INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL SYMPOSIUM ON IMPROVEMENTS IN WATER REACTOR FUEL TECHNOLOGY AND UTILIZATION

Stockholm, Sweden, 15–19 September 1986

IAEA-SM-288/ 33

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RECENT DEVELOPMENTS IN POWER REACTOR FUEL IN ARGENTINA

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ABSTRACT

At present two nuclear power stations are in operation in Argentina: a 350 MWe PHWR and a 600 MWe Candu type reactor. A third one of 680 MWe of the PHWR type is under construction. Planning for additional 700 MWe nuclear power is foreseen for the rest of the century. A small power reactor is under development (15 MWe-CAREM 15), designed for serving remote or low population areas in non-interconnected electrical grid systems.

Domestic participation in nuclear fuel design and development activities has increased greatly in the period since the start-up of the first nuclear power station to the present, as a result of the self-sufficiency policy in the nuclear fuel cycle.

PHWR fuel was optimized in order to simplify domestic material procurement, production operations or better fuel utilization.

Testing of the basic design of fuel assemblies and the supply of the first core for the 680 MWe power station under construction are being done in the country.

HWR fuel technology development was done in the country. While the Candu fuel basic design is well known, the detail design of this fuel is a task of fuel suppliers. For this reason, we have completed the detailed design, testing and manufacturing technology.

The development of special production equipment played a significant role in the domestic manufacturing technology and allowed us to become equipment suppliers for low scale production plants.

One version of the CAREM-15 fuel assembly is presented. The main features are the use of segmented fuel rods integrated in a 1.2 m long fuel assembly containing the guide tubes for control rods and heterogeneous burnable poison.

Compartmentalized fuel rod design with a low linear heat generation rate, the incorporation of geometrical and mechanical conservative parameters and the inclusion of PCI remedies, allow adequated operating margins and the flexibility for load following operations.

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1. INTRODUCTION AND BACKGROUND

The Argentine nuclear programme is based on both the use of domestic uranium reserves and on locally engineered projects of power stations installation with increasing domestic participation. (1)

Assured uranium reserves are approximately 30.000 Tn and the country has a highly probable potential for ten times this figure. (2)

A cautious planning for installation of nuclear power stations based on natural uranium and heavy water moderated reactors is being carried on since 1966.

Two nuclear power stations are at present in operation: a 350 MWe PHWR and a 600 MWe HWR type. A third one of 680 MWe of the PHWR type is under construction.

Additional 700 MWe nuclear power are foreseen for the rest of the century according to the economical and industrial growth expected in the near future.

The design and engineering of this station will start soon fully managed in the country.

Ours is a large and expanded country showing remote, isolated and scarcely populated areas. The long distances do not allow to install transmission lines which are highly expensive and subjected to climatological damages. For that purpose, we have under development a very small power plant, for production of electricity and industrial steam for urban heating or sea water desalination. It can also be used for building bigger electricity plants by adding more reactors, with an economic limit of around 150 MWe.

The nuclear fuel cycle is up to now industrially completed in the front end, while research installations are under construction or in operation for the back end.

Present capacity or potential expansion of industrial installations allow to supply fuel for all power stations planned for this century.

Domestic participation in nuclear fuel design and development activities have increased greatly in the period since the start-up of the first nuclear power station to the present, as a result of the self-sufficiency policy in the nuclear fuel cycle.

Participation in the design offices of the reactor designer for the first power station continues with a performance evaluation work and finally with their optimization.

PHWR fuel was optimized in order to simplify domestic material procurement, production operations or better fuel utilization.

The most important work was the study for the inclusion of a 37th fuel rod instead of a structural tube. This work included the modelling of fuel assemblies vibration behaviour, the test in a low pressure loop with fully instrumented prototypes and a performance test in the high pressure loop.

At present, these results can be used for the design of the fuel assemblies of the 680 MWe PHWR under construction.

Strip Zircalloy spacers instead of spark-eroded Zry spacers have been developed for the Atucha-I Power Station. Structural calculations, prototypes construction and testing, both in laboratory and in out-of-pile tests are being done or are planned.

Present status of these spacers design engineering encourage us to foresee its use in the Atucha-II Power Station fuel assemblies.

The introduction of slightly enriched uranium prototype fuels and a programme for MOX fuel recycling is also planned.

Testing of twelve prototypes 0,85% U-235 enriched is under way in Atucha-I Power Station. Changes in specifications for production and control charts are being done in our Design Group.

The development of MOX fuel technology started several years ago. Plutonium handling in an Alpha-Facility started early in 1980 and the production of MOX pellets and fuel rods is currently done.

Experimental irradiations are planned in research and power reactors.

The project for the installation of a second Alpha-Pu Facility in the Reprocessing Pilot Plant building is being prepared.

As suppliers for the 600 MWe HWR, we have done the detail design of fuel as well as the necessary out-of-pile testing and in-reactor irradiation to qualify our fuel bundles.

