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ARGENTINA: SPECIFICATIONS OF ARGOS 380 MW REACTOR

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ARGENTINA: SPECIFICATIONS OF ARGOS 380 MW REACTOR

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COMISION NACIONAL DE ENERGIA ATOMICA
DEPENDIENTE DE LA PRESIDENCIA DE LA NACION

ARGOS PHWR 380

ARGENTINE OFFER OF A SAFER PRESSURIZED HEAVY-WATER REACTOR OF 380 MW

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RESUMEN

Se presenta el diseño de una central nuclear de 380 MWe, equipada con un reactor de recipiente de presión y moderador de agua pesada. Se explican los motivos técnicos, económicos y financieros que condujeron a la realización de este diseño de central nuclear de potencia intermedia, compatible con los últimos adelantos en el campo de la seguridad nuclear.

ABSTRACT

This paper introduces the design of a nuclear power plant of 380 MWe, equipped with a pressure-vessel heavy-water reactor. An explanation is given of the technical, economical and financial factors that led to the design of this medium-sized nuclear power plant, which is compatible with the latest advances in the field of nuclear energy.

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**ENACE

INTRODUCTION

Reactor vendors in most countries have had lean pickings for the past decade, and ordering seems unlikely to show much growth until the shock wave from the Chernobyl accident has died away. Paradoxically, however, at least one firm sees a niche in the market.

ENACE - the Empresa Nuclear Argentina de Centrales Eléctricas, or Argentine Nuclear Power Plant Corporation - is stepping out into the market place with a newly-designed 380 MWe nuclear power plant. In its current version, the plant is equipped with a pressurized heavy-water reactor of the pressure vessel type (PHWR), but ENACE would also be prepared to configure it to use a pressure-tube reactor (PTHWR) design.

ENACE has adopted new boundary design conditions and has embodied a number of special features to assure safety and economy in operation.

The major shareholder in ENACE is the Argentine National Atomic Energy Commission (CNEA). ENACE is the architect-engineer for the NPP projects of the Argentine nuclear programme. It has a licensing agreement with Siemens AG's Kraftwerk Union AG, which is its minor shareholder. Under this agreement, ENACE has the right to use the Siemens-KWU PHWR technology, which was originally developed for the MZFR reactor in the Federal Republic of Germany [1], as well as their know-how in pressurized (light-) water reactors (PWRs) design and construction. The CNEA also has agreements with Atomic Energy of Canada Ltd. for the transfer of technology related to CANDU-type PTHWRs. The CNEA and ENACE have acquired considerable practical experience from the construction and operation of the 367 MWe Atucha I PHWR [2] and the 648 MWe Embalse PTHWR; ENACE is currently building Argentina's third plant, Atucha II, a 745 MWe PHWR. [3]

Throughout the development of its nuclear programme, Argentina has aimed for independence, using its own uranium and fuel cycle technology. Its choice of heavy water reactors (HWRs) was dictated, at least in part, by the fact that it enables the use of natural uranium fuel. However, the new design is configured to accept as well an optimized low-enriched fuel cycle, and fuel spiked with self generated plutonium or even spiked with thorium.

Flexibility is thus an important advantage of the new design. Couple that with the fact that in many countries - even in some which have already embarked on nuclear programmes - the grid is too small to accept the "big is better" 1000 MWe units which are commonplace in industrialized countries [4], and the Argentine offer begins to make sense.

WHAT IS ARGOS PHWR 380? ORIGIN

ENACE is offering a newly designed 380 MWe nuclear power plant (NPP) equipped with a heavy-water reactor. Plants with reactors of this type have been in commercial operation in Argentina since 1974. The new plant, named "ARGOS PHWR 380" (an acronym for "Argentine Offer of a Safer Pressurized Heavy-Water Reactor of 380 MWe"), is based on a heavy-water reactor of the pressure-vessel type (PHWR), although conceptually it could also house one of the pressure-tube type (PTHWR). It is medium sized (1123 MWth or 380 MWe) and reflects advanced boundary design conditions. It has a number of special features to assure its safety and economy in operation.

ENACE is an acronym for Empresa Nuclear Argentina de Centrales Eléctricas, or Argentine Nuclear Power Plant Corporation. Its major shareholder is the Argentine National Atomic Energy Commission (CNEA). ENACE is the architect-engineer for the NPP projects in the Argentine nuclear programme. It has a licensing agreement with Siemens AG's Kraftwerk Union AG, which is its minor shareholder. Under this agreement, ENACE has the right to use the Siemens-KWU PHWR technology, which was originally developed for the MZFR reactor in the Federal Republic of Germany [1], as well as their know how in the design and construction of pressurized (light-) water reactors (PWRs). The CNEA also has agreements with Atomic Energy of Canada Ltd. for the transfer of technology related to CANDU-type PTHWRs.

ENACE, with CNEA's support, can thus provide a wide spectrum of technical know-how gained through experience in engineering, constructing, commissioning and operating heavy-water reactors. ENACE has also considerable experience in the installation of nuclear industrial facilities in Argentina and is well prepared to participate in nuclear projects in other countries.

In connection with a request from the Argentine Government [2], CNEA asked ENACE to prepare technical specifications for a medium-sized power reactor, drawing on Argentine experience in operating heavy-water reactors. This request was the genesis of ARGOS PHWR 380.

Why a heavy-water reactor?

In the sixties, Argentina a country with reasonable reserves of uranium one-took the basic policy decision to develop an independent nuclear programme to meet a part of its domestic electricity demand. Argentina also decided to develop the industrial infrastructure necessary to permit the local supply of NPP software and hardware. Argentina's decision to choose a heavy water reactor fuelled with natural uranium as the basis for its nuclear power programme reflected this policy.

The Argentine programme got under way with the ordering of the 367 MW Atucha I PHWR NPP (CNA-I) [3] in 1967. The CNEA has operated this plant successfully since 1974. This plant was followed by the 648 MW PHWR Embalse NPP, which has been in operation since 1982. ENACE is currently building Argentina's third NPP, Atucha II (CNA II) [4], with another PHWR that will generate 745 MW.

Argentina's experience in NPP operation has confirmed the safety, reliability and availability of heavy water reactors, and their efficiency in fuel utilization. It has also confirmed their main advantage: the fact that they allow countries with limited economic or technological resources to attain fuel cycle independence.

Why a medium-sized NPP?

There has been a general trend towards the development of large NPPs (1400 MWe and even more). However, during the last few years, considerable attention has been given to the prospects for the so-called "small and medium power reactors" (SMPRs). The International Atomic Energy Agency (IAEA) has surveyed both the suppliers and the potential market for such reactors [5].

ENACE's design of a 380 MW NPP is intended to fill a gap in the range of NPP designs available for use in Argentina. It will permit greater flexibility in future decisions on the Argentine NPP programme, taking into account the country's economic and financial situation and its power demand.

The most important advantages of SMPRs are:

- Lower financial impact: Large NPP projects may often suffer significant delays because of the scarce of financial resources. The consequent financial burden can offset or even outweigh the economic advantages derived from the lower cost per unit power of large units. Tight time schedules and lower financial costs are, therefore, advantages of SMPRs.
- Better promotion of domestic industry: Smaller NPP modules involve the fabrication of smaller equipment that can be manufactured in series. This is highly important when decision makers wish to promote the creation or the continuity of a domestic nuclear industry. A national decision to embark on a programme involving a larger number of smaller NPPs will certainly be advantageous, because it will enhance nuclear industrial development and production capacity.
- Lower indirect costs: Design and engineering expenses -two large contributors to NPP indirect costs- are less significant when spread over a larger number of small units.
- Easier adaptation to electrical distribution networks: Smaller NPPs can be more easily tailored to grids of limited capacity. This is a particular advantage in many developing countries. Moreover, SMPRs -adequately sited near load centres- can easily become economically competitive with some other sources of power supply.
- Easier acceptance: It seems that the public accepts, subjectively, smaller units more readily than nuclear "giants". There may be no good objective reasons for this, but it may become an important factor to be taken into account in the choice of an NPP module.

Why new boundary design conditions?

NPPs must now be designed taking into account various factors that were not sufficiently highlighted in the early days of the nuclear industry, such as:

- Nuclear safety: The accidents in Three Mile Island and Chernobyl have induced a severe crisis of credibility with respect to NPP safety, and have called into question many safety dogmas, such as that of the "maximum credible accident".
- Installation and operational economy: Ever-increasing investment and operational costs have made potential customers reluctant to embark on the use of nuclear energy.
- Utilization of energy reserves: Since fast breeder reactors will not be available as soon as it was originally expected, optimizing the use of indigenous uranium may become a critical issue in the medium term.

All these factors were taken into account during the design of ARGOS PHWR 380.

- As far as safety is concerned:

- . It fully complies with the demanding nuclear safety requirements of the Argentine national regulatory authority and -automatically- with all relevant international safety standards, guides and recommendations -notably, those issued by the International Atomic Energy Agency (IAEA).
- . It follows the most recent developments in the field of probabilistic safety assessment and complies with quantitative probabilistic safety criteria, as well as with deterministic criteria.
- . It reflects lessons learned in the nuclear safety area, particularly in relation to emergency power supply, leak-before-break monitoring, heat removal and containment integrity.
- . Even for the most unlikely accidents involving core meltdown, measures are provided to mitigate the radiological consequences to reasonable levels.

- As far as installation and operational economy is concerned:

- . The application of lessons learned in the engineering of the widely used PWR -which are fully applicable to PHWRs- in the design of ARGOS PHWR 380 ensure its economical optimization. Most of the systems and components are identical or similar to those in the CNA I and CNA II and to those used in many PWRs that are currently being built or in operation worldwide. Several features have been introduced in the design to limit the investment costs to a reasonable level.
- . Given the excellent operating performance achieved by CNA I, ARGOS PHWR 380 is comparable with "the" perfect reference plant and fulfills the important criterion of proven design set up by IAEA in its SMPR study [5].
- . Subsequently, simplicity in system design, the use of proven components and the existence of a good reference plant do also support the expectation of good operational performance.
- . Load flexibility, relatively low power and a self-powered start-up make the reactor suitable for use in grids of limited capacity -such as those usually found in developing countries- as well as a good replacement for obsolete fossil fuel plants.
- . ARGOS PHWR 380 can efficiently generate by-products -such as the radioisotope cobalt-60- without jeopardizing energy availability.

