STR on Pellet-Cladding Interactions (PCI and PCMI): Progress in the Nuclear Industry

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ACRONYMS AND EXPLANATIONS

1 Introduction

Pellet-Cladding Interaction (PCI) and Pellet-Cladding Mechanical Interaction (PCMI) remains an important phenomenon in the modern nuclear reactor fuel engineering. This topic was addressed in detail in ZIRAT-11 Special topical report on PCI issued in 2006 [Adamson et al, 2006]. Since then, a considerable number of developments took place in this field, such as changes in fuel design, initiated by the fuel vendors in collaboration with utilities, which promoted introduction of new cladding and fuel pellet materials to minimize the risks of the fuel damage due to these phenomena. Considerable research efforts were put forward in order to expand the experimental knowledge and develop the simulation tools to model and predict the PCI and PCMI.

In addition to this, fuel vendors and power utilities have developed methodologies and recommendations on reactor operation to counter the risks of PCI and PCMI which address reactor start up, transients and flexible power operation.

This report addresses the new developments in the following areas:

- Laboratory testing techniques for PCI/PCMI properties of fuel cladding
- Research of PCI and PCMI,
- Advance in PCI modelling,
- PCI/PCMI in ATF fuels,
- Vendor methodology developments

The information within this report is based on the review of open-source literature available mainly through journals, conference proceedings and publications. The intention of this report is to provide an overview of the current status of development. Much more detailed description and analysis is available in the publications referred to in this report.

Several technical meetings sponsored by IAEA and NEA took place over time period of 2006-2022 which addressed the PCI and PCMI phenomena and which established the current state-of the-art within this area. The most important meetings providing a comprehensive overview of the current status, which are also referred to in this report, were:

- Workshop on Pellet-Clad Interaction (PCI) in Water Cooled Reactors, NEA Working Group on Fuel Safety (WGFS), Lucca, Italy 22-24 June 2016 [NEA, 2018]
- Technical Meeting on Progress on Pellet–Cladding Interaction and Stress Corrosion Cracking, Experimentation, Modelling and Methodologies Applied to Support the Flexible Operation of the Nuclear Power Plants, (IAEA) Aix-en-Provence, France, 8–11 October 2019 [IAEA, 2021]

The extensive, continuous flow of conference proceedings and journal publications is being monitored by several literature searches of world-wide publications and the important papers are summarised and critically evaluated. This includes the following conference and journals:

- LWR Fuel Performance/Top Fuel/WRFPM conference,
- GLOBAL conferences,
- ICONE conferences,
- Zirconium in the Nuclear Industry: International Symposium
- Transactions of Korean Nuclear Society,
- Journal of Nuclear Materials,
- Journal of Nuclear Science and Technology,
- Nuclear Engineering and Design,
- Kerntechnik,
- Metallurgical and Materials Transactions,
- Journal of Alloys and Compounds,
- Journal of Nuclear Engineering and Radiation Science,
- Etc.

2 PCI and PCMI

Fuel failure induced by pellet–cladding interaction (PCI) or pellet-cladding mechanical interaction (PCMI) in water-cooled reactors has been considered as a re-appearing issue since it was first observed in the 1960s. Over the years, a vast amount of contributions has been made to the understanding the PCI and PCMI of Zircaloy cladding. With significant efforts to address this issue, for example, through mechanistic understanding of iodine-induced stress corrosion cracking (I-SCC) phenomena, and fuel design improvements and operational restrictions implementation, the PCI-induced fuel failure has become an infrequent event. These earlier efforts are described in detail in the ZIRAT-11 report [Adamson et al, 2006].

PCI failures may occur in both boiling water reactors (BWRs) and pressurized water reactors (PWRs). The failure mechanism is much more prevalent in BWRs since reactor operation to some extent is done by control rods movements. In PWRs, reactor power is normally not controlled by insertion and extraction of the control rods in the core. Instead, reactor power is controlled by the boron concentration that is continuously decreased during operation to compensate for the decrease in reactivity. This type of reactor power control is much smoother than in the BWR case and consequently PCI failures are less common in PWRs. However, during reactor power increases, and specifically during a class II transient (anticipated operational occurrence, AOO), PCI failures may occur in a PWR.

PCI failures were rather common in the 1970s and 1980s and due to this the fuel vendors revised fuel pellet and fuel rod design and reactor operation to mitigate this failure mechanism. However, in the early 2000s, several duty-related failures occurred in high-energy US PWRs during reactor startup and restarts. Hot cell examination of several of the failures confirmed the presence of fuel pellet missing pellet surface (MPS). While the affected fuel vendor revised manufacturing and inspection to minimize the occurrence of large MPS features, affected utilities implemented one or more of the following remedial actions [Kennard et al, 2022]:

- Reduced power ramp rates on startup,
- Added constant power holds to allow cladding stress relaxation as the reactor approached full power conditions,
- Implemented core design changes to reduce local power peaking,
- Performed cycle-specific startup analyses to explicitly evaluate margin to PCI failure in limiting rods, and
- Limited extent and/or duration of coast downs.

These remedies have proven effective in limiting the occurrence of PCI-type failures in the affected US plants.

Since 2010's, the PCI/SCC has again become a concern for the nuclear fuel community, since, due to the changes in the power generation mix, the nuclear power plants in some countries have to implement flexible power operation to compensate the frequent variations in the power grid demand. The flexible power operation is known to reduce PCI/SCC design margins of the fuel both during normal operation and anticipated operational occurrences (AOOs) [Sim et al, 2019]. In addition to this, the industry is moving towards the higher burnup and higher energy core designs for longer operating cycles, and towards higher operating efficiencies. These and other changes have the potential to reduce margin to PCI-type failures.

PCMI is the condition of an increased loading at pellet/cladding gap that creates a stress state in the cladding, which, in the presence of pellet cracks incident to the cladding, causes stress concentrations. The local chemical environment created by the presence of released fission products promotes cladding failure by inter-granular stress corrosion cracking (SCC) at the point of stress concentration. The term 'classic PCI' refers to this particular condition of failure by SCC.

