Some Remarks on Fast Breeder Reactor Block Antiseismic Design and Verification

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INTRODUCTION

The design of LMFBR reactor-block components and piping might be greately affected by the safety requirements related to capability of withstanding a seismic event. Indeed, rather flexible structures and non-negligible clearances (which are desirable to minimize the effects of thermal conditions) may lead to considerable amplifications of the seismic motion of safety - related structures and components. Furthermore, the use of thin walls in the large vessels (again due to the above-mentioned reasons) might lead to buckling.

In some countries, in which large earthquakes are expected, restraints have been applied to the main vessel and / or the core, to make these structures as stiff as possible, and thus, to avoid large seismic amplifications. In other countries and particularly those in which the seismic event is less important, flexible structures have been adopted (unconstrained vessels and cores, besides thin vessel walls), thus especially taking care of thermal deformations. Furthermore, seismic isolation concepts are under development in several countries: these could affect design of major components, analysis methodologies and safety considerations.

Thus, it appeared desirable to compare the seismic design philosophies adopted in the various countries and to analyse the respective numerical and experimental methods and results, in order to define the state-of-the-art on the subject. To this aim, the Specialists' Meeting on "Fast Breeder Reactor-Block Antiseismic Design and Verification" was organized by ENEA in co-operation with the International Working Group (IWGFR) of the International Atomic Energy Agency (IAEA). This meeting was held in Bologna, Italy, on October 12-15, 1987 (Martelli and Forni, eds., 1988).

GENERAL INFORMATION ON THE IAEA / IWGFR SPECIALISTS' MEETING

33 specialists, representing most countries involved in the fast reactor technology (Belgium, the FRG, France, Italy, Japan, the USSR, the UK and the USA), attended the IAEA / IWGFR Meeting. This consisted of seven sessions (Fig. 1).

7 Status Reports were presented in Session 1. They outlined the state-of-the-art of the seismic analysis methodologies adopted for those fast reactors whose construction has most recently been completed or which are in advanced construction. They also provided

information on the methodologies to be applied to those fast reactors which are still in the design phase. Finally, they pointed out the considerable improvements of seismic analysis methodologies with respect to those adopted for some of the first fast reactors.

In addition to the Status Reports, 31 technical papers were presented in Sessions 2-6 (Fig. 1). Finally, Session 7 was devoted to the approval of session summaries (prepared by session chairmen) and concluding remarks and recommendations. All the papers were published by in the meeting proceedingd (Martelli and Forni, eds., 1988), together with session summaries and conclusions and recommendations. They also formed the base of a IAEA Technical Report (Gibert and Martelli, 1989).

The next sections provide remarks based on the session summaries, conclusions and recommendations of the meeting.

STATE-OF-THE-ART

Presentations concerning the Status Reports were grouped into three categories: (1) design in low-seismicity countries; (2) general statements on regulations; (3) review of R & D studies for specific power plants and programmes for the future.

With regard to the first category, the FRG mainly gave a presentation of SNR-2 activity programme (which included work performed by Belgium on core analysis), stressing the importance and adequacy of criteria based on strain limits, as well as the necessity of a more realistic definition of ground motion and design rules. However, the application of the above-mentioned philosophy to high - seismicity countries has to be investigated (although the results of UK tests on piping at high excitations seem to confirm its adequacy), and furthermore, acceptability by licensing has still to be discussed, even in countries like the FRG.

A considerable amount of work was done in the UK, in spite of rather low seismicity of this country: extensive buckling and piping studies have been performed, and detailed core, shutdown system and component analysis is in progress, together with fluid-structure interaction (FSI) analysis. Studies on seismic isolation and in general, structural response reduction (embedment, etc.), have also been initiated in co-operation with Japan and the USA. Furthermore, sloshing studies will also be performed, accounting for non-linear effects. Finally, Probabilistic Risk Assessment (PRA) analysis was initiated.