The development of special production equipment played a significant role in the domestic manufacturing technology and allowed us to become equipment suppliers for low scale production plants. (3)

The design of the fuel assembly for the 680 MWe PHWR was done in a joint venture with the plant designers (KWU). At present, its last version is being tested in the high pressure loop after a test with an instrumented fuel assembly carried out in the low pressure loop of our laboratory. (4)

Self-sufficiency in design engineering capability is being completed with the design of fuel rods and assemblies for low power reactors described below.

2. LOW POWER REACTOR FUEL

2.1 Main Features of the Reactor

The low power reactor CAREM-15 is a modular concept with enriched uranium fuel and light water cooled. (Fig.1)

It has an integrated primary circuit, it is self-pressurized, cooled by natural convection and has a passive-type emergency cooling system.

The main features of this reactor design engineering are: modular design, as few as possible on-site work, possibilities of increasing the power of the system by adding more reactors, components or subsystems easy to transport, self-controlled reactor, passive emergency system.

From the point of view of economics, this type of reactor allows to have: low engineering cost, lower licensing cost, small industrial infrastructure, lower Q.A. cost, lower cost on account of the use of standardized components, lower interest-rates, lower dead times during the projects and in the industries, lower radiactivity inventory and hence, higher public acceptability.

Moreover, it has lesser requirements for the siting and for the stand-by machines in the girds, short distances between the plants and the customers, lower spare parts stock and in-turn lower maintenance cost.

2.2 Fuel Assembly Design

Several versions of fuel assemblies design were proposed and studied. Hexagonal shape was chosen because of the compactness of this array and the possibility of reaching a quasi-circular section of the core.

Three versions of fuel assemblies were selected for detail engineering and testing.

The first one consists in a conventional PWR fuel assembly with two grids, upper and bottom, integrated guiding tubes for control rods and heterogeneous burnable poison, and spacer grids in-between. (Fig.2)

Fuel rods are free in the "box" and have enough place for longitudinal differential elongation.

The main feature is the extremely low pressure drop required by the thermohydraulic design due to the natural convection cooling.

Fuel assemblies are fixed from the bottom by a kind of mast that keeps the assembly in position and presses it against the reactor grid.

A cluster of control rods or burnable poison are placed in the guiding tubes. Control rods are mechanically connected to the driving systems while the cluster of burnable poison rods are static in a fixed position.

A second version of fuel assembly consisting in a box formed by two end grids, six lateral edges and the control rods-burnable poison guiding tubes. (Fig.3)

This box accommodates three to six short bundles made of compartmentalized fuel rods welded in both end caps to each end-plate.

This assembly allows to have small bundles shaped for sectors of the cross section.

The short bundles are axially loaded to prevent vibration.

The third version of the fuel assembly is the pile-up of three 40 cm. long bundles laser welded in its end plate junction.

No standard end plates are foreseen but a bottom piece attached to the reactor grid where the assembly is placed and fixed. (Fig.4)

2.3 Fuel Rod Design

Characteristics of fuel rod design parameters are shown in Table I.

Low linear heat generation rate and the selection of adequate geometrical parameters allow to have a very low restrain of the cladding at a steady state condition.

Compartmentalized fuel rods were selected to be compared with the full length rods, with the intention of reducing primary coolant contamination in the event of fuel failures.

Two different fuel rod concepts were analyzed by using the BACO code (5): a full length, 120 cm long rod (FL), and three compartmentalized 40 cm long rods. Due to the assumed symmetrical axial flux distribution in the reactor core, only two compartmentalized rods were analyzed: CN and CL, for high and low power.

A two years long, constant power operation was studied. Due to pellet swelling and some cladding creep-down (despite the low stresses characteristic of a thick, small diameter sheath), mechanical contact between fuel and cladding was predicted in some cases. Nevertheless, the hoop stress at the cladding inner surface remained compressive until the end-of-life.

As already stated, compartmentalization is intented to minimize coolant contamination in case fuel failure occurs. In order to quantify this effect, it was assumed that -in case of cladding rupture- both the fission gases released to the rod free volume and those stored at the UO_2 grain boundaries could be released to the coolant. Fig. 5 shows the volume V of fission gas released and stored at the grain boundary as a function of irradiation time, for the FL and CH rods. It can be seen that the potentiality for coolant contamination of the CH, compartmentalized rod, is approximately one half the potentiality of a full-length rod.

A daily load-follow operation was also studied. Fig. 6 shows the assumed daily power cycle, where a 15% overpower -simulating the effect of Xe oscillations- was considered in each early morning power-up ramp. In order to obtain an upper-bound to fission gas release during power cycling, extremely high ramp velocities (625 W/cm/h) were considered during the whole power history.

Fission gas release during power ramping was considered the most critical phenomenon in this analysis. For this reason, two different FGR models were used in the BACO code: an overpredicting, highly conservative one (Model I), and a more realistic one (Model II).

Fig. 7 shows fission gas release predicted by both models for the CH rod under power cycling;as a reference, He content for these pre-pressurized rods is also plotted. Under constant power operation, fission gas release is neglectable in all cases ($<1\%$).

