DEVELOPMENT ACTIVITIES ON ADVANCED LWR IN ARGENTINA

S.E. GÓMEZ
Comisión Nacional de Energía Atómica,
Argentina

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Abstract

CAREM, an Argentinean project, consists on the development, design and construction of a small Nuclear Power Plant. CAREM is an advance reactor conceived with new generation design solutions and standing on the large experience accumulated in the safe operation of Light Water Reactors in the world. The CAREM is an indirect cycle reactor with some distinctive features that greatly simplify the reactor and also contribute to a high level of safety: integrated primary cooling system, self-pressurized, primary cooling by natural circulation and safety system relying on passive features. In this paper a brief description of the CAREM distinctive features and associated development activities are presented.

1. INTRODUCTION

Argentine Nuclear Development started in early fifties. Initially the activities of the Comisión Nacional de Energía Atómica (CNEA) of Argentina were oriented to research in nuclear physics, radiochemical studies, material science among others subjects. In 1957, the CNEA decided to build a Research Reactor. The RA-1 was the first nuclear reactor to be put in service in South America. Since then, Argentina has designed and constructed several Research Reactors in Argentina and another countries, and at the present competes with foreign developed countries as supplier of this technology.

In 1964, CNEA initiated the feasibility study for the construction of Atucha I Nuclear Power Plant (CNA I) which would be the first nuclear power plant in Argentina and Latin America designed for electric power generation. In 1967 entrusted its design and construction to Siemens. The construction began in June 1968 and the commercial operation started in June 1974. The station contains a reactor of the pressure vessel type and it is heavy water moderated and cooled being of the PHWR type; it is periodically refueled on power. CNA I 's original design considered only natural uranium as fuel, being its electric power of 340 MW(e). The station suffered two essential modifications that improved its performance:

- In 1977 the electric power was increased to 357 MW(e).
- Since 1995 a progressive loading with slightly enriched uranium (0.85% wt) began, so that at present the core contains not only natural uranium fuel elements but also slightly enriched ones.

In 1967, CNEA initiated the feasibility study for the construction of Embalse Nuclear Power Plant (CNE) and in 1973 signed a contract with Atomic Energy of Canada Limited (AECL) and Societa Italiani Impianti P.A. (IT) for a 600 MW(e) CANDU–PHW (pressurized heavy water) type nuclear power plant. The construction of the station began in May 1974 and the commercial operation started in January 1984.

On the other hand, Argentina started the design of its own nuclear power plant, CAREM. The CAREM concept was first presented in March 1984 in Lima, Peru, during the IAEA conference on small and medium size reactor. CAREM design criteria or similar ones have since been adopted by other plant designers, thus originating a new generation of reactor design, of which the CAREM was, chronologically, one of the first. The Argentinean CAREM project consists on the development, design and construction of an advanced, simple and small Nuclear Power Plant (NPP) conceived with new generation design solutions and standing on the large world wide experience accumulated in the safe operation of Light Water Reactors. This project allows Argentina to sustain activities in the nuclear power plant design area, assuring the availability of updated technology in the mid-term. This implies working with technology acquired in Research Reactors design, construction and operation, and Pressurized Heavy Water Reactors (PHWR) Nuclear Power Plant operation as well as developing advanced design solutions.

2. FEATURES OF CAREM DESIGN

CAREM is an indirect cycle reactor (100 MWt, approximately 27 MW(e)) with some distinctive features that greatly simplify the reactor and also contribute to a high level of safety:

The CAREM NPP is a light water integrated reactor. The whole high-energy primary system and the absorbers rods drive mechanisms are contained inside a single pressure vessel.

The flow rate in the reactor primary systems is achieved by natural circulation. The driving force for the coolant's natural circulation is produced by the location of the steam generators above the core.

Self-pressurization of the primary system is the result of the liquid-steam equilibrium.

The main criteria used in the design of safety systems were simplicity, reliability, redundancy and passivity.

2.1. Integrated primary cooling system

The CAREM reactor pressure vessel (RPV) contains the core, steam generators, the whole primary coolant and the absorber rods drive mechanisms (figure 1). The RPV diameter is about 3.2 m and the overall length is about 11 m.

