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FAST REACTOR FUEL ASSEMBLY
THERMAL HYDRAULIC
DEVELOPMENT PROGRAM -
SODIUM HEAT TRANSFER

February 1970

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FAST REACTOR FUEL ASSEMBLY THERMAL
HYDRAULIC DEVELOPMENT PROGRAM - SODIUM HEAT TRANSFER

By

J. M. Yatabe, R. E. Collingham,
V. R. Hill, T. I. McSweeney,
and W. L. Thorne

Engineering Department
FFTF Division

February 1970

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ABSTRACT

This document provides the program plan for the out-of-reactor sodium heat transfer testing required to support the design, safety evaluation and initial reactor operation of FFTF driver- and test-fuel assemblies. These tests will employ electrically heated models of FFTF fuel assemblies and will be conducted under conditions of flow and heat generation simulating FTR operating conditions. The program will provide directly applicable laboratory information to support evaluation of:

- Clad temperatures and temperature gradients under normal and structurally distorted geometries
- Void initiation and propagation in loss of coolant accidents
- Emergency cooling operation and limits
- Postexpulsion cooling
- Operational limits for driver and test fuels.

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FAST REACTOR FUEL ASSEMBLY THERMAL
HYDRAULIC DEVELOPMENT PROGRAM - SODIUM HEAT TRANSFER*

INTRODUCTION

This report presents the plans for the sodium heat transfer testing portion of the Fast Reactor Fuel Assembly Thermal Hydraulic Development Plan. The purpose of this test program is to provide laboratory information to directly support: LMFBR fuel assembly design, reactor safety evaluation and operation. Since the first major reactor to be built under the LMFBR program during the 1970 decade will be the Fast Test Reactor, the immediate goals of this program are to support FTR design and initial operation.

This report provides a detailed description of the development test program to be performed in terms of specific problems, test content, test schedules, test facilities, and specific information provided to the reactor designer.

The problems described herein and the specific tests to evaluate their magnitude are based on the information developed during the conceptual design and early stages of preliminary design. If major changes occur in the driver fuel assembly design and new problem areas arise during the preliminary and final design, revisions will be made in the form of replacements and supplements to the present development plan.

* This report complements the first program report, Thermal Hydraulic Development.⁽¹⁾

SUMMARY

The Fuel Assembly Thermal Hydraulic Program has been divided into four phases (Table 1):

- Local clad temperature tests
- Voiding initiation and propagation
- Specific accident tests
- Improved core performance operation tests.

TABLE 1. Summary of Planned Thermal Hydraulic Tests

Topic	Test Section Description	Range of Fixed Parameters	Measured Variables	Information Transmitted to Design
Local Clad Temperature Tests	Electrically heated and instrumented bundles of 7 and 39 pins prototypical of sectors of FFIF fuel assemblies capable of simulating several structurally distorted conditions.	<ul style="list-style-type: none"> Pressure to 55 psig. Coolant temperature to 900 °F. Distortions: bowed pins to contact, loose wire wraps, spirally deformed pins. Flow rates: 100-1500 lb/ft²-sec. Heat flux: 500,000 to 800,000 Btu/hr-ft². 	Local clad and coolant temperature, power, pressure, bulk coolant inlet and outlet temperature, coolant flow rate, coolant purity (including inert gas).	<ul style="list-style-type: none"> Edge row temperature gradients. Verification of analytical design tools. Spacer induced hot spots. Overheating caused by loose wire accommodation. Overheating caused by pin bowing. Temperature gradients in spirally deformed assemblies.
Voiding Initiation and Propagation	Electrically heated and instrumented bundles of 7 and 19 pins prototypical of sectors of FFIF fuel assemblies, sheath surface modeling fuel assembly clad at specific degrees of exposure. Additional single pin test assemblies to extend range of surface conditions to all levels of exposure.	<ul style="list-style-type: none"> Pressure to 55 psig. Temperatures to 1800 °F. Distortions: pins bowed to contact. Flow rates: 300-1500 lb/ft²-sec. Heat flux: to 1,200,000 Btu/hr-ft². <p>In addition, during the voiding state, transient surveys of parameters will occur.</p>	Local clad and coolant temperature, power, pressure (local), bulk coolant inlet and outlet temperature, coolant purity including inert gas, quality, void volume and clad surface conditions, strain and strain rates on ducts and pins.	<ul style="list-style-type: none"> The degree of local superheating (representative of all exposure). Strain rates on pins during voiding. Strain rates on ducts. Rate of expulsion. Response and sensitivity of reactor instrumentation to off normal conditions prior to and during voiding. Effect of entrained gas on superheat. Effect of reactor velocity on superheat.
Specific Accident Tests	Electrically heated and instrumented bundles of 7 and 37 pins prototypical of sectors of FFIF fuel assemblies.	<ul style="list-style-type: none"> Outlet pressures to 55 psig. Temperatures to 1800 °F. Distortions: pins bowed to touching. Flow rates: to critical flow Heat flux: to 1,200,000 Btu/hr-ft². 	Local clad and coolant temperature, power, pressure (local), bulk coolant inlet and outlet temperature, coolant flow rate, coolant purity including inert gas content, quality, void volume, clad surface conditions, strain and strain rates on pins and ducts.	<ul style="list-style-type: none"> Cooling rates during high void volume flow (expulsion). Strain rates on pin and ducts during expulsion. 2-phase pressure drop flow behavior at high void volume. Natural convection cooling rates. Natural convection boiling heat transfer rates and limits (e.g., critical heat flux).
Emergency Cooling				
Operational Limit Tests	Electrically heated and instrumented bundles of pins modeling 37 pin sectors of driver fuel assemblies.	<ul style="list-style-type: none"> Pressure to 55 psig. Coolant temperatures 1200 °F. Flow rate: 900-1500 lb/ft²-sec. Heat flux: to 2,200,000 Btu/hr-ft². 	Clad temperature, power, pressure, bulk coolant inlet and outlet temperatures, coolant flow rate, strain levels in pins and ducts.	<ul style="list-style-type: none"> Demonstration of heat flux operation to demonstrate advanced core operation and support the establishment of operating limits.

OBJECTIVES

The objectives of the Fast Reactor Fuel Assembly Thermal Hydraulic Development Program are to provide sodium heat transfer laboratory data, obtained from heated models of fuel assemblies, to directly support the design and operation of the Fast Test Reactor. The program is to provide support to a reactor meeting the following performance levels established by the primary requirements and criteria:⁽²⁾

- An integrated fast flux in the range of 0.7 to 1.35 $\times 10^{16}$ n/cm²-sec with 75% of the flux greater than 0.1 MeV
- Specific power in the range of 500 to 2000+ W/g plutonium
- Reactor bulk sodium outlet temperatures from 800 to 1200 °F
- A plant availability target of 75%
- Instrumented closed loops, about 6 in. OD for a 3 to 4 ft core, and about four to six in number
- Instrumented open loops, about six in number
- Capabilities to test fuel up to and including failure in dynamic sodium.

To meet these criteria and requirements, the reactor operation will progress in several major steps from an initial operation at lower neutron flux, test space and coolant outlet temperature levels to the full performance operation which will achieve the upper range of parameters set forth in the primary requirements and criteria.

The immediate goals of the Thermal Hydraulics Program are to support the design licensing and first core operation of the Fast Test Reactor. Specific information will be provided to support the following major milestones:

- Preliminary and final driver fuel design.
- Preliminary and final instrument package design.
- Final Safeguards Analysis Report.
- Initial reactor operation.
- Closed test loop operation.
- Full performance reactor operation.

The conceptual design of the initial core has been basically described in Reference 3 and is summarized in Table 2. Changes are expected during the preliminary and final design studies; however, the reference design is considered to be near enough to the initial core that parameter ranges can be defined for the purposes of planning this experimental program.

TABLE 2. Summary of Performance and Configuration of Specifications for Initial FTR Fuel Assemblies⁽³⁾

Reactor Power, MW	395 (76 driver fuel, 6 in-core control rods)
Pin OD, in.	0.230
Cladding ID, in.	0.200
Pin Separation: Method	wire wrap, all pins (0.056 in.)
Center-to-Center Spacing, in.	0.286 on a triangular pitch
Pin Support Span, in.	2 in. on a 60° progression
Sodium Coolant: Flow Direction	up
Max. Velocity, ft/sec (orificed)	25
Core Pressure Drop, psi (orificed)	69

TABLE 2. (contd)

Pin Linear Power (kW/ft, at 0.90 total power)

Average	7.0
Maximum	12.2
Maximum (with OPF)	14.6

Nominal Pin Heat Flux (10^3 Btu/hr-ft²):

Average	400
Maximum	690
Maximum (with OPF)	830

Power Peaking Factors:

Nuclear:	
Axial	1.24
Radial	1.40
Overpower Factor (OPF)	1.20

Hot Channel Factors

	<u>Direct</u>	<u>Statistical, %</u>
Coolant ΔT	1.22	± 10
Film ΔT	2.21	± 34
Clad ΔT	1.02	± 15
Gap ΔT	1.02	± 41
Fuel $\Delta Pwr.$	1.02	± 14

Fuel Subassembly Data:

Pins/Subassembly	217
Active Fuel Length, in.	36
Fuel Pin Length, ft	7
Plenum:	
Length, in.	40.5
Location	(37, equivalent) top
Reflector, in.	6 (top and bottom)

Other Supporting Data:

Probable Coolant Temperatures	
Inlet, °F	600
Outlet, Core Average, °F	900
Maximum Sodium-Clad Interface (Steady-State, Hot Channel), °F	1125 (orificed)

SCOPE

The Fast Reactor Fuel Assembly Thermal Hydraulic Development Program has not been planned as an independent entity, but rather as one part in the LMFBR Program to provide the thermal hydraulics information required to assure the safe and reliable operation of the fuel and core of the FTR, or of later LMFBR sponsored fast reactors. The role of this program is to serve as the focal point of the thermal hydraulics development programs supporting the FTR. The program will provide specific information obtained with electrically heated and instrumented models of multipin models of FFTF driver and test fuel assemblies. Testing will be performed under specific conditions covering the range of proposed FTR conditions. The data provided will directly support the design and operation of the Fast Test Reactor. To completely evaluate the thermal hydraulic behavior of the reactor system, information is required on specific problems not simulated in these tests (e.g., molten fuel-sodium interactions). The LMFBR test program forms the basis for extending the range of these specific FTR tests to cover a broad range of operating conditions. To attain this goal, the results of many complementary tests from the LMFBR program are required:

- Water flow distribution and mixing test results for single fuel assemblies and the core system
- Reactor instrumentation (e.g., flow probes, thermocouples) test results in water and sodium
- Physics test results on the distribution of void for void coefficients
- Vibration tests of individual core components and component systems in water and in sodium
- Clad chemistry and metallurgy test results (to determine failure modes and surface conditions)

- Life test results of prototype fuel assemblies in flowing sodium
- Fuel failure propagation initiated gas release, flow blockage or loss-of-coolant (and out-of-reactor studies)
- Basic studies of boiling initiation superheat and propagation
- Boiling detector development
- Two-phase flow and critical flow behavior
- Natural convection and pool boiling studies
- Fuel-sodium interaction
- Meltdown behavior
- Primary system component development, pumps, valves, and other system components
- Development of methods of cooling of fuel assemblies during fuel handling and inspection.

