HELICAL COIL STEAM GENERATOR FOR S N R 300.

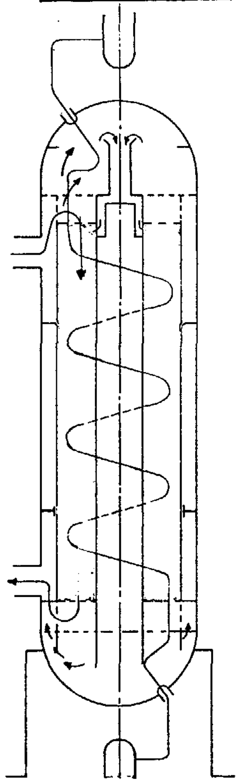

FIG 9

SNR HELICAL COIL DESIGN

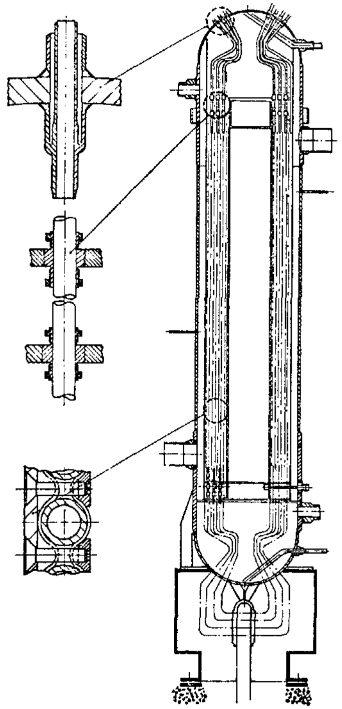

FIG 10

A.6. Some Engineering Aspects of the Steam Generator System for the United States' LMFBR Demonstration Plant

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ABSTRACT

This paper describes the main design features of the steam generator system for the Clinch River Breeder Reactor Plant and the engineering approach being employed for some of the critical elements of this system, including in particular the sodium-steam/water boundary, the efforts to have this boundary be of highest integrity, and the system features to safely accommodate any failure of the boundary.

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INTRODUCTION

The United States' Liquid Metal Fast Breeder Reactor Plant is to be located on the Clinch River near Oak Ridge, Tennessee, and is called the Clinch River Breeder Reactor Plant (CRBRP). Engineering of the CRBRP is proceeding with the goal of having the plant completed and delivering 350 MWe to the Tennessee Valley Authority power grid in the early 1980s.

The steam generator systems are major elements of the intermediate heat transport system (IHTS), which extends between the intermediate heat exchangers (IHX) on the sodium-side to the piping connections to the turbine system on the steam/water side. In the organization of work among the three participating reactor manufacturers for the CRBRP project (Atomics International, General Electric, and Westinghouse), the IHTS, including the steam generator systems, is under General Electric's scope of responsibility.

MAIN SYSTEM DESCRIPTION

The IHTS utilizes three parallel sodium loops to transfer heat from corresponding IHXs in the primary heat transport system through 24-inch (∿600 mm) diameter stainless steel piping to the steam generator systems. There are three independent and identical steam generator systems, one in each of the IHTS sodium loops (Figure 1); and each of these systems is housed with the major components of the associated IHTS loop in a separate cell of a "hardened" steam generator building (Figure 2). The steam generator systems accept flowing sodium at 936°F (503°C), cool it to 651°F (343°C), and in the process convert feedwater at 450°F (232°C) from the turbine system to superheated steam which is delivered to the turbine throttle at 900°F/1450 psig (482°C/99.7 atm). Each system is designed to transfer approximately 325 MWt at full power, with a load following range from 40 to 100 percent of full power.

The three steam generator systems each contain one steam drum and one water recirculation pump. At full power conditions, water is pumped from the steam drum at a recirculation ratio of 2:1 to the two evaporators, in which it is boiled to a two-phase mixture (50% by weight steam flow), and is then returned to the steam drum for separation of saturated steam which goes from the drum to the superheater. A portion of the saturated liquid water is drained continuously from the drum to the feedwater treatment system, for return to the drum as part of the feedwater flow after full-flow demineralization treatment.

Sufficient elevation difference between the IHX and the evaporators is provided to ensure adequate natural circulation head for reactor decay heat removal using any one of the three steam generator systems. An auxiliary heat removal system is connected to the steam drum in each of the steam generator systems for the purpose of removing reactor decay heat during shutdown.

