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ABSTRACT

The development of steam generators for the LMFBR was recognized from the onset by the AEC, now DOE, as a difficult, challenging, and high-priority task. The highly reactive nature of sodium with water/steam requires that the sodium-water/steam boundaries of LMFBR steam generators possess a degree of leak-tightness reliability not normally attempted on a commercial scale. In addition, the LMFBR steam generator is subjected to high fluid temperatures and severe thermal transients. These requirements place great demand on materials, fabrication processes, and inspection methods; and even greater demands on the designer to provide steam generators that can meet these demanding requirements, be fabricated without unreasonable shop requirements, and tolerate off-normal effects.

DOE has under development three types of LMFBR steam generators--a single wall, hockeystick, recirculation type unit for the CRBR Project (discussed in a companion paper for this session) with separate evaporator and superheater; and two base-program units. The latter are a helical-coil, single-wall-tube unit being developed by the Babcock & Wilcox Co., and a straight, double-wall-tube unit being developed by Westinghouse Nuclear Components Division. Each of these two base program units is a once-through steam generator. The base-program steam-generator-development activities are managed technically for DOE by Argonne National Laboratory's LMFBR Components Development Management (LCDM) Project.

In this paper, the scope and status of DOE's base program steam generator activities are discussed, and the designs of the helical-coil and double-wall-tube units are described briefly. Important issues that confront the designer are discussed, the various approaches in use to resolve these issues are identified, and the general status of the base technology supporting these matters is summarized.

INTRODUCTION

The development of the liquid-metal-heated steam generator for nuclear application in the U.S. originated in the late 1940s to early 1950s. From the onset it was recognized that these steam generators would require a degree of leak-tightness and reliability of the sodium-water/steam boundary that had not been attempted on a commercial scale in the power industry. Consequently, many early designs employed a double barrier between the heat transport fluids. Experience with these early steam generator units has been reported by various authors, such as J. S. McDonald,(1) and was summarized recently by C. C. Stone et.al.(2) As the development of the LMFBR spread internationally and intensified, the pace of steam generator development programs quickened. During the 1960s and 1970s, new concepts were developed in support of CRBR, Phenix, Super Phenix, PFR, KNK, SNR-300, BN-350, BN-600, and Monju. Several of these steam generators have seen extensive service with varying degrees of success.

In the late 1970s DOE initiated the development of two new steam generator concepts for LMFBR application. Babcock & Wilcox was selected to produce a helical-coil unit and Westinghouse Nuclear Components Division a straight shell, double-wall-tube concept. These two activities are ongoing and are managed technically for DOE by Argonne National Laboratory's LMFBR Components Development Management Project (LCDM).

OPERATING LMFBR STEAM GENERATORS

Currently there are seven operating LMFBRs that have steam generators: EBR-II, Phenix, PFR, KNK, BOR-60, BN-350 and BN-600. Table 1 lists relevant information for these seven steam generators, including an estimate of operating hours through the end of 1983. Table 2 summarizes the key problems encountered in most of the facilities including Fermi, which is no longer operating.

The degree of operational reliability achieved with the steam generator units for these plants has varied widely. EBR-II, which employs a straight shell, double-wall-tube concept,[3] has operated more than 19 years--approximately 90,000 full power hours--with no significant problems. (A minor steam-to-air leak during initial operation was readily repaired, and performance of one of the superheaters deteriorated and the unit was replaced during the normal yearly outage in April 1981.)

In France the Phenix facility uses a steam generator configuration of many hairpin-shaped shells, each containing seven tubes. Operating experience was excellent for about eight years. In Spring 1982 a steam-to-sodium leak occurred in a reheater module. This was followed by two additional leaks over the next 12-13 months.[4] The facility has been able to recover with minimal impact on plant operations, owing to the modular construction of the steam generator, the ordering of spare reheater modules prior to the first leak (distortion of some tube support elements had been discovered by x-ray examination during a prior routine shutdown), and the capability to operate the plant on two of its three loops.