Fig. 8 shows the time evolution of the internal gas pressure under load-following operation. For the FL rod, the conservative Model I predicts a slight internal overpressure (compared to the coolant pressure) at ECL; Model II, not shown here, gives no overpressure at all. For the low pressure, compartmentalized rod (CL), both Models predict the same, and gas pressure keeps below the coolant pressure. For the CH rod both Models predict internal overpressure at EOL: 83% using Model I, and 47% with the more realistic Model II. Nevertheless, under normal operation, the high overpressure predicted by the conservative Model I has no major effects: fuel centerline temperature keeps below 1300 degC (1100 degC if Model II is used), while the cladding diameter increase at EOL -mainly due to creep-out- is $<0.1\%$ (Model II predicts a small diameter contraction at EOL, due to the accummulative effect of early creep-down). Due to cladding creep-out, no PCMI is predicted with Model I.

Fig. 9 shows the hoop stress at the cladding inner surface for the same cases described in Fig. 8. Plotted values correspond to the: i) overpower at the end of the early morning power-up ramps (marked as "peak"); ii) the nominal power (marked as "std"); and iii) low power during the nights (marked as "low"). For the sake of clearness, the predicted daily oscillations between these three bounds were ommitted. Due to the low diameter to thickness ratio, cladding stresses are low in all cases, and well below any reasonable failure limit for Zry. Model II, compared to Model I, predicts not only lower stresses but also lower stress increments per unit time. The peculiar shape of the curve for peak power using Model II is due to the fact that pellet-clad contact is predicted between 420 and 670 days; afterwards, increasing gas pressure breaks the contact.

Under LOCA conditions, the high gas pressure predicted by using the conservative Model I for FGR may result in large ballooning and even rupture of the cladding. This is restricted to the high-power compartmentalized rods (one third of the total) at the central part of the reactor core. An analysis of the reactor radial power profile shows that under LOCA conditions, only 4% of the total number of fuel rods would fail.

3. CONCLUSIONS

Since 1970 we have continuously developed the capacity for engineering and performance analysis of power reactor fuel.

This capacity was used for improving PHWR fuel and detail design of HWR fuel type.

The most recent work is the development of CAREM-15 Reactor fuel conceived for servicing at load-following operation.

Compartmentalized fuel rods may be the best choice to reduce the amount of radioactivity released to the primary coolant in case of fuel failures.

Testing of different alternatives will provide the necessary support for the selection of the first core fuel.

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TABLE 1. CHARACTERISTICS OF CAREM-15 FUEL

FUEL ASSEMBLY

- Quantity	37
- Hexagonal array	
- Pitch	1.38 cm
- Fuel Rods	102
- Control or burnable poison rod per assembly	24
- Two end plates of SS 304 L	
- 3 Spacers of Inconel	

FUEL RODS

Zry Cladding
OD = 8.95 mm
ID = 7.71 mm
C = 0,6 mm
UO_2 pellet
d = 7.58 mm
δ = 92,6% td
Enrich.= 4% U-235
Active length = 1200 mm