EVAPORATOR/SUPERHEATER COMPONENTS

The evaporators and superheaters (Figure 3) are installed in a vertical position and are shell and tube heat exchangers, with fixed tube sheets and with a 90° bend in the shell and tube bundle ("hockey stick" configuration) to provide for differential thermal expansion between the tubes and between the tube bundle and the shell. Sodium flow is vertically downward, parallel with the tubes, on the shell side from the sodium inlet nozzle near the top of the active straight section, counter-current to the steam/water flow upward inside the tubes. The evaporators and superheaters are essentially identical in construction. There are 757 tubes, 5/8-inch (15.9 mm) OD with 0.109-inch (2.8 mm) minimum wall thickness, in each unit. There are only two welds in each tube, one at each end for joining to the tube sheet; and these will be full-penetration butt welds to machined bosses on the tube sheets, and will be fully radiographed.

This particular design for the evaporators and superheaters, employing single-wall tubes with fully radiographed butt-welds at the tube sheets and no intermediate welds in the tubing, was selected as the best available design approach for ensuring high integrity of the sodium-steam/water boundary, with acceptable economics and within the schedular constraints of timing for development required for the CRBRP application. Development of the hockey stick concept for sodium-heated steam generator application has been underway by Atomics International since the 1960s, and a recent test (over 4000 hours steaming) and post-test examination of a 30 MWt model (158 tubes) employing tubes almost identical to those for the CRBRP evaporator/superheater components affirm the basic feasibility and soundness of the design concept (Refs. 1,2,5).

The material of construction for the shell, tube sheets, tubes, and other elements of the evaporators and superheaters is 2 1/4Cr-1Mo (ASME T-22). The materials properties used for design, such as the allowable stress values, include conservative allowances for the effects of carbon removal from 2 1/4Cr-1Mo into the sodium over the 30-year design life of the units. Additionally, the minimum tube wall thickness (0.109-inch, 2.8mm) conservatively allows for sodium side corrosion, water side corrosion, and material removal during periodic cleaning of the units. In order to minimize the chance of any significant flaws in the tubing and in order to ensure highest quality welds for the tube/tubesheet joints, very high purity material is planned for the tubing and tube sheets--to be obtained by utilizing either the electro-slag remelt process or the vacuum arc remelt process for the material preparation. Reference 3 describes the main factors on which the material selection was based and the materials development work underway for the CRBRP evaporator/superheater components.

Preparations are underway for a full-scale test of the shell-side hydraulic characteristics of the evaporator/superheater design using a full-diameter model with water at flow rates up to about 30,000 GPM (6,800 m3/hr) to simulate the sodium flow. This test employing a facility at Rockwell International is scheduled to begin by mid-1975 and is intended to provide final design bases for the flow baffles, measurement of the shell-side flow distribution, and confirmation that no damaging tube vibrations exist.

A long-term endurance test of few-tube models of the evaporator/superheater, employing prototypical tubes, tube/tube sheet weld-joints and tube support geometries, at sodium and steam/water conditions representative of CRBRP is in the initial phases of planning and test equipment design for planned start of testing the first of 1976 using a facility at General Electric. This test is intended to provide thermal performance data for the specific conditions of CRBRP operation, including verification of the adequacy of the tube inlet orifices in the evaporator to prevent hydraulic instability, to verify absence of excessive tube wear at the tube supports, and to provide long-term endurance data for the tubes and weld-joints at representative CRBRP sodium and steam/water conditions (including temperature cycling).

Preparations are going forth for a test of a full-size prototype of the CRBRP evaporator/superheater using a facility to be prepared for this purpose (Sodium Components Test Facility) at the US Atomic Energy Commission's Liquid Metal Engineering Center. This test is planned to be conducted in 1978. The test will provide performance data for the full size unit at partial powers up to 70 MWt and, through a series of planned transient tests, will provide a final verification of the structural integrity of the prototype.

WATER CONDITIONS AND HEAT TRANSFER IN THE EVAPORATOR

One of the project guidelines for the CRBRP is that a water recirculation type of evaporator system should be used. This type of system, with the steam drum employed, provides some advantageous features, including: ease of control of the feedwater supply; mitigation of cold leg sodium transients during rapid power reductions; and stored water at near saturation temperature in the steam drum for immediate use for reactor decay heat removal through the steam generator auxiliary heat removal system.

