DYNAMIC RESPONSE OF FAST REACTOR PRIMARY CONTAINMENT

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SUMMARY

Dynamic response of LMFBR primary containment to the hypothetical explosive accident is discussed. The amount of experimental and analytical effort given to the problems in Czechoslovakia is presented.

The experimental program consists of a sequence of tests using models of increasing complexity in fast reactor primary structure. As the energy source, high explosives are used. The response of reactor materials subjected to dynamic load is investigated on an experimental facility based on "Hopkinson-Davies bar" which makes possible to study the influence of strain-rate, temperature and irradiation on material properties.

Theoretical effort of the Power Research Institute is based on the mechanistic approach. For the initial phase of the hypothetical accident, the programme SAS1A was adopted. The course of nuclear excursion is solved by the TWEXCO code. Dynamic response of fast reactor primary containment to the core disruptive accident is treated by the CEFRA code. Both programmes are two-dimensional (*r-z*) Lagrangian hydrodynamic codes. They treat the media under consideration as compressible, inviscid and non-heat-conducting fluids. Hydrodynamic equations and equations of state of reactor materials are used for the calculation of the shock waves propagation, the damage produced by these loads and the motion of coolant. They used plate-shell equations to determine the dynamic response of the reactor containment. Containment materials may be elastic-plastic, strain hardening and strain-rate sensitive. The plug and cover are treated as rigid-body model.

In the CEFRA code, the ANL parametric model of molten fuel-coolant interaction and a method for "hour-glass" distortion damping was incorporated. The code calculation has been extended to cover the shock waves, the liquid momentum and the early stages of the gass bubble.

The validity of the codes performance has been verified through experiments. Some results of this comparison and application to a 600 MWe fast reactor project are presented.

1. Introduction

This paper describes the Czechoslovak approach to fast reactor primary containment problems. It is based on the assumption that the safety assessment of the containment requires an analysis of hypothetical accidents.

Current Czechoslovak trends on primary fast reactor containment involve both theoretical studies and experimental programmes employing chemical explosives to provide calibration of computer codes which can then be used on fullscale reactor containment calculations.

The experimental programme consists of a sequence of tests using models of increasing complexity. The properties of model or reactor material are obtained with the help of "Hopkinson - Davies bar" apparatus.

The theoretical effort is based on the mechanistic approach. In recent years, a series of codes has been developed in the Power Research Institute. This paper describes mainly the CEFRA [1] code which can determine the dynamic response of fast reactor primary containment.

Comparison of the CEFRA code with experimental results and some recent improvements in the code will be discussed. The code system was applied to a 600 MWe fast reactor accident analysis and some results of the analysis will also be reported.

2. Experimental work

The experimental programme on fast reactor primary containment scale models using chemical high explosives was intended to predict reactor damage and containment potential on the base of TNT energy equivalent. The experimental work consists of two steps :

1. Preliminary research on small scale models. Bare charge in water, rigid or flexible cylindrical vessels.

2. Research on fast reactor primary containment scale models. Models including internal structures. Details of specific experiments will be modified to satisfy technical and/or economic constrains. At first, the coolant will be simulated by water, later sodium will be used.

Up to now, step 1 has been fulfilled. Plastic high explosive Semtex 1A was used as the energy source. Experiments studying the reactor plug-jump problem have been also carried out.

The work in step 2 will soon be started. Models of 600 MWe fast reactor containment are being prepared (1/12 linear scale). Effort will also be directed on the development of a chemical energy source which would simulate nuclear excursion and/or MFCI, but we believe that the development of a computer code system will eliminate the need for such scaled energy source. However, only an experimental programme can provide a true test of the capability of the codes.

A knowledge of the stress - strain characteristics of both reactor and model materials at high strain rates is essential for both direct assessment of experimental results and calculational purposes. These properties must also be known at high temperatures and under neutron irradiation; to gain them the "Hopkinson - Davies bar" apparatus was used. The principle of this device is evident from Fig. 1. The sample of a material in the form of a cylinder with radius r and height l_0 is placed between two steel bars. The loading stress wave $\sigma_i(t)$ is excited at the end of the test bar by the impact of the steel cylinder of length L, accelerated by a pneumatic device up to the velocity v_0.