CONTROL RODS

Material: Ag, In, Cd
Cladding: SS
OD: 8 mm

BURNABLE POISON RODS

Material: CB - Al_2O_3 - Si
Cladding: SS
OD: 8 mm

FUEL ROD OPERATING PARAMETERS

Max. LHGR = 235 W/cm
Max. Tc = 950°C

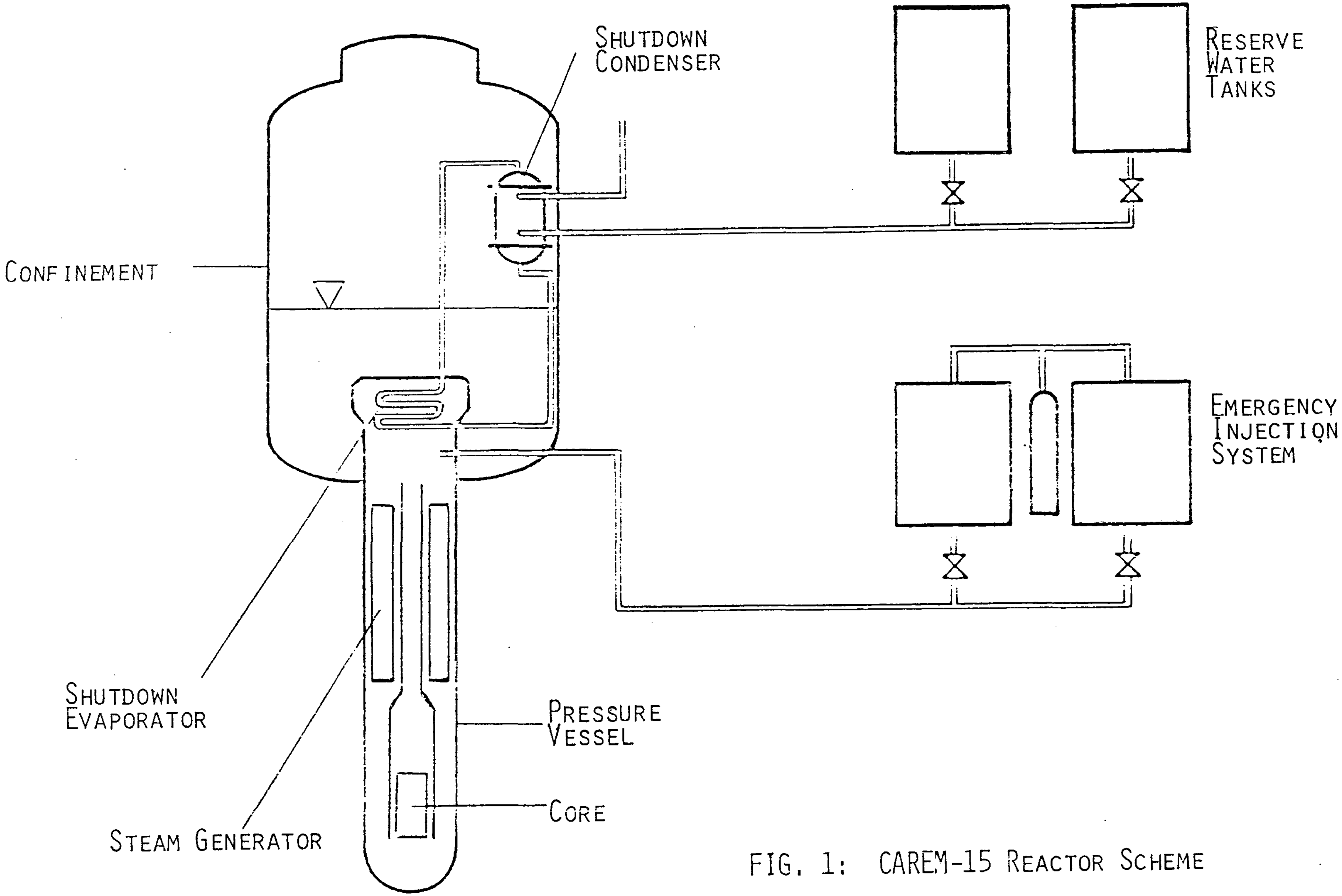

FIG. 1: CAREM-15 REACTOR SCHEME

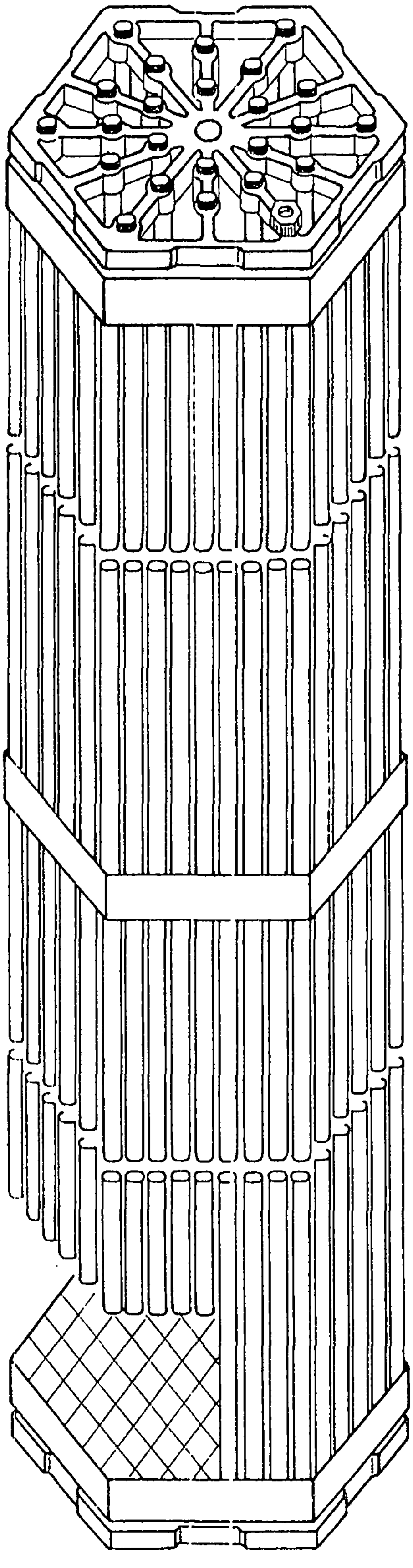

FIG. 2: CAREM 15 - Fuel Assembly (Version A)