The next subsections will shortly re-capture the mechanisms of both PCI and PCMI, with more detailed description available in the earlier ZIRAT report [Adamson et al, 2006].

3 PCI/PCMI Experimental Investigations

The rod failures in ramp tests involve a rather complex process of pellet-clad interaction (PCI) and pellet-clad mechanical interaction (PCMI). These processes include chemical, mechanical, thermodynamical, and radiological interactions. The failures during the majority of ramp tests can be viewed as a result of three different failure mechanisms: stress corrosion cracking (SCC), hydriding embrittlement strain failure (HESF) and delayed hydride cracking (DHC).

3.1 PCI test methods

3.1.1 In-reactor ramp tests

In-reactor testing has the best chance to reproduce the conditions present in PCI failures of commercial fuel. For sorting out PCI remedies and mechanisms, the use of test reactors has been useful in the past. Today, there are very few test reactors available for materials testing, and for change-of-power (ramp) tests outside of Russia. Ramp testing, and subsequent analyses are complicated and costly, but can provide essential input into PCI understanding.

Many types of ramp tests have been performed [Davies et al, 1984, 1977; Dahlbäck et al, 2005; Seibold et al, 1997; Cox, 1990; Massih et al, 2005] but the most common and most easily described is the "staircase" ramp illustrated in (Figure 3-1). Rods for the tests are irradiated in a power reactor at low power to burnups sufficient to generate ample fission products. Rodlets for ramp testing are necessarily short, so it is most convenient to irradiate segmented rods on the order of a meter long; if full length rods are used, they must be refabricated in a hot cell to lengths between a half to one meter. Great care must be taken to prevent contamination of the rod with air, oils, etc. and to prevent pellet fragmentation and relocation during refabrication. In the test reactor, irradiation is conducted for a short time (6 hours in Figure 3-1) at a power similar to the pre-irradiation in the power reactor. The power is then quickly raised a small amount (about 2 kW/ft (6 kW/m) in Figure 3-1) and held for a short time (1 hour in Figure 3-1).

This sequence is repeated until either the rod fails (detected by monitoring fission products in the coolant, or by dynamic monitoring of the rod length) or a power is reached that is above any power that would be reached in commercial service. The power increments, rates and hold times can be varied to test various hypotheses or fuel conditions (burnup, design, etc.).

Many in-reactor tests have been conducted. Usually, a failure threshold is determined as a function of burnup. (Figure 3-2 and Figure 3-3) show the results where various cladding designs or materials are compared. More information is given in [Adamson et al, 2006] and [Rudling et al, 2012].

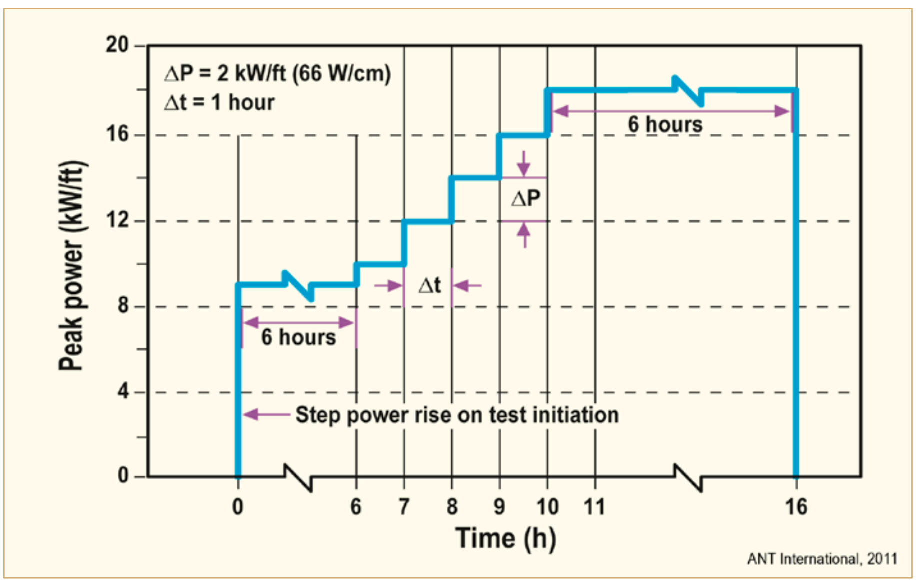

Figure 3-1: Power-ramp sequence illustrating the "staircase" ramp test, after [Davies et al, 1984].

