FR8501529

**RAPPORT DAS N° 97 e**

# CONTAINMENT LOADING DURING SEVERE CORE DAMAGE ACCIDENTS

J. FERMANDJIAN*, J.M. EVRARD*, C. CENERINO*
Y. BERTHION**, G. CARVALLO**.

5.International meeting on thermal nuclear reactor safety
Karlsruhe (Germany, FR) 10-13 Sep 1984
CEA-CONF--7653

*CEA/DAS
**CEA/IRDI/DEMT

Novembre 1984

# CONTAINMENT LOADING DURING SEVERE CORE DAMAGE ACCIDENTS

J. Fermandjian, J.M. Evrard, G. Cenerino

CEA/IPSN/DAS
B.P. 6 F-92260 Fontenay-aux-Roses (France)

Y. Berthion, G. Carvallo
CEA/IRDI/DEMT
B.P. 2 F-91190 Gif-sur-Yvette (France)

## ABSTRACT

The paper assesses the reactor containment loading during a severe accident sequence for a 900 MWe PWR unit. Calculations are performed by using the JERICHO thermalhydraulics code concerning the following cases: dry cavity, flooded cavity with corium not cooled and flooded cavity with corium cooled.

## INTRODUCTION

During the past several years, some work has been accomplished, in the field of the nuclear reactor safety, concerning the evaluation of hypothetical severe accidents on pressurized water reactors. These accidents involve core degradation and important fission product release from the fuel. The radiological consequences of such accidents are mainly linked to the residence time of the fission product in the reactor containment building (RCB). Indeed, fission product deposition, by settling on the floor and diffusion on the walls of the RCB, will reduce the amount of radioactive material which will be released to the environment if the containment fails.

Consequently, it is important to determine the mechanical behaviour of the RCB during the accident and the thermal-hydraulic conditions (especially, atmosphere moisture) in the RCB which influence the fission product behaviour. The JERICHO computer code has been developed in order to provide the thermal-hydraulics in the containment: gas pressure, gas and wall temperatures, atmosphere moisture, atmosphere composition and leakage rate.

The objective of the present paper is to study the influence of the state of the reactor cavity (dry or flooded) and of the corium coolability on the thermal-hydraulics in the containment in the case of an accident sequence involving core melting and subsequent containment basemat erosion.

## DESCRIPTION OF THE JERICHO CODE [1], [2]

Since the objective of the JERICHO code is to analyse accidents evolution over several days, it was necessary to write a computer code with a rather simple modelization and a sophisticated numerical treatment. The containment volume is modelized in one compartment with two phases: gaseous atmosphere and sump water. Both phases are assumed to be homogeneous, in pressure equilibrium, but in thermal non-equilibrium. Containment atmosphere can be composed of oxygen, nitrogen, steam, hydrogen, carbon dioxide and carbon monoxide. The main physical phenomena are described below (also, see Figure 1).

### Mass and Energy Sources

The JERICHO code provides the thermal-hydraulic conditions in the reactor containment building from the onset of the accident through the stages of blowdown, core heat-up, boiloff, core meltdown, pressure vessel bottom head failure and interaction of the molten debris with the concrete containment basemat. The energy and mass flowrates of the different gaseous species ($H_2O$, $H_2$, $CO_2$ and CO) entering the containment during the different phases of the accident are evaluated by various codes (described in [3]).

Besides, the following energy sources are taken into account :

- decay heat associated with fission products released into the containment. This energy, initially released in the gaseous phase, is rapidly transferred into the sump water in order to simulate the fission product deposition (the deposition rate being calculated by the AEROSOLS/B1 Code [3]),
- energy radiated by the corium surface during the corium-concrete interaction phase. The radiation generated by the corium surface is to a large extent stopped by the reactor cavity walls: it leads to the heat-up and the thermal degradation of the concrete, releasing steam and carbon dioxide,
- energy released from hydrogen and carbon monoxide burning.

### Heat and Mass Transfer Between Atmosphere and Sump Water

At the surface of the sump water, convection occurs in thermal non-equilibrium and condensation if the liquid temperature is lower than the saturation temperature or evaporation in the opposite case. Steam condensation in the bulk gaseous phase and sump water boiling are also considered.

### Heat and Mass Transfer Between Atmosphere and Structures

Heat and mass exchanges, between atmosphere and structures, occur by convection and condensation if the structure temperature is lower than the saturation temperature. In the JERICHO code, the condensation film is not modelized as a distinct system; the condensed water is assumed to reach the sump water instantaneously at the wall temperature.

### Thermal Conduction in the Structures

Containment walls and internal structures are modelized as one-dimensional slab geometry heat sink. Each structure consists of only one material. In the case of concrete structures with a steel liner, a separate calculation is performed for the steel liner. The outside atmosphere and the ground are treated as sinks at constant temperature.

### Containment Spray

In French 900 MWe PWR's, the containment spray system works either, in direct phase, from the storage tank, in common with emergency core cooling system, or, in its recirculation phase, from the sump water. In the second case, the liquid passes through heat exchanger in order to evacuate the decay heat in the long-term. The JERICHO code calculates the energy transferred to the heat exchanger and the inlet temperature of the spray water.

### Leakage and Venting

From the containment design pressure,venting of the RCB is designed through a rough filtration device in order to reduce the radioactive release to the environment. The gas is assumed to be expanded isoenthalpic and the flowrate is calculated from the SAINT-VENANT law. The containment leakages (natural containment leakage and containment isolation failure) are also modelized.

