SEVERE ACCIDENT IMPROVEMENTS FOR CAREM-25 TO ARREST REACTOR VESSEL MELTDOWN SEQUENCES

*L. E. Pomier Báez**, *J. E. Núñez Mac Leod***, *J. H. Barón****

CEDIAC Institute
Engineering Faculty, Cuyo National University
Ayacucho 666, 5500 Mendoza, Argentina
Phone: +54 261 4205115 int 2100 Fax: +54 261 4380120
*lpomier@cediac.uncu.edu.ar **jmacleod@cediac.uncu.edu.ar
***jbaron@cediac.uncu.edu.ar

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ABSTRACT

Given an accident sequence, that leads to sustained uncovering of the core, the progression of core damage involves several complex phenomena. The progression of these phenomena can lead to a breach of the reactor vessel followed by the discharge of molten core materials to the containment.

Advanced nuclear reactor designs, such as the CAREM reactor, include several improvements related to safety issues either enhancing the passive safety functions or allowing plant operators more time to undertake different management actions against radioactive releases to the environment.

In the development of the nuclear power plant CAREM, the possibility of including a passive metallic in-vessel container in its design is being considered, to arrest the reactor pressure vessel meltdown sequence during a core damaging event, and thereof prevent its failure.

The paper comprises the first analyses, via numerical simulation, for the conceptual design of such a container type; furthermore, the paper addresses simulation model characteristics helping to establish geometrical dimensions, materials and container compatibility with power plant engineering features.

The paper also presents the first model developed to analyze the complex relocation phenomena in the core of CAREM during a severe accident sequence caused by a loss of coolant. The PC version of MELCOR 1.8.4 code has been used to predict the transient behavior of core parameters. MELCOR is a fully integrated relatively fast running code that models the progression of accidents in light water reactor power plants.

This paper presents reactor variables behavior during the first hours of the event being studied, giving preliminary conclusions about the use and capability of a metallic in-vessel core catcher.

INTRODUCTION

One of the most important findings from the TMI-2 accident concerns to the role that plays the transfer mode of molten material from the degraded core to the lower head of the reactor vessel. This transfer is a meaningful factor to evaluate the in-vessel progression of a severe accident in nuclear reactors. The understanding of this phenomenon is a milestone with respect to the in-vessel debris cooling process and reactor vessel failure studies [1].

Depending on the scenario considered during the analysis of severe accident sequences, molten material relocation into the lower plenum may occur from the lateral periphery of the reactor core or from the bottom of the core. In both situations, the thermal-mechanical attack of the melt on the surrounding metallic support structures (for instance, lower support core plate, barrel, etc.) should be considered. The degradation process of these structures by ablation and melting could produce a pathway down to the lower plenum that may affect the rate of melt relocation and metallic content of debris in the lower head.

The reactor vessel integrity could be directly threatened if significant amounts of molten material relocate in the lower head. Phenomena such as steam explosion, or jet impingement on the lower head walls may cause severe damage to the reactor vessel.

The new generation of nuclear power plants (advanced design concepts) have implemented design features aimed to assuring in-vessel retention of molten debris. Two examples of the assumed measures to prevent lower head failure due to creep rupture are the following. First, the possibility to depressurize the vessel, thereby decreasing the differential pressure loads imposed on the lower head, and the elimination of penetrations through the lower head, so failure mechanisms associated with such penetrations are precluded.

CAREM is a project for an advanced, simple and small nuclear power plant (100 MW_t, 27 MW_e), jointly developed by CNEA (Comisión Nacional de Energía Atómica) and INVAP SE in Argentina.

Although CAREM is conceived as a new generation design reactor, standing on the large worldwide experience accumulated in the safe operation of Light Water Reactors (LWRs), the possibility to include other features to enhance its safety characteristics is still open. Among the possible features, the inclusion of a passive metallic in-vessel container is being considered to arrest the reactor vessel meltdown sequence during a core-damaging event, and thereof prevent its failure.

The in-vessel core catcher [2] is a totally new passive design feature aimed to assure in-vessel retention of molten debris, and to prevent lower head failure in CAREM. Nonetheless the concept seems to be quite promising and effective, it is necessary to establish the adequacy and conditions on which the container may operate and be more efficient (for instance, geometrical dimensions (mass of catcher), whether it will self-sufficient or needs a backup feature such as cavity flooding, etc.)

The objective of this paper is to analyze the severe accident sequence originated by the spurious opening of the first safety relief valve (SRV), and the relocation processes taking place in the core and lower head of reactor vessel, using the code MELCOR 1.8.4 [3].

It is assumed that relief valve remains in the stuck open position during the whole transient.

The predicted results provide useful information to dimension the in-vessel metallic core catcher (e. g., debris energies, events timing and the amount of relocated molten material in the lower plenum).

CAREM nuclear power plant (NPP) simulation model comprises the reactor and the containment system. A complete sequence of events and the corresponding fission product release fractions of CAREM NPP were simulated with MELCOR. The simulation ended at 300000 s after most significant reactor events took place.

BRIEF DESCRIPTION OF MELCOR

MELCOR is a fully integrated computer code (developed at Sandia National Laboratories for the U.S. NRC) that models the progression of severe accidents in LWRs. The current version of the code is MELCOR 1.8.4.

An important feature of MELCOR is its integrated approach to accident phenomenological modeling. Both in- and ex-vessel phenomena are treated in the context of a single, integrated code. All relevant phenomena and system are treated, although simplification is used where appropriate.

MELCOR contains a number of physics packages or modules, which model all essential phenomena and plant features. Either BWR or PWR systems may be modeled. The MELCOR core package (COR) calculates the thermal response of the reactor core and lower plenum internal structures. The core and lower plenum regions of the reactor pressure vessel are divided into a user-specified number of concentric radial rings and axial segments that define the so-called core cells. Inside the cells, one or more intact structures (fuel, cladding, control rods, reactor internals, etc) or particulate debris are modeled.

The core degradation model, included in the COR package, calculates the liquefaction of the intact structures caused by eutectic reactions and dissolution processes within the core cells. Molten portions of intact structures are transferred to the conglomerate debris associated with the structure. Molten materials are relocated downward by the candling model, and intact components are converted to debris if various debris formation criteria are met.