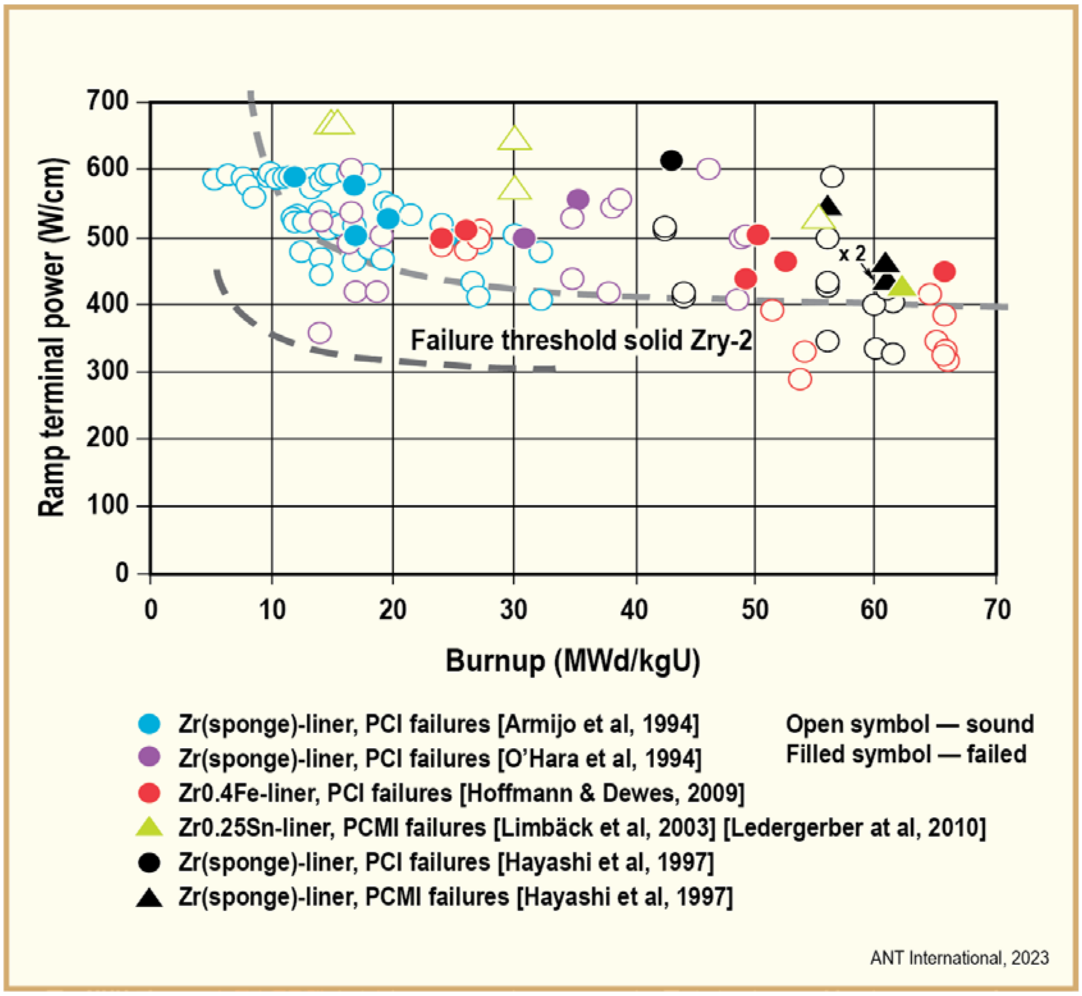

Figure 3-2: Ramp behaviour of BWR Zr liner fuel with sponge liner. The failures at burnups >50 MWd/kgU are not due to PCI but due to PCMI. All other failures at burnups <50 MWd/kgU are due to PCI, after [Adamson et al, 2006], [Hayashi et al, 2002], [Matsunaga et al, 2009].

4 PCI/PCMI Modelling and Simulation

In 2019, [van Uffelen et al, 2019] have written an overview paper on the current developments in the fuel performance modelling. This paper summarizes the historical developments for the fuel rod performance codes around the world, providing an overview of the developments rather than details of codes and equations. The paper is useful in demonstrating the current status of the modelling and simulations, including PCI and PCMI modelling.

The authors [van Uffelen et al, 2019] state that a consistent fuel rod simulation requires simultaneous consideration of several issues:

- Heat generation and transfer;
- The mechanical interaction between the fuel rod components;
- The isotopic evolution due to irradiation and the chemical interactions among fuel, fission products, cladding and coolant.

The strong links between these aspects of fuel performance required the introduction of assumptions right from the start in order to be able obtain a sufficiently accurate numerical solution of the set of equations after a reasonable computation time. The first important assumption is about the rod geometry: most fuel performance codes still consider a cylindrical geometry and adopt axi-symmetry. In such codes, the computational domain consists of a stack of cylindrical UO_2 fuel pellets and zircaloy cladding separated by a helium-filled gap as shown in (Figure 4-1). The second assumption deals with the axial temperature gradient, considered to be much smaller in comparison with the radial temperature gradient. As a result, a majority of the codes (e.g., FEMAXI, FRAPCON, START and TRANSURANUS) represent the cylindrical fuel rod in a so-called one-and-a-half dimension (1.5D). In such simulation tools, all transport processes are solved in one (radial) dimension, and the axial segments are coupled via balance equations. Hence, the solution of the transport processes is usually programmed in separate modules for each axial slice, using couplers to manage interactions between models and slices [van Uffelen et al, 2019].

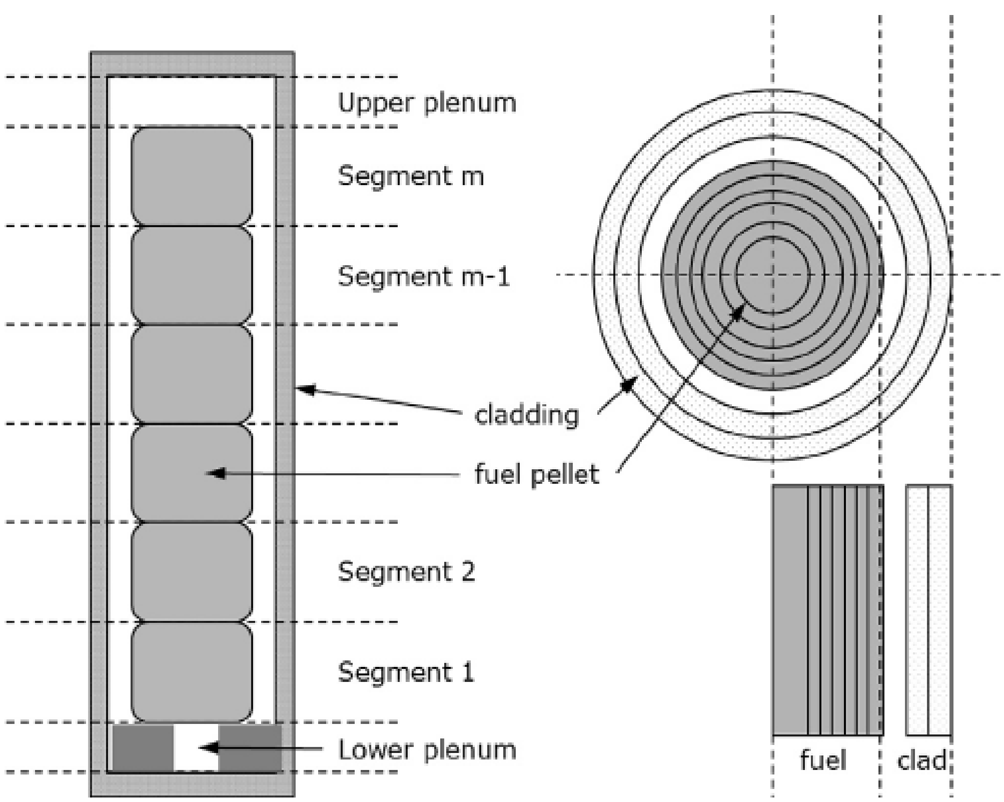

Figure 4-1: Typical schematic 1.5D representation of a rod in a fuel performance code [van Uffelen et al, 2019].

