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STATUS OF DEGRADED CORE ISSUES

Synthesis Paper prepared by G. Bandini in collaboration with the NEA Task Group on Degraded Core Cooling

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14th December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

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SYNTHESIS PAPER ON THE STATUS OF DEGRADED CORE ISSUES

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1. Introduction

The in-vessel evolution of a severe accident in a nuclear reactor is characterised, generally, by core uncovery and heat-up, core material oxidation and melting, molten material relocation and debris behaviour in the lower plenum up to vessel failure. The in-vessel core melt progression involves a large number of physical and chemical phenomena that may depend on the severe accident sequence and the reactor type under consideration.

Core melt progression has been studied in the last twenty years through many experimental works. Since then, computer codes are being developed and validated to analyse different reactor accident sequences. The experience gained from the TMI-2 accident also constitutes an important source of data. The understanding of core degradation process is necessary to evaluate initial conditions for subsequent phases of the accident (ex-vessel and within the containment), and define accident management strategies and mitigative actions for operating and advanced reactors.

This synthesis paper, prepared within the Task Group on Degraded Core Cooling (TG-DCC) of PWG2, contains a brief summary of current views on the status of degraded core issues regarding light water reactors. The in-vessel fission product release and transport issue is not addressed in this paper.

2. Status of understanding of core melt progression

The core melt progression [1,2] is characterised by two different phases. The so-called “early phase” refers to the damage occurring between the initial fuel and control rod damage and the melting and relocation of metallic materials. The “late phase” is characterised by the melting of ceramic material, loss of core geometry, and molten material relocation to the lower plenum of reactor vessel. The following subsections of the paper highlight the experimental basis and remaining uncertainties in the current understanding of core melt progression.

2.1 Early phase melt progression

The current knowledge concerning the early phase for PWRs and BWRs [1,2,3] is mainly derived from numerous experimental programs such as PBF-SFD tests and ACRR DF and ST experiments conducted in the USA, the PHEBUS-SFD experiments conducted in France, and the CORA tests performed in Germany. There is a common consensus among experts that the early phase of core melt progression is reasonably well understood. The experimental studies have provided a large database for code model development and validation. Most important physical and chemical phenomena are described in a mechanistic way in the available computer codes.

The issues related to core heat-up, fuel rod clad deformation and ballooning, clad rupture, metallic material oxidation in steam environment are practically solved. Adequate models are available in the system codes dealing with these phenomena, which are validated over an extended experimental database. The availability of steam and the extension of oxidation in the inner clad surface after clad rupture are still uncertain. They may affect the heat-up rate, in particular under oxidation runaway conditions. The uncertainty on the oxide layer failure has the largest impact on the peak temperature and the hydrogen production.

Control rod melting and relocation in PWRs and BWRs, which occurs earlier than fuel rod degradation due to eutectic interactions, is quite well understood and simulated in the system codes. Uncertainties still exist in the interaction between the relocating control rod material and the neighbouring fuel rod and structures, which may accelerate the core degradation process. Aerosols and/or gaseous reaction product release following control rod failure may affect fission product transport and hence the source term. The interaction between the spacer grids and rods is only partially understood.

Major uncertainties in early phase degradation concern the eutectic interactions among the different core materials (e.g. the simultaneous dissolution of UO_2 and ZrO_2 by molten Zircaloy), the integrity of the clad oxide layer and its ability to withstand the inside molten material, and the oxidation of material mixtures and B_4C which may significantly contribute to the hydrogen source. Furthermore, the behaviour of high burn-up and MOX fuel is almost not investigated yet (e.g. foaming due to expansion of fission gas in the fuel matrix). Largest uncertainties in these core degradation aspects are generally reflected in empirical and parametric models included in the present codes.

The uncertainty on oxide layer failure has the largest impact in the evaluation of timing and amount of hydrogen generation, and downward relocation of metallic melt and its solidification in colder region of the core, where flow area blockage may occur.

Recent results from the CIT project of 4th EU Framework Program on Nuclear Fission Safety have provided new insights on material properties and eutectic interactions, including quantitative data on oxide layer breach as mentioned above. Further separate-effect tests and integral experiments foreseen within the ongoing 5th Framework EU program will provide additional data in some areas where main uncertainties have been identified. The PHEBUS-FP program currently underway in France is adding experience in the field of irradiated fuel. On the basis of this additional experience, code model development and validation regarding the early phase of core melt progression could be successfully completed in the next decade.

The VVER experimental database is not as extensive as that for PWRs and BWRs [4]. A number of separate-effect and integral tests indicated that the behaviour of material used in VVERs is generally similar to that of PWRs, although reaction rates often differ. Some specific areas are not covered by experiments at all and their investigation should be given more attention (e.g. slumping in the guide tubes, relocation behaviour of VVER-440 absorber).