An additional advantage available with the recirculation type of system is that departure from nucleate boiling (DNB), with its attendant uncertainties of tube wall surface temperature fluctuations, potential for accelerated corrosion in the DNB zone, and reduced heat transfer effectiveness, can be avoided if the recirculation ratio is high enough. It was determined, however, based on the available DNB data, that for the reference evaporator design a large water recirculation ratio (more than 6:1) would be required in order to ensure absence of DNB; and that substantial savings in capital costs and pumping power costs could be obtained by using a smaller recirculation ratio. Furthermore, preliminary calculations indicated that the tube wall surface temperature fluctuations associated with DNB in the reference evaporator are tolerable in relation to estimated strain cycle fatigue limits for the tube walls; and initial results reported from the post-test examination of tubes from the AI 30 MWt model, which was operated for more than 4000 hours steaming at DNB conditions, indicated no disturbance of the oxide scale in the DNB zone (Ref. 2).

Based on the estimated substantial cost incentive to use a low recirculation ratio and the preliminary judgment that DNB can be feasibly accommodated, a nominal water recirculation ratio of 2:1 was selected, corresponding to 50% by weight exit steam quality from the evaporators at full power. Because of the existence of DNB in the evaporator tubes, full-flow demineralization treatment of the feedwater, together with continuous partial drain (blowdown) from the steam drum to the feedwater treatment system, is planned in order to minimize the buildup of corrosive contaminants in the recirculating water and to thereby avoid the likelihood of accelerated corrosion in the DNB zone of the evaporator tubes. The current approach (Ref. 3) is aimed at maintaining less than 2 ppm total dissolved solids, less than 80 ppb corrodent (sodium), and a pH of 9.0-9.5 for the water entering the evaporator.

As an alternative to the present scheme, some consideration is being given to reducing the recirculation ratio further, to approach once-through evaporator operating conditions if needed to maintain adequately low concentration of contaminants in the evaporator inlet water. A detailed test program is being implemented to obtain the necessary data required to fully confirm the feasibility or provide the basis for correction of the water recirculation ratio, the drain rate from the steam drum and the water quality for the CRBRP system. This test program includes: a sodium-heated single-tube boiling heat transfer test at Argonne National Laboratory to obtain detailed data for predicting the DNB point, the DNB wall temperature fluctuations, and the heat transfer coefficients in the post-DNB boiling zone (test startup in early 1975); both short-term and long-term tests at General Electric to determine any evidence of strain-cycle fatigue or accelerated corrosion at the most severe conditions of DNB surface temperature fluctuations and off-normal water quality expected for the CRBRP evaporators (test startup about mid-1975); and determination of any long-term corrosion effects at representative CRBRP water quality conditions during the few-tube model endurance test outlined in the preceding section.

LEAK DETECTION

A main element of the sodium/water reaction leak detection system is a hydrogen detector of the diffusion membrane type (nickel membrane). The hydrogen detector is being developed by Argonne National Laboratory (ANL); and prototypes of the detectors planned to be used will be tested for performance and reliability, beginning in early 1975, using a test rig being prepared for this purpose at ANL. In order to provide supplementary monitoring during periods of operation when the background of hydrogen in sodium may be too high to allow complete reliance on the hydrogen detectors, oxygen meters of the electro-chemical cell type are also planned to be employed in the CRBRP system.

The sodium is monitored for hydrogen at the exit from each superheater and evaporator and in the cold leg of the main sodium piping (Figure 1). There is a small bypass sodium flow from the sodium inlet through the horizontal expansion leg of each evaporator/superheater which is also monitored for possible leaks in the expansion leg by diverting this bypass flow to the detectors at the sodium exit (not shown in Figure 1). Additionally, the cover gas in the main sodium pump tank is monitored for hydrogen.

With this system, using the ANL hydrogen detectors, or their equivalent, it is estimated that leaks as small as about 10^{-5} lbs/sec (~0.005 grams/sec) water flow into sodium can be detected. Based on the available sodium/water reaction wastage data (cf., Ref. 4), it is planned that promptly after detection of a leak the operator action will be to perform a controlled shutdown, including de-pressurizing the steam/water and draining the sodium from the affected steam generator, in order to minimize possible growth (self-wastage) of the initial leak and sodium/water reaction wastage damage to adjacent tubes. Experimental work is in progress to determine more completely the growth and self-plugging characteristics of small leaks, and in particular to obtain test data needed to establish the shutdown procedures required in order to keep a small leak open long enough to identify the leaky tube. Once the leaky tube is found, it will be plugged at the tube sheets, and adjacent tubes will be inspected for sodium/water reaction wastage damage (internal bore inspection method), preparatory to returning the unit to service.

