THERMAL ELASTO-PLASTIC ANALYSIS IN REACTOR VESSEL NOZZLES

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SUMMARY

In a sodium cooled nuclear fast reactor the thermal analysis has a particular importance as a consequence of the rapid transients and the high sodium conductivity. One of the higher stressed components is the reactor vessel subjected to two different kinds of loads: mechanical (pressure and pipe thrust) and thermal (stationary and transient conditions). In the vessel the nozzles particulary have high stress concentration factors and it is necessary to analyse this area carefully. In a reactor scram there is a high sodium temperature change in a few seconds; then it is very important to verify the structural effects of this operational condition to specify the maximum number of reactor shut-down which can be supported by the vessel. The aim of this work is to estimate the value of transient thermal stresses in elastic and plastic fields.

In the first part, the loading conditions for different steady and transient operations are examined. It is not easy to evaluate theoretically the temperature distribution in the vessel wall as a consequence of different core transients. For this reason we have planned a number of tests to simulate in a mock-up the temperature changes in the upper part of the vessel for different thermal transients considered most dangerous: scram to full power, electric supply fault in a pump with reactor scram, pump seizure with reactor scram. A mock-up in scale 1 : 4 simulating the vessel area above the core was built-up. The Froude simulation law has been utilized (it is constant the ratio inertial and gravitational loads); the Reynolds numbers are not equal in the model and in the prototype, but are so high in the two conditions that the flow distribution can be considered similar.

The heat transfer is simulated with a mass transfer; with this simulation it is not considered that the heat conduction in the sodium and the thermal gradients measured are higher than in the reactor. The temperature transients have been obtained with a different salt concentration: the water simulates the hot sodium and the water salted represents the cold sodium. We have found the temperature transients in the core outlet to be the most dangerous condition and also the temperature distribution around the nozzle.

In the second part of the work is considered the thermo-elastic structural response using the FEM analysis. The elastic approach has had two functions: the first is to find a simplified structure in two dimensions that has the same behavior as the 3-D nozzle; the second is to follow the thermal transient to look for the time of maximum stress. For different geometry, in a previous paper, the stress distribution and the influence of some characteristic parameters have been examinated. As a result of this part an axisymmetric geometry has been chosen and also the time of maximum stress.

The results of the second step have been utilized to look for the most critical thermal condition and perform the thermal plastic analysis with the FEM approach. We have found the isothermal level curve at the time of maximum stress and the plasticity areas. There are two zones where the nozzle is going in plastic field: the first is near the connection between the cylindrical and spherical shell, the second area is in the spherical vessel.

1. Introduction

In a sodium colled nuclear fast reactor the thermal analysis has a particular importance as a consequence of the rapid transients and the high sodium conductivity. One of the more stressed components is the reactor vessel subjected to two different kinds of loads: mechanical (pressure and pipe thrust) and thermal (stationary and transient conditions). In the vessel the nozzles particulary have an high stress concentration factors and it is necessary to analyse carefully this area. In a reactor scram there is a high sodium temperature change in a few seconds; then it is very important to verify the structural effects of this operational condition to fix the maximum number of reactor shut-down supported by the vessel. The aim of this work is to estimate the value of transient thermal stresses in elastic and plastic fields.

2. Thermohydraulic experiments [1]

In the first part are examined the loading conditions for different steady and transient operations. It is not easy to evaluate theroretically the thermal distribution in the vessel wall as a consequence of different core transients. For this reason we have planned a number of tests to simulate in a mock-up the temperature changes in the upper part of the vessel for different thermal transient considred most dangerous: scram to full power, electric supply fault in a pump with reactor scram, pump seizure with reactor scram. A mock-up in scale 1:4 simulating the vessel area above the core was built-up (fig. 1).

The Froude simulation law has been utilized (it is constant the ratio inertial and gravitationl loads); the Reynolds number are not equal in the model and in the prototype, but are so high in the two conditions that the flow distribution can be considered similar.

The reduction scale for the principal parameters is (m = model, p = prototype).

