LMR large accident analysis method

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ABSTRACT

This paper presents the methodology used by NOVATOME to predict the mechanical consequences of large accidents which are considered in the design and beyond design of Liquid Metal Reactor: core disruptive accident, sodium-water reaction and pipe whip after a failure. These accidents lead to transient dynamic non linear phenomena. Their analyses is based jointly on an experimental and theoretical approach.

ABBREVIATIONS

ACS Above Core Structure
ALE Arbitrary Lagrangian Eulerian
BDB Beyond Design Base
CDA Core Disruptive Accident
DEGF Double Ended Guillotine Failure
DHX Direct (Decay) Heat Exchangers
EFR European Fast Reactor
FSI Fluid-Structure Interaction
IHX Intermediate Heat Exchanger
LMR Liquid Metal Reactor

NPP Nuclear Power Plant
PC Primary Circuit
RV Reactor Vessel
SGU Steam Generator Unit
T temperature
2D, 3D two-Dimensional, three Dimens.
(P₀, P) (initial) Pressure of CDA bubble
(V₀, V) (initial) Volume of CDA bubble
ε strain

1 - INTRODUCTION

Since the early stage of LMR development, safety concerns led people to devote strong efforts to the analysis of the consequences of large accident such as:

- hypothetical CDA,
- sodium-water reaction,
- pipe whip after a failure,
The common nature of these events is to pertain to the dynamic non-linear phenomenon field. The structures are submitted to severe transient solicitations involving large displacements of solid or fluid masses, non linear material and geometrical behavior: plasticity, shocks, impacts, water hammer.

In this paper, we present the methodology used by NOVATOME to predict their mechanical consequences.

2 - LIQUID METAL REACTOR, DESIGN AND BEYOND DESIGN APPROACHES

Figure 1 presents a scheme of a LMR in the case of the pool type, integrated, reactor such as French Phénix and Superphénix reactors and EFR project. The main characteristics can be observed on this figure: the RV, a thin shell (diameter/thickness ratio about 500 to 1000), contains the whole primary circuit. A secondary sodium circuit carries the heat from the sodium-sodium exchanger, the IHX, towards the sodium-water exchanger, the SGU, where the wall of thin exchange tubes separates the sodium from the vapor-water mixture. Not shown on this idealized picture, independent direct reactor cooling circuits are implemented with sodium-sodium DHX exchangers immersed in the RV and sodium-air exchangers out of the PC.

When dealing with LMR dynamic behavior, one shall consider two main features of these reactors: first, most of the structures can be considered as thin shells because of the low pressure conditions within the circuits. Second, FSI plays an important role due to the great quantities of liquid coolant, sodium, contained in the above mentioned thin shells.

For the LMR, as for other types of nuclear reactors, the design practices are ruled by a design code. In France, it is the RCC-MR code [1], developed by CEA, NOVATOME and EDF. The design process is based on the idea that the events are classified following a decreasing probability of occurrence. The designer distinguishes four situation categories in correspondence with three criterion levels A, C and D. A similar classification is used in the U.S. ASME code. When the event becomes less and less likely, the criteria become less and
less stringent. Generally the design process does not consider events the probability of which is negligible, (less than $10^7$ per year). Such accidents lie in the BDB region.

But, whatever the estimated probability of BDB accidents, the safety approach led to analyze their consequences and to demonstrate that they are reasonably acceptable.

The large accidents we will present in the following belong to the design (for sodium water reaction) and BDB (for CDA and pipe whip) domains.

3 - CORE DISRUPTIVE ACCIDENT

3.1 - Description

In a LMR, some highly hypothetical accidental scenarios may lead to a core accident. A pressurized bubble (CDA bubble), containing a mixture of fission gas and of sodium and molten fuel vapor, is created inside the damaged core. When the CDA bubble expands within the PC, it pushes violently the primary sodium against the containment wall, RV and roof. The inner structures (ACS, PP, IHX, inner vessels, neutronic shields) moderate the motion and dissipate a significant part of the bubble energy by means of their plastic deformations. The cover gas volume below the roof is compressed and a sodium slug may impact against the roof and the upper part of the RV. For high energy CDA bubble, plastic deformations occur along the RV, mainly in its upper part, where a bulge is formed. The central part of the roof submitted to the high pressure slug impact rises up and some sodium may flow on its upper surface through the connections with the rotating plugs or the components penetrations. High displacements of pipes connected to the exchangers and lift-off risks of the plugs and components may also be considered.

CDA is a BDB accident. Nevertheless one must evaluate the capacity of the reactor to sustain with reasonable margins the loadings due to this accident.

3.2 - Methodology for mechanical consequences evaluation

Due to the great uncertainty on the hypothetical CDA bubble characteristics, the regulatory body has prescribed a given envelope load. It was defined as the energy released by the bubble $E_{bubble} = 800$ MJ. This energy is the work done by the bubble from its initial, $V_0$, to its final, $V_f$, volume:

$$E_{bubble} = \int_{V_0}^{V_f} PdV$$

The final volume, $V_f$, is the sum of the cover gas volume and of the increased volume of the deformed RV, the sodium being quasi-incompressible. Then, the released energy is a result of the bubble expansion. The requested objective to release 800 MJ had obliged to perform iterative analyses.

