IAEA
International Atomic Energy Agency

Institute for Nuclear Research Pitesti activity developed in the light of the accident at the Fukushima Daiichi Nuclear Power Plant

RATEN ICN PITEŞTI

Cristian Gentea
Institute for Nuclear Research, Pitesti, Romania
Campului Street, No.1, 115400 - Mioveni, POB 78 - Pitesti, Arges County, Romania, www.nuclear.ro
cristian.gentea@nuclear.ro

INR has as main objective of its activity the scientific research, the fundamental and applied technological development in order to ensure the scientific and technical support for Romania's nuclear energy sector (TSO).

The research is carried out following multi annual strategic and programmatic plans (i.e. 5 years) concerning 18 programs for specific field from which we mention Nuclear Safety, Fuel Channel, Nuclear Fuel Development, Spent Fuel and Radioactive Waste, Analysis of operational experience of NPP. Most of the programs are accomplished by topic works, contracted each year in order to ensure the resources for research activity and to contribute to nuclear safety enhancement.

Following the accident at the Fukushima Daiichi Nuclear Power Plant and the IAEA "Action Plan" the program for Nuclear safety was completed during 2011 with a sub program concerning the applications of lessons learnt to CANDU Power Plant operation in Romania, dealing with severe accidents.

In general terms the actual safety analysis is based on the Design Basis Accident which is a "stilised" accident given by prescription is considering a "design basis accident" which is a postulated event, in correlation with specific power plant technology and the historical developed in time, but the lessons learnt show that beyond design basis events happen leading to severe accidents.

Prevention of accidents, development of accident, time of intervention and mitigation depends on understanding of responses to Systems Structures and Components to a broad consideration of events and on the materials deep knowledge of properties in the given conditions time dependents.

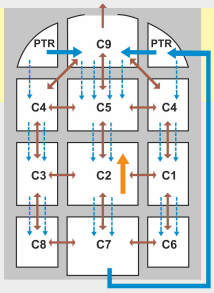
Considering the above assumption the recent research activity engaged in 2011 at the Institute for Nuclear Research is oriented to "Severe accident analysis" concerning accident progress which encompass specific conditions of CANDU 6 PHWR. The following abstracts present the specific research in this area performed by Institute's researchers:

Nuclear 2013 (I.2.4)

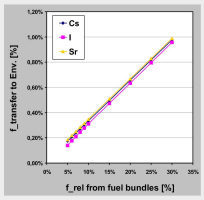
Source Term Calculation for an Accident Initiated by End Fitting Failure

Marin Constantin
Institute for Nuclear Research Pitesti, Romania
e-mail: marin.constantin@nuclear.ro

End Fitting Failure (EFF) is a very specific accident for CANDU type reactors initiated by a failure of the re-fuelling machine. From the point of view of the source term calculation, EFF is included in direct transfer to the containment accident. In this paper, the source term in the containment and also the source term to the environment are calculated by CPA-IODE-ISODOPEA-DOSE modules of the ASTEC code. The evolution of the distributions for the most important released fission products (FPs) is presented for different rooms of the containment and for the environment zone, and also for the different hosts (aerosols, walls, sumps). The main assumptions of the calculations are: all fuel bundles for one channel are ejected; channel is closed after failure by recovering of fuel handling machine; ejected fuel bundles are assumed to be broken into pieces; UO2 pellets are assumed to be completely ejected from the sheath and broken into fragments; after a short time, the fuel bundles are exposed in air; the inventory at a burnup of 8,000 MWd/tU is considered; different release speciation directly into containment atmosphere; the gap inventory and exposed fraction of the grain boundary inventory (i.e. surface area of particle size) are released at the time of failure; releasing fractions for volatile FPs are postulated at 8% according with literature data; dousing system is supposed to work normally or in the failure mode (stopped); a connection to the environment zone, immediately after ejection, is supposed (for example, by an unfiltered venting). At the same time, the paper investigates the uncertainties in the main data and their influence on the source term amount and structure.



Nodalization of the containment.



Variation of the total transfer fractions versus the releasing fractions from the fuel bundles.