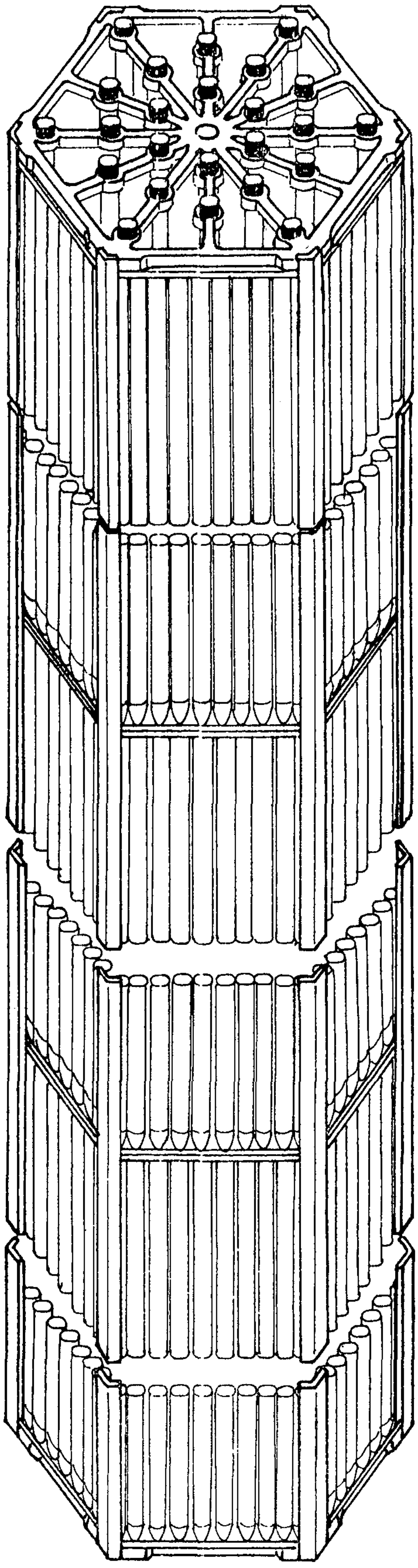

FIG. 3: CAREM 15 - Fuel Assembly (Version B)

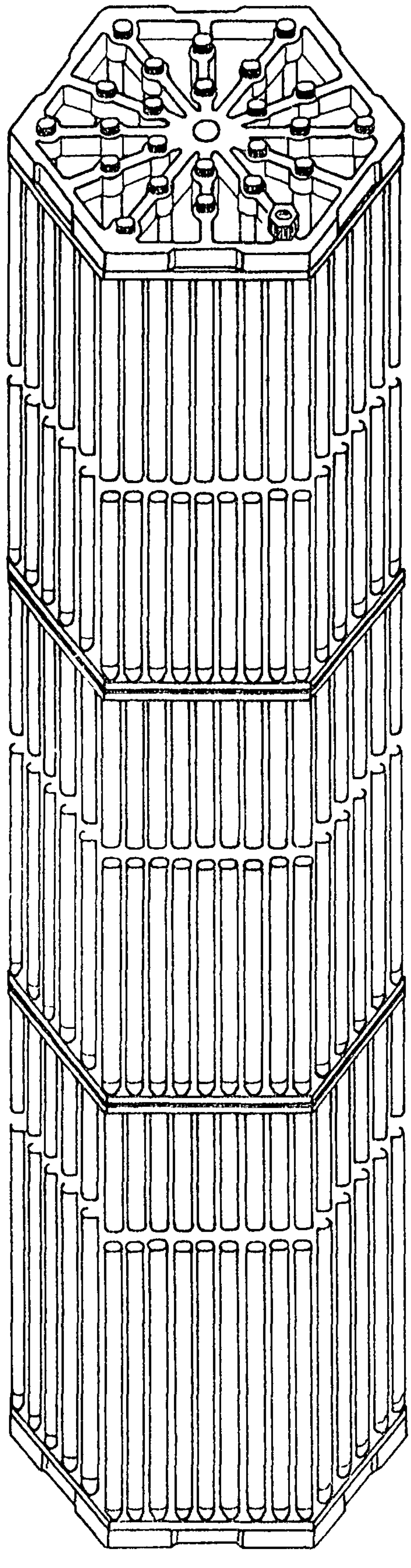

FIG. 4: CAREM 15 - Fuel Assembly (Version C)

