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Increased Accident Tolerance of Fuels for Light Water Reactors

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 31 countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Republic of Korea, the Russian Federation, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission 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 the 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.

Foreword

The Fukushima accident in March 2011 raised concerns about the safety of current and future nuclear power plants both inside and outside the international nuclear energy community. With a view to learning lessons from this accident a large consensus emerged on the need to strengthen each level of Defence-In-Depth, reinforcing both prevention and mitigation.

The fuel performance characteristics identified as being central to increased accident tolerance for long-term loss of coolant include reduced clad-steam reactions, reduced hydrogen production and improved fission product retention. New fuel designs which offered the potential to incorporate these characteristics, while retaining the operational performance of existing designs, would therefore be considered as suitable candidates for further investigation.

Under the auspices of the NEA Nuclear Science Committee, a workshop has been organised to bring together international experts from the modelling, safety, operations and regulatory technical disciplines to discuss the various issues related to increased accident tolerance of fuels for Light Water Reactors and to help establish a co-ordinated international approach in this field. The organisation of this workshop was also supported by the NEA Committee on the Safety of Nuclear Installations.

These proceedings include all the abstract papers presented at this workshop. The opinions expressed are those of the authors only and do not necessarily reflect the views of the NEA, any national authority or any other international organisation.

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Executive summary

After the events at the Fukushima Nuclear Power Plant in Japan in March 2011, enhancing the accident tolerance of Light Water Reactor (LWRs) fuels became a heavily discussed issue. One outcome of those discussions has been to promote research into the development of advanced fuels and more robust reactor system technologies (e.g. instruments, auxiliary power sources) with improved performance, reliability and safety characteristics during normal operations and accident conditions.

In this context, this workshop organised by the OECD Nuclear Energy Agency (NEA) represented the first step towards gathering international experts from the international community with the aim of:

- drawing the status of knowledge in the field of advanced fuel for LWRs;
- analysing design requirements and potential options, defining metrics;
- identifying the key elements of future collaborations in this field.

The programme was comprised of 4 sessions:

- Session 1: Lessons learned from the Fukushima accident;
- Session 2: Accident-tolerant fuel design;
- Session 3: Reactor operation, safety, fuel cycle constraints, economics & licensing;
- Session 4: Synthesis and future programmes.

A total of 55 participants from 16 countries attended the workshop, with 26 technical presentations and 2 breakout parallel sessions (one on safety issues, the other on reactor performance, R&D and technological issues).

The attendees represented a broad spectrum of stakeholders involved in different nuclear energy organisations, mainly R&D and industry and representatives from regulatory bodies. The list of participants is given in Annex 2.

Opening Session

The workshop opened with the welcome address from **Th. Dujardin (OECD/NEA)**, NEA Deputy Director. Th. Dujardin recalled the integrated NEA response to the dramatic Fukushima-Daiichi events performed by three standing technical committees: the Committee on Nuclear Regulatory Activities (CNRA), the Committee on the Safety of Nuclear Installations (CSNI) and the Committee on Radiation Protection and Public Health (CRPPH).

J. Gulliford (OECD/NEA) placed the workshop in the context of the activities of the Nuclear Science Committee within the framework of the NEA response to Fukushima-Daiichi.

K. Pasamehmetoglu (INL, US) explained the main goals of the workshop oriented towards defining requirements for selection among various options during the feasibility phase of the development process, and not towards identifying and proposing design solutions.

Session 1: Lessons Learned from the Fukushima Accident

Chair: K. Pasamehmetoglu (INL, US)

This session was devoted to setting up the context for the workshop, summarising the current level of understanding of the accident at Fukushima-Daiichi, the lessons learned to date, and the recent activities at the OECD/NEA:

- **J. Gulliford (OECD/NEA)** presented the current understanding of the sequence of events on Units 1-3, strengthening the difficulty for the operators to know the precise reactor conditions. Currently available information comes from TEPCO and from numerical simulations. Additional information will be provided at the beginning of 2014 through an OECD/NEA Benchmark Study of the Accident at the Fukushima-Daiichi NPS (BSAF) Project. The role of the fuel behaviour on the progression of the accident appears clearly through the zirconium-steam reaction and the hydrogen production.

- **S. Massara (OECD/NEA)** gave more insights into the numerical simulations by means of severe accident codes. The accident behaviour is well known on the qualitative level, but for the quantitative level the recourse of modelling is mandatory: a relatively good agreement on the pre-fuel melting is observed, whereas much higher uncertainties remain in the post-melting phase, and particularly on the final status of the primary reactor vessel. Numerical simulations show the very quick kinetics of hydrogen production. There are no available elements to extrapolate the validity of current severe accident tools for the study of accident tolerant fuels.

- **K. Pasemehmetoglu (INL, US)** discussed the main outcomes of a recent US National Workshop on Accident-Tolerant Fuels, focusing on results for ATF metrics development. All thermal, mechanical and chemical properties are relevant in defining the metrics for accident tolerance, but considerable testing and analyses are needed to identify the dominant attributes and quantify the metrics. Current analysis tools are not fully adequate to complete the task, and a need was highlighted for strong collaborations to complete the experimental data to qualify the new tools.

- The main lessons learned from the Fukushima accident were presented by **J. Nakoski (OECD/NEA)**, covering a large spectrum from accident management to crisis communication, from radiological protection and public health concerns to decontamination and recovery. Defence-in-depth, deterministic vs. probabilistic assessments and regulatory infrastructure were also addressed, as well as the actions undertaken by the NEA standing committees (CNRA, CSNI and CRPPH) in response to the accident.

- **M. Petit (IRSN, France)**, on behalf of the NEA CSNI/Fuel Safety Expert Group, which he chairs, focused on recent and current activities inside this expert group, which include analysis of LOCA and RIA accidents. Discussions on new concepts such as accident tolerant fuels are only in the initial phase as yet, and links and discussions with other NEA Expert Groups (like WPRS/EGRFP) should be strengthened on the basis of the outcomes of this workshop.

- **K. Pasemehmetoglu (INL, US)** introduced the current status of the lessons learned from the Fukushima accident, and in particular, the recommendations released by a NRC Near-term Task Force to enhance reactor safety in the 21th century. The near-term recommendations are focused on emergency power and emergency cooling availability during station blackout accidents.

Session 2: Accident-tolerant Fuel Design

Chair: K. Pasamehmetoglu (INL, US)

This session covered the analysis and discussion of the main elements referring to the design of accident-tolerant fuels, mainly focusing on on-going studies on innovative materials (fuels and cladding) carried out by the organisations attending the workshop.

- **F. Goldner (DOE-NE, US)**, gave an overview of the DOE development programme of accident-tolerant fuels, which is being implemented as a collaborative effort among national laboratories, industries and universities within the US but for which, for the very nature of the problem, international collaboration would be extremly beneficial. The main requirements of innovative fuels were indicated, with a voluntarily ambitious timeline (feasibility assessment for 2016, development and qualification for 2022, commercialisation beyond 2022). The establishment of future collaboration and partnerships stimulated by actions initiated at this workshop was suggested as one of the major outcomes expected from this workshop.
- **M. Le Flem (CEA, France)** illustrated on-going studies on innovative cladding materials at CEA, which follow two major axes: protecting current zirconium alloys with a robust thin coating, and the development of new cladding based on SiC/SiC composites. Future studies will cover selection and optimisation of materials, as well as prototype manufacturing and first experimental tests on the most promising materials.
- **T. Hinoki (Kyoto University, Japan)** presented the current status of and issues related to SiC cladding, which seems a promising material for LWR in terms of excellent stability of dimension and strength under neutron irradiation and excellent resistance to high-temperature steam. Future studies will address main issues, including the effect of impurity, the synergistic effect of irradiation and high-temperature water, compatibility with fuel, coating, large scale fabrication techniques and material cost.
- **J.-Y. Park (KAERI, Republic of Korea)** summarised on-going activities at KAERI, focusing on two axes: coated zirconium alloys and metal-ceramic hybrid cladding. It has been found that Cr and Si show the best performances as coating materials for Zr. Current studies on hybrid cladding focus on manufacturing processes, and the joining technology for SiC.
- **T. Sawabe (CRIEPI, Japan)** presented the innovative design of accident-tolerant control rod, implementing an axially heterogeneous neutron absorber: a higher neutron absorber than in current control rod concepts (Gd, Sm, Eu, Dy) located in the low-flux zones – in order to increase the shut-down margin in the case of severe accident – and more traditional absorbers (B_4C or Ag-In-Cd) in the high-neutron flux regions, to minimise the penalties on reactor operation and to increase the lifetime of the control rod itself. Future studies will focus on the choice of innovative absorbers and the investigation of the miscibility with molten structural and molten core materials.
- **L. Snead (ORNL, US)** illustrated on-going activities on fully ceramic microencapsulated fuel design, associating TRISO particles in a SiC matrix. In particular, the irradiation campaign at the ORNL High Flux Isotope Reactor and the post-irradiation examination campaign devoted to exploring the stability of the SiC matrix were addressed in this presentation.
- **J. Tulenko (University of Florida, US)** highlighted performances obtained by doping UO_2 fuel pellets with SiC whiskers. The reduced thermal expansion, thermal cracking and fission gas release resulting from the increase of the pellet

thermal conductivity are expected to produce an enhanced accident-tolerant fuel. If numerical results on the fresh fuel appear promising, the behaviour on irradiated condition will be validated against experimental results in a research reactor in the framework of collaboration with AREVA.

- **J. Strumpell (AREVA-NP, US)** gave an overview of the ATF Project at AREVA. A 10-year window is embraced, and as costs for licensing, testing and fabrication could be prohibitively expensive for any one vendor, strong partnerships with government and fuel vendors similar to the historical development of zirconium alloy cladding should be established.
- **C. Back (GA, US)** illustrated fuel technologies at GA, including the development of β-SiC-SiC composites for ATFs. GA β-SiC/SiC development addresses all aspects of fuel cladding fabrication, including research done to accelerate SiC-SiC fabrication time, research to produce irradiation resistant joints, material and part characterisation.
- **M. Pouchon (PSI, Switzerland)** presented an innovative design of ATF developed at PSI, associating an annular pin design and a sphere-pac fuel, with an attempt to reduce the fuel temperature by an increased heat exchange with the cladding (no pellet-to-cladding gap due to sphere-pac fuel) and with the coolant (increased exchange surface due to the annular geometry). The presentation also focused on SiC related activities at PSI, mainly analysis and testing methods.
- **B. Heuser (University of Illinois, US)** illustrated the US DOE NEUP integrated research project, devoted to the fabrication and evaluation of modified zircaloy LWR cladding, following two pathways: the first one is to add a protective coating; the second is the modification of the bulk cladding composition itself to shift the reaction away from oxide formation during temperature excursions.

Session 3: Reactor Operation, Safety, Fuel Cycle Constraints, Economics and Licensing

Chair: K. Pasamehmetoglu (INL, US)

This session aimed at gathering elements on the safety behaviour of innovative fuels and cladding materials identified as possible candidates for ATF during accidental conditions, up to severe accident. Also, items related to the requirements from the point of view of the fuel cycle, fuel licensing process, in-reactor testing capabilities and fuel performance code were discussed.

- **L. Ott (ORNL, US)** opened the session with an assessment of the impact of candidate ATF on the predicted reactor behaviour at normal operating conditions and under DB and BDB accident conditions. If in normal operation the main benefit is a much lower centreline temperature and fission gas releases, the increase of grace time in BDB accident depends on the considered scenario (with a magnitude ranging from minutes in a STSBO without water injection, to a few hours in the case of water injection). In the end it was stressed that reactor safety is determined by the system performance, which includes fuel, ECCS and operator actions.
- **B. Cheng (EPRI, US)** focused on the state of fuel rods under severe loss of coolant conditions, potential candidate alloys for accident-tolerant fuel cladding, and a novel design of molybdenum-based fuel cladding. The cladding was developed to maintain the fuel rod integrity and reduce the hydrogen production and exothermic heat in accident conditions. The demonstration of forming composite Mo-based cladding via mechanical reduction will be performed in the future.
- **G. Hache (IRSN, France)** presented the outcomes of a bibliographic study by IRSN on the safety of some fuel cladding materials alternative to Zr alloys. The different

response of the SiC oxidation kinetics below and beyond 1 700°C was highlighted (with a cliff-edge effect due to the loss of the protective effect, a result of the molten SiO_2) as well as results on reaction heat and physic-chemical interactions. The behaviour of stainless steel is also discussed and completed with 304L results from Westinghouse and Fauske MAAP numerical simulations.