The core has 61 fuel elements of hexagonal cross section and about 1.4 m active length. Each fuel element contains 108 fuel rods, 18 guide thimbles and 1 instrumentation thimble. Its components are typical of the PWR fuel assemblies. The fuel is enriched UO_2. Core reactivity is controlled by the utilisation of Gd_2O_3 as burnuble poison in specific fuel rods and movable absorbing elements belonging to the Adjust and Control System. Chemical shim is not used for reactivity control during normal operation. Absorbing elements belonging to the Fast Extinction System are used to produce the sudden interruption of the nuclear chain reaction when required. Each neutron absorbing element is a cluster composed of a maximum of 18 individual Ag-In-Cd rods which are put together in a single unit. Each unit fits well into the fuel assembly guide tubes.

The primary coolant is light water that also acts as moderator. The large coolant inventory provides relatively smooth transients and large response time for recovery actions. Strong negative temperature coefficients are the consequence of no boron use for reactivity control during normal operation, allowing the control system to keep control of the reactor power through transients and load variations with minimum control rod motion.

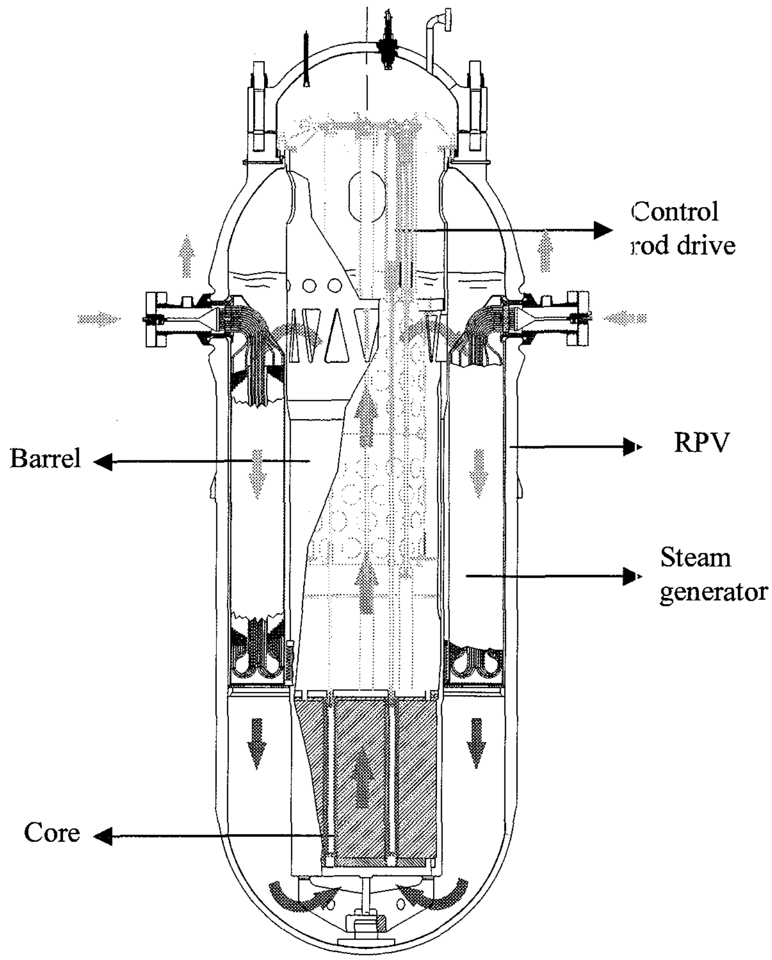


FIG. 1. CAREM primary circuit

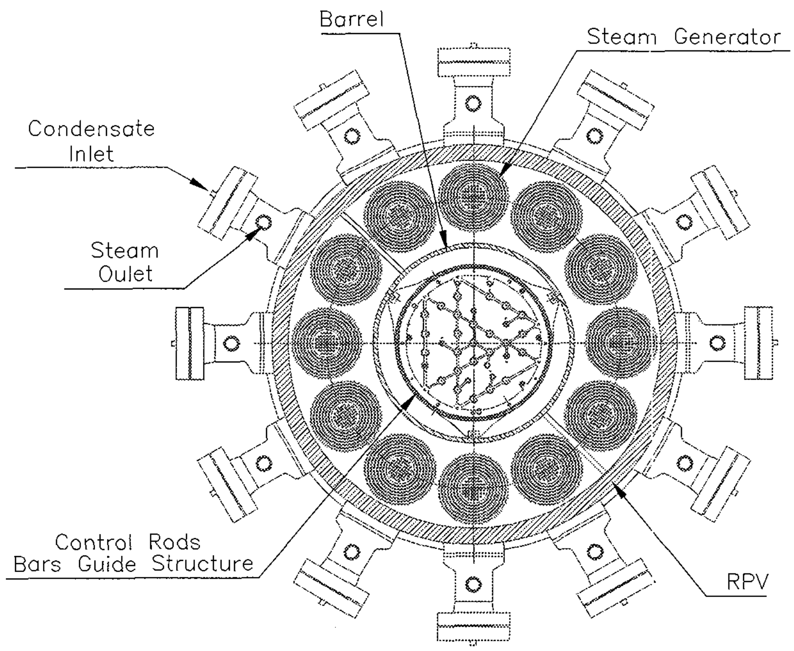


FIG. 2. Steam generators layout inside the RPV

The CAREM steam generators are twelve identical 'Mini-helical' of the "once-through" type placed equally distant from each other along the inner surface of the RPV area (figure 2). The secondary system circulates upwards within the tubes, while the primary does so in counter-current flow. An external shell surrounding the outer coil layer and adequate seals form the flow separation system, that guarantees that the entire stream of the primary system flows through the steam generators. In order to achieve a rather uniform pressure-loss and superheating on the secondary side, the number of tubes per coil layer is changed to equalize the length of all the tubes. Thus, the outer coil layers will hold a larger number of tubes than the inner ones. For safety reasons, steam generators are designed to withstand the pressure from the primary up to the steam outlet / water inlet valves even without pressure in the secondary.

