SUMMARY

1. INTRODUCTION

In liquid-metal cooled fast reactors (LMFR) or in accelerator driven sub-critical systems (ADS) with LMFR like sub-critical cores, the primary coolant pipes (PCP) connect the primary coolant pumps to the grid plate. A rupture in one of these pipes could cause significant loss of coolant flow to the core with severe consequences. In loop type reactors, all primary pipelines are provided with double envelopes and inter-space coolant leak monitoring systems that permit leak detection before break. Thus, the PCP rupture event can be placed in the beyond design basis event (BDBE) category. Such an arrangement is difficult to incorporate for pool type reactors, and hence it could be argued that the PCP rupture event needs to be analysed in detail as a design basis event (DBE, category 4 event).

However, the primary coolant pipes are made of ductile austenitic stainless steel material and operate at temperatures of the cold pool and at comparatively low pressures. For such low stressed piping with negligible creep and embrittlement effects, it is of interest to discuss under what design provisions, for pool type reactors, the guillotine rupture of PCP could be placed in the BDBE category.

The topical Technical Meeting (TM) on Primary Coolant Pipe Rupture Event in Liquid Metal Cooled Reactors' (Indira Gandhi Centre for Atomic Research, Kalpakkam, India, 13–17 January 2003) was called to enable the specialists to present the philosophy and analyses applied on this topic in the various Member States for different LMFRs.

The scope of the technical meeting was to provide a global forum for information exchange on the philosophy applied in the various participating Member States and the analyses performed for different LMFRs with regard to the primary coolant pipe rupture event. More specifically, the objectives of the technical meeting were to review the safety philosophy for the PCP rupture event in pool type LMFR, to assess the structural reliability of the PCP and the probability of rupture under different conditions (with/without in-service inspection), to review the classification of the PCP rupture event in DBE/BDBE categories and discuss the applicable design safety limits, to assess the need for consequency analysis, like pipe whip effects, primary pump seizure and multiple pipe rupture, and, last but not least, to present the results of analyses of the event *per se* for flows and/or temperatures and improved design concepts for minimizing the consequences to the core.

The TM was attended by 25 participants from six Member States and one international organization.

2. NATIONAL PRESENTATIONS

A brief overview of the respective national fast reactor programmes was given by the representatives of the participating Member States, before addressing the specific objectives of the TM.

2.1. China

Fast reactor research and development activities are pursued by the China Institute of Atomic Energy (CIAE). CIAE has evolved out of the Institute of Modern Physics, Academia Sinica, founded in 1950. It is the birthplace of nuclear science and technology in China. CIAE has a staff of 3 400, including 660 senior scientists and senior engineers, 43 supervisors of doctoral students, and 9 members of Chinese Academy of Sciences and Chinese Academy of Engineering. CIAE has eight departments and several research centres, and it can count on five eminent nuclear facilities, as well as on many small laboratories. Presently, CIAE is implementing four significant projects: the Chinese Experimental Fast Reactor (CEFR), the Chinese Advanced Research Reactor (CARR), the HI-13 Tandem accelerator upgrading project, and the radiochemistry laboratory.

With regard to nuclear energy, China has adopted a strategy based on pressurized water reactors, fast breeder reactors, and fusion reactors. Nuclear power will contribute to China's long term sustainable energy supply, and help meet the challenges of the 21st century. However, due to economic reasons and to allow for a solid experience base to be built, nuclear power technology development is presently pursued at a moderate pace. In mainland China, there are two nuclear power plants (NPPs) in operation with a total capacity of 2.1 GW(e). Four NPPs are under construction, and two NPPs are planned for the Tenth Five Year Plan (2001–2005). One or two additional NPPs are still under discussion. A new NPP, located is near the Lin Ao NPP in Guang Dong, was licensed in December 2002. As of 2001, nuclear power represents 1% of the total installed electric power in China. The share of nuclear power will attain 3% before 2005. In absolute terms, it is foreseen that the total nuclear power capacity will reach 8.5 GW(e) by the year 2005, and 14–15 GW(e) before 2010. With CEFR presently under construction, it is hoped that a prototype fast breeder reactor will be commissioned before 2020.

The conceptual design of the CEFR was started in 1990 and completed in 1993. The CEFR preliminary design was started in early 1995 and finished in August 1997. In the period 1995–1996, CIAE had technical design cooperation with a Russian liquid metal fast breeder reactor association whose members were Institute of Physics & Power Engineering, Experimental Designing Bureau of Machine Building, and Atomenergoproekt. The CEFR detailed design work started in early 1998, and now is almost finished (as of beginning 2003).

The CEFR is a sodium cooled, bottom supported 65 MW(th) experimental fast reactor fuelled with mixed uranium-plutonium oxide (the first core, however, will be loaded with uranium oxide fuel). Fuel cladding and reactor block structural materials are made of Cr-Ni austenitic stainless steel. It is a pool type reactor with two main pumps, and two loops for the primary and secondary circuit, respectively. The water-steam tertiary circuit has also two loops, with the superheated steam collected into one pipe that is connected with the turbine. CEFR's has a natural circuit decay heat removal system.