4.1 Fuel performance codes used in different countries

The important aspects of fuel behaviour are inherently multidimensional, a good example of this being pellet-clad mechanical interaction (PCMI), which can be described as a localized phenomenon. The fuel performance codes tend to incorporate models which simplify this complex behaviour using an axisymmetric, axially stacked, one-dimensional radial representation of fuel rods. Many of them use axial symmetry in the pellet-gap-cladding, dividing the rod in different height nodes to simulate different points of thermo-mechanical interaction (1.5D and 2D axisymmetric approach). Quite frequently coupled code systems are used, which incorporate FEM packages to simulate the thermo-mechanical interaction, linking them to fuel codes. The trend in the code development for the PCMI analysis under the last decades is to develop full 3D codes which could address the problem of PCMI without the use of any symmetry. This section provides a short overview of the computer codes used in different countries and the current state of the art of the codes.

A list of codes used by fuel vendors, research organizations and safety authorities in different countries is provided in (Table 3-1). There is a vast variety of fuel performance codes in use nowadays, and this section summarizes the main codes used in different countries as identified by [van Uffelen et al, 2019].

Table 4-1: List of fuel rod performance codes developed in different parts of the world for light water reactor fuel [van Uffelen et al, 2019].

Country	Organization	Code name
Argentina	CNEA	BACO, DIONISIO
Belgium	Belgonucleaire SCK-CEN Tractebel	COMETHE MACROS (ASFAD) FRAPCON, FRAPTRAN
China	Xi'an Xiaotong Univeristy CIAE NPIC CGNPC	FROBA FTPAC FUPAC JASMINE
Czech Republic	UJV	PIN-MICRO (GAPCON-THERMAL2)
France	CEA Framatome EdF IRSN	ALCYONE (METEOR-TRANSURANUS) COPERNIC (TRANSURANUS), GALILEO (COPERNIC/RODEX/CARO) CYRANO SCANAIR
Germany	Siemens Framatome GRS JRC	CARO GALILEO (COPERNIC/RODEX/CARO) TESPA-ROD (TESPA) TRANSURANUS (URANUS)
Hungary	MTA EK	FUROM (PIN-MICRO)
India	BARC PNC	FAIR, PROFESS FUDA
Japan	CRIEPI JAEA SEPC NFD	EIMUS (FEMAXI-III) FEMAXI, RANNS IRON (FEMAXI-III) TRUST
Korea	KAERI	COSMOS, INFRA
Russian Federation	VNIINM TRINITI IBRAE	START, RAPTA RTOP SFPR (MFPR)
Sweden	Westinghouse Sweden Electric	STAV
United Kingdom	NNL, EDF Energy	ENIGMA (MINIPAT, SLEUTH, HOTROD)
USA	USNRC Siemens EPRI INL Framatome Westinghouse	FRAPCON, FRAPTRAN (FRAP), FAST RODEX FALCON (FREY, ESCORE) BISON GALILEO (COPERNIC/RODEX/CARO) PAD

5 PCI/PCMI in Accident Tolerant Fuels

Accident Tolerant Fuel (ATF) is being developed to meet several goals: to make the fuel cycle more economical, to increase fuel reliability and to increase fuel operational flexibility in order to meet energy demand.

Additive fuel pellet technology has been investigated since 1976. At that time, a comprehensive program was initiated to characterize and test additive fuel, with a specific focus on conditions where pellet clad interaction (PCI) occurs. As part of this program, additive fuels were characterized, and material properties were determined for implementation into thermal-mechanical methods. Several lead test assemblies and segmented rods were inserted into power reactors in both the US and abroad. Instrumented fuel assemblies were inserted into a test reactor to develop thermal and mechanical data to support design and licensing [GNF, 2009].

The major nuclear fuel vendors of AREVA, Westinghouse, and GNF have been developing their own additive-doped large grain UO_2 pellets technology in order to improve the PCI behaviour since the early 90ies.

(Table 5-1) briefs the characteristics of developed advanced nuclear fuel pellets. The common feature between the developed pellets is using the small amounts of additives in the UO_2 pellet to increase the grain sizes and enhance the deformation behaviour. The developed pellets are planned to become a new standard for BWR fuel and is being considered for advanced PWR fuels [Yang et al, 2009].

Table 5-1: Developed large grain UO2 pellets [Yang, et al, 2009].

	Cr-doped UO2	ADOPT	Additive fuel
Vendor	Framatome	Westinghouse	GNF
Additives	Cr_2O_3	Cr_2O_3+Al_2O_3	Alumino-silicate
Contents	1600 ppm	700 ppm	Ca 2500 ppm
Grain size	40~50 μm (1500ppm)	~35 μm	above ~20 μm

The addition of a small amount of chromia has been proven to significantly enhance grain growth during the sintering process employed in fuel manufacture. This leads to larger grain sizes than standard UO_2 fuel pellets for shorter sintering times. There is also greater densification of the pellets during sintering than for standard UO_2, leading to a greater mass of uranium in the pellets and a reduced porosity volume fraction. These factors make the manufacturing process more economical, although careful control of the dopant concentrations and sintering conditions is needed to avoid volatilisation of the chromium and to ensure solubility of the chromia in the fuel matrix. Westinghouse began development of a new type of doped fuel pellet, ADOPT in 1998. They experimented with many of the additives; this resulted in the selection of chromia as their preferred choice.

