COMPONENT DESIGN FOR LMFBR'S

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For Presentation at the
Annual Meeting of the American Power Conference
and Publication in the Proceedings of the
American Power Conference
April 21-23, 1975

Chicago, Illinois

Westinghouse Advanced Reactors Division

This paper is based on work performed for the Energy Research and Development Administration for the Fast Flux Test Facility
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INTRODUCTION

Pursuant to a systematic development of energy independence in the United States, the development of a commercial breeder reactor is considered a major part of the national program to provide this independence. The commercial breeder reactor concept is the most expeditious approach to a reliable, safe, economic, and environmentally attractive long term source of electrical energy. A vital link in the total program is the design and construction of the Fast Flux Test Facility (FFTF). In addition to providing a testing facility for breeder reactor structural and fuel material, the FFTF program has the goals of developing: analytical techniques for high temperature plant design; cultivating an industrial base for design and fabrication of liquid metal reactor and plant components; and, training personnel in design, fabrication and plant operations. As a result of this program, there will exist experienced design and manufacturing staffs who are well aware of the special considerations which must be factored into the design and fabrication of liquid metal fast breeder reactor (LMFBR) components.

The FFTF is a high temperature liquid sodium-cooled reactor fueled by a mixture of plutonium and uranium dioxide. Although FFTF cannot breed additional fuel, it does provide the thermal, hydraulic and neutron environment of a commercial breeder reactor for the testing of material and components. Components and plant are of such a size that FFTF can truly be considered as a first vital step in the evolution of the commercial breeder reactor. The FFTF power level of 400 MWe is comparable to that of the French Phenix reactor and the British Prototype Fast Reactor. Operating conditions such as temperature and pressure are comparable to those planned for both the first demonstration plant and the prototype commercial breeder reactor. A full technical description of the FFTF has been presented elsewhere in open literature. It is intended to review in this paper the progress made in developing the design and fabrication capability of LMFBR components.

DEVELOPMENT OF HIGH TEMPERATURE DESIGN METHODS

Prior to FFTF, the state of knowledge for design of liquid metal cooled reactors operating at elevated temperatures was quite limited. The operation of LMFBR's, in particular at temperatures above 1000°F, required significant extension of mechanical design methods used in the design of lower temperature plants. Of the two most notable U.S. breeder reactor designs, EBR-II operates at a coolant temperature of 883°F and Fermi-I had a 800°F* operating temperature. In contrast, coolant temperatures of existing light water reactors are below 650°F. As a result, the ASME structural design criteria embodied in Section III, "Rules for the Construction of Nuclear Power Plant Components," has concentrated on a lower temperature thermal environment employing in the main, linear elastic analyses. For FFTF, the design temperature is 1050°F, and the only ASME rules for elevated temperature Class I components available in early stages of design were contained in the four pages of Code Case 1331-4. In addition, the response and failure characteristics of austenitic stainless steel above 800°F were not well defined.

*Fermi-I was designed for a 900°F operating temperature.
Structural design criteria developed during the past six years as part of the FFTF program includes failure mode effect analyses, stress limits which depend on the duration of load, and recognition and reliance on inelastic analysis. Extensive determinations of high temperature material properties were obtained from a variety of sources to support the analytical efforts. These new concepts have led the ASME to expand Code Case 1331-4 to over 100 pages, and more recently, to adopt Code Case 1592 as the elevated temperature design guide for Section III components. Today we have progressed to a level of capability where we can perform the necessary analyses to insure the safe and reliable design of commercial LMFBR's.