As of the beginning of 2003, 95% of the CEFR detailed design was finished, and 70% of the components had been ordered, of which 20% have been delivered to CIAE. The status and further planning of the construction works were as follows:

- Reactor building construction had reached the top, i.e. 57 m above ground;
- More than one hundred components and shielding doors had been installed;
- Systems installation works was to start in March 2003;
- Sodium procurement was to start later in 2003;
- Equipment and system tests was to start later in 2003;
- Reactor vessel sodium filling is scheduled for 30 May 2005;
- First criticality is scheduled for the end of 2005.

2.2. France

For many years, France was involved in an important sodium cooled fast breeder reactor R&D, which has seen the construction of Rapsodie (40 MW(th)), Phénix (350 MW(e)) and Superphénix (1200 MW(e)). Given the extensive experience available, R&D for the liquid metal cooled fast reactor option appears to be no longer necessary, and efforts are instead concentrated on a new R&D programme on gas cooled reactors. However, it is clearly recognized that the knowledge and experience gained from the liquid metal cooled fast reactor development must be preserved. Hence, knowledge preservation efforts in France are directed towards the creation of a database on R&D and reactor design, as well as on the operational experience of the French fast reactors. This database does not consist of a mere compilation of technical data and results, but it includes also the collection of interviews of senior engineers involved in the design and/or operation of Rapsodie, Phénix, or Superphénix. Another area where continuing efforts are necessary is sodium technology in support of Superphénix decommissioning.

Within the framework of the 30 December 1991 law concerning management of long lived radioactive waste, CEA is implementing an irradiation programme in Phénix. As outlined below, Phénix operation will resume in June 2003. It is planned to operate the reactor for another 6 irradiation cycles, corresponding to a period of approximately 5.5 years (at 2/3 nominal power). The main scope of this irradiation programme is basic data acquisition for material and innovative fuel testing, plutonium incineration, as well as minor actinides and long lived fission product incineration and transmutation.

CEA is also continuing validation work for the re-criticality risk analysis code SIMMER III. An important aspect of this research is the analysis of the molten and boiling mixed pool behaviour of taking into account the CABRI/Raft ball-trap experiments.

As already mentioned, Phénix was restarted in June 2003. The reactor was shut down in November 1998 for plant inspection, normal maintenance, and renovation works in view of its lifetime extension. The renovation works comprised buildings seismic reinforcement, sodium fires protection measures, installation of new emergency cooling circuits, and steam generator (SG) repair after the detection of a significant crack on one SG module. Inspection work comprised the ultra-sonic inspection of the welds of the core support conical skirt, and the visual inspection of the hanging structures of the vessel, and of the upper internal structures of the reactor block (above-core structures and subassembly heads).

2.3. India

As of 31 March 2002, the installed electric capacity increased to 105 GW(e). The power generation in the period April to December 2002 was 397.6 billion kW·h that is 3.7% higher than in the corresponding 2001 period. The plant load factor during April to December 2002 was 71.1% for thermal power, and 78.7% for nuclear, respectively. While thermal and nuclear power generation were higher by 6.3 and 0.6% respectively, hydro-power generation decreased by 9.6% for the same period.

The Indian nuclear electric capacity of 2.77 GW(e) consists of 2 BWRs of 160 MW(e) each and 12 PHWRs of various capacities up to 220 MW(e) each. The performance of the nuclear power stations was very good with an average capacity factor of 85% in the 2001–2002 period.

The following new nuclear projects are under way: Tarapur 3 and 4 (2×540 MW(e) PHWR), Kaiga 3 and 4 (2×220 MW(e) PHWR), Rajasthan 5 and 6 (2×220 MW(e) PHWR), and Kudankulam 1 and 2 (2×1000 MW(e) WWER).

The fast breeder test reactor (FBTR) achieved a higher power level of 17.4 MW(th) corresponding to higher linear power rating of 400 W/cm (compared to 13.4 MW(th) and 320 W/cm earlier). The reactor has logged 1228 days of cumulative operation generating 164 GW·h of energy and 2.17 million units of electricity. 101.5 GW·d/t_{HM} peak burnup has been reached without any fuel failure, and clearance for further operation has been obtained. A subassembly at peak burnup was discharged for examination after cooling. An earthquake of intensity 5.6 on the Richter scale occurred in the sea 80 km south of the FBTR and had no effect on the plant. In the FBTR biological shield cooling system some coils were found leaking which limited reactor power operation. A consultant company was hired to chemically seal the leaks and the system was put back into operation after satisfactory testing for leak tightness.

