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MATURITY OF THE PWR.

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1. INTRODUCTION

At the end of 1982, the breakdown of PWR units throughout the world (23 countries excluding Eastern Bloc and satellite countries) which is provided in Figure 1 was as follows :

- 110 PWR units were in operation or 47 % of the total number of 234 nuclear power plant units of all the different reactor systems; these 110 units have a net generating capacity of about 90 GW, which represents 60 % of the world's total generating capacity, and have accumulated 750 reactor-years of operation (Figure 2),

- Based on the number of units currently under construction or on order, the number of PWR units should increase to 226 between 1990 and 1995 or, in other words, 52 % of all the units in operation. These 226 units would provide a net electrical capacity of 210 GW or 63 % of the total electrical capacity of all reactor systems in service.

These figures illustrate the predominant position of the PWR system. The question is whether on the basis of these figures the PWR can be considered to have reached maturity.

The following analysis, based on the French program experience, is an attempt to pinpoint those areas in which industrial maturity of the PWR has been attained, and in which areas a certain evolution can still be expected to take place.

2. AREAS IN WHICH THE PWR CAN BE CONSIDERED TO HAVE REACHED MATURITY

2.1. Choice of main design options

The various manufacturers of PWR components in the different countries evolved simultaneously toward a large degree of uniformity in their choice of the main design options.

This resulted in an increasingly uniform design for the following main PWR components and systems :

- cyclindrical shaped fuel rods which are similar in diameter, arranged in a 17 x 17 square array of approximately the same size and which are reloaded during shutdown of the reactor,
- a loop type reactor coolant system with steel pressure vessels and vertically installed components,
- steam generators for the most part the recirculation type, generating dry and saturated steam and fitted with "U" shaped tubes,
- controlled leakage seal type reactor coolant pumps,
- auxiliary and engineered safeguard systems, ensuring identical main functions.

The validity of the design chosen for the PWR components was confirmed recently in France at the end of the preliminary feasability studies performed by FRAMATOME in association with EDF and CEA for the N4 project. These studies also underlined the advantages due to the relative simplicity in the design of certain PWR components.

2.2. Plant performances and component sizing

The various manufacturers of PWR Nuclear Steam Supply Systems have also reached a certain stability in the evolution of the performance parameters and of the size of the main components.

The main PWR design parameters specific of plant performances (reactor coolant temperature and pressure and steam pressure) are nearly the same. The capacity of the loops and size of the components which equip current PWR units have now reached a technical and economical optimum level of development which is considered to be practically close to the maximum technological limits of the existing industrial capacities.

This trend is illustrated in Figures 3 and 4 which show the changes in size and steam pressure over the years, for different series and models of steam generators, including those manufactured for the French Program, CPY , P4 - P'4 et N4.

2.3. Progressive disappearance of initial difficulties

The operating experience acquired with the first series of PWR's, in particular those put in operation between 1967 and 1972, has indicated a certain number of difficulties which have necessitated corrective measures.

These measures which have been analyzed and qualified with the assistance of important Research and Development programs, have been progressively incorporated in the following series of plants, particularly in the French units.

One example worth noting concerns the various forms of steam generator tube degradation for which appropriate modifications of operating conditions (such as water chemistry, systematic tube-sheet cleaning, ...) and local design modifications (such as improvements in thermo-hydraulics, tube supporting elements, manufacturing techniques) have been applied.

As a result, the steam generator operating experience on the first French twenty-two 3 loop units, having accumulated from one year, up to eight years of operation, is quite satisfactory : at the end of 1982, out of a total of more than 220,000 steam generator tubes in service, 63 tubes have been plugged for various reasons, that gives a damage ratio of 2.4×10^{-4} (Figure 5).

On the other hand, in-service performance of fuel loads has been excellent as shown by the following figures : out of a total of 52 reactor cycles either completed or near completion, the maximum iodine 131 activity level in the reactor coolant system was less than 5×10^{-4} Ci/t (equivalent to zero fuel leakage) for 20 of these cycles (Figure 6). Furthermore, the maximum activity detected was lower than 5×10^{-2} Ci/t which is only 2.5 % of the maximum permissible design criteria for the activity level in the reactor coolant system.

As a global result, the plant availability factor of the first six units of the 900 MWe series (from FESSNHEIM 1 to BUGEY 5) is comparable to - if not better than - that obtained by recently built fossil fueled plants (Figure 7).

2.4. Evolution in Research and Development Programs

One of the consequences of this trend is a change in the orientation and contents of R&D programs.

Prior to this change, R&D programs were aimed primarily at providing aid and support in the following areas :

- qualification of design, manufacturing processes and operating procedures, with the goal of solving initial experienced problems and of improving equipment performance and reliability,
- qualification of support analytical tools,
- demonstrations and justifications of assumptions and devices involved in the safety area.

To day, R&D programs are more oriented toward refining component performance and reliability as well as providing a greater ease in plant operation and maintenance.

The Figure 8 summarizes the main R&D programs performed or pursued in France from the mid 1970 s. Although a considerable period of time is required from the date at which a new concept is created and its actual industrial implementation, a number of significant benefits in PWR technology have occured, or are close to be available, such as :

- qualification of the Advanced Fuel Assembly (AFA) 3.7 m long, the first loading of which in an EDF 900 MWe reactor is scheduled for the end of 1984,

- in the area of NSSS adaptability to grid requirements, a solution has been provided in eliminating wear of control rod assembly and control rod driving mechanisms (problem which required improving wear resistance of these components to some eight million operating cycles), and also was given of the ability of the Reactor Advanced Maneuverability Package (RAMP) to ensure daily load follow,

- qualification of hydraulic design and bearing improvements of Reactor Coolant Pumps,

- design and performance improvements of moisture separators and dryer equipments of the steam generator.