It is well known from earlier work on fuel that pellets with high fuel creep rates can dramatically improve the PCI resistance. The PCI benefit has been attributed to fuel movement in the central high temperature regions in which the fuel becomes visco-plastic and under its internal stresses deforms so as to fill the as manufactured dimples. Axial sectioning after the ramp test shows the dimples are often filled to such an extent that it is no longer possible to distinguish the original pellet interfaces. Such fuel movement provides a PCI benefit since it reduces the stress which pellets transfer to the outer cladding. Rather than thermally expanding outwards, which increases both the cladding diameter and the stresses applied to it, if the fuel is mobile, it can instead deform to fill any free volume within the fuel stack. This can include dimples, central holes or even large internal pellet porosity. (Figure 5-1 and Figure 5-3) shows a schematic representation of the pellet deformation at high temperatures [Wright et al, 2016].

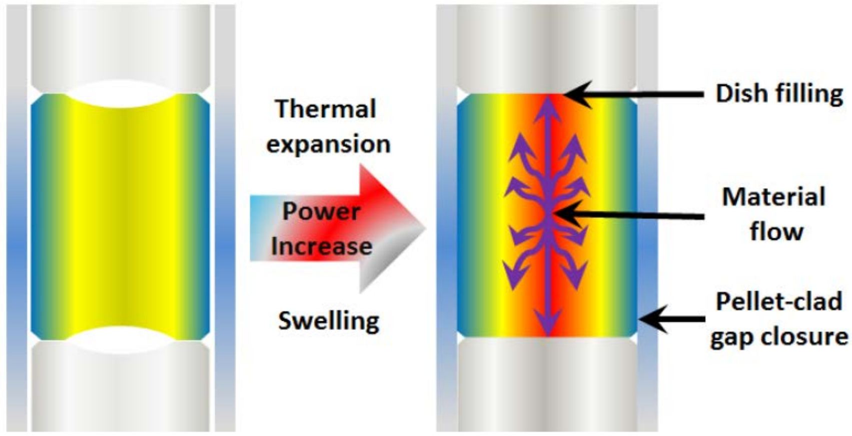

Figure 5-1: Schematic of pellet deformation at high temperature [Wright et al, 2016].

The addition of chromia also enhances the fuel thermal creep rate, without significantly affecting any of the other thermal or mechanical properties, so there is an improvement in the pellet-clad interaction (PCI) behaviour during power transients, making the cladding less susceptible to failure by stress-corrosion cracking. The PCI behaviour is further improved through increased radial cracking of the thinner brittle outer zone of the fuel pellets (noting that the fracture strength is reduced due to the large grain size), and, to some extent, by chemical effects (such as improved corrosive fission product retention, and chromia dissociation and associated generation of free oxygen which can oxidise the cladding inner surface, thereby protecting it from SCC) [NNL, 2020].

Cr_2O_3 is a neutron absorber, so to reduce this effect, alumina (Al_2O_3) was also added to the fuel; magnesia (MgO) was used as an alternative to alumina in the earlier stages of development. The alumina also enhances the grain growth induced by the chromium oxide (Figure 5-2). This synergy between the two dopants allows for a smaller chromia content (less neutron absorption), while still having the beneficial grain growth effects.

There are two main vendors of chromia doped fuel: Westinghouse and Framatome. The Westinghouse product uses a combination of chromia and alumina as dopants (ADOPT fuel), while the Framatome product uses only chromia. The chromia content of Westinghouse fuel is limited to 1000 ppm or less, which can be compared with the 1600 ppm value for Framatome's doped pellets. Both Westinghouse and Framatome have significant operating experience with their chromia doped pellets in commercial LWR. Ramp tests have also been performed to determine the PCI behaviour, and various experimental irradiations have been carried out to investigate the phenomenology. Further irradiations are planned in the near future.

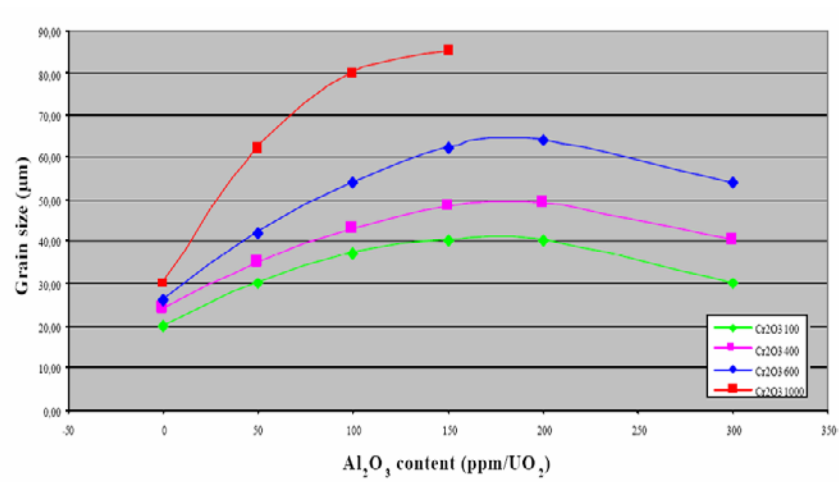

Figure 5-2: ADOPT grain size against Al_2O_3 content [NNL, 2020].

6 PCI/PCMI limits in Reactor Operation

Although PCI/SCC is not considered as a safety related issue in most countries, the fuel vendors and the operators have developed specific methodologies to quantify the actual PCI/SCC design margins to ensure sufficient operational margins preventing costly in-reactor fuel failures. The reactor operating guidelines include, for a given preconditioning power and for various operating modes, limitations

- Of the operating time at reduced power,
- Of the power ramp rates and
- Of the hold times at the ramped power.

These operating guidelines are valid for a specific plant design and a specific fuel design. The formulation of the guidelines strongly relies on the experimental data gathered over the years through numerous national and international experimental power ramp programmes in material test reactors and coupled with commercial in-reactor experience feedback [Sim et al, 2019].

The PCI/SCC failures were more frequent in boiling water reactors (BWRs) than in Pressurized Water Reactors (PWRs). This is due to the frequent use of control blades to compensate for changes in reactivity with burnup and for optimization of the core-wide power profile. Therefore, BWR operating guidelines impose limits on power ramp rate at start-up, after refuelling or after changing the control rod sequence, as well as on the control rod withdrawal rate in the high-power regions when the fuel is not preconditioned.