Works towards the detailed design, manufacturing technology development, and safety clearances for the 500 MW(e) prototype fast breeder reactor (PFBR) project were continued. The radial blanket was re-designed with two rows, and the middle of equilibrium cycle breeding ratio was estimated as 1.049. The analysis of core reactivity fluctuations during seismic event showed that Category 1 design safety limits are met. The scenario of collapse of the core support structure was analysed and suitable preventive design measures implemented. It was concluded that the leak before break criterion cannot be fulfilled for the primary inlet pipe, and the guillotine rupture was assumed for the analysis of this Category 4 event. Transient analyses of the main vessel cover gas system showed that main vessel buckling failure risk is not present. However, as a measure of defence in depth a pressure relief valve is provided. Seismic analyses were completed for the primary sodium pump, the intermediate heat exchanger (IHX), and the rectangular shaped reactor containment building (RCB). The manufacturing technology development of the main nuclear steam supply system (NSSS) components and materials of PFBR was completed. A 75° sector of the roof slab of 12.9 m diameter and 1.8 m height was manufactured, meeting the required specifications.

As of beginning 2003, the review of the PFBR Preliminary Safety Analysis Report (PSAR) by a two level safety committee is nearing completion. Documents required for the environmental clearance were completed, and a public hearing was held for the project on 27 July 2001, being the first time that such a hearing was held for a nuclear power plant in India. Consequent to the clearance from the Tamilnadu Pollution Control Board, the project has been appraised by the Ministry of Environment and Forests, and the formal clearance is expected. A final report was prepared on the investigation of mechanical consequences of a core disruptive accident (CDA) in PFBR based on simulated tests on scaled down models. This covered all the tests done since 1997, and it is concluded that the integrity of the main vessel and roof slab are maintained for an energy release of ~ 1200 MJ, while the structural integrity of the IHX and of the safety grade decay heat removal system is assured up to 200 MJ. Primary sodium release to the RCB under CDA is estimated as 350 kg.

Fast reactor related R&D was continued in reactor physics, engineering development, safety engineering, structural mechanics, thermal-hydraulics, metallurgy, non-destructive evaluation, chemistry and reprocessing. Important works include: obtaining calculation vs. experiment (C/E) factors from flux measurements PFBR radial shield models using leakage neutrons from a thermal reactor; structural mechanical testing of a sodium pipe model, a main vessel model, a core support structure model and a full size T-joint; and stress corrosion crack detection and nitric acid corrosion tests on special steels developed in India. Neutron radiography of FBTR fuel at burnups of 25 GW·d/t_{HM} and 50 GW·d/t_{HM} was completed and showed no major changes in the fuel region. The examination of the fuel at 100 GW·d/t_{HM} is to start.

2.4. Japan

In Japan, there are two fast reactors, JOYO and MONJU.

The loop type experimental fast reactor JOYO (2 loops, no steam generator installed) achieved initial criticality in 1977. The reactor was operated from 1978 to 1982 with the 50 MW(th) (increased to 75 MW(th) after appropriate modification works) MK-I breeder core, and from 1982 on with the 100 MW(th) MK-II irradiation core. Currently, JOYO is being upgraded to the MK-III core that will have an increased irradiation capacity (both increase of the fast neutron flux and of the number of irradiation rigs). The status of these upgrading works is as follows: the modification of the cooling system was completed in September 2001, the core replacement was started in the summer of 2002, and the MK-III initial criticality was scheduled for the summer of 2003. The fast neutron flux will increase by 30% in the MK-III core as compared to the MK-II one, while the number of irradiation rigs will be increased by a factor of two. At the same time, the MK-III core will have higher plant availability due to the reduction of the periodic inspection periods and reduced fuel exchange time. Overall, the MK-III core will lead to an increase of JOYO's irradiation ability by a factor of four as compared to the MK-II core.

The 714 MW(th) (280MW(e)) loop type prototype fast reactor MONJU has three loops. Sodium outlet and inlet temperatures are 529°C and 397°C, respectively. At turbine inlet, the steam temperature is 483°C, and the steam pressure is 12.5 MPa. The primary and secondary coolant pipes are made out of an austenitic 304-type stainless steel. MONJU construction works were started in October 1985, and by March 1991 component installation was completed. MONJU achieved initial criticality in April 1994, and was connected to the grid in August 1995. In December 1995, with the reactor at 40% power, a sodium leak incident took place in the secondary coolant pipe. In the aftermath of the sodium leak incident, JNC, performed a thorough investigation of the cause of the incident. A comprehensive safety review of all aspects of the MONJU design and operation has been conducted. At present, work is concentrated on the countermeasures against sodium leakage. The safety licensing examination by the Japanese government has been completed in December 2002. After the agreement of the local government, the actual improvement work will start and will take approximately 18 months to be completed.

In 1999, JNC started a feasibility study for the early commercialisation of the fast reactor cycle system. The scope of the study is to assess the feasibility of the fast reactor cycle system to meet the following requirements: ensure safety, achieve high economic competitiveness, establish effective utilization of resources, reduce the environmental burden, and enhance non-proliferation characteristics. The feasibility study is checked and reviewed approximately every 5 years. Looking at 2015 as time horizon, the feasibility study is carried out with the goal of establishing the most attractive fast reactor cycle system. In July 1999, JNC and Japanese electric utilities jointly started the first phase of the study, with the participation of the Central Research Institute of Electric Power Industry (CRIEPI) and the Japan Atomic Energy Research Institute (JAERI). During this phase, a wide range of technical alternatives incorporating innovative technologies were reviewed and evaluated. Data and materials required for clarifying the commercialisation strategy were collected and developed, and the most promising concepts for the fast reactor cycle were selected. The first phase of the study was completed in March 2001. The second phase of the study was started in April 2001. During the second phase, the overall consistency of the fast reactor cycle is being assessed, on the basis of some engineering-scale tests. The second phase is scheduled to last till March 2006. At the end of the second phase, the number of promising concepts selected in the first phase will be narrowed down, and the essential research and development areas will be identified.

