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COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

TECHNICAL OPINION PAPER ON FUEL-COOLANT INTERACTION

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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) and the Republic of Korea (12th December 1996). 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.

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.

Technical Opinion Paper on Fuel-Coolant Interaction

I. Introduction

The interaction of the reactor core melt with water, known as Fuel-Coolant Interaction (FCI), is one of the most complex technical issues involving a number of thermal-hydraulic and chemical phenomena. FCI's may occur in-vessel, during flooding a degraded core or when a molten core relocates into the lower head filled with water. They may also occur ex-vessel, when molten core debris is ejected into a flooded reactor cavity after the vessel failure or when the molten debris is flooded in the containment. Each of the scenarios may lead to an energetic FCI, commonly known as "steam explosion", which represents potentially serious challenge to the reactor vessel and/or containment integrity. The case of most concern is when molten debris is poured or ejected into a pool of water. Non-explosive FCI's lead to melt quenching to varying degrees during fall in the water and debris relocation into the lower head or the cavity pit.

The issue of understanding the phenomena involved in a FCI and its potential effect on reactor vessel and containment integrity have been studied and discussed for the last several years, most recently at the SERG-2 meeting, in 1995, and the Specialist Meeting in Tokai, in 1997. The reactor safety issues associated to FCI are the alpha-mode failure, core melt relocation, lower head integrity, and debris characteristics, retention and cooling. These issues are relevant to accident management strategies for operating reactors (cavity flooding in case of ex-vessel or late reflood in-vessel) as well as accident mitigation features for advanced reactors (e.g., in-vessel retention by ex-vessel cooling).

This Technical Opinion Paper (TOP) contains a brief summary of current views of the leading OECD FCI experts, based on the characteristics of modern western-like power plants. It has been prepared by PWG-2 with contribution of PWG-4. It is important to stress, however, that this paper does not address regulatory and policy issues. The expressed experts opinions are summarized here in support of regulatory decision making process in order to help the nuclear power regulators in addressing the FCI issue consistent with their own regulatory philosophy.

II. Summary of Experts Opinions

The experts agree that significant progress has been made in FCI research in the last two decades. Many experts also agree that research has attained a sufficient degree of maturity **in some areas** to adequately address a number of FCI issues from a risk perspective.

One of these issues, which retained particular attention over the years because of its potential dramatic short-term consequence for the containment, is the so-called alpha-mode failure, i.e. containment failure due to the impact of the RPV upper head launched as a missile against the containment wall as the result of an in-vessel steam explosion. A general conclusion from the SERG-2 meeting, which was confirmed at the Tokai meeting, is that **the alpha-mode failure issue can be considered resolved from a risk perspective**, meaning that this mode of failure is of very low probability, that it is of little or no significance to the overall risk from a nuclear power plant, and that any further reduction in residual uncertainties is not likely to change the probability in an appreciable manner. It is noted that on-going programmes (e.g., BERDA and ECO at FZK) have the aim to confirm this conclusion by performing deterministic assessment of the issue.

The other FCI issues are debris quenching and coolability, and energetic FCI with implications to RPV lower head integrity and reactor cavity structure integrity under shock loading. These issues are relevant to accident management strategies for operating reactors (cavity flooding in case of ex-vessel or late reflood in-vessel) as well as accident mitigation features for advanced reactors (e.g., in-vessel retention by ex-vessel cooling). These issues are recognised as important reactor safety issues by the experts, while they recognise also that, in general, **the current experimental database on them is insufficient.** However, concerning energetic FCI in particular, the opinion is that **the consequences of a steam explosion under reactor conditions may not be as severe as previously envisioned,** and that a limited additional effort in some areas would be sufficient to support this conclusion. This statement is based on a number of recent experimental data, which will be reviewed in the next section.

There is a fair consensus that fully validated codes are not required for reactor application and that **fit for purpose validation** would be sufficient. This approach includes three mains steps to be performed: 1)- set-up and validate models on physical basis and analytical experiment support, 2)- verify and assess the model and code performance on integral small-scale and large-scale experiments, 3)- apply the codes to the reactor scale. Experiments with prototypical materials are required at least for step 2. However, due to the lack of physical understanding and/or lack of data, **the codes have not yet achieved this stage of validation for all the phases of a FCI.** Details are given in the next section.

