m*h) to a pre-selected ramp terminal level (RTL) of 41-49 kW/m.
- Holding at ramp terminal level held for about 12 hours or until failure.

PK1 and PK2 fuel rods survived the tests. A small incipient (non-penetrating) crack crack was found in rod Pk2-s that was ramped up to 44 kW/m using an inlet coolant water temperature 50°C below the normal one.

Table 1: PWR-SR and PWR-SRX, summary of the main data.

Group	Pellet OD [mm]	Clad ID [mm]	Clad wall [mm]	Rod label	Measured Burn-up [MWd/kgU]	Inlet Cool. T [°C]	Ramp rate [kW/m-h]	RTL [kW/m]
PK1 (PWR-SR)	9.110	9.310	0.73	Pk1-4	33.1	314	570	47.5
				Pk1-3	35.2	314	510	47.5
				Pk1-2	35.6	314	480	44.0
				Pk1-1	35.4	314	540	41.5
				Pk1-s	34.4	314	360	42.0
PK2 (PWR-SR)	9.138	9.280	0.74	Pk2-4	41.4	314	510	44.0
				Pk2-3	44.6	314	510	49.0
				Pk2-2	45.1	314	570	46.0
				Pk2-1	45.2	314	510	41.0
				Pk2-s	43.4	**264**	510	44.0
PK1X (PWR-SRX)	9.110	9.310	0.73	Pk1x-4	34,3	--	Not ramped	--
				Pk1x-3	35.6	**265**	600	43.5
				Pk1x-2	34.8	314	600	42.0
				Pk1x-1	33.0	**265**	600	41.0
				Pk1x-s	30.5	**265**	600	46.0

Additional design data

Parameter	PK1-PK1X	PK2
Pellet enrichment / O/M ratio	3.2 % / 2.00	3.21 % / 2.00
Pellet density / average grain	10.360 g/cm^3 / 6.0 µm	10.340 g/cm^3 / 5.5 µm
UO_2 active length	[309.8 – 313.6] mm	[317.2-319.0] mm

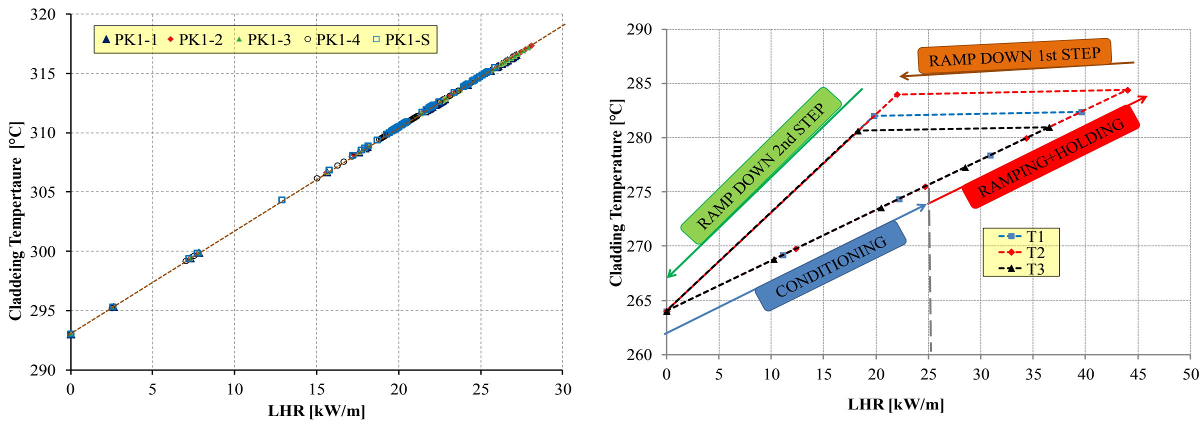
The PWR-SRX investigated 5 KWU (group PK1X) rods of same design than PK1 subjected to base irradiation in the Obrigheim power reactor up to 35 MWd/kgU. At the end of the irradiation, four out of five rods were ramp tested in the Studsvik R2 reactor using the same technique adopted in the PWR-SR. Three rods were tested at 265°C (inlet coolant temperature) the remaining one (Pk1x-2) was tested at 314°C. The ramp test was conducted to explain the unexpected resistance of PK1 and PK2 groups on the basis of the behavior of rod Pk2-s. Rod Pk1x/s experienced PCI/SCC failure at 46 kW/m. The cladding temperature was low, and the ramp terminal level was the highest one for the PK1X group. The main data of these tests are summarized in Table 1.