2.5. Experience in manufacturing

The manufacture of PWR components and, in particular, the main primary system components has required tremendous efforts in laboratories and factories in order to design the best adapted production tools, finalize equipment specifications, establish and enhance manufacturing methods and codify and normalize operating modes and calculations.

In the field of welding technology for example, which is extremely diversified (from the welding of thin Inconel tubes on steam generator tubesheets to the welding of reactor vessel elements of up to 150 mm thick or more) , the following evolution in industrial techniques was applied :

- first of all, the development of processes and equipment in test laboratories and then in workshop in order to determine and verify the various parameters, and also ensure the repeatibility of the process (often requiring test runs longer than 12 months) ,
- industrial development by specialized manufacturers of reliable and high precision components.

The high degree of precision frequently required in machining operations (up to several $1/_{100}$ mm on parts measuring several meters) made it necessary to select or develop high precision machines such as :

- deep drilling machines for tubesheets with a thickness of more than 600 mm,

- special machines for broaching steam generator tube support plates; these machines were developed solely for this specific application with the participation of a specialized manufacturer.

The industrialization procedure was also applied to non-destructive testing methods and involved the same above mentioned stages of development in test laboratories and in manufacturers workshops.

It is now widely recognized that expertise in new methods presently being developed (involving the use of automatic devices for certain processes) has been acquired or soon will be.

Acquiring this expertise was made possible by the experience encountered in solving various problems or malfunctions which required modifying specifications or methods. This expertise resulted in the establishment in France of a set of rules governing the design and construction of mechanical components for the nuclear steam supply systems, the RCC, illustrated in Table 1.

The RCC-M constitutes a set of rules for the design, construction and inspection of mechnical components which describe the standard practice of the French nuclear industry.

2.6. Energy generation costs

The competitiveness of the PWR, as compared with the other thermal generating plants has been recently reconfirmed in France by the study of a working group comprising representatives from government Ministeries, Electricité de France and the Commissariat à l'Energie Atomique (CEA). The main results are the following for plants that are scheduled for commercial operation by 1992 :

- with a postulated evolution of fuel costs of 7.6 c/th for coal (1982 cost 5 c/th) and 24.4 c/th for fuel oil (1982 cost 10.7 c/th), the previsional generating cost per kWh will be as follows : (French Francs on January 1982 basis) PWR units = 0.20 FF COAL fired units = 0.33 FF , OIL fired units = 0.68 FF , that is to say with the following ratio : PWR 100 , COAL 170 , OIL 350,

- in an extreme case with no evolution in fuel costs between 1982 and 1992, the above mentioned ratio would be PWR : 100 , COAL : 145 , OIL : 200 ,

- compared with coal-fueled plants, the PWR remains competitive at partial plant utilization down to 2000 hours/year ,

- to become competitive with the PWR, the fuel oil cost would have to be decreased by a factor of more than 7 and 3 respectively based on the two above mentioned assumptions of the evolution in fuel cost.

3. CONSEQUENCES FOR THE PWR EVOLUTION

3.1. Standardization

The stability in the design and size of PWR components referred to above, led manufacturers to produce components in standardized series. In France, where this policy was encouraged by EDF which had practised the same policy on earlier fossil-fueled plants, standardization was first applied to the main components which resulted in :

- two types of 17 x 17 fuel assembly which differ only in length and in the design of the nozzles at the end of each assembly,
- two main steam generator models (model 51 and 68/19) already in-service and two improved models incorporated in recent proposals,
- two reactor coolant pump models already on the market (93 D and 100) and a new model, the N 24 under final development.

This standardization of PWR components was further applied to Nuclear Steam Supply Systems and resulted in the following standardized Models (see Table 2) :

- the 3-loop 870-920 MWe CPY Model , 34 units of which have been commissionned or are under construction after the completion of an initial series comprised of seven units,

- the subsequent 3-loop Models (M 30 and M 31) with an increased net capacity of 975-1000 MWe,

- the 4-loop P4-P4' Models with a capacity of 1270-1300 MWe, 16 units of which are under construction and options have been taken for an additional four units of this series,

- a 4-loop Model , N4 , which is presently in the negotiation stage with EDF.

Standardization has been extended, as applied to the various NSSS and complete nuclear power plants, to cover the general plant layout and the main civil works structures.

This policy of standardization, combined with the implementation of a large number of units , results in the setting up of extensive industrial capabilities which thus ensure greater PWR expertise; these capabilities are furthermore well adapted to a standard product and to acquiring working techniques which allow to reduce lead times and mitigate the risks involved in the manufacturing stage.

3.2. Setting up large industrial capabilities

Large industrial capabilities have been set up to ensure full coverage of the design and construction of nuclear power plants.

These capabilities include highly efficient analytical tools created for specific purposes such as :

- thermohydraulic design codes used for reactor core and steam generator analysis (3-dimensional codes are now operational),

- tools and methods for performing stress analysis, dynamic strentgh analysis, fracture mechanic behaviour, etc. (an example is the performant TITUS computer code system , currently running at FRAMATOME Mechanical Analysis Department),
- methods for analysing operating sequences and physical phenomena under postulated accident situations.