Probabilistically based approaches should also be considered as a methodology for resolving specific steam explosion issues, and are most useful for addressing the significant uncertainties which can exist in initial and boundary conditions (e.g., melt mass, composition and temperature as well as jet diameter and number). In some cases, it may be expedient to address specific uncertainties in a bounding manner. An evaluation of the impact of FCI's in reactor situations based on appropriate boundary conditions, a conservative approach to triggering and sufficiently validated FCI codes constitutes an adequate procedure, which may also benefit from independent peer review.

Future work should focus on improving both probabilistic and deterministic methodologies, and understanding of the physics of corium jet break-up and fine fragmentation mechanisms, in particular. In addition, any future experimental programme should take into account problems of extrapolation to the reactor case. Working both analytically and experimentally with **real material** is necessary, with a few confirmatory experiments able to reproduce as far as possible reactor conditions.

III. Status of understanding of the various phases of a FCI

One can identify four "stages" in a steam explosion: premixing, triggering, propagation and expansion. The knowledge of the first two is the most important for evaluation of the steam explosion probability, while the knowledge of the last two is basic for an estimate of potential consequences. The scenario considered the most challenging for the vessel and containment is when a molten core drops into the lower head or into the cavity filled with water.

III.1. Premixing/Quenching

The pre-mixing phase is characterised by a (partial) break-up of the jet into droplets during its fall in the water and further fragmentation of the droplets. In this phase, the melt and the coolant are in a meta-stable equilibrium characterised by film boiling of the coolant. The kinetics of this mixing process and relative heat transfer have been and are being extensively studied 1)- because it is the initial condition from which a steam explosion may trigger and 2)- because of its implication for debris coolability in the lower head in case of no steam explosion (quenching).

Of particular importance is the FARO programme at the Joint Research Centre (JRC), Ispra, which has been providing experimental data on quenching of large masses of prototypic core melt. It is considered that the analysis of these experiments considerably improved the general understanding of melt/water non-energetic mixing. The initial and boundary conditions for these experiments are mostly representative of in-vessel conditions during an accident progression, while two tests have been performed with subcooled water at low pressure (ex-vessel conditions). Several tests have been also performed in subcooled water at a smaller scale in the KROTOS facility of JRC. Recently, NUPEC of Japan has conducted ex-vessel FCI in COTELS project in collaboration with National Nuclear Centre (NNC) of Republic of Kazakstan. In this project, typical ex-vessel conditions of BWR, including relatively high content of metallic components in the corium melt and nearly saturated water, were simulated.

The results to date from these programmes indicate that prototypic core melt shows no potential for a spontaneous explosion. Also, the simulated thermal load on the vessel lower head imparted by the quenched debris was found, in FARO, to be sufficiently small though it is recognised that the decay heat was not simulated in the experiments. At low pressure and saturated water **part of the debris was dispersed** and did not collect on the bottom structures. The presence of metallic phases in the molten corium enhanced melt break-up and debris dispersal. Due to increasing void fraction and inert gas (hydrogen) production by chemical reaction during the interaction, corium melts with metallic components exhibit, from the thermal-hydraulic point of view, even less propensity than pure oxidic corium to induce steam explosion. Recent tests with melts involving more than 60% of zirconium, the rest being zirconium oxide (ZREX series at ANL), have also shown that no spontaneous steam explosion occurred. However, steam explosion occurred when an external trigger was applied to these mixtures, and the explosion energetics, although low, were augmented by the zirconium-water interaction.

Generally, it may be concluded that in the absence of any external trigger mechanism, **the lower head integrity is not likely to be challenged by an in-vessel FCI**. However, it is noted that experiments to date have only been performed in a very limited range of ex-vessel conditions and that more experiments involving melt compositions with metallic components, and possibly, higher melt release rates or higher melt mass would be necessary. Therefore, a definite statement cannot be made at present regarding the potential of a spontaneous steam explosion under these conditions; nor can a definite statement be made regarding debris quenching and the resulting thermal load on the reactor cavity.

Both SERG-2 and CSNI FCI meetings concluded that substantial progress had been made in premixing research verifying the mixing limit concept, albeit, using data from experiments involving melt particles rather than melt jets. A number of premixing models and/or codes have been developed and validated against these separate effect experiments. In the recent CSNI-sponsored International Standard Problem exercise (ISP-39), many of these codes were used for the post-test prediction of the FARO integral experiment L-14 involving a melt jet.