- **L. Snead (ORNL, US)** presented, on behalf of K. Terrani, results of steam attack studies in SiC clad and advanced steel claddings recently carried out at ORNL. The most promising candidates are ferritic alloys that contain some Al in addition to Cr. FeCrAl alloys exhibit exceptionally low oxidation in high-temperature steam up to 1 300°C due to the formation of a protective alumina film on the surface of the material.
- **P. Van Uffelen (EC-JRC)** gave a brief overview of the structure of the TRANSURANUS code and the corresponding input requirements before summarising the needs for simulating new cladding materials such as those considered in the workshop. Two concrete examples were given, relatively to SiC and T91, implemented in the framework of collaborations with CEA and Politecnico di Milano.
- **C. Vitanza (OECD/Halden)** outlined the type of reactor testing that is currently required in the development of a new fuel, and that is likely to be necessary for ATF. In-reactor testing experiments were addressed such as material property assessment, rod integrity in normal operation conditions and rod behaviour in accident conditions.
- **M. Moatti (EDF, France)** highlighted the main issues related to the introduction of a new fuel in EDF reactors, focusing on in-core operational feedback, coherence of the nuclear fuel cycle and the issue of mixed cores. The EDF experience illustrates that entering a new fuel design on an industrial scale in EDF reactors can be a very long process and in the case of an ATF fuel, some additional time may be necessary to finalise the design. However, it may be worth considering a breakthrough technology for PWRs if safety margins are significantly improved.
- **N. Waeckel (EDF, France)** presented the requirements for the licensing process of a new fuel technology. In addition to identifying the promising ATF concept, the developers will have to address the manufacturing issues, the accidental and normal operation behaviour issues, as well as the economics vs. benefits vs. risk issues related to the implementation of a new fuel concept. Considering the licensing experience accumulated in France in increasing the fuel burn-up, and the time needed to get any irradiation feedback, the entire development/licensing process will be likely much longer than expected.

Session 4: Synthesis and Future Programmes

Chair: K. Pasamehmetoglu (INL, US)

The session opened with a report from the discussions in the breakout sessions, followed by a presentation on international collaboration and the synthesis of the workshop and conclusion.

- **Report from Breakout Session A. Safety Issues:**

M. Petit (IRSN, France) summarised the major outcomes of the discussion, covering the definition of a reference scenario (SBO), the notion of "grace period" as a figure of merit for the evaluation of the candidates, the importance of ensuring that normal operation conditions would not be worsened with ATF, a proposal of categorisation of testing priorities, the need for improving simulation tools, stressing the importance of establishing an appropriate experimental programme in parallel with the development process. It should be noted that during an extended station blackout, failure of components other than fuel may be important during the accident progression.

- **Report from Breakout Session B. Reactor Performance, R&D and Technological Issues:**

N. Waeckel (EDF, France) summarised the outcomes of the second breakout session. The importance of stressing that ATF should be as good as current fuel during normal operation ("first do no harm" principle), together with the researched improvement in DBA and BDBA conditions, but also the need for defining measurable objective of ATF in terms of metrics (Which target? Which criterion for success? Which tests to run to justify the new fuel?) – to cite only a few of the questions raised. A few recommendations were also made concerning the organisation of the general development process, the importance of the international co-operation, as well as the key role that the NEA should play in this activity. It is noted that, for truly revolutionary fuel designs with extensive safety benefits, the 10-year development time to reach lead test assemblies may be too optimistic. For some of the advanced concepts, 25-30 years may be needed for full commercialisation.

- **A. Sowder (EPRI, US)** presented the EPRI proposal for establishing an international collaboration for the development of ATF fuel. The recourse to international collaboration is mandatory given the fact that the financial support to commercialise ATF is estimated beyond 1 billion US dollars, and scheduled around 2030. A tentative schedule and a proposal of structure for this collaboration are also addressed in the presentation.

In the discussion that followed this presentation **T. Liu (Nuclear Power Technology Research Institute, People's Republic of China)** explained that China is starting to finance R&D on fuel to accompany its massive development of nuclear power. Sustainable development of nuclear fuel has been a major issue in recent years in China, as proved by the launching of a new programme in June 2012 devoted to testing a new zircaloy cladding. The integration of China in an international collaboration on advanced fuels would be highly beneficial to the country, contributing to filling a gap in fuel testing facilities and associated experiments.

- **The main conclusions of the workshop** were summarised by the chair of the workshop:
  - **Severe accident modelling methods:** Current tools are not (cannot be) well validated for assessing the tolerance of fuels for all types of postulated severe accident. Many of the codes used for accident analyses (especially for beyond design basis accidents) are 1980s vintage. As such they are limited by the old computational technologies. It is difficult (if not impossible) to add new properties, design information and new phenomenology to these codes. However, with the availability of high-performance computer and advanced software development techniques these limitations may now be overcome. Such modernisation will enable the exploration and provide insight into the physics associated with accident progression. However, verification and validation of such tools are expensive and lengthy. Well-validated high fidelity modelling of the basic phenomena is therefore a priority. A review of those phenomena should be carried out to identify key needs. It is suggested that a small group of code developers might be established to identify the main development needs and that this group should work closely with the team working on the OECD/NEA Benchmark Study of the Accident at the Fukushima-Daiichi NPS (BSAF) project. The aim would be to establish some validation of the models for Fukushima-like accidents and to apply any improved understanding of key processes and phenomena to the further development of methods.

- **Experimental facilities:** Experimental capabilities look reasonably adequate to start basic testing of candidate materials and fuel designs, considering the type of testing run historically for LWR fuels. However, the advanced fuel designs that are considered may require additional capabilities and/or modification to the existing capabilities (e.g. higher temperature testing). In addition, the multi-scale, multi-physics codes, discussed as part of the modelling needs, may require experimental data for validation beyond what the older models needed. It is concluded that more detailed guidance on which properties should be measured, and to what accuracy, would be useful. As new materials emerge as candidates, checks should be made to identify any relevant new phenomena. In the longer term in-core testing of prototype fuel designs will be needed. While the experimental capabilities spread across the countries appear reasonable to begin testing, there is not a single place where all the capabilities exist. Thus, irradiated fuels may have to be transported to multiple laboratories internationally. The issue of an international transfer cask for timely transport of the fuels is raised.
- **Testing in an operating LWR:** While experimental irradiations are feasible for many of the designs considered under a variety of conditions, introducing lead test rods and/or lead test assemblies into an operating reactor may not be possible in the near-term for the new designs considered. The safety basis for introducing the new designs into an operating reactor will require extensive data and validated models. In some cases, the regulatory framework has to be modified in order to introduce the new materials. Many participants in the workshop indicated that, a 10-year goal of introducing fuel rods of new design into an operating LWR may be too optimistic unless the changes are very small. A more likely commercialisation timeframe may be about 30 years, with a lead test assembly target of 20 years.
- **Key pieces of information and knowledge**: An in-depth understanding of the oxidation of Zr-based alloys remains one of the key elements needed to progress the development of increased accident tolerance of fuels. While much work has been done in this area, some relationships (such as response to impurities) are only known empirically. Also, of high importance would be an improved understanding of fission product retention in degraded UO_2 fuel. In the context of accident tolerance this would be one of the most valuable pieces of information to emerge from the recovery programme at Fukushima. As the analysis methodologies are improved and a better insight into the physical phenomena is gained, there will be needs for additional data and, with the guidance of modelling and simulation efforts coupled with laboratory-scale experiments, such data may be collected from the accident site.
- **International collaborations:** There are multiple international institutions investigating various fuel designs and cladding materials to enhance the accident tolerance of LWR fuels. There are too many options for any single institution to explore within a reasonable period of time and within a reasonable budget. Therefore, the topic can benefit considerably from a strong international collaboration, at least during the feasibility phase until a small number of technologies are identified as candidates for large-scale development and qualification. The collaboration should include the modelling and simulation, validation and benchmarking studies, joint small-scale experiments, data sharing from individual experiments, and joint use of international testing capabilities.

Workshop follow-up

- Technical presentations were available to the workshop participants (no public access to avoid possible misinterpretation) a few days after the workshop.
- A discussion forum was established by the NEA to allow debate among the participants even after the workshop.
- A follow-up meeting will be organised in Spring 2013. This meeting will be prepared by identifying several technical areas in a future programme of work, and potential contributors from the organisations who have attended this workshop.
- An international collaboration should be established which allows technical experts to present and discuss the various aspects of the problem in the field.

Session 1: Lessons Learned from the Fukushima Accident

Chair: K. Pasamehmetoglu (INL, US)

Current understanding of the sequence of events

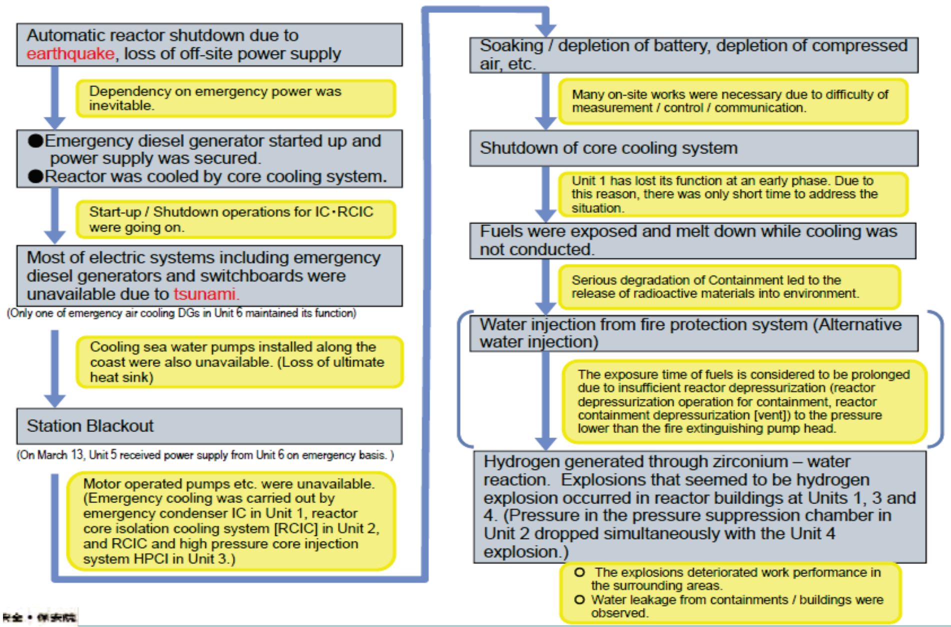
Jim Gulliford
OECD Nuclear Energy Agency

An overview of the main sequence of events, particularly the evolution of the cores in Units 1-3 was given. The presentation is based on information provided by Dr Okajima of JAEA to the June 2012 Nuclear Science Committee meeting.

During the accident, conditions at the plant were such that operators were initially unable to obtain instruments readouts from the control panel and hence could not know what condition the reactors were in. (Reactor Power, Pressure, Temperature, Water height and flow rate, etc.). Subsequently, as electrical power supplies were gradually restored more data became available. In addition to the reactor data, other information from off-site measurements and from measuring stations inside the site boundary is now available, particularly for radiation dose rates in air.