A present LMFBR task that may provide a direct complement to this program is the development of bundles of electron bombardment heaters. These heaters have not yet been completely developed for bundle heat transfer tests, but they offer more potential for a higher heat flux and temperature operation than the material properties limited resistance heaters to be used in this program. Since the latter stages of this testing program to support the FTR may require capabilities beyond the resistance heaters developed at that time, the use of electron bombardment heaters to complete the full range of testing is being considered.

DEVELOPMENT TEST PROGRAMLOCAL CLAD TEMPERATURE TESTSOverall Design Problem

There are several major thermal limits in the operation of driver and test fuel assemblies which establish a requirement for accurate laboratory measurements of fuel assembly local temperatures and temperature profiles. The problems can be divided into two interrelated parts, 1) maximum temperatures in clad, clad-fuel interface and fuel; 2) temperature gradients in the single pin, groups of pins and component hardware in the fuel assembly. The reference clad material is Type 316 SS. Temperature limits have been established based on the following phenomena:

- Maximum creep strain, which has tentatively been chosen to be 0.2% after 80,000 MWd/tonne (max) of fuel burnup. (Stresses are considered to arise from thermal strain, fission gas buildup and fuel swelling.)
- Interactions between mixed oxide fuel and clad.
- Mass transfer from clad to coolant (a limit on clad surface temperature rather than maximum clad temperature).
- Rupture strength of the clad.
- Stability of clad material (metallurgically and mechanically).

Based on these limits, with minimum experimental data on the material properties of highly irradiated materials, the clad maximum operating temperature has been tentatively set at 1200 °F. Exceeding this limit does not automatically result in clad failures.* Rather, it means that with presently

* Failures in this case are not restricted to the limited case of breach and release of fission product gas defined for the FEDAL system.

available analytical models based on limited available experimental data, assurance of the behavior of the FFTF fuel assemblies cannot be specified with complete confidence above this temperature. This is especially true for the higher burnups proposed for later cores.

The problem of temperature gradient often accompanies the maximum temperature limit. Typical of this is the creep-strain characterized diametrical growth, where the secondary stress caused by temperature gradients often will be an important factor. However, the temperature gradients of greatest specific interest are those that provide a driving force for single pins or pin groups toward structural deformation. For example, the large radial temperature gradient across the outer pin rows (estimated to range from 80 to over 140 °F) may potentially cause these rows to bow inward to touch adjacent rows. The radial temperature gradients are aggravated by this distortion, and the potential for a failure propagation throughout the fuel assembly due to the high temperature and gradients involved cannot be ruled out analytically. The example given considers only one case in which the gradient (derivative) becomes the critical value rather than the maximum temperature. Other examples, where gradients are important, include thermal ratcheting and thermal fatigue. It is also postulated that clad-sodium mass transport may be affected by gradient as well as surface temperature.

Test Program

The program will examine temperature profiles in FFTF fuel assemblies under specific simulations of normal and structurally distorted configurations. The tests will be performed in a flowing sodium environment with electrically heated and instrumented fuel assembly models. In addition to the direct data provided the designer, the test program will substantiate and improve the thermal analysis models presently used in design.

The first test group will be performed with 7-pin bundles because of the limitations of the present test facility (Small Heat Transfer Facility - Appendix A). Although the size of this test assembly is too small to exactly simulate many important effects, such as the edge row radial temperature gradient in an FFTF fuel assembly, the detailed temperature maps provided will serve as an excellent test of the accuracy of present analytical models. Water mixing and flow distribution tests from 7-pin bundles exactly modeling the sodium test assemblies will provide the hydraulic data required by the analytical model.

To extrapolate the data to the 217-pin fuel assembly, further water mixing studies will be made with larger assemblies (e.g., 37 to 217-pin assemblies) to permit the verified analytical model to predict actual fuel assembly temperatures and temperature profiles.

The 7-pin tests will be followed by large model tests in the Fast Reactor Thermal Engineering Facility (Appendix A). These tests will complete the data on a bundle size sufficient to prototypically simulate the conditions in the FFTF fuel assembly. It should be emphasized that all the test assemblies will not retain the hexagonal bundle shape. In several proposed tests, other arrangements will provide a preferred simulation of the problems.

7-Pin Bundle Testing

a. Test 1a: Edge Row Temperature Gradients and Spacer Induced Hot Spots

Test Purpose

The first 7-pin bundle tests will evaluate:

- Characterization of the temperature profiles between wall and central coolant channels.

- The potential hot spot in the clad caused by the presence of the wire wrap spacer.

The thermal gradient across the outer pin rows of an FFTF fuel assembly has been estimated to range from 80 to 140 °F.⁽⁴⁾ The considerable spread in the estimate is due mainly to uncertainty in configuration (e.g., does the bundle disperse to evenly fill the space available), coolant mixing, and fabrication tolerance buildup. An accurate evaluation of this gradient must be made to determine whether the thermal stresses of outer pin row can lead to strain levels that could lead to bowing distortion. Such a level of deformation is potentially serious since it results in fuel-coolant channel deformation and potentially high clad strain. This may be further extrapolated to be a potential initiating step for single fuel assembly failure propagation.

The potential hot spot in the clad caused by the presence of the spacer appears to be a problem of lesser magnitude. There have been approximate calculations made in Reference 4 which indicate that the hot spot temperature increase in the clad caused by the wire wrap is probably less than 15 °F. This increase is not serious in terms of cladding hot spots. However, in terms of the already high thermal stress levels in the outer row pins, the contribution of any effect that increases temperature gradients must be assessed.

Test Performance

The test assembly for this test is summarized in the following table:

TABLE 3. Test Assembly for Edge Row Gradient and Spacer Hot Spot Test

1. Number of heated pins: 7
2. Configuration: Hexagonal array; 0.056 in. minimum distance between pins

TABLE 3. (contd)

3. Pin diameter: 0.230 in., Type 316 SS
4. Heated length: 36 in., uniform axial heat generation
5. Spacers: Wire wrap, 0.056 in. spirally wound on
12 in. pitch
6. Instrumentation: One clad temperature thermocouple
per pin
Two local coolant temperature
thermocouples in wire wrap
7. Duct to bundle clearance: 0.004 in.

The small spacing is provided to force the assembly to retain a tight (pins in contact) configuration. Increased duct to bundle will be examined in Test IIa.

Presently, in the development, the pins are limited to a single clad temperature thermocouple per pin. Figure 1 shows the instrumentation setup of the first test.

Clad temperature thermocouples in Pins 1, 2, and 7 along with local coolant temperature thermocouples contained in wire wraps characterize the temperatures in two adjacent channels (edge and central). In addition, thermocouples in Pins 1, 3, and 6 provide a measure of the temperature gradient across an outer row pin. Pins 3 and 4 are instrumented to determine the hot spot (directly below) and the circumferential temperature gradient caused by the spacer. In addition, the clad temperature thermocouple in Pin 5 evaluates the temperature profile in the duct corner.

The range of test conditions are based on the flow conditions for an orificed core. Preliminary information indicated that the flow range will be approximately from 380 to 550 gal/min (edge-central assemblies). The first tests will cover the range shown and summarized in Table 4.

NOTE: ARROWS (→) REPRESENT CLAD TEMPERATURE THERMOCOUPLES
 CROSSES (X) REPRESENT LOCAL COOLANT TEMPERATURE
 THERMOCOUPLES INSERTED IN WIRE WRAP SPACERS

FIGURE 1. Basic Test Assembly Instrumentation for Local
 Clad Temperature Tests, Tests Ia, IIa, and IIIa

Test runs will be made at the conditions to obtain temperature profiles over an approximately tenfold range of Peclet numbers centering about the range of the proposed reactor flow condition.

In addition, to assure similitude relationships based on heat flux or temperature are valid, data will be obtained at the following two conditions:

1450	500,000	900
1450	400,000	800

TABLE 4. Test Conditions for Edge Row Temperature Gradient Test Ia

<u>Mass Flow Rates,</u> <u>lb/sec-ft²</u>	<u>Heat Flux,</u> <u>Btu/hr-ft²</u>	<u>Sodium Outlet</u> <u>Temperature, °F</u>
1450	500,000	800
1300	500,000	800
1200	500,000	800
1000	500,000	800
800	400,000	800
500	250,000	800
200	100,000	800
100	50,000	800

Prior to each test, isothermal thermocouple calibration runs will be made and then low heat flux tests performed to check instrument accuracy under imposed flow. At each test condition, steady-state flowrates, bulk and local coolant temperatures, local clad temperature and test assembly power will be established and recorded. Plugging temperature, a measure of overall coolant purity will be maintained below 225 °F. A sodium sample will be removed at the end of each test to provide accurate analysis of oxygen, carbon impurity levels. In addition, methods to check the presence of cover gas entrainment will be developed to provide test monitoring.

Information Provided to Design

The information on the potential hot spot will be directly useful in the fuel design. In the case of the edge row gradient the information from the 7-pin bundle represents a distortion of the 217 pin bundle case. However, the detailed temperature map provided by the test results will provide an absolute measure of the capability of analytical models such as MANTA or COBRA to accurately predict the temperature profiles in the fuel assembly. Complementary water

mixing and flow distribution tests with 7-pin models to provide direct data (mixing coefficients) for the analysis of the heated tests will be performed. After model verification, a water test of a large bundle representing flow conditions of an FFTF fuel assembly will be performed. Such a test will provide mixing coefficients and flow distribution data which will permit the analytical model to predict fuel assembly conditions.

The effect of hot spot magnitude under the wire wrap will be relatively unaffected by the bundle size and will be directly useful to the designer. The measured circumferential profile, however, is affected by flow distribution and mixing and, like the edge row temperature gradient, must be extrapolated with water mixing tests.

b. Test IIa: Effect of Fuel Subassembly to Duct Spacing on Edge Row Temperature Gradients

Test Purpose

The test will evaluate the effect of larger duct to pin bundle clearance spacing on edge pin row temperature gradients.