For the PWRs, the PCI/SCC surveillance and protection methods (similar to BWR operating guidelines) are intended to safely prevent fuel failures by allowing for adequate safety margins. An early approach developed in the 1980s, so-called RSST (power Range, Step, Speed and Time) has been successfully implemented in some German PWRs and in the N4 series of 1400 MW(e) class PWRs (developed in France). In the RSST, important model parameters are directly derivable form experimental data by using the 'RSST Approach' that in order to prevent PCI failures at least one out of four 'predictors' (i.e., power range, power step, speed of power increase, or time at transient overpower) has to be below a critical value at any operating time. An algorithm was provided for defining and monitoring an adequate 'conditioned power' as a reference power for acceptable power ramps [von Jan & Hering, 1981].

6.1 Startup ramps

Local power changes during reactor startup can introduce large local stresses when the power level increases above the conditioned power level of the cladding. Operators of pressurized water reactors (PWRs) employ some type of power ramp rate restrictions during startup following a refuelling outage. Fuel vendors imposed such operating restrictions in the late 70's to minimize the occurrence of PCI failures. These restrictions consist of a threshold power level above which a limit on the power ascension rate is imposed.

The fundamental aspect of the power conditioning is to allow relaxation in the cladding before next step of power increase. The relaxation is caused by creep in the cladding that lowers the stress in the cladding and thereby makes it more resistant to next power step.

Once conditioned to a certain level, power can be changed freely up to this value as long as the deconditioning was not allowed. Deconditioning takes place when a reactor operates under reduced power for a longer period which is usually a consequence of testing, maintenance or coastdown at the end of cycle. It decreases the cladding-pellet contact pressure which, in case of further power reduction, can result in re-opening of the hot gap between the pellet and the cladding. Deconditioning is a much slower process than conditioning (usually takes months while conditioning is measured in hours). The deconditioning rate depends on the level of power reduction and the time of operation at reduced power.

A broad spectrum of startup strategies following refuelling outages exists in currently operated pressurized water reactors (PWRs). An important factor of a plant's restart strategy is power ramp rate restrictions. The intention of power ramp rate restrictions is to prevent fuel rod damage and failure by PCI. Reactors undergoing restart are potentially vulnerable to PCI failures. Under restart conditions, the pellet/cladding gap may be closed at relatively low linear powers for fuel rods beyond a threshold burnup level. If the power level increases rapidly under these conditions, differential thermal expansion can result in stress concentrations in the cladding that may cause cladding failure.

The PWR power ramp restrictions consist of a threshold power level above which a limit on the power ascension rate is imposed. The threshold power levels vary from plant to plant depending on design and can range between 20% and 50% of core full power. Most plants use threshold power levels near 20% core power. The power ascension rate below the threshold power is plant-dependent and can vary between 10% to- 30%/hr. To limit the extent of this stress concentration, very slow ramp rates, that is, power increases of ~3% per hour above a threshold power level are prescribed to minimise the likelihood of cladding failure during reactor startup. Below the threshold value (which can be typically 30% of full power), the ramp rates are significantly less restrictive (e.g., ~15% per hour). There is additional time required in a start-up procedure due to certain procedural hold points that take place at intermediate power levels in PWRs. These procedural hold points must be adhered to during a typical restart and are summarised in (Table 6-1).

Taking into account the hold point durations and the ramp rates between the hold points (varying from ~3% of nominal power per hour to ~15% of nominal power per hour), the whole process to take the reactor back to nominal power after a refuelling shutdown can take 72 to 100 h. Start-up rates after a shutdown without refuelling can be faster than the 3% per hour, with rates as high as 10% per hour. In addition, typical manoeuvring rates during power operation can be much faster than the start-up rates described here.

Table 6-1: Summary of hold points during power increase after a shutdown [Peakman et al, 2020].

Power level %	Duration, h	Activity
10	1-2	Turbine heatup and rotation, synchronize generator
30	24	Core flux mapping, water chemistry checks
50	1-2	Verification of compliance to technical specifications
75	18-24	Core flux mapping, water chemistry checks
90	≤1	Nuclear instrument calibration
95	1	Calorimetric heat balance between primary and secondary sides prio to full power

Even though safety margins are maintained, the insertion of control rods and the response of the local xenon concentrations can lead to significant changes in local pin powers. If a utility chooses to ramp down the reactor during times of excess supply, the fuel pellets can contract. When the decision is made to return the reactor back to full power, the rate at which the power can be increased is limited by the hoop stress on the cladding. To determine the magnitude of the cladding hoop stress, a large number of the parameters has to be identified. The shape of the fuel pellets controls the volume of the gap, affecting the gap pressure, and in the case that the gap has closed, the fuel pellet's shape is restrained by the cladding. Many different factors contribute to the shape of the fuel pellet including temperature of the fuel, fission product inventory, power history and manufacturing characteristics. Temperature and irradiation are the primary contributors to fuel pellet thermal expansion during power manoeuvres. To further complicate the geometry of the fuel pellet, nonuniform thermal expansion during the initial ramp to power causes radial cracks to form within a pellet. These cracks, in addition to thermal expansion, lead to shear forces and hoop stress on the inner radius of the fuel cladding.

7 Fuel Vendor Design and Methodology Developments

Although fuel vendors incorporated improved manufacturing process to reduce the incidence of pellet defects and cladding has evolved to materials with a higher capacity of stress relaxation, there is a trend in the industry toward the use of additional experimental limits based on the behaviour of commercially irradiated rods subjected, later on, to power ramp tests. The use of this kind of limits constitutes an additional barrier to PCI failures, especially in non-base operating conditions as returns to power after long periods at reduced power, load follows or power modulations. This kind of limits is especially useful to ensure fuel integrity during flexible operation.

Various PCI design analysis methodologies were developed by fuel vendors. ENUSA proposes a methodology to evaluate the risk of fuel rod failure due to PCI mechanism throughout the development of an effective stress limit, called “PCI Technological Limit” based on power ramps to discriminate failed from non-failed rods. Framatome proposes a set of different PCI methodologies for PWR reactors to support utilities for a more flexible nuclear energy production.