DESIGN AND EVALUATION OF MAJOR PLANT COMPONENTS

With few exceptions, the design and manufacture of critical components for the FFTF plant were performed by suppliers with little or no previous experience with high temperature components. Although design of the components, in most cases, did not pose undue manufacturing difficulties, fabrication of stainless steel components in shops where primary emphasis had previously been based on carbon steel component fabrication did present problems in handling, cleaning, and machining. One area of particular difficulty was maintaining an adequate cleanliness level in a shop where such levels were previously unnecessary. The relatively thin walls of large vessels presented unique problems in handling and precise machining to maintain concentricity. Difficulties encountered in achieving satisfactory weld quality caused significant delays in component manufacture. Considerable time and effort was expended in achieving the required welder and inspector qualifications needed to meet the rigorous requirements. In general, after capabilities were developed to achieve the necessary quality levels, welding proceeded on a satisfactory schedule.

In addition, FFTF design introduced the application of Reactor Development and Technology (RDT) Standards. The RDT Standards set exacting requirements for a wide range of topics such as materials, welding, material forming processes, non-destructive test procedures and acceptance criteria, cleanliness and quality assurance. These standards supplement existing nationally recognized standards to satisfy more stringent sodium reactor technology, safety and reliability requirements. Lack of familiarity with these standards contributed to manufacturing difficulties.

During the manufacture of FFTF components, significant advances were made in the welding of complex shapes without incurring unacceptable weld distortion. Three such examples were the fabrication of the instrument tree, the core basket, and the core support structure. In the case of the core basket (Figure 1), 151 receptacles for the fuel assemblies had to be welded in place with precise alignment control. This was achieved through use of fixturing and controlled weld sequencing. Significant distortion was encountered in welding the top plate with its receptacles to the barrel section and caused major misalignment of the receptacles. After recovery was made through straightening and machining operations, a girth weld to the bottom plate assembly was necessary. Following significant changes and requalification of the weld procedure, a satisfactory method was developed and the girth weld was performed without incident.

The three instrument trees, which provide operational data for every core position and guide tubes for the control/safety rods (Figure 2), are unique to FFTF and similar components will not be used in commercial breeder reactors. Alignment of the instrument trees with respect to the core and reactor head is exacting, and distortions resulting from welding or relaxation of weld stresses due to immersion of the trees in 1050°F sodium cannot be tolerated. The 28-foot column was built up by welding a series of seven tubular cross section pieces. Four of these have a non-symmetric "D" cross section and the remainder are circular. This column was required to meet a length and straightness tolerance of ± 0.25 and ± 0.095 inch, respectively, in 28 feet at weld completion with no additional machining.

After the column was completed, two box frame weldments were attached to it, each requiring 36 inches of one-inch thick weld. These frames were premachined to support the control rod guide tubes. The box welding was controlled to position the guide tubes within 0.045 inch of true position with respect to the column centerline. Additionally, the upper and lower frames, which control the positions of the control rod guide tubes and which are approximately 12 feet apart, were required to be oriented within ± 0.020 inch of each other.
During the fabrication of the prototype instrument tree, several areas where weld distortion could occur were identified. Methods to correct these potential sources of distortion were evolved as the prototype was being fabricated. These techniques proved successful and have been incorporated into the fabrication of the three plant units which are now in the final stages of manufacturing. Success of these techniques was highly dependent upon the mock-ups made of the various configurations. Weld shrinkage was measured so that the actual components for the final product could be dimensionally corrected to account for the expected shrinkage. In addition, local peening, fixturing, and sequential welding were employed.

The success of these measures is best demonstrated by the fact that critical measurements made on the prototype instrument tree before and after a 20-day exposure in 1100°F sodium were virtually identical. The fabrication experience gained in developing the complex weldments has provided valuable information to component designers faced with predicting and controlling weld distortion.

The third example of the development of exceptional distortion control during welding was in the fabrication of the core support structure (CSS). The CSS, which is attached to the reactor vessel by a support skirt, provides support for the core assemblies, the inner core reflectors, and the radial neutron shield which surrounds the core. In addition, it provides passages for coolant flow distribution to these core assemblies and to the annular space between the core barrel and the reactor vessel wall. The CSS forms the pressure boundary between the high pressure inlet plenum and the low pressure outlet plenum of the reactor vessel. The CSS, excluding the core basket, is designed for the 20-year life of the plant and no maintenance is planned or envisioned.