Modelling of PWR-super-ramp extension

Should be pointed out that PWR-SRX was subjected to the same irradiation cycles than PK1 (see Ref. [2] pp 133). The modeling of PWR-SRX required the development of appropriate boundary conditions. In particular, linear power and cladding temperature histories have been reconstructed as function of time [6].

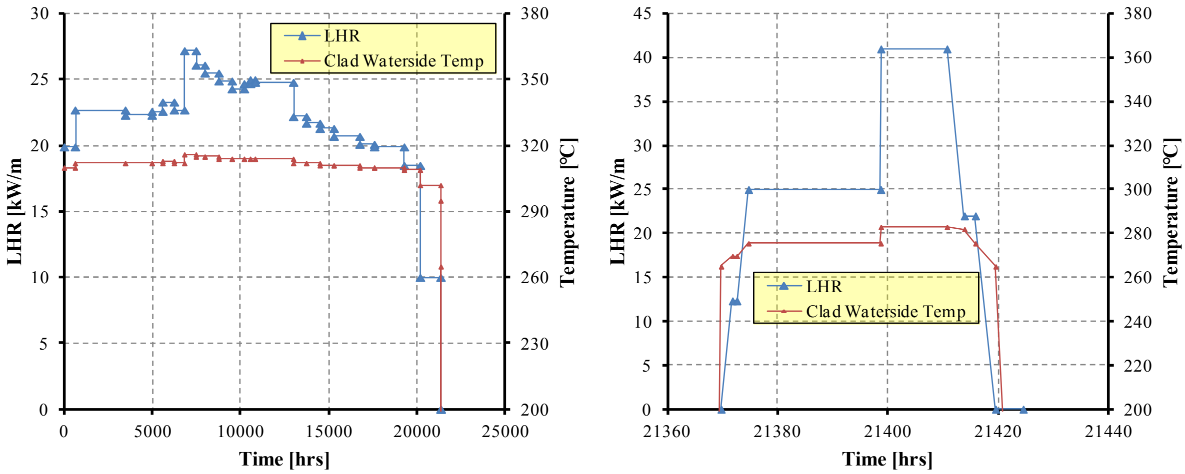
- Linear power histories in the base irradiation have been taken from figures reported in Ref. [2] by means of Tech-dig program. Each figure provides the rod linear power evolution at one node (zero power period removed) as function of burn-up.
- The conversion from burn-up to time dependent linear power has been obtained by iteration using the well validated TRANSURANUS-LWR burn-up model [7]. According to the PWR-SR, the data are treated as histograms in which the power change between two constant values is assumed 6 kW/m-h. Three axial nodes have been modeled each of them displaying identical power as in the ASCII files of PK1 and PK2 [3][4]. This artificial is necessary to model the final ramp.
- The cladding temperature in base irradiation has been obtained as function of linear power (three identical nodes). The function was developed from the PK1 ASCII files by means of simple interpolation techniques, Figure 1-a.
- The power ramps have been modeled based on Table 1. Three axial nodes have been developed (according to the ASCII files of PK1 and PK2). The linear power axial form factor is given in Ref. [2].
- The ramp cladding temperature have been obtained based on interpolation from the ASCII file of rod Pk2-s, Figure 1-b. This rod was tested at the same inlet coolant temperature than PWR-SRX.

The final boundary conditions are reported in Figure 2 for the sample rod Pk1x-1. The modeling of PWR-SR has been documented in Ref. [8].



a) Base irradiation *b) Power ramp*

Figure 1: Correlations adopted to reconstruct the cladding waterside temperature histories.



a) Base irradiation *b) Power ramp, peak axial position*

Figure 2: Rod #Pk1x-1, linear power and cladding waterside temperature histories.