Logical processes (instead of rate processes) model the relocation of particulate debris downward by gravitational settling through consideration of volume, porosity and support constrains. Gravitational leveling of molten pools and debris bed across the core rings is calculated with a user adjustable time constant. Whenever mass is relocated or debris formed, material energies in the new or changed component are re-evaluated and the temperature updated to maintain thermal equilibrium, and relevant geometric variables are recalculated to reflect the change in geometry.

Failure of the core support plate (or any steel structure providing support) due to decreasing yield strength at higher temperatures is triggered whenever the steel temperature in a cell reaches a user-specified failure temperature or by a user-specified control function. The relocated core material may eventually collect in the lower head if all the intermediate barriers (e.g. lower core support plate) fail.

The falling debris quench heat transfer to liquid water pools may be calculated (through a user-specified quench heat transfer coefficient), when failure of the lower core support plate triggers the relocation of hot debris from the core region to the lower plenum. In the case that falling-debris quench heat transfer is deactivated, the debris is assumed to relocate instantaneously from the core region to a dried-out debris bed in the lower plenum with an overlying pool of water if present.

The MELCOR 1.8.4 code, released in July 1997, is used in this analysis. This version contains many code improvements, including the capability to maintain fuel integrity based on the thickness and temperature of zirconium and zirconium oxide, modeling of TMI-type core blockage, and detailed modeling of the lower plenum.

CAREM GENERAL DESCRIPTION

CAREM is a project for an advanced, simple and small nuclear power plant, conceived with new generation design solutions and standing on the large worldwide experience accumulated in the safe operation of Light Water Reactors.

The CAREM is an indirect cycle reactor with some distinctive and characteristic features that greatly simplify the reactor and also contribute to a higher level of safety:

- Integrated primary cooling system.
- Self-pressurised.
- Primary cooling by natural circulation.
- Safety systems relying on passive features.

The primary system is integrated, that means the whole high energy primary system (core, steam generators, absorbers rods drive mechanisms and pressurising system) is contained inside a single pressure vessel.

Reactor coolant circulates by natural convection. The driving force is the density difference between the hot and cold legs of the loop. Self-pressurisation of the primary system is the result of natural trend towards the liquid-steam equilibrium. The large dome volume contributes to the damping of pressure perturbations. Due to self-pressurisation, bulk temperature at the core output corresponds to saturation temperature at the primary pressure. Conventional PWR's heaters and sprinklers are thus eliminated.

The steam generators are of a 'Mini Helical' vertical, 'once through' design. Coolant in the primary and secondary sides flow in counter-current with secondary flow coolant flowing upwards inside the tubes. The secondary system exits the steam generator with ample superheating.

The fuel element cross section is hexagonal. Each fuel element contains 108 fuel rods, 19 guide tubes, and 1 instrumentation tube. The core has 61 fuel elements.

The reactor protection system activates the residual heat removal system in case of blackout. The reactor decay heat is transferred to the pressure suppression pool by natural convection, and the pool water inventory is large enough to receive the residual heat for a period of at least 48 hours after the reactor scram without the need for makeup supply.

The emergency injection system prevents the core exposure in case of LOCA. The system injects borated water flooding the RPV. The system provides at least 48 hours of protection to the core after the accident initiation.

The containment system, where the primary system, the reactor coolant pressure boundary and important ancillary systems are enclosed, is a cylindrical concrete structure with an embedded steel liner. It is a pressure suppression type containment and assures that at least 48 hours after initiated the accident, without any external action, the pressure in its interior is kept below the maximum design value of 5 bar.

The pressure relief system has three relief valves to protect the integrity of the reactor pressure vessel against overpressure.

The CAREM safety systems design follow standard state of the art trends in nuclear energy regarding redundancy, physical separation, diversification, independence, testability, fail-safe principle and by-passing resistance.

CAREM NPP NODALIZATION FOR MELCOR

The MELCOR NPP nodalization used for this calculation is shown in Fig. 1. It consists of eight control volumes (CV) (7 for the reactor vessel and internals, and 1 CV for the containment system) representing the major hydrodynamic characteristics of CAREM. CV310 represents the downcomer annuli. The lower head and core volumes are represented by CV320 and CV330 respectively. The so-called chimney comprises volume CV340, which is connected with the upper plenum, CV350. The twelve steam generators (SG) were lumped into one volume, and this is represented by CV360. The bypass zone between SGs housings and reactor barrel is depicted by CV370. The SG secondary side is modeled using control functions representing the power conversion system.

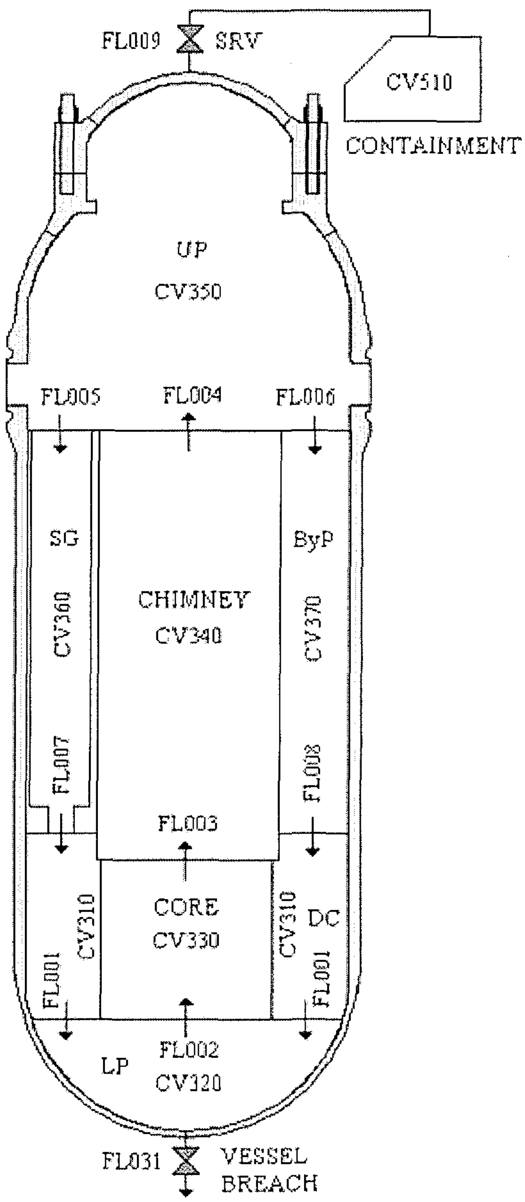

Fig. 1. MELCOR nodalization of CAREM

The leak was modeled using a valve positioned at the top of reactor vessel. Another valve, at the bottom of the vessel, represents the reactor lower head failure when conditions for that are reached. Both valves are controlled by suitable control functions enabling their operations.