The spacing between fuel pin bundle and duct may be increased through fabrication tolerance buildup or by differential swelling.* While more in-reactor data on swelling are required to define the bundle duct clearance ranges, the major thermal hydraulic problem is to determine the sensitivity of the edge row temperature gradient to spacing changes. In addition, tests must be performed to verify the capability of thermal analysis models to predict the effect of these spacing changes.

* During certain periods of reactor life, the fuel subassembly may also swell more, thus reducing the duct clearance.

Test Performance

The basic test assembly and test conditions will be identical to those employed in Test Ia, Edge Row Temperature Gradients and Spacer Induced Hot Spots. The major change to be made will be a new flow duct which increases bundle duct spacing from 0.004 to 0.030 in. The test conditions and performance will be identical to those described in Test Ia.

Restraining pins, passing through the test assembly duct, will be used to hold the pin bundle in a tight (in contact) configuration. The pins will also be used to hold the bundle centered within the duct to assure controlled test conditions. Centering will be checked by in-operation radiography. Potential flow distortion from the presence of these small pins appears to offer no serious problem because of the small size of these pins and the small equivalent diameter of the test assembly (~ 0.1 in.). The length between the pins and measurement point represents approximately 80 equivalent diameters, and fully developed flow conditions are predicted.

The use of an unrestrained bundle to examine dispersion or twiggling effects has been considered. However, the differences in mechanical behavior between the test assembly and an actual fuel assembly (e.g., especially end loading and stiffness) would make an application of these results very difficult. In addition, the attractive forces between high amperage current-carrying conductors is predicted to strongly affect bundle configuration by causing the pins to remain in a tight, or in-contact, configuration.

Information to Design

The information from these tests will establish the base point on the temperature gradient across the edge row of pins for the worst or tightest bundle configuration. These data, when combined with that of the preceding tests and water mixing

results, will provide an evaluation of the most temperature gradients across edge pin rows as a conservative function of bundle to duct spacing.

c. Test IIIa - Temperature Profiles and Overheating in Loosely Wrapped Fuel Assemblies

Test Purpose

The purpose of these tests is to determine the temperature profiles caused by spacer redistribution in a loose wire wrapped fuel assembly.

Radiation induced differential swelling in combination with differences in cold work may cause the wire wraps to elongate significantly more than the fuel pin. A secondary contribution may come from stress relaxation of the wires.* Confident evaluation of the amount of looseness (differential elongation) is not possible, due to a paucity of materials properties data and fuels swelling data. However, a conservative estimate indicates that wires may elongate up to 0.25 to 0.30 in. more than the fuel pin at exposure levels expected in first core operation. The safety and design supporting development programs must evaluate whether safe and reliable operation is assured under these conditions.

There are two major thermal hydraulic problems: the first problem, that of a support point shifting to permit pin bowing will be examined in Test IVa, Overheating Caused by Pin Bowing. The second problem, that of changing coolant channel geometry by the accommodation of the excess amount of wire, is difficult to analyze because of the complex arrangement of coolant passage geometry. The detail of description in thermal hydraulic models such as COBRA-II does not permit

* This effect was observed in the out-of-reactor sodium-life tests of a fuel assembly model.

a solution of the problem; nor do mechanics models predict the location of the accommodation (distribution or accumulation) of the loose wire.

The emphasis in this test will be to examine an extreme case, that is, one in which accumulation of wire looseness is made over one pitch length (12 in.) for maximum looseness 0.15 in. excess wire length (based on 0.30 in. excess length over the entire fuel pin). Test conditions are summarized in Table 5.

TABLE 5. Test Assembly for Temperature Profiles
in Loose Wire Fuel Assemblies

1. Number of heated pins: 7
2. Configuration: Hexagonal array, 0.056 in. minimum distance per pin
3. Pin diameter: 0.230 in., Type 316 SS clad
4. Heated length: 36 in., uniform axial heat generation
5. Spacers: Wire wrap, 0.056 in. diam, length 0.15 in. greater than required for zero tension wrap. Wound on 12 in. pitch with excess length accumulated in third spiral, spot welded at end points to restrain accumulation and redistribution.
6. Instrumentation: One clad temperature thermocouple per pin
Two local coolant temperatures per wrap
7. Duct to bundle clearance: 0.020 in. (requires additional clearance to assure fit of excess wire material)

The instrumentation set up will be identical to that of the first test series Ia - IIa (Figure 1). The instrumentation will be at a single axial position midway between the point of support of the spiral loop that accumulates wire looseness. The accommodation of the spacer excess length will be in a single 12 in. spiral with the ends of the spiral spot welded in place to retain all excess in this region.

The range of test conditions are based on the flow conditions for an orificed core. The tests will be initially operated at very low heat flux to assure that severe damage to the test assembly will not occur at the desired operating levels. Test conditions are summarized in Table 6.

TABLE 6. Test Conditions for Loose Wrapped Fuel Assembly Temperature Profile Tests

<u>Mass Flow Rates,</u> <u>lb/sec-ft²</u>	<u>Heat Flux,</u> <u>Btu/hr-ft²</u>	<u>Sodium Outlet</u> <u>Temperature, °F</u>
1450	200,000	600
1450	500,000	800
1300	500,000	800
1200	500,000	800
1000	500,000	800
800	400,000	800
500	250,000	800
200	100,000	800
100	50,000	800

At each test condition, steady state flow rates, bulk and local coolant temperature, local clad temperature and test assembly power will be established and recorded. In addition, plugging temperatures, a measure of overall impurity levels, will be maintained below 225 °F. A sodium sample will be removed at the end of each test to provide accurate analysis of oxygen and carbon impurity levels.

Information Provided to Design

The information will be a direct comparison of temperature profiles between normal fuel assembly configuration and that caused by a conservative accumulation of the excess wire in a single pitch length under conditions of maximum differential elongation. This case does not represent the most probable

behavior of the accommodation of excess wire length; however, if the temperatures and temperature gradients are not excessive in this worst case, the test results will provide strong support for operation under these conditions.

Since the measured temperatures are relative to the previous heated tests (Tests Ia - IIa), no water mixing tests will be performed.

d. Test IVa: Overheating Caused by Pin Bowing

Test Purpose

The 7-pin bundle tests will evaluate clad overheating and temperature profiles in fuel assemblies in which three fuel pins are bowed to contact.

There are several potential causes of fuel pin bowing in FFTF fuel assemblies. Causes, single and in combination, include:

- Movement of support points by the shifting of a broken wire wrap.
- Spiral deformation caused by wire wrap pin mechanical interaction.
- Secondary stresses from temperature gradients across the outer pin rows in the fuel assembly.

There are several spatial types and degrees of bowing that depend on the combination of mechanisms acting to cause the bowing. There is also a temporal ordering in multipin bowing, that is the sequence of bowing propagation through the assembly.

Two pins bowed to touching is the simplest case of bowing analyzed.⁽⁵⁾ This results in local overheating in the clad of the pins at the contact point of approximately 100 °F. While this degree of overheating is not expected to lead to

rapid failure, in the hottest pins, the permissible creep limit will be exceeded and expected life will be decreased. The analysis performed⁽⁵⁾ establishes a reasonable limit for two pin bowing, and no tests are planned in this area.

The evaluation of overheating becomes progressively more complex as the bowing of more than two pins is considered. For example, if the unlikely event of pins bowing in a triad to form a completely closed channel, the coolant enthalpy rise rate in the small channel is approximately twelvefold greater than that of a normal subchannel. If, however, the three pins bow to leave a partially open channel (Figure 2), the enthalpy rise rate drops and coolant mixing becomes a potentially important attenuation effect. Analysis methods to treat this problem are being developed in a present code, COBRA II.

When the problem of evaluating potential propagation through the thermal stress and the mechanical interaction is considered, the location of the bowing in the fuel assembly assumes a high degree of importance. For example, if three pins are bowed together in the interior of the bundle (say at the hottest channel), the major problem is overheating of the coolant and clad in the vicinity of the bowed region. However, if the bowing is of pins in edge rows, the overheating in the vicinity of the bowed position is less severe (lower coolant temperature and pin power), while the gradient across the pins is greatly magnified over the already large radial temperature gradient across that row. In the 7-pin bundle case, the test assembly is not large enough to examine either case with assurance of an accurate simulation. The final results will have to await a test with a bundle model sufficiently large to provide conditions prototypic of a full fuel assembly. This testing will be performed in the Fast Reactor Thermal Engineering Facility in Test VIIIA, Final Pin Bowing Tests.

NOTE: ARROWS (→) REPRESENT CLAD TEMPERATURE THERMOCOUPLES
 CROSSES (X) REPRESENT LOCAL COOLANT TEMPERATURE
 THERMOCOUPLES INSERTED IN WIRE WRAP SPACERS

FIGURE 2. Test Assembly Instrumentation for Bowed Pin Tests

Test Performance

The test assembly for this test is summarized in Table 7.

TABLE 7. Test Assembly for Bowed Pin Testing

1. Number of heated pins: 7
2. Configuration: Hexagonal array, 0.056 in. minimum distance between pins
 - a) distortion: Two outer pins bowed to point contact with central pin (axial point 10 in. from down stream end)
3. Pin diameter: 0.230 in. diam, Type 316 SS clad

TABLE 7. (contd)

4. Heated length: 36 in. uniform axial heat generation
5. Spacers: Wire wrap, 0.056 in. diam with support points shifted to permit bowing (loose wire accommodation in one spiral)
6. Instrumentation: Two clad temperature thermocouples per affected pin
Two local coolant temperature thermocouples per wire wrap
7. Duct to bundle clearance: 0.020 in.

The instrumentation of the test assembly (Figure 2) specifically includes clad temperature thermocouples in Pins 3, 4, and 7 for examining clad temperature at the contact point and for determining pin temperature gradients at this point. The remainder of the pin thermocouples will examine temperatures in the perimeter channels around the affected one. The coolant temperature measurement thermocouples will be positioned to measure temperature 4 in. upstream and 2 in. downstream of the bowed location in each coolant subchannel.

Table 8 presents a summary of the test conditions, with the first two test runs performed at low heat flux to assure that overheating leading to failure will not occur at the main test conditions.

TABLE 8. Test Conditions for Bowed Pin Tests

<u>Mass Flow Rates,</u> <u>lb/sec-ft²</u>	<u>Heat Flux,</u> <u>Btu/hr-ft²</u>	<u>Outlet Coolant</u> <u>Temp (bulk), °F</u>
1450	200,000	800
1450	500,000	800
1300	500,000	800
1200	500,000	800
1000	500,000	800
800	400,000	800
500	250,000	800
200	100,000	800
100	50,000	800

For each test condition, steady flow rates, bulk and local coolant, local clad temperature, and test assembly power will be established and recorded. In addition, plugging temperature, a measure of overall impurity, will be maintained below 225 °F. A sodium sample will be removed at the end of each test to provide accurate analysis of oxygen and carbon.