Assessment of pk1x vs. Pk1 and pk2 by transuranus code

Fission gas release analysis

The FGR was measured at the end of the ramp test with exception of rod Pk1x-4 (not ramped) by means of the puncturing technique (uncertainty ±8%) [2][3]. The assessment of FGR requires the selection of two models to simulate the intra-granular and the grain boundary gas behavior. The reference simulation models the diffusion coefficient inside the grains according to Matzke (exponential function of local fuel temperature). The grain boundary processes agree with the Koo model [9]. It is specific for power ramps, it assumes that the gas at grain boundaries is released if the power excursion is greater than 3.5 kW/m and the local temperature overpass a burn-up dependent limit (constant saturation concentration is applied otherwise). Both the models are recommended for power ramps [1]. Two intra-granular and three grain boundaries additional models are available in the code. Among them, the grain boundaries model labeled as CASE B is reported in Figure 3. It assumes a saturation concentration proportional to the local temperature ($1*10^{-4}$ / T). Rod Pk1x-4 gives indication of the FGR during base irradiation of PK1 and PK1X groups, Table 2 column 1a,b. The reference simulation over predicts PK1X group. This is in agreement with the trend of PK1 group [10]. PK2 are generally under predicted (excepts Pk2-s). This could be related to swelling over prediction of PK2 (that fail in the simulations, Table 2 columns 5a,b). CASE B better represents the behavior of PK1-PK1X (both base irradiation and ramp).

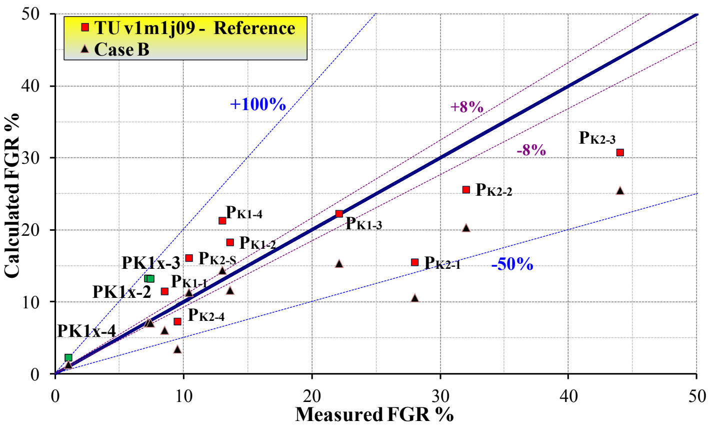


Figure 3: FGR analysis.

Grain size analysis

The grain dimension is subjected to increase with the fuel temperature. In particular, the recrystallization temperature represents the limit between equal-axial grain increase and columnar grain increase. This last phenomenon produces large deformation within the pellet centre. The assessment of this parameter is connected with the simulation of the fuel centerline temperature during the ramping phase [11]. Several measurements have been acquired after the power ramping in the peak axial position, at the pellet centre. Equal-axial grain increase was observed in the experiment with the exception of rod Pk1-2 that experiences columnar grain growth. The simulations (Figure 4) are in good agreement with the experimental value. The grain increase is in the range [1.2 – 2.5] times with respect to the initial average value. Rod Pk1-2 is an exception, it is under predicted even if columnar grain development is visible in the simulations (it increases of about 4 time its initial value).

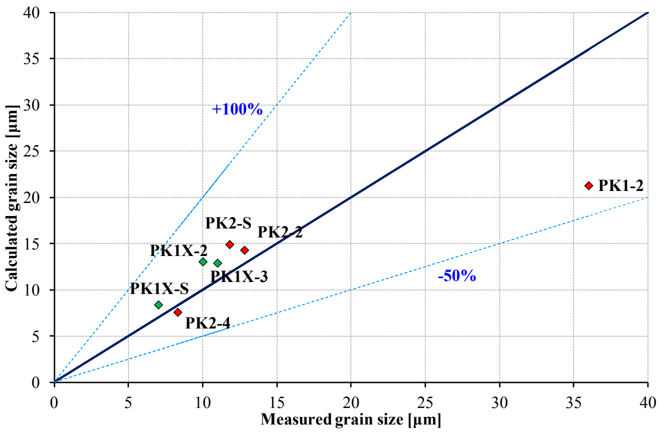


Figure 4: Grain size analysis.

Dimensional analysis

The average cladding diameter increase is derived from several diameter measurements performed prior to ramping (PTR) and after ramp (AR), at ambient conditions (20 °C). Two series of measurements are available: between ridges and at ridges. The first set of data (i.e. diameter increase between ridges) are considered for the comparison with the code results. Diameter increase at ridges is considered in the next section. The simulations are under predicted, Figure 5. This trend is correlated with the limits of the code in the geometrical modeling. First, the schematization is one-dimensional, plane, axisymmetric, and characterized by plain strain condition. Second, local variations of the geometry as ridges cannot be modeled. In addition, the diameter change is calculated based on 1 position (the peak), whereas in the experiments the measures are taken as mean over 9 axial positions located at the power peak.