An important experimental and theoretical approach has been developed by the CEA in order to demonstrate the resistance of the reactor to the hypothetical CDA. We can summarize this as follows:

* **Experiments**: it was firstly necessary to calibrate explosive charges in order to be able to release in a given mock-up the scaled energy, MANON tests [2]. Then several mock-ups of the reactor were fired in order to provide data for a sufficient validation of the computer
code SIRIUS, MARA tests [3]. Finally a very detailed demonstration mock-up was fired in order to prove the resistance to the 800 MJ release, MARS test [4]. Figure (2) presents two of these mock-ups.

* Theoretical: the SIRIUS computer code was developed in parallel and validated by comparison to the MARA (8 and 10) tests. It was based on a Lagrangian description of the fluid and of the structures. The hypothetical CDA of 800 MJ was finally calculated for the reactor geometry.

Figure 2, MARA10 (left) and MARS (right) mock-ups

In the recent years, NOVATOME had to make new estimations, mainly in the frame of the EFR project, and we took advantage of the availability of the PLEXUS code from the CEA CASTEM package [7]. The development of the ALE technique in PLEXUS allows much easier and powerful calculations than the elderly SIRIUS code. We had to validate the PLEXUS code in the domain of CDA calculation.

This was done by comparison with the above mentioned experimental tests MARA [5] and MARS, [6]. It provides us a set of modeling rules:

- 2D axisymmetric models which take into account the various fluid media (sodium, bubble and cover gas) and FSI effects shall be sufficient.

- The containment structures (RV, roof) shall be described accurately, particularly the roof flexibility (even if very low)

- The modeling of inner structures (ACS, inner vessels and baffles, core surrounding shells...) is important and their plastic deformations allow to reduce the estimate of the containment damages.

- The material behavior shall be sensitive to the strain rate at the mock-up scale, but much less at the full reactor scale.

Development are still under progress in order to describe the effect of the components immersed in the sodium (PP, IHX) and of the neutronic shields. They create a porous 3D barrier against the sodium and bubble flow. At the present stage, we obtained reasonably conservative estimates of the CDA loads on the containment, [6].

Our present approach of the CDA is to check the risk of cliff-edge effects. It consists in determining the maximum CDA bubble energy which can be sustained by a given design and to verify that the behavior of the containment is not of a fragile-type when the initiator bubble is increased up to the limit value.
The criteria used are not provided by the design codes for this BDB accident. Generally, a strain-related criterion is retained: \( \varepsilon < 5\% \) to check the integrity of the structures. Numerous experimental tests proved this to be conservative. But particular attention shall be paid to the failure mode of the considered structure.

4 - SODIUM WATER REACTION

4.1 - Description

The only place in a LMR where sodium and water are close is the SGU bundle, Figure(1). Assuming a leak of one exchange tube, the steam-water mixture inside the tube will come into contact with the secondary sodium. This will initiate the sodium water chemical reaction \( Na + H_2O \rightarrow NaOH + \frac{1}{2} H_2 + \text{large heat release} \). This reaction is characterized by its high temperature, up to 1000°C, and by an important release of hydrogen gas. The reason for the initial leak should be associated to chemical corrosion, vibrational wear or fatigue, initial defects or to a combined cause.

One can separate the accident into two phases. In a first phase, the high pressure hydrogen gas bubble expands violently in the secondary sodium. It generates an acoustic wave which propagates along the secondary circuit towards the IHX. The duration of this phase is about 100ms. During this phase, rupture disks, implemented on the loop in order to drain quickly the SGU and stop the reaction, will be broken by the pressure wave. In a second phase, the gas and soda produced by the reaction will flow along the loop, to the discharge line. This is the mass transfer phase.

Contrarily to the CDA, the sodium-water reaction has been observed in operating LMR, see the Prototype Fast Reactor accident at Dounreay (U.K.) in 1987.

The sodium-water reaction is included in the design process. The integrity of the secondary circuit, and especially of the IHX tubes bundle, shall be verified in the case of an envelope sodium-water reaction.

4.2 - Methodology for mechanical consequences evaluation

In order to evaluate the mechanical consequences of a sodium water reaction in a SGU the following methodology is used:

- Definition of the envelope accident to be considered for the design. This envelope accident corresponds in most cases to a number of simultaneous DEGF of water tubes.

- Evaluation of the time dependent water flowrate in the SGU due to the defined accident. As long as this flowrate depends on the operating condition of the plant (nominal, shutdown,..) and on the position of the rupture in the bundle, a parametric study is performed in order to define an envelope flowrate law.

- Calculation of the transient pressure on the outer shell of the SGU by a local sodium water calculation. For this purpose we use a three dimensional finite element model of the part of the steam generator affected by the sodium water reaction.

- Calculation of the transient pressures in the secondary loop and in the IHX. For this purpose, the loop is modelized with pipe finite elements. Pipes are filled with
sodium and at the reaction zone the previously calculated water flowrate is 
introduced. The sodium water reaction model represents then the injection in the 
loop of hydrogen produced by the reaction. We consider that the reaction is total and 
immediate. Some specific elements like the rupture disks are modelized. With such a 
model we simulate the propagation of acoustic waves and the mass transfer in the 
loop.