These types of information, combined with detailed knowledge of the plant design and operations history up to the time of the accident are being used to construct detailed computer models which simulate the behaviour of the reactor core, pressure vessel and containment during the accident sequence. This combination of detailed design/operating data, limited measured data during the accident and computer modelling allows us to construct a fairly clear picture of the accident progression. The main sequence of events (common to Units 1, 2 and 3) is summarised in the figure below:

Figure 1: Sequence of events for the Fukushima–Daiichi accident



The OECD/NEA is currently co-ordinating an international benchmark study of the accident at Fukushima Daiichi known as the BSAF Project. The objectives of this activity are to analyse and evaluate the accident progression and improve severe accident (SA) analysis methods and models. The project provides valuable additional (and corrected) data from plant measurements as well as an improved understanding of the role played by the fuel and cladding design.

Based on (limited) plant data and extensive modelling analysis, we have a detailed qualitative description of the Fukushima–Daiichi accident. Further analyses of the type described above are expected to provide the more quantitative descriptions needed to inform recovery plans and improve safety at other plants. The SA modelling methods being developed (and tested against the Fukushima event) should also provide the means to make quantitative assessments of the benefits which might be realised from alternative, accident-tolerant fuel designs.

Overview of fuel behaviour and core degradation, based on modelling analyses

Simone Massara
OECD Nuclear Energy Agency

Since the very first hours after the accident at Fukushima-Daiichi, numerical simulations by means of severe accident codes have been carried out, aiming at highlighting the key physical phenomena allowing a correct understanding of the sequence of events, and – on a long enough timeline – improving models and methods, in order to reduce the discrepancy between calculated and measured data. A last long-term objective is to support the future decommissioning phase.

The presentation summarises some of the available elements on the role of the fuel/cladding-water interaction, which became available only through modelling because of the absence of measured data directly related to the cladding-steam interaction. This presentation also aims at drawing some conclusions on the status of the modelling capabilities of current tools, particularly for the purpose of the foreseen application to ATF fuels:

- analyses with MELCOR, MAAP, THALES2 and RELAP5 are presented;
- input data are taken from BWR Mark-I Fukushima-Daiichi Units 1, 2 and 3, completed with operational data published by TEPCO.

In the case of missing or incomplete data or hypotheses, these are adjusted to reduce the calculation/measurement discrepancy.

The behaviour of the accident is well understood on a qualitative level (major trends on RPV pressure and water level, dry-wet and PCV pressure are well represented), allowing a certain level of confidence in the results of the analysis of the zirconium-steam reaction – which is accessible only through numerical simulations. These show an extremely fast sequence of events (here for Unit 1):

- the top of fuel is uncovered in 3 hours (after the tsunami);
- the steam line breaks at 6.5 hours.

Vessel dries at 10 hours, with a heat-up rate in a first moment driven by the decay heat only (~7 K/min) and afterwards by the chemical heat from Zr-oxidation (over 30 K/min), associated with massive hydrogen production.

It appears that the level of uncertainty increases with the progression of the accident (which is not surprising); if a good agreement among most of the numerical predictions and the plant parameter is observed in the pre-fuel melting phase, much higher uncertainties affect the post-melting phase and, ultimately, the final status of the RPV.

Finally, nearly no elements allow confirming whether the capabilities of current severe accident tools could easily be extrapolated to accident-tolerant fuels. The major efforts should be focused on:

- input data;

- adaptation and improvement of physical models;
- overall validation against experimental results;
- The development of advanced modelling techniques, including multi-scale modelling.

Key regulatory and safety issues emerging NEA activities

John Nakoski
OECD Nuclear Energy Agency

A presentation was provided on the key safety and regulatory issues and an update of activities undertaken by the NEA and its members in response to the accident at the Fukushima Daiichi nuclear power stations (NPS) on 11 March 2011. An overview of the accident sequence and the consequences was provided that identified the safety functions that were lost (electrical power, core cooling, and primary containment) that lead to units 1, 2, and 3 being in severe accident conditions with large off-site releases.

Key areas identified for which activities of the NEA and member countries are in progress include accident management; defence-in-depth; crisis communication; initiating events; operating experience; deterministic and probabilistic assessments; regulatory infrastructure; radiological protection and public health; and decontamination and recovery. For each of these areas, a brief description of the on-going and planned NEA activities was provided within the three standing technical committees of the NEA with safety and regulatory mandates (the Committee on Nuclear Regulatory Activities – CNRA, the Committee on the Safety of Nuclear Installations – CSNI, and the Committee on Radiation Protection and Public Health – CRPPH).

On-going activities of CNRA include a review of enhancement being made to the regulatory aspects for the oversight of on-site accident management strategies and processes in light of the lessons learned from the accident; providing guidance to regulators on crisis communication; and supporting the peer review of the safety assessments of risk-significant research reactor facilities in light of the accident.

Within the scope of the CSNI mandate, activities are being undertaken to better understand accident progression; characteristics of new fuel designs; and a benchmarking study of fast-running software for estimating source term under severe accident conditions to support protective measure recommendations. CSNI also has on-going work in human intervention and performance under extreme conditions; evaluations of metallic components and structures under high-seismic loads; risks assessments for natural external initiating events; and defence-in-depth, including the robustness of electrical systems. A recent joint research project has also been started that will include a benchmarking study of accident codes and the collection of data from the damaged reactors at the Fukushima Daiichi NPS.

CRPPH activities in response to the Fukushima Daiichi NPS accident include an update of report on Short-term Countermeasures in Case of a Nuclear or Radiological Emergency that was last updated in 2003; the performance of a survey on emergency management lessons learned; developing lesson learned in the management of occupational exposure in high-radiation areas; and providing support to the Japanese Government by co-ordination and participation in workshops on decontamination and recovery and other technical topics.

Recent and current activities of the OECD/NEA Working Group on Fuel Safety (NEA/CSNI)

Marc Petit
Institut de Radioprotection et de Sûreté Nucléaire (IRSN), France

The Working Group on Fuel Safety (WGFS) is part of the Committee on the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency and has the main mission of advancing the current understanding and addressing fuel safety issues.

Recent and current activities of the working group have addressed mainly the loss of coolant accident (LOCA), the reactivity initiated accident (RIA), the fuel safety criteria and leaking fuel issues, as well as Fukushima-related fuel topics.

In the area of LOCA, the group issued different documents, the most notable being a very comprehensive state of the art report [NEA/CSNI/R (2009)15].

Regarding RIA, some documents were finalised and issued in the recent years, as well as a state of the art report [NEA/CSNI/R (2010)1].

The question of leaking fuel and how it is handled in the reactors is an activity that is just starting.

Of particular interest to people developing new fuel concepts is the Nuclear Fuel Safety Criteria Technical Review – Second Edition [NEA/CSNI/R (2012)3]. This document provides a broad overview of the numerous criteria used in the NEA member countries to demonstrate to safe use of fuel in light water reactors.

The WGFS has started discussions about fuel related issues raised by the Fukushima accident, in particular, hydrogen production. New concepts have been proposed to solve these issues but it appears that these concepts will need to go through a long qualification process to assess their adequacy for the different situations considered in the evaluation of fuel safety, from normal operation to accident conditions.

Session 2: Accident-tolerant Fuel Design

Chair: K. Pasamehmetoglu (INL, US)

Overview of the accident-tolerant fuel development

Frank Goldner
Department of Energy (DOE), US

After the March 2011 events at the Fukushima Daiichi Nuclear Power Plant in Japan, enhancing the accident tolerance of light water reactor (LWRs) fuels became a heavily discussed issue. In funding for Fiscal Year 2012 the U.S. Congress provided funding and guidance to the US Department of Energy Office of Nuclear Energy (DOE-NE) to start developing nuclear fuels and claddings with enhanced accident tolerance.

Fuels with enhanced accident tolerance are those that, in comparison with the standard UO_2-zircaloy system currently used by the nuclear industry, can tolerate loss of active cooling in the reactor core for a considerably longer time period (depending on the LWR system and accident scenario) while maintaining or improving the fuel performance during normal operations, operational transients, as well as design-basis and beyond design-basis events.

Design objectives identified as potentially important to enhance accident tolerance, include: reduced hydrogen generation, improved fission product retention, improved cladding reaction to high-temperature steam, and improved fuel cladding interaction for improved performance under extreme conditions.

An overall strategy timeline for development and demonstration, shown in Figure 1, extends to 2022. This date was developed in consultation with nuclear industry and national laboratory experts, and is comprised of three phases:

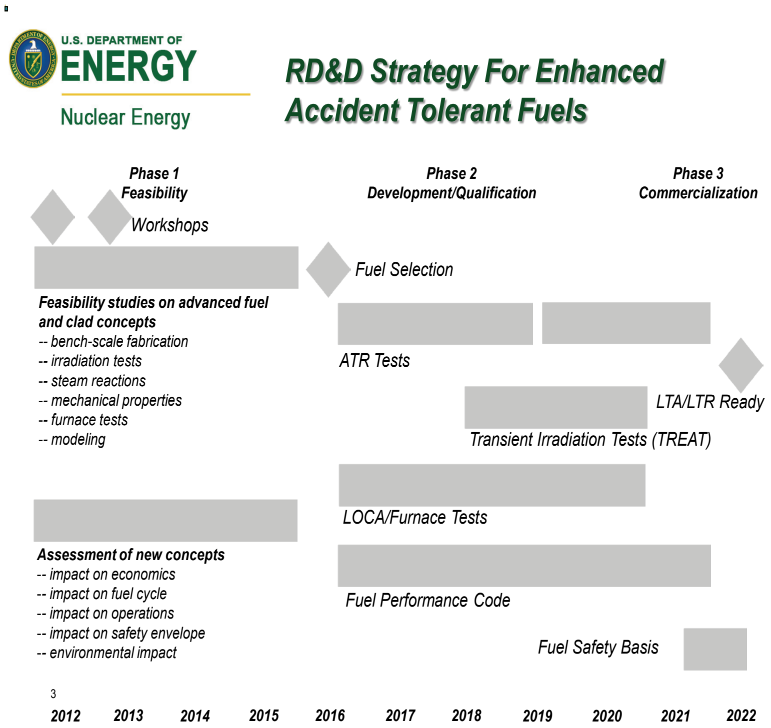
- Feasibility Assessment and Down-Selection – Fiscal Year (FY) 2012 through FY 2016.
- Development and Qualification – FY 2016 through FY 2022.
- Commercialisation – FY 2022 and beyond.

Activities performed during the feasibility assessment phase include laboratory scale experiments; fuel performance code updates; and analytical assessment of economic, operational, safety, fuel cycle, and environmental impacts of the new concepts.

The development and qualification stage will consist of fuel fabrication and large scale irradiation and safety basis testing, leading to qualification and ultimate US NRC licensing of the new fuel.

The commercialisation phase initiates technology transfer, and programme completion responsibility to industry for implementation.

Figure 1: Provisional schedule for ATF development



Path forward:

As this calendar year comes to a close we have already accomplished a good start on our programme to pursue development of LWR fuel with enhanced accident tolerance, including:

- Funding three industrial groups (Westinghouse Electric Co, LLC, General Electric Global Research, and AREVA Federal Services, LLC) to develop fuels that can enhance the ability to withstand extreme conditions and limit damage in the unlikely event that a reactor loses coolant and temperatures rise above their normal operating levels.
- Funding university led groups to dealing with advanced light water reactor design related activities with inherently safety features and/or advanced materials for fuel cladding concepts that would enhance accident tolerance of the nuclear fuel system.
- Holding "metric" related meetings in the US and France to engage with international parties to discuss the state of existing accident-tolerant related activities and potential future programmes of work.

Further co-operation and partnerships stimulated by actions initiated at this meeting are needed.

Evaluation of potential alternative nuclear fuel cladding materials for LWRs applications with increased margins in LOCA and beyond LOCA conditions at CEA

Marion Le Flem, Aurore Michaux, Cédric Sauder, Jean-Christophe Brachet
Commissariat à l'énergie atomique et aux énergies alternatives (CEA), France

The Fukushima accident has highlighted issues concerning fuel cladding resistance at high temperatures: in the event of loss of cooling water (LOCA), a temperature increase over 800°C results in rapid high-temperature oxidation of the zirconium alloy cladding material, leading to a significant embrittlement (risk of fuel containment rupture), a large release of hydrogen (explosion risk) and even core melt. Reducing susceptibility of cladding to high-temperature oxidation in steam can limit and even eliminate these risks.