### Hydrogen and Carbon Monoxide Combustion

The combustion of hydrogen and carbon monoxide becomes possible when each of these components exceeds a given volumetric concentration. More accurately, the flammability limit is determined in the ternary diagram ($H_2$+CO, $H_2O$+$CO_2$, air).

Hydrogen and carbon monoxide burning is modelized in two ways:
1) continuous burning as soon as the flammability limits are reached,
2) storage and instantaneously burning (deflagration) at input specified time.

### Numerical Treatment

The evolution of gas components and liquid water mass and internal energy, as well as structure surface temperature, is governed by a system of first order non-linear differential equations. In order to modelize the engineered safety feature performance and some physical phenomena (fission product influence and leakage), some additional equations are considered. This system is solved by an ADAMS-PECE method. Gas and liquid temperatures, as well as gas component partial pressures, are obtained by solving a system of non-linear equations with a specific residues method.

## ACCIDENT SEQUENCE DESCRIPTION

The accident sequence to be studied, on which the following study is based, is a break loss-of-coolant accident (2" diameter) with failure of both the emergency core cooling system and the containment spray system ($S_2$CD sequence according to the Reactor Safety Study terminology [4]). Note that, in this case, the containment building cooling is not operational. The reactor to be considered is a 900 MWe PWR unit with a large dry containment (Figure 2). The reactor building is a prestressed concrete containment with an internal steel liner (steel thickness = 6 mm, air gap = 0.5 mm). We present herein-after the successive stages of the accident and the corresponding containment response. The timing of the main events occurring throughout the accident is given below:

| Time (minutes) | Events |
|---|---|
| 0 | Start of blowdown |
| 0.8 | Reactor scram (primary pressure less than 12.9 MPa) |
| 0.8 | Secondary circuit: feedwater and steam outlet shut-off |
| 1.8 | Auxiliary feedwater pumps start |
| 4 | Primary pumps stop (cavitation) |
| 31 | Core uncovery begins |
| 47 | Accumulator discharge |
| 150 | Fission product release starts |
| 155 | Core melting begins |
| 175 | Core slump |
| 200 | Reactor vessel dry |
| 262 | Bottom head fails |
| 262 | Corium-concrete interactions begins |
| 622 | Corium solidification |
| 10,080<br>(7 days) | Containment floor melt-through |

## Primary Circuit Blowdown

The primary circuit depressurization initiates the reactor scram and the starting of the auxiliary steam generators feedwater supply system. Main pumps are stopped because of the cavitation. The primary circuit blowdown progresses and the core uncovery occurs at about 30 minutes. This stage has been computed by using the RELAP 4 Mod 6 code [5].

## Core Heat-Up and Meltdown

The core uncovery involves its heat-up, enhanced by the zircaloy cladding oxidation by steam, which releases large amounts of energy and of hydrogen. During this stage, the primary circuit depressurization continues and the pressure falls below the accumulators pressure threshold (4.2 MPa): the accumulator water discharge temporary refloods the core. After this water is vaporized, the core melting process resumes: the most volatile fission products are released by the core. This stage comes to an end with the core slumping in the vessel bottom head (at 195 minutes). Some water is still present in the bottom head and its sudden contact with the melting core causes its rapid vaporization. The metallic components of the core are supposed to be oxidized and hydrogen is again released in the containment building. The computer code used here is a version of the EOIL 2 code [6], modified in order to calculate small break and transient accidents; moreover the decay heat and fission product release models have been markedly improved. According to a simplified evaluation, the bottom head rupture occurs after about one hour. This long delay is due to the low primary pressure, which does not lead to important mechanical stresses.

## Concrete Basemat Ablation

The melting fuel and internal structures mixture (corium) falls down on the concrete basemat in the reactor activity. The concrete basemat is progressively eroded by a thermal degradation process: concrete is decomposed and free water, bound water and carbon dioxide are successively released.

Water and carbon dioxide are partially reduced into hydrogen and carbon monoxide when passing through the corium metallic layer. At the same time, large amounts of energy are radiated by the very hot (temperature decreasing from 2000°C to 1500°C) upper layer of the corium. The concrete ablation kinetics, as well as the gas masses released into the containment atmosphere, are highly dependent upon the composition of the concrete.

Main characteristics of a limestone concrete are presented in Table I. In the case of limestone concrete, the ablation rate is estimated to be 30 cm/h during the first hour of interaction, 18 cm/h thereafter and 2 cm/h after the solidification of the corium metallic layer in contact with the concrete basemat. It is to be noted that considerable uncertainties still exist in the knowledge of these phenomena and so we have simply adopted the approximate values currently accepted. With this hypothesis, the basemat rupture occurs at about 7 days (thickness of the concrete containment basemat: 4.2 m).

## PRESENTATION OF THE JERICHO CALCULATIONS

After the vessel rupture, the corium forms a deep bed about 60-70 cm high, according to the amount of core and structures involved, and progressively ablates the concrete basemat. During the long period of time (several days) preceding the complete basemat pass-through, it is likely to set up or recover means of providing cooled water on the corium surface. Hence the problem of corium coolability is of primary importance as concerns the management of severe accidents. Three key-questions are to be answered: (a) what kind of interaction between the corium and water does occur, (b) does it bring to a stable coolable configuration and (c) what are the consequences on containment integrity ?