With regard to the USSR (presentation of the second category), it was pointed out that at present USSR fast reactors are constructed in low-seismicity areas. The design rules do not differ from those usually adopted for thermal reactors. Reactor shutdown is provided for earthquakes exceeding Design Earthquake (DE), which practically corresponds to Operational Basis Earthquake (OBE): this is an automatic scram, similar to the Japanese reactors, as well as Superphénix-1 (SPX-1) and PEC in Europe (on the contrary, manual shutdown is provided in the FRG, the UK and the USA). The USSR is interested in the development of improved methods, for future design purposes.

The first presentation of the third category was that given by France. The French Status Report gave an overview on the numerical and experimental studies performed for SPX-1 (core - partly in co-operation with Italy, shutdown system, buckling of thin shells, etc.) and summarized research items in progress for SPX-2: three-dimensional (3D) effects including FSI, improvement of core analysis by development and use of 3D codes and tests on new shaking tables, piping analysis, probabilistic methods).

It was mentioned that these studies are being performed by France in the framework of the European co-operation on LMFBR development (AGT-9B and AGT-5 groups). Finally, it was pointed out that SPX-2 might be seismically isolated.

Italy presented an overview of studies mainly carried out in support to PEC seismic design. In particular, numerical and experimental analyses performed for the core, core elements and shutdown system were outlined. Furthermore, on-site tests (with various excitation methods) of the reactor building and vessel were mentioned, together with the related detailed numerical studies. It was pointed out that such a large effort was necessary due to the quite severe design earthquakes (0.3 g peak acceleration for the Safe-Shutdown Earthquake - SSE - in the two horizontal directions) and specific PEC features. Feed-backs on PEC design, due to seismic conditions, were shown to be important.

Studies performed by Italy on buckling of thin shells and tests carried out in sodium in the CPV-1 test rig of Brasimone on a SPX-1 primary pump shaft (in co-operation with France) were also recalled: it was pointed out that the CPV-1 rig might be used for further seismic tests on components in sodium. The general interest for LMFBR seismic analysis of studies performed for PEC was stressed.

The Japanese Status Report was devoted to studies performed for both MONJU and the future, pool-type, large-scale plants. With regard to MONJU, the numerical and experimental analyses of the reactor-block (including sloshing), core and shutdown system were mentioned, together with those concerning non-linear effects for piping supports. R & D in progress for future large-scale plants includes rational aseismic design development (embedment, core bottom support, fluid support), improvement of methods for integrity evaluations (post-buckling behaviour, sloshing) and improvements of "siting" technology (seismic isolation, floating plants, soft-rock sites).

Finally, the USA Status Report noted that Department of Energy (DOE) was supporting innovative Liquid Metal Reactor (LMR) design concepts such as SAFR and PRISM. The DOE seismic support was mostly to these two projects, concerning modular small-sized reactors. (This support is now limited to PRISM). Improvements of studies performed, from FFTF to CRFBR and the above-mentioned projects were mentioned, with regard to core analysis, sloshing effects, piping studies. Interest for seismic isolation for the new projects was pointed out. The present and planned DOE programmes address this topic, together with fragility analysis in the framework of PRA, piping studies, core aseismic design and accomodation of site-to-site seismic variability.

During the general discussion, the importance of full-scale tests for the system including fluid effects and non-linearities was emphasized by Japan. Italy also judged that these tests are important, due to the many uncertainties that still affect some data of the calculations. It was agreed, according to the FRG and

France, that reliable computer codes already exist for an adequate seismic analysis of the reactor-block and its components, but some physical phenomena need to be supported by experimental evidence.

WHOLE REACTOR-BLOCK ANALYSIS

Session 2 dealt with problems and solutions involving seismic analysis of entire reactor vessel, internals and support. The Japanese reported the results of analysis of a 1/8-th scale model of a large LMFBR pool-type reactor vessel for shell vibrations (including FSI), sloshing and other effects from seismic loads. This analysis showed good agreement for horizontal excitations and presented good data for vertical excitation effects.

Most persons agreed that the understanding of the horizontal sloshing was much more complete than that of the vertical effects. As to the validity of using a 1/8-scale model to "validate" the analytical approach, it was concluded (1) that the validation is okay as long as the basis is based on physics, and (2) that size effects are important if the analysis depends on certain parametric relationships.