- As far as the utilization of energy reserves is concerned:

- . The natural uranium fuel cycle not only lends itself to support of an independent nuclear programme -even in countries suffering from technical or economic constraints- but ensures an optimum utilization of uranium reserves.
- . ARGOS PHWR 380 does also offer several fuel cycle alternatives that allow for further optimization of uranium consumption.

HOW IS ARGOS PHWR 380?

General description

ARGOS PHWR 380 uses a PHWR as a nuclear steam supply system (NSSS). The simplified flow diagram of the plant is shown in Figure 1. The design bears close resemblance to standard pressurized (light) water reactors (PWRs). The NSSS includes a reactor vessel and a reactor coolant system. The moderator system - a characteristic feature of HWRS - includes a moderator tank (similar to a calandria tank) within the pressure vessel and a moderator cooling system. The reactor is on-load refuelled and can be fuelled not only with natural uranium in its basic fuel cycle version but also with fuel elements resulting from several advanced fuel cycles. The balance of plant includes a reactor building, an auxiliary building and conventional buildings. The reactor, the reactor coolant and moderator systems, the refuelling equipment and the spent fuel pool are housed inside the reactor building. The main technical data of ARGOS PHWR 380 are presented in Table 1, together with data from CNA I and CNA II.

The Nuclear Steam Supply System

- Coolant and moderator systems

The components of the reactor coolant system of ARGOS PHWR 380 are fully comparable in design and arrangement with those in similar PWRs (see Figure 2): the reactor pressure vessel (RPV) is positioned vertically in the center of the system; two identical coolant loops, each comprising one coolant pump and one steam generator, are connected to the RPV radially. Additionally, there is the moderator system with three loops flowing the moderator from the moderator tank to the moderator coolers through moderator pumps.

The reactor coolant is heavy water (D_2O) flowing upwards by the coolant channels and driven through the steam generators by the main coolant pumps. The moderator is also D_2O . Under normal operating conditions, the moderator flow inside the moderator tank is directed upwards and driven through the moderator coolers by the moderator pumps. The moderator system has also an important safety function (see description under "Why is ARGOS PHWR 380 safer?").

There is only a small difference between the pressures of coolant and moderator. Coolant channels need therefore be only thin-walled. As the coolant and moderator systems are interconnected, common auxiliary systems can be used to maintain the necessary water quality. The number of auxiliary systems is thus reduced to a minimum. The heat generated in the moderator by neutron moderation and heat transfer is equivalent to approximately 10% of the total energy, and is used to pre-heat the feedwater, thus enhancing the efficiency of the plant.

- Reactor Core

The design of the ARGOS PHWR 380 pressure vessel and core is similar to that of the Atucha reactor type (see Figures 3 and 4). The moderator tank contains 244 vertical fuel channels producing 1067 MW of thermal power and arranged in a triangular lattice array with a pitch of 27.5 cm. The fuel assemblies (one per fuel channel) are identical to those used in the Atucha design (37 pins with an active length of 530 cm).

Hydraulically-actuated control and shutdown absorber rods are vertically inserted from the upper part of the vessel through the moderator tank. A total of 60 rods is used. Forty-five of them are shutdown rods allowing for quick shutdown and maintenance of safe long-term subcriticality, and the rest are power and power density control rods. A suitable selection of these vertical control rods provides flattening of the power distribution from the very beginning of the reactor operation, allowing the reactor to be operated at 100% rated power with a nearly fresh core. This system provides outstanding load follow-up capability of the reactor.

- Refuelling

An operational advantage of the ARGOS PHWR 380 is that it is designed for automated on-load refuelling (see Figure 5). There is a single refuelling machine. The fuel assembly transport system is located within the reactor building and includes a fuel pool with a capacity which can be tailored up to the design life time of the plant. The refuelling procedure is monitored from the control room. A special feature is the possibility of inserting irradiated fuel assemblies into the core in the same way as fresh ones. This may include used fuel from other NPPs.

ARGOS PHWR 380 is also an efficient generator of by-products. The fuel transport system can be used during full reactor power operation to remove specially designed fuel assemblies containing rods for cobalt-60 generation by irradiation. A production of $5.55 \cdot 10^{16}$ Bq ($1.5 \cdot 10^6$ Ci) of cobalt-60 per full power year can be obtained, without jeopardizing energy availability.

The balance of plant

- Plant layout

Some features determining the arrangement of the ARGOS PHWR 380 NPP buildings are: safety-related systems and components located in specific buildings protected against all relevant external events; clear definition of the controlled area; short piping and cable runs; easy access for construction to obtain a minimum construction schedule; a turbine building oriented so as to avoid the risk of turbine missiles to other buildings; and two physically separated cooling water intake structures protected against external events for the secured service cooling water system (see Figure 7).

- The reactor building

The reactor building (see Figure 8) consists of two concentric spherical structures. Its main purpose is to contain radioactive materials that, otherwise, would -in hypothetical accidental situations- be released into the environment. The internal sphere is metallic and has been designed to withstand the maximum pressure which might result from any conceivable loss of coolant or moderator accident. The spherical form was chosen as optimal, taking into account stress distribution parameters.

The steel sphere is -in turn- surrounded by an external spherical structure of concrete, which is intended to act as a secondary containment and as a radiation shielding, as well as to resist all relevant external events which might damage the inner containment. The space between the metallic and the concrete structures forms the reactor building annulus, where -in its lower part- the most important auxiliary systems are placed. Thus, they are located close to the primary system and to the reactor itself, and well protected against external events.

The reactor building houses not only the nuclear steam supply system but also the refuelling equipment and a 12-year-capacity fuel storage pool. The pool could be easily enlarged to meet users' requirements, up to the whole lifetime of the plant. Compartments inside the containment are classified in two groups, taking into account the radiation levels expected during normal operation: equipment and operating compartments. The latter are accessible without restriction during normal plant operation.

- The reactor auxiliary building

Other systems containing radioactive materials are housed in the reactor auxiliary building. They include the coolant and moderator cleaning and upgrading systems, the nuclear ventilation system and all the equipment necessary for handling liquid and gaseous radioactive wastes. The building is subdivided into D₂O and H₂O areas, according to the types of systems. The relevant areas of the reactor auxiliary building, as well as the reactor building, conform the "controlled area" from the radiation protection point of view.

- Other buildings

Since all the other buildings form the so-called conventional part, no special requirements were needed for their design, except those mentioned in the description of the plant layout. However, it is worth mentioning that the switch-gear building -reflecting the importance to safety of the equipment it houses-is protected against external events and internally subdivided into three redundant modules.

HOW CAN ARGOS PHWR 380 BE FUELLED?

- Efficient fuel-cycle options

The natural uranium consumption of HWRs is around 20% lower than that of standard PWRs with a high burnup cycle. However, there is also the feasibility to optimize the fuel consumption further and, thus, to raise the average discharge burnup, by using advanced fuel cycles.

The ARGOS PHWR 380 is prepared to accept a fuel of advanced design without significant system or component changes. Several different options have been investigated [6]. Two important alternatives to the use of pure natural uranium are the optimized low-enriched uranium (LEU) fuel cycle and the plutonium-spiking fuel cycle.

- The natural-uranium fuel cycle

The use of D_2O both as a coolant and as a moderator allows for a neutron balance that makes it feasible to use natural uranium as a fuel. To attain a high discharge burnup in a natural uranium cycle, it is necessary to make a suitable selection of a radial shuffling scheme compatible with the radial power profile. Using natural uranium fuel, a fuel consumption of 0.99 fuel assemblies per full power day (corresponding to a 6000 MWd/MgU discharge burnup) is compatible with a load cycle of 100-80-100% of rated power. In the equilibrium burnup core condition, the fuel loading scheme is 3 zone-1 way: that is, fresh fuel is introduced into a channel at an intermediate radial position. From there, partially depleted fuel is shuffled to the core centre and finally to the reactor periphery, from where the fuel assemblies with the highest burnup are discharged. By adopting this radial shuffling scheme and by flattening the radial power distribution, an average discharge burnup of 6600 MWd/MgU can be obtained in case of minimum reactivity reserve (see Fig. 6).

- The low-enriched-uranium cycle

A small increase of the U-235 content in the ARGOS PHWR 380 fuel, from the natural 0.71% to 1.0% - turning the core into a LEU core - reduces the natural uranium requirement by a factor of 1.6. Only a small amount of separative work is needed to support such cycle.

Considerable experience in the use of LEU fuel in PHWRs has shown that no changes in the thermal design or fuel assembly design are to be expected in the transition from the natural uranium to the LEU cycle without power reduction [7]. The converse transition -that is, from homogeneous LEU fuel to the natural uranium cycle- is also possible at any time. A smooth core transition without power reduction is a result of the design of the ARGOS PHWR 380 fuel assembly-- there is no additional power peaking in axial direction because a full length column is used--and of the further flattening of the radial power distribution allowed by the natural uranium core."

The insertion of LEU assemblies yields a greater channel power jump than that of natural uranium assemblies, which require a modified fuel loading scheme, involving fresh fuel insertion at the reactor periphery and more radial shuffling operations of the fuel elements in the core.

- The fuel cycle with plutonium spiking

Another way to achieve a more efficient fuel cycle is using, as a fuel, natural uranium mixed with plutonium bred in the reactor. The operation of ARGOS PHWR 380 will generate approximately 160 kg of fissile plutonium per year; it would nearly double the plutonium production rate of light water reactors at the same power level. The use of that fissile material may double the effective worth of reserves of natural uranium.

There are at least two methods for recycling the plutonium built up in the reactor: by homogeneous distribution of plutonium over all fuel assemblies and by plutonium enrichment of some fuel assemblies only -the so-called "spiked elements" option. The first alternative, although feasible, is not recommended because the over-cost of plutonium fuel fabrication is, for the time being, high and may cancel the economical advantages gained by the increased burnup. The se-

cond alternative offers a potential advantage, as far as the number of spiked assemblies does not exceed a reasonable limit.

The insertion of plutonium spiked assemblies in the reactor increases burnup not only in these assemblies but also in the surrounding natural uranium assemblies -because of the great reactivity increase provided by the spiked assemblies- and increases the thermal power in the channels where they are inserted. The amount of plutonium inserted in the spike is, therefore, as large as possible, within thermal-hydraulic and technological limits. The plutonium spiked assemblies are introduced into the outer core region, where the thermal-hydraulic margin is high. Once they have accumulated a burnup increment of about 6000 MWd/MgU, they are moved to the inner zones of the reactor. From there, after some radial shuffling operations, they are removed with a burnup of more than 25000 MWd/MgU. In this way, more radial flattening of power distribution can be obtained than in the natural uranium case. No change in axial power distribution, as compared to the natural uranium case, is observed.