The molten corium and the water pool overlying it will probably be separated by a solid crust and a vapor film, which limit the heat transfer to rather low values. Yet crust dislocation by gas release and destabilization of the vapor film, with direct water-corium interaction, are likely to occur. These mechanisms could lead to a certain extent to corium fragmentation, but it is difficult to imagine the formation of a coolable debris bed in this way, according to the corium depth.

Nevertheless it is impossible up to now to answer the question of corium coolability, and the present paper is aimed only at addressing the third question, with arbitrary assumptions on coolability. It allows therefore to bracket the containment response during this stage of the accident, assuming the containment building cooling is not available.

To this end, three calculations were performed: dry cavity, flooded cavity with corium assumed being not coolable and flooded cavity with corium assumed being coolable.

### Case 1 (dry cavity)

The first calculation concerns the study of the $S_2$CD sequence in which the reactor cavity is dry. In this case, the concrete basemat is heated to high temperatures before water vapor and gases are released from it. During the first hours, a considerable amount of the available energy in the core debris is therefore transferred to and stored in the concrete, leaving only a fraction of the energy to be transferred to the containment building atmosphere as steam and non condensible gases (mainly, carbon dioxide).

Case 2 (flooded cavity - corium not cooled)

The second calculation relates to the study of an accidental sequence similar to the preceding, except that the reactor cavity is flooded (for example, by recovery of the emergency core cooling system) after the beginning of the corium-concrete interaction. Therefore, the radiation heat generated by the corium surface does not heat up the reactor cavity walls but rather vaporizes the water poured into the reactor cavity.

Case 3 (flooded cavity - corium cooled)

For the third calculation, we assume that the cavity contains an unlimited amount of water to cool the debris to the extent that the core could not reheat and interact with concrete. After the core is brought into thermal equilibrium with the water (in 1 hour) and the rapid steam production and rapid pressure rise have occured, the decay heat in the core materials continues to boil water (at the rate of 400 kg/minute).

For the previous three cases, the sources, in kg, released (after 4 days) to the reactor containment building during the "corium-concrete interaction" are given in the following table:

| | Case 1 | Case 2 | Case 3 |
|---|---|---|---|
| Steam released to the RCB | 34,128 | 573,873 | 1,303,010 |
| Hydrogen released to the RCB | 276 | 276 | 80 |
| Carbon dioxide released to the RCB | 74,437 | 41,125 | |
| Carbon monoxide released to the RCB | 4,338 | 4,338 | |

The main characteristics of the reactor containment building are presented in Table I and the different sources released into the containment throughout the accident (dry cavity) in Table II, among which the main sources are the two-phase water flow during the primary circuit depressurization and the carbon dioxide generated by the concrete basemat erosion.

## RESULTS OBTAINED - DISCUSSION

### Containment Atmosphere Pressure and Temperatures

The curves of pressure and average temperature of the gaseous and liquid phases (Figures 3, 4 and 5) show two peaks corresponding to the primary circuit depressurization and to the bottom head water vaporization when the core slumps. These pressure peaks reach respectively 0.20 and 0.28 MPa and so are far lower than the containment building design pressure (0.5 MPa). For case 3, the third peak (0.35 MPa) corresponds to the rapid steam production due to the thermal equilibrium between core and water (see previous section).

The containment atmosphere pressure exceeds the design pressure: after 6 days for case 1, after 3 days for case 2 and after 1 day for case 3.

When the containment design pressure is reached, the containment atmosphere temperatures are equal to 117°C, 131°C and 138°C for cases 1, 2 and 3 respectively.

Note that:

- the containment basemat is supposed to be penetrated by the corium after 7 days,

- the best-estimate assessments of the actual strength limits of reactor containment with an internal steel liner had led to the conclusion that the rupture occurs suddenly by fracture of prestressing cables at about 1.0 to 1.3 MPa leading to loss of containment integrity [7].

Therefore, the above-ground containment rupture occurs only for case 3, after 3 days. For case 2, the above-ground containment rupture and the basemat rupture occur at about the same time (after 6-7 days).

The pressure and temperature increase rates are:

- 1.8 kPa per hour and 0.29°C per hour for case 1;
- 4.4 kPa per hour and 0.54°C per hour for case 2;
- 10.5 kPa per hour and 0.83°C per hour for case 3.

## Containment Atmosphere Moisture

Figure 6 gives the containment atmosphere moisture versus time for case 1. This curve shows that the containment atmosphere is not always saturated during the first hours of the accident. Similar results have been obtained for cases 2 and 3.

## Containment Atmosphere Composition

Figures 7, 8 and 9 give the atmosphere composition ($O_2$, $N_2$, $H_2O$, $H_2$, $CO_2$ and CO) as functions of time for cases 1, 2 and 3. These curves show clearly:

- the importance of steam and carbon dioxide for case 1,
- the importance of steam for cases 2 and 3.

When the containment design pressure of 0.5 MPa is reached, the atmosphere compositions are the following (%, in mass):

| | Case 1 | Case 2 | Case 3 |
|---|---|---|---|
| $O_2$ | 5.9 | 7.4 | 8.4 |
| $N_2$ | 19.5 | 24.4 | 27.9 |
| $H_2O$ | 23.4 | 44.9 | 63.1 |
| $H_2$ | 0.5 | 0.6 | 0.6 |
| $CO_2$ | 48.6 | 20.1 | 0 |
| CO | 2.1 | 2.6 | 0 |

Concerning the hydrogen combustion:

- For case 1, the hydrogen concentrations always exceed the flammability limits: hydrogen volume fraction more than 4% and steam volume fraction less than 60%.
- For cases 2 and 3, the steam volume fractions are higher than 60% after 3.5 days and 0.5 days respectively.