Information to Design

The major design information will be an initial evaluation of coolant and clad overheating caused by the bowing to contact of three pins. The test results will verify the accuracy of analytical models with water testing to provide an evaluation of the flow redistribution from bowed pins effect in the FTR fuel assembly.

The 7-pin bundle tests are insufficient to provide prototypical data of temperature profiles in edge or internal pin bowing and will be followed with large assembly tests in Test VIIla.

e. Test Va: Clad Temperature Profiles in Bundles of Spirally Deformed Pins

Test Purpose

The purpose of these tests is to provide information of temperature profiles to support stress evaluation in fuel assemblies with spirally deformed pins.

The potential of radiation induced differential swelling causing significant mechanical interactions between fuel pin and wire wrap spacers is considered to be remote for the burnup goals for the first FTR core. However, as burnup levels and coolant temperatures are increased in higher performance operation such interactions appear to be possible. One potential result is that the spacer system may force the pin to assume a spiral deformation, somewhat akin to a corkscrew. The problem is that the strain may exceed the material ductility and that on further stress, the clad could rupture.

The other possibility is that the deformation would become so great that buckling, leading to pin-to-pin contact, could occur. Either result would lead to a potential fuel failure propagation within the fuel assembly.

Even if the spacers retain integrity and support positions, local overheating is expected due to the distortion of coolant subchannels. This will occur despite the fact that bundle overall heat transport may be slightly enhanced by the additional diversion or swirl imparted to the coolant by the new geometry. A second problem lies in the temperature profile in the individual fuel pin clad material. The temperature profiles caused by the spiral deformation cannot be analytically predicted with accuracy or confidence. Therefore, experimentally determined profiles are required to permit a mechanical prediction of whether or not the fuel assembly can safely and reliably operate under these conditions. These data, along with in-reactor fuel swelling behavior and irradiated materials properties, form the basis of the evaluation of the problem.

Test Performance

Two tests are planned in this program, with performance of the second test dependent on successful completion of the first. The experimental thermal hydraulics program must progress through certain critical development stages because of the difficulty in producing reliable heated pin models which will simulate the spiral deformed configuration. The first stage will concentrate on the production of the configuration desired on pins which have already been rejected from use in the other bundle heat transfer studies due to size or other inspection test criteria. These pins will be spirally deformed in stages by excessive wire tension or by the application of a local torque. The pins will then be passed through the quality assurance program to assess their performance capability, with the exception of the original rejection defect.

When the technique of producing deformed pins that appear capable of operation at modest heat fluxes and temperatures,* a single heated pin will be prepared and tested as the central pin of a 7-pin bundle with the outer row of six pins unheated. Table 9 contains a summary description of the test assembly.

TABLE 9. Test Assembly for Spirally Deformed Fuel Assembly Temperature Tests (First Test)

1. Number of heated pins: One spirally distorted to maximum axial deviation to 0.030 in.
2. Number of pins: Seven spirally distorted to maximum axial deviation 0.030 in.
3. Configuration: Hexagonal array; 0.056 in. minimum distance between pins
4. Pin diameter: 0.230 in. Type 316 SS cladding
5. Heated length: 36 in. uniform heat generation
6. Spacers: Wire wrap on 12 in. pitch, spot welded at 2 in. intervals to assure position
7. Instrumentation: Three local clad temperature thermocouples per pin
Three local coolant temperature thermocouples per wrap
8. Duct to bundle clearance: Minimum clearance

In the instrumentation of the test assembly, the data required must be tempered with the ability to install thermocouples. Three thermocouples are considered to be the limit at a single axial position (Figure 3).

The thermocouples are placed at 35° intervals (azimuthal referenced to pin center) from the first directly under the

* Here the values are set by the thermocouple accuracy. For example, if a clad thermocouple is accurate to $\pm 1^\circ\text{F}$, the temperature differences must be on the order of 10°F or more for sufficient accuracy for design purposes.

spacer which contains the local coolant thermocouple.
Table 10 provides a summary description of test conditions.

NOTE: ONLY CENTRAL PIN IS HEATED
ARROWS (→) REPRESENT CLAD TEMPERATURE THERMOCOUPLES
CROSSES (X) REPRESENT LOCAL COOLANT TEMPERATURE
THERMOCOUPLE INSERTED IN WIRE WRAP SPACERS
ALL PINS SPIRALLY DEFORMED

FIGURE 3. Test Assembly Instrumentation for First
Spiral Deformation Test

TABLE 10. Test Conditions for Spiral Deformation Tests

Mass Flow Rates, lb/sec-ft ²	Heat Flux, Btu/hr-ft ²	Outlet Temperature, °F
1450	300,000	800
1300	300,000	800
1200	300,000	800
1000	300,000	800
800	300,000	800
500	100,000	800
200	100,000	800
100	50,000	800

In addition to the above measurements, bulk coolant conditions (pressure, temperature and flow) at the inlet and outlet of the test assembly will be measured. Coolant purity will be controlled on the basis of plugging temperature (measure of overall impurity levels) during test operation and sodium samples will be removed for quantitative analysis of oxygen and carbon levels during test operations.

The results to design from this first test will provide an initial scoping of the problem. If the single heated pin tests are successful, two courses of action will be followed:

- Comparative data will be provided by performing a single-heated pin test in a 7-pin bundle of normal configuration. This will be performed with the assembly used in Tests Ia and IIa.
- Seven-pin heated bundle tests will be performed with spirally heated, deformed pins.

Of these, the latter test is more important since the accuracy of the results must be as high as possible to provide confidence that operation can proceed to increasing levels of burnup. While even a 7-pin bundle does not provide an

undistorted model of the thermal hydraulic environment of the fuel assembly, it will be a great step beyond the first test.

The second test assembly is described in Table 11.

TABLE 11. Test Assembly for Spiral Deformation Tests
(Second Test Assembly)

1. Number of heated pins: Seven spirally deformed
maximum axial deviation
0.030 in.
2. Number of Pins: 19 spirally deformed maximum axial
elevation 0.030 in.
3. Configuration: Hexagonal array; 0.056 in. minimum
distance between pins
4. Pin diameter: 0.230 in. Type 316 SS clad
5. Heated length: 36 in. uniform axial heat generation
6. Spacers: Wire wrap on 12 in. pitch
7. Instrumentation: Three local clad temperature
thermocouples
Three local coolant temperature
thermocouples per wrap
8. Duct to bundle clearance: Minimum clearance
9. Duct heated externally to reduce external temperature
gradient

The instrumentation will be similar to the first test; however, the additional heated pins will make a fuller characterization of clad temperature profile possible (Figure 4).

Thermocouples will all be at a single axial position to provide maximum measurement concentration. This axial position is open at the present time. The three clad thermocouples per pin will be at 15° intervals with one local coolant temperature thermocouple monitoring each coolant channel.

NOTE: 7 INNER PINS HEATED
 ALL PINS SPIRALLY DEFORMED
 ARROWS (→) REPRESENT CLAD TEMPERATURE THERMOCOUPLES
 CROSSES (X) REPRESENT LOCAL COOLANT TEMPERATURE
 THERMOCOUPLES INSERTED IN WIRE WRAP SPACERS

FIGURE 4. Test Assembly Instrumentation for Final
 Deformation Tests

The test conditions will be the same as in the single heated pin test, except that external heating on the duct will be used to reduce the gradient on the edge row to 0 °F (based on the central coolant channel temperature) to 50 °F. A greater range of edge row gradients should be examined, but the limit on pin heat flux may prevent this.

Information to Design

The information from the single-heated pin, 7-pin bundle test will provide a comparison of the temperature gradients in a single spirally deformed and a normal pin. It will also provide an important decision point on whether further testing is valuable to the designer or feasible.

The 7-heated pin bundle test will provide quantitative temperature profiles in deformed inner row pins which can be extrapolated to reactor conditions. The results of these tests, along with reactor fuel swelling behavior and irradiated materials property, will provide the information required for a complete evaluation of whether or not significant spiral deformation can be permitted in FTR driver fuel assemblies or if this condition must be defined as end of life condition.

f. Test VIa: Temperature Rise Caused by Blockage with Foreign Matter by the Shifting of a Broken Spacer

Tests to measure overheating due to a flow blockage of one or more coolant subchannels by either foreign matter or from a broken wire are not being planned under this program. The reason for this action is that the information will be obtained from planned tests at ORNL to support FFTF needs, in this area.

Fuel assembly caused by the bowing together of pins due to the shifting of support points due to the movement of a wire wrap are covered in Test IVa, Overheating Caused by Pin Bowing.

Large Bundle Tests

g. Test VIIa: Final Edge Row Temperature Gradient Tests

Test Purpose

The purpose of this test is to provide final evaluation of temperature gradients across the outer pin rows of FFTF

fuel assemblies. The heated test assemblies will contain a sufficient number of pins to represent an accurate simulation of the flow and temperature conditions of FFTF fuel assemblies.

Test Performance

The present estimates of the influence of neighboring coolant subchannels on the thermal hydraulic conditions on a single subchannel indicate that the simulation should extend about three channels radially from the subchannel of interest. In the case of a structurally distorted channel, the radius of influence is dependent on the specific distortion; however, it generally is larger. In the large bundle tests, one basic test assembly will be fabricated to accommodate all normal configuration and structural distorted configuration testing (Figure 5). The test assembly simulates thermal hydraulic conditions in central pins and edge pin rows. The assembly also provides a reasonable simulation of a single duct corner. The reason for proposing the assembly rather than a 37-pin hexagonal bundle, is that the affect of turning at the duct corners cannot presently be analytically predicted. In the small hexagonal assemblies, the duct corners are very close together compared with the 217 pin bundle assembly, causing a potential distortion of test data. In addition, the subchannels of interest are four or more subchannels removed from a nonprototypical boundary.

In the test assembly, the outlined section is removable to permit insertion of distorted or other off-normal test configurations without complete dismantling of the test assembly.

The test assembly summary descriptions are shown in Table 12.

TABLE 12. Edge Row Temperature Gradient Test Assembly

1. Number of heated pins: 39
2. Configuration: Minimum pin spacing 0.056 in.
(Figure 5)
3. Pin diameter: 0.230 in. clad material Type 316 SS
4. Heated length: 36 in. uniform axial heat generation
5. Spacers: Wire wrap, 0.56 in. spirally wound on
12 in. pitch
6. Instrumentation: Two clad temperature thermocouples
per pin (18 pins)
One thermocouple in 9 peripheral
pins
Two local coolant temperature
thermocouples per wire wrap
7. Bundle to duct spacing: 0.030 in.