The UK's PFR employs a U-tube configuration, single-wall-tube concept that, from the very onset, has had serious and continuing problems with

Table 1. Currently Operating LMFBR Steam Generators

Plant	Steam Cycle	Steam Conditions	Configuration	Material*	Estimated Operating Time, hr**	Startup Date
EBR-II	Recirculation w. superheat	820°F (438°C) 1265 psia (87 bar)	Straight shell & tube, double-wall tubes	E 2-1/4 Cr-1 Mo S 2-1/4 Cr-1 Mo	90,000	Dec. 1962
Phenix	Once-through w. reheat	961°F (516°C) 2364 psia (163 bar)	Hairpin shell & tubes, single-wall tubes	E 2-1/4 Cr-1 Mo S 321 SS R 321 SS	50,000	July 1974
PFR	Recirculation w. superheat & reheat	957°F (514°C) 2400 psia (166 bar)	Straight shell, U-tube, single-wall tubes	E 2-1/4 Cr-1 Mo S 316 SS R 316 SS	5700 (>100,000 steaming)	Late 1974
KNK	Once-through	905°F (485°C) 1176 psia (81 bar)	Hairpin shell & tube, one tube per shell, single-wall tubes	10 Cr Mo Ni Nb 910	>10,000	Aug. 1972
BOR-60	------------------------------ Insufficient information available ------------------------------					
BN-350	Recirculation w. superheat	815°F (435°C) 735 psia (50 bar)	Straight shell, bayonet tubes (evap.), single-wall tubes	E 2-1/4 Cr-1 Mo S SS	>60,000 (est.)	May 1983
BN-600	Recirculation w. superheat & reheat	932°F (500°C) 2030 psia (140 bar)	Straight shell & tube, single-wall tubes	E 2-1/4 Cr-1 Mo S SS R SS	>10,000 (est.)	Apr. 1980

*E - evaporator, S - superheater, R - reheater.

**Equivalent full-power hours

Table 2. Summary of Key LMFBR Steam Generator Problems

Plant	Service Date	Problem
Fermi	1962-1968	Numerous leaks caused by stress corrosion cracking, leaky T/TS welds, and flow-induced vibration (caused by major leak).
EBR-II	1964	One water-air leak in evaporator T/TS weld; repaired <u>in</u> <u>situ</u>.
		Anomalous superheater performance; unit removed from service.
BN-350 (Original Units)	1973	Defective tube welds and end cap material leading to numerous water leaks into sodium.
		All units retubed.
		One unit removed from service due to major leak after retubing.
		Modified units appear to give adequate service.
Phenix	1973	Water-to-air leakage at feedwater inlet lines caused by erosion/corrosion downstream of inlet orifice.
		Distortion of tube supports in reheaters caused tube distortion.
		Three water-sodium leaks in reheaters since March 1982; reheaters being replaced.
PFR	1974	Numerous leaks, mainly at evaporator T/TS welds due to absence of post-weld heat treatment.
		Stress corrosion cracking in reheater tube sheet; unit removed from service.
		Stress corrosion cracking in ferritic evaporators reported.
BN-600	1980	Two cases of leakage--no details released; units removed from service.

small leaks. An extensive program to resolve these difficulties has been ongoing for many years, ranging from local repairs and sleeving to—in one instance—permanent removal of the unit from service. The continuing incidence of leaks is reportedly attributable to the aftermath of a small number of early leaks caused by hard, highly stressed tube welds resulting from a lack of post-weld heat treatment.[5] Consequently, the yearly plant factor for PFR has been low—in the range of 10% or less.

The KNK steam generator consists of a number of units, each a single-wall tube within an outer tube with sodium in the annulus that operates as a once-through unit. Early in generator life, a manufacturing fault led to a leak.[6] The unit was repaired and returned to service.

BOR-60 is used to test 30 MWt steam generator models for the USSR LMFBR program. It has tested a unit of Czechoslavakian design and a prototype unit for BN-600. No leakage has been reported, but some leak tests have been performed.