The flow paths for the reactor vessel consist of twelve junctions, while the containment system comprises only one junction (reactor relief valve path).

To represent the heat transfer surfaces in the model thirty-four heat structures (HS) were included in the reactor vessel and one HS in the containment.

The core region was divided into three radial rings (containing 7, 30, and 24 fuel elements). This subdivision obeys to the CAREM core fissile-control material composition. A total of 51 calculation nodes were established to analyze the core behavior. The axial subdivision comprises seventeen levels. The axial subdivision in the lower plenum (including the lower core support plate level) consists of five levels. There are ten active core levels and two non-active core levels representing the inlet and outlet core regions of the fuel elements.

A nodalization scheme of the core and lower plenum for the COR package is shown in Fig. 2. Four temperature nodes were selected to study temperature profile in the lower head.

Decay heat in core after the scram, is generated using a control function and the information from the radionuclides package. Values for the control function were obtained from the preliminary safety report of the NPP [4]. Initial fuel-cladding gap inventory fractions are considered in the model.

An out-of-code initialization procedure was applied to obtain steady-state initial conditions for CAREM reactor simulation model.

Table 1 shows the comparison between CAREM design features and MELCOR calculation parameters.

TABLE 1.
Initial Full-Power Steady State Conditions

Variables	Design	MELCOR
$N_{reactor}$ (MW)	100.0	100.0
$P_{reactor}$ (MPa)	12.25	12.25
H_{water} (m)[1]	7.76	7.75
G_{core} (kg/s)	410.0	410.5
T_{out} (K)	599.15	599.4
T_{in} (K)	557.15	558.0

[1] reactor nominal water level

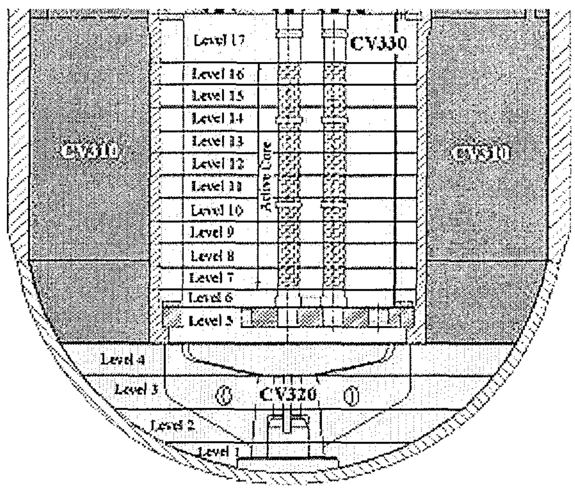

Fig. 2. CAREM COR nodalization.

BRIEF DESCRIPTION OF ACCIDENT SCENARIO

Prior to the accident sequence, the steady-state initial conditions correspond to the nominal full-power state of the NPP as shown in Table 1. When the accident begins the SRV opens (spuriously), and remains in this position during the whole transient, leading the system to a loss of coolant accident (LOCA). Once the accident has begun, the SG feed-water flow is immediately reduced to zero and no emergency feed-water is considered. The reactor scram follows reducing core power to the corresponding decay heat level.

No credit has been given to operators during the accident to apply recovering actions, aiming to create the most adverse conditions in the system.

The calculations encompass major reactor events, and stop when lower head fails and debris are ejected to the cavity.

ANALYSIS OF CAREM SRV STUCK OPEN ACCIDENT SEQUENCE

The SRV stuck open accident was initiated at 0 s. The main system responses are originated from the LOCA through the open valve. The sequence of events is shown in Table 2. The corresponding major plant responses are shown in Figs. 3 through 17.

Once the accident begins, reactor pressure drops sharply in the first 100 s as shown in Fig. 3. The quick reactor depressurization leads to a flashing of primary system coolant. Natural circulation stops due to the disruption caused by the leak in the uppermost part of reactor vessel. At 1764 s, the water level in the reactor vessel has dropped to the inlet part of the SGs. From this moment on, the SGs heat transfer surfaces start to uncover and SG bypass section to empty. Chimney is also subject to the same conditions, but the emptying process inside is slower. The responsible for this delay, is the resistance created by the presence of barrel internal structures (lower and upper bushing supporting plates, etc.). Fig. 4 shows the reactor levels behavior.

TABLE 2.
Sequence of events in SRV stuck open accident

Event	Time (s)
SRV stuck open starts	0
UP water level lost	1764
Core uncovers (TAZ)	15000
Gap release in Ring 1	24231
Gap release in Ring 2	24291
Oxidation starts in Ring 1	25000
Oxidation starts in Ring 2	25000
Oxidation starts in Ring 3	25000
Gap release in Ring 3	25191
Core uncovers (BAZ)	50003
Core support plate fail in ring 2	55127
Reactor vessel dryout	60000
Reactor equals cont. Pressure	70000
Core support plate fail in ring 1	73001
Core support plate fail in ring 3	73056
Lower head fail in ring 1	253319
Debris ejection to cavity starts	253319
Simulation ended	300000

UP= upper plenum
TAZ= top of active zone
BAZ= bottom of active zone

The core starts to uncover at ~ 15000 s, when water level drops to the top section of fuel channels, and chimney is occupied only by steam. No core heat-up is observed at this moment, because the channels are still refrigerated by the upward flow created by the leaking valve (0.4 kg/s). This process continues up to 50000 s, when the bottom of the core is reached.

The fuel channels heat-up begins due to the combination of several factors. The loss of coolant and the deterioration of heat transfer (steam atmosphere surrounding the fuel elements), and the decay heat generated in the core, are the promoters of this heat-up. At 24231 s, a gap release takes place in ring 1. Relocation in ring 1 started at ~25000 s as the cladding temperature reached its melting point. Fig. 5 and 6, show liquid and gas temperatures in the reactor vessel, respectively. The gap release in ring 2 and 3 follows at 24291 s and 25191 s, respectively. The oxidation process triggers at ~25000 s almost simultaneously in all the rings. The relocation process in each ring, called the candling phenomenon, is shown in Fig. 7, 8, and 9.