Significant improvement in the safety of light water reactors (LWRs) is possible with the use of other materials that are more robust than those currently serving as the first fuel containment barrier. Based on the large R&D feedback throughout extensive CEA/EDF/AREVA collaborative studies on the behaviour of the current cladding materials in LOCA conditions, CEA is assessing two potential alternative solutions using innovative materials that have already received initial operating experience feedback from very recent preliminary research performed at the CEA.

- The follow-on solution consists of protecting current zirconium alloy cladding from oxidation with a sufficient, robust thin coating. The optimisations led this last seven years on flat specimens have resulted in the selection of very promising coatings: oxidation tests in autoclaves (360°C) highlighted a significant decrease in the weight gain (i.e. for nominal in-service conditions), oxidation at 1000-1200°C in steam (for prototypical LOCA oxidation times) resulted in limited oxidation of the coating only, without significant oxygen diffusion in the Zr alloy substrate (no ZrO_2 formation, no sub-oxide α_{Zr}(O) layer growth), and the post-quench ductility is enhanced (bending and tensile testing). Besides, the transposition of the coating process to tubular geometry has been recently demonstrated. Additionnal further improvements are planed in terms of composition, adhesion, microstructure and the fabrication of long objects must be done.
- Another direction for the R&D, which is less conventional, relies on heat-resistant cladding made of silicon carbide (SiC/SiC) composites. Such a solution, first driven by R&D on gas cooled reactors should enable cladding to reach very high temperatures (approximately 2 000°C) without significant oxidation or hydrogen production. The dimensional stability of the material also ensures a coolable and then reliable core geometry. The "sandwich" design patented by CEA includes a metal layer, which ensures leaktightness, inserted between two layers of SiC/SiC composites that provide overall mechanical resistance. The transposition to LWR is suitable and achievable with the present manufacturing process. The preliminary oxidation tests at 1 200-1 400°C in moisture atmosphere have highlighted the superiority of the SiC/SiC material: at 1 200°C, the Zr cladding was completely destroyed by oxidation and cracking after 1h whereas the SiC/SiC tube

was barely affected by oxidation after 110 h and kept the starting dimensions. Tensile test after oxidation confirmed the very good behaviour of the composite.

The next studies on these two concepts will cover research (selection and optimisation of materials) as well as prototypes manufacturing (process configuration and facility adaptation), experimental development (new analytical devices) and tests that reproduce conditions found in pressurised water reactors (PWRs) to demonstrate the benefit of such designs in the event of an emergency as well as during normal operating conditions. The targets are to:

- accurately assess the reduction of hydrogen production in accidental conditions;
- assess the residual robustness of the claddings after oxidation at high temperatures;
- specify improvement in terms of melting temperature of the cladding+fuel combination;
- estimate the production cost;
- assess the new designed clads behaviour in representative in-reactor conditions (irradiation).

Silicon carbide materials for LWR application: current status and issues

Tatsuya Hinoki
Institute of Advanced Energy, Kyoto University, Japan

Silicon carbide (SiC) is a very attractive engineering ceramic in particular for high-temperature use and nuclear application due to its high-temperature strength, oxygen resistance, chemical stability, low activation, radiation resistance, etc. Silicon carbide composites have pseudo ductile behaviour by debonding and sliding at fiber/matrix interphase. Fundamental mechanical properties of highly crystalline nuclear grade SiC composites are stable following neutron irradiation. Silicon carbide composites are promising materials for accident-tolerant fuel.

The sophistication of the technology infrastructure for safety has been requested by the Ministry of Economy, Trade and Industry (METI) in Japan. The research and development of fuel such as SiC cladding are expected to be described in a new road map by METI. Silicon carbide is a promising material for LWR application in terms of excellent stability of dimension and strength under neutron irradiation and excellent resistance to high-temperature steam. Fundamental fabrication technique and joining technique have been established. Current SiC/SiC composites have C interphase and environmental coating is required to prevent oxidation. Novel porous SiC/SiC composites do not have C interphase and have excellent oxidation resistance, although hermetic coating is required.

The issues of SiC composite development for LWR application are as follows: The SiC/SiC composites have impurities depending on fabrication methods. It is important to understand the effect of impurities on the resistance to high-temperature water under normal operation and the resistance to high-temperature steam in the case of severe accident. The synergetic effect of irradiation and high-temperature water is also important. The reaction with fuel under neutron irradiation needs to be clarified. As for material development, coating, joining technique and large scale fabrication should be considered as important issues. Material cost should be reduced extremely.

Development of advanced claddings for suppressing the hydrogen emission in accident conditions

Jeong-Yong Park
Korea Atomic Energy Research Institute (KAERI), Republic of Korea

The development of accident-tolerant fuels can be a breakthrough to help solve the challenge facing nuclear fuels. One of the goals to be reached with accident-tolerant fuels is to reduce the hydrogen emission in the accident condition by improving the high-temperature oxidation resistance of claddings. KAERI launched a new project to develop the accident-tolerant fuel claddings with the primary objective to suppress the hydrogen emission even in severe accident conditions. Two concepts are now being considered as hydrogen-suppressed cladding. In concept 1, the surface modification technique was used to improve the oxidation resistance of Zr claddings. Like in concept 2, the metal-ceramic hybrid cladding which has a ceramic composite layer between the Zr inner layer and the outer surface coating is being developed.

The high-temperature steam oxidation behaviour was investigated for several candidate materials for the surface modification of Zr claddings. From the oxidation tests carried out in 1 200°C steam, it was found that the high-temperature steam oxidation resistance of Cr and Si was much higher than that of zircaloy-4. Al_3Ti-based alloys also showed extremely low-oxidation rate compared to zircaloy-4. One important part in the surface modification is to develop the surface coating technology where the optimum process needs to be established depending on the surface layer materials. Several candidate materials were coated on the Zr alloy specimens by a laser beam scanning (LBS), a plasma spray (PS) and a PS followed by LBS and subject to the high-temperature steam oxidation test. It was found that Cr and Si coating layers were effective in protecting Zr-alloys from the oxidation. The corrosion behaviour of the candidate materials in normal reactor operation condition such as 360°C water will be investigated after the screening test in the high-temperature steam.

The metal-ceramic hybrid cladding consisted of three major parts; a Zr liner, a ceramic composite layer and a surface coating layer.

- The Zr liner is not a main concern in the hybrid cladding, though the manufacturing process needs to be modified to adapt the design change.
- The ceramic composite layer has to be developed to meet the requirements of the fuel cladding. SiC composite was selected as a primary candidate. The filament winding process is being developed in collaboration with the specialised industry using Tyranno-S fiber. The polymer impregnation and pyrolysis (PIP) process was investigated for a matrix impregnation method for the SiC composite layer of hybrid cladding. The manufacturing technology for SiC composite layer will be optimised by changing the variables in each step of PIP process.
- The surface coating layer acting as a corrosion or oxidation barrier is being developed using a sol-gel process, which can make the overall manufacturing process simpler since the sol-gel coating is basically the same process with the PIP.

The joining technology for SiC was also developed, although the joining of SiC was not necessary in the hybrid cladding using Zr liner. Many types of interlayer such as Ti, Zr and Mo were inserted among the SiC specimens and then a laser welding was carried out to join the SiC specimens. The joining technology for SiC will be improved by changing the interlayer types and the laser welding condition.

Accident-tolerant control rod

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Central Research Institute of Electric Power Industry (CRIEPI), Japan

Boron carbide (B_4C) and hafnium (Hf) metal are used for the neutron absorber materials of control rods in BWRs, and silver-indium-cadmium (Ag-In-Cd) alloy is used in PWRs. These materials are clad with stainless steel. The eutectic point of B_4C and iron (Fe) is about 1 150°C and the melting point of Ag-In-Cd alloy is about 800°C, which are lower than the temperature of zircaloy – steam reaction increases rapidly (~1 200°C). Accordingly, it is possible that the control rods melt and collapse before the reactor core is significantly damaged in the case of severe accidents. Since the neutron absorber would be separated from the fuels, there is a risk of re-criticality, when pure water or seawater is injected for emergency cooling.

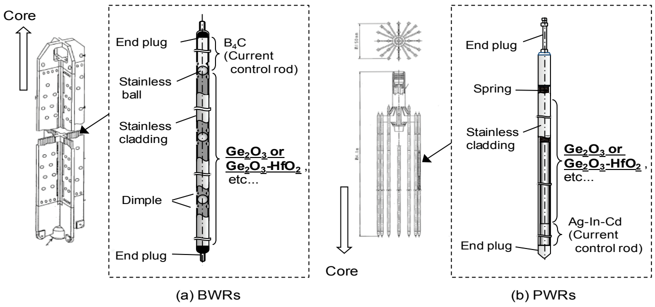
In order to ensure sub-criticality and extend options of emergency cooling in the course of severe accidents, a concept of accident-tolerant control rod (ACT) has been derived. ACT utilises a new absorber material having the following properties:

- higher neutron absorption than current control rod;
- higher melting or eutectic temperature than 1 200°C where rapid zircaloy oxidation occurs;
- high miscibility with molten fuel materials.

The candidate of a new absorber material for ATC includes gadolinia (Gd_2O_3), samaria (Sm_2O_3), europia (Eu_2O_3), dysprosia (Dy_2O_3), hafnia (HfO_2). The melting point of these materials and the liquefaction temperature with Fe are higher than the rapid zircaloy oxidation temperature. ACT will not collapse before the core melt-down. After the core melt-down, the absorber material will be mixed with molten fuel material.

The current absorber materials, such as B_4C, Hf and Ag-In-Cd, are charged at the tip of ATC in which the neutron flux is high, and a new absorber material is charged in the low-flux region, as shown in Figure 1. This design could minimise the degradation of a new absorber material by the neutron absorption and the influence of ATC deployment on reactor control procedure. As a result, the new absorber material may be recycled.

Figure 1: Design of ACT, (a) BWR type and (b) PWR type



Fully ceramic micro-encapsulated fuel design and irradiation testing

Lance Snead
Oak Ridge National Laboratory (ORNL), US

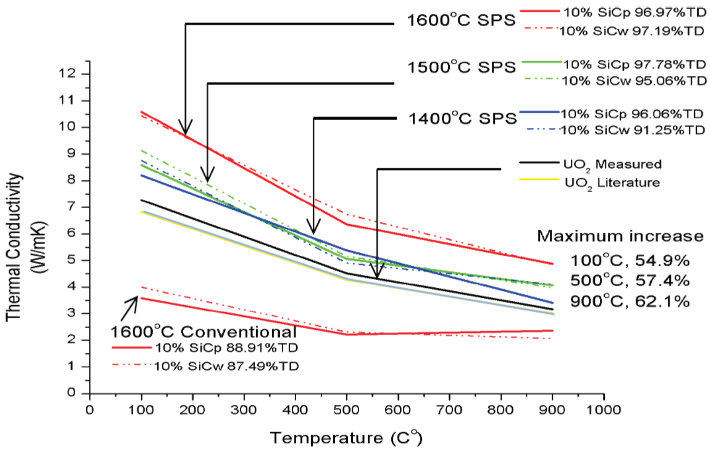
Fully ceramic microencapsulated fuel design and irradiation testing light water reactor fuels consisting of a silicon carbide matrix and specifically engineered TRistructual ISOtropic (TRISO) particles are currently under development and analysis as part of the DOE Fuel Cycle R&D Programme. This concept is specifically referred to as the fully ceramic microencapsulated (FCM) fuel form. As part of this effort SiC matrix processing (compaction) optimisation, uranium nitride (UN) kernel fabrication, and irradiation studies are underway. This report described the development programme and the irradiation campaign, and post-irradiation examination campaign was carried out to determine the stability of the SiC matrix and surrogate fuel form. A series of surrogate FCM fuel pellets with TRISO fuel particles (zirconia substituted for the fissile containing kernel) were fabricated and irradiated at ORNL's High Flux Isotope Reactor to examine SiC matrix irradiation behaviour. The pellets were clad in PWR 17×17 zircaloy-4 and irradiated at 425°C. By selecting the zircaloy cladding this experiment also provides insight into any potential fuel-clad interaction. Four capsules, each containing 5 pellets, were irradiated up to ~8 dpa (SiC matrix, or about 8x1 025 n/m^2 E>0.1 MeV) at fluence intervals of ~2 dpa. TRISO volume loadings up to 41% were studied. Of note is no apparent cracking, deformation, discoloration, or pellet clad interaction (standard microscopy inspection of both surfaces) was observed. Thermal conductivity, which has been optimised pre-irradiation and is a significant design attribute of this fuel, will be reported on post-irradiation and compared with existing algorithms for irradiation-induced degradation. Similarly, swelling of the compact will be compared with the existing database for CVD SiC and SiC composite-based materials. Also of note was the lack of interaction between the fuel matrix and zircaloy holder suggesting the potential for nil fuel clad interaction in the final fuel form.