BN-350 has had considerable difficulty with its original steam generator equipment, which has natural-circulation evaporators employing single-wall, bayonet-type tubes. Leaks, some of sizeable magnitude, occurred in five of six steam generator complexes. In at least one instance an evaporator experienced more than one large leak and was retired from service. These problems were attributed to the low quality of tube welds, the poor material of the end caps of the bayonet tubes, and inadequate leak detection instrumentation to mitigate the consequences. Each of the units that leaked was retubed and has subsequently performed sufficiently well to allow a satisfactory plant factor.[7] In at least one loop the original steam generator has been replaced recently by a unit of Czechoslovakian design (previously tested at BOR-60).

BN-600, which has been in operation since 1980, uses a highly modularized steam generator system of straight shell and tube design with eight evaporators, eight superheaters, and eight reheaters per loop and with three loops per plant. Little operating information is available, but, it is believed that small leaks have occurred. However, it also is believed that facility operation has not been unduly hampered, due to the modular nature of the system.

Clearly the spectrum of experience is wide, ranging from essentially trouble-free operation of the EBR-II units, to the severe problems experienced with the continuing evolvement of small leaks in PFR and to the spate of small and large leaks in the BN-350 original equipment.

In summary, a few LMFBR steam generators are acquiring significant service history (see Table 1.) Those that have had no leaks or a few infrequent, relatively minor leaks have given excellent service; those with large leaks or frequently occurring small leaks have encountered extensive repair or replacement with severe impact on plant performance in some cases.

IMPORTANT STEAM GENERATOR DESIGN CONSIDERATIONS

The design of an LMFBR steam generator must assure a high-integrity water/steam-sodium boundary; operation with leakage is intolerable due to the tendency of small leaks to escalate rapidly into large ones. Boundary integrity must be the overriding concern throughout the design process, including concept selection, design feature development, and determination of suitable fabrication and QA/QC processes. Failure to avoid leaks inevitably leads to unacceptable plant downtime and costs. But where leaking has been avoided, LMFBR steam generators operate reliably, with satisfactory availability and maintainability.

The key factors affecting the integrity of the sodium-water boundary that have been evident to date are listed in Table 3. It is of interest to note that few significant difficulties have been reported concerning the kinds of problems that have affected PWR steam generators. Moreover, steam generators in at least three LMFBR plants (EBR-II, Phenix, and BN-350) have acquired significant operating times (up to 90,000 hours) without the appearance of sodium-side corrosion or unacceptable waterside corrosion.

Experience shows that the most critical portions of the sodium-water boundary are tube welds. Tube-to-tubesheet joining undoubtedly is crucial, particularly in the single-wall-tube concepts. Each steam generator unit employs thousands of tube joints, each of which must be of unquestionable quality. For example, very small leaks (10^{-3} to 10^{-5} lb/sec) that would be of little concern in PWR or fossil-fired units are a major problem to the LMFBR, because small leaks can propagate into large leaks within minutes. This high degree of weld integrity requires a total steam generator design dedicated to the delivery of highly reliable welds. Moreover, the tube welds must be suitable for commercial joining processes, forgiving of day-to-day shop floor environments, and capable of final quality confirmation. These factors are especially important where the joining process is a single or few-pass weld, or similar continuous, single-barrier process.

Although sophisticated welding equipment is available with programmed power supplies and microfocus x-ray equipment, consistent production of high-quality welds has proven difficult. The microfocus x-ray permits the detection of very small voids--2-3 mil in tube wall thicknesses of interest. This vastly improved inspection capability, coupled with the

Table 3. Key Factors Affecting Integrity of Sodium-Water Boundary of LMFBR Steam Generators