Dimension	$D_p = 4\ D_m$
time	$t_p = 2\ t_m$
Mass	$M_p = 0{,}85.64\ M_m\ (T_{na} = 375°C)$
	$M_p = 0{,}87.64\ M_m\ (T_{Na} = 525°C)$

The heat transfer is simulated with a mass transfer; with this simulation it is not considered the heat conduction in the sodium and the thermal gradients measured are higher than in the reactor. The temperature transients has been obtained with a different salt concentration: the water simulated the hot sodium and the water salted represents the cold sodium. We have found the temperature transients in the core outlet in the most dangerous condition and also the temprature distribution around the nozzle. In fig. 2 it has been represented at different levels the thermal gradient reduction factor for the three different incidents levels the thermal gradient reduction factor for the three different incidents in the correspondance of the centre and wall respectively. In fig. 4 it has shown the particulars about the points of measure along the height and the plan of the vessel. At the end, the figure 5 shows the temperature transients in the core outlet in the most dangerous condition (electric supply faults in a pump with reactor scram); it is shown also the temperature distribution around the vessel-nozzle. There is an attenuation in the maximum temperature value and in the thermal gradient. This experimental result is the input for the second part of this work.

3. Thermoelastic structural response

In this part of the work is considered the termo-elastic structural response using the FEM analysis. The elastic approach has had two function: the first is to find a simplified structure in two dimensions that have same behavior of 3-D nozzle; the second is to follow the thermal transient to look for the time of maximum stress.

For different geometry in a previous paper it has been examinated the stress distribution and the influence of some characteristics parameters [2] .

As a result of this part, it has been chosen an axisymmetric geometry where the cylindrical vessel is approximated with a spherical vessel ($R_{sph} = 2 \cdot R_{cyl}$).

In fig. 6 is reprensented the nozzle mesh at intersection nozzle-vessel; the analysis was done with 3 nodes (EX6) and 6 nodes triangular elements (EX12), keeping the number of elements unchanged.

The thermal analysis has been performed assuming the following heat transfer coefficients:

sodium-steel $\quad Nu = 7 + 0{,}025\ (R_e)^{0{,}8}$

steel-Nitrogen $\quad h = 0{,}84\ 10^{-4}\ (\Delta t)^{1/3} \quad \frac{W}{cm^2\ {}^\circ C}$

where
- Nu = Nusselt number
- Re = Reynolds number
- h = heat transfer coefficient
- Δt = temperature difference

In fig. 5 is shown also the stress behaviour as a function of time to find the maximum stress valued.

In fig. 7 are reported the isothermal level curves at the time of maximum stress.

In fig. 8 is shown the equivalent stress at the same time; in fig.9 is reported the structure deformed and mesh plot.

4. Thermo-plastic structural response

The elastic-plastic analysis were carried out on this structure according to the Bersafe system |3| ,| 4 |.

In fig. 10 is shown the result of this analysis. There are two different areas where the nozzle is going in plastic field: the first is near the connection between the cylindrical and spherical shell, the second area is in the sperical vessel.

In fig. 10 the black elements represents the plastic nozzle zone.

5. Conclusions

This work is preliminary to realize the vessel behaviour as a consequence of nuclear excursion. The most interesting consideration at this moment is that the vessel analysis must be extended to plastic field because the plastic volume is about 5% of total volume and it is not only in the corner but also in spherical vessel.

References

|1| Contract RVC(72)081 PEC, Alsthom - CNEN, 1975

|2| F. Cesari, 'The equivalent nozzle in thermomechanical problems',4° SMIRT, S. Francisco , 1977