The maximum pressures obtained during the previous calculations are used for the 
sizing of the IHX tube bundle, the SGU outer shell and all the other parts of the 
secondary loop.

Water flowrate and pressure calculations are performed by mean of the PLEXUS code 
developed by CEA DMT. In the same model one can include the calculation of the water 
flowrate coupled with the calculation of the pressure and mass transfer in the sodium loop. 
During the reaction the counter pressure in the steam generator will then limit the water 
flowrate. It is also possible to couple fluid with structure in order to obtain the movement, 
the stresses and strains in the loop during the accident. A less conservative evaluation of the 
pressure in the fluid is obtained when the deformability of pipes is taken into account.

The validation of PLEXUS for sodium water reaction simulation has been performed by 
comparison with the other codes or by comparison with analytical calculation (water 
flowrate) and large scale sodium water reaction tests performed during the development of 
the Superphénix SGU.

5 - PIPE WHIP AFTER A FAILURE

5.1 - Description

Pipe whip is a general concern for NPP. In the case of the LMR, one shall distinguish the 
water/steam circuits from the secondary sodium circuit. The first are characterized by their 
high pressure conditions (steam: \( P = 18 \text{ MPa} \) and \( T = 490\degree \text{C} \), feed water: \( P = 22 \text{ Mpa} \) and \( T 
= 240\degree \text{C} \)).

Thanks to its low pressure conditions (less than 1 MPa), the secondary sodium circuit 
cannot be the source of violent pipe whip consecutively to an assumed failure. The analysis of 
the consequences shall mainly deal with the risk of sodium fire.

The pipe whip analyses are not included explicitly in the design approach which shall 
provide sufficient provisions against these events. However, protective measures are 
implemented and dimensioned according to design rules in order to mitigate the 
consequences of such accidents: pipe anti-whip devices, sodium fire protections.

5.2 - Methodology

As mentioned in §5.1, pipe whip analyses are performed for two different type of pipe, the 
high energy steam or water pipes (\( P>2\text{Mpa} \) or \( T>100\degree \text{C} \)), the sodium pipes.

5.2.1 - High energy water or steam pipes

The methodology is based on the application of the design rules, [8]. These rules allow to 
define, on a given line, the location of several postulated break points. Then, for each of these 
points, one must evaluate if it is aggressive for its environment (i.e. if during the whip, the 
pipe can hit a safety related equipment or structure). If it is aggressive, one must check if it is
possible to erect at its location an anti whip device. If not possible a whip calculation is performed by mean of the PLEXUS code [7].

The main hypotheses of this calculation are:

- rupture of the pipe in 1 ms,
- maximum thrust force on the pipe reached 1ms after the break, and maintained,
- elastoplastic behavior of the material is taken into account.

The results of this calculation are the movement and the velocity of the pipe, the risk of impact on a safety related equipment or structure. In case of impact, the impact force can be evaluated by mean of conservative methods. When necessary, a detailed and less pessimistic PLEXUS impact calculation can be performed. The impact force is used to check the resistance of an existing barrier (wall, roof), or to design a protection device.

5.2.2 - Sodium pipes

In case of a pipe break in a sodium pipe, the leak will lead to a sodium fire. The magnitude of this sodium fire is linked to the amount of sodium released by the break. The aim of whip calculation for a sodium pipe is then to evaluate with a good accuracy the evolution in time of the size of a break. The main steps of such an analysis are:

- location of the break point to be postulated at the ‘a priori’ place which maximizes the whip,
- initial hydrodynamic calculation of the flow out of the break considering a DEGF and the operating transient of the NPP in case of pipe rupture,
- simulation of the nominal operating conditions (before the break) which induce thermal stresses in the pipe,
- simulation of the rupture and of the pipe whip by canceling in 1 ms the prestressed state and applying at the break the thrust force due to the flow calculated previously,
- calculation of the movement of the line and evaluation of the break area evolution during time, Figure (3),
- with this break size law, new hydrodynamic calculation and if necessary iteration with whip calculation.

Figure 3, whip of a sodium pipe.

<table>
<thead>
<tr>
<th>Positions at several times</th>
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<tbody>
<tr>
<td>1 : t = 0 ms</td>
</tr>
<tr>
<td>2 : t = 100 ms</td>
</tr>
<tr>
<td>3 : t = 200 ms</td>
</tr>
<tr>
<td>4 : t = 300 ms</td>
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</tbody>
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The whip calculations are performed by mean of the PLEXUS code. The resulting sodium release law is the input data for the evaluation of the sodium fire consequences.

6 - CONCLUSION

The large dynamic accidents considered here pertain to the design or BDB domain. A methodology to analyze their consequences is available. It is based on the powerful PLEXUS software which takes advantage of the important experimental and theoretical program performed by CEA for French LMR development.

The dynamic (fast transient) nature of the solicitation is taken into account as well as the fluid structure interaction. The calculations provide reasonably conservative results and their accuracy has been verified by means of comparisons with experimental tests.

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