The production plants include facilities specialized in the manufacture of components for which the production tools can be adapted to a standard product and benefit from increased automation which enhances both quality and productivity. Examples of such facilities often mentioned are : the ZIRCOTUBE factory at PAIMBOEUF (France), the FBFC PWR fuel assembly manufacturing factories at ROMANS (France) and DESSEL (Belgium) , and in 1984 CFC fuel assembly manufacturing factory of PIERRELATTE (France), FRAMATOME's heavy component factories at CHALON and LE CREUSOT in France, where the reactor vessels, steam generators and pressurizers are manufactured and the JEUMONT factory in France which produces the reactor coolant pumps and control rod drive mechanisms, etc.

It should also be mentioned that the human factor has not been neglected regarding the training of plant personnel in specialized fields. For example, a training center was created at CADARACHE (France) to offer training to personnel who participate in on-site testing and startup operations. More than 220 engineers have received specialized training since the center began operating in 1978.

These extensive industrial facilities have provided France with a considerable technological and industrial potential which is capable of ensuring rapid corrective action in the case of unexpected difficulties. The best known example involves the problem of underclad cracks in reactor vessels and steam generators which was successfully analyzed and treated in 1979 and 1980, and is now completely solved. Another more recent problem concerns the failures of the RCC guide tube alignment pins. The problem was solved by appropriate on-site corrective action which kept outage of the units involved to a minimum.

3.3. Reduction of lead times

Proof of the expertise acquired in manufacturing techniques is illustrated by the reduced lead times recorded in France for the commissioning of the 900 MWe units.

Figure 9 illustrates these reduced lead times. Overall figures show that the total time required for the construction of a 900 MWe unit from the signing of the contract to its connection to the power grid dropped from 75 to 80 months for the first 900 MWe units, FESSENHEIM and BUGEY 1 to around 60 months for the last units of this series to be commissioned with an average completion time of 65 months for the 23 units in service.

4. PERSPECTIVES

The above described maturity achieved through stability in design and standardization of components and plants does not exclude, however, the possibility of a controlled evolution while maintaining continuity in order to :

- benefit from any changes in technological limits as they occur as well as from new developments in existing techniques,

- enhance the quality of a product in certain respects.

Concerning the first point, the possibilities offered by the development of forgings which are either larger or produced by new techniques allowing to simplify or improve reactor vessel manufacturing techniques (fewer welds, improved quality of the base metal) should be mentioned. Generic developments in welding, instrumentation and data processing methods can also contribute to such an evolution.

The second point pertains to extending existing R&D programs which is aimed primarily at :

- increasing reliability and safety,
- promoting operator aid,
- reducing maintenance personnel exposure to radiation.

R&D efforts are also continuing toward obtaining a reduction in fuel cycle costs and in the amount of fissile material consumed using current designs. This R&D concerns only modifications in the reactor core and does not extend to the main NSSS components which remain basically unchanged.

5. CONCLUSION

In the light of the analysis presented above, everyone is free to draw his own conclusion concerning the degree of maturity achieved by the PWR.

It should be noted, however, that if the decision initially taken by the French government, EDF and French Industry to develop the PWR system has since proved to be the best choice, this is because France's nuclear program was based in 1969 on a reactor type which had at that time the greatest potential for ensuring full success of this program. This potential was subsequently developed to a much larger extent by the policy applied in France and which can be summarized by three key words : concentration, standardization and continuity :

- concentration of the industrial and engineering capabilities,
- standardization of components and models,
- continuity in the program .

TABLE 1

DESIGN AND CONSTRUCTION RULES

FOR PWR NUCLEAR ISLANDS COMPONENTS

R. C. C.

MECHANICAL : RCC - M	AVAILABILITE
- 1981 EDITION	IN FRENCH IN ENGLISH
- 1983 EDITION	IN FRENCH BY FEBRUARY 83 IN ENGLISH BY JUNE 83
ELECTRICAL : RCC - E	
- 1981 EDITION	IN FRENCH IN ENGLISH
SYSTEM DESIGN : RCC - P	
- 900 MW - 1979 EDITION	IN FRENCH IN ENGLISH
- GENERAL	IN PREPARATION
FUEL DESIGN : RCC - C	NEARLY COMPLETED
CIVIL WORKS : RCC - G	SOON AVAILABLE
FIRE-FIGHTING : RCC - I	SOON AVAILABLE