## Energy Balance

For cases 1, 2 and 3, the energy balances after 4 days are the following (in % of energy injected into the reactor containment building):

| | Case 1 | Case 2 | Case 3 |
|---|---|---|---|
| Energy located in the concrete walls | 23 | 21 | 19 |
| Energy located in the internal structures | 58 | 42 | 46 |
| Energy located in the atmosphere | 6 | 9 | 12 |
| Energy located in the sump water | 6 | 13 | 19 |
| Energy transferred to the environment | 7 | 5 | 4 |

Note the importance of the energy located in the internal structures.

Concerning the decay heat, after 4 days, 5.334 MW is still located in the corium (71.3%), 2.026 MW in the sump water (27.1%) and 0.118 MW in the gas phase (1.6%).

## CONCLUSION

The JERICHO code has been developed in order to study the thermodynamic behavior inside the reactor containment building for the complete spectrum of accident sequences likely to occur on a PWR.

Sensitivity analyses performed with the JERICHO code on the $S_2CD$ sequence (involving core melting and subsequent containment basemat erosion) have shown the importance of the state of the reactor cavity (dry or flooded) and of the corium coolability on the thermal-hydraulics in the containment.

For a dry cavity, the containment building failure is due to the basemat rupture, after 7 days. For a flooded cavity with corium not cooled, the basemat rupture and the above-ground containment rupture occur at the same time, after about 7 days. For a flooded cavity with corium cooled, the containment building failure is due to the steam overpressurization, at 3 days. For this case, if the containment building cooling is recovered the above-ground containment rupture will be prevented and the fission products will not be released to the environment.

A substantial buildup of combustible gases resulting from the corium-concrete interaction can be prevented if the core can be flooded with water. The formation of a coolable debris bed remains to be demonstrated. Nevertheless, in the long-term, the rates for corium penetration and for combustible gas production are expected to be lower compared with the dry cavity sequence, even if the coolability is not demonstrated.

In the near future, we intend to check the adequacy of the input data for the JERICHO code (in particular, mass and energy flowrates released into the reactor containment building during the corium-concrete interaction) and to validate the JERICHO code from analytical and integral experiments.

# REFERENCES

[1] J. FERMANDJIAN, J.M. EVRARD, Y. BERTHION, G. LHIAUBET
JERICHO computer code: PWR containment response during severe accidents. Description and sensitivity analysis.
OECD-NEA-CSNI Specialist Meeting on Water Reactor Containment Safety.
Toronto, Canada - October 17-21, 1983.

[2] A. L'HOMME, P. PEPIN, Y. BERTHION, G. CARVALLO
Code de calcul réaliste du comportement thermodynamique et radiologique des enceintes de confinement des réacteurs PWR.
AMS Conference - Paris, France - July 1982.

[3] J. FERMANDJIAN, J.M. EVRARD, Y. BERTHION, G. LHIAUBET, M. LUCAS
Source term associated with two severe accidents in a 900 MWe PWR.
International Meeting on LWR Severe Accident Evaluation - Cambridge, USA
August 28 - September 1, 1983.

[4] Reactor Safety Study - An assessment of accident risks in U.S. commercial nuclear power plants.
WASH-1400 (NUREG-75/014), October 1975.

[5] RELAP 4 Mod 6. A computer program for transient thermal hydraulic analysis of nuclear reactors and related systems. User's manual.
EG & G, Idaho, Inc-Report CDAP TR 003, January 1978.

[6] R O. WOOTON, H.I. AVCI
MARCH (Meltdown Accident Response CHaracteristics). Code description and user's manual. NUREG/CR.1711 (BMI-2064), October 1980.

[7] J. BROCHARD et al.
Study of the behavior of containment buildings of PWR-type reactor, until complete failure in case of LOCA.
6th International Conference on Structural Mechanics in Reactor Technology - Paris, France - August 17-21, 1981.

# TABLE I _ MAIN CHARACTERISTICS OF THE REACTOR CONTAINMENT BUILDING, OF THE PRIMARY COOLANT SYSTEM AND OF THE CORE

| | | |
|---|---|---|
| **Reactor containment building** | | |
| Volume | 49,900 $m^3$ | |
| Design pressure | 0.5 MPa | |
| Failure pressure estimate | 1.0 - 1.3 MPa | |
| Floor area | 830 $m^2$ | |
| Total surface | 50,000 $m^2$ | |
| **Concrete composition** | | |
| $CaCO_3$ | 0.557 | weight fraction |
| $Ca(OH)_2$ | 0.013 | |
| $SiO_2$ | 0.219 | |
| Free $H_2O$ | 0.038 | |
| $Al_2O_3$, $Fe_2O_3$ | 0.173 | |
| **Primary coolant system** | | |
| Mass of water ( including accumulator tanks ) | 280,000 kg | |
| **Core** | | |
| Mass of $UO_2$ ( fuel ) | 79,600 kg | |
| Mass of Zr ( fuel cladding ) | 18,200 kg | |
| Mass of Fe, Ni, Cr ( structure material ) | 4,600 kg | |
| Mass of Ag, In, Cd ( control rods ) | 1,900 kg | |
| Mass of active fission products * | 385 kg | |
| Mass of inactive fission products * | 1,260 kg | |
| Mass of actinides * | 580 kg | |