2.5. Republic of Korea

In Korea, nuclear power plants generate about 39% of the total electricity, and the role of nuclear power plants in electricity generation is expected to become more important in the future years due to the country's poor natural resources. The significance of nuclear power will become even greater, considering its practical potential in coping with the greenhouse gas emission control. The increase of nuclear power capacity eventually raises the issues of efficient utilization of uranium resources and of spent fuel storage. Liquid metal cooled fast reactors (LMR) will eventually be the most promising nuclear power option, considering their enhanced safety based on inherent safety characteristics, their transuranics (TRU) reduction capability (thus contributing towards solving spent fuel storage problems), and their proliferation-resistant actinide recycling capability.

The Korea Atomic Energy Commission (KAEC) revised the National Nuclear Energy Promotion Plan in 1997 and approved the LMR Design Technology Development Project as a national long term R&D programme. The objective of the LMR Design Technology Development Project is to develop the design technologies necessary for the design of an economically competitive, inherently safe, environmentally friendly, and proliferation-resistant fast reactor concept. Based upon the KAEC decision, the Korea Atomic Energy Research Institute (KAERI) has been developing KALIMER (Korea Advanced LIquid MEtal Reactor), a pool-type liquid metal cooled reactor generating 392 MW of thermal power. The first three-year phase of the project was completed in March 2000. During this first phase, the basic computer codes and methodologies necessary for design and analyses have been developed, and an effort has been made to establish a self-consistent conceptual design of the system configuration, arrangement and key features satisfying the design requirements. Efforts have also been made to develop basic sodium technologies, such as measurement or detection techniques, as well as investigations on thermal-hydraulics and sodium fires. The conceptual design of KALIMER was finalized during phase 2 of the project that started in April 2000 and was completed in March 2002. During the second phase, computer code improvements and experiments for model validation were performed. These codes and methodologies were utilized to develop a conceptual design based on the preliminary conceptual design obtained in the first phase. Phase 3 covers the three-year period from 2002 to 2004. The focus in this phase is on the development of basic key technologies and the establishment of advanced concepts. Any innovative trends in the development of fast reactor design will be reviewed thoroughly and included in the KALIMER design if it improves the safety and efficiency. Supporting R&D will be carried out to further validate the design codes and methods.

As an effort to enhance the key LMR technologies, KAERI decided to join the I-NERI, a three-year collaboration program between Argonne National Laboratory (ANL) and KAERI. The objective of this collaboration is to identify and quantify the performance of innovative design features in metallic-fuelled, sodium cooled fast reactor designs. Korea also expects to contribute to the development of Generation IV nuclear energy systems through the worldwide R&D collaboration within the framework of the Generation IV International Forum (GIF). Korea considers the sodium cooled fast reactor (SFR) and the very high temperature reactor (VHTR) to be the most interesting GEN-IV reactor concepts.

The detailed design of KALIMER and the feasibility of the construction are to be examined sometime during the mid 2010s.

3. TECHNICAL SUMMARY

The TM heard ten papers. There was ample time allocated to discussion, and the participants did engage in a very lively dialogue. From the overall deliberations, the following four main themes emerged: safety philosophy, structural integrity assessment, thermal hydraulics, and new concepts.

In the following sections, the countrywise assessment of these four themes is summarized. Following that, the discussions that evolved around the major technical issues addressed in the meeting are summarized. Issues on which an overall agreement was reached amongst the participants are discussed as ‘resolved’, while the others, on which there was no agreement, or where it was difficult to reach an unanimous position, are discussed as ‘open’ issues. Finally, indications on further Member States information exchange and collaborative R&D needs.

3.1. Safety philosophy

3.1.1. China

For CEFR, safety reviewers demand the designers to assess the consequences of the Double Ended Guillotine Rupture (DEGR), including the identification of the most critical location for the rupture. The designers’ safety philosophy is to consider the leakage through a crack area equal to Dt/4 (D: pipe diameter, and t: pipe thickness) as the Design Basis Accident (DBA), and to classify the DEGR of the primary pipe as a Beyond Design Basis Accident (BDBA). However, in the analyses carried out for the DEGR, instantaneous primary coolant pipe (PCP) rupture has been assumed, and the most critical location obtained is the junction of the PCP to the grid plate. Temperature limits used to assess the consequences are such that mechanical characteristics of the cladding and the structure of the fuel assembly are guaranteed and there is no coolant boiling in the reactor core.