Development of an enhanced accident-tolerant composite UO_2 fuel pellet with a dopant of SiC whiskers

James Tulenko, Ghatu Subhash
University of Florida, US

The University of Florida research sponsored by the Department of Energy has shown beneficial performance characteristics for UO_2 –10 vol% SiC composite pellets with thermal conductivity improvements exceeding 50% at 900°C. The thermal conductivity results to date, shown in Figure 1, have been extremely encouraging, where we have noted significant improvements in the thermal conductivity by almost 55% at 100°C and by 62% at 900°C, when compared to theoretical values in literature for UO_2.

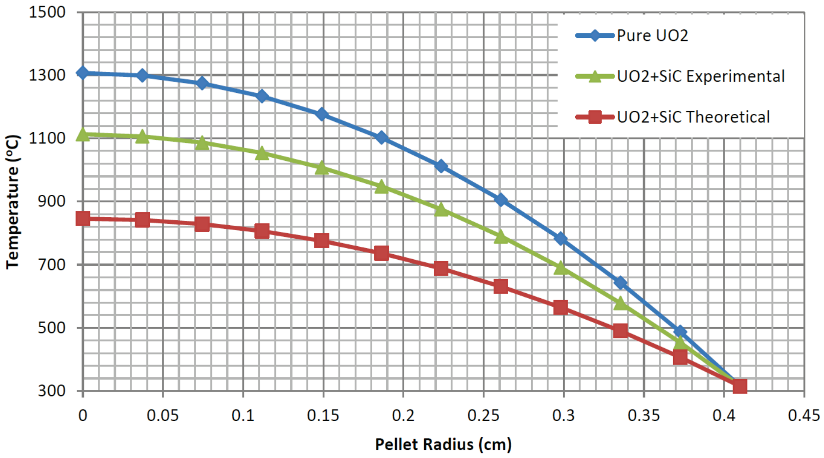
Figure 1: Thermal conductivity values of UO2 +SiC composite pellets



The maximum increase refers to the increase in SPS sintered UO_2+SiC pellets at 1 600°C compared to the theoretical UO_2 values in literature at various temperatures.

The reduced thermal expansion, reduced thermal cracking and reduced fission gas releases resulting from the increased pellet thermal conductivity are expected to produce an enhanced accident-tolerant fuel. According to the rule of mixtures, the addition of silicon carbide should show a greater increase in the thermal conductivity of the pellets. This initial inability to reach the thermal conductivity values projected by the rule of mixtures is believed by the University of Florida to result from defects (voids, cracks, debonding, etc.) in the microstructure of the composite pellet resulting from differences in the thermal expansion coefficients during the cool down after the sintering process. The University of Florida is researching the development/production of a defect free composite UO_2 pellet which will allow for the full benefit by the rule of mixtures of composite properties. Figure 2 shows the projected pellet temperatures in a fuel pellet at 10 kilowatts per foot power for UO_2, the as measured UO_2-SiC thermal conductity with defects and the rule of mixtures defect free projected UO_2SiC thermal conductivity.

Figure 2: Projected fuel pellet temperatures as a function of pellet radius at 0 GWd/MTU burn-up –10 kW/ft (32.8 kW/m)



The University of Florida will be preparing composite UO_2-SiC whiskered pellets with AREVA for irradiation in a research reactor to validate the enhanced thermal conductivity performance under irradiation conditions.

Enhanced accident-tolerant fuel

John Strumpell
AREVA-NP Inc., US

The Fukushima accident provided a strong reminder that the exothermic reaction between zirconium and steam, and the attendant hydrogen generation, can significantly affect the course of a severe accident. Part of the response to the accident was increased interest in the extent to which the fuel itself can mitigate the consequences of a severe accident. Improved fuel alone is not sufficient to provide the desired increase in reactor safety, but it can provide an important contribution.

With support from the US Department of Energy, AREVA has brought together a team that includes researchers (AREVA, Electric Power Research Institute, Savannah River National Laboratory, University of Florida, and University of Wisconsin), a fuel vendor (AREVA), and utilities (Duke Energy and Tennessee Valley Authority). The goal of the project is to develop new technologies that can be deployed in a lead assembly within ten years. The researchers have proposed a variety of approaches for improving the performance of the fuel, including new cladding and structural materials, fuel pellets with improved thermal characteristics, and coatings on the fuel rods. The expected performance of fuels that apply these technologies will be judged against the requirements of the vendor and utilities to determine those that are most promising for immediate development and those that may be suited for development in the future. The first review will consider the manufacturability of the proposed designs; the second will focus on performance. Materials that are suitable for immediate development will be considered for irradiation in a test reactor and subsequent use in lead assembly designs.

Advanced fuel technologies at General Atomics

Christina A. Back
General Atomic (GA), US

General Atomics (GA) has made significant contributions since its founding in the 1950s to develop nuclear power for peaceful means. With the conception and construction of the TRIGA reactors and research on TRISO particles, GA has long recognised the importance of "accident-tolerant" materials.

Before the accident at Fukushima Daiichi, GA had already initiated work on silicon carbide (SiC) and SiC-related technologies for application in nuclear reactors. At that time, the work was initiated in support of the GA advanced gas-cooled fast reactor concept called the Energy Multiplier Module, EM^2. This work continues, however, the reasons that make SiC materials attractive for fast reactor concepts also make them attractive for advanced light water reactors. These include superior performance over zircaloy for high-temperature strength, especially above 1 500°C, and significantly reduced hydrogen production in accident scenarios.

The current focus on "accident-tolerant" components is to develop cladding made of silicon carbide fiber and silicon carbide matrix, SiC-SiC composites. The goal for this work is to produce a cladding that provides strength and impermeability to meet reactor performance and safety requirements. To date, GA has examined the trade-offs between processing time and infiltration uniformity to reduce fabrication time, fabricated cylindrical prototypes, and refined material properties for fracture toughness, impermeability, and thermal conductivity.

Generally, the GA programme is developing innovative fuel elements that employ both high density uranium-bearing fuels that enable longer lifetime with higher burn-up, and claddings that are more resistant to neutron damage. In addition to fabrication, significant effort is devoted to measuring the critical parameters, such as thermal conductivity, mechanical strength and component performance at reactor-relevant operational conditions, using a mix of commercial equipment, customised in-house test rigs, and specialised fixtures. Furthermore, GA strives to iteratively refine models and simulations with benchtop experimental data to accelerate process development and optimise component design. Throughout the programme, GA maintains active collaborations with industry, universities, and national laboratories.

This work has been supported by General Atomics internal funding.

Suggestion of a novel failure tolerant fuel element

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Paul Scherrer Institute (PSI), Switzerland

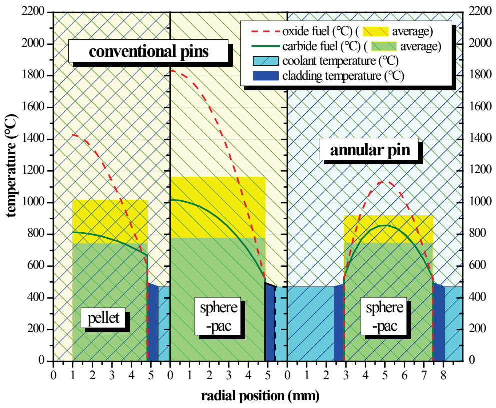
Introduction

A new internally cooled pin design is proposed as being very similar to the concept suggested by MIT [1] and KAERI [2], where an additional concentric inner cladding tube provides an internal cooling channel. The major difference to the already proposed concepts is the fuel type, which is proposed to be a sphere-pac fuel, which, similarly to vibropack, proposed in [1], is a particle fuel. Because of the much reduced maximal distance from the fuel to the cladding a better heat transfer towards the cladding is expected. Additionally, the increased cooling surface to fuel volume ratio provides a heat transfer through the cladding. This becomes especially important for new materials considered for the accident tolerance, which after irradiation can show an important reduction in conductivity (e.g. SiC composite).

Calculation and preliminary evaluation

The thermal conductivity, the thermal hydraulics and the neutronic aspect of a sphere-pac filled annular pin design has already been studied for a fast reactor design [3], and a new method of producing the fuel kernels by microwave internal gelation was studied in the framework of a Swiss national research programme MeAWaT [4].

Figure 1: Innovative annular pin design characteristics



The graph shows the temperature calculation for a fast reactor according to the ESFR design for a standard pellet and sphere-pac pin with a linear heat rate of 293 W·cm^{-1} (two left temperature profiles) and an annular design with a linear heat rate 626 W·cm-1 (higher because of increased cross-sections, temperature profile on the right-hand side). Both oxide and carbide fuel are calculated (for details see [3]). The annular design shows for the oxide fuel (red dotted lines and yellow boxes) an important temperature reduction, the maximal and average temperatures are reduced by about 300°C and 100°C, respectively. For the hereinafter discussed LWR fuel, a similarly better heat transfer is to be expected. Of relevance for an accident scenario is the reduced stored energy in the system. This is, however, a minor contribution and will not greatly influence the accident development. More important here is the better heat transfer through the cladding, together with the geometrically enhanced component stability, with its double structure and the increased diameter. These factors are contributions to the usability of novel, failure tolerant, cladding materials such as SiC.

Conclusions and remarks

An annular pin design with particle fuel is suggested, which on the one hand, provides better heat transfer in the fuel and the cladding section, and on the other hand, promises better mechanical stability. Both are factors which can promote novel materials which are especially designed to withstand high temperatures, but have lower thermal conductivity and less favourable mechanical properties (e.g. fracture toughness).

Besides, the fuel concept, SiC related activities at PSI were presented at the meeting. These are different analysis and testing methods, taking advantage of synchrotron light, the spallation source and the mechanical testing devices in PSI's Hotlab [5] [6].

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Engineered zircaloy cladding modifications for improved accident tolerance of LWR fuel: US DOE NEUP Integrated Research Project

Brent Heuser
University of Illinois, US

An integrated research project (IRP) to fabricate and evaluate modified zircaloy LWR cladding under normal BWR/PWR operation and off-normal events has been funded by the US DOE. The IRP involves three US academic institutions, a US national laboratory, an intermediate stock industrial cladding supplier, and an international academic institution. A combination of computational and experimental protocols will be employed to design and test modified zircaloy cladding with respect to corrosion and accelerated oxide growth, the former associated with normal operation, the latter associated with steam exposure during loss of coolant accidents (LOCAs) and low-pressure core re-floods. Efforts will be made to go beyond design-base accident (DBA) scenarios (cladding temperature equal to or less than 1 204^{0}C) during the experimental phase of modified zircaloy performance characterisation. The project anticipates the use of the facilities at ORNL to achieve steam exposure beyond DBA scenarios. In addition, irradiation of down-selected modified cladding candidates in the ATR may be performed. Cladding performance evaluation will be incorporated into a reactor system modelling effort of fuel performance, neutronics, and thermal hydraulics, thereby providing a holistic approach to accident-tolerant nuclear fuel. The proposed IRP brings together personnel, facilities, and capabilities across a wide range of technical areas relevant to the study of modified nuclear fuel and LWR performance during normal operation and off-normal scenarios.