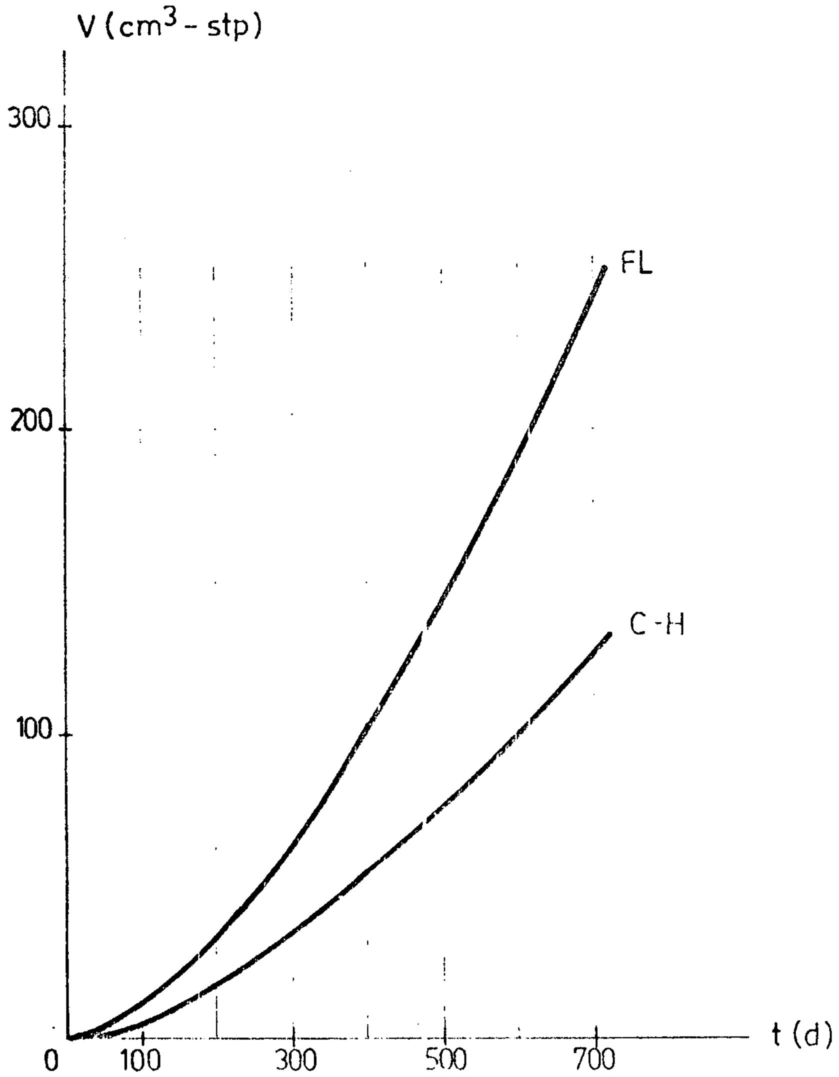

FIG.5: Time evolution of the volume V of fission gas released plus fission gas stored at the grain boundaries.

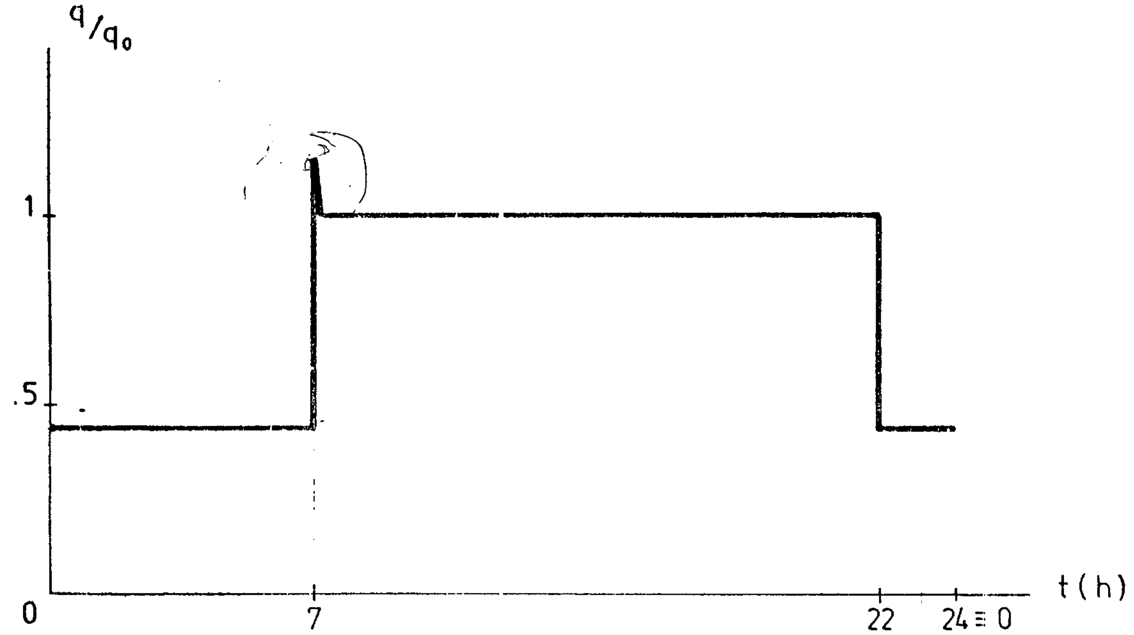

FIG.6: Assumed daily power cycle during load-following operation.