The stress wave σ_i impinging on the sample (see detail A in Fig. 1) is partly reflected as wave σ_r and partly transmitted through the sample into another test bar as wave σ_t. The time-dependencies of the stress waves $\sigma_i(t)$, $\sigma_r(t)$, $\sigma_t(t)$ are detected by radial capacitance transducer placed on the test bars. The set of the curves $\sigma_i(t)$, $\sigma_r(t)$, $\sigma_t(t)$ contains a complete information about the behaviour of strain rate $\dot{\varepsilon}(t)$ and deformation $\varepsilon(t)$ according to the relations [2], [3] :

$$\dot{\varepsilon} = (\sigma_i + \sigma_r - \sigma_t)/\rho c l_0 \tag{1}$$

$$\varepsilon = \left[\int_0^t (\sigma_i + \sigma_r - \sigma_t)\,d\tau\right]\Big/\rho c l_0 \tag{2}$$

$$\sigma = \sigma_t = \sigma_i - \sigma_r \tag{3}$$

where ρc is the specific acoustic impedance of the test bars.

To determine the stress σ in the sample we must necessarily take into consideration its space dependance.

In work [4] it was shown that eq. (3) is valid for :

$$l_0 \leq 0.04\, c_s \tag{4}$$

where c_s is the propagation rate of the stress wave in the sample material and λ is the wave length.

The effect of the loading stress wave can be described by the parameter S_i according to the relation :

$$S_i = \sigma_{im}/\tau_{im} \tag{5}$$

where τ_{im} is the time needed to achieve σ_{im} ;

This description is correct for the short loading wave σ_i of an approximately sinusoidal form ($\lambda \doteq 30\ \mu sec$).

The main characteristic determining the plastic material properties is the dynamic yield point σ_{yd}. This is defined at the inflexion point of the curve $\dot{\varepsilon}(t)$ [5] .

In Fig.2, the curves $\sigma(t)$, $\varepsilon(t)$ and $\sigma(\varepsilon)$ are given for a material A2. In Fig.3, the function $\sigma_{yd} = \sigma_{yd}(\dot{\varepsilon}_{yd})$ is given. The values of $\dot{\varepsilon}$ are within the range $< 10^{-4}, 10^{3} >$ sec^{-1}. We can see that the yield point increases with the strain rate. In Fig.3, the same function for irradiated material is shown.

The small volume of the tested sample enables the study of the influence of neutron irradiation together with some other metallurgical factors (e.g., ageing).

We can also perform all tests at lower or higher temperatures : from -196 °C up to 200 °C. A new heating system will be supplied to carry out the tests up to 700 °C.

We must also take into consideration some limitations of the given method. This experimental procedure enables the study of only one - dimensional pressure stress waves. The stress wave length λ is limited by the radius of the bar.

In order to study the effect of shorter waves, we must change the radius according to the relation :

$$r \leq 0.1 c \lambda \tag{6}$$

The height of the sample is also limited by the condition (4).

3. Theoretical analysis

A series of codes has been developed in the Power Research Institute to assess the time development of hypothetical fast reactor accidents.

The initial phase of the accident analysis requires the proper description of complex accident phenomena : coolant dynamics, fuel dynamics, fission gas release, molten fuel-coolant interaction. This phase of accident is solved by the modified SAS 1A code [6] . For better description of the accident phenomena involved, calculational modules for fission gas release and MFCI will be incorporated into the code [7].

As soon as prompt criticality or high pressures are achieved, the course of accident is calculated by the TWEXCO [8] disassembly code which is of the VENUS [9] type.

When the neutron kinetics deceases, the computation is continued by the CEFRA code which resembles REXCO-HEP code. Recently some significant modifications of the code has been made. They include a model for MFCI and a method for "hour-glass" grid distortion damping. It enables calculation of the dynamic response of fast reactor primary containment to any core disruptive accident up to early stages of the gas bubble expansion.

3.1 Verification of the CEFRA code

The results of the CEFRA code were compared with model experiments reported in [10] in which bare charges of plastic high explosive Semtex 1A in water were used. The details of the comparison are presented in [11] .

At first, the experiments with rigid cylindrical vessel were treated. The geometry of the experiments is shown in Fig.4 along with the Lagrangian grid used in the computations. The pressure was measured at the tank wall directly opposite to the charge and at the center of the bottom with piezo--electric transducers.

In computation, the TNT equation of state (constant-beta form) was used for Semtex 1A. In Fig.5, the CEFRA computed pressures on the tank wall are compared with the measured values for the charge of 10g. The computed peak shock pressure is 1400 bars, about 30% greater than the measured value of 1100 bars. This discrepancy is mainly due to the equation of state of the explosive but the computed pressure profile agrees quite well with the measured one. The same results were obtained for rigid vessels partially filled with water. Results obtained suggest that the pressure history at any point in (r-z) geometry can be computed by the code provided that adequate equations of state of materials are used. Therefore many forms of equation of state were included in the code.