3.1.2. France

The safety approach for Superphénix was deterministic. However, for historical reasons, the safety approach followed by CEA later for the European fast reactor (EFR) was a semi-probabilistic one, named ‘lines of defence’ (LOD). The deterministic approach leads to establish a list of events for each category of operating conditions, and to study the consequences of each event in order to verify that these consequences are in the allowed domain associated to the category. The LIPOSO (‘LIaison POmpe SOmmier’) break is classified in the last design basis class (Category 4, hypothetical event). In order to consider this event as an envelope of any fast loss of core-flow, it is assumed that it occurs in the most penalizing reactor conditions, so the break is a DEGR in 1s. To these assumptions, a fuel cladding failure in the hottest core channel is added, in application of the ‘single failure’ criterion. The radiological safety target for Category 4 events is 0.15 Sv at the site boundary, and the proof that the reactor is in a safe state. These safety targets are translated into two decoupling criteria: first, the sodium temperature has to be below the boiling point in all the channels, and, second, the number of cladding failures must be limited.

In the frame of the LOD approach, in-service inspection (ISI) is considered to be the first level (prevention) of the defence in depth philosophy. While ISI was not possible for Superphénix, the MIR device in EFR was designed to measure relative displacements at junctions by ultrasonic telemetry.

3.1.3. India

The DEGR of one of the four pipes in PFBR has been considered as DBE, under Category 4. The worst location is found to be the junction of the pipe with the spherical header. The rupture time is assumed to be very close to zero, which can be considered as instantaneous. Under this event, the Category 4 temperature limits are to be respected, which means that bulk sodium boiling must be avoided, which, in turn, is achieved by limiting the mean subassembly sodium hotspot below 940°C, and the cladding hotspot below 1200°C, thus ensuring a coolable geometry. The design basis criteria such as to maintain the coolable geometry of a fuel subassembly is the same as in other reactors. However, the design safety limits on temperature are higher compared to other reactors. Such a prescription has been made mainly due to the fact that the coolable core geometry must be maintained after the occurrence of the event. However, currently efforts are being made to categorize the DEGR event under BDBE.

3.1.4. Japan

MONJU is a loop type reactor. The primary pipe, which is below the level of the system sodium, is located in the guard vessel whose function is to ensure the adequate sodium level. The leak-before-break (LBB) principle has been demonstrated for this design. Hence, the DEGR is not considered as DBE.

3.1.5. Republic of Korea

For KALIMER, the DEGR of the primary pipe is considered to be an extremely unlikely event. Hence, it is not included in the bounding event category. However, the inherent safety characteristics of KALIMER in the event of DEGR have been demonstrated.

3.2. Structural integrity assessment

3.2.1. China

The following reasons support the claim of high structural reliability of CEFR's primary coolant pipes:

- Large thermal inertia due to large sodium inventory (260 t in vessel);
- Low pressure in the primary coolant pipes;
- Good primary coolant pipes material (09×18G9 stainless steel);
- Large negative reactivity coefficients.

In the CEFR design, the primary coolant pipes are located between the upper and the lower core support. Thus, in the event of a DEGR of a primary coolant pipe, the pressure shock will not damage the bottom of the vessel. The results of comprehensive accident analyses have shown that the integrity of the pressure boundary of the primary vessel is maintained.

3.2.2. France

The prevention of accidents and ensuring structural integrity depends in the first place on design, dimensioning and manufacturing, i.e.:

- Number of pipes and pumps: in the case of Superphénix, each of the four pumps is connected by two primary coolant pipes to the core grid plate in order to minimize the consequences of a pipe rupture;
- The design of the coolant pipe and grid plate connections: their design ensures flexibility in order to avoid stresses in case of differential thermal expansions, pump vibrations, and pump seizure;
- Design and dimensioning considering the appropriate seismic loads;
- Stringent design rules and quality controls for manufacturing and welding;
- Specific tests (e.g. pump seizure) and periodic inspections during reactor start-up and operation (e.g. for the EFR, by means of the MIR device, which enables the measurement by ultrasonic telemetry of relative displacements at junctions).

3.2.3. India

A very comprehensive presentation of the structural mechanics analysis was given, starting from layout optimisation, including dimensioning of pipes. The CASTEM 2000 code has been used for the stress analysis, including fracture mechanics. The French code RCC–MR–1993 is used for checking the design compliance, and the fracture assessment is done according to the French Guidelines A16. The analysis shows that primary stresses due to internal pressure (0.8 MPa) and seismic moments have a margin of about a factor 2, over and above the safety factors provided in the design codes. The vibration loads are found to be insignificant.

Thermal transients following one secondary pump trip impose thermal loading on the primary pipe. All other transients have insignificant effects to this component. Due to these, the cumulative creep-fatigue damage is found to be < 0.2. An analysis, to check whether the critical crack length can be detected by means of core outlet temperature increase, indicates that the temperature rise in the core is very small (< 0.2 K). Hence, core outlet temperature monitoring cannot indicate any leak rate resulting by a critical crack opening. Alternative means are required.