One general conclusion from the ISP-39 exercise is that **none of the codes performed consistently well in post-test prediction of all measured quantities**. The calculated energy release to the steam/water system was in general too low with respect to the experiment. This poor performance of the codes was attributed mainly to a deficiency in modelling the energy partitioning between steam and water, and the heat transfer at the steam/water interface. Actually, most of these codes either do not have jet breakup models at all or do not have well validated models. Consequently, there is a need to improve jet breakup models and validate these models against melt jet experiments. This would help reducing the parametric factors still present in many of these codes.

In this regard, it can be stated that from the debris quenching and coolability perspective **sufficient data has been collected for the integral behaviour of oxidic melts** to allow an adequate validation of the aforementioned models. The data base is still **lacking results for prototypical melts which contain metallic phases** and there is also a need to adequately instrument these future experiments so that local mixture conditions and properties can be evaluated for code validation.

III.2. Triggering – Propagation - Expansion

From the meta-stable equilibrium discussed above some internal or external "event", the trigger, (e.g., pressure wave) may induce locally the collapse of the vapour film, liquid (melt) - liquid (coolant) contact, fine fragmentation of the melt and rapid heat transfer to the coolant. The local pressurisation induced by the rapid heating of the coolant may in turn produce the collapse of the film of the adjacent regions and so on, resulting in a self-sustained propagation to the rest of the pre-mixture like in a chemical detonation**. This concept of "thermal detonation" is widely accepted among the experts.**

The understanding of the triggering mechanisms is still poor. It has been observed that a steam explosion may spontaneously trigger when the melt comes in contact with structures, and may be artificially triggered by pressure waves induced by firing a small amount of black powder or by the rupture of a gas capsule. As an example, artificially created pressure pulses of amplitude of the order of 10 MPa and having energy of a few hundred of Joules have induced very energetic steam explosions with alumina melts.

To date, the few experiments performed at a system pressure of 0.1 MPa and subcooled water with small masses of prototypic core melt in the KROTOS facility **did not produce an explosion even when an artificial trigger was applied.** The few higher pressure experiments performed at system pressures in the range of 0.2 to 0.4 MPa showed very weak explosive events when artificially triggered. Conversely**, very energetic steam explosions easily occurred when pure alumina melt was used** in similar geometrical conditions.

While the KROTOS data demonstrates that it is difficult to trigger an explosion involving prototypic core melt, both the SERG-2 and the CSNI FCI meetings concluded that, due to its very stochastic nature, such an event could not be positively excluded at present in reactor geometries under all accident conditions of interest. Consequently, **a conservative approach to triggering is recommended by the experts**, i.e., steam explosion energetics and potential consequences must be assessed under the assumption that an explosion will be triggered at the worst time during the premixing transient.

It is believed that the **material characteristics play a fundamental role** in steam explosion triggerability and explosivity, possibly explaining the low yield observed in corium tests. However, there is still a lack of understanding on the exact role each single material property may have in this process. **The role of non-condensable gases** (e.g., hydrogen generation due to melt oxidation during pre-mixing) is poorly understood. In general, the kinetics of chemical interactions during the time-scale of the pre-mixing is not known and, consequently, not modelled in the codes. Leaving aside the possible role of chemical reactions during the propagation, non-condensables gases in the pre-mixture are believed to make the triggering of an explosion more difficult and to limit the energetics. **Venting** possibilities exist in a reactor, which may decrease the constraint during the propagation thus reducing the dynamic loading of the structure and/or absorb part of the energy during the expansion phase (e.g., downcomer).

Various analytical models have been developed based on the thermal detonation concept. However, the phenomenology of fine fragmentation of droplets and resulting rapid heat transfer is believed to be not thoroughly understood for lack of appropriate experimental data involving prototypic core melt. Therefore, the need exists in both experimental and analytical research to understand better the droplet fragmentation during the explosion or propagation phase, to develop and/or improve fine fragmentation models, and to assess these models for use in reactor applications.

It is stressed that reliable material **properties of high temperature core melt mixtures** and **experiments involving prototypic core melt in reactor-like configurations are crucial** in extrapolating experimental findings to a real reactor situation.

IV. References

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