The castings which form part of the bottom head of the CSS are composed of two sections of a cone joined together by welding across the ligaments of the 12 equally spaced 16 inch diameter flowholes in the bottom head. The two-piece welded casting was necessitated since a single casting of 80,000 pounds was beyond the melting capacity of stainless steel casting suppliers. The core support structure required a total of six tons of weld metal to complete its fabrication. The control of weld distortion and weld shrinkage and the production of radiographically acceptable welds in the heavy wall sections required continuous procedural planning during production. Submerged arc welding was used wherever possible and a low defect and repair rate was experienced. Good accessibility for welding was not always present due to the close proximity of parts. For example, two-inch thick gussets, full penetration welded to the inside of the bottom head, were made with the welder standing inside 16-inch diameter flowholes. Welding was performed from alternate sides of double welded joints to minimize distortion and internal spiders were used in cylindrical parts to maintain roundness during girth welding. Even with these precautions, some rerounding and straightening operations were required.

Both the lower core support structure and the core barrel were thermally stabilized before these parts were final machined and joined together. This was the most critical weld in the core support structure due to the tight perpendicularity tolerances imposed. Extreme precautions were taken to insure uniform and symmetrical drawdown of the barrel assembly with the lower structure by sequencing welding and monitoring relative position of the two structures until welding was completed.

With the use of 304 stainless steel as the primary material of construction in FFTF, great care was exercised throughout manufacturing to control contamination of the finished surfaces and to protect against stress corrosion. In several components, where dimensional stability was of concern, to maintain precise dimensions under the elevated temperature conditions, heat treatments were performed at temperatures in excess of the maximum operating temperature for which they were designed. Most FFTF components which were stress relieved utilized temperature-time cycles which resulted in no sensitization.

An exception to this practice was the heat treatment of the primary pump tank, shaft, and internals. These were annealed at 1150°F for four hours to assure that dimensional stability was obtained at the 1050°F service temperature. In the case of the pump tanks, only the upper section of the pump was given this heat treatment. Although it was recognized that this heat treatment could cause severe intergranular carbide precipitation, it was thought that intergranular corrosive attack could be controlled during manufacture.

During liquid penetrant examination of the welds on the first primary pump tank, an unusually large concentration of very small penetrant indications was noted. While the general size of individual indications was
such that they met Code inspection requirements, the large number of indications caused considerable concern and prompted further investigation. Information obtained by means of surface replication and metallographic examination of boat samples removed from the surface of the upper tank section revealed intergranular separations having a depth from a few mils up to 100 mils. It was further determined that light surface grinding was required prior to performance of the liquid penetrant examination to reveal the indications. This technique was slow and in many instances revealed less than 50 percent of the indications present, but, it was the only means of assuring that the pattern of intergranular attack could be identified. Subsequently, eddy current inspection was substituted as an alternate to liquid penetrant inspection and proved to be both reliable and efficient in determining location and depth of intergranular attack.

It is important to note that no evidence of intergranular attack was identified in weld metal, in either the as-welded or annealed condition, and in the pump shaft in spite of the fact that the shaft had been given a stabilizing anneal at 1150°F followed by approximately 100 hours of exposure to temperatures in the sensitizing range. For some unknown reason, the extent of the intergranular attack on the pump internals was significantly less than that observed on the upper portion of the pump tanks, even though the same heat treatment was involved and fabrication was performed in the same supplier's shop. The pump internals were judged acceptable and no remedial action was instituted. Additional restrictions, however, were instituted on the internals to keep them clean and dry packaged (less than 40 percent relative humidity). Further, it was experimentally determined that no further intergranular attack would occur with exposure of the pump internals during specification testing in water at the pump supplier's plant.