Substructuration of the reactor vessel and internals, in order to use 3D analysis to more accurately predict the seismic responses of the overall system, was discussed in a French paper. Imperfections in the structure involving FSI effects were analyzed. The results, permitted the introduction of a significantly simplified model which reproduces the results of interest quite well. The 3D work shows increasing number of modes compared to 2D axisymmetric approach and additional detail, and cannot be compared directly with the 2D. In general, the 3D results are 20-30% lower than the 2D results. Revisions were made to the seismic response by use of 3D and some unconservativism was uncovered (i.e., used from 2D models). It was pointed out that this work was carried out subsequent to the actual reactor design (SPX-1).

Finally, the analysis made on the PEC reactor-block, which supports the use of a top-core restraint around the reactor core, was presented by Italy. The method deals with the non-linearities in the core taking into account the mechanical interactions between core and vessel. Studies also included examination of the influence of fluid coupling. The conclusion was that the core-vessel coupling, which was quite strong, was adequately handled by the analysis made for PEC.

One point made was that the restraint ring has greater effect (in terms of amplifications) at lower earthquake loads, such as OBE. (This is due primarily to high radial compaction at greater earthquake loads, such as SSE). Differences between transfer functions amplitudes in the SSE and OBE were due to large non-linear behaviour of the restrained core.

The general discussion was mainly devoted to post-buckling behaviour and use of scaled models. The main conclusions are :

(1) The UK supported the Japanese point that the vessels which had sufferred shear buckling had large energy absorption capability. It was also agreed that even after buckling, lateral displacements were small enough so as not to cause concern about ability to shutdown the reactors.

(2) On the topic of problems of using scale models, it was noted that certain factors are omitted from either the full size or scale model, which may have a greater effect on actual system behaviour than that involved in the full size versus scale model issue.

SLOSHING ANALYSIS

As far as linear sloshing is concerned (Session 3), a Japanese study of a MONJU mock-up (scale 1/4) allowed the FINAS code to be validated and efficiency of dip plate to be demonstrated, to reduce the height of sloshing wave. With regard to non-linear sloshing, the impact of the sodium wave on the roof slab was studied again in Japan. Tests for a pool-type reactor were performed on a rectangular mock-up with various roof slab types. These tests led to following results: -

- the maximum pressure acting on the roof slab occurs when the first effective wave strikes at the roof slab;

- the law between this pressure and the impact velocity is linear and depends on roof slab features (type, stiffness);

- the pressure on the lateral walls can be estimated from wave propagation analysis.

Extrapolation from the above-mentioned results was performed for the analysis of a real plant. The fact that the pressure is maximum when the velocity is not maximum was attributed to the effects of gas mixture present in the fluid after the first wave strikes at the roof slab: the presence of this gas mixture is equivalent to a reduction of fluid compressibility.

BUCKLING ANALYSIS

About buckling (Session 3), and in particular that due to pressure, caused by fluid motion, French and Italian studies showed that: -

- comparisons between calculations and static test results stress the need of taking into account the imperfections, and in some cases, plasticity;

- for some geometries , dynamic loads cause an increase of the pressure value that leads to buckling.

An explanation was proposed about the lower efficiency of dynamic pressure compared to the static value. It is related to the rate of the two frequencies affecting the dynamic phenomenon: the frequency of the load and that of the vibrational mode associated with the buckling mode shape. An additional explanation was given about the non-linear variation of pressure versus excitation level: such behaviour is due to coupling of the equations governing the various harmonics of the motions [$\delta = \Sigma \delta_n \cos(n\theta)$], and especially the harmonic of the load and that of the buckling mode. This coupling is due to the defects, and simplified calculations show that the instability level depends on the damping value.

On the contrary, the Japanese showed that the effects of the imperfections are less important on shear bucklig, with respect to the case axial compression or that of pressure load. In addition, buckling initiation was observed to occur at the same loading level in both dynamic and static tests. It appears that analytical and experimental work done in Japan and UK reach the same conclusions.

SEISMIC ISOLATION STUDIES

Seismic isolation (Session 3) plays an important role in the design of the whole plant, including its components and systems. World-wide interest in recent years in this concept is mainly confined to the LMRs, for the reasons that the cost and technical/safety bene - fits are more obvious for LMRs than other types of reactors. Also the necessity of portability and standardization of LMRs is more urgent in the context of a collaboration in Europe and world-wide.