The control rod drives (CRD) are of the hydraulic type. They are wholly contained in the RPV avoiding the use of mechanical shafts passing through the primary pressure boundary. Rods are kept in position by an external hydraulic circuit that pumps water to a lower chamber of a piston/cylinder assembly. Absorbing elements belonging to the Fast Extinction System are kept in the upper position during all normal operation and, at that position, the piston partially closes the outlet orifice and reduces the flow to a leakage. The CRD for this system is designed using a large gap between piston and cylinder in order to obtain a minimum dropping time. The inner wall of the cylinder and the outer wall of the piston of the CRD of the Adjust and Control System have grooves which produces changes in the pressure losses depending on their relative position. A steady flow keeps the piston in a fixed position and stepwise movement is achieved by applying pressure/flow pulses. It is designed to guarantee that each rod can be moved step by step so manufacturing and assemblies' allowances are stricter and clearances are narrower. In this case there is not a stringent requirement on dropping time. Both devices type, perform their SCRAM function by the same principle: "rod drops by gravity when flow is interrupted", so malfunction of any powered part of the hydraulic circuit will cause the immediate shutdown of the reactor.

2.2. Natural circulation

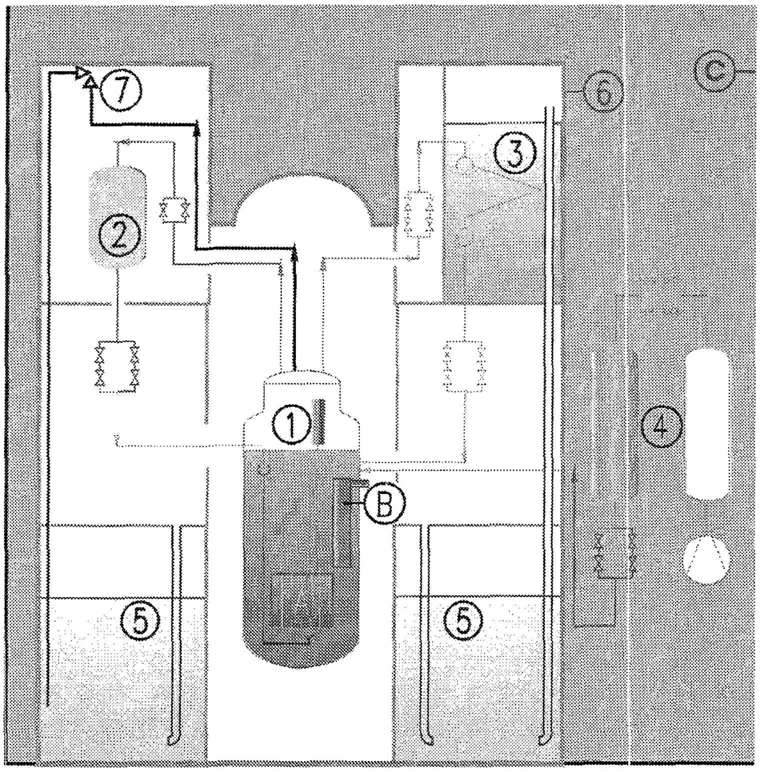
The flow rate in the reactor primary systems is achieved by natural circulation. Figure 1 shows a diagram of the natural circulation of the coolant in the primary system. Water enters the core from the lower plenum. After been heated, the coolant exits the core and flows up through the riser to the upper dome. In the upper part, water leaves the riser through lateral windows to the external region. Then it flows down through modular steam generators, decreasing its enthalpy. Finally, the coolant exits the steam generators and flows down through the down-comer to the lower plenum, closing the circuit. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form losses, producing a flow rate in the core that allows to have a sufficient thermal margin to critical phenomena. Core internals are designed to minimize the pressure drops. The natural circulation of the coolant produces different values of the flow rate in the primary system according to the power generated (and removed). Under different power transients a self-correcting response in the flow rate is obtained. CAREM has not primary pumps therefore there is not possibility of Loss of Flow Accident (LOFA).