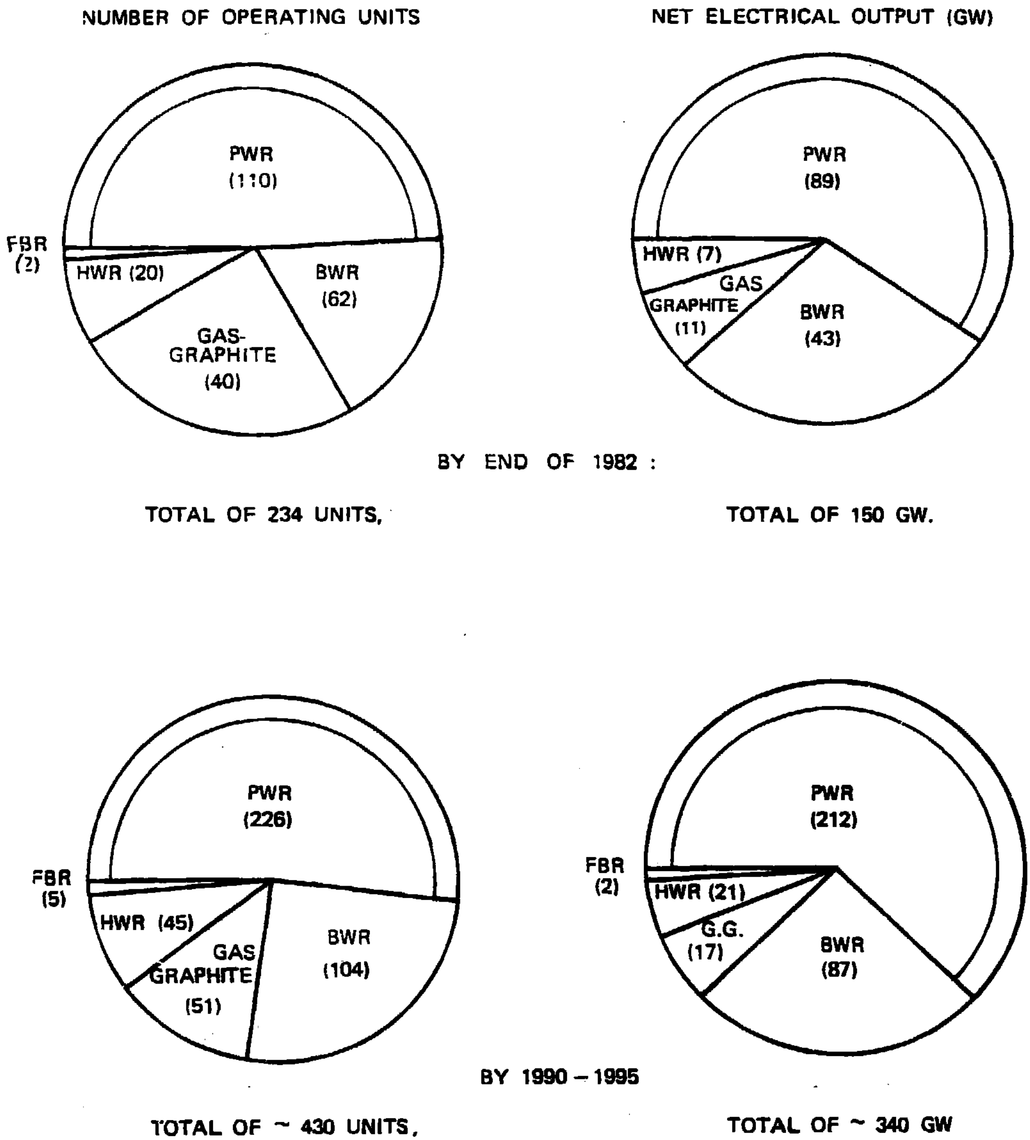
★★★

OTHER RULES (RRC) ARE ALSO IN PREPARATION FOR THE CONVENTIONAL ISLAND

TABLE 2

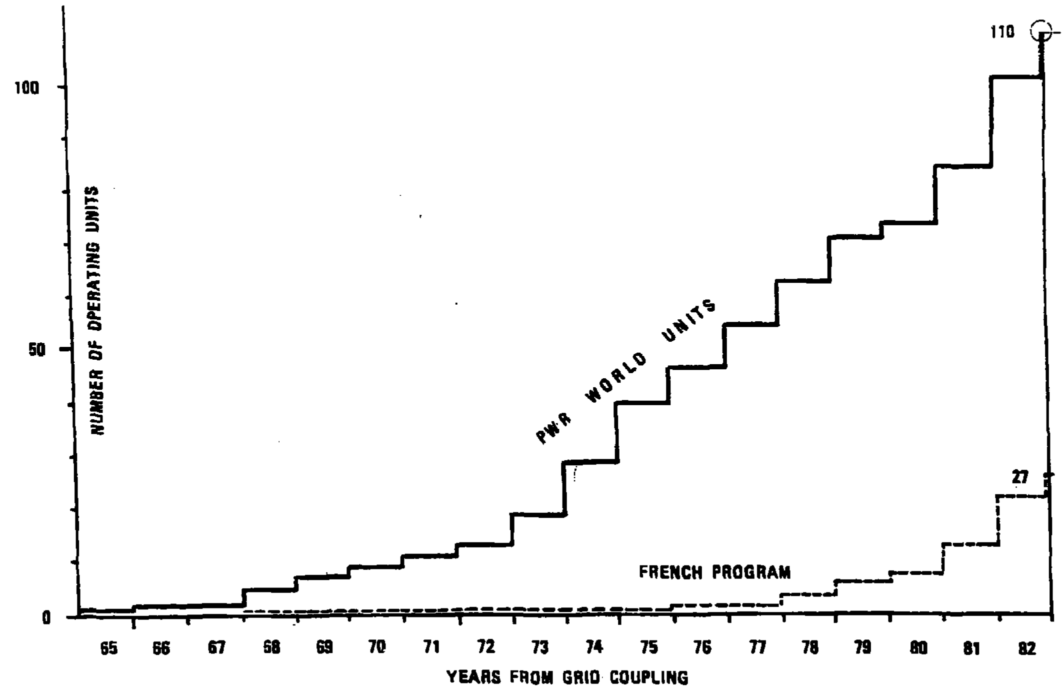
FRAMATOME STANDARD NSSS MODELS

MODELS		CPY	M 30	M 31	P4 - P'4	N4
NSSS THERMAL POWER	MW	2785	2905	2905	3817	4270
NUMBER OF LOOPS			3		4	
FUEL ASSEMBLY			17 X 17		17 X 17	
ACTIVE FUEL LENGTH	M		3.66		4.27	
REACTOR VESSEL DIAMETER	M		4.0		4.4	4.5
PRESSURIZER VOLUME	M^3		40		60	
STEAM GENERATOR MODEL		51 B	51 BS	55/19	68/19	73/19
REACTOR COOLANT PUMP MODEL		93 D	100	100	100	N 24
NUMBER OF ORDERED UNITS		7 + 34			16 + 4	