- Conclusion on fuel cycle alternatives and additional options

Table 2 summarizes the main data and results of three fuel cycle alternatives. The first one is the standard natural uranium cycle, which is taken as a reference case. The second one refers to a 1% homogeneous enriched uranium cycle, while the third one is a case of plutonium-spiked assemblies using 'self generated plutonium'; in this last cycle, 20% of the coolant channels contain plutonium elements with an average enrichment of 2.1%."

Conceptually, the ARGOS PHWR 380 design offers other potential means of improving fuel utilization, such as: (i) reuse of fuel (fuel rods or fuel assemblies) irradiated in pressurized water reactors -the so called "tandem solution"; and, (ii) insertion of thorium mixed with some fissile materials in a few fuel assemblies -the so called "Th-spiking".

WHY IS ARGOS PHWR 380 SAFER?

The basic safety criteria applied to the design of the ARGOS PHWR 380 are the following:

- * It should ensure normal operation within the internationally recommended system of dose limitation [8] [9], which has been implemented by the Argentine regulatory authority [10] [11]; particularly, it should comply with the requirement of optimization of radiation protection.
- * It should follow -as a necessary but not sufficient safety condition- all relevant international safety standards, guides and recommendations and, in particular, issued by the International Atomic Energy Agency [12].
- * It should comply with the demanding Argentine nuclear safety requirements, which are based on quantitative probabilistic safety criteria [13]. In particular, it should comply -as a necessary safety condition- with a risk limit line or criterion curve (see Figure 9) [14], following specific probabilistic regulations on failure analysis (see Table 3) [15]. Besides, through probabilistic safety assessment and decision aiding techniques, it should also ensure that all risks to people be kept as low as reasonably achievable, well below the risk limit line. Additionally, it should comply with deterministic requirements including: reactor core design, residual heat removal systems, primary boundary, fuel behaviour, protection and instrumentation systems, shutdown systems, containment system, main electrical supply and quality assurance.
- * It should reflect the lessons learned in the nuclear safety area. In particular, the power supply to safety systems should be highly reliable, sensitive early detection of coolant and moderator leakages had to be ensured, containment integrity should be assured under any conceivable circumstances and the radiological consequences of severe accident sequences -even those leading to core melt-down- should be mitigated.

Compliance with these criteria, coupled with the well-known intrinsic safety advantages of HWRs, makes ARGOS PHWR 380 a safer NPP option built on the basis of existing experience in nuclear engineering and industry.

Design features for normal operation

The basic design criterion for normal operation of the ARGOS PHWR 380 is that it should comply strictly with the system of dose limitation recommended by the International Commission on Radiological Protection (ICRP) [8] and adopted -inter alia- by the International Atomic Energy Agency, the World Health Organization, the International Labour Organisation and the Nuclear Energy Agency of the OECD [9]. The Argentine regulatory authority has implemented the ICPR system by issuing regulations on occupational exposure [10] and on limitation of releases [11]. The system's requirement for the optimization of radiation protection -which is still being implemented by the nuclear industry -has been applied to the design of ARGOS PHWR 380.

Radiation protection measures for the ARGOS PHWR 380 in normal operation are optimized to keep doses as low as reasonably achievable (the ALARA principle). Optimization is achieved by using internationally recommended decision-aiding techniques (such as cost-benefit analysis)[16]. Optimization is carried out under the constraint that the design should ensure -under any relevant circumstances- that the annual dose to exposed individuals be lower than the internationally recommended dose limits. During the design process, protection by technical means was preferred to that achievable by operational procedures.

The following design limits (note: limits, rather than objectives or goals) have been, in every case, respected:

- For occupational exposures

- . Access to any area where the dose index may exceed 0.5 mSv/h is prevented by physical barriers.
- . In areas without restrictions, the concentration in air of radionuclides must be lower than 0.1 of the Derived Air Concentration (DAC) (or the concentration of radioactive materials in air which would expose workers to the recommended limits).
- . No individual may be exposed to concentrations higher than 0.1 DAC and, in any room where this concentration may exist, specific protection devices are provided.

- . Maintenance and in-service inspection can only be performed at dose rates lower than 0.5 mSv/h.

- . Normal repair can be performed with dose rates lower than 3 mSv/h, while infrequent repairs can be performed with dose rates lower than 12 mSv/h.

- For exposure of members of the public

- . Design is constrained by a limiting annual dose in the critical group of 0.3 mSv. The collective dose commitment must not exceed 0.015 man Sv per megawatt year of electrical energy generated.

- . Effluents can be discharged to the atmosphere only through the stack. Provision is made for continuous monitoring of the discharge of radioactive effluents into the environment in accordance with international recommendations.

Adherence to such limiting conditions implies, in practice, that ARGOS PHWR 380 includes particular radiation protection systems that are not commonly found. For instance, the design incorporates -inter alia- systems for the retention of Carbon-14 and for on-line monitoring of tritium in the environment.

Design criteria for potential accidental situations

Probabilistic safety criteria

A unique feature of ARGOS PHWR 380 is that its safety design is mainly based on probabilistic safety criteria (PSC), which is a requirement from the Argentine regulatory authority [13]. A priori probabilistic safety analyses (PSA) were carried out at the design stage and their results fed back into the reactor design taking into account the most recent reliability figures and design criteria. Such analyses are not purely theoretical: they are substantiated by the experience gained with the CNA II safety design, which was also based on PSA [17]. Moreover, the PSA results are checked against quantitative PSC issued by the Argentine regulatory authority, which are in line with the most recent international developments in PSC [18],[19].

These requirements were used as an a priori comprehensive condition for design, rather than as an a posteriori confirmation of design compliance.

The limiting criterion is that an annual risk upper bound of one-in-a-million must be respected for any individual who might hypothetically be subject to accidental exposures from a NPP. This criterion is consistent with the philosophy of dose limitation for exposures assumed to occur with certainty. Since accidental exposures may result from several accident sequences, and it is difficult to be sure that all such sequences have been identified, about ten relevant sequences are being identified and an annual risk upper bound of one-in-ten-millions assigned to each. As each sequence may result in different doses, a criterion curve or limit line was used for the ARGOS PHWR 380 design: this is a relationship between the annual probability of sequence occurrence and the expected individual dose, each point of the curve representing a constant level of annual risk equal to one-in-ten-millions. The criterion curve is shown in Figure 9 [14]. The ARGOS PHWR 380 design also complies with applicable regulations on failure analysis for PSA [15] (see Table 3). The CNA II design also complies with this limiting criterion [17].

However, the fact that ARGOS PHWR 380 complies with these individual-related PSC is a necessary but not sufficient condition for its safe design. Not even this advanced methodology for individual risk limitation is considered as enough to ensure a "safe" design from a probabilistic point of view. In fact, ensuring that no single individual will incur an unduly high probability of harm because of potential radiation exposure is not sufficient to ensure the appropriateness of the safety measures. They may need to be improved by taking into account, for instance, that from a high number of individuals incurring an acceptably low probability of harm may still result in an unacceptable high expectation of harm.

The ARGOS PHWR 380 design, however, reflects the fact that, for exposures which have a very low probability of occurrence, the use of the concept of expectation to reduce risks further, below the limits, is not straightforward [20]. The problem of comparing two or more engineering options, then, reduces to that of comparing different (mathematical) distributions of individual risks. There is also the additional problem of how to include, in the comparison process, quantities or preferences which may not be translated into commensurate units. Such preferences, which can be explicitly accounted for in the ARGOS PHWR 380 design pro-

cess, may include: the degree of risk aversion for higher-consequence accidents, social costs for restrictions or inconveniences, and the degree of relative importance of the various possible manifestations of radiation health effects.

The problem in comparing these quantities, which are not directly and linearly comparable, can be solved in the process for optimizing the ARGOS PHWR 380 design by using utility functions and decision theory [21]. Preferences for quantities of differing types are expressed using a utility function which prescribes how the different types of quantities are to be combined for the purposes of comparison. The resulting utility functions are then processed by a decision mechanism so as to arrive at a "best under the circumstances" (i.e., optimized) safety option.

The result of this process is a design for the ARGOS PHWR 380 that can be considered as safe-as-is-reasonably-achievable. Again, it is more than a theoretical exercise: should ARGOS PHWR 380 have been designed with two moderator loops, it would have complied with the criterion curve of the individual-related PSC; however, a third loop has been added following a decision-making process aimed at safety optimization. Nevertheless, the optimization process will only come to an end when the full spectrum of site-related information is available.

Deterministic safety criteria

ARGOS PHWR 380 also complies with all the deterministic criteria supplementing the probabilistic requirements described above, including the following:

- . Compliance with deterministic requirements from the Argentine authority

These requirements include the following main features: general safety criteria in the design [22] (minimizing the consequences of any eventual failure); reactor core design [23] (ensuring safe operation during the whole lifetime of the reactor); residual heat removal systems [24] (ensuring that fuel elements shall not suffer damage); pressurized primary boundary [25] (preserving integrity under any operational, testing or failure conditions); fuel behaviour in the reactor [26] (minimizing the possibilities of activity releases); safety-related protection and instrumentation system [27] (considering all the tentative situations under operation and failure conditions); shutdown system [28] (ensuring

reliable shutdown, under any conceivable conditions); containment system [29] (ensuring a proper activity confinement function); main electrical supply [30] (ensuring the necessary power supply for the protection, instrumentation and safety-related systems); and, a quality assurance system [31] (assuring the adequate quality of the NPP safety systems).

. Compliance with international safety standards, guides and recommendations

The ARGOS PHWR 380 complies -as a necessary but not sufficient condition for a safe design- with all applicable safety codes, standards, guides and recommendations issued by the International Atomic Energy Agency and, in particular, with those of the NUSS program [12]. ENACE is prepared to guarantee contractually the applicability of those regulations.

. Compliance with other deterministic criteria based on practical experience and current engineering judgement.