NOTES: OUTLINED SECTION (NUMBERED PINS) REMOVABLE
 ARROWS (+) REPRESENT CLAD TEMPERATURE THERMOCOUPLES
 CROSSES (X) REPRESENT LOCAL COOLANT TEMPERATURE
 THERMOCOUPLES

FIGURE 5. Test Assembly for Final Local Clad
 Temperature Tests (39 Pin Bundle)

Clad thermocouples in Pins 1 through 15 (Figure 5) along with the local coolant thermocouples characterize the temperature profiles in edge row pins both in the corner and the flat region. A perimeter row, Pins A-I, will be instrumented to further characterize the edge row temperature profile and to provide information on central subchannel temperature profiles. The other pins will be heated, but not instrumented, providing an environmental rather than active role due to their adjacency to nonprototypical surfaces. The inner duct portion (dotted outline) will be shaped to represent an additional pin row to add to the simulation of peripheral flow conditions.

The test conditions are described in Table 13.

TABLE 13. Test Conditions for Edge Row Gradient Tests

<u>Mass Flow Rates,</u> <u>lb/sec-ft²</u>	<u>Heat Flux,</u> <u>Btu/hr-ft²</u>	<u>Outlet Coolant</u> <u>Temperature, °F</u>
1450	500,000	800
1300	500,000	800
1200	500,000	800
1000	500,000	800
800	500,000	800
500	200,000	800
200	100,000	800
100	50,000	800

To evaluate the effects of gas content, Test Condition 1 will be repeated with argon gas content raised in stages from saturation, 0.05, 0.1, and 0.5% by volume. Metered quantities of gas will be introduced at the test section inlet.

At each test condition, steady-state flow rates, bulk and local coolant temperatures, local clad temperature and test assembly power will be established and recorded.

Plugging temperature, a measure of overall coolant purity will be maintained below 225 °F. A sodium sample will also be removed to analyze oxygen and carbon level.

Information Provided to Design

The major results will be edge pin row temperature profiles which will be directly representative of conditions in FFTF fuel assemblies. The data will also provide verification of the ability of the previous 7-pin bundle and water mixing results to predict full fuel assembly thermal hydraulic conditions.

h. Test VIIIA: Final Pin Bowing Tests

Test Purpose

The purpose of these tests is to provide final evaluation of clad overheating and temperature gradients caused by three pins bowing to contact at the outer row and internally in FFTF fuel assemblies. A description of the design problem is contained in Test IVa: Overheating Caused by Pin Bowing. In addition, if other potential bowing possibilities are shown to exist, the test assembly contains a sufficient number of pins to accurately simulate bowing of more than three pins.

Test Performance

The basic test assembly is the 39-pin assembly described in Table 12. In the first test, two edge row pins will be bowed to contact (spot welded to retain configuration) a second row pin (Figure 6). The support point of one wire wrap spacer will be shifted to permit the outer row pins to bow to contact the inner pin. The major measurements in Pins 11, 12, and 15 are to determine clad temperatures at the contact point and to determine pin temperature gradients at this point. The remainder of the pin-clad thermocouples will be held as in the edge row gradient tests to determine the effect of the structural

distortion on adjacent fuel pins. Local coolant temperature measurements will be made to monitor coolant in each sub-channel 4 in. upstream and 2 in. downstream of the bowed location.

NOTE: ARROWS (+) REPRESENT CLAD TEMPERATURE THERMOCOUPLE
CROSSES (X) REPRESENT LOCAL COOLANT TEMPERATURE
THERMOCOUPLES

FIGURE 6. Test Assembly Instrumentation for Final Pin Bowing Tests (39 Pin Bundle)

In the internal pin bowing test, the test setup is the same with Pins 4, 7, and D replacing Pins 11, 12, and 15.

The test conditions for both tests are described in Table 14.

TABLE 14. Test Conditions for Bowed Pin Tests

<u>Mass Flow Rates,</u> <u>lb/sec-ft²</u>	<u>Heat Flux,</u> <u>Btu/hr-ft²</u>	<u>Outlet Coolant</u> <u>Temperature, °F</u>
1450	500,000	800
1300	500,000	800
1200	500,000	800
1000	500,000	800
800	500,000	800
500	200,000	800
200	100,000	800
100	50,000	800

For each test condition, steady state flow rates, bulk and local coolant temperature, clad temperatures, and test assembly power will be established and recorded. In addition, plugging temperature, a measure of an overall coolant impurity will be maintained below 225 °F. Sodium samples will be removed for analysis of oxygen and carbon.

Information to Design

The major design information will be a direct evaluation of coolant and clad overheating and pin temperature gradients caused by the bowing together (point contact) of three fuel pins in outer rows and internal rows. The data will support evaluations of potential failure and failure propagation mechanisms. The test results will also provide directly applicable information on the thermal hydraulic effect of the bowed triad on adjacent pins to assess secondary effects in potential fuel failure propagation in a single assembly.

VOIDING INITIATION AND PROPAGATION TESTS

Overall Design Problem

Several severe reactor accidents may lead to voiding of the reactor coolant. For these accidents initiated in an

intact single fuel assembly, the cause of boiling mainly arises from the loss of coolant and flow blockage. In the evaluation of these accidents, the major question is the determination of whether the accident can propagate radially to other fuel assemblies in the core. The most probable cause of such propagation is through the combined effects; mechanical interaction with other fuel assemblies caused by transmitted pressure waves on structural distortion of the affected duct, disruption of coolant through other flow channels by the pressure surges caused by the boiling channel, and reactivity changes caused by changes in mechanical configuration of the affected and adjacent fuel assemblies. The knowledge of the initial voiding phase of the accident is crucial to assure the accuracy and/or conservatism of the total evaluation of the accident.

Several quantities must be determined in this initial voiding phase. The major item is the superheat required to initiate boiling. Several experimental studies in liquid metal pool boiling, and forced convective boiling^(6,7) have indicated that liquid metals may undergo a significant metastable, superheating before voiding is initiated. On the formation of the first vapor void, this condition of stored energy in combination with the large volume change of transition (liquid to vapor) can lead to a rapid void growth rate and rapid coolant expulsion. The degree of superheating assumed in the analysis of the voiding phase has an extremely strong influence on the calculated progress of the accident by strongly affecting the maximum pressures, coolant expulsion modes and rates, fuel assembly temperature and stress (clad failure mechanisms), reactivity changes (void coefficients and the time sequence), and response of safety circuits. In a real sense, it even affects the choice of the analytical model with the single bubble model limited in growth by inertia

and interface heat transport becoming a better and better simulation for the elevated superheat cases. The more complex dispersed void models may be required to simulate coolant behavior for the case of low superheat voiding.

General Test Program

The test program will characterize the initial stages of coolant voiding under conditions representative of voiding initiated in a structurally distorted single subchannel at full flow conditions and then coolant voiding initiated due to low flow conditions. Major predicted effects on superheat including clad surface conditions representative of several degrees of reactor exposure, reactor flow rates, gas content and heat flux will be simulated. The effect of surface conditions is expected to play an important role in the degree of superheat required to initiate voiding. The basis for the statement is that the prediction of superheat in liquid metals have the same basis as those used by water boiling experimenters, who characterize nucleate boiling as originating at specific sites on the surface of the boiler. It has been postulated and experimentally verified⁽⁸⁾ that these sites are small pits and scratches on the surface, with certain shapes preferred (e.g., re-entrant cavities) due to the difficulty of filling such sites with liquid. The models used to predict superheat are based on the Gibbs (force balance) and Clausius Clayperon equations, which involve the site surface equivalent diameter as a prime parameter in the energy required to initiate voiding. The models predict a high superheat required for liquid metal voiding initiation because the high wettability of sodium causes much smaller sites to be active (as compared with water or organic fluids).*

* *An analogous illustration of the type of behavior is shown by water when heated in a polished glass container. The boiling bubbles are formed at extremely small imperfections in the surface. This bubble which is initiated only after significant superheat is created, grows at a violent rate, drawing on the excess stored energy in the fluid.*

factors in predicting superheat are the temperature-pressure history of the system⁽¹⁰⁾ which determine active cavity sizes and the amount of gas entrained in the coolant.

The surface condition and material (e.g., wettability) are important and predictive models exist to estimate superheat for a given surface plus the pressure-temperature history of that surface. There is evidence, however, that the surface conditions may change radically during boiling. Test results from Atomics International⁽¹¹⁾ show a distinct reduction of superheat with successive boiling operation until a surface condition existed which produced boiling at very low superheats. The photomicrographs of their surfaces on a pre- and post-test basis verifies a significant change in the surface conditions. The major question is, can significant surface changes occur due to normal exposure of the fuel assembly in the FTR environment which would change the boiling initiation superheat requirement. The program for FFTF cladding has been explored in a preliminary manner; however, no quantitative results can be provided at this time.

The possible causes of surface condition changes in the FTR are a result of a combination of:

- Radiation induced clad material property changes.
- Surface changes due to mechanical and thermal stresses (steady state and cyclic).
- Mass transport of elements from clad to sodium.
- Related mechanical wear rates such as erosion or fretting.

These clad material effects are being examined in other fuel development programs, though not with the intent to characterize potential voiding initiation requirements, and in FFTF-LMFBR chemistry and metallurgy programs. It is not the purpose of the thermal hydraulic test program to perform

research in these areas, but rather to use the results of these programs to establish a range of surface conditions versus exposure time for simulation in the voiding initiation tests. Other effects on superheat such as flow velocity, gas content, are described in the test conditions for the individual tests.

a. Test Ib: Initial Voiding Phase-Single Subchannel Initiation

Test Purpose

The test will determine the initial phases of coolant voiding and expulsion when local coolant overheating occurs due to coolant blockage in a single coolant subchannel. If the FFTF fuel assembly instrumentation (e.g., flow probe sensor and thermocouples) are available at the time of these first tests, their response of and sensitivity to the off-normal conditions will be tested. In addition, if an incipient voiding detector has been developed by the time of this test, its performance will be tested.

In the FTR, it is considered possible that individual flow subchannels may distort or become partially blocked. It is considered much less probable, though still possible, that the blockage or distortion will be sufficient to permit the coolant to reach the saturation temperature. However, the safety evaluation must consider the possibility and be able to evaluate the possible consequences of the accident. The most important evaluation is to predict the potential for failure propagation to other fuel pins in the assembly, and from a single fuel assembly to other assemblies in the core.