The progress in this field has been reflected in the three papers presented at the meeting. The benefits and real or potential problems still outstanding are better focussed now than even year or two ago, and have been well summarized by the USA. The Japanese introduced several isolation concepts including inshore floating plant which were seriously considered in Japan (a large government funding might be available in Japan to study the seismic isolation concepts in the near future). The UK described a particular analytical study and provided an outline of the R & D programme to investigate the dynamic characteristics and performance of an isolation device.

A consensus of opinion was reached that the seismic isolation concepts offer sufficient advantages to warrant further work to resolve some of the outstanding problems. The remaining problems were considered to be technical rather than feasibility problems.

The question of testing a complete isolation system rather than a single unit which has dominated the experimental work so far has been discussed. Also a need for multi-excitation has been pointed out. These two issues are best answered by installation of seismic isolation devices to real structures in an environment of high seismicity. Such tests are already underway in Japan, where different types of seismic isolation systems can be installed to a building structure. On-site tests using forced excitations, such as those performed for PEC in Italy, might also be useful.

DETAILED CORE ANALYSIS

Session 4 was devoted to core analysis. The design calculations performed by use of the non-linear CEA/ENEA code CORALIE for the seismic verification of the SPX-1 reactor core were illustrated by France. In particular, the main parameters characterizing the seismic behaviour of the core and the subassemblies (like top displacements, shock forces) were given. Special attention was paid to the consequences of core seismic behaviour during the OBE and SSE events on the structural and functional safety verifications. For structural verification, specific criteria, taking into account modification of the steel mechanical properties due to irradiation, were used. For safety verification, the main evaluations concerned the reactivity fluctuations and behaviour of the shutdown systems.

Tests performed in France on the RAPSODIE core mock-up and especially the optical measuring methodology, were also described in detail. These tests, which stressed some important parameters in the core response, were used to validate CORALIE.

Furthermore, the fundamentals of the numerical and experimental studies performed for PEC core seismic analysis were presented by

Italy. The effort made for a complete experimental programme was stressed, starting from the vibrational tests on single simplified element mock-ups, through tests on element groups, shock tests, tests on prototype elements and FSI tests on 7- and 19-element groups. All the test results were used for validating CORALIE and the Belgian code CLASH.

CORALIE was used in Italy for PEC design calculations, together with the mentioned iterative procedure which involves the non-linear detailed core calculations and the linear reactor-block runs. It was shown that design analysis is adequate. It was noted that CORALIE was being used, together with CLASH, for evaluating the safety margins in the design.

The features of CLASH were given by Belgium, together with the typical results of SNR-2 preliminary calculations. The objectives of future R & D at Belgonucléaire were also mentioned.

As far as Japan is concerned, the experimental vibrational behaviour of a cluster of core elements with hexagonal cross-section located in a barrel, under dynamic excitation due to seismic events, was presented. Moreover, the numerical analyses of the test results, performed for the validation of analytical method (VIOLLON), were described. The conclusion was that the single central row model could provide a practical representation for design calculation of the whole core, with consideration of fluid forces and dampings.

Progress of core seismic assessment and current status and future programmes within the European collaboration, especially with regard to code development and verification on the basis of experimental results, was also reported by UK. It was pointed out that the seismic response of the reactor core is now taken into account at the design stage, and that it is expected that a joint European strategy will be developed for assessing seismic response.

Possible failure of irradiated pads was discussed, in particular for the core fuel elements in faulted conditions: about this problem, it was mentioned that, after analysis, this was not found as serious as first thought.

The last contribution to the session dealt with the homogeneization method developed by CEA in France, which approximates the FSI phenomena, in order to allow for the determinations of the global behaviour of complicated systems (such as steam generator tube bundles, nuclear reactor cores, etc.). The method is linear (thus, computer time cost is very limited). It consists in replacing the heterogeneous physical medium (tubes and fluid) by an equivalent homogeneous medium. Its applicability was verified using test results on LMFBR core geometries.

