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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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ONGOING AND PLANNED FUEL SAFETY RESEARCH IN NEA MEMBER STATES

Compiled from SEGFSM Members' Contributions

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Abstract

This report is in response to an action placed on SEGFSM members to compile ongoing and planned fuel safety research in NEA member states with the aim of providing CSNI an overview on related R & D international programmes and projects, along with the identification of current and future needs and priorities.

The report is based on replies to a questionnaire distributed to SEGFSM members requesting them to identify fuel safety research programmes and to provide information on achievements and future plans. The report is confined to the replies received, as a consequence it cannot be viewed as comprehensive; programmes may well be in progress in addition to those detailed here.

The report is organized in topic sections relating to: fuel and clad studies, integral fuel rod tests and PIE, LOCA and RIA studies including whole rods and bundles as well as single effects studies of fuel and cladding, code development for both steady state and transient fuel behaviour, thermal hydraulics, reactor physics codes and finally severe accident studies.

10. Severe Accident Studies

10.1.1 Fuel Moderator Interaction (MFMI) Experiments, (Canada)

These MFMI experiments will be performed at the Chalk River Laboratories of AECL. The CANDU Owners Group (COG) is sponsoring these experiments under joint funding from Ontario Power Generation, Hydro Québec, New Brunswick Power, and AECL.

The MFMI experiments are being performed to improve our understanding of the energetics of the interaction of molten material with the heavy-water moderator under CANDU single-channel accident conditions, and to provide data for use in reactor safety code validation. The stagnation feeder break and flow blockage scenarios both lead to fuel channel failure and have the potential for injection of small quantities of molten UO_2-Zircaloy mixtures into the heavy-water-filled moderator vessel. The results of the MFMI experiments will be used to verify the assumption currently made in CANDU safety analysis that classical steam explosions do not occur under these conditions.

The main feature of the MFMI facility is a robust confinement vessel (5.5 m tall, 1.5 m in diameter) located inside a concrete pit enclosed in a concrete building. The experimental plan provides for incremental increases in the molten material loading to help manage this risk. The MFMI experimental program has been defined and the facility is under construction; No experimental results have been obtained to date.

11. Conclusions

Normal Operations; UO_2 and MOX fuel

Over the last twenty years there has been a gradual increase in discharge burn-up from commercial power reactors. With the original 3 cycle operation, the discharge burn-up was of the order 30 MWd/kg but now, most countries have increased that to a level approaching 60 MWd/kg, see Table 11.1. This has been accompanied by intense R&D both in test reactors like that operated by the OECD Halden Reactor Project and R2 at Studsvik and LTA irradiation in power reactors.

It is anticipated that with the current research programmes there are sufficient data to develop and validate fuel performance codes to support these higher levels of discharge burn-up.

The main goal for MOX fuel is for it's safety issues to be treated indistinguishably from those of UO_2. Research to date has shown that MOX pellets have a slightly worse thermal conductivity but similar degradation with burn-up. At the same time, it exhibits slightly greater fission gas release, although the onset of release as a function of temperature and burn-up is little different from that of UO_2. FGR from MOX at high burn-up is exacerbated by a higher reactivity than UO_2 due to its neutronics characteristics. Improvements in MOX are being pursued by vendors by investigating the effect of homogeneity on thermal performance and FGR. However, MOX pellets are naturally more compliant than UO_2 because of their higher rate of thermal creep.

LOCA

The 1980s saw great attention paid to the LOCA scenario and much data are relevant today. However, the data on high burn-up behaviour are rather scarce. In this respect, aspects such as possible fuel relocation or 'slumping' into the ballooned area leading to higher clad temperatures need addressing as well as the effect of axial constraints during quenching. Most important is the need to re-visit the 17% Equivalent Clad Reacted (ECR) criterion in the light of new alloys and new geometries, (clad diameters and thickness, see Table 11.2).