7.1 Framatome

Framatome has developed two PCI methodologies for PWR with M5 cladding which allow to justify base load and flexible operation: the MIR methodology and the Allowable Power methodology. The Framatome PCI approaches is based on thermal-mechanics computations during class-II transients. The margin to rod failure is given by the comparison of the cladding Strain Energy Density to a threshold previously assessed on representative experimental power ramp tests. The methodologies differ regarding the neutronic calculations (3D kinetics for the MIR methodology vs 3D static for the Allowable Power Methodology) and the coupling between thermal-mechanics and neutronics calculations. The MIR methodology allows to justify considerable plant manoeuvrability but at a significant computational cost. With a more simplified approach, the Allowable Power methodology provides PCI justification without restraining plant manoeuvrability at a moderate computational cost [Bessiron et al, 2014].

7.1.1 MIR method (Méthod IPG Rénovée)

The MIR method was developed to adjust the reactor protection thresholds in order to guarantee the integrity of the fuel rod during any Anticipated Operational Occurrences (AOOs). The reactor protection thresholds shall assure the scram of the reactor prior to a PCI-related fuel failure.

The fuel performance codes and analysis methods were developed by EDF and Framatome in order to evaluate, for all the fuel rods of the core, the risk of clad failure by PCI during any class-II occurrence (excessive load increase, uncontrolled rod bank withdrawal at power, control rod drop, uncontrolled boron dilution, etc).

The MIR study is divided into five parts [NEA, 2018]:

- Generation of conditioning power histories,
- Simulation of the initial conditions of the PCC-2,
- Simulation of the AOO transients,
- Thermo-mechanical evaluation of PCI risk for base load operation, frequency control and load follow, ELPO de-conditioning or return to full power after ELPO re-conditioning, which allows defining PCI Operating Technical Specifications (OTSs),
- Modification of the reactor protection thresholds.

The flowchart for the MIR methodology is given in (Figure 7-1).

As the PCI phenomenon is a local issue, it is necessary to define the accurate thermal-mechanical state of each point in the core at the moment of the transient. The local thermal-mechanical state of fuel rods depends on the power histories. Thus, the evolution of the power of each fuel rod is computed from its first irradiation in the core until the time of occurrence of class-II transient, anytime during irradiation.

Since this method requires significant computational and human resources, only a generic study on a prototypical reference core reload pattern is carried out. The objectives of the work are:

- To define the protection thresholds for AOO operations,
- To limit the manoeuvrability of the plant in PCC-1 operations (ELPO, load follow, etc.) at a level that can guarantee the absence of PCI failure during a AOO event.

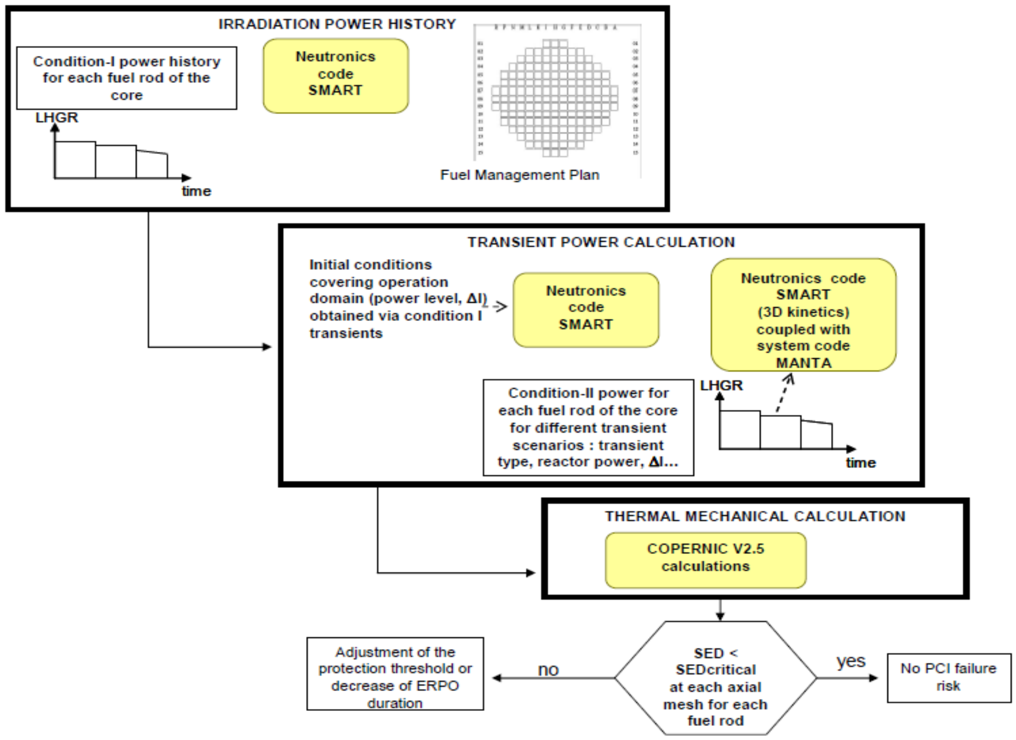

Figure 7-1: Flowchart for the MIR methodology [Bessiron et al, 2014].

The flow of the MIR analysis [NEA, 2018]:

- The rod power histories are generated for each rod starting from the introduction of fresh fuel rod into the reactor core. Those power histories are generated under varying conditions such as base load and ELPO with inserted and extracted RCCA's.
- The AOO transients are initiated from a conditioning state obtained after normal operations simulation. The initial conditions are selected such as to maximize the local power increase. For each transient, the initial conditions are characterized by the burn-up, the power level, the power axial distribution, the Xenon axial distribution, the position of the regulation groups, the Boron concentration. These assumptions can vary depending on the transient under investigation.
- The AOO transients are simulated for the whole core, including the reactor the regulation and the protection systems. The neutronic behaviour of the core is modelled kinetically. The transients are simulated starting from the initial states, at several moments in the irradiation cycle.
- The risk of cladding failure during the AOO transient is estimated by comparing the evaluation parameter calculated by the fuel performance code with the Technological Limit. The evaluation of the cladding failure risk consists in explicit thermo-mechanical calculations: the evolutions of stress or SED in the cladding are calculated according to the applied local power history. The PCI margin of a rod is defined as the difference between the maximum simulated stress (or SED) and the technological limit. Minimal PCI margins are assessed in every fuel rod for various operational states (base load operation, ELPO, return to base load operation after ELPO, frequency control and load follow).
- The reactor protection system activation threshold values are adjusted to take into account the analysis described above.