The information required by the evaluation of the accident is:

- The degree of local coolant superheating (representative of all fuel exposure times).

- The effect of cover gas entrained in the coolant on superheating.
- The effect of flow rate on superheat.
- Clad temperatures in the distorted region.
- The strain and strain rate produced in the fuel sub-assembly by voiding.
- The strain or strain rate produced on the duct by voiding.
- The rate of expulsion and direction.
- The response and sensitivity of the reactor instrument to the fuel assembly conditions before and during voiding.
- The void distribution during expulsion.

Test Performance

The performance of the test will be divided into two phases. The first phase consists of testing with multipin bundles to obtain the integrated data required for this phase of the accident evaluation. The second phase consists of ten tests with single short pin test assemblies to permit extrapolation of these data over the entire expected range of surface conditions for reactor exposure levels covering the entire proposed fuel assembly life [e.g., up to 80,000-100,000 MWd/tonne (max.)] for full performance operation. The multipin tests are summarized in Table 15.

TABLE 15. Phase I - Multipin Test Assemblies for First Boiling Initiation Tests

1. Number of heated pins: Seven heated (19 pins total)
2. Configuration: Hexagonal array; 0.056 in. minimum spacing between pins, 3 pins bowed together to touching (mechanically held in place)
3. Pin diameter: 0.230 in., clad material Type 316 SS

TABLE 15. (contd)

4. Surface condition simulating clad to: Test Ib-1, beginning of life, Test Ib-2, 0.6 goal exposure for first core
5. Heated length: 36 in. uniform axial heat generation
6. Spacers: Wire wrap, 0.056 in. spirally wound on 12 in. pitch, distorted to permit three pins to bow together
7. Instrumentation: One clad temperature thermocouple for pin
Three local coolant temperature thermocouples over wrap
8. Duct to bundle clearance: Open
9. Other fuel assembly flow resistances simulated in inlet and outlet piping.

The distorted condition to produce the coolant boiling in the single subchannel is a complete bowing together of three pins. The choice was based on controllability of test conditions and probability of occurrence in the reactor. The requirement to examine the effect of clad surface condition on superheat makes this distortion a more controllable one since in the distorted or small channel where voiding is expected, no wire wrap on blockage will be present to possibly prematurely trigger boiling. In addition, the certainty of where boiling will occur reduces the clad surface simulation to a section of three pins approximately 6 inches in length (e.g., distance for the coolant to reach saturation if the pins are bowed together to form a single closed channel).*

A test assembly problem may be premature failure before voiding. The stresses caused by the original bowing will

* In all voiding tests, the outlet piping will contain flow resistance devices (e.g., dummy pins and orifices) to simulate the presence of the fuel assembly gas plenum and reflector region and the instrument package. Flow resistance at the inlet to the test assembly will be treated in a similar manner.

relax before the high temperature conditions are reached; however, there will be a high secondary stress caused by temperature gradients. An initial evaluation of the problem indicated that the assembly should not fail from the cause during the brief duration of the test. In order to be sure before the detailed planning of the first test assembly is completed, more detailed calculations must be performed, based on the low heat flux results in Section A, Local Clad Temperature Tests.

The test assembly instrumentation is shown in Figure 7.

NOTE: 7 INNER PINS HEATED

ARROWS (→) REPRESENT CLAD TEMPERATURE THERMOCOUPLES

CROSSES (X) REPRESENT LOCAL COOLANT TEMPERATURE
THERMOCOUPLES INSERTED IN WIRE WRAP SPACERS

FIGURE 7. Test Assembly Instrumentation for Single
Subchannel Void Initiation Tests

The clad thermocouples in Pins 1, 2, and 7 will examine clad temperature at the contact point and pin temperature gradients. The remainder of the pins will examine temperatures in the perimeter channels around the affected one. The coolant temperature measurements in the wire wraps would be located such that in the reference of 6 in. of affect channel (wire deleted in affected region) a local coolant temperature measurement will be made in all channels, 4 in. upstream and 2 in. downstream from the voiding channel. Consideration has been given to instrumenting an unheated dummy to contain strain and strain rate measurement. With presently available strain measuring sensors, this does not appear feasible (temperature and size problem); therefore, the mechanics portion of the test has not been definitely included. However, at the test fabrication milestone, this issue will again be considered and final decision made. The duct will be instrumented to measure strain and strain rate during the voiding phases.

The test assembly can only be used to evaluate a single clad surface condition due to the surface condition simulation requirement (burnup level) before the distorted triad will be replaced. The test conditions will be based on a simulation of the conditions in full assemblies in the presently orificed core. Test conditions are summarized in Table 16.

TABLE 16. Summary of Test Conditions for Single Subchannel Voiding Initiation

<u>Mass Flow Rates, lb/sec-ft²</u>	<u>Argon Gas Content, vol%</u>	<u>Sodium Inlet Temp, °F</u>
1450	Saturated, 0.5, 1.0	600
1300	Saturated, 0.5, 1.0	600
1200	Saturated, 1.0	600
1000	Saturated, 1.0	600
800	Saturated, 1.0	600

In all tests, the outlet pressure will simulate a constant reactor plenum pressure.

The tests will be performed first by establishing coolant inlet temperature, inert gas content and flow conditions at a minimum test assembly power level corresponding to 600,000 Btu per hr-ft² and then raising test assembly power in measured steps until voiding is initiated. For each test condition, gas content will be varied from saturation to 1.0% void volume by forced introduction at test reaction inlet.

The second tests will be performed by establishing power and reducing flow until voiding is initiated.

The measurements to be made during the tests will be individual pin temperatures, local and bulk coolant temperature, power, pressure from taps at 6 in. increments on the test assembly, flow at inlet and outlet of the test assembly, duct strain and strain rates from externally mounted strain gauge. Void fraction at low void fraction (20%), will be measured by the difference between the two flowmeters. At low void fraction, this method will provide a reasonably accurate measure of both flow and void fraction. Coolant purity will be monitored and analyzed. In addition, response of the reactor safety instruments (non-nuclear) will be recorded during the entire test run.

Before test operation begins, the assembly will be operated for several days under conditions predicted to produce correct pressure-temperature history on the surface. The pin temperatures, local coolant temperatures, flow pressure and strain measurements will be rapid response data and obtained at rapid scan rates (see Appendix B, for data acquisition system description) during the initiation and propagation stage. This is especially important since pressures and pin temperatures will also be monitored and trip set for the safety of the facility and personnel.

The second phase of testing will consist of single pin boiling initiation tests over a range of clad surface conditions covering the full range of full exposure levels. These tests will be used to extrapolate the limited number of multi-pin results on voiding initiation superheat for the entire life of the fuel assembly. Clad simulation is described in the next section (*Surface Conditioning*). The test conditions will correspond to those in the distorted subchannel in the 7-pin bundle tests. Inert gas content of the coolant will be varied through the same range of concentrations as in the 7-pin bundle tests. The major measurement in these tests will be superheat required to initiate boiling.

Surface Conditioning

The importance of simulating surface material and condition in these tests has been previously described. The methods of producing the required heater surface simulations fall into three areas:

- Determining the mechanisms for clad surface condition during exposure.
- Characterizing the surface conditions to be simulated.
- Simulating the conditions in the electrically heated test assembly sheath material.

The first problem, determination of mechanisms, has considered the four possible effects.

Irradiation induced roughening caused by material property changes will be examined from irradiated clad material. Presently, there are indications that the roughness caused is very small in scale on the order of micron diameter pits. Samples of irradiated cladding from FTR fuels tests will provide an indication of the magnitude of this effect.

Surface changes due to stress (steady state and cyclic) and stress relaxation can cause grain size changes and, like the radiation effects, are expected to cause roughening on a micron scale. There are cladding metallurgy studies that will examine these mechanical properties, and samples from these will be used for surface examination.

Mass transport of elements from the cladding to the sodium can also affect surface roughness. It has been indicated experimentally that nickel and manganese are removed from the stainless steel surface by sodium, thus leaving a layer of iron-chrome rich alloy (ferrite). The surface condition of this layer and changes in this layer by further mass transport are of major interest. It should be emphasized at this point that the body of data in this area is limited, and very little data exist from experiments simulating both the reactor system heat input and removal in the cold leg. The effect of temperature gradient in the cladding and cold leg regulation on the concentration of the removed species appears to be primary in the mass transport process. Thus, the remainder of the discussion represents an estimate of the behavior based on limited data.

The ferrite layer may appear rapidly; however, there appears to be sensitizing effects in that experiments have been performed where no ferrite layer was present. The available photomicrographs of the ferrite surface show surface roughness of the micron order;⁽¹²⁾ however, a great deal more work needs to be evaluated before a conclusion can be drawn. After the ferrite layer is formed, two theories exist for further change: one is the mechanism of crystalline break-off of the chromium rich phase proposed by British workers;⁽¹³⁾ the other proposed mechanism is by molecular transport of the iron and chromium layer. The crystalline break-off model should result in a significant change in surface roughness although this has not been observed experimentally.

There are chemistry programs in progress at GE, AI, BNW, ANL, and BNL to examine the mass transport effects although not in specific terms of evaluating the surface condition at various exposure times. These programs will be informed of the thermal hydraulic surface characterization requirements through the offices of the LMFBR Program Office and AEC/RDT.

The effect of fretting erosion and other mechanical wear mechanisms may significantly effect the local condition of the clad surface. Very little data are available for these mechanical wear mechanisms. Test results, which will be directly applicable to FTR fuel assemblies, are from isothermal sodium life tests and in-reactor, small assembly irradiation tests. As these test results become available, they will provide direct input for the thermal hydraulic surface characterization requirements.

The second problem, that of characterizing surface conditions, does not present any real difficulty for the clad test specimens. The standard techniques of photomicrographs (size distribution) and sectioned profilometry (surface diameter and shape) will be adequate to examine maximum site size distribution. The major difficulty, if any, is to produce comparative measurements between the treated, heated pin sheath and clad samples to insure that the surface simulations are adequately reproduced. The direct comparison of photomicrographs and sectioned profilometry appears to be laborious but practical for the limited simulations planned with bundles. Discussions with metallurgy and nondestructive testing sections have indicated several other measurement techniques exist (surface activation energy comparison) that could be modified to meet the specific needs of this program. No selection has been made for this comparative method.

The third problem of simulating clad surface conditions is limited to mechanical and chemical processes for the three

bowed pins. It will be necessary to perform the conditioning on the assembled pin because of the swaging process used in the final fabrication step. The methods considered most practical are mechanical (e.g., grinding, lapping) and chemical (e.g., etching) method. A broad range of surface defect size and distribution can be produced by a combination of these methods. The major problem will not be one of producing the surface condition, but of providing quantitative comparison between them and the clad material samples.