In the case of the upper tank assemblies, however, it was judged that the attack was sufficiently severe that material must either be replaced or restored to a metallurgical condition not susceptible to this form of attack. Following a series of heat treat scoping studies, it was determined that a four-hour heat treatment at 1800°F followed by a water quench, to be carried out within two minutes of furnace removal, would restore the material to an acceptable condition. In addition to restoring immunity to intergranular attack, it was desirable that the reheat treatment cycle be selected so to minimize distortion of the final machined upper tank assembly. Prior to committing the upper tank section to this heat treatment, it was also determined, through heat treatment of a spare ring forging, that distortion during heat treatment of the pump tank sections would be acceptable. Following that determination, the three upper pump tank sections were heat treated successfully with distortions well within acceptable limits. These heat treated upper assemblies have since been rejoined to the lower tank sections, fabrication has been completed, and the tanks have been shipped to the FFTF site.

Efforts to determine an assignable cause for the intergranular attack have not provided sufficient data to establish a single agent as being responsible. Since the phenomenon was observed on both plate and forgings, the specific heat or form of the base material could not be established as being particularly susceptible to attack. It was demonstrated that the attack would occur in a vendor's plant only on sensitized material and that an incubation period of approximately five weeks was required to produce detectable attack.

While the remedial actions taken to restore immunity of the tank sections to intergranular attack were successful, they were most undesirable in terms of schedule and component costs. This experience again emphasizes the necessity to thoroughly examine the implication of any stress relief of unstabilized austenitic stainless steels in the sensitizing range of 900-1650°F. If such an anneal is found to be necessary, elaborate precautions to control cleanliness and moisture are mandatory through the fabrication, storage and erection operations.

**REACTOR VESSEL**

The reactor vessel is constructed of Type 304 stainless steel, and consists of four basic sections: the barrel, core support ring forging, inlet plenum and liner. The barrel, which comprises the upper cylindrical portion of the vessel, has an inside diameter of 20 feet, 3 inches, and a wall thickness of 2 3/8 inches. The barrel is joined to a ring forging located at about core midplane, which provides support for the core support structure. The ring forging is joined to a torispherical bottom closure which serves as a high pressure inlet plenum for the reactor coolant. The overall height of the reactor vessel is 43 feet, 1 1/2 inches.
The fabrication and in-shop handling of the reactor vessel was complicated by its sheer size and its large diameter to wall thickness ratio which resulted in significant distortion under its own weight when lying horizontally. Special fixturing was used to control this distortion for some final machining operations while in other cases, the machining operation factored in and compensated for the distortion. Distortion control in the welding of the core support structure and the thermal liner into the reactor vessel required the use of controlled weld bead deposition and continuous alignment checks during the welding.

Because of the low temperature, low expansion carbon steel reactor vessel head and the high temperature, high expansion stainless steel vessel, a typical bolting attachment would result in ratcheting of the vessel wall. Therefore, a specialized arrangement was utilized. A ring forging is welded into the top of the barrel directly below the top flange to which 30 support arms are welded. The reactor vessel is securely attached to the main support structure (MSS) by means of bolting the vessel support arms (Figure 3) to the MSS. Sealing is accomplished by an omega seal welded to the upper flange of the reactor vessel and to the I-ring and by two concentric metallic O-rings between the head and the I-ring. The long retention bolts are preloaded only in the middle to achieve sealing loads and still allow the head to move freely with respect to the reactor vessel support ring. The support arms remove the mechanical loads from the high thermal stress portion of the vessel wall. The vessel support arms are also slotted, which eliminate any stress due to the temperature gradient, but more important acts as a resilient spring to reduce loads on the concrete ledge.

The reactor vessel will receive an ASME Code N stamp in the field following the welding of the omega seal to the reactor vessel flange and helium leak testing in accordance with ASME Code Case 1595. Because reactor vessels for the future LMFBR plants will not be that much larger than the FFTF vessel and head, the experience gained in the design, fabrication, and installation of the FFTF vessel provides the technology needed for these future vessels.