Factor	Manifested	Little Evidence
Weld integrity	X	
Absence of post-weld heat treatment of ferritic welds	X	
Stress corrosion cracking	X	
Flow-induced vibration	X	
Differential expansion	X	
Feedwater inlet orifice erosion/corrosion	X	
Material/manufacturing defects	X	
Sodium-side corrosion, general		X
Water/steam-side corrosion, general		X
Tubing integrity		X
Fretting and galling		X
Flow distribution		X
Hydraulic instability		X
Thermal stress		X

intense desire for very high-quality leak-proof welds, inevitably imposes great demands on weld processes and shop personnel and production procedures. Processes that work well under laboratory conditions sometimes have proven difficult to transfer to the production shop. Cleanliness is extremely important in welding 2-1/4 Cr-1 Mo tubing and can be difficult to achieve consistently in a commercial shop environment. In some instances close tolerances on weld contour have created additional difficulties. Proper heat treatment is critical. Failure to achieve it can lead to hard, highly stressed welds that will be susceptible to stress corrosion cracking, allowing problems like those encountered at PFR to occur. Thus, even with improvements in automated welding techniques and equipment and the introduction of sophisticated NDE equipment, it remains to be demonstrated whether pitfalls will continue to be encountered and whether a suitably high level of integrity of the sodium-water/steam boundary can be sustained in a commercial environment within reasonable costs.

A satisfactory post-weld heat treatment is essential. This has been demonstrated vividly by the UK experience, where failure to provide post-weld heat treatment led to stress corrosion cracking. Steam generator concepts that employ small tube sheets may heat-treat the entire tube sheet as a unit. Designs that employ large tube sheets, or none at all, usually must heat-treat each tube individually. To do so requires adequate design provisions to avoid large thermal gradients in the tube-to-tube sheet or tube-to-shell attachment region.

Large tube sheet designs in the U.S. employ forged tube sheets with machined tube attachment bosses. The bosses must be of sufficient length for making the tube weld and for post-weld heat treatment, and the material of the bosses must be equal to the tube quality or better.

Therefore, the material quality in the subsurface region (first 1-2 in.) of the tubesheet forging is critical. The use of remelt forging stock provides a good material base, but the propensity of large forging to have microfissures in the central subsurface region is a concern; the larger the forging the greater the concern. This is an area where further work is needed to ensure tubing-quality material throughout the machined surface areas of the tube sheet.

Stress corrosion cracking is another serious design concern, particularly for steam generators using stainless steel. Many early steam generator designs employed austenitic material, especially in the superheater, for compatibility with sodium and for high-temperature strength. Numerous incidents have occurred with these units. For example, an early U.S. test article experienced such severe cracking that all testing was terminated. Similarly, one of the PFR rehaeters was damaged so badly that it was removed from service. More recent designs tend to avoid the use of austenitic material in favor of ferritic material, mainly 2-1/4 Cr-1 Mo or 9 Cr. The French have opted for Incoloy 800 for the Super Phenix units to obtain improved resistance to stress corrosion cracking, improved resistance to wastage in the event of a small water/steam leak, and good high-temperature strength. Current U.S. designs employ 2-1/4 Cr-1 Mo as the material of choice due to its good compatibility with sodium and water and resistance to stress corrosion cracking. Use of the more resistant ferritic materials does not assure complete relief from this problem, as shown by the problems that continue to develop in the PFR evaporators, which are 2-1/4 Cr-1 Mo.

Flow-induced vibration, particularly of the tubes, must be considered carefully during the design process; otherwise, excessive tube vibration could lead to severely shortened tube life and a major sodium-water leakage incident. Major areas of design interest are the inlet and outlet shell-side cross-flow regions and the regions where there are tube bends to accommodate tube-to-tube differential expansion. A satisfactory design requires careful balancing of fluid forces and flow-induced structural responses. This, in turn, requires balancing of numerous conflicting design considerations. For example, flow-induced vibration is influenced by, and in turn influences, considerations of cross-flow area and geometry, type and spacing of tube supports, flow distribution requirements, pressure drop, etc. These same factors affect strongly the tube support system. Wherever rubbing or impacting occurs between tubes and support structures, due consideration must be given to design loads and material choices to assure the avoidance of fretting and galling. Design uncertainties usually lead to flow testing of the critical regions of the steam generator. In the past few years, however, considerable progess has been made with analytic methods that eventually should reduce the need for such tests.