|3| T.K. Hellen, 'Bersafe Phase 2', CEGB Report RD/B/M2500

|4| T.K. Hellen, P.G. Harper, 'Bersafe Phase 3', CEGB Report RD/B/N3597

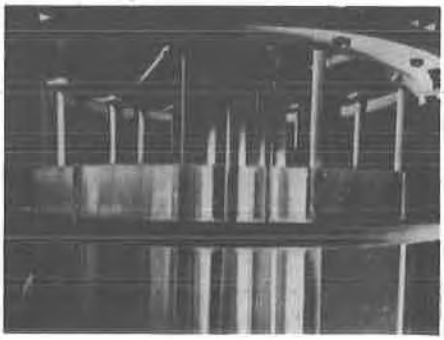

fig.1 - Hydraulic model

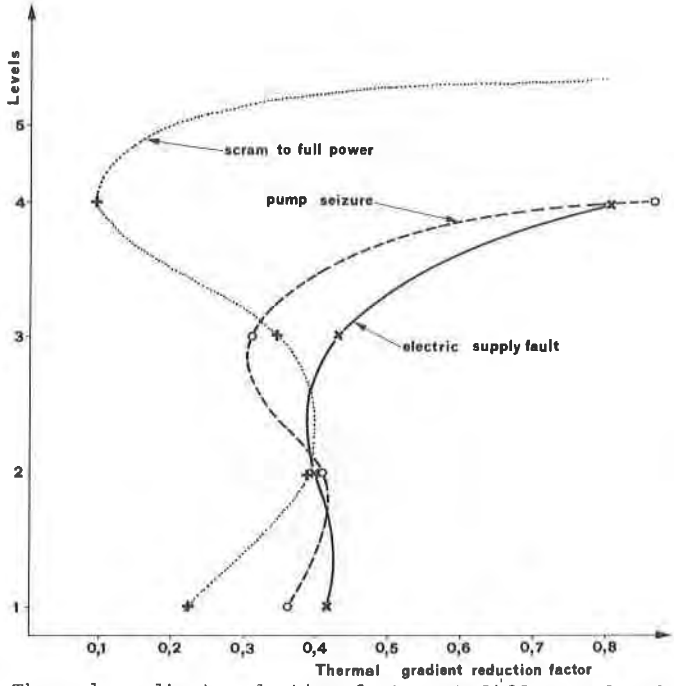

fig.2 - Thermal gradient reduction factor at different levels at the vessel centre