Table 11.2 Examples of PWR and BWR fuel designs

Assembly type	PWR 14 x 14	PWR 17 x 17	BWR 8 x 8	BWR 10 x 10
Cladding outer diameter, mm	10.72	9.50	12.52	9.62
Cladding inner diameter, mm	9.48	8.36	10.79	8.36
Cladding wall thickness, mm	0.62	0.57	0.87	0.63
Cladding cross section, mm^2	19.70	16.00	31.67	17.80
Fuel pellet diameter, mm	9.29	8.19	10.57	8.19
Fuel pellet cross section, mm^2	67.80	52.68	87.75	52.68
Rod pitch, mm	14.10	12.60	16.30	12.40
Rod-to-Rod distance, mm^2	3.38	3,10	3.78	2.78
Water cross section (subchannel) mm^2	108.5	87.9	142.6	81.1
Total clad cross section in assembly mm^2	3861	4624	2027	1780
Cladding surface to fuel volume ratio mm^{-1}	0.50	0.57	0.45	0.57
Cladding-to-fuel cross section ratio	0.29	0.30	0.36	0.34
Water to fuel cross section ratio, subchannel	1.60	1.67	1.63	1.54

Notes:
- Total clad cross section increased in PWR from 14x14 to 17x17, decreased 10% in BWR from 8x8 to 10x10.
- Cladding surface / fuel volume ratio increased in PWR and BWR.
- Cladding to fuel cross section ratio relatively constant.
- Coolant-to-fuel ratio changed by only ~5% in PWR and BWR.

Two aspects of fuel slumping are the impact of fuel-clad bonding on the propensity of slumping and FGR in the slumped region leading to increased local pressure. Another aspect is the impact of axial gas flow through a 'tight' fuel column on ballooning behaviour. From section 5.2 it is clear that there are several research programmes addressing the properties of high exposure cladding to LOCA but not so much on separate effects fuel studies. Halden have carried out axial gas flow studies in fuel rods over a range of burn-up and test have shown a severe restriction in volume flow at high burn-up thus restraining the rate of clad ballooning. What is lacking therefore is evidence for or against slumping and the internal pressure generated by FGR within the slumped region during the temperature/time envelope of the transient.

RIA

Early experiments on low burn-up fuel showed that fuel failure by rapid reactivity insertion only occurred after energy depositions around 200 cal/g. It was not until the first CABRI REP Na test on high burn-up fuel with severely oxidized cladding that it was realized that under these conditions a much lower energy deposition caused fuel failure and extensive fuel dispersal. This result initiated a renewed interest in this type of accident with integral and separate effects tests initiated as outlined in section 6. The main issues with respect to the cladding are its mechanical response during high rates of strain and the effect of hydides on the mechanical properties. Regarding the fuel, tests have shown that the clad strain was greater than that expected from thermal expansion of the fuel pellet. Consequently, there would appear to be a new loading force imposed by high burn-up pellets, so the goal is to explain this new force and quantify it. In this respect, the high burn-up structure at the pellet rim is under intense separate effects study as this is anticipated to be the root cause of the increased clad loading. The reduction in acceptible energy deposition at high burn-up is considered therefore to be a result of degraded clad mechanical properties and increased strain from restructured regions of pellets. It is clear that the several research programmes both separate effects studies on fuel, section 6.2 and on cladding, section 6.3 should in the near future lead to a better understanding of this type of accident. As a separate but parallel study, it is important to improve reactor physics codes and calculation to see whether or not it is possible to deposit energies as high as those required for fuel failure.

Resolution of Safety Issues

As a final comment on the inter-play between phenomena influencing fuel safety, the criteria used to assess compliance and the experimental database on which such criteria can be derived, Figure 11.1 shows a 'road map' linking these three components. From this it is easy to identify where supporting data already exist and where new data will be generated by ongoing programmes or programmes already in the planning stage.

This report has addressed safety issues as they are applied to current reactor systems, the so called 'Generation II' designs. There are now advanced reactor designs categorised as generations III, III+ and IV. A common element of these is the introduction of passive safety features. Thus one question which requires consideration is whether or not these new designs will operate within the safety envelope of the current design. If this is the case, then the scenarios currently being addressed should apply without extension to these new systems. Hence, future R&D will concentrate on

compliance of new materials to the current or a reduced safety envelope and not the consideration of new scenarios.

It is clear that the currently most important issues to the international nuclear industry are high burn-up performance both in normal operations, LOCA and RIA conditions. The survey of international research programmes outlined above demonstrates the large element of activity to address these issues. When put together, the individual programmes add up to a tremendous effort in both time and money and will ultimately lead to a much better understanding of materials and component behaviour in a wide range of postulated scenarios.

It is very important therefore, that these activities are well supported and that their results should be made available to the widest possible audience. Thus ensuring a common culture of safe and economic production of electricity from nuclear power generation.