The accommodation of thermally induced differential expansion is another particularly important design consideration for LMFBR steam generators. Low film resistances of sodium and water cause metal temperatures to respond rapdily to fluid temperature changes; coolant temperature transients are severe--as large as 5-10°F/sec (2.5-5°C/sec) over a range of 250-300°F (139-167°C) and the mismatch in the response of various structures of the steam generator can generate large stresses. Examples of areas of concern are shell nozzles, shell-to-tube-sheet transition, transition between the rim and inner tubed portion of the tube sheet, and the shell vs. tube bundle response. Mean shell temperatures normally are higher than the mean temperature of the tubes due to the large ΔT across the tube wall. This temperature differential can increase greatly during a transient because the thick shell responds more slowly than the thin-wall tubes. Similarly, there can be tube-to-tube mean temperature differences due to nonuniformity of conditions within the unit.

Frequently, thermal differential considerations lead to the need for thermal baffles which, in turn, complicate the design, increase cost, and may become the source of unexpected problems. So the various designs (such as the U-tube, helical coil, hockeystick, bent tube, and straight tube) are dictated significantly by the accommodation of thermal differential expansion; in most designs, tube-to-tube consideration is the controlling factor. For example, the helical coil design readily accommodates tube-to-tube expansion in the helical portion of the bundle. However, tube runs from bundle to tubesheet or from bundle to shell (e.g., Super Phenix) are more difficult because at one end, usually the lower, the tubes must accommodate both tube-to-tu and tube bundle-to-shell expansion as well as avoid excessive flow-induced vibration. Each design concept requires unique considerations in this regard, and it remains to be proven which is the better.

Some designs, especially the once-through units, require the use of an orifice at the feedwater inlet to each tube to guard against tube-to-tube flow instability. Owing to the high fluid velocities usually prevalent within the orifice region, the design must carefully avoid tube wall erosion. Overly small orifices are also to be avoided to guard against inadvertent plugging. Where particularly adverse conditions are encountered, testing of the orifice assembly is recommended.

Modern material and quality control practices should suffice for

LMFBR applications. Current electroslag or remelt material processes can provide suitable material for tubes and tube sheets with one potential concern. Large tube sheets can develop microfissures during the forging process near the surface area and jeopardize the integrity of machined tube attachments.

POSITIVE ATTRIBUTES OF THE LMFBR STEAM GENERATOR

The incompatibility of sodium and water and the problems of leakage notwithstanding, the LMFBR steam generator has many positive attributes, as listed in Table 4. In particular, sodium is a very well-mannered fluid. In the usual LMFBR system, where sodium oxygen concentrations are readily maintained in the 0.5-1.0 ppm range, corrosion of the sodium side is essentially nonexistent. Decarburization of the sodium boundary must be considered but is readily accommodated. Sodium pressures are modest (a few hundred psi) and heat transfer characteristics are good (high heat transfer coefficients with most fluid velocities.) Thus, sodium can be used on the shell side of the steam generator, eliminating many of the problems that have plagued the shell side of PWR steam generators (denting, impurity concentrations, flow tube thinning, etc.) As has been demonstrated by some 90,000 hours of operation at EBR-II,[3] no sodium-side problems were observed in the superheater that was completely disassembled and examined. The sodium-side surfaces were so unaffected that soapstone markings placed there during fabrication were still legible. Wear between rubbing surfaces such as tube-to-tube/support plate contact was small--mostly a slight buffing or polishing of the surfaces.