Information to Design

The test results will provide information to directly support the evaluation of the voiding stages of flow blockage accidents. The major data are:

- The degree of coolant superheating to initiate voiding for all levels of fuel assembly exposure.
- The effect of cover gas concentration in the coolant or superheat.
- The effect of initial coolant velocity on superheat.
- Expulsion rates associated with these superheats.
- The response of reactor safety instrumentation (if the non-nuclear instrumentation has been developed at the test milestone).
- Strain and strain rate on the duct and assembly.

The use of 7-pin bundles will distort test results in a conservative manner due mainly to the coolant inventory involved. The much larger void-liquid ratio provides a significant change in the accident rate even with the added row of pins (19 total) to increase the ratio. The reason that a larger bundle is not proposed in these first high heat flux, high temperature tests is that it is felt that 7-heated pins will represent the most reliable test assembly.

b. Test IIb: Initial Voiding Phase - Loss of Coolant
Initiation

Test Purpose

The test will determine the initial phases of coolant voiding and expulsion when coolant overheating occurs in the total fuel assembly by a severe reduction in coolant flow. In addition, the response and sensitivities of the fuel assembly safety instrumentation will be evaluated to examine their behavior in off-normal conditions.

In the FTR, several causes of loss of coolant serious enough to result in boiling to a single and intact fuel assembly may be postulated. The major causes which can lead to boiling are a complete loss of pumping and a major blockage of the inlet or outlet region of a fuel assembly. Those are not considered to be probable accidents, however, their progress and consequences must be evaluated in the safety evaluation of the reactor.

The safety information is the same as that required in Test Ib. These tests will also reduce the range of clad surface conditions already covered by Test Ib.

Superheat is a metastable (nonequilibrium) condition which besides the actual condition at the time of boiling, the condition of the fuel assembly may depend on the transient sequence of events leading to these conditions. For example, superheating to initiate boiling may not be the same for a flow blockage (instantaneous event) and a loss of pumping (slower reduction). This leads to the conclusion that each accident requires a separate transient simulation to assure an accurate evaluation of its progress. This statement cannot be refuted and extends to other phenomena (e.g., expulsion) in the course of the accident besides superheating. However, the experimental techniques and tools do not exist for the

large scale, transient simulation of loss of coolant accidents, nor will they be developed in time for the FTR safety requirements for startup. While transient simulation is planned for the future to directly support increased performance operation, for this case, the approach will be to produce data under controlled conditions to evaluate each phase of these accidents.

Test Performance

The multipin bundle to be used is summarized in Table 17.

TABLE 17. Test Assemblies for Boiling Initiation
Under Loss of Coolant Conditions

1. Number of heated pins: 18 (1 unheated dummy for strain measurement)
2. Configuration: Hexagonal array; 0.056 in. minimum spacing between pins
3. Pin diameter: 0.230 in. clad material Type 316 SS
4. Surface conditions: Based on the results of Test Ib
5. Heated length: 36 in. uniform axial heat generation
6. Spacers: Wire wrap, 0.056 in. spirally wound on a 12 in. pitch
7. Instrumentation: One clad temperature thermocouple per pin
Two local coolant temperature thermocouples per wrap
8. Duct to bundle clearance: Open
9. Flow resistance in outlet and inlet regions to simulate remaining fuel assembly flow resistance

The instrumentation of test bundles will be arranged as shown in Figure 8 for this test. Local coolant temperature will be measured at the downstream end and at a point 6 in. upstream. Pin thermocouples will be placed at the downstream points for Pins 1, 2, 4, 6, 9, 13, 14, 16; and 6 in. upstream for the other pins. Clad thermocouple positions are arranged

to provide measure of superheat within the limitations of the number of clad thermocouples permitted. Pin 19 will be an unheated dummy instrumented to evaluate the strain and strain rate of the fuel assembly to assist in the evaluation of failure modes. There is no instrumentation available at this time to make these measurements; however, by the test milestone, the position will be re-evaluated if present LMFBR sensor programs proceed on schedule. The duct itself will be instrumented to measure strain and strain rate during the voiding.

NOTE: PIN 19 INSTRUMENTED FOR STRAIN AND STRAIN RATE MEASUREMENT (UNHEATED)
 ARROWS (+) REPRESENT CLAD TEMPERATURE THERMOCOUPLES
 CROSSES (X) REPRESENT LOCAL COOLANT TEMPERATURE THERMOCOUPLES INSERTED IN WIRE WRAP SPACERS

FIGURE 8. Test Assembly Instrumentation for Voiding Initiation - Loss of Coolant Tests

Because of the surface simulation, the test assembly will only be useful for the simulation of a single burnup level.

Tests will be conducted by establishing a power level and coolant outlet temperature and reducing flow until boiling occurs. The conditions for each test are summarized in Table 18.

TABLE 18. Test Conditions for Loss of Coolant Voiding Initiation Tests

Heat Flux, Btu/hr-ft ²	Argon Gas Content, vol%
1,200,000	Saturated, 0.5, 1.0
1,000,000	Saturated, 0.5, 1.0
800,000	Saturated, 0.5, 1.0
600,000	Saturated, 0.5, 1.0
300,000	Saturated, 0.5, 1.0

In each test the initial outlet temperature will be established by controlling flow at 800 °F and then flow will be decreased in discrete steps of 50 lb/sec-ft². Gas will be introduced at the inlet to the test section by forced injection. System outlet pressures will be held at a constant simulation of the reactor plenum pressure. The measurements to be made for each test are individual pin temperature, local and bulk coolant temperature, power, pressure (axial taps at 6 in. increment), flow at inlet and outlet of test assembly, pin and duct strain and strain rates (transient). Void fraction will be measured by difference in flowmeter readings. In addition, coolant purity will be monitored and samples analyzed.

As in Test Ia, pin temperatures, coolant temperatures, flow, pressure and strain will be rapid response data and will require special data acquisition system requirements (Appendix B).

Information to Design

The test results will provide information to directly support the evaluation of loss of coolant and flow blockage accident. The major data are:

- The degree of superheat for the worst case of surface condition as determined in Test Ib.
- The effect of coolant flow rate on superheat.
- Coolant expulsion rates.
- Reactor safety system response (non-nuclear instruments only).
- Strain and strain rate on duct and pins.
- The effect of power level on superheat (expected to have weak effect).
- The effect of entrained gas on superheat.

SPECIFIC ACCIDENT TESTS

General Problems

In the evaluation of severe reactor accidents, resulting from power excursion, loss of coolant or flow blockages, specific information requirements exist beyond the initial voiding phases covered in the section on voiding initiation. Such requirements assure the accuracy or conservation of the analyses. In extreme cases where fluid behavioral mechanisms cannot be defined, or extremely nonequilibrium mechanisms are involved, complete transient accident simulations are performed to provide directly applicable information. Although such testing will be required in the future, no such tests are planned at this time.

Test Program

The test program will evaluate the behavior of specific phases of severe reactor accidents. The testing to support the FSAR will not include transient simulations of accidents, though several of the tests will require the determination of rapidly changing variables. For example, in the determination of heat transport during high vapor content cooling, sudden deterioration of the heat transfer coefficient may occur resulting in very rapid changes in flow, temperature, and pressure. The low heat storage capacity of both the fuel pin and test pin (test heat storage will be approximately 50% of the fuel pin) will emphasize the rapidity of response to system changes, e.g., adiabatic temperature (completely insulated) rises in a fuel pin at 14 kW/ft is over 1100 °F/sec. This implies that whereas the transient simulation of an entire accident will not be attempted, the acquisition of much of the data required will require very fast response instrumentation and data acquisition. This is also a requirement to conserve the test assembly and facility from significant structural damage during the course of testing.

Specific Accident Information from LMFBR Programs

The amount of thermal hydraulic information required for complete safety evaluation of the FTR is large. This single program is not intended to provide all of the information needed for the FFTF Final Safeguards Analysis Report. It is dependent upon several other LMFBR safety development programs. The major areas already in progress are briefly described below (Appendix D):

- Boiling expulsion - both analytical models and experimental verification are required. Work mainly performed in ANL and GE development programs.

- Power excursions: (in-reactor tests) GE, BNW.
- Basic superheat prediction and mechanism: AI, ANL, BNL.
- Materials properties (irradiated) and high strain rate data: BNW, ANL, BNL, GE.
- Fission product gas release: ANL, ORNL.
- Critical flow: ANL.
- Safety instrumentation: ANL, BNW, GE.

The purpose of this section is not to summarize all programs, but rather to put the specific BNW testing in proper context and perspective.

a. Test Ic: Cooling After Expulsion

Test Purpose

An important phase of severe (loss of coolant or coolant blockage accident) is the cooling of the fuel assembly after the major portion of coolant expulsion has occurred. Under high void volume fraction condition (e.g., over 20 void volume percent), the heat removal capabilities are expected to be significantly reduced. Clad failure is almost certain under these conditions due to the high temperatures involved. It should be noted that clad failure with release of fission product and fuel could occur previously in the voiding initiation and initial propagation phase of the accident due to pressure and temperature ranges. However, in low superheat cases, the clad may remain intact under the conditions of the initial phase accident. The effect of flow regime changes on duct strain and strain rate will also be investigated.

The determination of the heat transfer removal capability is important from two aspects in the accident evaluation:

- The overall values must be known to determine when temperature levels are reached which could lead to failure.

- The spatial relative values are needed to determine whether simultaneous failure or sequential failure can be expected. Also, to examine if gradient maximum temperature or momentum pressure changes cause the failure.

The determinations are required for a range of void volume fractions from 20 to 90+%.

Test Performance

The test assembly for the high vapor fraction cooling tests is summarized in Table 19.

TABLE 19. Test Assembly for High Vapor Fraction Cooling Tests

1. Number of heated pins: 18 (1 dummy to measure strain and strain rate)
2. Configuration: Hexagonal array; 0.056 in. minimum distance between pins
3. Pin diameter: 0.230 in. refractory or super alloy clad
4. Heated length: 36 in. shaped axially and radially to simulate the reactor
5. Spacers: Wire wrap, 0.056 in. spirally wound on 12 in. pitch
6. Instrumentation: One clad thermocouple per pin
One local coolant temperature in wire
7. Flow resistance of remainder of fuel assembly simulated in inlet and outlet piping

The test assembly will be instrumented with the single clad temperature thermocouple located at the predicted maximum clad temperature position (2/3 downstream of inlet for reactor axial generation curve) with an accompanying local coolant temperature thermocouple opposite. More instrumentation is desired, but appears to be impossible at this time from a reliability standpoint because of the extreme condition of pin operation. A single pin and the duct will be instrumented to measure strain and strain rate. Additional

instrumentation will include pressure taps located at 6 in. intervals to measure steady state two-phase pressure drop in the bundle. Transient expulsion data will not be measured in these tests (see Specific Accident Information from LMFBR Programs under this section).

