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LECOMTE C. *

COMITE TECHNIQUE AIEA SUR LA SOLLICITATION
ET LA PERFORMANCE DE L'ENCEINTE DE CONFINEMENT
EN SITUATION D'ACCIDENT GRAVE.
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RESEARCH AND DEVELOPMENT STRATEGY ON THE BEHAVIOR OF CONTAINMENTS DURING SEVERE ACCIDENTS

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In case of an hypothetical severe accident leading to core melting, the last barrier preventing radionucleide release in the environnment is the containment of the main reactor building.

The french research and development programmes aimed at understanding the containment behavior during severe accidents relate to several domains ; some of them are :

- assessment of hydrogen behavior
- corium behavior and coolability
- ultimate resistance of the containments and leaktightness
- caracterization of filtered venting procedure.

All these aspects are covered by code calculations and experimental developments.

1 - ASSESSMENT OF HYDROGEN BEHAVIOR

The behavior of hydrogen during a severe accident sequence has been the subject of special attention ; a new research program aimed at reducing the uncertainties for specific events is now starting.

Hydrogen in a containment can act by increasing the number of gazeous moles in the containment and by his ability to sustain a combustion ; the assessment of "hydrogen risk" implies that the potential sources of hydrogen are identified; these are mainly two during an hypothetical severe accident : zirconium oxidation and corium-concrete interaction ; other sources, radiolysis and steel structures oxidation in the containment, are negligeable for at least several weeks.

It has been shown that the integrity of the containment for a french nuclear power plant is not threatened in case of deflagration of all the hydrogen accumulated, under the conservative hypothesis that all the zirconium from the fuel cladding and the grids and plugs has been totally oxidized.

.Moreover, the containment of a french nuclear power plant is so designed that there is a large free volume at the top of the containment : calculations have shown that, under convection effect, the homogeneization of hydrogen in that volume is very fast and effective ; for the lower zone where compartments exist, the openings are large enough that the differences in concentration between the different zones are not of prime importance. As a consequence, the conclusion before has a general range of validity.

For the longer term of the accident, hydrogen production can arise from corium-concrete interaction after vessel melt-through, in the hypothetic case where core coolability has not been restaured. This hydrogen production is concomittant with steam, carbon monoxide and carbon dioxide production.

Because of the design of reactor pit in french nuclear power plants, it is highly probable that the hot gazes generated at the beginning of corium-concrete interaction will induce the combustion of the hydrogen in the containment when coming out the reactor pit via the holes (diam. 80 cm) which exist at the bottom of the reactor cavity for defueling.

For that reason, the maximum quantity of hydrogen which can be ignited at one time is the quantity which corresponds to the total oxidation of the zirconium present in the core, as it is the maximum quantity which can be present in the containment at the moment of vessel melt-through and beginning of corium-concrete interaction.

The specific phenomenology of combustion associated to the ignition by hot gazes, as it is the case at the beginning of concrete erosion, is a subject which is currently under study.

Detonation in the containment is not plausible because the ignition energy which corresponds to an homogeneous hydrogen concentration of physical meaning is very high (several tens of MJ) ; in the case where a local detonation in a specific cell would occur, it has been verified that this detonation would not be transferred to the containment, but would extinguish itself through the communication opening. Concerning the local mecanical effects, these can be studied specifically by mean of the PLEXUS code, which is the CEA code for dynamical fluid-structures interactions.

2 - CORIUM-CONCRETE INTERACTION

Corium-concrete interaction can occur after in-vessel phase of an hypothetic severe accident if no cooling water and no appropriate mitigating measure can be found. The molten core, together with a part of the vessel steel structures, falls then on the concrete raft at the bottom of the reactor pit and the thermal erosion of the concrete begins if there is no water or if this water is insufficient to cool the core.

.This interaction is the source of gazes which contribute to pressurize the containment ; it can also lead by itself to the loss of containment integrity if the concrete raft is eroded on its whole depth.

As this mode of containment failure it thought to be of great importance, CEA/IPSN has participated to a number of international experimental programs, as for example the ACE/MACE experiments, which are still under way.