The conclusions were that the homogeneization method allows the seismic behaviour of a LMFBR core to be studied and provides a good approximation of FSI at low excitation level. For the high excitation levels, at which non-linearities due to interactions among the subassemblies become important, a more complete formulation of the equivalent medium must be developed. The interest in such a further development was stressed, because this approach appears very useful for simplifying the analysis.

In the general discussion, it was stressed that core-vessel interaction is an important topic, which includes the development of equivalent linear models. Furthermore, the need of performing core analysis by use of detailed computer codes and that of improving the existing codes by introducing FSI effects were declared by all the participants in the meeting.

SHUTDOWN SYSTEM VERIFICATIONS

The extensive studies performed for the experimental and numerical safety and operational verifications of shutdown systems of reactors which have already been constructed or are in the design phase (SPX-1, PEC, MONJU and CDFR) were due to the complexity of the problem and the necessity of providing very convincing demonstration of the system reliability (Session 5).

The general features of testing methodologies adopted were very similar for all the systems: they were subjected to the displacements calculated in the core and the reactor-block studies, at the axial levels at which they are connected to or interact with the surrounding structures; furthermore, sodium was replaced by water in all the cases.

Besides performing tests, both France and Italy also developed (and applied to the design) numerical models, which were used to interprete the test results and to extrapolate them to the actual reactor conditions, taking into account parameters such as temperature effects.

The UK work dealt with tests performed on the "alternative" system especially developed for CDFR to eliminate most of the earthquake consequences on the system behaviour.

All the tests were successful and the design requirements concerning insertion time of antireactivity in the core were confirmed (although in some cases, some modifications of the original system design were necessary). It was noted, however, that some important parameters could not be taken into account in the tests: the most important look to be the friction coefficients in sodium and the two-directional nature of the seismic load in the horizontal plane.

Questions were also raised as to whether the margins shown by experiments and calculations (with regard to displacements, drop time and structural integrity) are adequate, taking into account the large uncertainties related to the sensitivity of system dynamic response to different types of excitation.

Furthermore, discussion concerned the predictive numerical models that have been developed. At the time being, it is possible to accurately calculate horizontal displacements of the system, taking into account shocks (due to the gaps existing among its parts), as well as fluid coupling. However, the vertical movement itself (which also depends on friction) has not been properly modelled yet: suitable numerical methods must be developed. These methods would be very useful for both extrapolating test results to the actual reactor conditions, and orienting and even qualifying new shutdown system concepts, without the need of complicated tests.

Finally, with regard to shutdown system design, it is noted that it may be difficult to obtain a good compromise between two aims: to get a very high level of shutdown reliability, and to avoid spurious shutdowns in normal operating conditions. Although the alternative shutdown system was considered very attractive from the seismic point of view, concern was raised about the lack of operational experience, in a prototype reactor, of the CDFR system.

CORE STRUCTURAL AND FUNCTIONAL VERIFICATIONS

The procedure developed and used to evaluate the seismically induced reactivity variation of the PEC reactor was also presented, by Italy, in Session 5. In the SSE case, this procedure combines a static evaluation of the maximum reactivity insertion by static and geometric compaction and a time function obtained with CORALIE. For the PEC geometry, it was found that the contribution of vertical motion is small as compared to that of the core compaction. In the final PEC core design, the maximum reactivity insertion was always found to be smaller than 0.9 $ for SSE. Furthermore, the maximum fuel temperature was always well below the fuel melting point.

The two-dimensional CLASH runs had led to higher values of reactivity insertion, with respect to the one-dimensional CORALIE results. However, the CLASH results were smaller than the static results.

As to the structural and functional verifications of the subassemblies (spikes, shrouds and fuel pin bundle), a contribution was provided again by Italy in Session 5. The experimental approach used for PEC is both a quasi-static and a dynamic one. The displacements and shock forces calculated with CORALIE were applied and procedures derived from the ASME III code, case RCCM-R, were used, together with specially developed criteria for irradiated components.