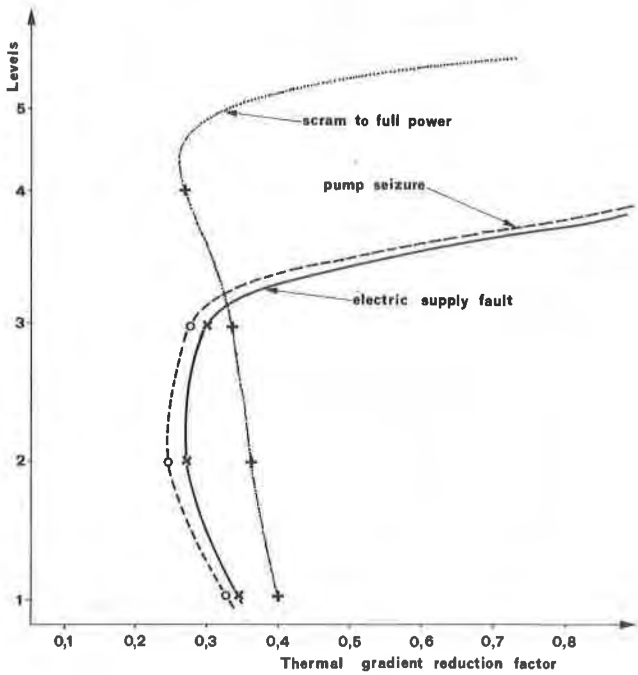

fig.3 - Thermal gradient reduction factor at different levels at the vessel wall

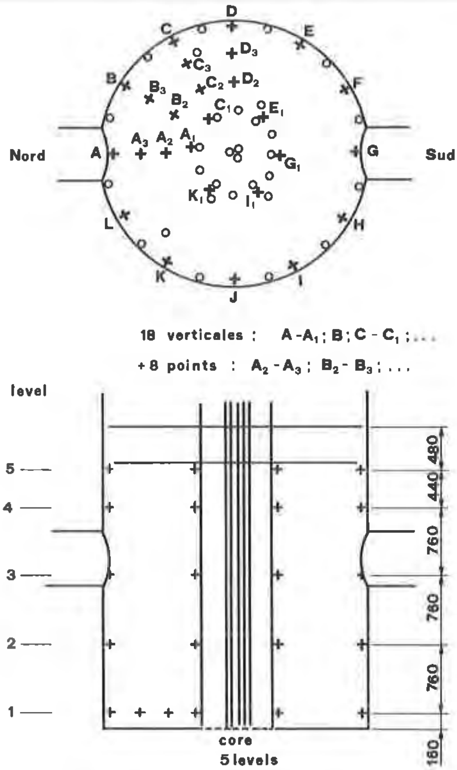

fig.4 - Test points in the hydraulic model

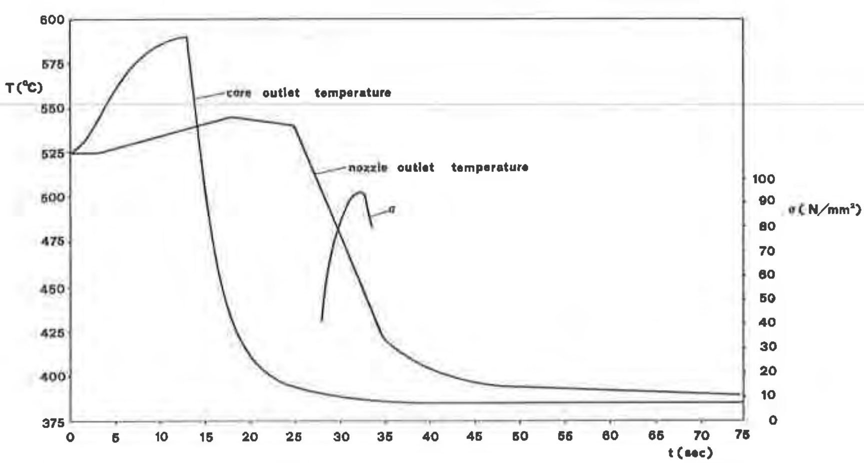

fig.5 - Temperature transient in the core outlet

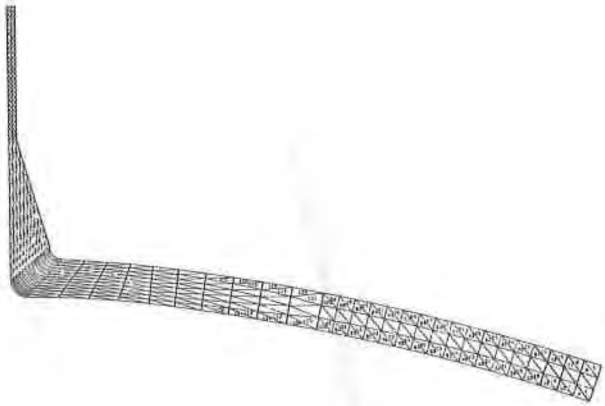