Water and steam are confined to the tubeside. The geometry is regular, crevices are avoided, fluid velocities are controlled easily, boiling is confined to surfaces that are regular and well-defined, with

Table 4. Positive Attributes of LMFBR Steam Generators
(in the absence of water/steam leaks)

Sodium (shell) side	
	Essentially corrosion free
	No crud or significant impurity deposits
	Minor carbon transport
	High heat transfer coefficients at all velocities
	Low operating pressure
	No tube denting observed to date
Water/steam (tube) side	
	Regular geometry
	Boiling confined to inside of tubes
	No reported problems from water-side scale
	Amenable to chemical cleaning
General	
	Essentially non-radioactive; facilitates maintenance & inspection
	Ferritic materials suitably immune to stress corrosion cracking

few places for impurities to concentrate or hide out, and chemical cleaning can be readily accommodated. Water quality should be maintainable by proper choice of steam cycle, operating criteria, and design of the feedwater/condensate systems. With the use of 2-1/4 Cr-1 Mo materials for the steam generator, water quality as shown in Table 5 has been used at EBR-II and has proven acceptable. Very little water/steam-side corrosion is evident in either the EBR-II evaporators or superheaters after more than 19 years of operation.

Finally, the LMFBR steam generators are essentially non-radioactive and maintenance can be performed readily.

DOE BASE PROGRAM STEAM GENERATOR DESIGN ACTIVITIES

DOE has in place an extensive and aggressive steam generator development program. Initially, work focused on developing the single-wall, hockey stick unit for the CRBR Project. An extensive program tested design features, a few-tube model, and a full-size prototype unit. Supporting the tests was an extensive program of materials development, detecting and recovering from small-to-large water-to-sodium leaks, heat transfer and hydraulic performance, vibration and wear, and tribology considerations.

Beginning in 1975-1976, DOE (then ERDA) initiated the development of two alternative steam generator designs. Technical management of the activities has been provided by Argonne National Laboratory. One steam generator concept was directed toward further development of a single-wall-tube approach using a helical-coil configuration similar in concept to the designs planned for SNR-300, Super Phenix, and Monju. The other concept being developed is a double-wall-tube, straight, shell-and-tube unit with an integral leak detection system.

Table 5. EBR-II Steam Generator Water/Steam Quality

Content, ppm	Blowdown	Feedwater	Steam
pH (Avg.)	8.8	8.9	8.9
O_2	—	<5	—
Hydrazine	—	0.01-0.03	—
NH_3	0.03-0.8	0.03-0.8	0.30-0.8
Cl	<0.01	<0.01	<0.01
Cu	<0.05	<0.05	—
Fe	<0.01	<0.01	—
Morpholine	2-8	2-8	2-8
SiO_2	<0.05	<0.05	<0.05
Total hardness	<0.05	<0.05	<0.05
Na	<0.01	<0.01	<0.01

The development program for each concept consists of the following activities:

- Preliminary design of a full-size unit for a 1000 MWe, three-loop plant,
- Design and fabrication of a 70 MWt model for testing at DOE's Energy Technology Engineering Center (ETEC),
- Posttest examination of the test model, and
- Supporting-feature tests.

The two steam generator development programs are nearing completion.

Helical-coil Steam Generator

The helical-coil steam generator is being developed for DOE by the Nuclear Equipment Division of the Babcock & Wilcox Co. The configuration of the steam generator is shown in Figure 1 and is described in detail in Reference 8. The steam generator is designed to operate as a once-through unit, transfer 438 MWt (two units per loop in a three-loop plant), and produce 155 x 10^6 lb/hr (195 kg/s) steam at 855°F (457°C), 2275 psig (15.7 MPa), with sodium inlet and outlet temperatures of 900°F (482°C) and 610°F (321°C). The design uses 240 helical-coil tubes that terminate in four tubesheets at each end. Each tube is about 352 ft (110.3 m) long and is formed by butt-welding together 50 ft (15.2 m) lengths.

The helical-coil bundle is supported at the top and expansion bends are provided in each tube above and below the bundle to accommodate tube-to-tube and tube-bundle-to-shell differential expansion. The tubes are butt-welded to tube-sheet bosses, permitting the use of butt-welded joints throughout for improved integrity of the water-sodium boundary. The material is 2-1/4 Cr-1 Mo except for Alloy 718 clamp bolts and Inconel 600 inlet tube orifices.