Cladding temperature response during the accident sequence is shown in Fig. 10, 11, and12. During the accident, because of metal-water-reaction (MWR), 56.6 kg of hydrogen were generated in the reactor vessel, with total water consumption of 506 kg. At 60000 s, the reactor vessel dries out, leading the accident sequence through dryout conditions.

As relocation started, the fuel relocated to the core lower support plate (level 5). The structure temperature starts to rise as shown in Fig. 13. The temperature rises from ~ 450 K (at 50000 s) to 1273 K (at 55127 s), where the support plate fails in ring 2. The other two rings failure took place at ~73000 s. When this happen, 7766 kg of molten material and debris are present in the lower head of reactor vessel as shown in Fig. 14. This mass represents the 96% of material composition of the intact CAREM core.

The existence of a radial temperature gradient between the radial rings at this level could be observed as follows. The particulate debris in ring 1 of the lower head has a temperature of 2111 K, meanwhile, the particulate debris in ring 3 has a temperature of 1563 K, and barrel support structures have a temperature of 1351 K.

The relocated mass induces a sustained temperature rise in the lower plenum wall. The weight and decay energy of this mass lead to lower head creep-rupture failure at 253319 s and the corium is ejected to cavity. The failure occurs as a response to mechanical loading of the vessel, under conditions of material weakening at elevated temperatures. The lower head temperature is shown in Fig. 15, 16 and 17.

The accident sequence simulation ended at 300000 s soon after debris ejection to the reactor cavity.

CONCLUSIONS

The MELCOR input deck for CAREM NPP was established. The severe accident sequence induced by the safety relief valve stuck open event was analyzed with MELCOR 1.8.4 starting from the full-power steady-state conditions. Major phenomena, such as boil-off, heat transfer degradation, reactor vessel dryout, core heat-up, oxidation, melting, relocation, and vessel failure, were simulated in this study.

The predicted sequence of events up to vessel failure spanned for 3.0 days. Compare to other non-advanced designs, results stated that CAREM vessel would stand more time, even if no injection were achieved. This is a clear advantage of the design.

During the accident sequence, the 96% of the reactor core (7766 kg) is relocated in the lower plenum. The results provide very useful information not only to dimension the in-vessel metallic core catcher, but also to the reactor designers, and Probabilistic Risk Assessment (PRA) of CAREM.

This tool will be applied also to the PRA, and the severe accident management study of the CAREM

NPP. The nodalization could be used to simulate other alternative features to arrest reactor vessel failures, e. g., such as cavity flooding.

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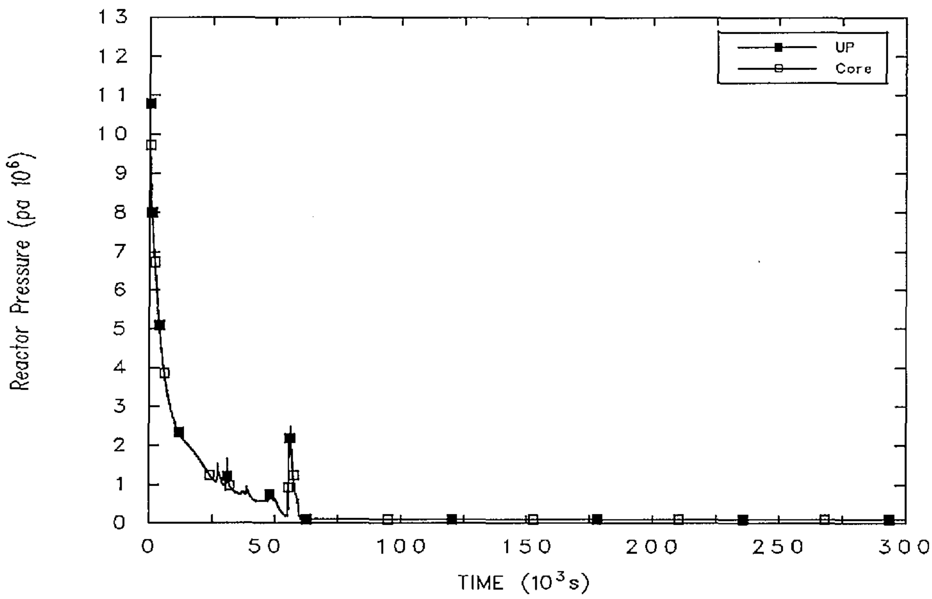

Fig. 3. Pressure in the reactor vessel.

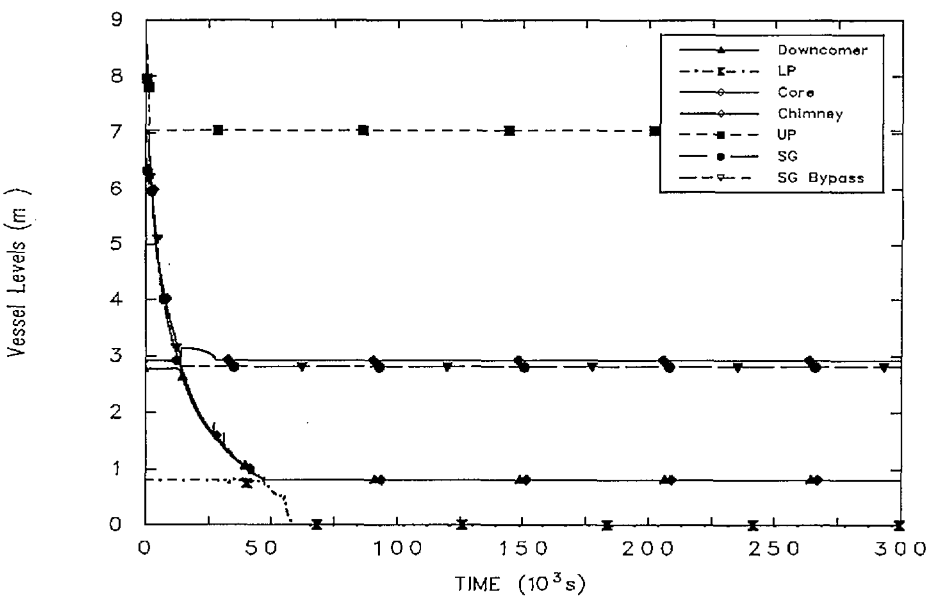

Fig. 4. Reactor vessel levels.