The radial gap was measured AR during metallographic examinations. The values are expressed as range. Similarly, the maximum and minimum gap predicted at the end of the experiment (at 20°C) is reported in Figure 6. This analysis allows to assess the relocation models and to improve the simulations. In fact, TU code incorporates 6 relocation models and, among them, both the Modified KWU-LWR and the Modified FRAPCON-3 (assumed as reference in this analysis) models are recommended [1]. The Modified KWU-LWR is applied to Pk1-2, Pk1x-2 and Pk1x-3 (it better fits the experimental trends). In general, the gap size of the rods that remain intact in the experiment and are predicted failed (see Table 2 columns 5a,b), is under-predicted (Pk2-2, Pk2-4 and Pk2-s). Opposite considerations apply to rod Pk1x-s. The remaining rods are in agreement with the experimental trend.

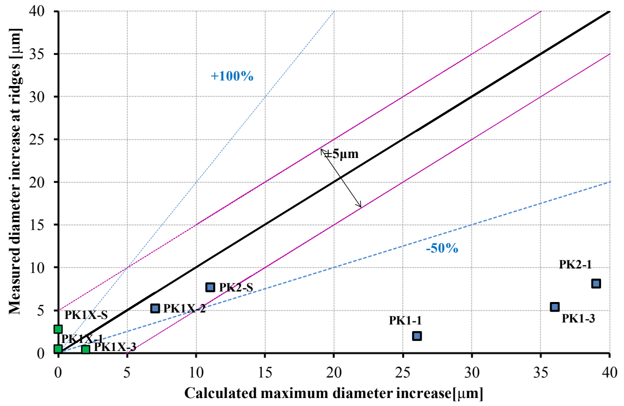


Figure 5: Diameter expansion between ridges.

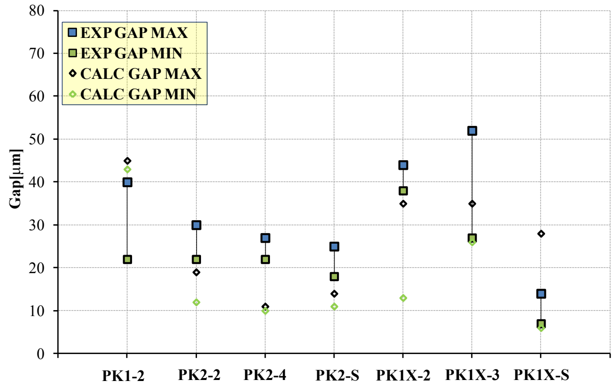


Figure 6: Gap size after ramping.

Analysis of the failure mechanisms

The summary of the results is reported in Table 2. The table includes all the parameters presented in the paper and the status of integrity of the cladding at the end of the experiments. The KWU rods tested in the PWR-SR at a coolant inlet temperature of 314°C survived the tests up to a RTL of 49 kW/m (see also Table 1). Should be mentioned that rod Pk2-s was tested at 265°C and revealed non penetrating SCC cracks (it is considered as failed). PWR-SRX revealed the existence of a failure limit at 46 kW/m (rod Pk1x-s). The coolant inlet temperature was 265°C. A possible explanation for this experimental trend may be connected to a smaller creep rate in the cladding at the lower temperature. This may result in a more severe stress situation, increasing the failure probability [2]. Furthermore, SCC phenomenon could be more aggressive at low temperatures.

The PWR-SR is conservatively predicted (PK2 rods are predicted failed) [8], including the high burn-up rod Pk2-s (44 MWd/kgU, tested at 265°C). None of the medium burn-up PWR-SRX rods are predicted failed. TU code models the cladding failure due to PCI/SCC according to Mattas [1]. It does not include directly the influence of iodine in the crack chemical grow-up. In particular, it is treated just as thresholds (i.e. after 5 MWd/kg the model is activated since it is assumed that iodine is enough to cause SCC). The chemical crack growth occurs when the cladding hoop stress is positive (PCI conditions) being the crack length increase calculated on the basis of a function that depends on cladding temperature, cladding stress and fitting constants.