The program has reached the stage of preparation for testing the

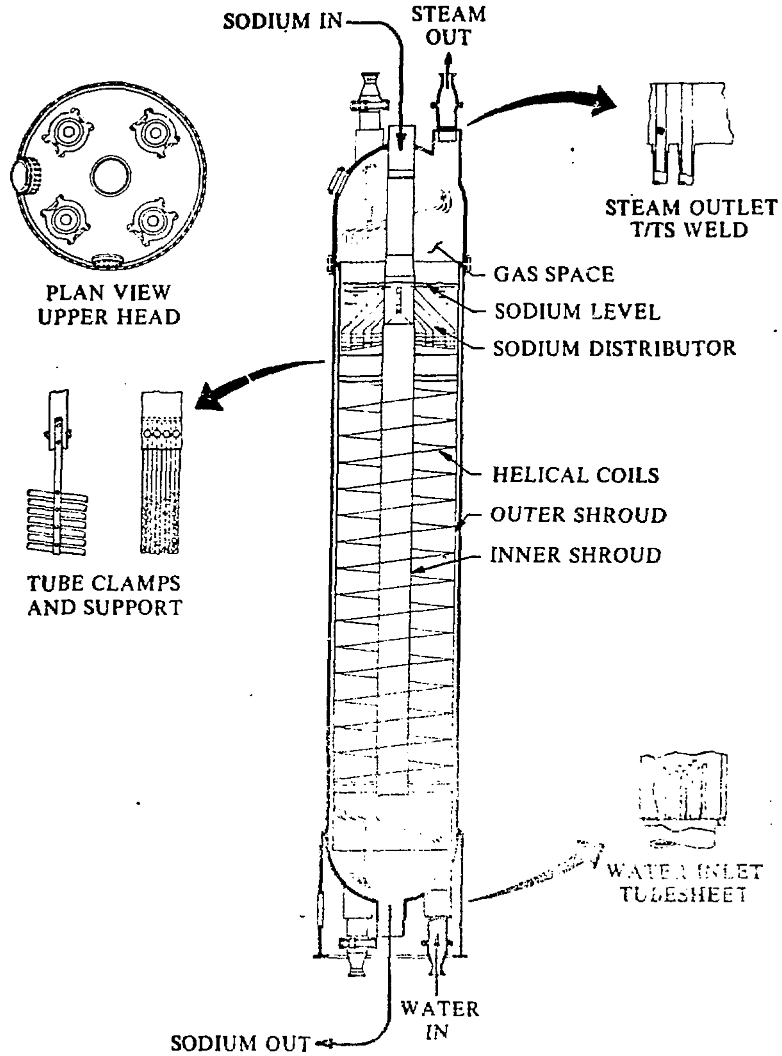

Fig. 1. Helical-coil Steam Generator Key Design Features (developed by Babcock & Wilcox Co. for DOE)

70 MWt model. The preliminary design of a 438 MWt, full size, plant unit is complete. Design and fabrication of the 70 MWt model is complete and the unit has been delivered to ETEC for testing in the 70 MWt Sodium Components Test Installation. Testing is expected to begin the fourth quarter of 1985.

Double-wall-tube Steam Generator

The double-wall-tube (DWT) steam generator is being developed for DOE by the Nuclear Components Division of the Westinghouse Corporation. The steam generator arrangement is shown in Figure 2 and is described in detail in Reference 9. As with the B&W helical-coiled unit, the Westinghouse DWT steam generator is designed to operate as a once-through unit and produce steam at 855°F (457°C), 2275 psig (15.7 MPa). Three units per loop are employed for a three-loop plant; each steam generator unit is rated at 292 MWt and produces 1.05×10^6 lb/hr (134 kg/s) of steam at full load with sodium inlet and outlet temperatures of 900°F (482°C) and 605°F (318°C).