For the PFBR, systematic experimental works have been carried out for assessing the integrity of the primary coolant pipes. This includes fracture propagation tests on plates and circular rings. The crack propagation behaviour predicted by the fracture mechanics analysis according to the A16 procedure have proved to be satisfactory when compared with test data. Tests have been conducted on 1/5 scale models of pipes to estimate the ratcheting strain including ovality of bend under simulated pressure, seismic loadings, as well as deflections corresponding to thermal transient loads. The data required for the analysis have been generated. Finally, it is concluded that ratcheting is not possible in the pipes, even under extreme loading conditions. The finite element prediction of ovalities and strains has excellently matched the experiments. Fatigue tests on the model indicate that cracks were not initiated, even after applying more than 500 plant life load cycles.
In view of the above results obtained for PFBR, it is recommended that DEGR need not be considered as a DBE. Nevertheless, as no leak detection is possible, DEGR is considered as a Category 4 DBE.

3.2.4. Japan

JNC presented the outline of the structural integrity assessment, as performed during the safety licensing process of MONJU. A large number of possible failure modes were investigated for the primary coolant pipes in MONJU, but most of them could be eliminated. Generally speaking, fatigue failure is the most likely failure mode. Therefore, fatigue crack growth evaluation was carried out in the structural integrity assessment. It was demonstrated that the crack growth under design duty cycles was negligible. Even if the crack penetrated under the loading conditions beyond design duty cycles, it resulted in detectable cracks, and there were large margins between the penetrated crack size and the critical crack size. As a design-base-leak path area, the conservativeness of the Dt/4 approach was demonstrated. As far as MONJU is concerned, the successful demonstration of LBB leads to the elimination of the DEGR as DBE, and supports the respective analyses for future fast reactor designs.

JNC presented the outline of the simplified creep-fatigue crack growth assessment procedure that will be applied for the structural integrity assessment of future fast reactor components. Within the framework of validation efforts, some experimental works have been carried out, and the results of the theoretical analyses were compared with the experimental results. Preliminary results of this validation effort indicate that there is the need for some improvement, especially with regard to the creep crack growth analysis method. For the time being, this simplified procedure can be adopted only to simple structures (e.g. plates, tubes) subjected to an idealized load. However, for all practical purposes, and in order to carry out the actual structural integrity assessment, it is necessary to improve the applicability of this simplified procedure to all structures (e.g. nozzles, elbows) under all loading conditions. JNC would propose to extend the validation basis for this simplified procedure by sharing the experimental data to an international benchmark exercise of the various fracture mechanics analysis methodologies.

3.2.5. Republic of Korea

Presently, efforts are being directed on basic studies of the creep crack growth and failure mechanisms with the objective of developing a model for the application of the LBB criterion to the KALIMER design.

3.3. Thermal hydraulics

3.3.1. China

The OASIS code has been used for the thermal hydraulics analyses. A large number of events were analysed. The objective of these studies was to identify the worst-case scenario. Therefore, rupture position, rupture time, and break size spectra must be analysed. Currently, the detailed thermal hydraulics analysis for the primary coolant pipe break accident as a function of the break position is being performed. This study is based on conservative assumptions, as required for Category 4 DBE analyses. The main results of this analysis are the core inlet and outlet temperatures, the break flow rate, the IHX inlet and outlet temperatures, as well as the and clad temperature distribution. The results of the calculations show that the hotspot temperatures for the cladding and for the coolant are below the design safety limits (DSL). These temperatures are lower than the sodium boiling temperature. The regulator accepted the methods applied, as well as the results of the analysis.

3.3.2. France

In the event of the LIPOSO break, the reactor thermal hydraulics depends on the core size and on the number of primary coolant pipes. In the case of Superphénix, with 8 primary coolant pipes, a large core, and the pessimistic assumptions of DEGR, the LIPOSO break leads to a fast over-heating of the sodium in the core, which causes both reactivity and a power increase. With all uncertainties taken into account, the accident is first detected by the measurement of the core sodium outlet/inlet temperature increase (ΔT), which causes fast reactor shutdown 3.6 s after the break.

Extensive studies, including experimental programmes, were performed, leading to the conclusion that the Superphénix safety provisions were able, under the more pessimistic conditions, to cope with this type of accident. Nevertheless, one conclusion, taken into account in the EFR design, was that it would be desirable to have the LIPOSO detection based on the power to flow ratio (P/Q). This would ensure earlier detection, and hence increase the safety margins on sodium boiling. Summarizing the French view, the LIPOSO break accident has two aspects: a mechanical and a safety demonstration aspect. The mechanical aspect is a design question, and it is evident that each country is able to design the pipe connecting the pump with the grid plate in such a way as to ensure that the DEGR is not a credible scenario. All improvements in the design are certainly desirable, but they do not change the necessity to tackle the second aspect, i.e. the safety demonstration under the assumption of the DEGR. With regard to this, the safety demonstration must be made in all cases that the DEGR accident can be detected, and that the reactor operational characteristics remain within the safety limits adopted in the respective safety philosophy.