2.1.1 Degraded core quenching

Quenching of a degraded core [6,7] could cause renewed oxidation of the Zircaloy cladding re-heating of the rods, sharp increase in the hydrogen generation, and increased fission product release. It could cause also a sharp increase in steam generation, that may cause a pressurisation of the RCS with possible induced break (e.g. steam generator tube rupture). Furthermore, fragmentation of embrittled cladding may lead to the formation of fuel debris region and accelerate the core degradation process towards the late phase, as observed in the TMI-2 accident. These phenomena are important regarding accident management, since the additional hydrogen may threaten the containment, and the increased fission product release increases the source term.

Separate-effect tests and integral experiments have been conducted or are in progress to investigate the quenching of a degraded core, where the fuel rods are in a mainly rod-like geometry at the time of core reflood. The INEL experiments, OECD LOFT LP-FP-2 and PBF SFD-ST, and some CORA experiments contribute to the general database on the quenching of hot, damaged bundles. More recently, single rod and bundle tests performed at FZK within the ongoing QUENCH program have provided valuable new data, supplemented by the VVER test CODEX-3 at AEKI Budapest. Bundle experiments have addressed integral effects, while single rod experiments and tests on hydrogen absorption and release by the cladding have sought to identify the mechanisms in detail.

Separate effects tests have clearly demonstrated the effects of oxide cracking and oxidation of newly exposed metallic surfaces, as well as the role of hydrogen absorption and release, under a wide range of cooling conditions (cold steam vs. reflood water from temperature up to 1873 K). No separate-effects data are however available for temperatures above 1873 K. QUENCH bundle tests have shown temperature excursions and excess hydrogen production for quench from high temperature (2300 K) with a non-pre-oxidised bundle, while smooth cooling with no significant excess hydrogen production has been observed for quench from lower temperatures (1750-1870 K) and with pre-oxidation.

No detailed mechanistic models are available in the system codes to describe core quenching. A simplified shattering model is available in SCDAP/RELAP5; models are under development and implementation in other system codes (ICARE/CATHARE, KESS, and MELCOR). The most detailed work has been performed by IBRAE who have extended the SVECHA code to include mechanistic models calibrated and assessed against FZK test data.

The needs for further experiments and development of more mechanistic models is recognised, in order to resolve the core quenching issue. The main requirement for further separate-effect and bundle quench tests would appear to be generally towards experiments at high temperatures where excursions are probable, and where control material and/or ballooning is present. For those experiments where the temperature is high enough to induce Zircaloy melting and subsequent interactions, the use of real material (irradiated fuel) is required. The effect of system pressure is another parameter not yet studied systematically. The LOFT LP-FP-2 experiments remains the only one (excluding TMI-2) performed with reflood from high temperatures where significant debris bed formation had occurred. Therefore, the possibility of conducting separate-effect and integral experiments on quenching debris bed should be considered.

2.2 Late phase melt progression

The different issues regarding the late phase of core degradation are dealt with in this section. They relate to in-core melt pool formation and spreading, crust failure and molten material relocation into the lower plenum, and the debris behaviour in the lower head of the reactor vessel. The fuel-coolant interaction issue has been discussed elsewhere [5] and is not addressed in this paper.

2.2.1 In-core melt progression

In the late phase, the core heat-up and material interactions lead to melting and relocation of ceramic materials, formation of debris beds and of molten pools. A significant number of experiments have been conducted to investigate the in-core melt progression in the late phase. Some PBF and LOFT-LP-FP experiments have shown evidence of the formation of fuel debris towards the termination of these tests. The first two PHEBUS-FP experiments have shown evidence of molten pool formation in the late phase of the tests. The ACRR DC and MP experiments conducted at Sandia Laboratories have provided information on fuel debris behaviour and melting leading to the formation of a molten pool and the advancement of the pool/lower crust into intact fuel geometry. These tests confirmed the observation from the TMI-2 core that the melting of ceramic fuel materials will results in the formation of a peripheral solid crust region that acts to retain the molten material. The recent PHEBUS FPT-4 experiment, the results of which are still under examination, is providing additional information on debris bed melting and molten pool formation.