A key feature of the design is the double-wall tubes. They are straight, 67.5 ft (20.6 m) long, and are prestressed during processing to assure contact at the expected design conditions. Each tube contains four longitudinal grooves that communicate with the space between the upper two tubesheets. The spaces are filled with helium for detecting leaks across either tube wall. With the exception of some Inconel 718 tube support plates and tubesheet baffles, the construction material is 2-1/4 Cr-1 Mo. Work on this program is advanced--development activities are essentially complete. The preliminary design of the 292 MWt plant-size units is complete. The design of the 70 MWt test article is essentially complete and fabrication has reached the assembly stage.

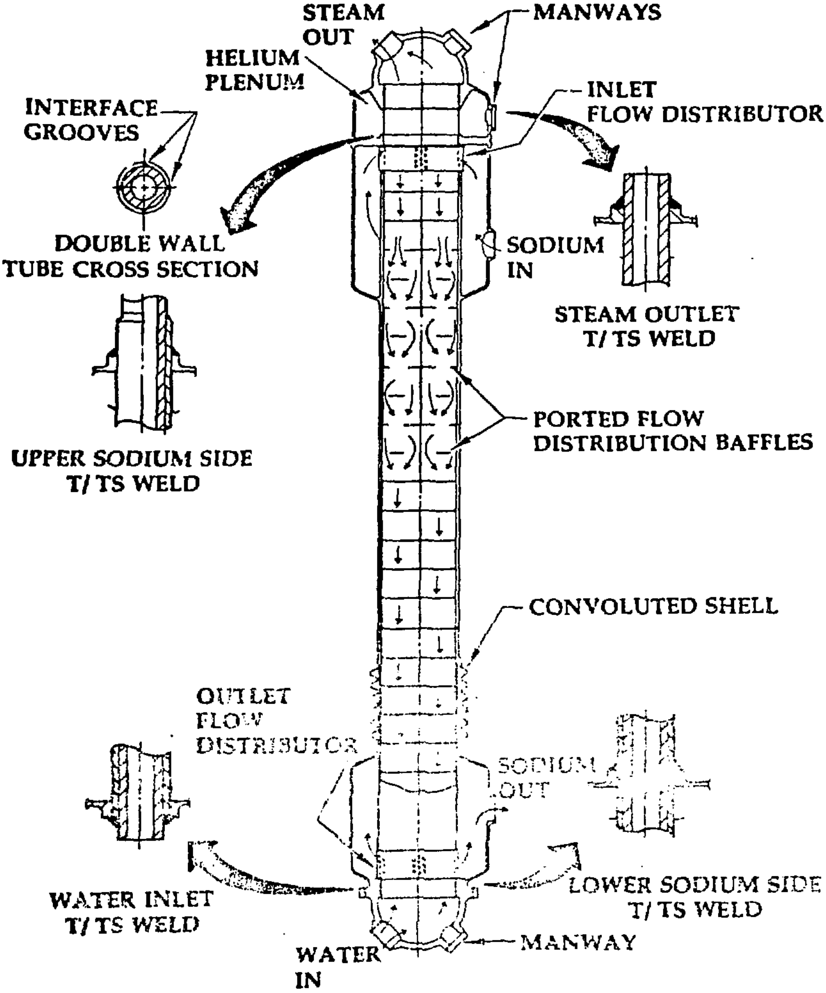

Fig. 2. Double-wall-tube Steam Generator Key Design Features (developed by Westinghouse Nuclear Component Division for DOE)

Recent Trends

The difficulties encountered in the nuclear industry are inevitably reflected in the activities and approaches taken in the developing technologies. Recent trends in the LMFBR Program have emphasized smaller, lower-first-cost, more cost-effective plants. Innovative plant designs are evolving that reflect these goals. Steam generator development is beginning to focus increasingly on cost-reducing ideas such as improving performance without increasing weight, cost-effectively employing material codes and standards, improving engineering practices to reduce cost yet achieve the requisite quality, re-examining requirements to achieve the goals of the new plant designs, and possibly using 9 Cr-1 Mo as the principal structural material.

ACKNOWLEDGEMENTS

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