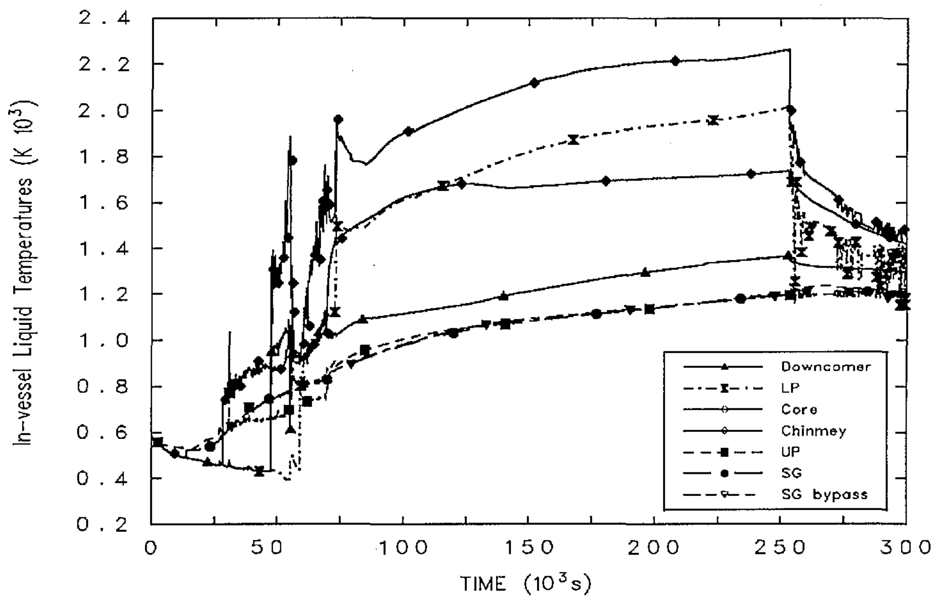

Fig. 5. Reactor liquid temperatures.

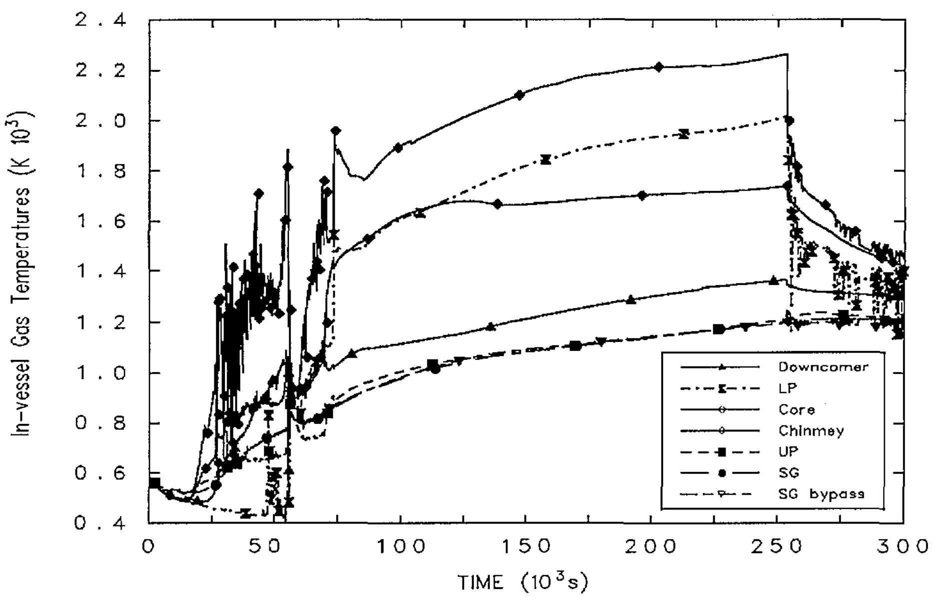

Fig. 6. Reactor vapor temperatures.

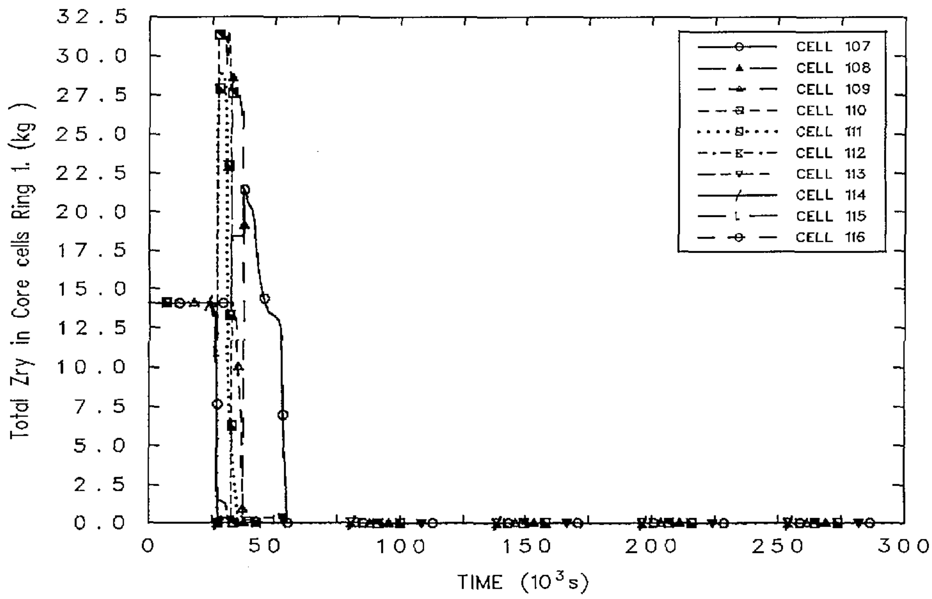

Fig. 7. Cladding mass variation in ring 1.