Obviously, ARGOS PHWR 380 complies with the conventional safety criteria intended to ensure safe reactor trip and long-term holding of subcriticality, as well as reliable residual heat removal, limiting the release of radioactive materials into the environment. Great emphasis was given in the ARGOS PHWR 380 design to the achievement of these basic objectives in the most reliable manner, by means of the following measures, among other:

a) Passive engineered safeguards

To control the release of radioactive materials, several passive engineered barriers are provided: the fuel matrix, the fuel cladding; the closed and seal welded high-pressure boundary of the reactor coolant system; the spherical full pressure steel containment; the secondary concrete containment; and the annulus between the steel and the concrete spherical structures, which is exhausted by a specially designed system. But, even for the most unlikely accident sequence resulting in uncontrolled core meltdown, a special venting system for the reactor building is provided allowing for a controlled release through a passive filtering system (see next section). Increase of pressure

within the steel containment beyond its design pressure is avoided, and thus its eventual disruption is prevented.

b) Active engineered safeguards

A wide spectrum of accidents and incidents was considered in the design. To keep the plant in a controlled state under these accidental conditions, active engineered safeguards are automatically actuated and controlled by the reactor protection system. They are designed to meet high reliability targets by means of: conservative and careful design; quality assurance and control; examination and in-service inspections; and inherently safe operating characteristics.

The active engineered safeguards are provided in a diverse manner and each diverse system is redundantly three-branched. Any one of these three branches is designed to cope with a given postulated accident sequence. This principle was also observed in the physical separation of sub-systems, thus avoiding consequent ("knock-on") failures.

c) Defence-in-depth concept

In addition to the instrumentation and control required for normal operation, a condition limitation system is provided, which acts between the normal feedback controls and the reactor protection system, keeping the plant variables within the range specified and enhancing not only the safety but also the availability of the plant. This condition limitation system is also triply redundant.

The reactor protection system also triggers and controls the active engineered features. This is a self-controlling three-channel dynamic system. Reflecting the state of the art, the reactor protection system detects all dangerous deviations from the parameters and triggers all necessary counter-measures automatically. No manual action is required during the first thirty minutes after any incident. This provides additional protection against any tentative improper human action.

Special safety features

Finally, as an ultimate effort towards safety, ARGOS PHWR 380 offers some unique safety features, such as:

. High pressure heat sink

The reactor has a unique safety feature: its moderator system can be used to remove heat in a high-pressure mode. Under normal shutdown conditions, the residual heat can be removed via the steam generators -as in PWRs-, maintaining coolant recirculation either by operating the main coolant pump or by simple natural convection. And ARGOS PHWR 380 includes the additional possibility of using the moderator system as a high pressure heat sink. For this operation mode, the moderator is pumped from the bottom of the moderator tank, cooled in the moderator coolers and injected into the main coolant system. In an emergency core cooling condition, the moderator system serves also as a high pressure injection system. The necessary commutations for the different operation modes are performed automatically according to the already mentioned general design philosophy requiring that no operator action should be necessary within the first 30 minutes after any conceivable incident.

The high pressure residual heat removal (RHR) system is designed for high pressure and temperature. In all conceivable incidents, it can keep the reactor in a hot condition after shutdown as long as it is required or -if convenient- it can cool it down following a predetermined temperature gradient. All branches of the RHR chain are triply redundant and physically separated. Since the steam generators are also available for heat transfer, the plant has two diverse, high pressure, highly available heat sinks for the different accident sequences which may need to be considered.

. Auxiliary and emergency power

One confirmation from the PSA of ARGOS PHWR 380 is that the power supply is in the critical pathway for risk. Accordingly, the auxiliary power supply has been designed to assure adequate reliability levels and a high degree of protection against interruptions. The system is divided into two diverse, redundant and independent systems, which are located in separate sections of the

switchgear building. Moreover, in case of a common mode failure of the normal power supply, an emergency system takes over the feed of safety-related loads. This emergency power supply system is divided into three redundant systems, that are physically separated. Each of these systems comprises: one diesel motor generator, a non-interruptible AC bus bar and a DC supply with batteries, rectifiers and converters. The design also provides that, in case of a prolonged loss of power, the installation can switch an external transportable generator to the emergency power net.

. Leak detection

One of the most important advantages of the ARGOS PHWR 380 design is the possibility of early detection and location of potential leakages in the coolant and moderator systems. This is achieved by tritium detection, the most sensitive method for this purpose, which can only be effectively implemented in this reactor type. Therefore, even the smallest leaks can be detected long before they can threaten the integrity of the primary boundary of the reactor systems. In case of leakage, the localization and further repair can be achieved with a minimum effort and at a very early stage.

. Vented containment

The ultimate lesson learned in the field of nuclear safety is the need to ensure the confinement of radioactive material also in cases of severe hypothetical accidents involving core melt-down. For that purpose, ARGOS PHWR 380 was equipped with a venting system whose objective is preventing the disruption of the steel containment and the consequent uncontrolled release of radioactive materials into the environment that could occur in such an extreme case. Should the pressure increase unexpectedly within the containment, the venting system is designed to stabilize the pressure at a safe value by regulating the release of excess gases and steam into the atmosphere (see Figure 10). The design criterion is that the result of this hypothetical and extremely unlikely situation will be such that even the critical group of the population would not be exposed to projected doses higher than 0.1 Sv. This level of projected dose would not usually justify radiological intervention or counter measures.

In fact, the IAEA has recommended (for radiological emergencies) that "the level of projected dose liable to be received in the short term, below which evacuation is unlikely to be justified, will usually be about an order of magnitude greater than the annual dose limits for members of the public"[32]. Since the applicable limit for unique situations is 5 mSv, such a short term dose should be in the order of 50 mSv. An integrated dose rate over the long term of the order of 0.1 Sv should therefore not usually trigger radiological countermeasures. This is consistent with recent developments in the optimization of radiation protection concerning emergency measures [33].

THEREFORE...

ARGOS PHWR 380 is a recommendable option to break through the current worldwide stagnation of nuclear power programmes. It is so, because:

- as a medium power reactor, it can imply a lower financial impact, a better promotion of domestic industry, lower indirect costs, a simpler adaptation to electrical distribution networks and -possibly- an easier public acceptance;
- its design is featured to ensure installation and operational economy; it reflects the worldwide experience achieved through its kins, the PWRs;
- it can efficiently generate by-products without jeopardizing energy availability;
- it can not only use the independent natural-uranium fuel cycle, but also be fuelled with optimized low enriched uranium assemblies and even with the plutonium it generates; and -last but not least-,
- it offers the usual outstanding safety features of heavy water reactors and, furthermore, its design has been upgraded following the ultimate developments in the nuclear safety field...

... all this has not been done starting from zero but using all the past experience of an already economic and extremely safe industry.

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TABLE 1
MAIN TECHNICAL DATA
(Comparative table)

	CNA I	ARGOS PHWR 380	CNA II
Reactor Type		P H W R	
Gross Generator Output (MW)	367	375	745
Thermal Reactor Output (MW)	1179	1123	2160
REACTOR CORE			
Type of Fuel	Sintered Pelletized Natural Uranium Dioxide, 37 Rods		
Refuelling		on Load	
Number of Fuel Assemblies	253	244	451
Active Length (mm)	5300	5300	5300
Burnup (MWd/Mg)	6000	6600	7500
Mean Fuel-Rod Power (W/cm)	232	223	232
Number of Control Rods	29	60	18
MAIN CIRCUITS			
Number/Main Coolant Loops	2	2	2
Number/Moderator Coolant Loops	2	3	4
Coolant Flow Rate per Loop (Kg/s)	3080	2573	5150
Moderator Flow Rate per Loop (Kg/s)	222	150	222
Operating Pressure (bar)	113	115	115
Coolant Temperature (°C)	262/296	277/314	278/312
Average Moderator Temperature (°C)	140/210	165/220	170/220
REACTOR PRESSURE VESSEL			
Internal Diameter (mm)	5360	5366	7368
Weight of Bottom Portion (Mg)	320	320	670

TABLE 2
FUEL CYCLE ALTERNATIVES
MAIN DATA

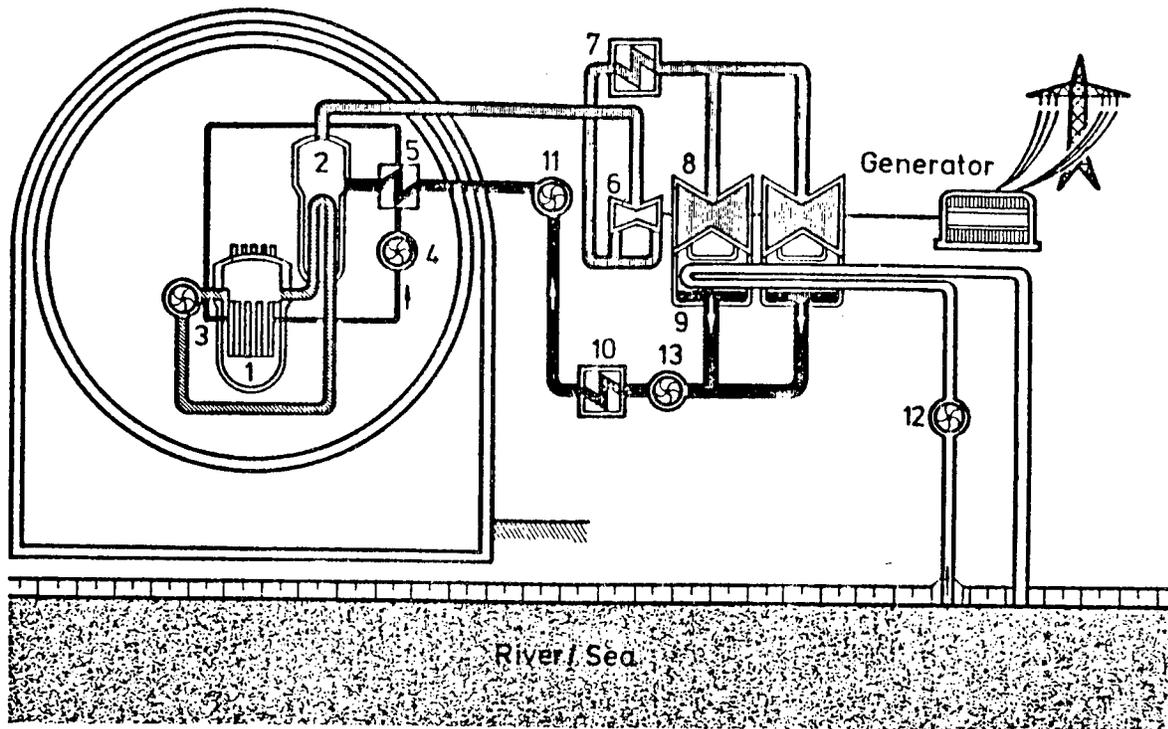
	NATURAL URANIUM (reference case)	L.E.U.	Pu SPIKED	
			NATURAL URANIUM ASSEMBLIES (80%)	PLUTONIUM ASSEMBLIES (20%)
Fuel consumption (f.a./f.p.d.)	0.99	0.41	0.48	0.06 0,54
Average residence time (f.p.d.)	246	594	407	814 453
Fissionable material (%)				
- Fresh U-235 Pu	0.71 0	1.00 0	0.71 0	0.68 2.10
- Discharged U-235 Pu	0.29 0.24	0.15 0.29	0.18 0.28	0.18 0.81
Average burn up (Mwd/Mg.H.M.)	6,000	14,500	9,200	26,000 11,040
Average thermal power per channel (MW)	4.6	4.6	4.3	6.0 4.6
Pu production (kg fiss. Pu/GWe f.p.y.)	430	215		