The issues related to debris bed and molten pool formation and spreading are not fully solved yet. Furthermore, the actual knowledge is not reflected in adequate modelling in most of the system codes used in severe accident analysis. Major uncertainties concern the timing in the transition from a rod-like core geometry to debris bed and molten pool, and the melting temperature of ceramic material mixtures. The PHEBUS FP experiments have shown that the eutectic melting temperature can be as low as 2500 K, due to presence of different materials in the corium such as iron oxides. Irradiated fuel may react more quickly with liquid metals such as Zircaloy, and may swell (foam) under such high temperature conditions; such effects are not yet modelled in system codes. MOX fuel is also not considered in the present models. Molten pool spreading downward and towards the core periphery can be predicted with high uncertainty. Debris bed heat-up and cooling and oxidation need to be further investigated.

2.2.2 Molten material relocation

Once the core materials are melted within the core there is the potential for material relocation into the lower plenum of the reactor vessel [3]. Two different relocation mechanisms can be envisaged: “coherent” or “gradual” melt relocation, depending on the reactor plant and the accident scenario under consideration.

Coherent melt relocation may occur when the core melting proceeds through the formation of a large molten pool in the core and then the molten pool crust fails due to thermal and mechanical stresses. The presence of water in the bottom of the core (“wet-core” conditions) at the time of core degradation favours the formation of lower core blockages and the development of in-core molten pools. Gradual melt relocation could occur if there are no thermal or mechanical factors causing melt retention in the lower part of the core.

There is a large consensus in the international community that the gradual relocation process is not likely to occur, even in BWRs under “dry-core” core conditions. The coherent melt relocation scenario seems much more likely for light water reactors. The TMI-2 accident proceeded along this general melt relocation scenario, that involved the formation of a large molten pool in the core region that was ultimately released to the lower plenum through a breach in the side wall of the pool crust. Gradual melt relocation could be potentially relevant for VVER-440 designs with large steel-made control assemblies in the core, which can be already melted at the time of fuel assemblies degradation, providing a preferential way for downward material relocation into the lower plenum.

The duration of the process of melt relocation may vary according to the location of the molten pool crust breach, the hole ablation rate, and the presence of various structures in the core periphery and in the lower plenum. The different design of the core periphery and bottom structures, and lower reactor vessel structures in PWRs, BWRs and VVERs may strongly influence core melt interactions during movement into the lower plenum.

The experimental database for core melt relocation is limited. The present knowledge is mainly based on the analysis of the TMI-2 accident. The XR experiments conducted at Sandia Laboratories have investigated relocation of melt in dry-core conditions in BWR-specific geometry. The database regarding structure and behaviour of the in-core molten pool crusts is poor. Some experiments have been conducted to evaluate the impact of the melt onto structures: ablation, spreading, and jet impingement. The lack of data is reflected in the simplification of models presently incorporated in the severe accident codes. On the contrary, a larger number of experiments have been devoted to investigate the melt-coolant interaction during the relocation process (steam explosions and quenching), and to characterise the debris that relocates into the lower head of the vessel.

From the point of view of timing, pouring rate and amount of involved materials the melt relocation process is an almost completely unresolved issue. Further investigation should be addressed to evaluate molten pool crust behaviour, and melt interactions with core surrounding and support structures, which may influence the pouring rate and the transfer mode of molten material to the lower head of the vessel. Models able to deal with these phenomena should be improved and implemented in the present system codes. This work should be supported by experimental analyses allowing theoretical model validation, where the highest uncertainties exist.

The reduction of uncertainties in the analysis of core material relocation would provide more accurate initial and boundary conditions for the analysis of the debris behaviour in the lower head of the reactor vessel. At present, we are not certain about the possibility to describe the melt movement only by purely deterministic tools. In that case, sensitivity studies may be used to address uncertainties in this area.

2.2.3 Debris behaviour in the lower head

The characteristic of the debris that relocates in the lower head of the vessel (composition, size, porosity, etc.) depends on the relocated melt and its degree of interaction with the structures and the water remaining in the vessel bottom during the relocation process. If the relocated debris is in a non-coolable configuration, it may re-heat and melt, and a large molten pool may develop in the lower head of the vessel. Debris and molten pool behaviour in the lower head of the vessel has been analysed in several experimental studies, with the main aim to verify possible in-vessel debris retention and coolability [8,9].

The formation and the cooling of corium debris bed resulting from corium melt jet quenching tests have been investigated in FARO and KROTOS facilities at JRC/Ispra. Progress in understanding and modelling of debris bed heat transfer is underway. No priority needs are identified, nevertheless, efforts should be maintained to enrich the knowledge of basic phenomena involved in severe accidents, and increase the capability of numerical tools to predict scenario. The FARO facility has provided quantitative results, but there are difficulties to transpose them to reactor scale.