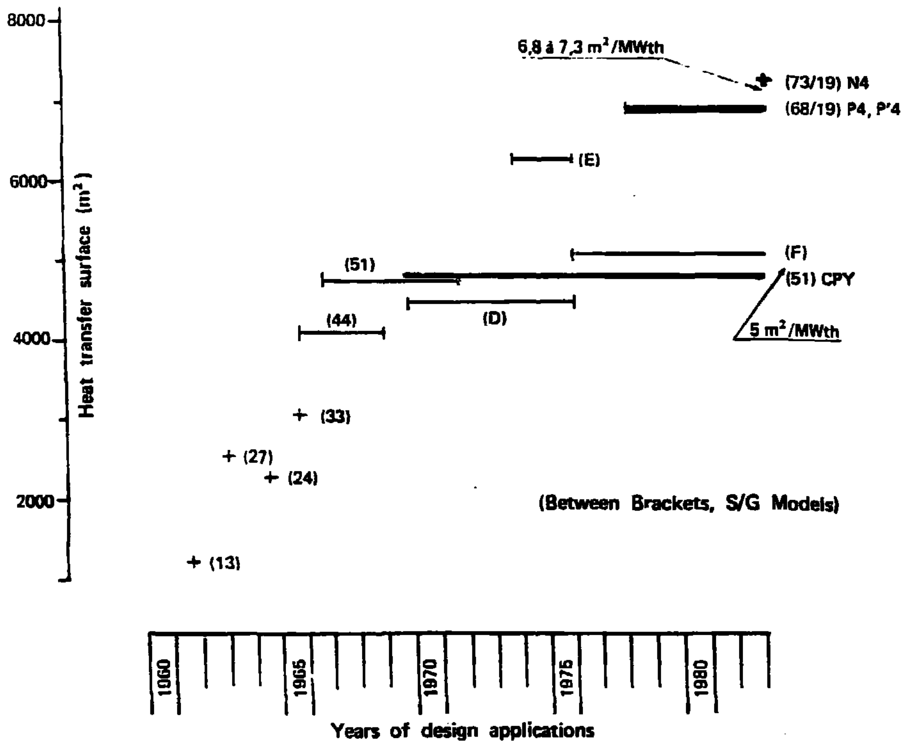
THE P.W.R. PART INSIDE THE TOTAL WORLD NUCLEAR PLANTS (East Countries - non included)

Fig. 1



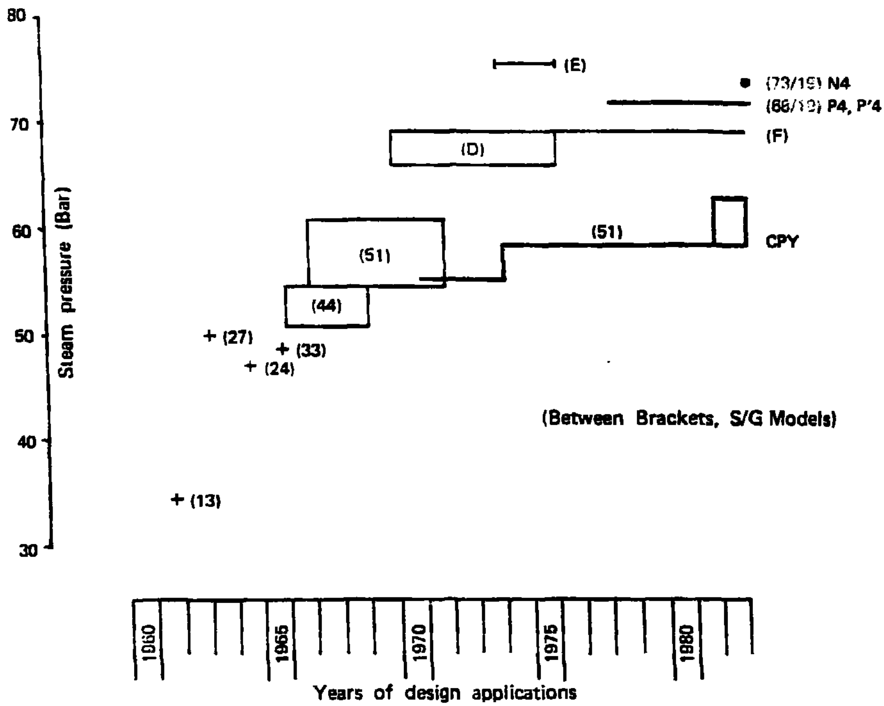
PWR OPERATING EXPERIENCE
750 UNITS X YEAR BY END OF 1982