Two pathways towards accident-tolerant LWR fuel are envisioned, both based on the modification of existing zircaloy cladding. The first is the modification of the cladding surface by the application of a coating layer designed to shift the M+O→MO reaction away from oxide growth during steam exposure at elevated temperatures. This pathway is referred to as the "surface coating" solution. The second is the modification of the bulk cladding composition to promote precipitation of minor phase(s) during fabrication. These precipitates will be stable under normal operation, but dissolve during the temperature excursions; the migration of solute elements to the free surface will then shift the reaction away from oxide formation. This pathway is referred to as the "bulk self-healing" solution. A synergistic response of the fuel rod is anticipated in which the combined mitigation of brittle *exothermic* oxide formation and associated reduction in cladding temperature lead to accident tolerance with respect to cladding failure. The proposed cladding modifications potentially may influence neutronics and thermal hydraulics, both under normal operation and off-normal scenarios; a favourable reactor system response must therefore be demonstrated for both solution pathways.

The objectives of the proposed IRP is four-fold:

1) demonstration of the performance of modified cladding material under normal BWR and PWR operation with respect to corrosion, in particular, stress corrosion cracking (SCC) and irradiation-assisted stress corrosion cracking (IASCC);

2) the mitigation of accelerated cladding oxidation during off-normal scenarios that fall below unchecked LOCA events, as well as uncovering scenarios that involve used fuel in on-site storage pools;

3) the benchmarking of the fuel performance code against the databases developed in 1 and 2;

4) demonstration of overall reactor system performance with the proposed modifications to the pellet and cladding.

Session 3: Reactor Operation, Safety, Fuel Cycle Constraints, Economics and Licensing

Chair: K. Pasamehmetoglu (INL, US)

Preliminary assessment of the impact of candidate accident-tolerant fuels/cladding on the predicted reactor behaviour at normal operating conditions and under DB (LOCA and RIA) and BDB (STSBO AND LTSBO) accident conditions

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Currently, the United States Department of Energy (DOE) has initiated the study of advanced accident-tolerant fuel/cladding (ATF) configurations that exhibit 1) slower reaction kinetics with steam, 2) lower enthalpy of oxidation, 3) less susceptibility to unfavourable core material interactions, and 4) provision of additional barriers to fission product release. Whenever changes, whether minor or major, are made to commercial NPP fuel/cladding systems; then the effect of these changes must be evaluated on all phases of the fuel/cladding lifetime (from fabrication through operation through eventual storage and reprocessing). This presentation focuses on preliminary assessments of several potential ATFs on the impact of these materials on predicted reactor behaviour 1) at normal operating conditions, 2) under postulated design basis (DB) accidents (LOCAs and RIAs), and 3) under beyond design basis (BDB) accident conditions [for short- and long-term station blackouts(SBO)]. These preliminary reactor response predictions are compared against the responses of UO_2/Zr cores. For the ATFs evaluated, during normal operation, the most significant features are much lower fuel centerline temperatures and fission gas releases; and for LOCAs the peak cladding temperatures are lower with significantly lower hydrogen generation rates and for a RIA the ATF ejected worth is very similar to the UO_2 ejected worth. The use of higher melting/lower hydrogen producing core components (ATFs) will not preclude a BDB accident. Without core cooling the severe accident will march-on; however, the ATFs do allow an increase in margin (time) to initiation of core component degradation – although this may be measured in minutes rather than hours. The ATF core responses (with oxidation kinetics about two orders of magnitude lower than that for Zr) are nearly the same as for components with no oxidation (for a STSBO, the increased time to vessel dry-out is approximately 4.5 hours). There is a need to consider additional materials interaction experiments and component interactions with steam; and, besides the ATFs, considerations must be given to other components within the core (ergo, the control elements and their housings) and to components above and below the core. Finally, it must be remembered that the reactor safety is determined by the system performance which includes the fuel/cladding combination as well as the ECCS and operator actions.

Fuel behaviour in the case of severe accidents and potential ATF designs

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This presentation reviews the conditions of fuel rods under severe loss of coolant conditions, approaches that may increase coping time for plant operators to recover, requirements of advanced fuel cladding to increase tolerance in accident conditions, potential candidate alloys for accident-tolerant fuel cladding and a novel design of molybdenum (Mo) -based fuel cladding. The current Zr-alloy fuel cladding will lose all its mechanical strength at 750-800°C, and will react rapidly with high-pressure steam, producing significant hydrogen and exothermic heat at 700-1 000°C. The metallurgical properties of Zr make it unlikely that modifications of the Zr-alloy will improve the behaviour of Zr-alloys at temperatures relevant to severe accidents. The Mo-based fuel cladding is designed to (1) maintain fuel rod integrity, and reduce the release rate of hydrogen and exothermic heat in accident conditions at 1 200-1 500°C. The EPRI research has thus far completed the design concepts, demonstration of feasibility of producing very thin wall (0.2 mm) Mo tubes. The feasibility of depositing a protective coating using various techniques has also been demonstrated. Demonstration of forming composite Mo-based cladding via mechanical reduction has been planned.

Safety of some fuel cladding materials, alternative to Zr-alloys

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Introduction

The Fukushima accident underlined the impact of hydrogen production on LWR core melt accident behaviour. New fuel cladding and structural materials are under development by the industry. IRSN performed a bibliographic study on the behaviour of these materials during LWR core melt accidents.

Method

This presentation is focused on cladding oxidation by steam and more precisely on:

- number of H_2 moles produced per cladding length unit at thermochemical equilibrium;
- oxidation kinetics;
- heat of reaction;
- physic-chemical interactions between material or oxidation products and fuel.

Silicon carbide (SiC)

- During SiC oxidation by steam, nearly 3 times more explosive gases ($CO+H_2$) moles are produced per cladding length unit (dimensions from [1]) at thermochemical equilibrium than for Zr-alloys.
- SiC oxidation kinetics below 1 700°C: According to early tests performed by NASA [2] and ORNL [3], the oxidation is linear but slow, there is an effective protection by a thin vitreous SiO_2 layer; these tests underlined the importance of the steam pressure and flow rate. Recently, published MIT [4] & ORNL tests [5] confirm that under large break LOCA conditions (~5 bars [6]) and up to 1 200°C, SiC recession is much slower than for Zr-alloys. Tests under small break conditions (3 inches LOCA: ~40 bars [7]) were not performed or not published.
- SiC oxidation kinetics above 1 700°C (melting point of SiO_2): Molten SiO_2 loses its protective effect; this is known in the literature as “catastrophic oxidation by molten oxides” [8]. There will be a cliff-edge effect. For un-inerted containments, H_2 recombiners will be saturated, leading to a risk of $CO+H_2$ explosion in these containments.
- During SiC oxidation by steam, the heat of reaction produced per cladding length unit at thermochemical equilibrium is of the same order of magnitude as for Zr-alloys.

- Molten SiO_2 will interact with UO_2 to form molten mixtures at temperatures well below UO_2 melting temperature [9].
- Calculations were published by Westinghouse and Fauske with MAAP code, assuming zero kinetics and no physicochemical interactions [10]. SiC is probably tolerant to DBAs and short duration severe accidents (TMI-2 like) but more intolerant than Zr-alloys to long-duration severe accidents (Fukushima Daiichi unit 1 like).

Stainless steels (SS)

- During SS oxidation by steam, nearly the same number of H_2 moles are produced per cladding length unit at thermochemical equilibrium than for Zr-alloys.
- SS oxidation kinetics below 1 600°C: According to early tests performed by GE, 304L oxidises faster than Zr-alloys above 1 150°C [11]. Recent ORNL tests [5] show that Fe-25 Cr or PM2000 (19% Cr, 5% Al) are similar to SiC at 1 200°C.
- SS oxidation kinetics above 1 600°C (melting point of FeO_x): In molten FeO_x, oxygen diffusion is very fast; this is known in the literature as "catastrophic oxidation by molten oxides" [8]. There will be a cliff-edge effect. For un-inerted containments, H_2 recombiners will be saturated, leading to a risk of H_2 explosion in these containments:
  - The main advantage of SS is the lower heat of reaction produced per cladding length unit at thermochemical equilibrium compared to Zr-alloys.
  - Molten FeO_x will interact with UO_2 to form molten mixtures at temperatures well below UO_2 melting temperature [12].
  - Calculations with system codes (ASTEC, MELCOR, MAAP) are necessary and were published by Westinghouse and Fauske with MAAP code, assuming no catastrophic oxidation above 1 600°C and no physicochemical interactions [10]. 304L is intolerant to long-duration severe accidents (extended station blackout).

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Steam attack studies in SiC clad and advanced steel clad

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The most recent generation of zirconium alloys used as nuclear fuel cladding in LWRs offer exceptional performance under normal operating conditions with failure rates below 1 ppm. This level of performance is due to six decades of active research and development by nuclear operators, vendors, government laboratories, and academia. During this same period, iron alloys, which were successfully used in the early days of nuclear power but supplanted by zirconium alloys, have undergone broad-based development including specialty specialised steels for high-temperature corrosion and steam resistance. This coupled with the recent trends in fuel handling, reactor operation and the extreme control of coolant chemistry begs for re-examination of the application of advanced iron-alloys as LWR fuel cladding, specifically, oxidation resistant iron alloys that offer large margins of safety under severe accident conditions.

Recently, steam oxidation tests have been performed on a wide range of austenitic and ferritic iron alloys at ORNL where the dependence of the oxidation kinetics on alloy chemistry has been determined. As the temperature increases the minimum Cr content in the ferritic and austenitic alloys needs to be increased for oxidation resistance. At temperatures ≥1 200°C ~25% Cr content is necessary for adequate oxidation resistance. At the fixed Cr content, an increase in Ni content in the austenitic alloys enhances the oxidation resistance. Note that the 25% minimum Cr content disqualifies the conventional austenitic (18Cr-8Ni) alloys such as 304L as oxidation resistant materials in steam. The most promising alloys are ferritic alloys that contain some Al in addition to Cr. FeCrAl alloys with 20% Cr and 5% Al exhibit exceptionally low oxidation in high-temperature steam up to 1 300°C due to formation of a protective alumina film on the surface of the material. High-pressure steam oxidation tests (up to 2 MPa) were also performed that showed the effect of steam partial pressure is important for alloys with poor oxidation resistance. Finally, no effect from the presence of hydrogen gas (up to partial pressure of 0.5) was observed on oxidation behaviour of iron-based alloys in high-temperature steam.

Contributing to the design of accident tolerant fuels by applying the TRANSURANUS fuel performance code

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The TRANSURANUS fuel performance code is used by safety authorities, industry, research centre and universities in the EU and across the globe. Accordingly, only a very brief overview was provided about the structure of the code and the corresponding input requirements before summarising the needs for simulating new cladding materials such as those considered in the framework of the workshop by means of TRANSURANUS. Two concrete examples were then provided. The first deals with the implementation of the material properties from the CEA for SiC based cladding in the frame of the GoFASTR Project, which is funded by the EU. The second deals with material properties for T91, which have been implemented in the framework of a collaboration agreement between Politecnico di Milano and JRC-ITU. In order to illustrate the impact of replacing one cladding by another, an example irradiation has been selected, and some of the relevant cladding properties shown as a function of irradiation time when considering SiC-based cladding, T91 cladding, compared to standard zircaloy and stainless steel cladding.

Finally, it was pointed out that despite the fact that the fuel performance codes may be very useful for the current scoping studies based on available material properties, there are limitations in terms of material properties under representative irradiation conditions, or in terms of representativeness for heterogeneous and anistropic materials such as Complex Matrix Composite cladding materials (e.g. the so-called sandwiched SiC_f-SiC material). More experimental data are therefore required for more refined and reliable predictions.