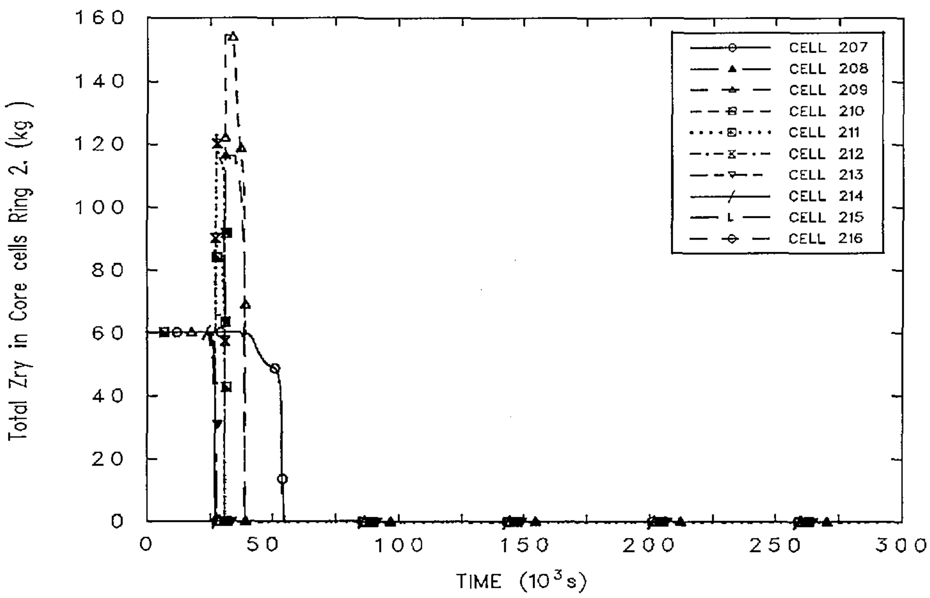

Fig. 8. Cladding mass variation in ring 2.

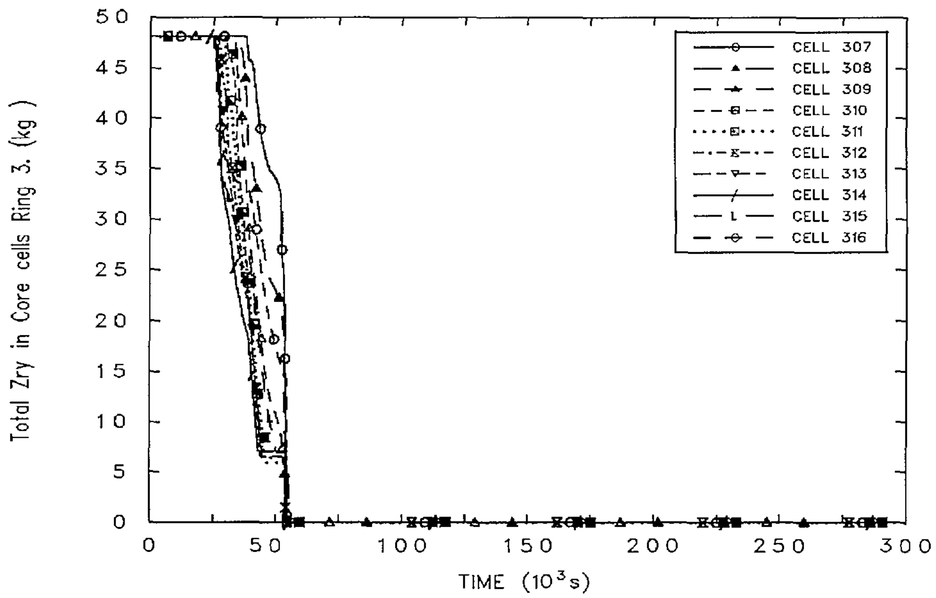

Fig. 9. Cladding mass variation in ring 3.

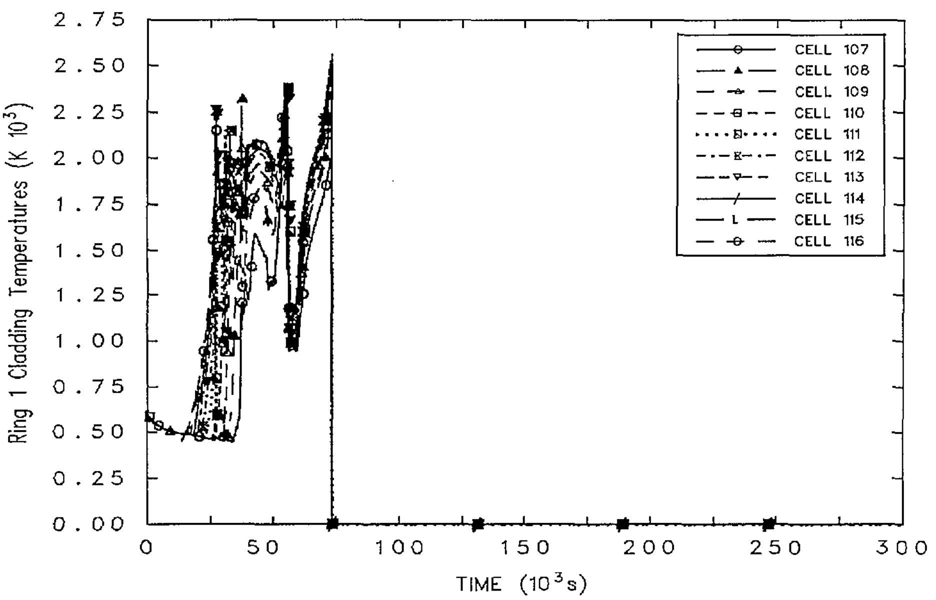

Fig. 10. Cladding temperatures in ring 1.

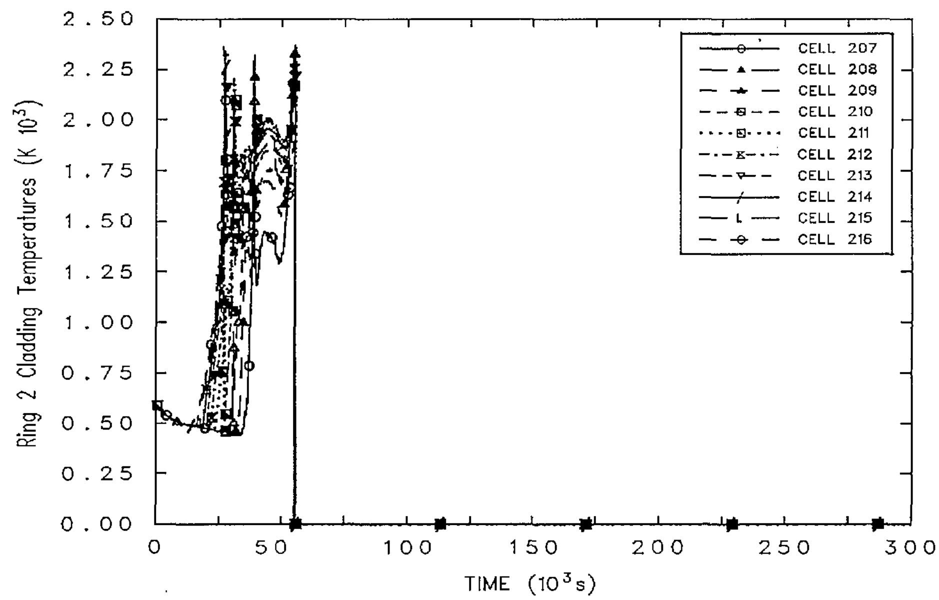

Fig. 11. Cladding temperatures in ring 2.