### MAIN COOLANT PUMPS

Prior to FFTF, the stage of development for liquid metal pump design in the U.S. was reflected in EBR-II and Fermi-I. The major characteristics of the sodium pumps used in those facilities and FFTF are presented in Table I. It is clear that the FFTF pump must satisfy significantly more demanding requirements than its predecessors; consequently, a definite advancement in liquid metal pump technology will be achieved with the FFTF design.

The pump (Figure 4) is a free surface, shaft seal, single stage, single suction pump with a variable speed drive and is located in the hot leg of the reactor heat transport system. Inlet flow enters the hydraulic assembly via the suction elbow and is discharged from the impeller radially into the diffuser/turning vane assembly which in turn redirects the flow from a radial to a downward direction through the discharge nozzle.

The 1050°F design temperature and thermal transient requirements for the pump necessitated in some cases the attenuation of transients by mixing the transient fluid with constant temperature fluid. The hydraulic assembly represents a unique

### TABLE 1 U.S. LIQUID METAL PUMP DESIGN COMPARISONS

<table>
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<tr>
<th>CHARACTERISTIC</th>
<th>FACILITY</th>
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<tr>
<td></td>
<td>EBR-II</td>
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<tr>
<td>PUMP TYPE</td>
<td>MECHANICAL-CENTRIFUGAL</td>
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<tr>
<td>PUMP HEAD (Ft Na)</td>
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<tr>
<td>PUMP CAPACITY (GPM)</td>
<td>4,500</td>
</tr>
<tr>
<td>OPERATING TEMP (°F)</td>
<td>700</td>
</tr>
<tr>
<td>APPROXIMATE BHP</td>
<td>500</td>
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*Designed For A Maximum Temperature Of 1000°F
approach to this problem, in that only non-pressure containing walls are exposed to the full transient. Strain ratcheting is practically eliminated by this approach.

The pump shaft is fabricated by welding two solid forgings to a tube section. The dynamic design criteria imposed in conjunction with the fact that only two radial bearings are used required the use of an 18 foot, 28 inch diameter section made from a seamless extruded tube. A thermal baffle system is located in the upper end of the tube to minimize the heat transferred into the shaft seal region.

Briefly, some of the pump development programs carried out or still in progress are:

1. Design verification of the seal in a conventional test rig and testing of a complete shaft seal package in a simulated operating environment with seal operation over a sodium pool, in argon cover gas.

2. To confirm bearing performance calculations, tests were performed in a water test rig followed by bearing operational tests in sodium.

3. Bearing hardfacing material selection evolved from a series of thermal shock tests of coupons hardfaced with candidate materials. Stellite proved superior with respect to tolerating repetitive thermal shocks. A follow-up to this work consisted of thermal shock tests on full-size hard-faced journal and bearings.

4. An air model test of the inlet configuration was conducted which confirmed the calculated fluid velocity profiles at the impeller suction.

5. Two development programs aimed at supporting the dynamic design effort were performed. One test provided data on the effects of shaft motion transmitted to the pump structural members via the surrounding fluid to enable accurate analytical modeling. The other test exposed a full-size shaft to simulated plant thermal gradients for evaluation of thermal distortions. As a result, design modifications were made and the shaft satisfactorily completed retesting.

6. A comprehensive prototype pump test program in water followed by sodium testing at the Liquid Metal Engineering Center is planned prior to installation of this equipment in the FFTF.

The primary purpose of the prototype is to demonstrate performance characteristics and mechanical reliability at operating temperatures prior to actual in-plant use. At the completion of the scheduled testing, the prototype is potentially capable of serving as a spare for the FFTF plant.

The water test of the prototype pump has been completed, and it has verified the basic mechanical design and hydraulic performance characteristics. During performance of this phase of the prototype pump test, two problems were identified for which design changes were required. A shaft orbit was observed as measured by journal position sensing instrumentation in the hydrostatic bearing. The shaft orbit has been reduced to an acceptable level by a reduction in the hydrostatic bearing supply orifice size and by an improved concentricity between the shaft and its surrounding chamber. A slight dip in the head flow curve was detected and attributed to flow separation in the diffuser. This condition was also brought within acceptable limits by a modification to the impeller discharge configuration.