** for fuel burn-up* { *1/3 11,000 MWd/MTU*; *1/3 22,000 "*; *1/3 33,000 "*

# TABLE II _ MASSES RELEASED TO THE REACTOR CONTAINMENT BUILDING (in kg) FOR THE REFERENCE CASE

| | |
|---|---|
| **Initial conditions** | |
| Nitrogen | 41,048 |
| Oxygen | 12,400 |
| Steam | 1,276 |
| **Blowdown** *(from 0 to 31 min.)* | |
| Water - steam | 129,914 |
| **Core heat-up_Boiloff_Core meltdown** *(from 31 to 195 min)* | |
| Steam produced | 129,825 |
| Steam released from the break to the RCB | 124,965 |
| Hydrogen released from the break to the RCB | 540 |
| **Core slumping and vessel failure** *(from 195 to 262 min)* | |
| Steam produced | 24,078 |
| Steam released from the break to the RCB | 21,683 |
| Hydrogen released from the break to the RCB | 266 |
| **Corium-concrete interaction** *(from 262 min to 4 days)* | |
| Steam released to the RCB | 34,128 |
| Hydrogen released to the RCB | 276 |
| Carbon dioxide released to the RCB | 74,437 |
| Carbon monoxide released to the RCB | 4,338 |
| **Sump water radiolysis** *(after 4 days)* | |
| Hydrogen released to the RCB | 80 |