SODIUM/WATER REACTION PRESSURE RELIEF

It is estimated that leaks in the order of about 0.1 lb/sec (~50 grams/sec) water into sodium can cause a rise in the sodium pressure of the affected loop which is sufficiently rapid that the automatic pressure relief devices (rupture discs) may actuate before preventative action can be completed by the operator. Consideration is being given to using a rupture disc at the cover gas space of the sodium expansion tank (Figure 1) to provide pressure relief through a vent line to the sodium dump tank. It is estimated that with this arrangement adequate pressure relief can be provided to accommodate leaks up to about 2 lbs/sec (~1 kg/sec), or larger if the results from planned tests of leak propagation characteristics indicate it to be desirable.

Leaks sufficiently large that the pressure rise is not halted by operator action, nor by the pressure relief from the sodium expansion tank gas space, will result in bursting of one or more of the three rupture disc assemblies in the main sodium piping immediately adjacent to the nozzles of the evaporators and superheaters (Figure 1). The sodium and the sodium/water reaction products in the affected loop will be expelled through 24-inch (~600 mm) diameter piping to reaction products tanks located below the evaporator/superheater components (Figure 2). Gross separation of liquid, solid, and gaseous products takes place in the reaction products tanks, which are vented to a centrifugal separator located at a higher elevation. Further separation of caustic material is done in the centrifugal separator, after which the gaseous products are vented to atmosphere through a stack which contains apparatus to ignite and burn the discharging hydrogen and to prevent back-flow of air into the system.

Each main rupture disc assembly contains two rupture discs in series with a sodium leak detector between them. Bursting of the discs activates a sensor located downstream in the pressure relief pipe, which automatically causes steam/water isolation valves to close and rapid blowdown of water from the affected steam generator system to be initiated, in order to terminate as rapidly as possible the steam/water flow into the sodium (in the order of about 30 seconds blowdown time planned). When the pressure has been reduced to a predetermined level, nitrogen gas is admitted automatically to the steam/water side, which will be maintained at a pressure above the shell-side pressure until a sodium dump can be accomplished, in order to minimize entry of sodium into the tubes. The sodium dump is initiated by operator action.

Design of the sodium/water reaction pressure relief system is proceeding on an approach of having the system capable of accommodating an incident of gross failure of several tubes without excessive pressures occurring in the IHTS piping and components, nor in the IHX. A typical incident being studied as representative of an upper limit on the extent of tube failure propagation, for example, assumes instantaneous guillotine failure of a tube at the estimated worst location (in the evaporator), followed immediately by failure of a number of nearby tubes such that altogether the water flow rate into sodium is about 90 lbs/sec (~40 kg/sec). For this case, calculations using a computer code originated by Atomics International (under development for CRBRP design use as TRANSWRAP II), indicate that : the maximum water flow rate from the guillotined tube (assumed double-ended) is about 13 lbs/sec (5.9 kg/sec); the peak pressure at the IHX (about 0.4 sec. after initiation) is about 400 psi (~27 atm); and the peak pressure in the failed evaporator is about 1500 psi (~100 atm). These estimated pressures are below the ASME Code allowable over-pressures for "emergency" and "faulted" conditions in the IHX and the evaporator/superheater components, respectively.

A large leak test rig (LLTR) is being prepared at the Liquid Metal Engineering Center, in which a series of tests will be performed to confirm or provide data for correction of the design-basis pressure calculations. Additionally, from these tests it is intended to obtain more adequate data than now exist regarding the tendency for gross failures of additional tubes as a consequence of an initial tube break. For this latter purpose, all the tubes in the test model will be filled with steam/water at nominal CRBRP steam generator pressures and the tubes will be connected to a pressurized water reservoir, so that the conditions for tube failure propagation are reasonably representative of those in the plant units. Preparation of test equipment is proceeding for start of the LLTR tests by mid-1975, using for the first test article the 158-tube AI steam generator model, which has been modified for this purpose and incorporates a guillotine leak injector device. Planning and preliminary equipment design is underway for later tests (beginning late 1976) using a test article that will represent the full 757-tube cross-section of the CRBRP evaporator/superheater design.