The second series of comparison was made for flexible stainless steel cylindrical vessels (0.1 cm thick). In Fig.6, the time dependent deformation of the vessel wall opposite to the 10g charge is shown. The computed time--dependancy of wall deformation is equivalent to 15g charge. During these calculations, the validity of plate-shell equations of the vessel was tested.

3.2 Modification of the CEFRA code

The CEFRA code is a Lagrangian code and thus a gross mesh distortion can limit the time of accaptable solution and it may become impossible to continue calculation for longer times, especially in liquid momentum phase. In many cases, the mesh distortions are not of physical origin, but are rather induced by the numerical method used. It is the case of the so called "hour-glass" distortions. The procedures of overcoming these distortions were proposed in [12] , [13] . The programme CEFRA is provided by a method for "hour-glass" distortion damping. It uses damping rotating forces like the programme TENZOR [14] . From Fig.7, the damping rotating force Q at the center of the side 34 will be :

$$Q = K c \rho \Delta x (\omega_{14} - \omega_{23}) \tag{7}$$

$$\omega_{14} = -\left[(u_1 - u_4)(z_1 - z_4) - (v_1 - v_4)(r_1 - r_4) \right] \Big/ l_{14}^2 \tag{8}$$

where u , v are velocities in r and z direction, l is length of a side, c is local sound speed, ρ is material density and Δx is the characteristic transverse length of the Lagrangian cell.

The constant K must be determined empirically. In Fig.8, the Lagrangian mesh grid at various times with and without the "hour-glass" distortion damping is shown for 10g bare charge in flexible vessel. For the cases analysed, the best value for K was 0.1.

It is evident that in the charge vicinity "hour-glass" distortions cannot be eliminated (they are of physical origin), but for longer times we are interested in the regions quite far from the charge or core center. In these regions, the proposed method is very effective. For longer times, the Lagrangian cells near the charge or core are also very much distorted, thus accordingly the White convergence criterion, the time step is very small. However, interesting processes occur in the boundary regions, thus the CEFRA code can ignore gross distortion near the center and the computation continue without decreasing the time step.

This approach enables study of the plug-jump problem with the help of the Lagrangian CEFRA code. The verification of the code in this phenomenon is being prepared.

In order to treat the analysis of postulated LMFBR accidents with the CEFRA code for longer times, effects of heat transfer should be included. From these, molten fuel-coolant interaction is most significant. The modified parametric ANL model devised by Cho and Wright [15] was used. It also respects the presence of noncondensable gases in MFCI region and the effect of fuel latent heat of freezing. With this computational module it is possible to study both whole core accident and/or MFCI in smaller configuration such as single subassembly. The MFCI module was incorporated into the code [16] and various sensitivity studies of MFCI are being carried out.

3.3 Application of the codes

The question that remains to be answered is how well the code CEFRA will predict the response to a hypothetical core disruptive accident postulated for an LMFBR. The scale models for a 600 MWe fast reactor are prepared and thus proper comparative studies will be done soon. Without this comparison we applied the TWEXCO and CEFRA codes to the preliminary analysis of hypothetical accidents of this reactor.

The analysis of the causes of hypothetical accidents showed that two categories of the initial core accidents should be analysed :

- coolant flow decrease,
- positive reactivity insertion.

Credible starting mechanisms of these initial events are well-known. The course of the accidents in the predisassembly phase was described by the SAS 1A modified code without models of fission gas release and MFCI. To evaluate these phenomena we performed calculation of disassembly phase by the TWEXCO at various conditions at the moment of the pressure buildup in the fuel. The reactivity insertion rate at prompt criticality was ranged from 60 to 100 $/sec.

The results of the disassembly phase are presented in Table I. Maximum value of the core pressure is quite small (up to 18 000 bars) in spite of the pessimistic value of the reactivity insertion rate. The available work (determined as isoentropic expanse to 1 bar) is large and accident intensity (available work per thermal reactor power) is about 2.5.

Disassembly of "wet" cores gives smaller values of available work than of "dry" cores, but the amount of liquid fuel (see Table II) is considerable and the available work will be increased by violent MFCI. However it must be noted that a considerable part of fuel in case of "wet" core disassembly is vaporized at the end of the disassembly process - in the range of 10-25%.

On the base of the TWEXCO calculation we evaluated primary containment by the CEFRA code. It meant to investigate the consequences of two hypothetical core disruptive accident :

- disassembly of "dry" core,
- disassembly of "wet" core and of delayed MFCI.

Computational reactor primary system model for the code is shown in Fig.9. This model retains the main components of primary containment. In the analysis, two-phase equation of state for the fuel was used. The behaviour of other materials was described by the Murnaghan equation. The whole core MFCI analysis was made for various values of the mixing time and of the effective particle radius.