Failure correlation

The cladding permanent strain represents one of the fuel criteria to limit PCMI. In particular, it is required that the maximum circumferential elastic and plastic strain does not exceed 1%, and maximum permanent axial and tangential strain caused by fuel swelling at end of fuel life remains lower than 2.5% [12]. However, the chemical attach may jeopardize the cladding integrity because the SCC phenomenon occurs also at very low strains [13]. The last process is usually connected with the power increase and the power ramp features [14].

In the case of PWR-SR and PWR-SRX, the diameter expansion at ridges gives fruitful indication about the minimum deformation capable to cause the cladding failure, Figure 7. The experiments evidences two different trends (that confirms the partial adequacy of the PCMI limit). The rods tested at 314 °C remained intact and highlighted the largest deformation. The rods tested at 265 °C highlighted low deformation and the existence of a failure limit at 14 μm (Pk1x-s). Pk2-s and Pk1x-2 fit these correlation too. TU code highlights the existence of a minimum deformation causing failure of about 8 μm, Pk2-1 and Pk2-s. It is not dependent upon the coolant inlet temperature.

Table 2: PWR-SR and PWR-SRX, summary of the analysis.

0	1a	1b		2a	2b	3a		3b	4a	4b	5a	5b
Rod label	Exp FGR [%]	Calc. FGR [%]		Exp. Grain [μm]	Calc. Grain [μm]	Exp.AR-PTR [μm]		Calc.AR-PTR [μm]	Exp. Gap [μm]	Calc. Gap [μm]	Exp F/NF	Calc F/NF
		Ref	Case B									
Pk1-4	13.0	21.3	14.4	--	--	--	--	--	--		*NF*	NF
Pk1-3	22.1	22.2	15.4	--	--	36	68	5.4	--		*NF*	NF
Pk1-2	13.6	18.3	11.7	36.0	21.3	--	--	--	22-40	43-45	*NF*	NF
Pk1-1	8.5	11.5	6.1	--	--	26	28	2.0	--		*NF*	NF
Pk1-s	--	--	--	--	--	--	--	--	--		*NF*	NF
Pk2-4	9.5	7.3	3.5	8.3	7.7	--	--	--	22-27	10-11	*NF*	F
Pk2-3	44.0	30.8	25.6	--	--	107	138	27.0	--		*NF*	F
Pk2-2	32.0	25.6	20.4	12.8	14.3	--	--	--	22-30	12-19	*NF*	F
Pk2-1	28.0	15.6	10.6	--	--	39	71	8.2	--		*NF*	F
Pk2-s	10.4	16.1	11.4	11.8	14.9	11	36	7.7	18-25	11-14	*F*	F
Pk1x-4	1.0	2.3	1.3	--	--	--	--	--	--		--	--
Pk1x-3	7.4	13.2	7.1	11.0	12.8	0	10	2.8	27-52	13-35	*NF*	NF
Pk1x-2	7.2	13.3	7.2	10.0	13.0	7	30	5.2	38-44	26-35	*NF*	NF
Pk1x-1	--	--	--	--	--	2	13	0.4	--		*NF*	NF
Pk1x-s	--	--	--	7.0	8.3	0	14	0.5	7-14	6-28	*F*	NF

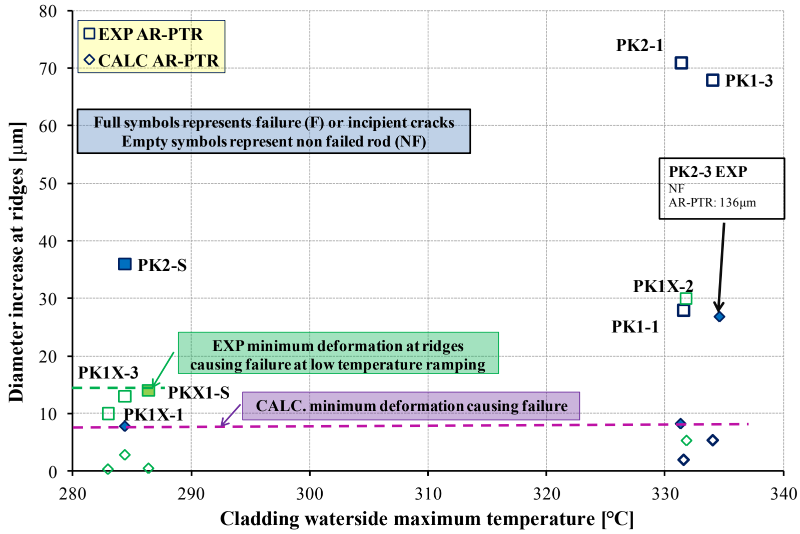


Figure 7: Minimum diameter expansion causing failure as function of ramp temperature.