In-reactor testing capabilities at Halden relevant for ATF fuel

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The development of the new accident-tolerant fuel for LWRs is a long-term endeavour, which will require a very extensive selection and validation process consisting of:

- laboratory assessments;
- out-of reactor testing;
- in-reactor testing at representative nuclear, thermal-hydraulic and water chemistry conditions.

While new fuel compositions might also be considered, it is expected that the focus will be on new cladding materials, mainly because the cladding is the main barrier to radioactive release and because it constitutes the fuel rod overall structural support. The cladding reaction with steam at high temperatures as can occur in beyond design basis LOCA conditions also constitutes a weakness for the cladding materials currently used in the industry.

The report is intended to outline the types of in-reactor testing that are currently required when the nuclear industry proceeds in the development of new fuel types or materials, and that are likely to be needed also in the development of ATF fuel. The existing experience at Halden and at other laboratories constitutes the basis for the above.

The presentation focuses on in-reactor testing and in particular on the following items:

- in-reactor material property assessments;

The tests considered under this item can normally be carried out with small size specimens, as well as with one or more fuel rods suitably designed for the purpose. These tests should be carried out at representative LWR conditions or at somewhat more demanding conditions in terms of radiation, temperature or water chemistry environment. Typical tests are:

  - stress corrosion cracking susceptibility;
  - corrosion and irradiation growth;
  - irradiation creep and stress relaxation;

- rod integrity in normal operation;

These tests are carried out with fuel rod segment, and follow the same procedure used today for validating modified fuel designs. The tests are normally performed at the upper envelope of the normal operating conditions, or at somewhat more demanding conditions. Typical tests discussed in the presentation are:

  - integrity of un-fuelled rods (for e.g. first SiC cladding qualification);

  - integrity of fuel rods, normal operation;
  - PCI/PCMI margin in power ramps.
- rod behaviour in transients;

Currently, only reactivity initiated accident (RIA) and design basis (DB) LOCA tests are considered for new fuel developments. However, it is expected that ATF demonstrations will also require studies in beyond-design basis (BDB) conditions. Typical tests addressed in the presentation are:

  - RIA transients;
  - Design Basis LOCA;
  - Beyond Design Basis LOCA (severe accident scenarios).

Introducing a new type of fuel in EDF reactors: Main issues

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The main issues associated with the introduction of a new fuel design at an industrial level in EDF reactors are the following: in-core operational feedback, coherence of the nuclear fuel cycle and mixed cores.

In-core operational feedback

Once all possible experimental results have been obtained for a new fuel design, in-core qualification is necessary to assess its robustness and reliability prior to an industrial use.

That qualification involves several steps to gather the relevant operational feedback at minimum risk:

- irradiation of a few LTA's during a number of operating cycles greater than what would be needed on an industrial scale, with specific exams and rod extractions at the end of each fuel cycle: that first step is 5 to 6 years long;
- loading whole reloads with the new fuel design in a dedicated reactor is then requested if the new fuel design is associated with a new structure design; in that case, only a full core with the new design will enable checking the behaviour of that fuel with respect to GTRF and assembly bow: that second step may take another 5 to 6 years.

Altogether, the whole in-core qualification process might need more than 10 years. Considering that 10 other years could be necessary to finalise a design for LTA's, the qualification on an industrial case could be targeted around 2032. By then, the average age of EDF existing reactors will be 47 years old.

Coherence of the nuclear fuel cycle

The feasibility of a new fuel design needs to be assessed for each step of the fuel cycle. The specificity in the French nuclear fuel cycle stems from the spent fuel processing performed at La Hague plant, and the recycling of the recovered recyclable materials (uranium and plutonium). When a new fuel is used at an industrial scale, it is necessary to keep the feasibility of plutonium and reprocessed uranium recycling safe, taking into account the reactors'safety limitations. For example, the increase in the average fuel assembly discharge burn-up from about 33 GWd/t to about 45 GWd/t between 1990 and 2010 led to an increase in the share of even isotopes in recovered Pu and RepU, with reduced energetic equivalences and radioprotection issues. Some adaptations have been or will be implemented in the nuclear fuel cycle to deal with that change in isotopic qualities.

Mixed cores: Another critical issue

When we enter a new fuel design, it is necessary to justify its compatibility with co-resident fuels. That justification is all the more important and difficult because the new fuel design is significantly different. If the compatibility cannot be demonstrated, the ultimate solution would be to entirely discharge the core and reload it with the new design only, which would be very costly.

Conclusion

Entering a new fuel design at an industrial scale in EDF reactors can be a very long process, and in the case of an ATF fuel, some additional time is necessary to finalise the LTA's design. However, depending on the remaining life duration of the reactors, it might be worth considering a breakthrough technology for PWRs, especially if significantly improved safety margins might be an incentive to consider life extension for the reactors.

Fuel licensing process for an industrial use

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To develop and license a breakthrough nuclear fuel technology for commercial use is becoming challenging. All the former safety analysis design limits (SAFDLs) defined in the 1970s for the standard UO_2-Zr fuels, might no longer be applicable.

Identification of the appropriate safety analysis design limits

For each type of innovative fuel, the developers will have to identify and investigate all the possible failure/ruins scenarios (not only those related to severe accidents but also those related to normal operation). In order to save time and to focus on the best options, those failure scenarios (which could be "killers" for the ATF concept) have to be determined early enough in the development process.

Based on the above failure scenarios, the developers will have to propose the licensing limits (and the experimental protocol to determine and to justify them).

As mentioned earlier, the licensing limits should not be defined against the accidental conditions only. For the operators, the (good) in-reactor fuel behaviour is crucial. As an example, in the case of the new fuel concepts coming with an outer coating, it is important to include the analysis of the consequences of the loss of this protective outer layer in the licensing process due to a manufacturing defect or an inevitable in-reactor fretting wear.

Obviously, the new/specific SAFDLs will have to be endorsed by the regulators (which could be a long process by itself).

Identification of a commercial reactor to irradiate the first ATF

A commercial NPP is not a material test reactor (MTR); irradiating lead test fuel rods (LTFRs) or lead test assemblies (LTAs) implies strict requirements regarding the manufacturing processes [which should not include chemicals (additives or solvants) potentially incompatible with the nuclear technical specifications], the compatibility with the hosting fuel core (in terms of geometry, enrichment, thermal hydraulic performances, etc.) and the robustness and the reliability of the ATF rods (within the hosting FA). It is recommended to perform out-of pile endurance tests, in-pile irradiation tests (in prototypical conditions) before starting an irradiation in a commercial reactor. It is known that a single "bad" rod failure could result in a costly premature shut-down of the NPP.

If the ATF rods or assemblies are not compatible with the current NPPs fuel cores designs, the ATF concept validation will rely on MTRs irradiations only and the whole core will have to be removed before introducing the new fuel concept. This process is going to be risky and costly.

Conclusion

In addition to finding the right ATF concept, the developers will have to address the (small and large scale) manufacturing issues, the accidental and normal operation behaviour issues (cladding, fuel pellet and their interactions with the fuel assembly structure), the economic vs. benefit vs. risk issues regarding the implementation of a new fuel concept.

Defining an appropriate licensing process (new concept means new requirements and limits) is a motivating challenge.

Considering the current discharge burn-ups observed in the nuclear industry (>60 Gwd/t) and the related time to get any irradiation feedback, the entire development/licensing process will be likely much longer than expected (30 years).

Session 4: Synthesis and Future Programmes

Chair: K. Pasamehmetoglu (INL, US)

Breakout Session A: Safety Issues

Chair: Marc Petit
Institut de Radioprotection et de Sûreté Nucléaire (IRSN), France

The first issue discussed during the breakout session on safety aspects of accident-tolerant fuels was the objective that must be assigned to the development of such concepts. The first goal should be to avoid, or at least limit, the release of radioactive materials into the environment in case of an accident. This implies that severe accidents (core melt) situations must be avoided. To reach this goal, the core geometry must remain coolable, even for accident scenarios worse than what current fuel designs are able to sustain.

There was a consensus that the station blackout (SBO) is a good reference transient to evaluate the potential benefits from new, more robust, fuel designs. With respect to the present situation, the merits of new designs can be analysed with respect to three figures:

- the “grace period”, i.e. the additional amount of time before the onset of core melt, during which more recovery actions can be made;
- the amount of combustible gases produced;
- the amount of radioactive materials released.

It is important to note that those three values are not independent from one another. They may be understood as three different ways to measure the improvements arising from accident tolerant fuels.

The notion of “grace period” was discussed and it was suggested that it should be compared to the amount of time needed to switch from normal operation to accident management type of procedures. The participants agreed that the “grace period” should be counted in hours (or even days but the realism of this last goal was questioned). In other words, there was a consensus that a “grace period” of some minutes is pointless and definitely not worth the effort of developing and characterising the behaviour of new concepts.

Although the purpose of accident-tolerant fuel development is to improve the core robustness in design basis accidents (DBA) and situations somewhat beyond like SBO, it was recognised that new concepts must not lead to situations worse than the present one if, despite all efforts, core melting cannot be avoided. Thus, it is important to measure the oxidation rate and to characterise the materials interactions above 1 200°C for the proposed designs.

It was then recalled that the fuel is part of the core system as a whole. Potential improvements must be sought not only for fuel pellets and cladding, but also possibly for channel boxes, control rods, supporting structures. Furthermore, the core design must remain homogeneous: for example, the efficiency of a very robust fuel pin design may be challenged if the supporting structures cannot sustain the accident conditions.

When developing improved fuel concepts, one should not forget about normal reactor operation. Indeed, if using new fuel, safety during normal operation must be at least as good as it is at present. There are a number of physical phenomena to assess and criteria to be verified. Those are, for example, dimensional stability, corrosion resistance, departure from nucleate boiling, irradiation assisted stress corrosion cracking, fission gas release, etc. It was stressed that qualifying a new fuel for safe operation is a long and difficult process. Moreover, ensuring a very high-quality level of manufacturing is also an important requirement that may not be easily fulfilled with some of the materials currently contemplated.

Based on the above considerations, there was an agreement on the testing priorities for new fuels. Those are:

- corrosion tests under in-service conditions, including a representative primary water chemical composition; the phenomena to be characterised are oxidation, hydriding, CRUD deposition;
- oxidation experiments at temperatures above 1 200°C; the behaviour of the materials must also be known for beyond DBA conditions;
- degradation tests under steam environment; material interactions, for example between fuel pins and control rods, may be more complex than just what thermodynamic equilibrium predicts.

To evaluate the merits of the different accident-tolerant fuel concepts proposed, simulation tools will be required. In particular, adequate models will need to be developed and validated. Also, assessment of uncertainties in the new modelling will be required. It should be kept in mind that again a significant amount of time and efforts will have to be spent in order to fulfill these requirements and experimental support will be needed.

The above considerations were made with the assumption that the new fuel concepts are only variations of the current ones (e.g. new cladding material) and the mode of reactor operation is not dramatically changed. If this is not the case, the safety issues, the physical phenomena to be considered, and the criteria to be applied would need to be revisited. This was beyond the scope of the discussion during this session.

Finally, the participants agreed that the development of fuel with improved properties is only one aspect of the more general goal of enhancing the defence-in-depth. Other options are also of great interest for improving the robustness of nuclear power plants to accidents, such as assuring more reliable means of core cooling (ECCS, steam generators), strengthening the emergency preparedness procedures. The industry added that in the end cost matters and that the most effective solutions to enhance safety in the frame of a cost/benefit analysis should be favoured.

Breakout Session B: Reactor Performance, R&D and Technological Issues

Chairs: Nicolas Waeckel[1], Larry J. Ott[2]
[1]Électricité de France (EDF), [2]Oak Ridge National Laboratory (ORNL)

Three main outcomes resulted from this break-out session, as well as a few recommendations.