L.E.U. Low enriched uranium
f.a. fuel assembly
f.p.d. full power day
H.M. heavy metal
f.p.y. full power year
fiss.Pu fissionable plutonium

TABLE 3

MAIN FEATURES OF THE REGULATIONS ON FAILURE ANALYSIS APPLIED TO THE ARGOS PHWR 380 DESIGN

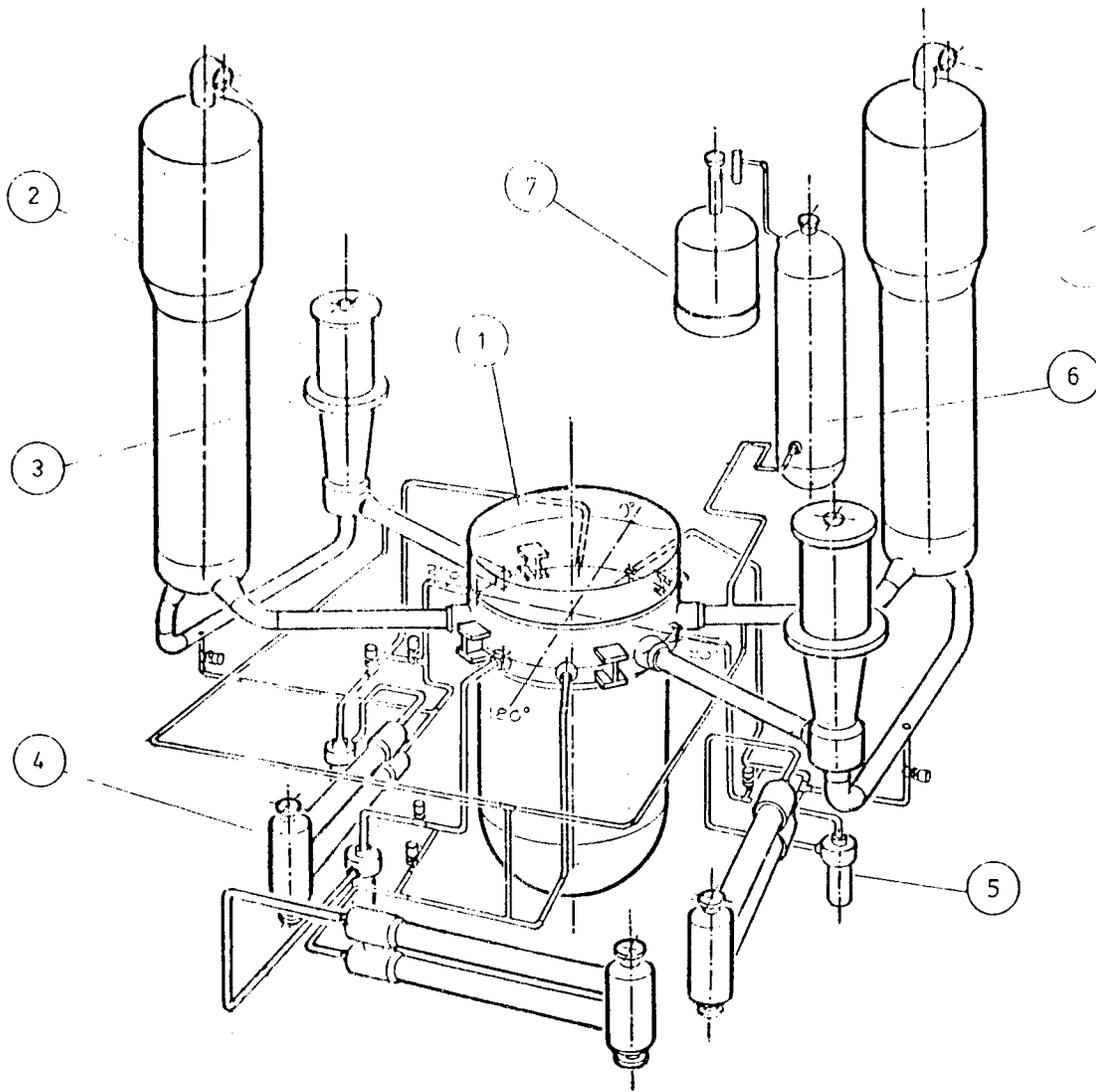
- The probability of occurrence of each identified failure sequence, as well as the corresponding activity of released radionuclides, are assessed by using event and fault trees, while taking into account the following criteria:
 - . The failure analysis must systematically encompass all foreseeable failures and failure sequences, considering the common-mode failures, the failure combinations and -most important- the situations exceeding the design basis. (Failure in this context means an alleatory event preventing a component from performing its safety function, as well as any other event which may additionally occur as a necessary consequence of such deficiency. Failure sequence, on the other hand, means a sequential series of possible failures which can, although not necessarily, occur after an initiating event).
 - . A failure or a failure sequence is selected as representative of a group of failures or of failure sequences. The failure or failure sequence that is selected from the group is that delivering the worst consequences and the analysis takes into account the sum of the probabilities of the failure or failure sequences in the group.
 - . The analysis considers that a protection function may have lost operativeness, either before the occurrence of the failure or of the failure sequence or as a result of such occurrence.
 - . The analysis of failures, of failure sequences or of any part thereof is based on experimental data as far as it is possible. If this cannot be done, the valuation methods are validated through appropriate tests.
 - . Failure rates assigned to safety-related components for evaluating the probability of system failure must be justified. In case that justifiable values were not available for some of the components, levels of failure rate prescribed by the competent authorities are used.
 - . Failure analyses take into account maintenance and testing procedures, and the time interval between successive maintenance and testing actions.
 - . The failure rates postulated for human actions are justified taking into account the complexity of the task, the psychological stress involved and any other factors which might influence that failure rate, balanced with the level of automatization for each interaction in concern.
- The doses on the critical group, that would result from the release of radionuclides due to a failure or failure sequence, must be assessed by accepted methods. (The critical group is defined as a group of people, neighbour to the nuclear power plant, sufficiently homogeneous with regard to the doses expected to be incurred, and representative of the most exposed individuals in case of an accident.) The assessment should take into account the meteorological conditions of dispersion at the site and their probabilities. The assessment should not take into account the eventual application of countermeasures, even if they are forecasted in the emergency planning.
- The annual probability of occurrence of any failure sequence, if plotted as a function of the resulting effective dose equivalent assessed as indicated above, must result -as a necessary but not sufficient design condition- in a point located outside the non-acceptable area of the limit criterion curve. Otherwise, the design must be adjusted accordingly.



- | | |
|---------------------------|----------------------------|
| 1 Reactor pressure vessel | 7 Moisture separator |
| 2 Steam generator | 8 Low pressure turbine |
| 3 Reactor coolant pump | 9 Condenser |
| 4 Moderator pump | 10 Preheater |
| 5 Moderator cooler | 11 Feedwater pump |
| 6 High pressure turbine | 12 Main cooling water pump |
| | 13 Main condensate pump |
-
- | | |
|--|--|
|  Reactor cooler |  Cooling water |
|  Moderator |  Condensate / Feedwater |
|  Main steam | |

Figure 1 - SIMPLIFIED FLOW DIAGRAM

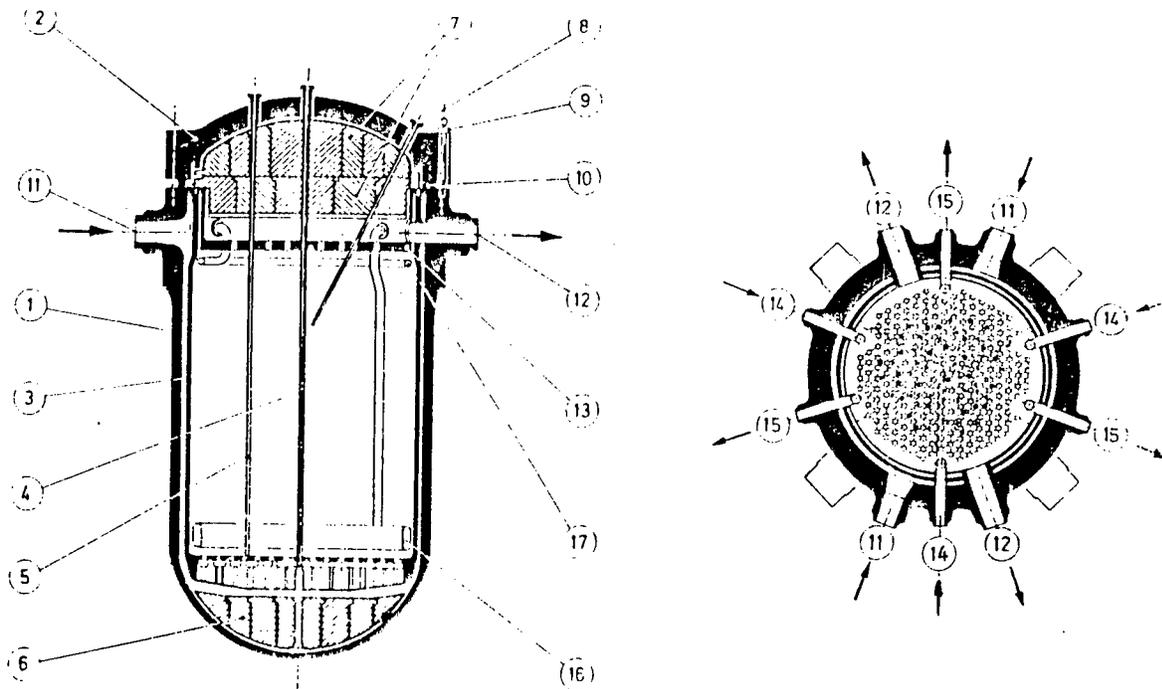
The simplified flow diagram of ARGOS PHWR 380 is equivalent to that of Atucha NPPs. The heat produced in the core is transferred, via the reactor coolant loops, to the steam generators, where the secondary feedwater is transformed into steam. This steam is processed in the turbogenerator, producing electrical energy. The heat generated in the moderator by neutron moderation and heat transfer is extracted by independent moderator loops and used for preheating the feedwater, enhancing the high efficiency of the plant.