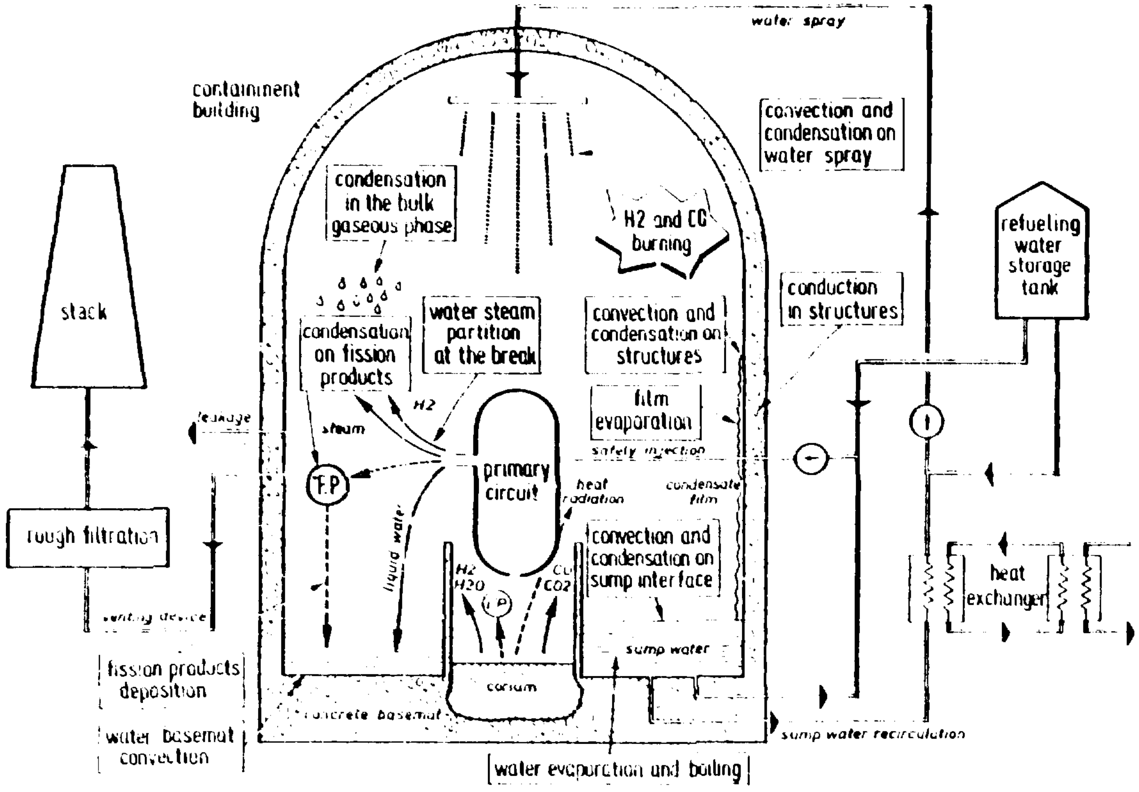

Fig 1_ DIAGRAM OF SYSTEMS AND PHYSICAL PHENOMENA IN THE JERICHO CODE

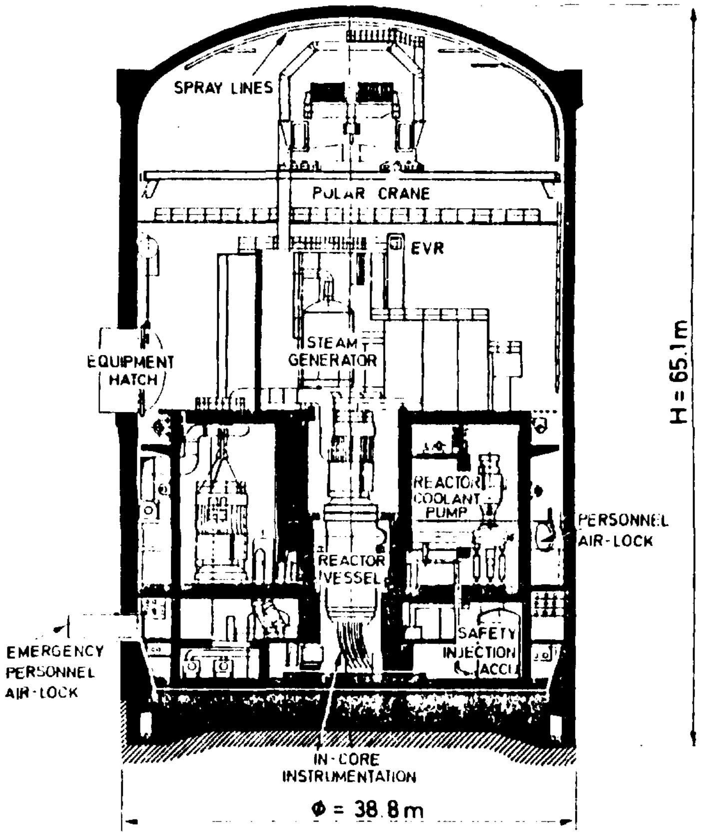

Fig. 2 - REACTOR CONTAINMENT BUILDING LARGE DRY - 900 MWe

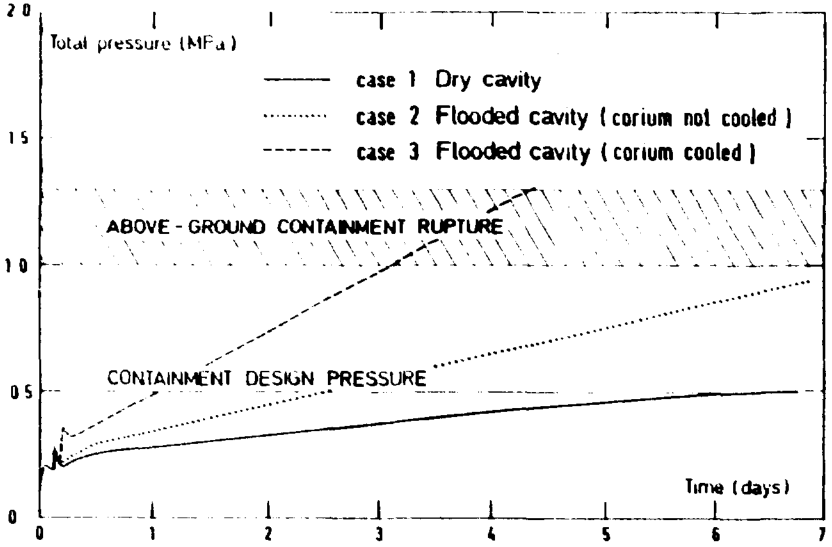

Fig 3 _ CONTAINMENT BUILDING INTERNAL PRESSURE VERSUS TIME for $S_2CD$

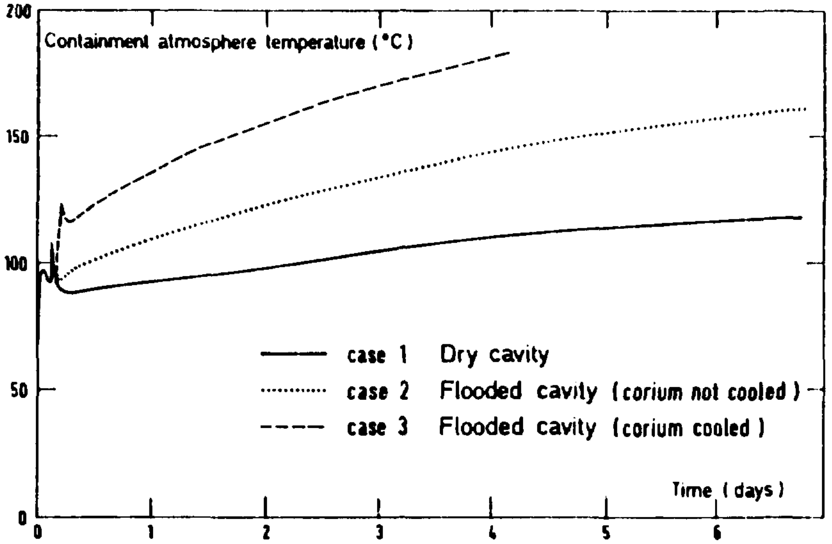

Fig. 4 _ CONTAINMENT BUILDING ATMOSPHERE TEMPERATURE VERSUS TIME for $S_2CD$

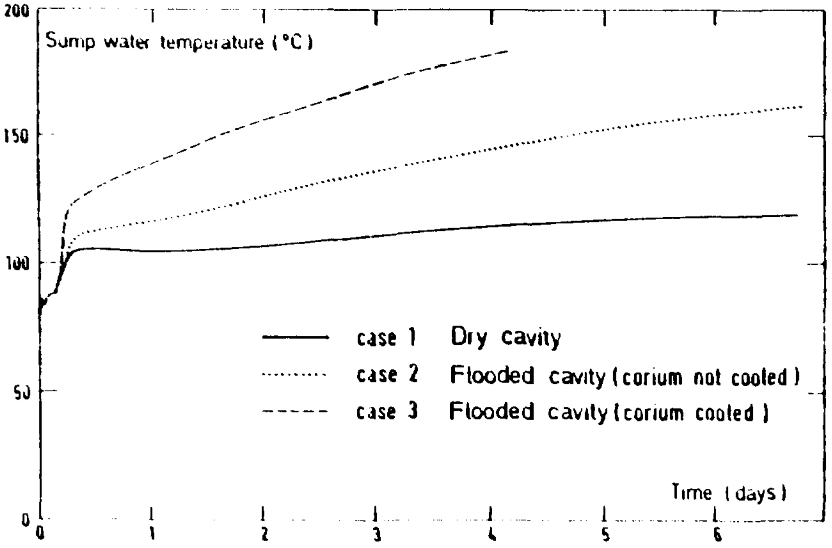

Fig 5 _ CONTAINMENT SUMP WATER TEMPERATURE VERSUS TIME for $S_2CD$

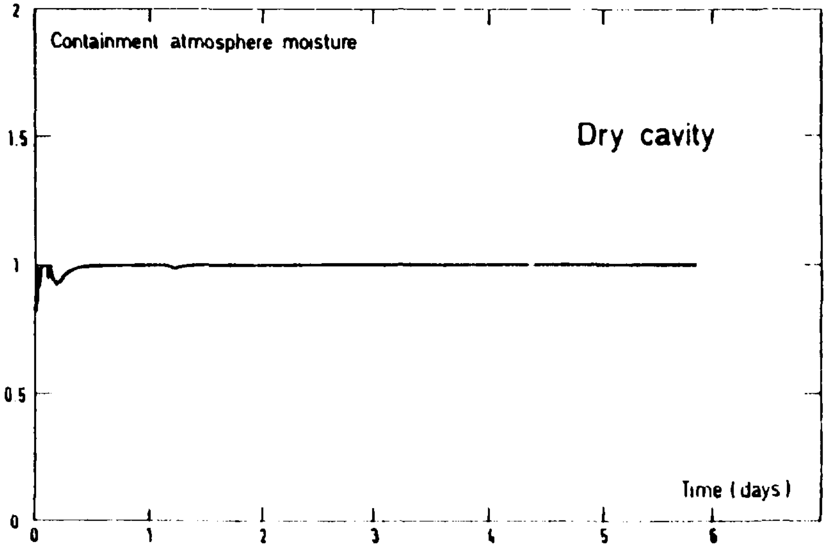

Fig. 6 _ CONTAINMENT BUILDING ATMOSPHERE MOISTURE VERSUS TIME for $S_2CD$

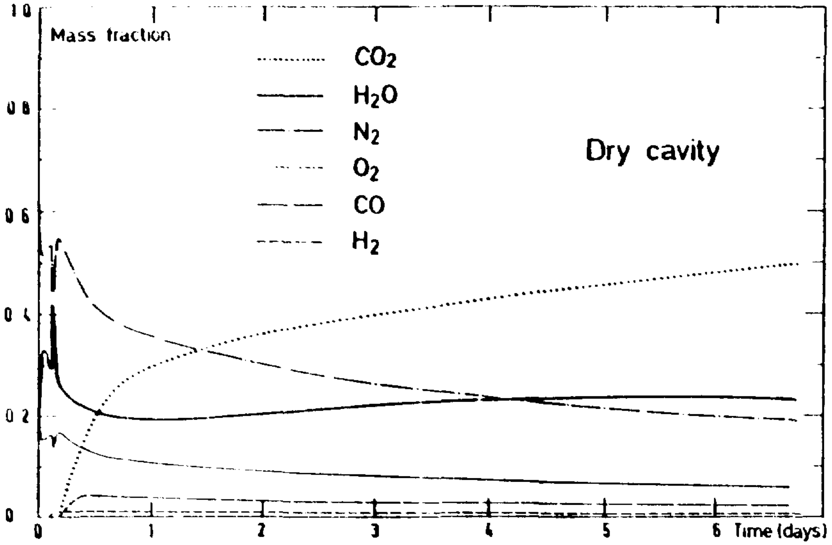

Fig 7 _ ATMOSPHERE MASS COMPOSITION VERSUS TIME for $S_2CD$

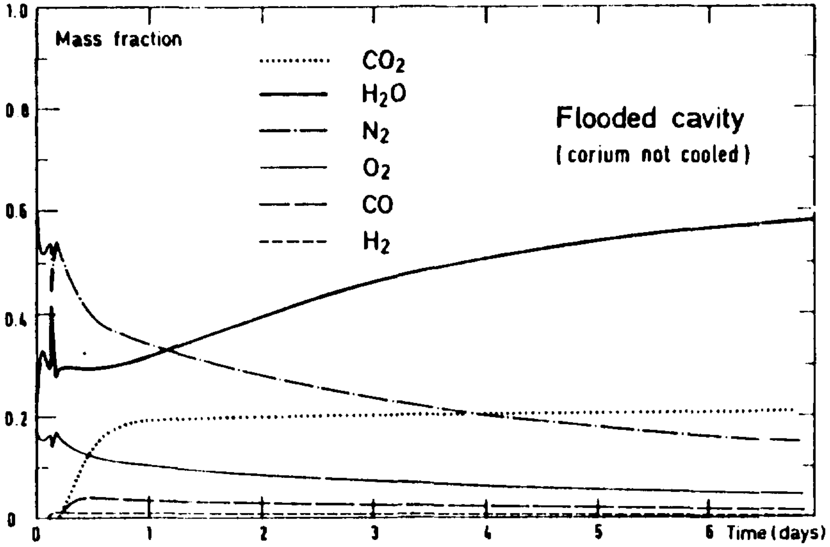

Fig. 8 _ ATMOSPHERE MASS COMPOSITION VERSUS TIME for $S_2CD$

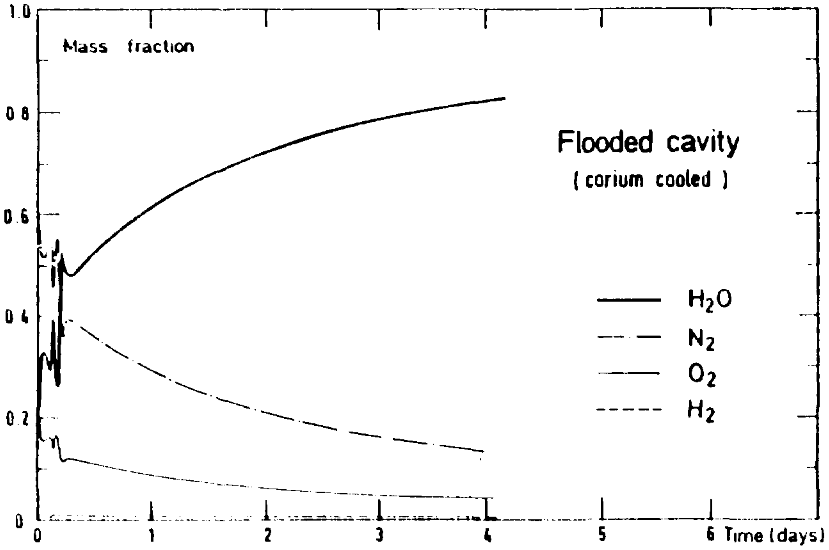

Fig 9 _ ATMOSPHERE MASS COMPOSITION VERSUS TIME for $S_2CD$

# DESTINATAIRES

## DIFFUSION CEA

M. le Haut Commissaire
DSE
DDS
IPSN
IPSN/D.SN
IPSN/D.SN : M. CANDES
IPSN/D.SN : M. PELCE
IPSN/D.SN : M. SCHMITT