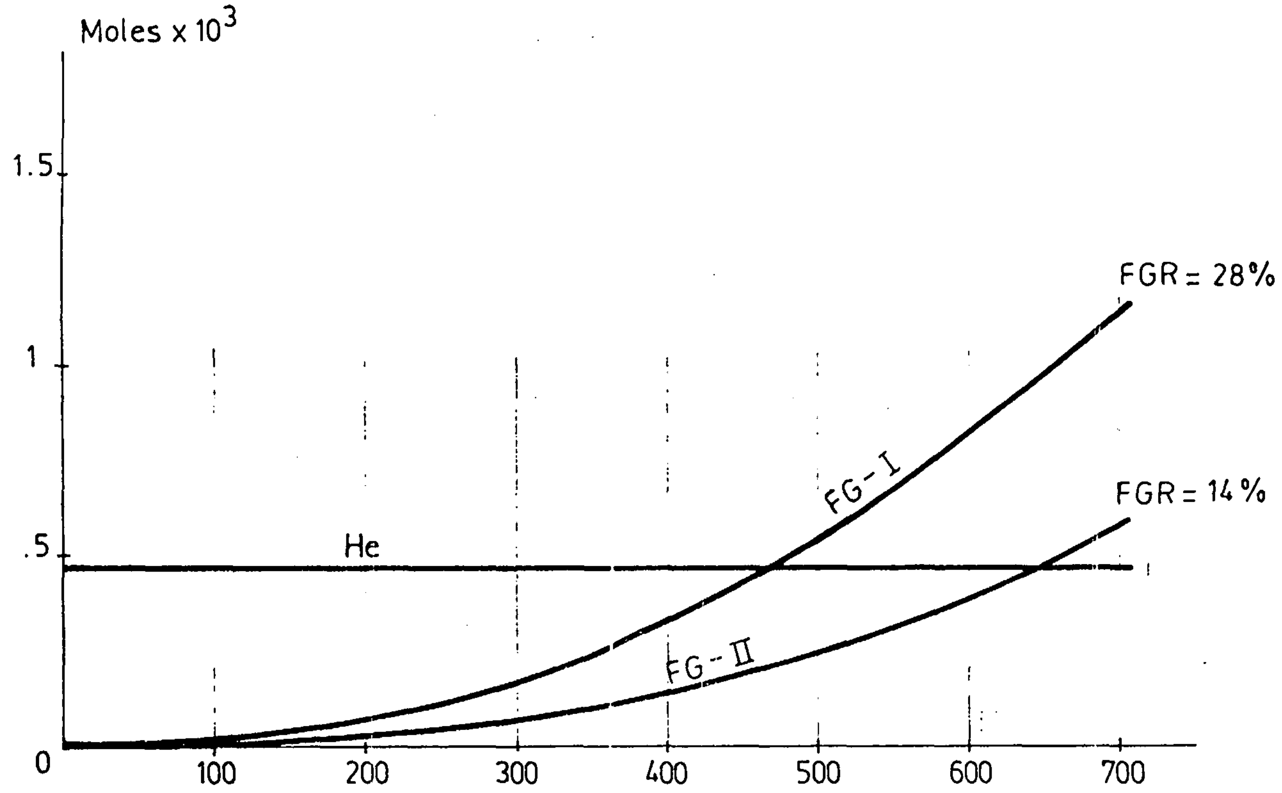

FIG. 7: Time evolution of the number of gas moles in the rod free volume. He refers to the filling helium; FG-I and FG-II to fission gases released according to Models I and II. FGR is the fraction of fission gas released at EOL.