- | | |
|----------------------------|----------------------------|
| 1- Reactor Pressure Vessel | 5- Moderator Pump |
| 2- Steam Generator | 6- Pressurizer |
| 3- Reactor Collant Pump | 7- Pressurizer Relief Tank |
| 4- Moderator Cooler | |

Figure 2 - REACTOR COOLANT AND MODERATOR CIRCUIT

The reactor coolant system of ARGOS PHWR 380 is fully comparable in design and arrangement with those of PWRs. It consists of the reactor pressure vessel (RPV), in the center, and two identical coolant loops, each comprising one coolant pump and one steam generator. In addition, as a typical HWRs feature, there is a moderator cooling system, which is subdivided into three identical loops, each comprising one moderator pump and one moderator cooler. The moderator sys-



- | | |
|--|------------------------------------|
| 1. RPV (reactor pressure vessel) | 9. Stud |
| 2. Closure head of RPV | 10. Closing joint |
| 3. Moderator tank | 11. Coolant inlet |
| 4. Coolant channel | 12. Coolant outlet |
| 5. Guide tube of control rod shut-down rod | 13. Closure head of moderator tank |
| 6. Lower filler pieces | 14. Moderator inlet |
| 7. Upper filler pieces | 15. Moderator outlet |
| 8. Boric acid injection-line | 16. Moderator piping lines |
| | 17. Moderator piping lines |

Figure 3/4 REACTOR PRESSURE VESSEL AND INTERNALS

The reactor pressure vessel holds the reactor core with 244 vertical coolant channels containing one fuel assembly each. The coolant channels penetrate the moderator tank. The pressure between coolant and moderator is equalized by openings in the moderator tank closure head, resulting only in a slight pressure difference and, therefore, requiring only a thin-walled coolant channel.

The 244 coolant channels producing 1067 MW of thermal power are arranged in a triangular lattice array with a pitch of 27.5 cm. Control and shutdown absorbed rods, hydraulically moved, are vertically inserted. A total of 60 rods is used, 45 of them for shutdown and longterm subcriticality. The remaining rods are power control and power density control rods, designed for flattening the power density distribution over the core.

- 1 New fuel store
- 2 Manipulating bridge
- 3 Spent fuel pool
- 4 Transfert vessel
- 5 Refueling machine

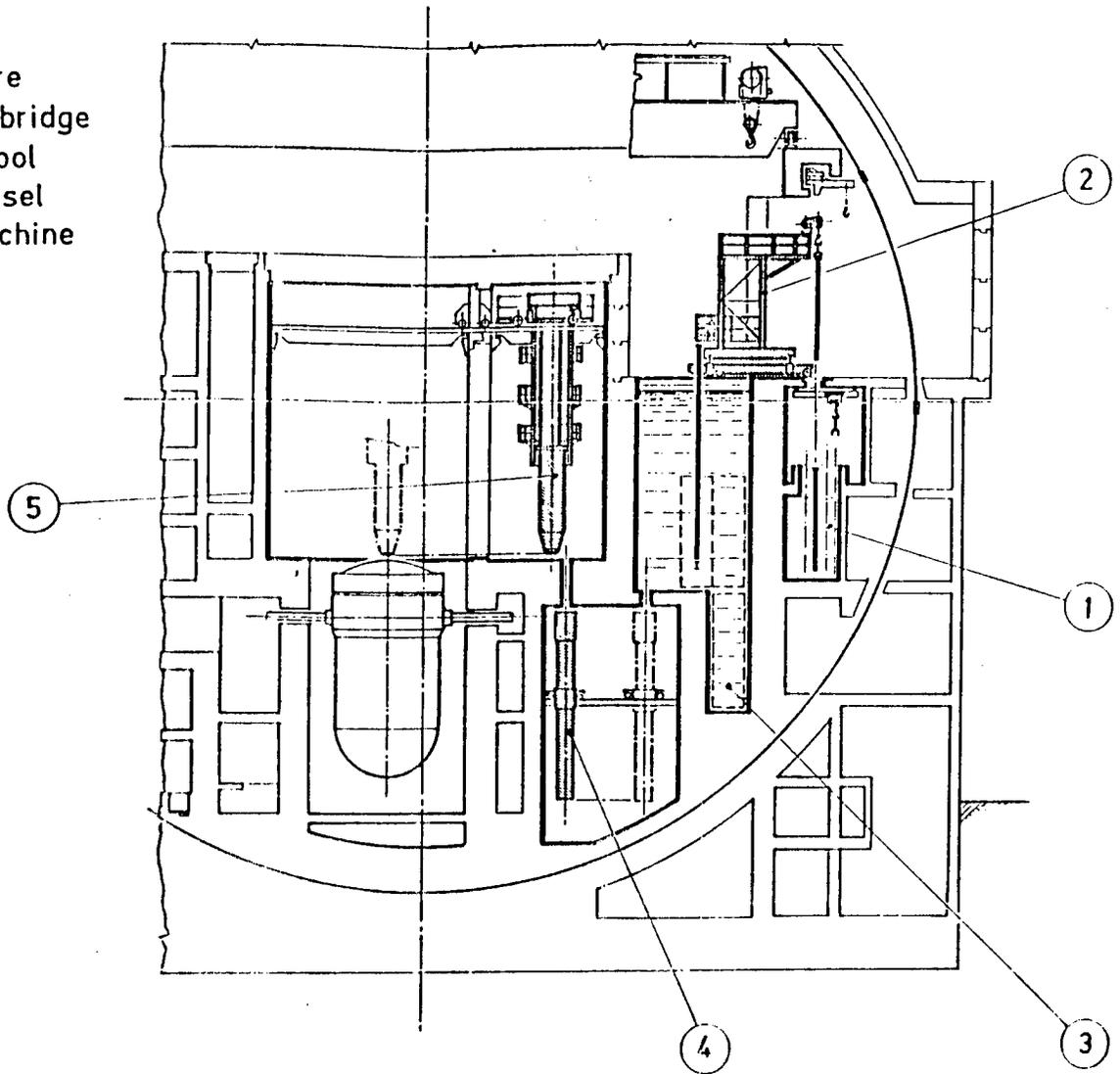


Figure 5 - REFUELLING

The refuelling system of ARGOS PHWR 380 is designed for the automatic refuelling of the reactor during full power operation. New fuel is inserted into the transfer vessel, where a fluid change from H_2O to D_2O takes place. The refuelling machine takes over the fuel assembly and performs reloading and reshuffling operations on top of the reactor pressure vessel. The spent fuel is carried to the spent fuel pool into the opposite direction. The capacity of the spent fuel pool can be tailored up to the design lifetime of the plant.

A special feature of this reactor type is the possibility to insert irradiated fuel assemblies in the same way as the fresh ones, as well as special assemblies to produce cobalt-60 by irradiation.