TIMING OF THE WORK

The main development and test work for the CRBRP steam generator system is in various stages of implementation, under USAEC sponsored programs, according to the general timing illustrated by Figure 4. Preparations are in progress for ordering critical long lead-time materials and for selection by the end of 1974, of the contractor to prepare the detailed design and manufacture the full-size prototype and the plant units. These preparations are oriented toward a target of delivery of the prototype to the test site in early 1977 and delivery of the plant units to the CRBRP site in 1978.

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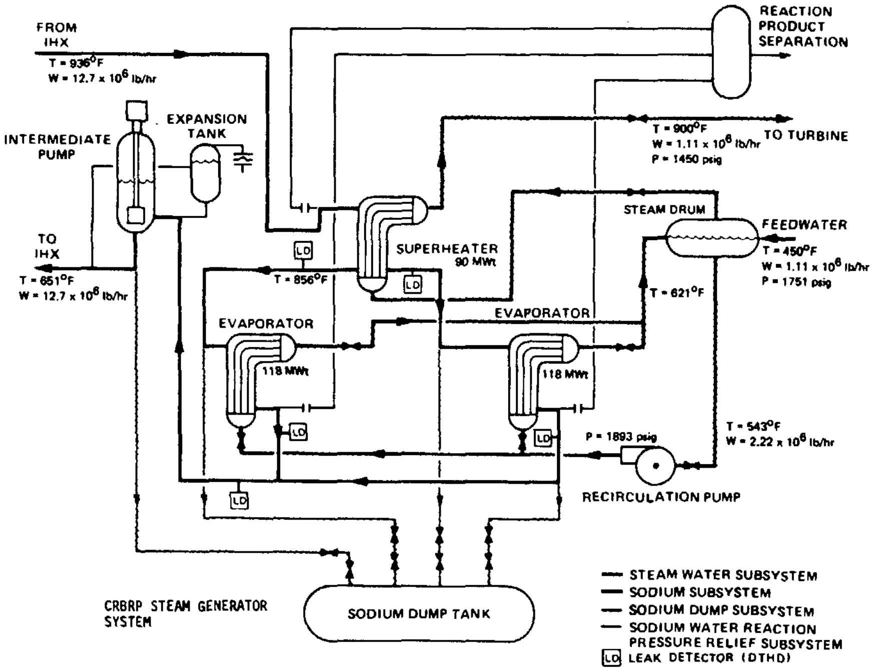
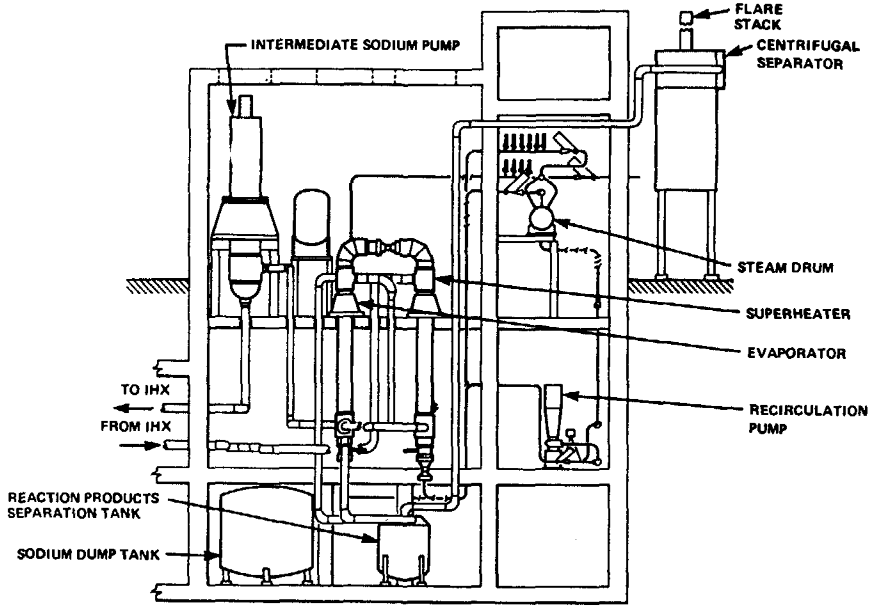


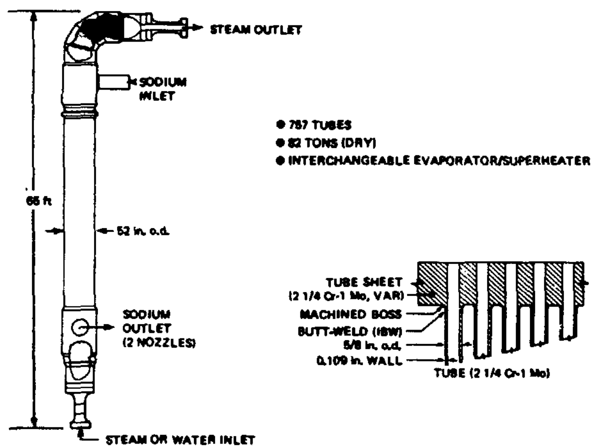
CRBRP STEAM GENERATOR SYSTEM

FIG. 1.