During the analysis the effect of structural components was studied. The results showed that the stress acting úpon the reactor and guard vessel resulted in small plastic strain mainly due to the energy dissipation of shock wave which proceeds through sodium.

The most significant is the second destructive component of energy released - sodium momentum, i.e., kinetic energy of sodium which moves towards the reactor plug or platform. In the case under consideration the effect on the reactor plug is the most important. Due to the design of primary system, the energy released during a hypothetical core desruptive accident is directed above the reactor core and results in large kinetic energy of upper sodium slug and therefore in great energy which must be absorbed by the plug holt-down bolts. The results of the analysis gave requirements for designers.

4. Summary

In the paper, the Czechoslovak effort in the field of fast reactor primary containment was presented. Its direction is the same as in most other countries.

For the analysis of hypothetical fast reactor accidents, a series of codes was developed. Contemporary with analytical work, the experimental research is performed.

The comparison of the CEFRA code with the experimental results suggest that this direction of research is very promisible and hopeful. The codes can be effectively used to advise designers of fast reactors.

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Table I. Results of disassembly calculations for a 600 MWe fast reactor

Core type	reactivity ramp	Energy released	Kinetic energy	Available work	Maximúm pressure in core center	Excursion time	Initial fuel temperature
	$/sec	MJ	MJ	MJ	bar	msec	°K
dry	100	17463	90,09	3764	17850	2,882	4000
dry	80	15880	75,32	3067	15060	3,167	4000
dry	60	14033	57,52	2384	11760	3,585	4000
dry	100	5625	7,99	818	1608	4,038	5000
dry	80	2347	2,10	335	334	5,646	5000
wet	100	8012	18,51	670	6764	2,926	4000
wet	80	7424	12,03	559	6135	3,230	4000
wet	60	6821	7,68	476	4238	3,6867	4000
wet	100	14824	80,59	1846	15020	2,884	3000
wet	80	13852	46,59	1472	13410	3,178	3000
wet	60	12663	30,46	1112	10400	3,601	3000

Table II. Mass of liquid fuel after disassembly of the "wet" core

Reactivity ramp	Mass of liquid fuel	Average temperature of fuel
$/sec	kg	°K
100	7450	5560
80	5130	5570
60	5980	5000

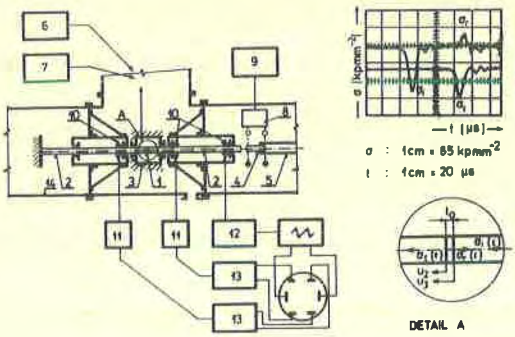

Figure 1 Scheme of the experimental device for dynamic tests.

1-sample of material
2-test bar
3-thermostat
4-punch
5-pneumatic accelerating device
6,7-media for cooling or heating
8-photoelectric velometers
9-computer
10-strain gages
11-preamplifier
12-start and delay circuits
13-amplifier

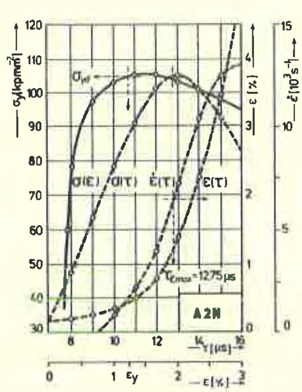

Figure 2 Experimental results of dynamic tests for A2 steel

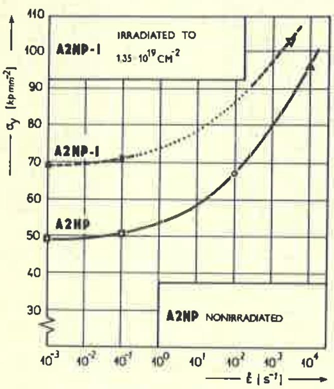

Figure 3 Dependency of the yield point on the strain rate for irradiated and nonirradiated A2 steel

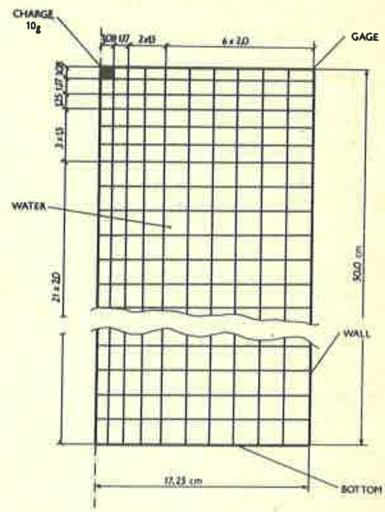

Figure 4 One quarter of 10g bare charge model test. Lagrangian mesh used in the computatäon is also shown