2.3. Self pressurization

Self-pressurization of the primary system is the result of the liquid-steam equilibrium. Due to self-pressurization, bulk temperature at core outlet corresponds to saturation temperature at primary pressure. The steam dome pressure is very close to the saturation pressure, and at all the operating conditions this has proved to be sufficient to guarantee a remarkable stability of the RPV pressure response. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in case of power ramps. The large volume of the integral pressurizer also contributes to the damping of eventual pressure perturbations. Heaters and sprinkles typical of conventional PWRs are thus eliminated.

2.4. Passive safety systems

The main criteria used in the design of safety systems were simplicity, reliability, redundancy and passivity. Special emphasis has been put on minimizing the dependence on active components and operators' actions (figure 3).



SAFETY SYSTEMS

1- First shutdown system
2- Second shutdown system
3- Residual heat removal system
4- Emergency injection system
6- Containment
7- Relief valves

REFERENCES

A- Core
B- Steam generator
C-Secondary containment
5- Suppression pool

FIG. 3. CAREM safety systems

The First Shutdown System (FSS) is designed to shut down the core, when abnormal or deviated from normal situations occur, and to maintain the core sub-critical during all shutdown states. This function is achieved by dropping the neutron-absorbing elements into the core by the action of gravity when the water flow in the CRD mechanism is interrupted, so malfunction of any powered part of the hydraulic circuit will cause the immediate shutdown of the reactor. Six out of twenty-five absorbing elements are part of the Fast Extinction

System capable of shutdown the reactor immediately. The rest belongs to the Adjust and Control System capable to introduce enough negative reactivity to keep the reactor in shutdown mode, with appropriate safety margin, during all cooling conditions.
Gravity driven injection system of borated water at high pressure makes up the Second Shutdown System. It actuates automatically when the Reactor Protection System detects the failure of the First Shutdown System or in case of LOCA. The system consists of tanks connected to the reactor vessel by two piping lines which valves are opened automatically when the system is triggered. Then one of the pipes -from the steam dome to the upper part of the tank- equalizes pressures, and the other -from a position below the reactor water level to the lower part of the tank- discharges the borated water into the primary system by gravity.

The Residual Heat Removal System has been designed to reduce the pressure on the primary system and to remove the decay heat in case of loss of heat sink. It is a simple and reliable system that operates condensing steam from the primary system in emergency condensers. The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected via piping to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside of the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed, therefore the tube bundles are filled with condensation. When the system is triggered, the outlet valves open automatically. The water drains from the tubes and steam from the primary system enters the tube bundles and is condensed on the cold surface of the tubes. The condensation is returned to the reactor vessel forming a natural circulation circuit. In this way, heat is removed from the reactor coolant, and simultaneously water is supplied to the reactor vessel. During the condensation process the heat is transferred to the water of the pool by a boiling process. This evaporated water is then condensed in the pressure suppression pool of the containment.

The Emergency Injection System prevents core exposure in case of LOCA. The system consists of tanks with borated water connected to the RPV. In the event of such accident, the primary system is depressurized with the help of the emergency condensers and at low pressure the rupture disks break starting the RPV flooding with borated water.

Three safety relief valves protect the integrity of the reactor pressure vessel against overpressure, in case of strong unbalances between the generated and removed power.

The pressure-suppression type primary containment is a cylindrical concrete structure with an embedded steel liner type with two major compartments: a drywell and wetwell. The lower part of wetwell volume is filled with water that works as the condensation pool, and the upper part is a gas compression chamber. The blow-down pipes from the safety relief valves are routed to the pressure suppression pool.