3.3.3. India

The required thermal hydraulic analysis has been carried out using the one dimensional plant dynamics analysis code DYANA-P. The analysis of the instantaneous sodium coolant pipe rupture requires using two different sets of equations in the primary circuit hydraulic model for the initial steady state and subsequent transient.

Reactor shutdown (scram) parameters based on power to flow ratio, reactivity, power, and central subassembly (SA) sodium outlet temperature, are effective in limiting the consequences within Category 4 design safety limits (DSL). Sensitivity analyses have been carried out for input parameters like location of pipe break, amount of scram reactivity, and the value of the core by-pass flow pressure drop coefficients. In the worst case scenario, when all the above three parameters are assumed pessimistically, core flow reduces to a minimum of 30%, the maximum cladding hotspot is found to be varying between 1284 to 1406 K, and the mean SA sodium hotspot to be varying between 1120 to 1209 K, depending on the first or last scram parameters being effective. It has also been demonstrated that power to flow ratio and central SA sodium outlet temperature scram parameters are effectively available for all power levels from 50 to 100%. At power levels less than 50%, the DSL are not exceeded even without any safety actions The important assumptions made in the one-dimensional modelling are grid plate as a single pressure plenum, and integral momentum balance for rather long flow segments. These assumptions have been substantiated through separate three-dimensional hydraulic analyses of the grid plate, and through pressure wave propagation in the primary circuit (especially in fuel SA), respectively. Experimental investigation on a 1:2.75 scaled model pump under severe cavitation has been carried out to substantiate the assumption of cavitating pumps to give the stated flow.

3.3.4. Republic of Korea

In some advanced pool type reactors, the consequences of a postulated break of the primary coolant pipe can be mitigated by the inherent safety functions. This is achieved through the reactivity feedbacks induced by the Doppler, as well as sodium, radial and axial expansion reactivity effects. The control rod driveline length increase, and the operation of gas expansion modules (GEM) are also important reactivity feedback mechanisms.

The break of a pipe reduces the core flow, which results in power-to-flow mismatch for some seconds after the pipe break occurred. However, the power is stabilized, even though the automatic reactor scram is not activated. The GEM play an important role in providing the dominant reactivity feedback in case of an important core flow reduction resulting in the decrease of the GEM level below the top of the active core. The results of the primary coolant pipe break event analyses performed for the Korea Advanced Liquid Metal Reactor (KALIMER), which adopts several advanced design features, prove both a coolant sub-cooling margin of more than 400 K, and a stable system response.

3.4. Innovative concepts

3.4.1. China

The mechanical design of, and the supporting arrangement for the primary coolant pipes have to be carried out so as to prevent and eliminate the assumption of DEGR. This design will have to be substantiated and certified by detailed mechanical analyses and experiments. A double walled primary coolant pipe design should allow leak monitoring in the inter-space, thus permitting the assumption of leak-before-break, and, at the same time, strengthen the mechanical characteristics of the primary coolant pipes.

3.4.2. India

Various design concepts, to minimise the loss of coolant to core in case of pipe rupture, to improve ISI and to eliminate DEGR from DBE were presented for discussion.

Designs with increased number of pipes with reduced diameters have shown apparent advantages. Apart from bringing down the loss of core flow, it is possible to go for seamless pipes with reduced thickness, by which the structural reliability can be improved significantly. However, a general feeling has been expressed that increased number of pipes may decrease the structural reliability due to the overall increase of the pipe lengths. The incorporation of a flow diode device to increase the friction drop for break flow may also help to increase the core flow.

There are also various in service inspection (ISI) possibilities, by incorporating double wall pipes filled with argon in inter-space, which can be monitored for a sodium leak. As an example, the approach followed in BN-600 was discussed. In this concept, the outer wall is not leak tight, and a tube is incorporated through which sodium can flow from the inter space to the relatively hotter sodium in the space between inner vessel and thermal baffle. During no-leak condition, the temperature of the sodium in the tube is at its immediate surrounding. Any sodium leak in the primary pipe can cause flow in the tube that decreases the temperature of the sodium in it. Thereby, the leak can be detected. However, detailed discussions on this concept with Russian specialists are necessary.

The outcome of the discussion among the participants was that, while the safety implications of incorporating an increased number of pipes require careful considerations of all aspects, the idea of improved ISI, particularly the concept used in BN-600, was welcomed by all.

Finally, a possibility to relegate the DEGR event into the BDBA area is to demonstrate the primary coolant pipe integrity by performing the structural reliability analysis, based on probabilistic methods in conjunction with deterministic studies, evaluating the consequences for the maximum leak through a design basis crack opening area, e.g. Dt/4, as applied for MONJU.

3.5. Resolved issues

This section summarizes the issues on which consensus was reached among the participants.

- Number of ruptured pipes: only single pipe rupture needs to be considered, which itself is considered to be pessimistic.

- Availability of a reliable scram parameter: all the participants consider the power to flow ratio to be a reliable scram parameter. This requires a sodium flow meter in the circuit, which will truly represent the core flow. All the participants are unanimous on the choice of a core by-pass flow meter at the pump discharge, as in EFR.