Conclusions

The paper discusses the analysis of 10 KWU rods power ramped in the PWR-SR Project and 5 KWU rods of same design power ramped the PWR-SRX. The burn-up ranges between 30-45 MWd/kgU. The aim of the activity is the assessment of TRANSURANUS code in predicting fuel and cladding behavior under pellet cladding interaction. In order to model the PWR-SRX, suitable boundary conditions have been developed according to PWR-SR (PK1 and PK2 groups). The analyses bring to the following conclusions:

- FGR: PK1X and PK1 rods are over predicted. The Koo model seems to overestimate the contribute to FGR due to base irradiation. PK2 are underestimated, this could be connected with swelling over-estimation (since they are predicted failed).
- Grain size: the simulations are in good agreement with the experimental trends. Columnar grain development was observed only in rod Pk1-2.
- Dimensional changes: the cladding diameter expansion caused by the ramps is under predicted. This is connected with code limits and the low number of axial nodes. The gap size of the rods that remain intact in the experiment and are predicted failed is under-predicted (Pk2-2, Pk2-4 and Pk2-s). Opposite considerations apply to rod Pk1x-s.
- Cladding integrity: the KWU rods tested in the PWR-SR at a coolant inlet temperature of 314°C survived the tests up to a RTL of 49 kW/m. Rod Pk2-s was tested at 265°C and revealed non penetrating SCC cracks. PWR-SRX revealed the existence of a failure limit at 46 kW/m (rod Pk1x-s). The coolant inlet temperature was 265°C. The code predicts correctly 6 out of 10 rod from PWR-SR, the errors are conservative. Four out five rods from PWR-SRX are correctly simulated, the error is not conservative.

In conclusion, the PCI/SCC phenomenon is conservatively simulated for ramp test performed at high coolant inlet temperature and for the high burn-up rod Pk-2s ramped at low coolant inlet temperature. The medium burn-up rod Pk1x-s is predicted intact, this may indicate a lack in the failure subroutine in treating the impact of low coolant temperature during power ramps. Should be mentioned that only 5 rods are tested at low coolant inlet temperature and additional data are necessary to point out more accurate conclusions.

List of symbols

AR After Ramp
CRP FUMEX-III Coordinate Research Project Fuel Modeling at eXtended burn-up, III
ENEA Agenzia nazionale per le nuove tecnologie, l'energia, e lo sviluppo economico sostenibile
FGR Fission Gas Release
IAEA- International Atomic Energy Agency
ID Inner Diameter
IFPE International Fuel Performance Experiments database
KWU/CE Kraftwert Union / Combustion Engineering
LWR Light Water Reactor
OD Outer Diameter
OECD/NEA Organization for Econ. Co-operation and Development / Nuclear Energy Agency
PCI/SCC Pellet Cladding Interaction / Stress Corrosion Cracking
PCMI Pellet Cladding Mechanical Interaction
PTR Prior To Ramping
PWR Pressurized Water Reactor
PWR-SR PWR Super-Ramp project
PWR-SRX PWR Super-Ramp eXtension project
RTL Ramp Terminal Level
TU TRANSURANUS code
UNIPI UNIversity of PIsa

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ALL. 1: RAPPORTO TEST SU COMBUSTIBILI E MATERIALI AVANZATI PRESSO HRP

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

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Luglio 2012

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

CIRTEN

Consorzio Interuniversitario per la Ricerca TEcnologica Nucleare

UNIVERSITY OF PISA

S. PIERO A GRADO NUCLEAR RESEARCH GROUP

Rapporto Test su Combustibili e Materiali Avanzati presso HRP

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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"

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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.