Fig. 2



EVOLUTION OF STEAM GENERATOR SIZE

Fig. 3



EVOLUTION OF STEAM PRESSURE

Fig. 4

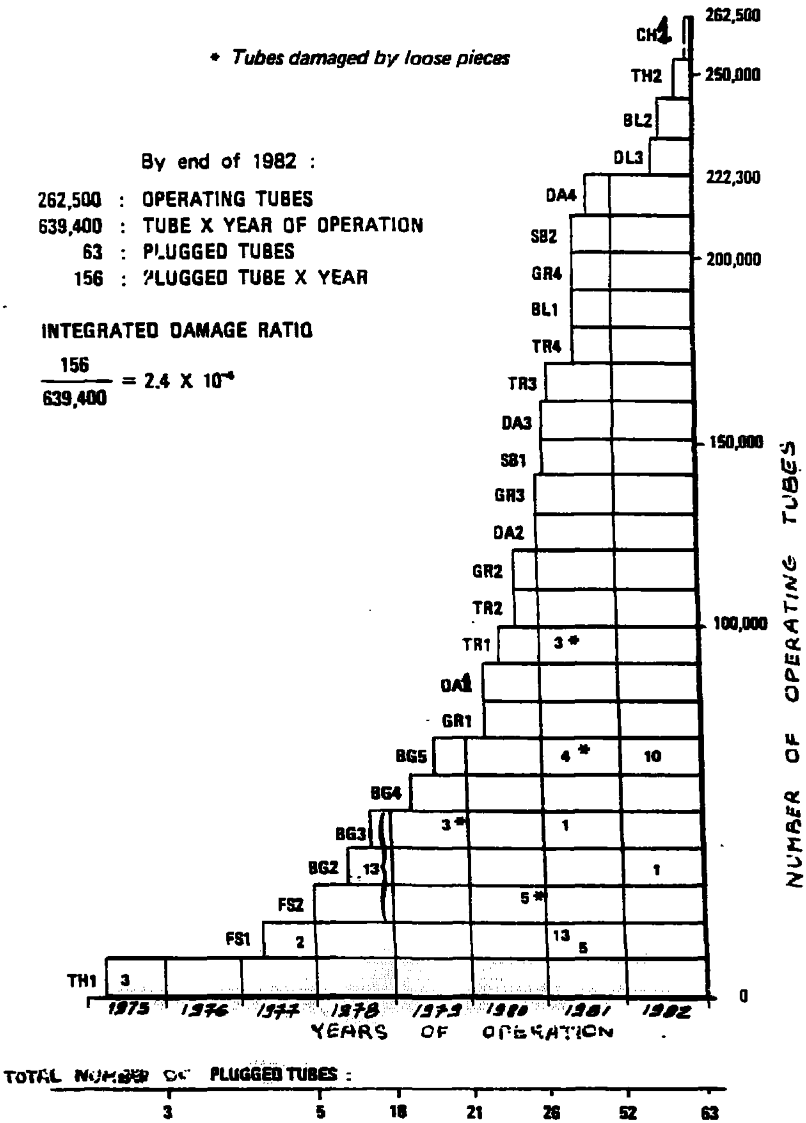
* *Tubes damaged by loose pieces*

By end of 1982 :

262,500 : OPERATING TUBES
639,400 : TUBE X YEAR OF OPERATION
63 : PLUGGED TUBES
156 : PLUGGED TUBE X YEAR

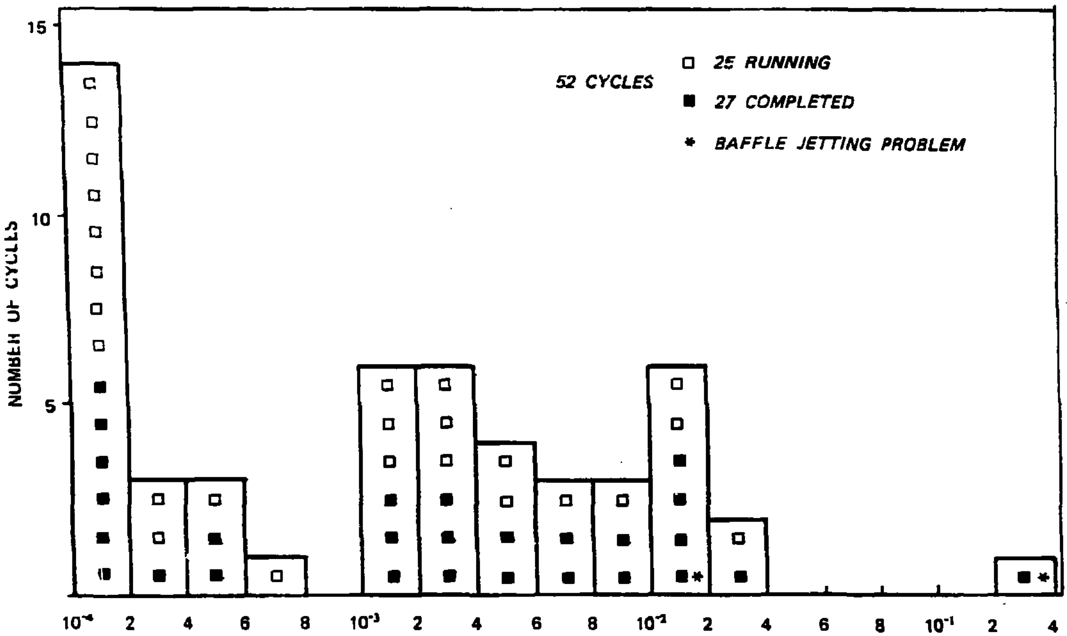
INTEGRATED DAMAGE RATIO

$$\frac{156}{639,400} = 2.4 \times 10^{-4}$$



FRAMATOME OPERATING EXPERIENCE WITH STEAM GENERATOR TUBE DAMAGE

Fig. 5



IODINE 131 ACTIVITY IN THE PRIMARY COOLANT (ci/t)
(Maximal level observed in normal operating conditions)