It was noted that the method might overlook one of the major potential failure modes of the subassemblies: the brittle fracture of irradiated tubes. The problem does not exist in the PEC case (for this specific subassembly geometry, shocks only occur on the pads, which are not located in a strongly irradiated zone). However, in a more general perspective, there is an urgent need of experimental data related to impact resistance of wrapper tubes embrittled by irradiation, or artificially.

COMPONENT ANALYSIS

Session 6 was devoted to component and piping analysis. As far as components are concerned, the concept of a fluid-filled gap support system, to be mainly used for internal components suspended from the deck, was presented by Japan. The support is based on the hydraulic coupling between co-axial cylinders separated by a narrow gap. Seal rings control axial fluid leakage. Analytical procedures to evaluate added mass and damping values were outlined; the adequacy of these procedures was demonstrated on the basis of comparisons with experimental results obtained for a mock-up in reduced scale. Extrapolation of the experimental results to the full-scale sizes is judged feasible by the Japanese. No cavitation was observed in the large amplitude experiments.

Furthermore, the experimental on-site tests performed by exciting the PEC main vessel at different axial levels were presented by Italy, together with the related numerical analysis. Tests and detailed analyses (the latter performed in co-operation with France) aimed at improving the simplified calculation models used for whole reactor-block seismic design. This work gave an example of the usefullness of on - site testing techniques for components .

A second work performed by Italy in co-operation with France concerned the seismic tests of the SPX-1 primary pump shaft, carried out in sodium at full reactor temperature. These tests demonstrated the capability of the hydrostatic bearing system to withstand the seismic load. Results at different mass-flow rates through the bearing and revolution per minute values of the shaft were given.

It was pointed out that the CPV-1 rig used for the tests is the only one available at the time for seismic tests in sodium at full reactor temperature. The need of checking the coast-down behaviour of the pumps was mentioned by the USA, because sometimes there can be a marked reduction in bearing effectiveness during coast-down.

Finally, seismic analysis performed for the BN-800 plant was presented by the USSR. Response acceleration levels of the main components were reported. The calculations showed that seismic stresses are quite below the allowable values. Verification of the calculation procedures was carried out by shake table tests of a mock-up which included reactor vessel and internals.

PIPING ANALYSIS

Piping analysis is the last item which was discussed at the meeting (Session 6). The seismic design criteria and procedures used for the design of BN-800 power station were presented by the USSR. It was pointed out that the BN-800 primary piping was designed to be very flexible (2.5 Hz frequency) due to thermal expansion, but the seismic loads did not require specific seismic supporting devices, because of low seismic intensity levels.

The seismic evaluations made in the design stage were based on static methods using Floor Response Spectra (FRS) acceleration values at the first piping natural frequency. The coefficients to be applied to the allowable stress value, when combined normal operation stress and seismic stress are checked, are defined in the standards applicable in the USSR. The results will be verified according to new standards, which require a full dynamic analysis.

The FRG commented on the non-primary nature of the seismic load and inadequacy of the failure criteria based on stresses, which are used at present to evaluate seimic effects on piping. A research work is ongoing at Interatom, which aims at defining methods to seismically qualify piping components by means of strain criteria. Comparison of strain results obtained both by fully non-linear ANSYS analysis and experiments, with those evaluated using the methods which are under development, are quite satisfactory and promising.

Experiments stressed the overconservatism entailed by the present design procedures. No cracks nor fatigue effects were detected (only a few maximum amplitude cycles were experienced by the piping during tests, thus there was no possibility of fatigue).

The need of an increased Quality Assurance (QA) work - if strain methods are applied - was stressed by Japan: this concept is already being applied in Japan to a part of the building code, and it is also intended to be extended to other codes. The need of research work to get a sound definition of strain criteria for the appropriate materials and conditions was also noted; the method and quality of weldments inspection might have to be reconsidered. Finally, it was stressed that problems may arise if seismic loads must be combined with thermal loads using strain criteria.

Japanese dealt with tests performed and analytical procedures developed to account for non-linear pipe supporting structures (such as mechanical snubber free travel or gaps) using standard Response Spectra (RS) analysis. The conclusion was that non-linearities can be taken into account using an iterative procedure to adjust "effective stiffness" of the supports (5 to 6 iterations are required). The iteration loop was performed following standard RS analyses. The method was applied to the MONJU primary piping.