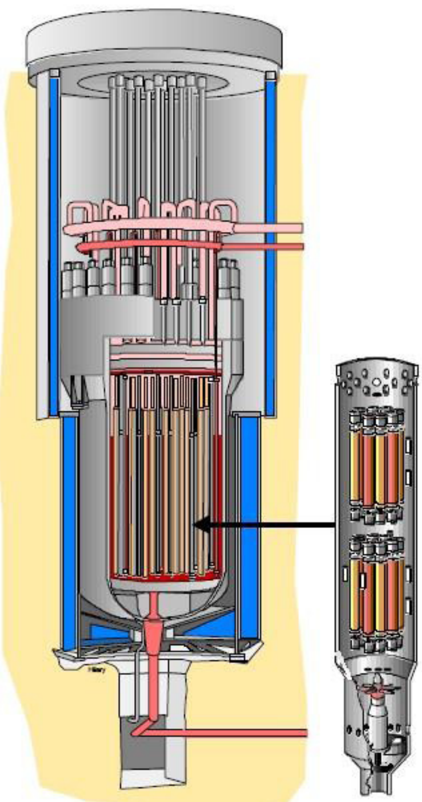

Figure 1 – Simplified flow sheet of the reactor system

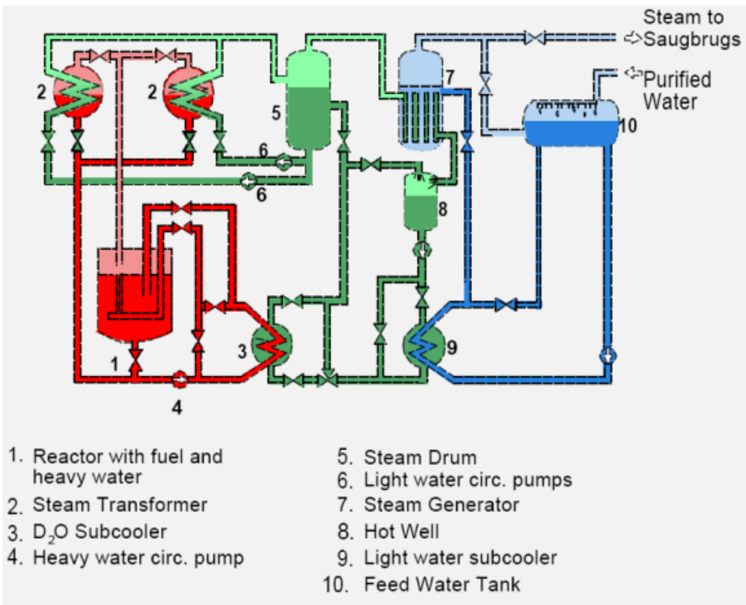

Figure 2 – Simplified scheme of the reactor

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.

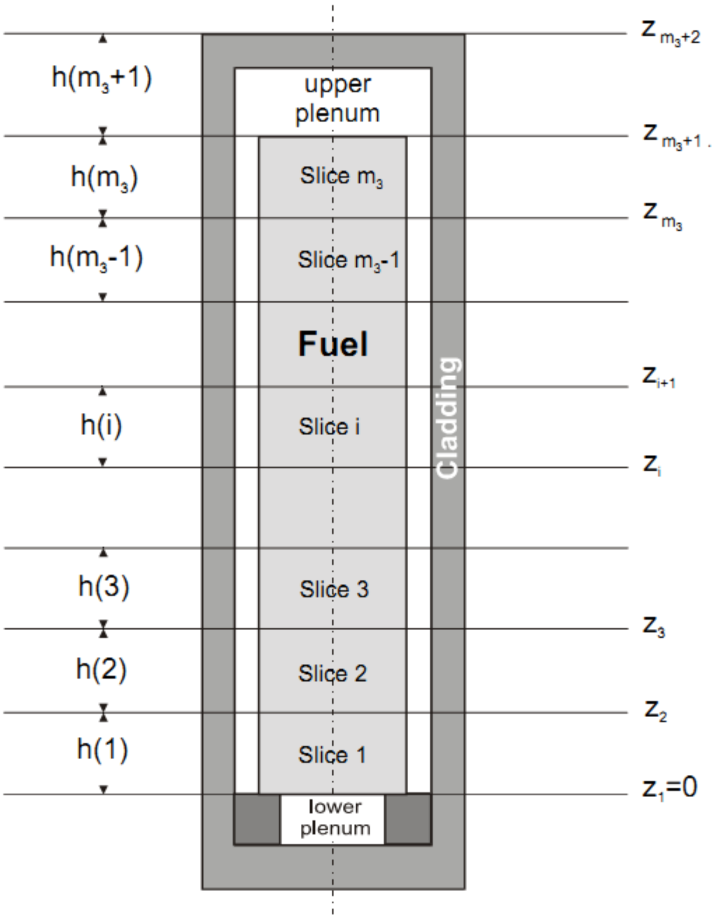

Figure 3 – Axial discretization of the fuel rod

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.