8 Summary

Multiple experimental programmes have shown that very low concentrations of aggressive species are enough to drive PCI. Chemistry effects play an important role and the PCI benefits seen in modern PCI resistant fuel, and the further developments in PCI resistance will likely not be explained through mechanical effects alone. This means that fuel-performance models must also deal with chemistry effects if they seek to make accurate predictions of PCI failure limits. In general, however, it appears that mechanical aspects are better understood than chemical aspects of PCI benefits.

Regarding the role of cladding design on the mitigation of SCC driven by PC(M)I mitigation, the following can be stated:

- Liner fuel remains of interest to fuel designers.
- Texture controlled cladding shows improved resistance to PCI failure.

Regarding the role of fuel pellet design on PCI mitigation it can be stated that:

- Available experiments and analyses on pellet additive effects do not fully explain all aspects of the potential PCI benefits.
- There is some evidence that additives trap aggressive species in the fuel.

Experimental programmes and analyses focused on identifying the relevant parameter to characterise the cladding failure risk (linear heat rate, cladding strain, cladding stress, strain energy density, etc.) still present a complex picture. LHGR appears to be the primary driver but is not sufficient to identify the outcome of failure/non-failure in sufficient degree of detail. In addition to LHGR and the increase of heat generation rate during a transient are the main focus of the experiments, even though time appears to play a role as well. Therefore, at the current state of development, despite a number of research programmes which focus on separate effects and efforts to model the existing PCI experimental database, it appears that the existing PCI models are not able to always distinguish “failure” from “non-failure” points in code predictions.

Mechanistic PCI/SCC models, through a multi-dimensional thermomechanical analysis should include the simulation of the effect of the irradiation history on nonlinear materials properties (pellet and cladding behaviours under thermal, mechanical and neutron irradiation loadings), the behaviour pellet-cladding interface, the gaseous swelling at high power and the fission gas releases during the transient, the corrosive species, the cladding stress corrosion cracking behaviour in iodine environment and the potential coupling effects of physics phenomena such as neutronics and thermal hydraulics.

There are remaining issues of in-reactor fuel licensing of doped fuel pellets, such as the effect of reduced densification and increased intra-granular gas bubble swelling of chromia doped pellets may reduce margin to cladding hoop strain and rod growth limits during normal operation and reduce margin to clad hoop strain increment limits during frequent faults, which, in turn, may increase the number of rod failures during RIA. There still is a lack of data for doped pellets on ramp test behaviour to justify proposed PCI thresholds in a statistically meaningful manner.

Flexible power operation (FPO) and related power changes may challenge the fuel rod thermomechanical behaviour in the reactor core. As such, PCI/SCC failure risk needs to be assessed by considering AOO transients in addition to flexible operation.

Despite the worldwide R&D efforts, including a large number of experimental investigations associated to refined multi-physic models developments, predicting the PCI/SCC failure threshold of a given fuel concept still remains a challenge. This is likely due to the complexity of some of the mechanisms involved in the PCI/SCC process (e.g., fuel pellet cracking pattern which determines the local stress at the pellet-clad interface).

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ACRONYMS AND EXPLANATIONS

AOO	Anticipated Operational Occurrence
ALPS	Advanced Light Water Reactor Fuel Performance and Safety Research
ATF	Accident Tolerant Fuel (Advanced Technology Fuel)
BNFL	British Nuclear Fuels plc (BNFL
BWR	Boiling Water Reactor
CASL	Consortium for Advanced Simulation of Light Water Reactors
CEA	Commissariat à l'Energie Atomique
CEB	Closed End Burst test
CIPS	CRUD-induced power shift
CNEA	"Comisión Nacional de Energía Atómica" – Atomic Energy National Commission of Argentina
CRUD	Corrosion-related unidentified deposits or Chalk River unidentified deposits
COG	Core Operating Guidelines
DHC	Delayed Hydride assisted Cracking
FPO	Flexible power operation
GTRF	Grid-to-Rod Fretting
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
I-SCC	Iodine-induced Stress Corrosion Cracking
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
HESF	Hydriding embrittlement strain failure
EBT	Equal Biaxial Tension test
EDC	Expansion-Due-to-Compression
ELPO	Extended low power operation
EPRI	Electric Power Research Institute
LFDRM	Local Fuel Duty Risk Monitor
LHGR	Linear Heat Generation Rate
LPO	Low Power Operation
LPPRA	Loading pattern PCI risk assessment
MBT	Modified Burst Test

MDA	Mitsubishi-Developed Alloy
MPS	Missing Pellet Surface
NDA	New Developed Alloy
NPP	Nuclear Power Plant
PCI	Pellet-cladding interaction
PCMI	Pellet-cladding mechanical interaction
PCIOMR	Pre-Conditioning Interim Operating Management Recommendations
PIE	Post-Irradiation Experiments
PNNL	Pacific Northwest National Laboratory
RIA	Reactivity Initiated Accident
RCCA	Rod Cluster Control Assembly
RTS	Ring Tensile Sample
SCC	Stress Corrosion Cracking
SCIP	Studsvik Cladding Integrity Project
NEA	Nuclear Energy Agency
NNL	National Nuclear Laboratory
NP	Nominal Power
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSSR	Nuclear Safety Research Reactor
MNF	Mitsubishi Nuclear Fuel
MPS	Missing Pellet Surface
PWR	Pressurized Water Reactor
TL	PCI technological limit
TS	Technical Specification
OEB	Open End Burst test
OIC	Outside-Inside Cracking
OTS	Operation Technical Specifications
VERA	Virtual Environment for Reactor Applications
WGFS	Working Group on Fuel Safety
WRFPM	Water Reactor Fuel Performance Meeting