The first FFTF plant pump was modified to incorporate the changes made to the prototype pump and is presently being tested in water. Over 300 test hours, to date, have verified the modifications made for the two problems identified during the prototype pump tests were acceptable.

Testing of the prototype pump in high-temperature sodium is scheduled to start in July 1975. This testing represents a first in that the pump will be operating at the 1050°F temperature level for extended periods. Basic testing will be essentially the same as that performed in the water facility, with the additional capability of subjecting the pump to thermal transients approximating those imposed by the FFTF plant.

THE INTERMEDIATE HEAT EXCHANGERS

Three intermediate heat exchangers (IHX) of 133 MWt are used in FFTF. The units are vertical, counterflow, shell and tube type heat exchangers. The IHX is composed of three main subassemblies: the hanging support, which is a cylinder bolted to the containment building operating floor and supports the other two subassemblies, the shell, and tube bundle. The IHX has an overall length of approximately 35 feet, an inside diameter of 76 inches, and has an empty weight of 90 tons.
In contrast to heat exchangers used in light water reactor systems, the primary sodium is on the shell side and the secondary sodium circulates through the heat transfer tubes. Because the tubesheets are welded to a rigid central downcomer pipe, provision was made for the differential thermal expansion between the tubes and the cooler downcomer pipe. This was accomplished by adding a large compound bend in each tube (Figure 5). The IHX for the Clinch River Breeder Reactor Plant (CRBRP) will use a floating lower tubesheet with a bellows expansion joint in the central downcomer to accommodate the differential thermal expansion and will, therefore, use straight heat transfer tubes.

The manufacture of the heat exchangers was accomplished without an unusual number of delays and problems, considering the complexity of the design and the several thousand individual parts involved. The extensive use of well designed jigs and fixtures minimized dimensional deviations and weld distortions and permitted interchangeability of parts between units.

One fabrication operation for which little production experience existed was the tube to tubesheet spigot welding. To eliminate the crevice between the tube and tubesheet hole, which exists when a conventional front face tube-to-tubesheet weld is used, this design employs a full penetration fusion butt weld of the tube to a machined spigot on the back side of the tubesheet. An automatic tungsten inert gas torch fuses the joint from the inside of the tube. The tubesheet downcomer assembly was erected vertically in the clean room and tubes were welded into both tubesheets simultaneously. All welds were accepted by a visual as well as a radiographic inspection before the next row of tubes were fitted. Most of the weld defects found were porosity and linear indications due to sharp changes in weld contour which were corrected by refusing the weld.

While welding the eight row of tubing on the lead tube bundle, it was noted that some welds were not fusing smoothly at the outer corner of the spigot and the corner produced a linear indication on the radiographs. Visual examination disclosed an unusual oxide coating on the weld which upon chemical analysis was determined to be aluminum oxide. Further chemical analyses of the tubing heats involved in the oxide coated welds revealed an aluminum content which varied from 0.005 to 0.40 percent. Testing determined aluminum content up to 0.15 percent does not degrade the strength of 304 stainless steel. However, an aluminum content above 0.07 percent does increase the fluidity of the molten pool during the fusion welding, producing sharp changes in weld contour and an oxide coating which could possibly crack off and be carried into the pump bearings. All tubes containing aluminum in excess of 0.07 percent were replaced with other tubes. While FFTF did not specify an aluminum content for heat transfer tubing, the specification for the CRBRP IHX incorporated a 0.07 percent maximum aluminum content.

Major testing programs performed to confirm operating and design features of the IHX are briefly described below:

1. A seven-tube model was tested with water to measure pressure loss characteristics through the tube support plate flowholes and to measure tubeside flow induced vibration at five different flow combinations. No vibration occurred from 10 to 150 percent of full flow.