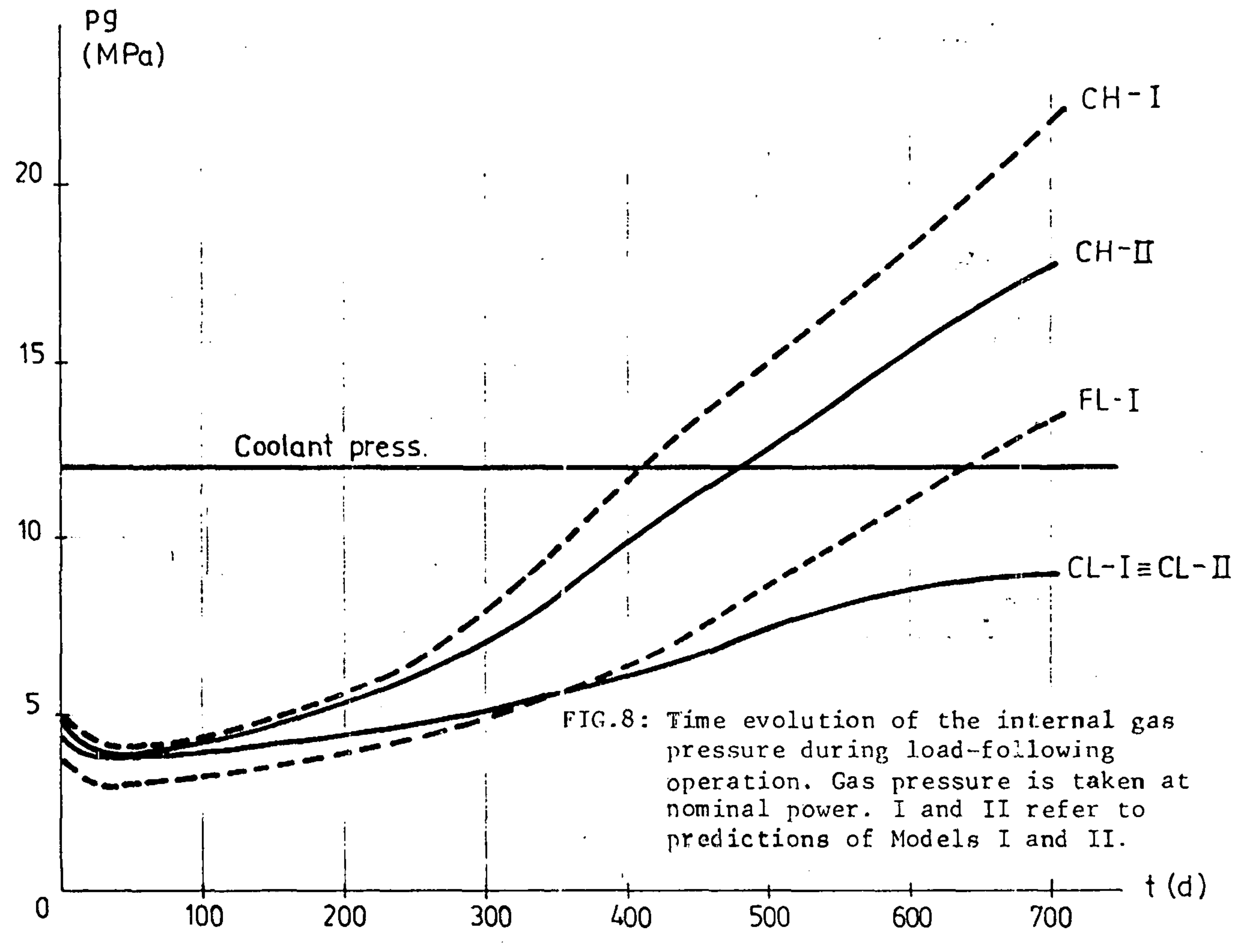

FIG.8: Time evolution of the internal gas pressure during load-following operation. Gas pressure is taken at nominal power. I and II refer to predictions of Models I and II.

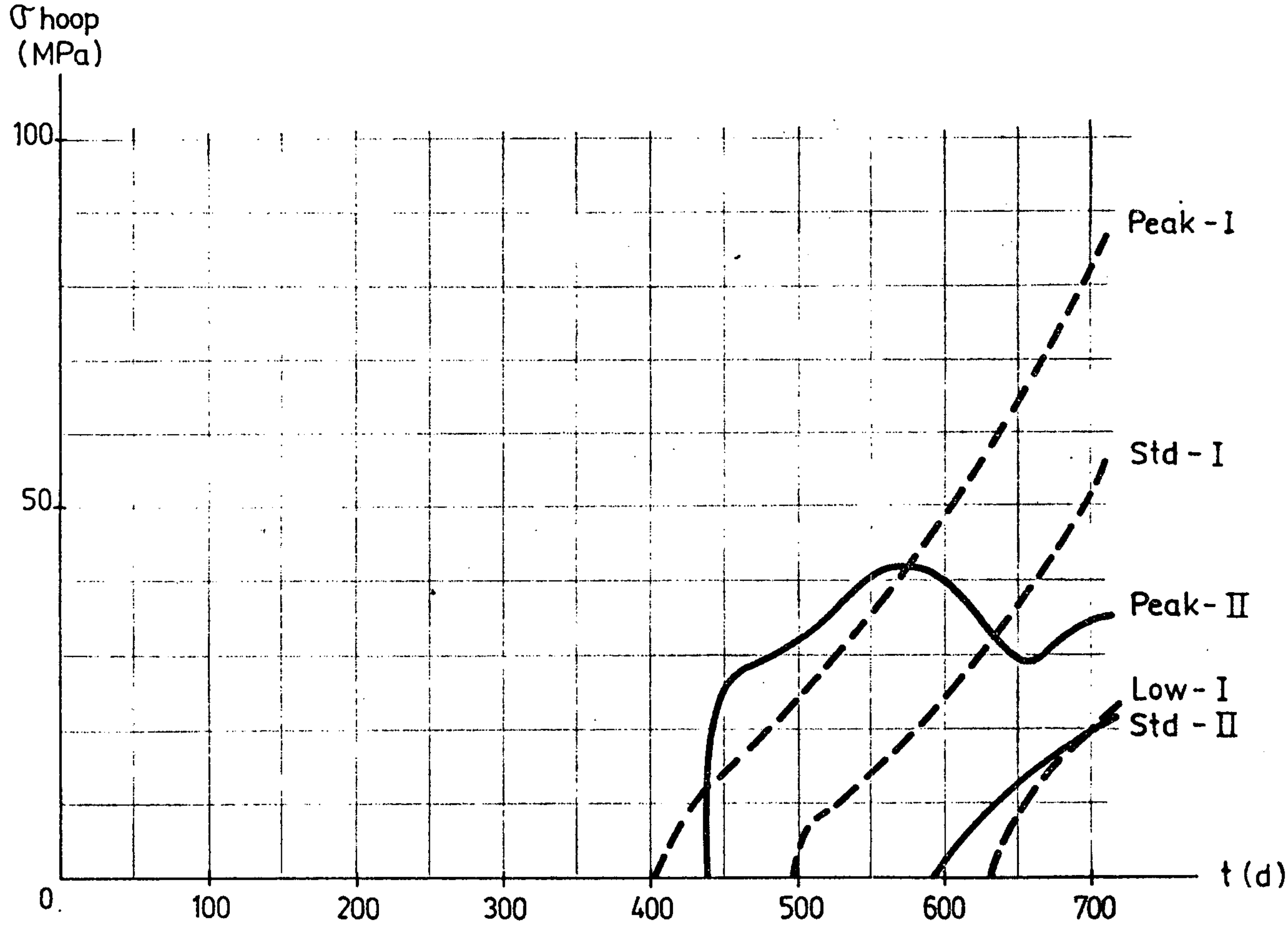

FIG. 9: Time evolution of tensile hoop stresses at the cladding inner surface during load-following operation. "Peak" correspond to the overpower at 7 a.m. (Fig. 6); "std" to nominal power, and "low" to low power during the night.