Nuclear 2013 (I.1.11)

Self-Disintegration of Fuel Pellets in Steam-Air Atmosphere

Iulia Dumitrescu, Dumitru Ohai
Institute for Nuclear Research Pitesti, Romania
e-mail: iulia.dumitrescu@nuclear.ro

The physical phenomena involved in severe accidents are extremely complex and demand the development of specific research. The aim of this research is to understand the physical phenomena and to reduce the uncertainties regarding their quantification. The final goal is to develop models that can be applied to reactors. These models grouped in computer codes should allow the prediction of severe accident progression. Because in this field is not possible to conduct experiments on a real-world scale, elementary tests must be used. This type of tests allows each physical phenomenon to be studied separately. Then global tests should follow to confirm the interaction between phenomena. The severe accident scenarios show the evolution of reactor core damage finalized with the corium and debris bed formation. Generally located above the corium, the debris bed has the temperature range evaluated between 1300 0C and 300 0C. At the air or steam ingress, in the debris bed the main chemical phenomena contributing to subsequent degradation and fission products release are the following: oxidation of the Zircaloy 4 sheaths of the still intact rods, oxidation of the mixtures composed of Zr and UO2 in the configuration of solid debris and oxidation of pure UO2 in the fuel pellets remnants. The impact of UO2 fuel pellets oxidation on reactor safety is important due to its influence on fission products release. This paper describes the UO2 fuel pellets behaviour during oxidation tests in mixed steam-air atmosphere and temperature ranging between 400 – 800 0C in the FIPRED_EQ set-up. During oxidation, the UO2 fuel pellets specimens are spalled due to the crystallographic system modification. The compacts resulted from pellets oxidation were analysed by Scanning Electron Microscopy and X-Ray Diffraction. Also the particle size distributions of powders were measured using Laser Diffraction technique.

Nuclear 2013 (I.2.6)

Operating Experience Insights Supporting Ageing Assessments

Mirela Nitoi
Institute for Nuclear Research Pitesti, Romania,
e-mail: mirela.nitoi@nuclear.ro

Physical ageing means in the end, a gradual deterioration of Systems, Structures or Components (SSC) performances. In order to avoid critical failures, the ageing effects should be assessed and an ageing management program should be implemented. Be effective in ageing management means looking at the right spots with the right techniques, and one of the most effective tool which could be used for this purpose is the analysis of operating experience. This paper has as objective to perform a review of available operating experience, with the aim to provide a better picture about the impact of ageing effects, image based on the number of events reported that were caused by ageing related failures of systems and components. The analysis of operating experience aimed to identify the following: proportion of ageing related events from among all reported events; the most vulnerable components to ageing (in terms of most frequently components represented in the data bases); the most frequent occurred degradation mechanisms;
consequences and impact of events caused by SSC ageing. The IAEA International Reporting System for Operating Experience and NRC Licensee Event Reports were chosen as reference databases to identify relevant ageing events, both databases being internationally recognized as important sources of information about events occurrences in the nuclear power plants. The event distribution in the selected time window will be graphical presented and the evolution in time of ageing related events (in terms of reported events) will be analyzed. The contributions of each major degradation mechanisms that have induced the ageing related events (specific to each defined group of components) will be represented and also discussed. The paper demonstrates the possibility to use operating experience insights in highlighting the ageing effects.

Nuclear 2014 (I.3.11)

Average Diffusion Coefficient Measurement of the Oxygen in Zy-4 Cladding, at Temperatures between 873 and 1673 K

Tiberiu Meleg, Mariea Deaconu
Institute for Nuclear Research, Pitesti, Romania
e-mail: tiberiu.meleg@nuclear.ro

The main purpose of this paper is to present the results obtained for diffusion coefficient measurement of the oxygen in Zy-4 cladding material, from the initial gradient of the sorption curves. Because the oxidation kinetic of the zirconium alloys are mainly governed by the diffusion, the curves of absorption can be directly related to the diffusion coefficient. If the initial gradient of the sorption curves give the average value of the diffusion coefficient, in the later stage the influence of the tensile stresses at oxide - metal interface, the evolution of the surface of this and the porosity development in the oxide layer will strongly affect the diffusion. Zy-4 cladding samples were oxidized in dynamic oxygen atmosphere at 16ml/min flow rate and 1bar pressure, under isothermal conditions, at temperatures between 873K and 1673 K. With the average values of the diffusion coefficients obtained from the initial gradient of the sorption curves plotted against the square root of time, the temperature dependence of the diffusion coefficient were obtained, both from alfa and beta structures of the zirconium. The results obtained, related also from the structural changes, the oxide layer thickness, the contributions of the tensile stresses and the surface of diffusion are presented and discussed.