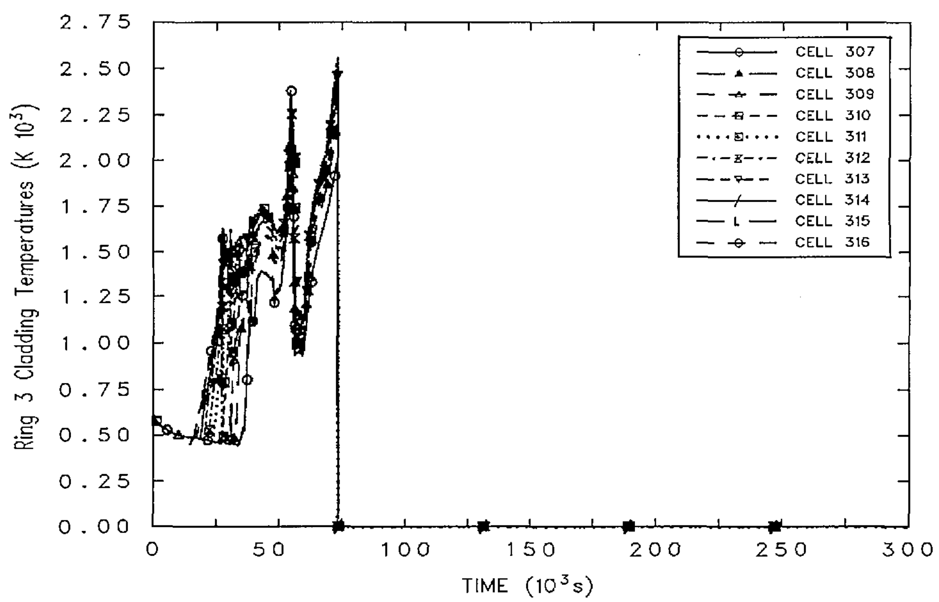

Fig. 12. Cladding temperatures in ring 3.

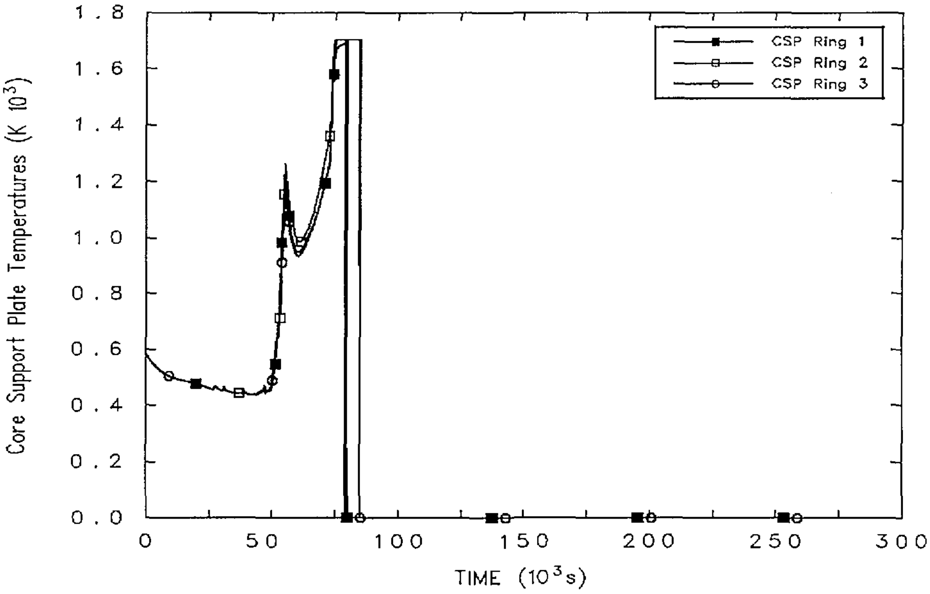

Fig. 13. Core support plate temperatures.

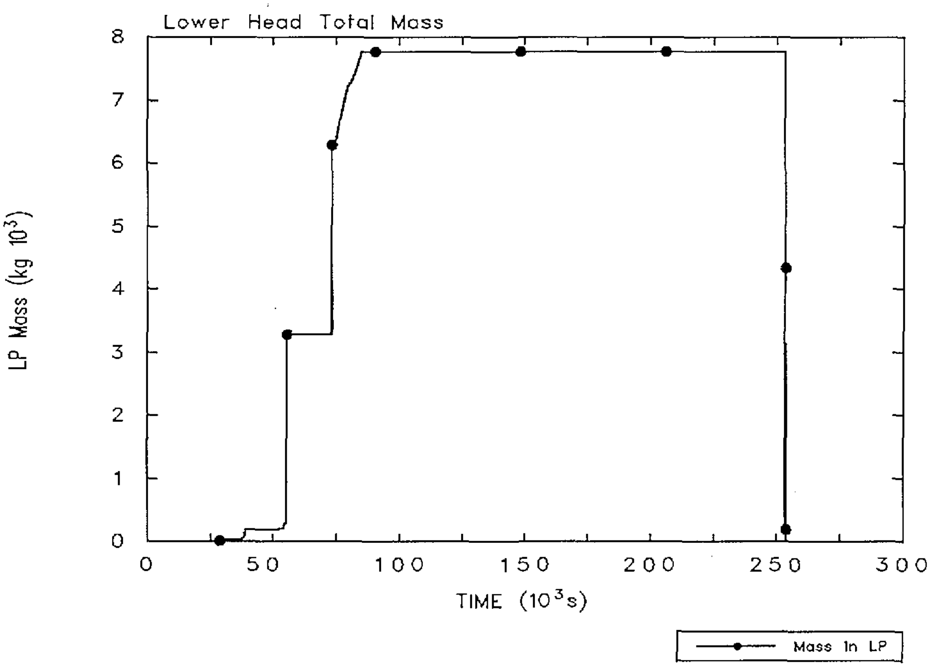

Fig. 14. Lower plenum mass inventory.