3. RESEARCH AND DEVELOPMENT

The CAREM project involves technological and engineering solutions, as well as several innovative design features that have been properly proved during the design phase. Within CAREM project, the effort was focused mainly on the nuclear island (inside containment and safety systems) where several innovative design solutions require developments. This comprises mainly: the reactor core cooling system, the reactor core and fuel assembly, the reactor pressure vessel internals and the hydraulic control rod drive mechanisms.

The utilisation of specific codes (modelling tools) is required to obtain design parameters of some important systems (e.g. primary cooling system, reactor core, etc.). In some cases these codes must be developed and/or validated with experimental data to build confidence on their results.

A High Pressure Natural Circulation Loop (CAPCN) was constructed and operated to produce data in order to verify the thermal hydraulic tools used to design CAREM reactor. The CAPCN reproduces the dynamics phenomena of the CAREM primary cooling system, except for the three-dimensional effects. Dynamical experiment data are being used to test our numerical procedures and codes [1].

The neutronic and thermal-hydraulic behaviors of CAREM are strongly coupled, so to take this effect into account, the neutronic code CITVAP and the 3D two fluid model THERMIT code were linked. Benchmarck data available worldwide were used to validate the neutronic data, codes and modeling. Ad-hoc experiments, to generate a substantial database in the operate range and fuel geometry of the CAREM core, to develop a prediction methodology for critical heat flux, were performed at the Institute of Physics and Power Engineering (Obninsk, Russian Federation).

Hydrodynamic and structural test are being conducted to qualify the fuel assembly design.

The development of the hydraulic control drive for the Fast Extinction System and the Adjust and Control System comprises different stages. First, a series of test using prototypes to determine preliminary operating parameters were finished. After that, a series of test were and are being conducted in the Low Pressure CRD Rig (CEM) to characterize the mechanism and the hydraulic circuit used to control the mechanism movement. Now, a High Pressure CRD Rig (CAPEM) is being designed and it will be used to qualify the mechanism at the CAREM RPV operation conditions including some test under abnormal conditions and to test the drive mechanism instrumentation developed for determining the position of the absorber elements in the core.

Related to mechanical design (structural, dynamic, seismic, etc.) of the core and other reactor pressure vessel internals, different mock-up facilities are being constructed. Among others, the evaluation of manufacturing and assembly process for the "Mini-helical" steam generators is being done using mock-up. The design of the cinematic chain of the First Shutdown System is of particular interest. A dummy of the core, up to the extension of three fuel assemblies, core support and upper structure including control rod guides were used to study the behavior of the control rod movement (stepwise and rapid fall) under horizontal seismic load on a wide range of frequencies and magnitudes. One vertical full-scale model of the control rod drive structure with the absorbing elements and a dummy fuel assembly were used to study the static/dynamic friction loads of the cinematic chain for different insertion/extraction velocities of the absorbing elements and different misalignment of the structure.

4. CONCLUSIONS

The development activities on advanced LWR in Argentina are carried on in connection with CAREM, a reactor conceived with new generation design solutions and standing on the large experience accumulated in the safe operation of Light Water Reactors in the world.

Technical and economical advantages are obtained with the CAREM design compared to the traditional design. The large loss of coolant accident was eliminated due to the absence of large diameter piping associated to the primary system and the rod ejection accident was also eliminated due to the development of the innovative hydraulic mechanisms located completely inside the reactor pressure vessel. In addition, hydraulic control rod drives mechanism significantly cost down compared with the traditional PWR's control rod drive mechanisms. The large coolant inventory in the primary results in large thermal inertia and long response time in case of transients or severe accidents and the large water volume between the core and the wall leads to a very low fast neutron dose over the reactor pressure vessel wall. The elimination of chemical shim (borated water) during normal operation reduces the maintenance and occupational doses and the operational costs and results in strong negative temperature coefficients that simplify the operation control. The integration of the primary system inside a single pressure vessel reduces the shielding requirements, by the elimination of gamma sources of dispersed primary piping and parts and the elimination of primary circuit branch pipes systems reduces the primary circuit failure probability. The elimination of primary pumps and pressurizer results in lower costs, added safety, and advantages for maintenance and availability.

Improvements of the reliability of CAREM reactor come partly from the use of natural circulation in the primary circuit.

Since CAREM is a LWR conceived with new generation design solution, an extensive research and development program related with the innovative design solutions are been carried on to properly verify them during the design phase.

REFERENCE

[1] DELMASTRO, D.F., "Thermal-hydraulic aspects of CAREM reactor", IAEA TCM on Natural Circulation Data and Methods for Innovative Nuclear Power Plant Design, Vienna, 18-21 July 2000.