DAS
LEFH
SASR
SASCEL
SAF
SAER
SGNR
SAREP
BRTSN
SASICC
SASLU
SASLU/VALRHO
SASLU/SPI
SEC
SESECT

DERS Cadarache
SAED
SES Cadarache
SERE Cadarache
SIES Cadarache
SESRU Cadarache
SRSC Valduc
SEAREL
IPSN/D.Pr/FaR

DPS/FaR
DPT/FaR
UDIN/VALRHO
DEDR Saclay
DRNR Cadarache
DRE Cadarache
DER Cadarache
DEMT Saclay
DMECN/DIR Cadarache
DMECN Saclay
DTCE Grenoble
DSMN/FaR
Service Documentation Saclay

## DIFFUSION HORS CEA

Secrétariat Général du Comité Interministériel de la Sécurité Nucléaire : M. AUGUSTIN
Service Central de Sûreté des Installations Nucléaires : M. de TORQUAT (+ 3 ex.)
Service Central de Sûreté des Installations Nucléaires - FAR
Direction Générale de l'Energie et des Matières Premières : M. FRIGOLA
Conseil Général des Mines : M. MEO
FRAMATOME : M. le Directeur Général
NOVATOME : M. le Directeur Général
TECHNICATOME : M. le Directeur Général
TECHNICATOME : Service Documentation
EDF / Etudes et Recherches
EDF / SEPTEN (2 ex.)
EDF / SPT
M. SCHNURER - Bundes Ministerium des Innern - BONN (RFA)
M. KREWER - Bundes Ministerium für Forschung und Technologie - BONN (RFA)