OPERATING EXPERIENCE WITH FRAGEMA 17 X 17 FUEL LOADINGS BY END OF DECEMBER 1982

Fig. 6

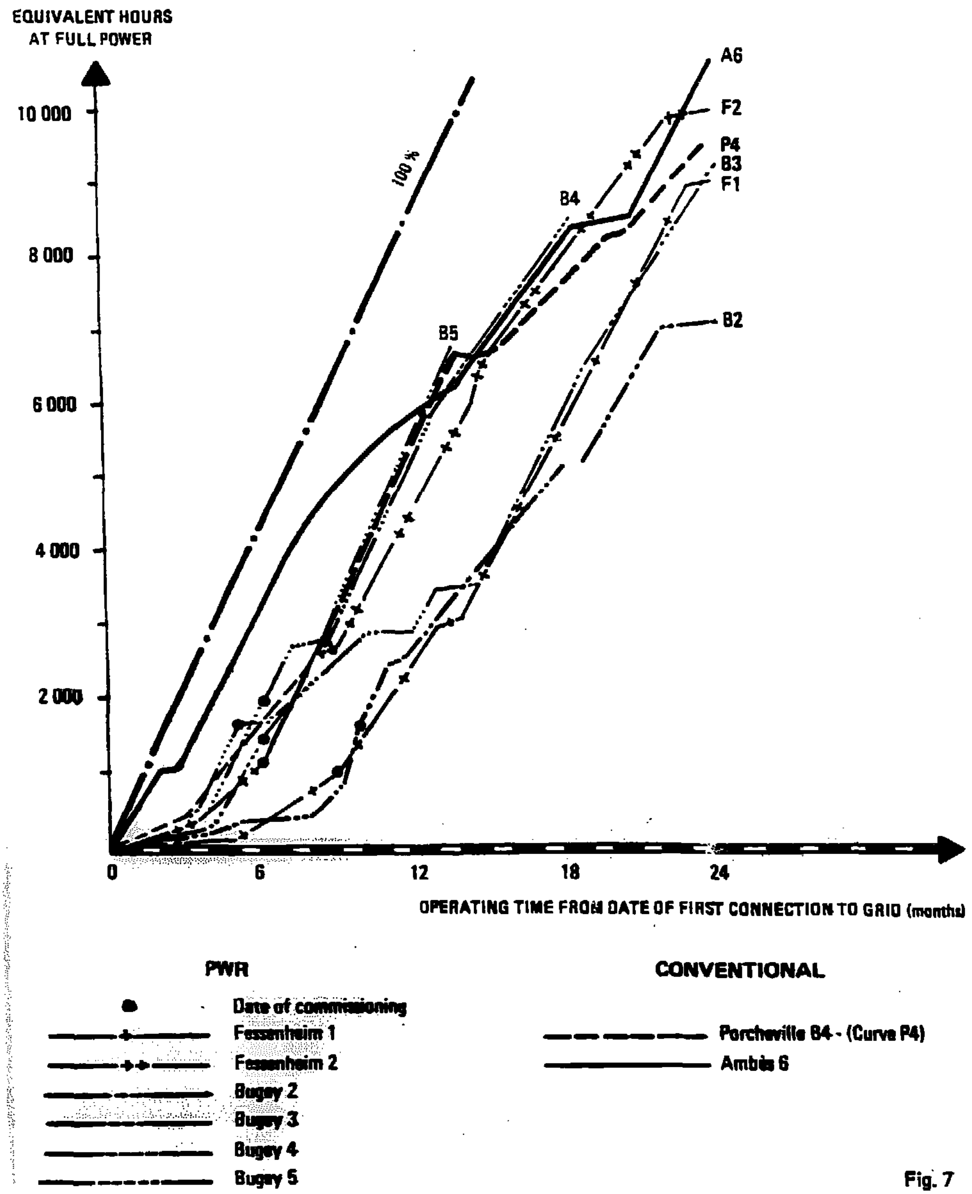

EQUIVALENT HOURS AT FULL POWER
10 000
8 000
6 000
4 000
2 000
100 %
A6
F2
P4
B3
F1
B4
B5
B2
0
6
12
18
24
OPERATING TIME FROM DATE OF FIRST CONNECTION TO GRID (months)
PWR
Date of commissioning
Fessenheim 1
Fessenheim 2
Bugey 2
Bugey 3
Bugey 4
Bugey 5
CONVENTIONAL
Porcheville B4 - (Curve P4)
Ambès 6