- Location of rupture for study of consequences: the analysis of the primary coolant pipe rupture event has to be carried out for the rupture occurring at various locations of the pipe, and the worst location, including the junction between the grid plate and the coolant pipe, should be identified for DEGR analysis.

- Analysis method: the participants agreed that one-dimensional analyses to obtain the consequences of the event are good enough. However, flow redistribution amongst the various core subassemblies should be assessed separately. For this, three-dimensional hydraulic analyses or experiments, as performed by India, are required.

- Computer codes validation: considering the complexity involved in the analytical simulation of the entire phenomenon, agreement was reached on the necessity to quantify the approximations involved in the one-dimensional simulation codes with the help of experimental data from mock-up experiments of increasing complexity. This becomes even more important for larger cores.

3.6. Open issues

This section summarizes the status with regard to the contentious issues.

Categorization of the DEGR: there was consensus among the participants that the assumption of DEGR is very pessimistic, due to the fact that the structural reliability of the primary coolant pipe is very high, since it operates at low temperature and relatively low primary stresses. Moreover, the thermal stresses are moderate, and with the choice of highly ductile material, as well as adopting high quality design and manufacturing with high level of quality control. The participants agreed, therefore, that the DEGR is analysed within the framework of a ‘safety philosophy’, not because of any strong mechanistic reasons. The participants from China, India, Japan, and the Republic of Korea propose that DEGR should be categorized as

BDBA, and, provided the design ensures LBB characteristics, the design basis leakage area as Dt/4 should be considered as adequate.

- Time before primary coolant pipe rupture: coolant pipe rupture has been considered to be instantaneous in the DEGR event analyses performed in China, India, and the Republic of Korea. The analyses performed by the French colleagues considered 1 s time before rupture. The assumption made for the time before rupture highly influences the evolutions of the fuel temperatures, as well as of the cladding and coolant following the event. It is agreed that for highly ductile materials, the assumption must be made that crack propagation from critical length to DEGR should take a finite time. However, the choice of 1 s for this period in the French analysis seems to lack justification (it is surmised that this was the shortest time that could be used in the course of the numerical simulations during the 1970s). Therefore, the need for a comprehensive study in determining this parameter is recognized.

- Temperature limits: the analyses performed by the Chinese and Korean specialists have shown that the cladding temperatures are below the boiling point of sodium so that local and sub-cooled sodium boiling are avoided. This is most likely a consequence of the design characteristics of their reactor concepts (i.e. negative sodium expansion reactivity effect, long and small diameter pipelines, and higher flow from the other pumps unaffected by the DEGR). The results obtained by the French and Indian experts showed that only the bulk sodium temperature could be restricted below the boiling point. The argument put forward by the latter is that, since DEGR is a Category 4 event, maintenance of a coolable core geometry after the event is sufficient. Hence, avoiding bulk sodium boiling, and ensuring a coolable geometry in the fuel subassemblies is to be considered as adequate for the DEGR event. The participants concluded that a better resolution on this issue is required.

- Demonstration of leak detection and the LBB criterion: the LBB criterion has been applied for loop type reactors, and thus the DEGR event is eliminated. There was general agreement on the fracture mechanics methodology to be adopted for this purpose. However, LBB justification is not possible for the primary coolant pipes in pool type reactors, because of the on-line leak detection requirement. If this problem can be adequately resolved through improved design concepts, the LBB criterion and appropriate fracture mechanics methodology can then be adopted to eliminate the DEGR as a DBE also for pool type fast reactor designs.

- Innovative design concepts: while no final judgement on particular improved design concepts was possible, the participants agreed that emphasis should be put on the study of those innovative concepts that can facilitate leak detection, ISI, and limit core flow reduction in the DEGR. In this context, it is useful to consider the double wall concept adopted for Russian reactors (BN-600).

3.7. Conclusions

The technical meeting has attained its objectives of in-depth information exchange on the primary pump rupture topic.

The technical meeting identified R&D needs and opportunities for collaboration in the following areas:
- Code validation;
- Innovative concepts;
- Guidelines for safety analyses.

Based on the discussions during the technical meeting, international collaboration was noted several times as being of significant value to the Member States programmes. The IAEA's role as promoter and facilitator of information exchange and collaborative R&D was clearly acknowledged.

More specifically, the participants identified the following R&D needs:

- Analytical benchmark exercises focusing on the validation of computer codes used to assess the consequences of a primary coolant pipe rupture. The benchmark model would be a reference pool type LMFBR design. The objective includes studying the sensitivity of certain input parameters.

- Code validation efforts in the area of crack growth methodologies, including JNC's efforts (both analytical and experimental) to develop and validate a simplified creep-fatigue crack growth methodology.

- Develop innovative concepts with regard to in-service inspection and liquid metal fast breeder reactor primary coolant pipe leak detection.

- Develop guidelines for safety analysis of the primary coolant pipe-rupture event in liquid metal fast breeder reactor.