Corium pool convection and crust formation have been investigated in several experimental work using simulant materials and for different shapes of the lower head. The ACOPO experiments at the University of Santa Barbara (California) and the COPO I experiments at IVO (Finland) were aimed at demonstrating the in-vessel debris retention capability by external vessel flooding for the AP-600 and the Finnish VVER-440 reactor, respectively. The COPO-II-Lo and COPO-II-AP have investigated more deeply heat transfer phenomena in corium pools, for tori-spherical and hemi-spherical shapes of the lower head. In BALI experiments at CEA Grenoble and SIMECO tests at RIT (Sweden), the melt stratification and focusing effect have also been studied. Within the ongoing OECD RASPLAV project, a large number of experiments have been performed, including both simulant and real material tests. Data have been obtained for the properties of various core melt compositions and the behaviour of prototypic core melt in a relatively large-scale melting facility. Stratification or segregation behaviour has been observed for a mixture of UO_2, ZrO_2 and Zr, however, the influence of test specific materials (graphite, carbon) is still unclear.

Convective corium pool behaviour with all chemical and physical aspects requires still confirmatory research, especially for the corium properties and the transposition of simulant test results to the reactor problem. The pool formation and the initial conditions in the lower plenum are plant and accident sequence dependent and are subject to large uncertainties.

The formation of narrow gaps between the core debris and the lower has been confirmed after relocation of corium melt masses up to 200 kg in ALPHA, LAVA, FAI, FARO and FOREVER experiments. The thermal-hydraulic cooling mechanisms and critical heat fluxes that occur in the gap have been investigated in VISU II, BENSON, CTF and FOREVER facilities. The total power transfer to the fluid in narrow gaps is limited by dry-out rather than critical heat flux. The critical heat flux increases with gap width, pressure and sub-cooling.

Internal gap cooling requires long-term availability of water and stable crust formation. The latter is questionable for larger melt pools and for metal-rich melt. The safety relevance of this phenomenon is questionable, since the gap formation and stabilisation conditions are not enough understood for a practical application to the reactor.

2.2.4 Core degradation after vessel failure

The TMI-2 examination showed that the peripheral fuel assemblies remained intact after the relocation of the central core region into the lower plenum and molten pool formation. It can be expected that even after the vessel failure, a large mass of core components remains in the reactor vessel. Following the lower head damage air can enter the primary system and the character of the degradation process can be changed compared to steam atmosphere conditions.

The interaction of air with Zr alloys and UO_2 pellets can strongly affect the evolution of severe accident scenarios through heat generation, increased core degradation and fission product release. The interaction of air with fission products can influence the radiological source term due to the volatilisation of ruthenium oxide [10]. The Zr oxidation in air produces 85% more heat than in steam. The UO_2 pellets can be further oxidised and this results in a lower melting point.

Experimental and computational studies with air ingress conditions were performed in the OPSA project of the 4th Framework EU program [11]. Oxidation kinetics of Zr in air was investigated at FZ Karlsruhe and at AEKI Budapest in separate effect tests. Experiments in the DRESSMAN facility Dresden with single Zircaloy fuel rod simulators showed that the pre-oxidation period in steam can reduce the intensity of later air-Zr interaction. Two 9-rod bundle tests were carried out in the CODEX facility at AEKI Budapest under air ingress conditions. The experiments indicated the acceleration of oxidation phenomena and core degradation process during the air ingress phase of the severe accident scenario. The degradation process was accompanied with zirconium nitride formation and release of U-rich aerosols.

The fission product release during air ingress conditions will be addressed in a later PHEBUS test (FPT5). A future bundle experiment with air oxidation is considered in the QUENCH test matrix as well. The consequences of air oxidation on the core degradation and fission product release in a reactor case cannot be well estimated today and this field may need further analytical and experimental work.

3. Conclusions

The status of understanding of the degraded core issues has been presented and discussed in this paper. The areas with remaining uncertainties and the needs for further experimental investigation and model development have been identified.

The early phase of core melt progression is reasonably well understood. Remaining uncertainties may be addressed on the basis of ongoing experimental activities, e.g. on core quenching, and research programs foreseen in the near future. The late phase of core melt progression is less understood. Ongoing research programs are providing additional valuable information on corium molten pool behaviour. Confirmatory research is still required. The pool crust behaviour and material relocation into the lower plenum are the areas where additional research should be mainly envisaged.

Reducing uncertainties is a reasonable objective of additional research where existing uncertainties limit the understanding of potential plant behaviour and the assessment of risk. Best estimate calculations, in conjunction with bounding calculations, engineering judgement, and systematic probabilistic assessment, have been used when the physics of core degradation phenomena is not precisely known.

Better understanding of in-vessel core degradation phenomena will also help to define more precisely preventive and mitigative actions for severe accident management strategy, in order to limit the consequences of severe accidents and reduce the risk for operating and advanced reactors.

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