M. BIRKHOFER - Technische Universität München - GARCHING (RFA)
M. KELLERMAN - Gesellschaft für Reaktorsicherheit - KOLN (RFA)
M. LEVEN - Gesellschaft für Reaktorsicherheit - KOLN (RFA)
M. LAFLEUR - U.S.N.R.C. - WASHINGTON (E.U.)
M. MINOGUE - U.S.N.R.C. - WASHINGTON (E.U.)
M. KINCHIN - U.K.A.E.A. - Safety and Reliability Directorate - RISLEY (G.B.)
M. HANNAFORD - Nuclear Installations Inspectorate - LIVERPOOL (G.B.)
M. ALONSO - Junta de Energia Nuclear - MADRID (ESPAGNE)
M. PERELLO - Consejo de Seguridad Nuclear - MADRID (ESPAGNE)
M. CARLBOM - Department of Safety and Technical Services - NYKOPING (SUEDE)
M. NASCHI - Direttore Centrale della Sicurezza Nucleara e della Protezione Sanitaria
ROMA (ITALIE)
M. TANIGUCHI - MITI (JAPON)
M. ISHIZUKA - Science & Technology Agency - Nuclear Safety Bureau (JAPON)
M. TAMURA - Science & Technology Agency - Nuclear Safety Bureau (JAPON)
M. KATSURAGI - JAERI - Center of Safety Research (JAPON)

## COPIE (SANS P.J.)

M. CHAVARDES (Attaché près de l'Ambassade de France aux Etats-Unis)
M. FELTEN (Attaché près de l'Ambassade de France au Japon)
M. WUSTNER (Attaché près de l'Ambassade de France en RFA)