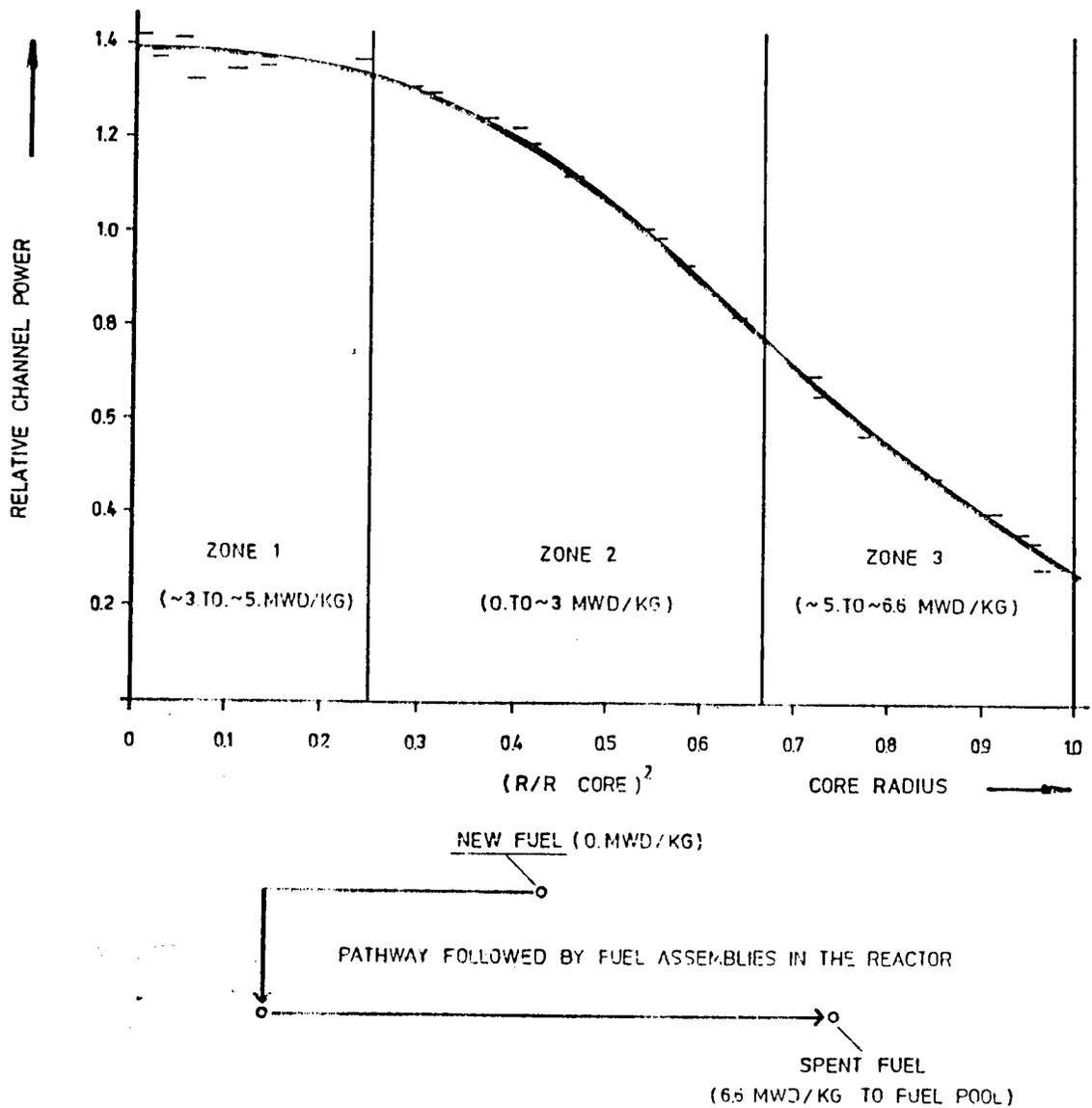
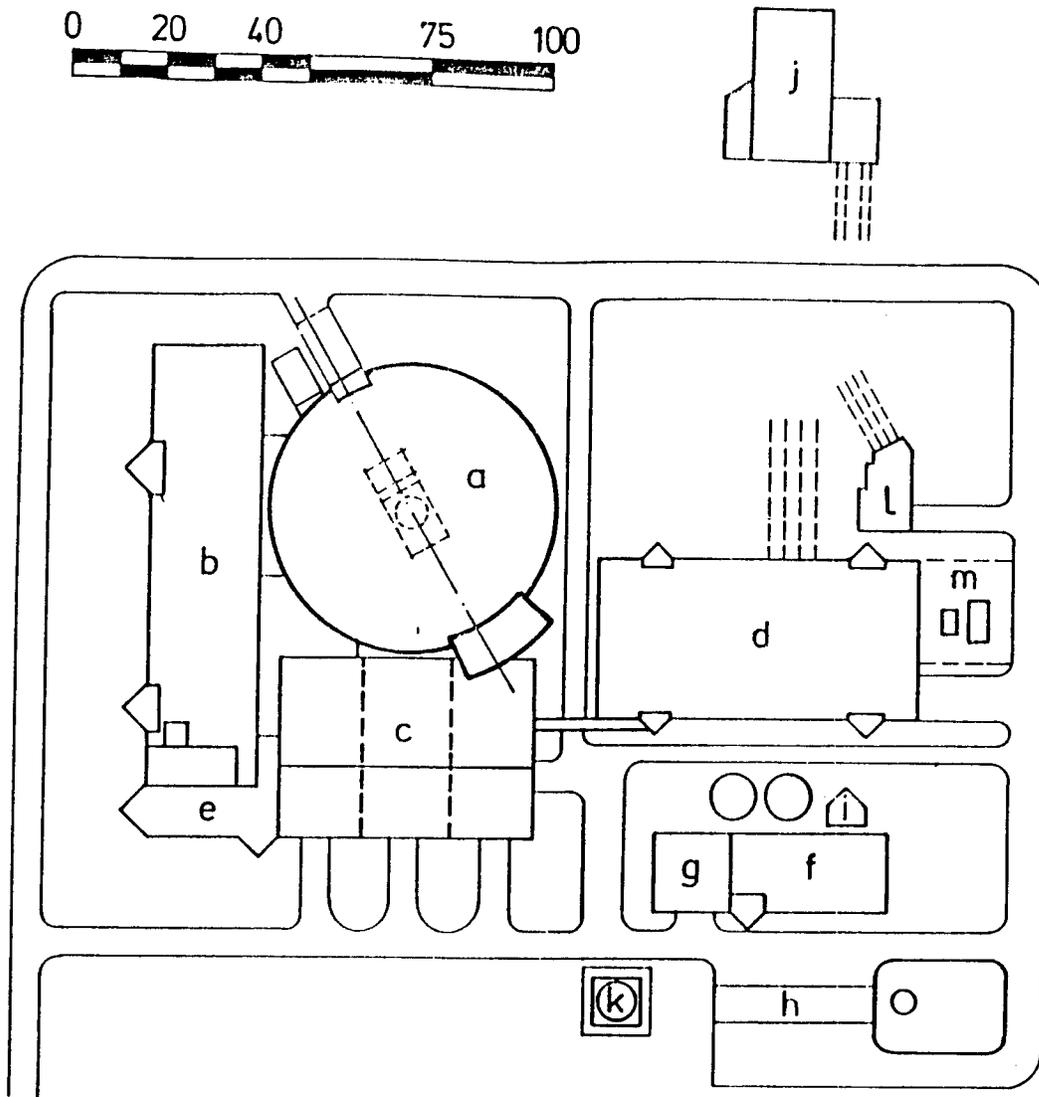


Figure 6 - RADIAL POWER DISTRIBUTION

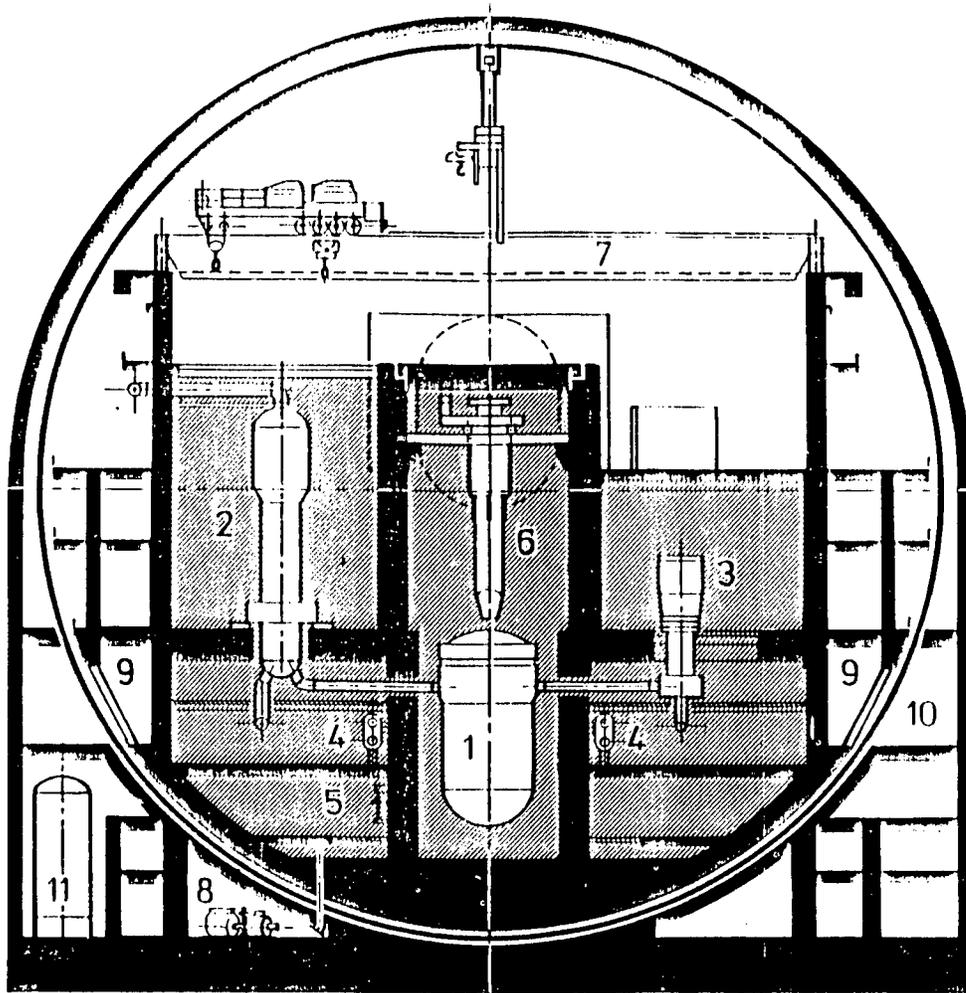
In order to assure a flat power density distribution, a radial reshuffling scheme of 3 zones/1 way is applied. This implies that the fresh fuel is introduced into a channel at an intermediate position. From there, the partly depleted fuel is shuffled to the core center and, finally, to the reactor periphery. In this way, an average discharge burn up of 6600 MWD/MgU can be achieved with natural uranium fuel.



- a. Reactor building
- b. Reactor auxiliary building
- c. Switchgear and emergency power supply building
- d. Turbine building
- e. Staff facilities and office building
- f. Demineralizing system building
- g. Auxiliary boiler and air compressor building
- h. Gas cylinder store
- j. Cooling water intake structure
- k. Fuel oil tank
- l. Service cooling water collecting pit
- m. Transformer park

Figure 7 - PLANT LAYOUT

The balance of plant is designed to ensure economy in construction and operation. The following aspects are considered in the arrangement of the building: clear energy flows, short piping and cable runs; good access for construction,



▨ Equipment compartments

□ Operating compartments

1. Reactor pressure vessel
2. Steam generator
3. Reactor coolant pump
4. Moderator cooler
5. Moderator pump
6. Refueling machine

7. Reactor building crane
8. Safety injection pump
9. Pipe duct
10. Cable spreading area
11. D₂O Storage tank

Figure 8 - REACTOR BUILDING CROSS SECTION

One of the most important features of the ARGOS PHWR 380 is its double containment, which consists of two concentric spherical structures: an inner metallic sphere and an outer concrete shield building. The optimized steel containment encloses the nuclear steam supply system and is designed to resist the maximum pressure derived from any conceivable loss-of-coolant accident. The outer concrete shield building acts as a secondary containment and is designed to protect the steel sphere from any relevant external events.