fig.6 - Nozzle mesh at intersection sphere-cylinder

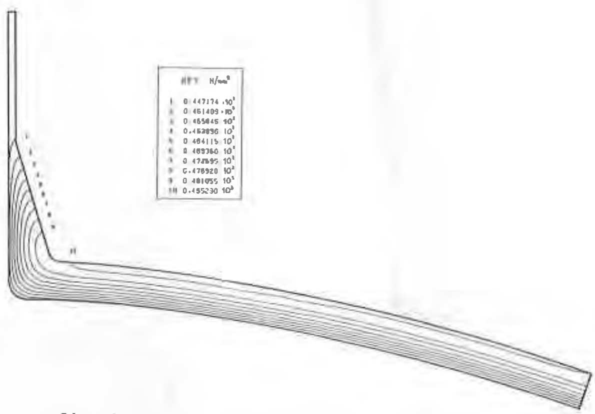

fig.7 - Nozzle temperature distribution

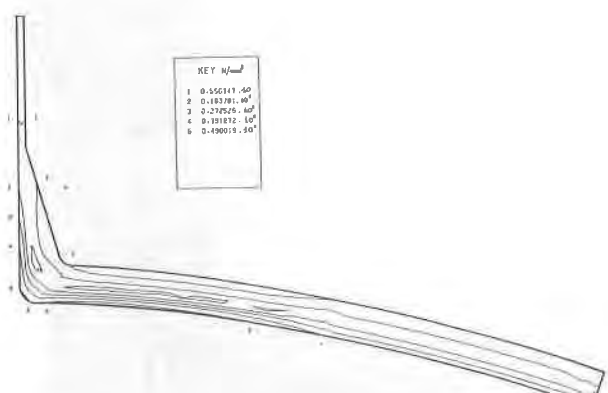

fig.8 - Nozzle equivalent stress

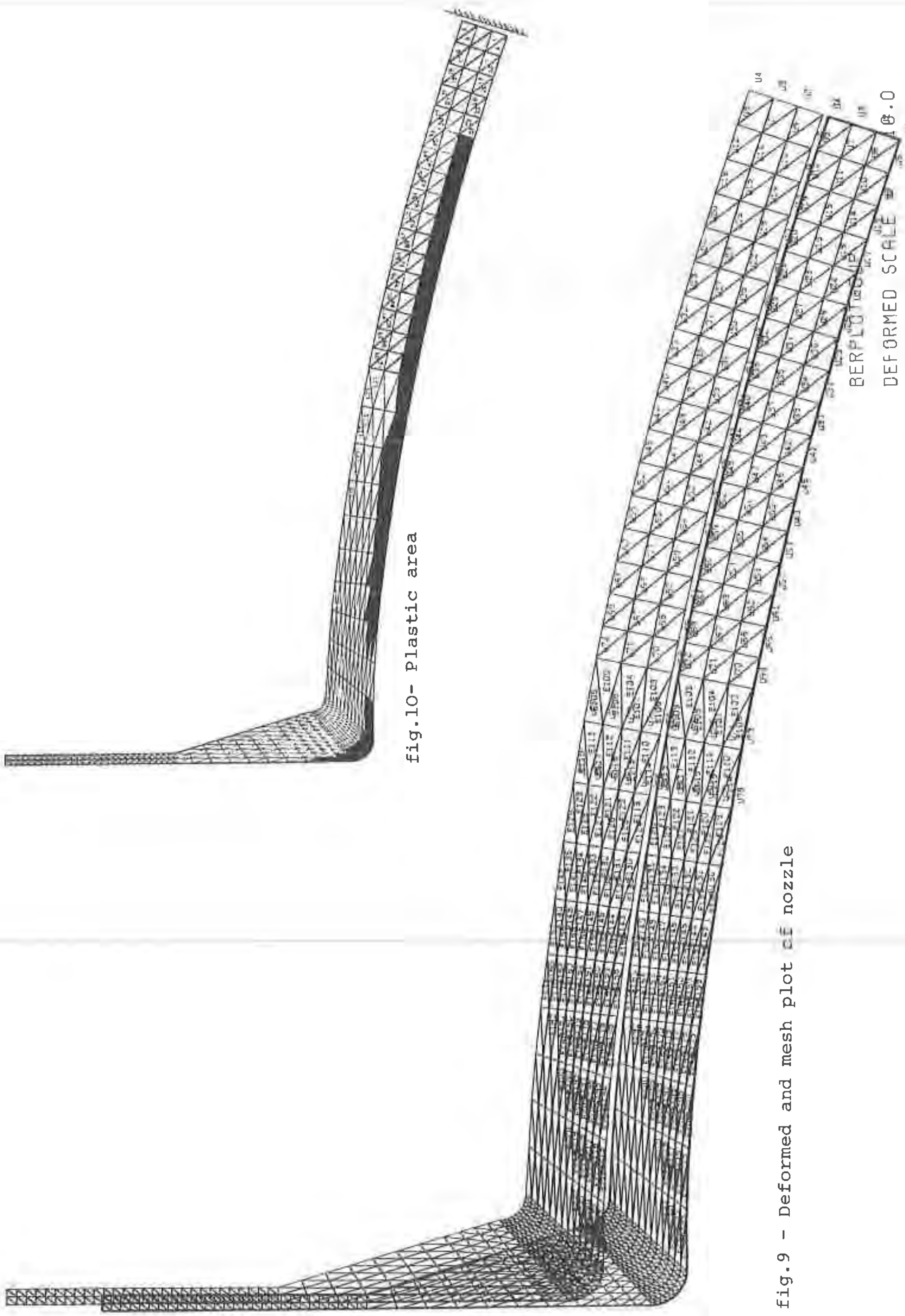

fig.10- Plastic area

fig.9 - Deformed and mesh plot of nozzle