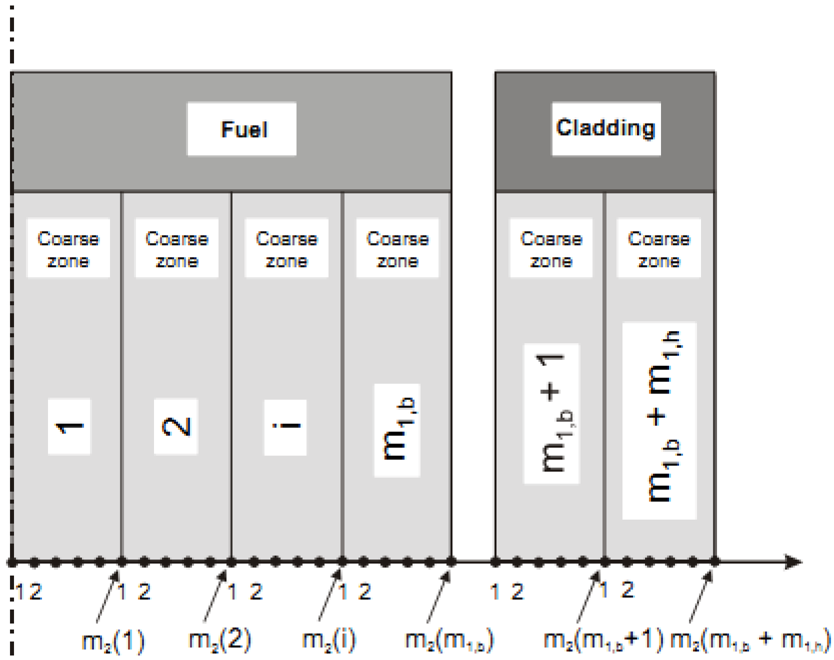

Figure 4 – Radial discretization of the fuel rod

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.

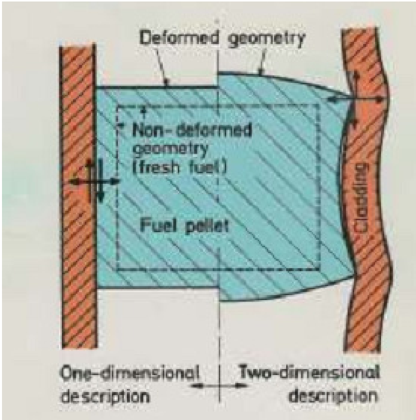

Figure 5 – Schematic view of a deformed fuel pellet; comparison between a one-dimensional and a two-dimensional description

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.

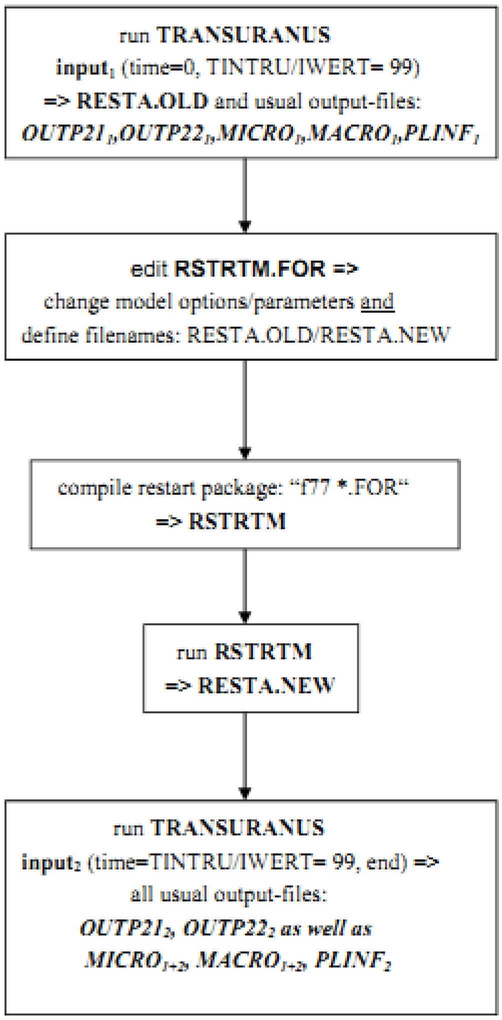

Figure 7 – Restart option flowchart

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.

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