CEA/IPSN has also developped the WECHSL code in connection with KfK ; this code is validated on the BETA experiments. It analyzes the phenomenon of corium-concrete interaction in order to calculate the erosion rates of the concrete, the shape of the cavity, the mass and enthalpy flowrates of the generated gazes, and the power radiated on the different structures.

An extensive work has been undertaken by CEA/IPSN with the assistance of THERMODATA to determine the thermodynamical properties of oxides and oxides mixtures (up to 5 constituents) which can be present at the time of corium-concrete interaction.

In the frame of the WECHSL User Group, CEA/IPSN has launched at the beginning of 1990 an informatic amelioration of the WECHSL code ; it has also developped the connection of WECHSL with an other module named CALTHER, which makes it possible to calculate the influence of the surrounding structures in the containment on the progression of corium-concrete interaction.

All this knowledge and the work done make the WECHSL code a powerful tool to analyze corium-concrete interaction and corium coolability under various geometrical configurations ; it relies upon international experience and french analytical studies.

3 - RESISTANCE AND LEAKTIGHTNESS OF THE CONTAINMENTS

The various fields covered are the behavior of the containment under quasi-static or dynamic pressure and temperature sollicitation, and the mode and influence of the fissuration.

Specific items are the influence of the steel liner present in a number of french nuclear power plants, the behavior of penetrations and locks, and the extent and influence of vapor condensation in cracks.

Calculations have shown that, for reactors with a steel liner, penetrations should not induce any loss of leaktightness before the pressure at which this would occur in the absence of penetrations.

.The extent of condensation in cracks is also the subject of experimental work aimed at validating a meshed code for distribution and thermal exchanges.

4 - FILTERED VENTING PROCEDURE

The general aim of the filtered venting procedure, as it is adopted on french PWR, is to limit the containment pressure increase by allowing planned releases, and to reduce the associated radioactive releases by the use of a filtered pathway.

By the application of this procedure, named U5, the potential source term from a severe accident can be reduced to a level which can be handled by emergency plans, taking into account confinement and evacuation possibilities, as well as international radiological recommendations. (P.P.I. : Plan Particulier d'Intervention = Local Emergency Plan)

A number of requirements have been formulated for the design of the filtering device :

- efficiency

- reliability

- no interaction with plant normal operation and other safety systems

- independance of external power

- easy to operate

As a consequence of medium scale experiments, the technological choice has been the installation of a sand bed filter on french nuclear power plants ; this sand bed filter is made of about 70 tonnes of so-called "Cattenom sand" (average diameter 0,6 mm), the filtration area being 42 m2 and the thickness of the sand bed 80 cm. It is installed on all french power plants ; for 900 MWe units, there is one sand bed filter for each pair of twin units ; for 1300 MWe, there is a sand bed filter for each unit.

The filter is designed to operate at a pressure close to the atmospheric pressure after a pressure reducing device (throttling orifice).

The exhaust of the filter is connected to an independant release duct located in the plant discharge stack ; the connecting line is equipped with a radioactivity measurement device.

In order to have a full technological validation of the sand bed filter system, a series of full scale experiments has been jointly ordered at the CADARACHE Nuclear Center by CEA/IPSN and Electricité de France.

.The experimental loop is designed to inject in the sand bed filter an experimental aerosol : it is pure CsOH in a water/air flow at composition, temperature and flowrate representative of the conditions of the containments 24 hours after the beginning of the accident, under conservative assumptions. The use of this hygroscopic aerosol, in the presence of water vapor, is the more severe challenge to which the filter can be submitted.

The sand bed filter is tested from differents points of view :

- efficiency towards CsOH aerosols and gazeous iodine

- thermalhydraulic behavior

- thermal behavior in the presence or not of a calorifuge coating, when simulating the residual power of retained fission products

These experiments are not yet fully completed ; nevertheless, first results indicate that the efficiency of fission product retention is pretty good and higher than the design value ; also, the thermalhydraulic calculations of the transient phase at the moment of valves opening are validated.

5 - CONCLUSION

CEA/IPSN has undertaken a number of studies which aim at a better comprehension of the role of the containment during a severe accident in a nuclear power plant. These studies have already contributed to the definition of the ultimate procedures to be used under such hypothesis. In the future, they will continue to help both the evaluation of the present plants and the conception of more performant nuclear power plants for the future.

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