The last contribution was provided by the USA. It dealt with tests performed at ETEC seismic test facility with the aim of investigating the ability of nuclear piping to withstand seismic loads which are well above those permitted by current design criteria. Tests were performed on piping which had been manufactured following standard procedures according to ASME III, class 1, requirements. The objective of the experiment was fully attained: failures occurred only at loading levels which were one order of magnitude higher than those permitted by the code. Localized failures occurred, rather than system collapse.

Since these tests were performed on piping representative of BWRs, some caution must be adopted in extrapolating the results to the case of LMRs: in particular, some effects such as fluid inertia, localized collapses, temperature and low primary pressure stresses shall be taken into account. (The need of additional specific tests on actual LMR piping, and additional theoretical work to extrapolate the results now available on PWR/BWR piping, to the LMR conditions, was also stressed by France).

The importance of the results obtained, in connection with seismic isolation, was emphasized. There was a general agreement that the criteria based on stress limits, currently used to design LMR piping against seismic loads, lead to a decrease of safety, because of the negative consequences of these criteria on the design, and thus, the piping behaviour in normal operating conditions.

CONCLUSIONS

The main items discussed at the IAEA/IWGFR Specialists' Meeting on Fast Breeder Reactor Block Antiseismic Design and Verification have been reported. It has been shown that, in spite of the fact that the existing methodologies are adequate to demonstrate fast reactor seismic safety, improvements are necessary for the future reactors, in order to achieve a better knowledge of some physical phenomena, and thus to allow more realistic, less conservative, criteria and approaches to be adopted.

The aim is a general improvement of the design, which may result not only in a decrease of plant costs, but also in a further increase of the already high overall reactor safety level: indeed, the use of unnecessarily too severe seismic criteria and methods, besides uselessly complicating plant design, might also make reactor behaviour worse in normal operating conditions and complicate plant safety demonstration with regard to accidents which may be much more frequent than the very improbable design earthquakes.

It has been recommended by the participants in the Specialist Meeting that further R & D work should be carried out, from both the numerical and the experimental points of view , mainly on fluid-structure interaction, sloshing, buckling, core and pump seismic behaviour, how to exploit ductility margins, and use of PRA methodology. These studies should lead to the definition of more realistic design criteria and methods.

Furthermore, the benefits , needs and problems of new design solutions, which are potentially capable of reducing the effects of the seismic event on the reactor dynamic behaviour (mainly seismic isolation), should be analysed. Finally, exchange of information on the physical phenomena of interest and advances of R&D studies should be enhanced and existing co-operations should be extended.

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REFERENCES

Gibert, R.J. and Martelli, A. (1989). Seismic Analysis of Liquid Metal Fast Breeder Reactors. IAEA Technical Report (in press), Vienna, Austria.

Martelli, A. and Forni, M., eds. (1988). Proceedings of the 1987 Specialists' Meeting on Fast Breeder Reactor-Block Antiseismic Design and Verification. IAEA Report IWGFR-65 , ENEA, Bologna, Italy.

COUNTRY	S.1	S.2	S.3	S.4	S.5	S.6	TOTAL
Belgium	-	0	0	1	0	0	1
FRG	1	0	0	0	0	1	2
France	1	1	1	2	1	0	6
Italy	1	1	1	1	3	2*	9
Japan	1	1	4	1	1	2	10
USSR	1	0	0	0	0	2	3
UK	1	0	1	1	1	0	4
USA	1	0	1	0	0	1	3
total numbers	7	3	8	6	6	8	38

Fig. 1 : Contributions of the different countries to the various sessions of the IAEA/IWGFR Specialists' Meeting (*= joint Italian - French papers, prepared and presented by Italy; S= session number).

- Session 1 : Status Reports ;
- Session 2 : Whole Reactor-Block Analysis (General Items) ;
- Session 3 : Sloshing and Bucking ; Seismic Isolation Effects ;
- Session 4 : Detailed Core Analysis;
- Session 5 : Shutdown System and Core Verifications ;
- Session 6 : Component and Piping Analysis.