Nuclear 2012 (I.1.6)

The Flame Front Behaviour Following a Severe Accident in a Generic CANDU 6 Containment

Gabriela Radu1, Ilie Prisecaru2
1Institute for Nuclear Research Pitesti, Romania; 2Univesity Politehnica Bucuresti, Romania
gabriela.radu@nuclear.ro

During a severe accident in a nuclear power plant, oxidation of the metallic components of the reactor core will produce hydrogen. The radiolysis of water and the corrosion of metals in containment are additional sources of hydrogen. This could potentially lead to the formation of flammable mixtures of hydrogen/air/steam in reactor containments. Combustion of these mixtures could lead to pressure and temperature levels that may challenge the containment integrity. Hydrogen combustion can cause containment building failure by static or quasi-dynamic pressure loads (deflagration or diffusion flame) or dynamic pressure loads (detonation), equipment failure due to thermal or pressure effects, and missile generation. Deflagration (slow flame) and detonation (very fast flame) are two completely different modes of combustion, with different propagation mechanisms. As detonation leads to spatially non-uniform pressures that exceed the maximum pressure caused by deflagration, this combustion mode is of interest in reactor safety analyses. A hydrogen detonation can be developed by either direct initiation, which requires high energy source, or flame acceleration (a Deflagration to Detonation Transition, DDT). This paper presents a study of the behaviour of the flame front following in vessel phase of a severe accident initiated by a Station BlackOut (SBO) in a generic CANDU 6 station containment. The calculations are done, using the CPA-Front model of the code ASTEC, with different sets of model parameters with the best fit on the experimental data. The aim is to investigate the influence of some varied input parameters for the flame front model with respect to pressure and temperature histories in different containment zones.

Nuclear 2012 (I.1.4)

Investigating the Relation between Common Cause Coupling Factors and Ageing

Mirela Nitoi1,2, Jaroslav Holy3, Irina Dinu4
1Institute for Nuclear Research Pitesti, Romania; 2 EC JRC - Institute for Energy, Petten, The Netherlands
3Nuclear Research Institute Rez, Prague, Czech Republic; 4Cernavoda NPP, Romania
mirela.nitoi@nuclear.ro

Usually, probabilistic safety analyses (PSA) are using the assumption of the constant component failure rate, and the assumption that the probability associated to common cause failure will be constant in time. Still, as ageing phenomena could have some influence on components or systems performances, is probable that it could influence also the probability of occurrence for common cause failures (CCF). This paper is devoted to investigation of the most important common cause failures initiating factors from point of view of relation to ageing phenomena. The connections between common cause failure potential and ageing phenomenon, as the influence of the individual prevention factors against CCF were evaluated. A questionnaire has been developed on this topic and all the inputs provided were summarized, compared and commented. The analysis has not been limited only to the list of negative factors that would increase the CCF potential, it has included also positive factors representing the prevention and correction measures applied with the aim to avoid occurrence and recurrence of CCF events, whether related to ageing or not. Two main topics were discussed:
- the strength of relation between the individual CCF coupling factors (contributing to total CCF potential) and ageing phenomenon;
- the effect of CCF prevention measures on the coupling factors versus ageing relation (estimated for those CCF coupling factors, which have been evaluated as coincident with ageing phenomenon, with at least medium level of coincidence).

Conclusion

The Institute annual programs 2012-2014 contains 28 works dedicated to study, understanding and analysis of phenomena and progression of events during severe accidents, with peculiarities to CANDU 6 HWP, in order to ensure an effective response to beyond design basis events. The research in this area will continue in the future and will be applied to existing and new power reactors.

Poster ID Number: IAEA-CN-235-93
IEM on Strengthening Research and Development Effectiveness in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant.
IAEA Headquarters, Vienna, Austria, 16-20 February 2015