Fig. 7

R & D MAIN PROGRAMS

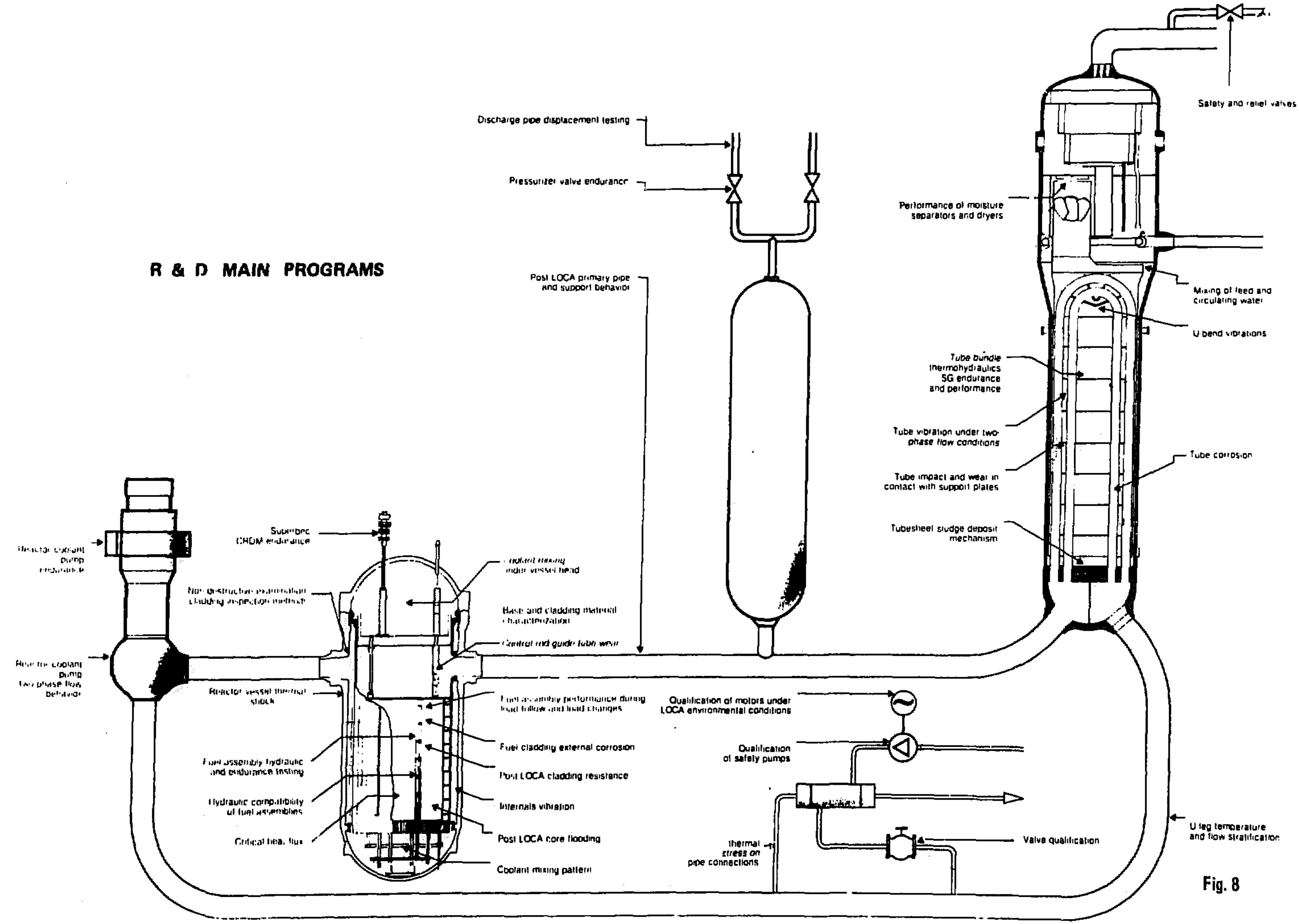
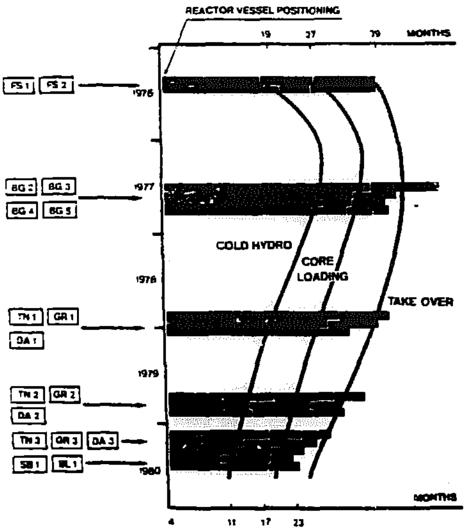
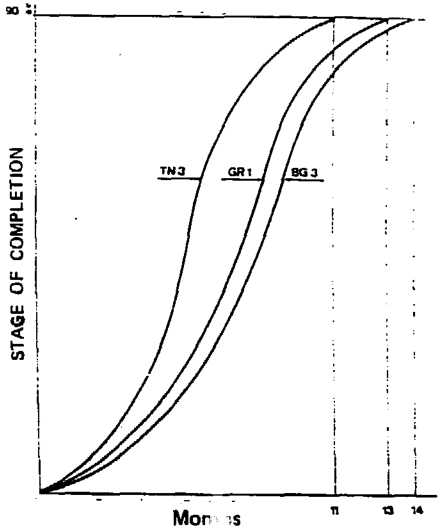


Fig. 8

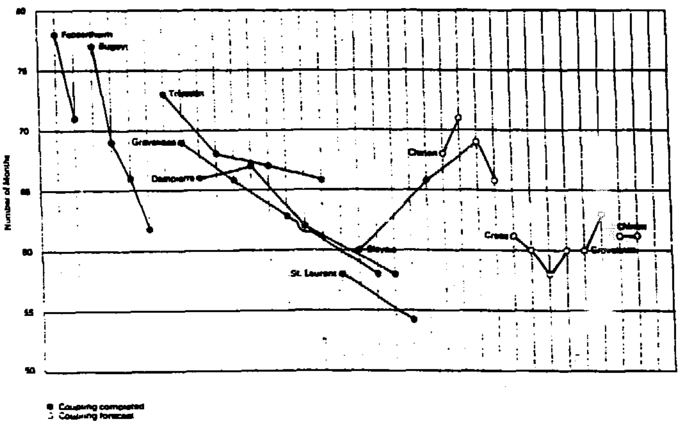




MAIN MILESTONES

AUXILIARY SYSTEMS

REDUCTION IN FIELD WORK SCHEDULE



OVERALL FROM ORDER TO COUPLING

REDUCTION IN CONSTRUCTION SCHEDULE

Fig. 9