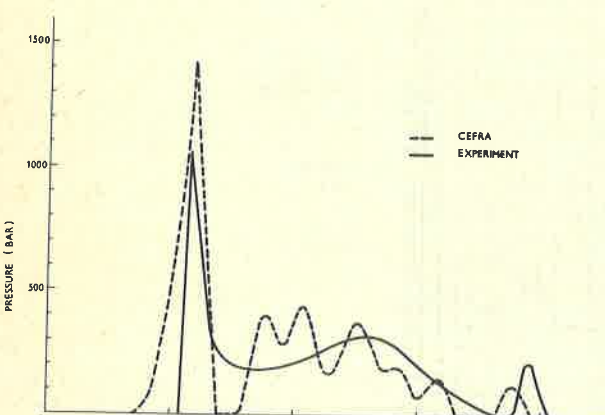

Figure 5 Comparison of computed and measured pressures in bare charge model test

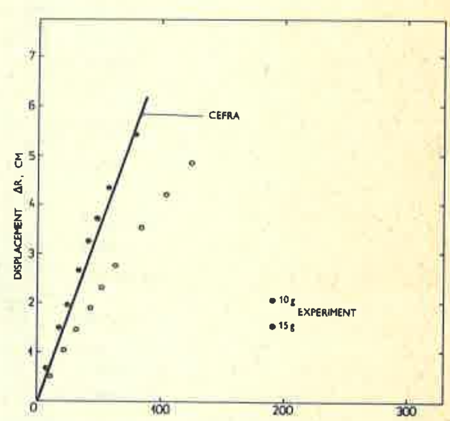

Figure 6 Displacement of flexible steel vessel wall for 10g bare charge experiment

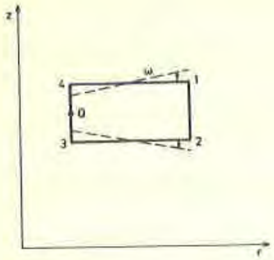

Figure 7 Notation used for a method of "hour-glass" mesh distortion damping

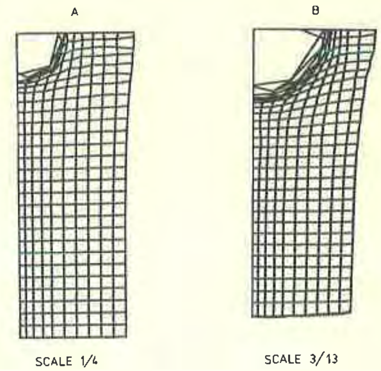

Figure 8 CEFRA simulation of 10g bare charge experiment with flexible vessel :
a) "hour-glass" distortion at time 344 μsec
b) mesh-regularizing (K = 0,3) at time 886 μsec

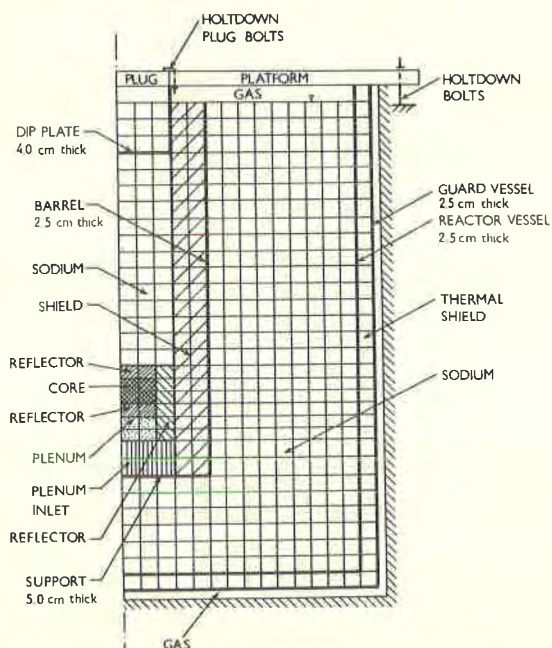

Figure 9 CEFRA model of a 600 MWe fast reactor.
15 division in r direction : 2x51,25, 1x55,0, 2x52,5, 8x50,8, 1x51,1, 1x50,0 cm, 2x37,5,
31 division in z direction : 7x50,0, 2x55,0, 1x40,0, 2x37,5, 1x40,0, 16x50,0 cm.