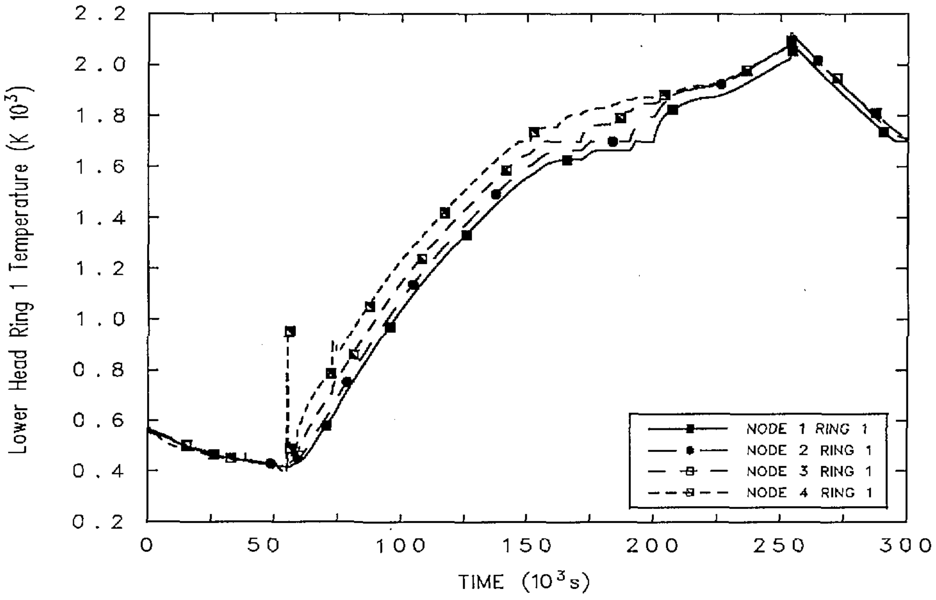

Fig. 15. Lower head temperatures at ring 1.

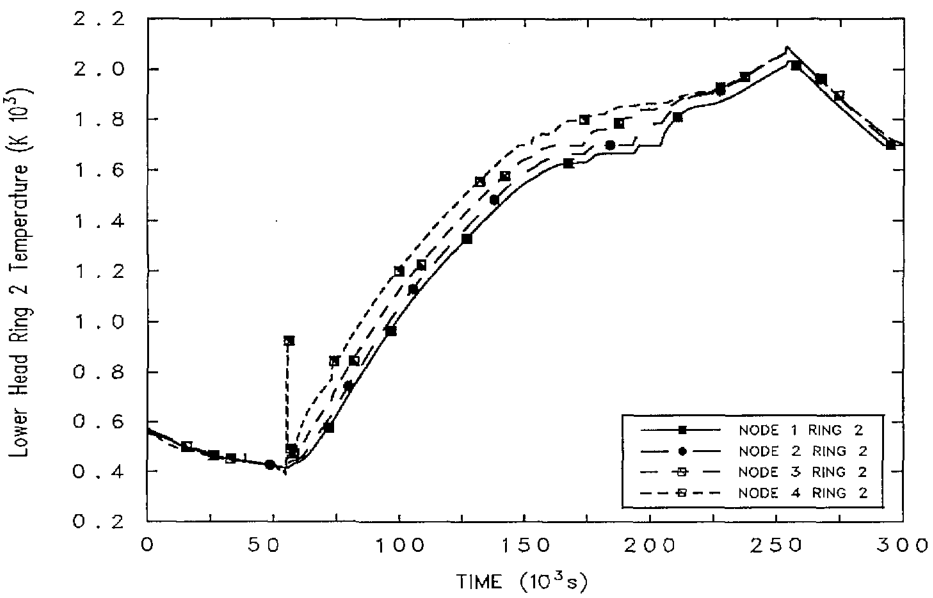

Fig. 16. Lower head temperatures at ring 2.

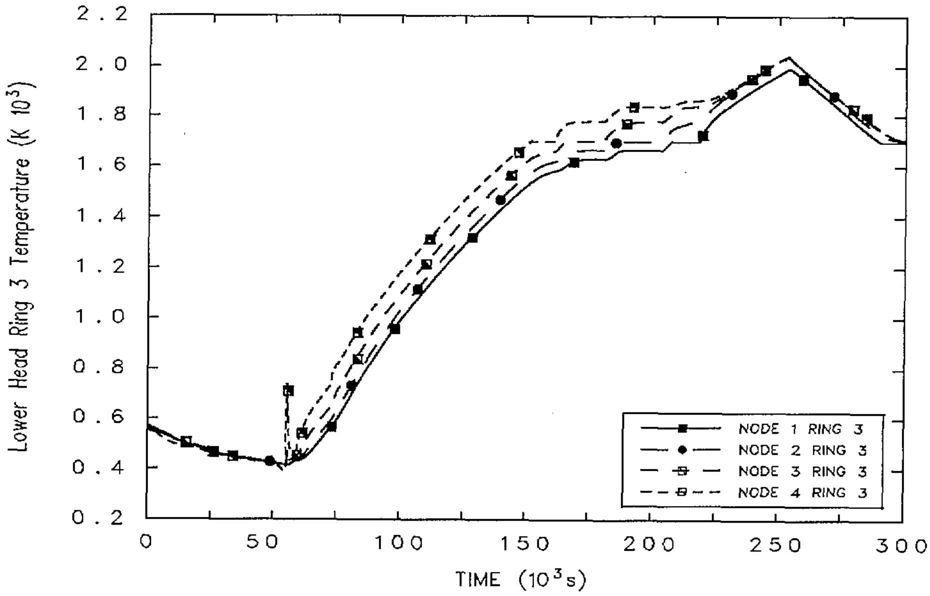

Fig. 17. Lower head temperature at ring 3.