Outcomes:

- The development of ATF is likely to be a long, stepwise/iterative approach, for which a broad consensus on the objectives is essential:
  - ATF should be as good as current fuels during normal operation ("first do no harm" principle), including good mechanical properties, and for DBA;
  - ATF should exhibit higher performances during severe accidents.
- For normal operation and DBA, the qualification path is well known and should follow the regulatory requirement:
  - the same as for an advanced fuel (properties characterisation, justification tests regarding SAFDLs of SRP4.2, etc.);
  - need to confirm though that the ATF candidate does not exhibit a specific behaviour or damage mechanisms that may require a specific justification (specific tests, specific SAFDL).
- For severe accidents, the recurrent questions are as follows:
  - What is the target? (in terms of expected performance);
  - What is the criterion for success?
  - Which tests should be run to justify the new fuel?

Recommendations:

- In terms of organisation of the general development process:
  - identify the key factors or physical parameters that will really reduce the accident consequences;
  - calculate one BWR scenario (3-F) and one PWR (TMI-2) with the appropriate simulation tools;
  - run sensitivity analysis to scale the relative (beneficial) effect of various physical parameters (thermal conductivity, oxidation kinetics, H_2 production, exothermal reaction, FP retention, mechanical behaviour, etc.);

  - these calculations will be run with data progressively qualified on ad-hoc experience.

- International co-operation is recommended, especially during the feasibility phase:
  - no proprietary concern at this stage;
  - from utility point of view, it is better if all fuel vendors propose an ATF, to ensure the security of supply;
  - cross-irradiation of ATF candidates (ATR, HRP, JHR) can accelerate the selection process, benefit from advanced capabilities (instrumentation, PIEs) and be cost effective.

It is stressed that the NEA is the right vehicle to streamline this collaboration (cross-check tests, shared irradiations of coupons) and suggested that a follow-up workshop meeting could be scheduled in one year, in which different candidates for ATF should be discussed and define:

- those with high priority;
- those with lower;
- nice to get;
- waste of time;

This process will be fed by a first selection of candidates which will be carried out by DOE in the early summer of 2013, in the framework of identifying candidates for experimental irradiation in ATR in 2014.

International collaboration for development of accident-resistant LWR fuel

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Following the March 2011 multi-unit accident at the Fukushima Daiichi plant, there has been increased interest in the development of breakthrough nuclear fuel designs that can reduce or eliminate many of the outcomes of a severe accident at a light water reactor (LWR) due to loss of core cooling following an extended station blackout or other initiating event. With this interest and attention comes a unique opportunity for the nuclear industry to fundamentally change the nature and impact of severe accidents. Clearly, this is no small feat. The challenges are many and the technical barriers are high. Early estimates for moving maturing R&D concepts to the threshold of commercialisation exceed one billion USD. Given the anticipated effort and resources required, no single entity or group can succeed alone. Accordingly, the Electric Power Research Institute (EPRI) sees the need for and promise of co-operation among many stakeholders on an international scale to bring about what could be transformation in LWR fuel performance and robustness.

An important initial task in any R&D programme is to define the goals and metrics for measuring success. As starting points for accident-tolerant fuel development, the extension of core coolability under loss of coolant conditions and the elimination or reduction of hydrogen generation are widely recognised R&D endpoints for deployment. Furthermore, any new LWR fuel technology will, at a minimum, need to (1) be compatible with the safe, economic operation of existing plants and (2) maintain acceptable or improve nuclear fuel performance under normal operating conditions. While the primary focus of R&D to date has been on cladding and fuel improvements, there are a number of other potential paths to improve outcomes following a severe accident at an LWR that include modifications to other fuel hardware and core internals to fully address core coolability, criticality, and hydrogen generation concerns.

The US Department of Energy is providing substantial support for initial R&D on accident-tolerant fuel concepts with an aggressive target of a lead test assembly (LTA) in an LWR by 2022. EPRI proposes an additional stretch goal of commercialisation of a new LWR fuel by 2030. The scale of and resource demands associated with these R&D targets require a global collaborative structure to leverage resources, create an environment for innovation and co-operation, and foster necessary partnerships and arrangements among the many key players and roles spanning government, academic, and industrial sectors. EPRI is proposing a voluntary, open, and non-binding structure to quickly build momentum and to maximise early engagement and information exchange among key stakeholders. The flexibility of this organisational model offers an environment that is compatible with and encourages engagement, innovation, and development of the more formal arrangements and partnerships that will be needed to commercialise current R&D concepts. The opportunity for transformation of LWR fuel performance under normal and accident conditions is now. Accordingly, the time for action is now. Commercialisation of accident-tolerant fuel in the near future can only be realised with collaboration among governments, industry and academia on a scale commensurate with the challenges at hand.

Presentations not included in the programme

Simple process to fabricate nitride alloy powders

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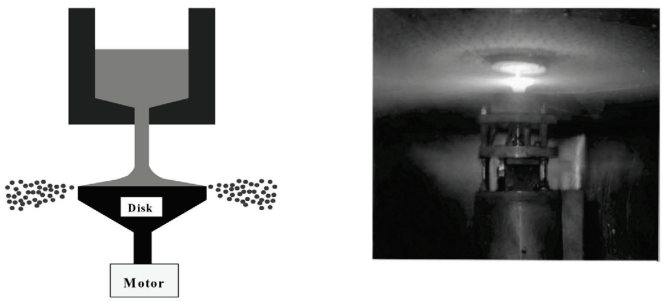
Introduction

Uranium mono-nitride (UN) is considered as a fuel material [1] for accident-tolerant fuel to compensate for the loss of fissile fuel material caused by adopting a thickened cladding such as SiC composites.

Uranium nitride powders can be fabricated by a carbothermic reduction [2] of the oxide powders, or the nitriding of metal uranium [3]. Among them, a direct nitriding process of metal is more attractive because it has advantages in the mass production of high-purity powders and the reusing of expensive $^{15}N_2$ gas. However, since metal uranium is usually fabricated in the form of bulk ingots, it has a drawback in the fabrication of fine powders.

Figure 1: (a) Schematics of atomisation process, (b) Video capture image



(a) (b)

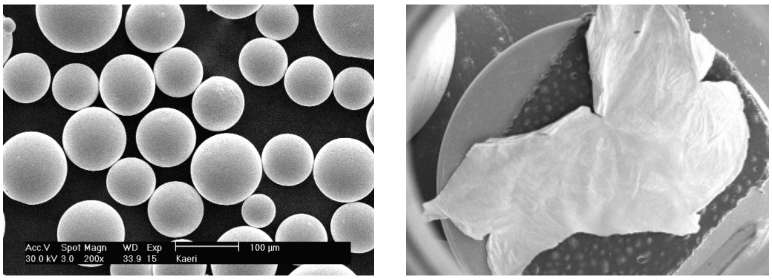
The Korea Atomic Energy Research Institute (KAERI) has a centrifugal atomisation technique to fabricate uranium and uranium alloy powders. In this study, a simple reaction method was tested to fabricate nitride fuel powders directly from uranium metal alloy powders. Spherical powder and flake of uranium metal alloys were fabricated using a centrifugal atomisation method. The nitride powders were obtained by thermal treating the metal particles under nitrogen containing gas. The phase and morphology evolutions of powders were investigated during the nitriding process. A phase analysis of nitride powders was also part of the present work.

Centrifugal atomising process

KAERI has developed the centrifugal rotating disk atomisation process to fabricate spherical uranium metal alloy powders which are used as advanced fuel materials for research reactors [4].

As shown in Figure 1, the rotating disk atomisation system involves the tasks of melting, atomising, and collecting. A nozzle in the bottom of melting crucible introduces melt at the center of a spinning disk. The centrifugal force carries the melt to the edge of the disk and throws the melt off the edge. Size and shape of droplets can be controlled by changing the nozzle size, the disk diameter and disk speed independently or simultaneously. By adjusting the processing parameters of the centrifugal atomiser, a spherical and flake shape alloy powders were obtained. Figure 2 shows the typical morphologies of the obtained particles.

Figure 2: Typical powder morphologies, (a) Spherical U-10 wt% Zr powder (b) U flake



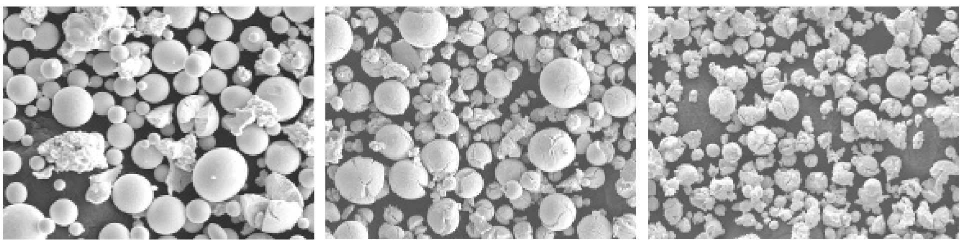
(a) (b)

Nitride powders fabrication

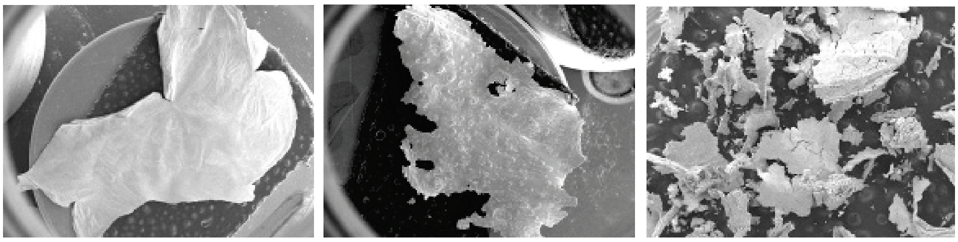
Two types of the simple thermal treatment procedures were tested to fabricate nitride powders.

First, the procedure is a direct nitriding process in which the metal powders were annealed at 1 000^{0}C under nitrogen gas and then further annealed at 1 500^{0}C under hydrogen containing Ar gas atmosphere. Figure 3 shows the shape evolution of U metal particles during the successive annealing step. It was revealed that the particles were fragmented to smaller particles during the annealing. The XRD results showed that the uranium metal converted to UN_2 phase during the annealing at 1 000^{0}C and then decomposed to UN phase during the further annealing at 1 500^{0}C.

Figure 3: Powder morphology evolution during the direct nitriding process (a) U spherical powder (b) U flake



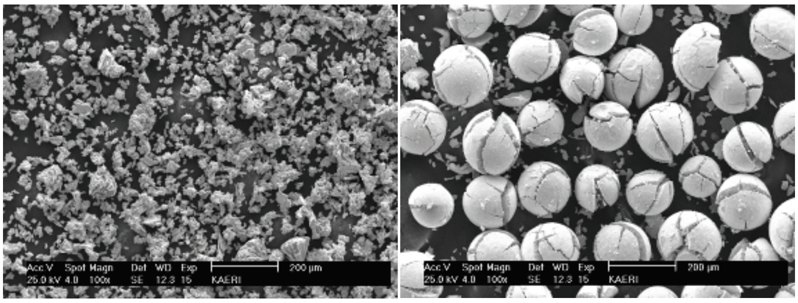
(a)



(b)

Observed fragmentation and cracking of particles were caused by sequential volume changes of expansion and contraction which were accompanied by the formation and decomposition of uranium nitrides. Although uranium nitride powders were successfully fabricated during the simple nitriding process, it seems that milling of the obtained powder might be necessary to fabricate sintered nitride fuel pellets.

Figure 4: Powder morphology evolution after the modified nitriding process (a) U metal, (b) U-10 wt% Zr



(a) (b)

In order to fabricate finer nitride powders, a nitriding procedure has been modified. In the modified process, the particles were heat-treated at 250^{0}C in H_2 before nitriding. Figure 4 shows the nitride powders obtained by the modified process. The addition of a hydriding step [5] was effective in obtaining fine uranium nitride powder as shown in Figure 4(a). In the case of U-10 wt% Zr-alloy, however, only a few large cracks were developed on the particle surface and the particle maintained its size as shown in Figure 4(b). This result reveals that hydriding and nitriding kinetics or mechanisms of U-10 wt% Zr alloy are quite different from those of U metal.

Summary

Nitriding the U metal and alloy particles obtained through centrifugal atomisation is a simple and effective method to fabricate the nitride fuel powders.

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