2. A single-tube vibration test was performed which verified that tubeside flow does not induce significant vibrations in the bent tube section and that support was not required at the center of the bend.

3. A tube fretting wear test was performed in 1050°F sodium utilizing a three-tube, two-span mockup of the tube bundle in which flow induced vibrations were simulated. It was found that the fretting wear, for up to 200 million cycles, was too small to express in quantitative terms.

4. The IHX employs a perforated and slotted conical flow distribution plate at the top of the primary side inlet plenum to evenly distribute the flow to the shell side of the tube bundle. In order to determine the hole and slot pattern which would provide even flow distribution around the periphery of the tube bundle, a one-fifth scale model of the inlet plenum chamber and inlet elbow was built of plastic and tested in a water test loop. The model was divided into twelve equal sections. By varying the total hole area in each section, an equalized flow pattern was determined. Flow rates up to the equivalent of 100 percent of design flow were used and a satisfactory flow balance was established. The final configuration flow balance is ± 2.5
percent down to 60 percent of full flow and +19 and -6 percent at 40 percent flow.

5. A full-scale model covering the heat transfer section between the tubesheets was built. The model was a 72 degree segment of the heat exchanger cross section. The model was used to determine the velocity distribution on the primary side between the tube support plates from the center to the outside of the tube bundle and to measure the pressure drop across each support plate. Measurements were taken in the range of 25 to 100 percent of design flow. The tests revealed that adjustments were necessary in the number and size of flow holes in some regions. Modifications were made and tested to achieve a more constant velocity distribution through each support plate. The pressure drop measurements taken after the modifications confirmed that the pressure drop on the primary side would meet specification requirements.

Following the above testing, the model was reduced to a 48 degree wedge section to permit flows up to 150 percent of the design flow rate. Testing was performed to determine the amplitude and frequency of flow induced vibrations. It was shown that no resonance occurred between the driving frequency and the tube natural frequencies. The tube response was less than 0.001 inch rms and typically random in nature.

6. In order to verify acceptable flow distribution into the secondary side, a one-third scale model was built which simulated the center secondary sodium downcomer liner, the lower hemispherical head, and the tube bundle. The model was tested without any flow distribution structure in the head and unacceptable flow balances resulted. Testing revealed that an annular ring, attached to the inside wall of the head, would produce the desired distribution.

7. The ability of the tube-to-tubesheet butt welds to withstand imposed thermal shocks was demonstrated by means of a seven-tube, double tubesheet, stainless steel tilting autoclave-type vessel using liquid sodium as the heat transfer medium. A total of 252 thermal shocks, using the most severe temperature changes the welds will experience, were performed by heating the tubesheets and sodium to the two temperatures required to produce the desired differential temperature and then rotating the autoclave 180 degrees such that the sodium ran through the tubesheets and tubes to the opposite end of the autoclave. Following the shock testing, the 14 tube-to-tubesheet welds were subjected to inspections and examination and no evidence was found of cracking or separation of tubesheet cladding from the base metal.

PROTOTYPE TESTING

To confirm design adequacy of the more developmental reactor and plant components, a number of prototypes were built and are being tested at operating temperatures. Included in the prototype test program are the primary sodium pump, in-vessel handling machine (IVHM), core restraint mechanism (CRM), instrument tree (IT), and control rod drive mechanism (CRDM). At this point in time, the primary sodium pump has completed water testing and is awaiting testing in sodium, the CRM has satisfactorily completed air testing and is currently in sodium testing, and the CRDM has satisfactorily completed a life test in 1100°F sodium. Test requirements for the IVHM and IT are briefly described below.

The incorporation of the special in-reactor testing components imposed the development of IVHM and IT designs unique to FFTF. These components are designed for a sodium environment of 1100°F at one end and an ambient environment 30 feet away at the control end. The components remain static during reactor operation but are required to move with precision during shutdown. These components have bearings immersed in sodium and bearings and seals exposed to sodium vapors which can cause a sodium frost problem. These components have sliding surfaces with close tolerances subject to sodium frost formation which must continue to slide freely in order to adequately perform their function. The test program was specifically focused on determining the retention of these operating capabilities throughout design lifetime.