CRBRP STEAM GENERATOR BUILDING ARRANGEMENT

FIG. 2.



CRBRP EVAPORATOR/SUPERHEATER (AI HOCKEY-STICK DESIGN)

FIG. 3.

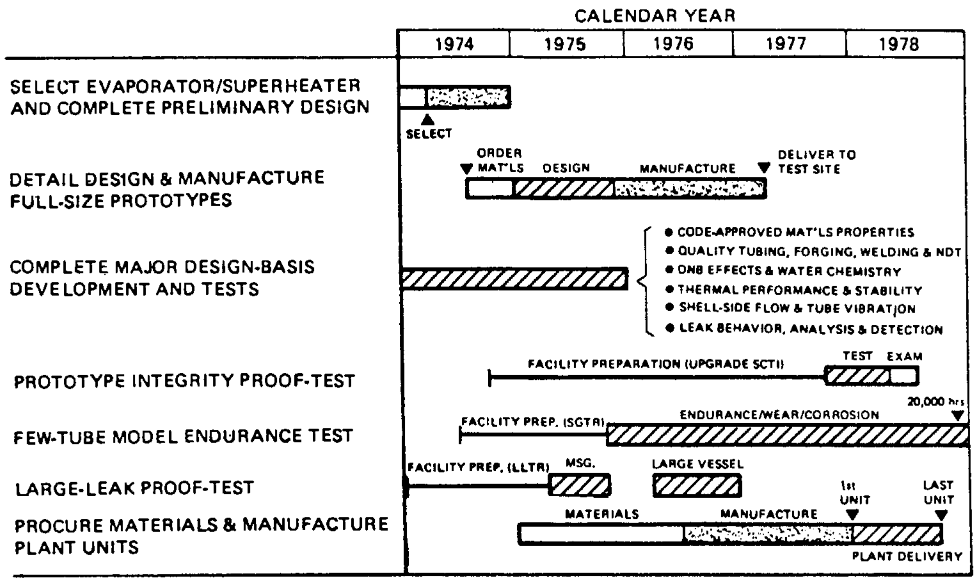


FIG. 4. CRBRP STEAM GENERATOR SCHEDULE

A.7. Design Phylosophy and Criteria for NIRA straight Tubes Steam Generator

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ABSTRACT

The criteria why the NIRA straight tubes steam generator has been designed are analyzed.

Montainance and repair procedures are descibed.

1. PROBLEMS INVOLVING SODIUM HEATED STEAM GENERATORS DESIGN

In a liquid metal fast breeder reactor, the steam generator is one of the most critical components;in fact it involves both the development of special technologies and the safety problems of the component itself and of the whole plant.

The most important requirements for a LMFBR steam generator are :

- Realiability and safety
- Low cost per unit
- Low cost of operation
- Wide range of operation
- Easy transport

1.1 Reliability and Safety

In a LMFBR's steam generator, damages and outages probability must be very low and however if a large accident occurs (sodium water reaction) the integrity of the system's container must be assured to avoid harm to people or to the neighbouring equipment.

1.2. Low cost for unit

This can be reached by a simple and light design and using as much as possible standard technologies and repeated solutions.
It means that all the parts might be easy to build and easy to assemble.

1.3. Low cost of operation

The steam generator must produce low pressure drops but essentially it must be easy to be repaired or, in any case , the system must be forseen to minimize the outage time.

1.4. Wide range of operation

Steam generator can be coupled to different power plants with different thermal cycles. Modularity is a big advantage.

1.5. Easy transport

Transport may, in many cases, involve a large part of the unit cost. Units, which are difficult to be transported, need assembling in site, this, involves a cost increase owing to the lacking of specialized equipment.
Modules must be designed to allow an easy transport by railway or road.

2. CHOICE OF STEAM GENERATOR CONCEPT

N.I.R.A. components division tackled the problem of the choice of steam generator concept taking into account all the above said considerations.