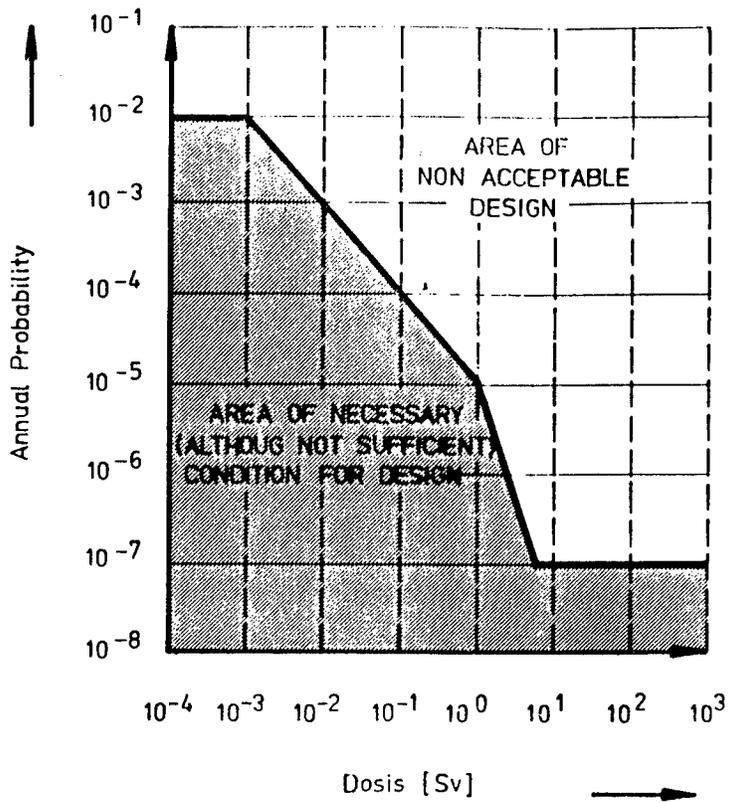


Figure 9 - LIMIT CRITERION CURVE

The probabilistic safety assessment performed for the ARGOS PHWR 380 shows compliance with the above criterion curve, which is a necessary but not sufficient condition for NPP licensing in Argentina. The authority's limiting criterion is that an annual risk limit of one-in-a-million must be respected for any individual who might hypothetically be subject to accidental exposures from a NPP. Since accidental exposures may result from several accident sequences, an annual risk upper bound of one-in-ten-millions must be respected for around ten selected relevant sequences. As each sequence may result in different doses, the above criterion curve or limit line is used.

The logics behind this criterion curve is as follows. For the range of doses from which only stochastic effects of radiation can be incurred, the criterion curve must show a constant, negative, 45° slope in a -log annual probability versus log individual dose- coordinate axis plane. This would ensure that the risk, i.e., annual probability of incurring the dose multiplied by the probability of serious deleterious effects given the dose (the latter being in the order of 10^{-2} per sievert) will be kept constant. One of the coordinate points in this part of the curve would obviously be the following: | Annual probability = 10^{-5} ; Individual dose = 1 Sv |, because the product $10^{-5} \text{ annum}^{-1} 1 \text{ Sv } 10^{-2} \text{ Sv}^{-1}$ results in an annual risk of 10^{-7} which is the risk upper bound for any scenario from the postulated initiating events. In the dose range where non-stochastic effects of radiation may occur (i.e., for individual doses higher than approx. 1 Sv), the slope of the curve should increase, in order to take into account the higher risks of death at these levels of dose. For doses higher than approximately 6 Sv, the probability of death approaches unity. From this level to higher doses, the criterion curve should remain constant at an annual probability of 10^{-7} (because the exposed individual would inevitably die regardless the level of the dose). Between the coordinate points defined by | Annual probability = 10^{-5} ; Individual dose = 1 Sv | and | Annual probability 10^{-7} ; Individual dose = 6 Sv |, the criterion curve should show a shape inverse to the dose-response relationship (which, at that range, is approximately S-shaped; however, for the sake of simplification, the regulatory authority has decided to approximate these two points by means of a linear-shaped relationship). Finally, the criterion curve has been truncated at an annual probability level of 10^{-2} , because the occurrence of incidents having a higher annual probability (regardless the dose) is unacceptable for the regulatory authority.

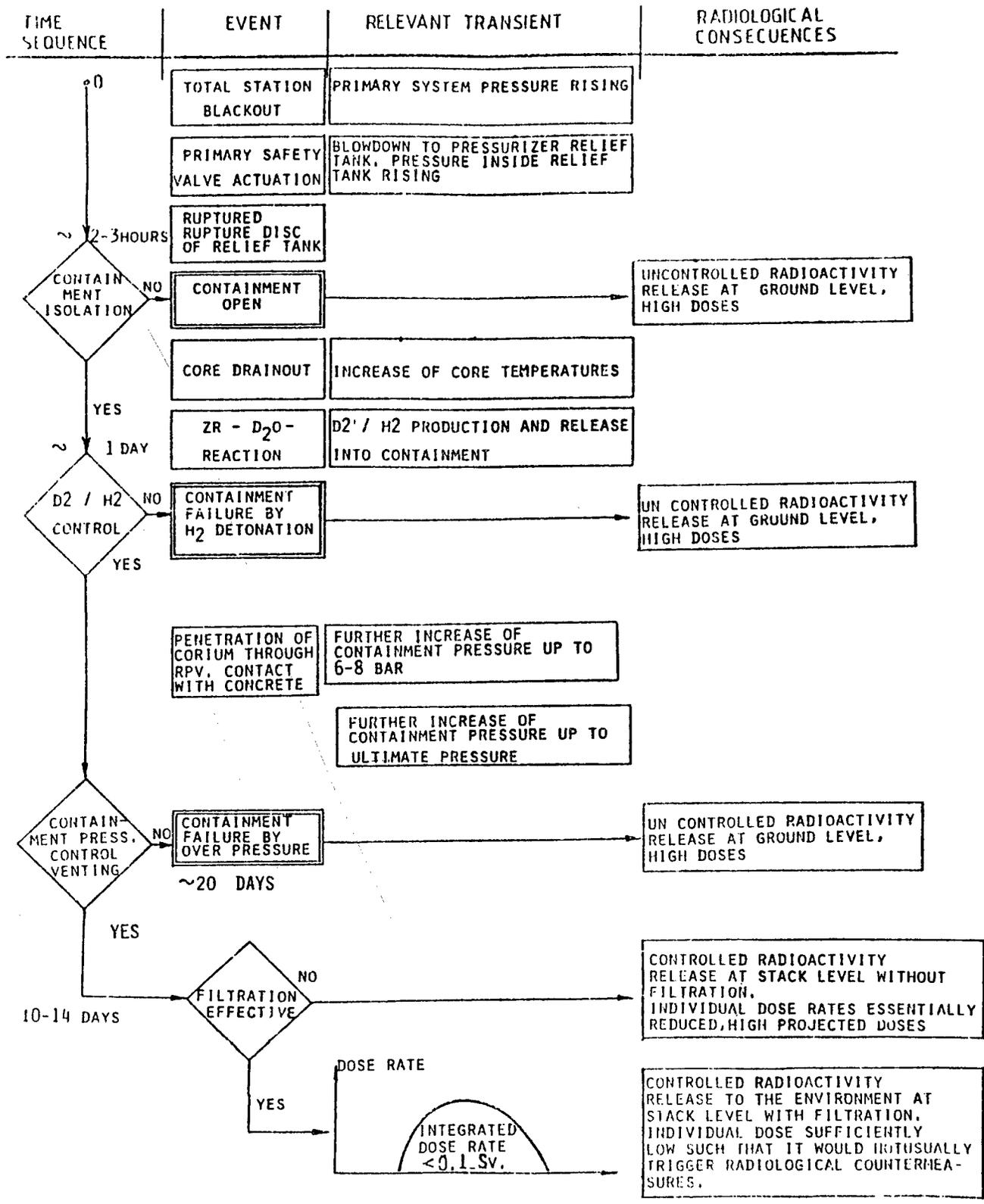


Figure 10 - EXAMPLE OF THE OPERATION OF THE ARGOS PHWR 380 VENTED CONTAINMENT

The intended functioning of the vented containment of the ARGOS PHWR can be exemplified by following a postulated accident scenario, the so-called "emergency power" case, in which the highly reliable emergency power supply would be unavailable and non-recoverable. In such a case, after reactor shutdown, the temperature of the reactor coolant would increase as a result of residual heating in the core. The primary pressure would reach a level that would actuate the safety valve. Primary coolant steam would then flow into the pressurizer relief tank, where the pressure would eventually lead to bursting of a rupture disk. Radioactive coolant steam would then be released into the containment environment, where it would be retained. The design of this extremely important boundary includes provision for sufficiently reliable isolation of pipes and ducts penetrating the steel sphere by redundant valves. Otherwise, there would be an immediate release of radioactive materials into the reactor building annulus, followed by an uncontrolled release into the environment and a subsequent unacceptably high public exposure and environmental contamination.

Following the containment isolation, which can be ensured either by automatic or by human action, all consequences would be limited within the containment environment. The core would dry out and eventually melt. The cladding of the fuel rods would burst and its temperature would increase until a zirconium water reaction would take place. Hydrogen, along with fission products, would be released into the containment environment. There, another important system of the containment would become effective: the early ignition of the hydrogen by appropriate catalysts which will operate without auxiliary power. This effect will ensure the integrity of the containment, by avoiding a later hydrogen deflagration with possible rupture of the steel sphere.

Following with this catastrophic scenario and still in the absence of electrical power supply, the melting core is assumed to penetrate the bottom of the reactor pressure vessel and come into contact with the floor of the reactor vault. There might then be an exothermic reaction between the melted core material and the concrete. The released energy would increase the pressure inside the containment significantly (that pressure would already amount some bars, because of the blowdown of the primary coolant).

One of the most significant features of the ARGOS PHWR 380 design would now become effective. The containment design pressure of about five bar would be reached only after some 10-14 days, when all the short-lived fission products would have already decayed and most of the radioactive material would have been deposited. This is due to the very favourable relation between the very large containment volume and the comparatively small energy content of the primary system. Without any counter-measures at this stage, pressure would increase inside the containment up to its rupture, which would take place some 20 days after the initiation of the described hypothetical scenario. In such a case, the consequences would be extremely modest compared with those from the Chernobyl accident, where such containment function did not exist; however, a containment rupture would lead to an uncontrolled release of the radioactive materials still remaining in the containment into the environment. The ARGOS PHWR 380 design therefore incorporates venting, to permit the controlled release of gases and aerosols from the containment into the environment through a filtering system, relieving the pressure and thus avoiding the containment failure. The efficiency of the filtering system ensures that the resulting projected dose to the most exposed individual must not exceed under any circumstance 0.1 Sv.