In the case of the IVHM (Figure 6), the design task was to make a 50 ton machine mounted on an adjustable foundation, index 94 different remote locations over an area of approximately 104 square feet and be able to locate its grapple within ± 1/4 inch. It must maintain capability to lift the component from the core and position it, either in another core location, or in a storage module for later use or examination. During these operations, the arc made by the grapple is dependent on accurately controlling the rotating plug and rotating arm to insure that it does not contact other reactor components. All of these operations are performed without visual aid under 10 feet of sodium and must be done accurately over the lifetime of the component.
Although operating requirements for the IT are not as stringent as for the IVHM, a disconnect must be made of the CRDM driveline prior to each refueling, the in-reactor assembly must be moved to a parked position; after refueling, the IT is moved in place over the core and a reconnection must be made of the CRDM driveline.

The prototype IT and IVHM are presently undergoing testing at Hanford Engineering Development Laboratory. This test program consists of two major phases: (1) air testing in the core mechanical mockup (CMM); and (2) high temperature sodium testing in the Composite Reactor Component Test Activity (CRCTA).

Each program consists of individual and composite, functional, dimensional and interface tests with a simulated core and other reactor components. In CRCTA these components are immersed in sodium and are exposed to the thermal environment that the plant units will be subjected to during plant operation.

The IT has successfully completed functional testing in CMM and initial functional and dimensional tests in CRCTA. The crucial phase of CRCTA testing occurred after the tree was soaked in 1100°F sodium for 20 days. Prime concern centered around the potential relaxation of welds and resulting stress relief which could cause in-reactor distortions making the tree inoperable. Functional testing of the IT following the 1100°F soak was successfully completed and dimensional measurements before and after the 20 day exposure to 1100°F sodium indicate relatively little or no distortion of the in-reactor assembly.

Air testing of the IVHM prototype revealed that extensive changes in design details were necessary to correct problems which were not evident on paper. These changes have been incorporated into the plant unit components. The IVHM has successfully completed air testing in CMM and is currently being installed in CRCTA for composite test with the IT and other reactor components.

All the test programs, models, mockups, prototypes, etc. are too numerous to discuss here. But the significant accomplishment of the test programs is that the testing to date has confirmed the adequacy of the designs established by the codes, standards, and analytical methods developed to produce FFTF components.

**SUMMARY**

Just as FFTF has prototype components to confirm their design, FFTF is serving as a prototype for the design of the commercial LMFBR's. The operating conditions of LMFBR's, in particular the elevated temperature conditions, required an extension of the state-of-the-art of mechanical design which has been accomplished by FFTF. Development of the FFTF design to meet stringent safety and performance requirements necessitated the extension of many analytical tools needed for LMFBR's.

Design and manufacture of critical components for the FFTF system have been accomplished primarily using vendors with little or no previous experience in supplying components for high temperature sodium systems. The exposure of these suppliers, and through them a multitude of subcontractors, to the requirements of this program has been a necessary and significant step in preparing American industry for the task of supplying the large mechanical components required for commercial LMFBR's.

In conclusion, FFTF is a vital, necessary step in the United States program for the development of an economical, reliable, and safe commercial LMFBR. The FFTF has currently completed many of its originally established goals and accomplishment of the remainder will provide further significant information necessary to the LMFBR development.

**REFERENCES**


Figure 1. Core Basket Upended During Fabrication
Figure 2. Instrument Tree
Figure 3. Reactor Vessel/Closure Head and Main Support Structure Interface
Figure 4. Primary Sodium Pump
Figure 5. IHX Tube Bundle During Tube-to-Tubesheet Welding
Figure 6. In-Vessel Handling Machine