Quality inlet will be established in a preheat section upstream of the test assembly. Quality will be measured on a heat balance from flow rate (all liquid main loop), coolant temperatures and power input from the preheater and test section. Void volume will be obtained by correlation.⁽¹⁴⁾ The direct measurement of two-phase flow rate and void volume will only be made if developed instrumentation exists at the time of the testing.

Test conditions will begin at low test section powers representing predicted scram levels for an entire range of void fraction percentages from 20 to 95%. Outlet pressure will be held constant, simulating reactor plenum pressures. Flow conditions in the test assembly will be at critical flow (based on outlet pressures of 20 psig) and at 80 and 85% of that value. Power levels will then be increased by 20%, and the tests repeated to cover as broad a range as the test assembly can withstand.

In each test, the desired quality and flow conditions will be established (fixed test assembly power) and measurements of local and bulk coolant temperature, pressure and pressure differential will test assembly local coolant temperature recorded. Strain and strain rate on ducts will also be measured to determine if changes in flow regime will affect duct failure. Coolant purity conditions at or below reactor purity standards will be established prior to the test and checked after the test by direct sampling and chemical analysis.

Information to Design

The important test results will be:

- Overall heat removal rates during high void fraction cooling to predict when failure may occur.
- Spatial temperature distribution within the assembly to predict if failure is sequential or simultaneous.
- Two-phase pressure drop-flow behavior in fuel bundle geometries to assist in the prediction of expulsion rates.
- Strain and strain rates on ducts and pins to evaluate if the change in flow regimes (momentum change) has an effect on structural failure modes.

b. Test IIc: Cooling Under Emergency Cooling Conditions

Test Purpose

The test will evaluate fuel assembly cooling under emergency cooling conditions in the event of a complete loss of flow. During a complete loss of pumping, the mode of cooling the fuel assembly changes completely in character. The power will be reduced quickly by the safety circuits; however, coolant flows will drop to the natural circulation rate (e.g., an order of magnitude flow reduction to 2 to 3 ft per sec). The major information required to support the evaluation of the emergency cooling system is the degree of cooling available in natural circulation, temperature gradient (e.g., edge row gradients), the potential superheat to initiate voiding, void propagation rate, and the rate of heat transport during voiding, and the limits in cooling involved such as the critical heat flux.

Test Performance

The test assembly will be the same as that employed in Test IIb.

The elevation difference between core and heat exchange will not be simulated by the test facility; thus, the major driving force for natural circulation will be different. However, the overall driving force or pressure differential can be predicted and simulated with a reduction in sink temperature or by low level mechanical pumping.

In these tests, flow conditions will be established along with outlet pressure. The test assembly power will be raised in discrete stages with steady state conditions of flow, local and bulk coolant temperature, power, pressure, pin and duct strain will be measured and recorded. Power increases will continue until boiling occurs at which point test measurements will be made (some, such as strain rate under transient conditions) at increasing power until test assembly temperatures are reached which endanger its structural integrity.

The test assembly surface conditions represent a simulation of only one condition of reactor exposure. However, here again, the single pin tests evaluating conditions from a broad range of clad surfaces representing full reactor exposure times will provide a means of extrapolation of these specific pin bundle tests to other conditions.

As in the other voiding initiation tests, a second series of test runs will be made with varying concentrations of cover gas in the coolant (to 1% void volume).

Information to Design

The major information will be:

- Overall heat transfer rates in natural circulation cooling.
- Spatial temperature distribution within the fuel assembly.
- The effect of very low flow on voiding superheat.
- The effect of gas content on low flow voiding superheat.
- Potential heat flux limits in natural circulation cooling.

OPERATIONAL LIMIT TESTING

General Test Purpose

After initial reactor operation, increases in operating performance are proposed to raise the reactor's test capabilities to the upper range of the primary requirements and criteria. To support these performance increases, such as the elevation of coolant outlet temperatures to 1200 °F, or the increase of linear power in advanced cores, a specific body of information will be required in the safety evaluation of the reactor.

In addition, as fuel tests in the closed loops are proposed where failure is possible, specific supporting information is required to assure the safe operation of the reactor under these conditions. This discussion should also mention the proposed closed loop that would contain a coolant other than sodium. If this coolant were to be supercritical steam or a high-pressure gas, an extensive body of information on the interaction between this coolant and the reactor core will be required. However, the schedule for this specific closed loop operation is several years from today, and detailed planning is not merited.

Test Program

The test program falls into two categories. The first is demonstration testing to support increased performance in early cores or plan advanced core operation. These tests would be similar to the local clad temperature tests providing information on fuel-assembly hot-spot temperatures and temperature gradients at elevated temperature (to 1100 °F outlet) or heat flux ($2,200,000 \text{ Btu/hr-ft}^2$). The quoted values were proposed for an advanced carbide-fueled core for the FTR. While the conditions are not firm, the proposed advanced-core fuel assemblies all appear to involve high

heat fluxes to attain desired peak-reaction flux performance levels. The geometries of these assemblies, with the possible exception of spacers, do not appear to be significantly changed from the current design presented in Reference 2.

The second test category is the performance of specific accident testing at conditions corresponding to higher performance operation or accident conditions not covered in the first program years.

a. Test Id: High Heat Flux - Outlet Temperature Operation

Test Purpose

The purpose of these tests is to provide data on hot-spot temperature and temperature gradients to support the evaluation of higher performance operation and operating limits with first cores and the operation and design of advanced cores.

Test Performance

Conditions of proposed test performance can only be tentatively described at the present time. Test assemblies will be of sufficient size (e.g., 37 pins or greater) to permit direct applicability of the data to the fuel assembly. The heat flux-coolant temperature requirements for the assemblies are:

- Heat fluxes of $1,500,000 \text{ Btu/hr-ft}^2$ at 1200°F coolant outlet temperatures for providing data to establish operational limits for first core, higher performance operation,
- Heat fluxes of $2,500,000 \text{ Btu/hr-ft}^2$ at 1200°F coolant outlet temperatures for providing design and operational support to advanced cores.

The instrumentation goals will be two thermocouples per pin to measure temperature gradients; however, the high specific power levels of the test may require a limitation on the amount of in-pin instrumentation.

Tests will be conducted the same way as in the local clad temperature test program, with emphasis on measurement of specific test assembly temperature points and differences under steady-state conditions of constantly increasing power.

Information to Design

The information will be direct temperature profile data to demonstrate steady state operation at increased power and temperature levels for both first and advanced performance cores. These data will also support the establishment of reactor operational limits.

The second part of the test program will provide specific input to the evaluation of accidents in higher performance core operation. These safety evaluations will be required to support any increase in reactor operating conditions.

APPENDICES

The purpose of these appendices are to provide specific supporting information for the test plans described in the main body of the report. The appendices cover the following topics:

Appendix A, Test Facilities, a summary description of the sodium test facilities to be used in this program.

Appendix B, Test Schedule, milestone program schedule.

Appendix C, Interfaces, interfaces with other LMFBF development programs.

Appendix D, References.

APPENDIX ATEST FACILITIES

The test facilities used in this program are:

- Small Heat Transfer Loop (HTL-I), presently in operation, to be used for 7-pin bundle local clad temperature testing (Tests Ia - Va) and the development of high flux, high temperature test pins for voiding initiation and specific accident tests.
- Fast Reactor Thermal Engineering Facility (FRTEF), to be used for all large bundle tests and for those in which coolant boiling is expected. This includes Tests VIIa, VIIIa, and all tests described in Sections titled, Voiding Initiation and Propagation Tests, Specific Accident Tests, and Operational Limit Testing.

The Small Heat Transfer Loop (HTL-I) is a moderate capability, forced convective sodium heat transfer facility which has been in operation for approximately 1 1/2 years at BNW. A summary of the test capabilities of this facility are:

- Sodium flow rate: 50 gpm
- Maximum sodium temperature: 1500 °F at 50 psi (single phase)
- Pressure head (pump) available: 100 psi
- Maximum system pressure (test section outlet):
100 psi (at 1200 °F)
50 psi (at 1500 °F)
- Power supply: 300 kW, (multitapped)
- Heat dump: 300 kW (at 1500 °F inlet temperature)
- Purification: MSAR type OCI system, plus sampling station.

The facility was designed and built by Mine Safety Appliance Research to perform small bundle, low heat flux

testing and in the development of single high flux, high temperature (testing to 1500 °F) capability single pins for later use in voiding tests. It will be used in FY-1970 and 1971 to perform all 7-pin local clad temperature tests. Beyond that period, it will be used to support the FRTEF test program.

The Fast Reactor Thermal Engineering Facility (FRTEF) will be a large capability--a forced, convective sodium-heat transfer facility where all large bundle and safety supporting tests will be performed. Design of the facility will begin in FY-1970, and the facility will be ready for testing in the first half of FY-1971.

The list of capabilities of this facility are:

- Sodium flow rate: 600 gpm in the main loop
175 gpm through the test section
- Maximum sodium temperature: Initially 1200 °F,
ultimately 1800 °F (test section outlet)
- Pressure head (pump) available: 200 psi
- Maximum system pressure (pump outlet): 250 psig
- Power supply: 1.5 MW_e, multitapped
- Heat dump: 5 MW_t (at 1200 °F inlet temperature)
- Purification system: MSAR type OCI system, 10 gpm
at 1200 °F

The facility will be designed and fabricated to permit operation over a wide range of flows and temperatures with up to 1.5 MW_e fuel pin electric resistance heating for evaluating the heat transfer characteristics of fast reactor fuel assemblies. The first series of tests will use simulated FFTF fuel assemblies of up to 39 pins to evaluate standard and off-standard nuclear fuel assembly operating conditions. Provision will be incorporated into the initial design to allow safety testing up to 1800 °F (test section outlet temperature) by providing

at a later date additional equipment and piping, and an increased capacity of electrical power supply.

The data acquisition system will be a general purpose system designed for use in conjunction with several large sodium test facilities. Since the FRTEF will be the first large test facility at BNW, its capabilities have been established to meet the needs of this facility and the Thermal Hydraulics Program. The planned capabilities of the system will be:

- A general purpose digital controller with up to 16,384 words of core storage. Word length will be up to 16 bits (nominal) and memory cycle time as low as 2.5 sec (nominal). This equipment will include a tele-printer input/output unit, binary input/output interface, and a paper tape input/output unit.
- An analog input system including a 125 to 200 channel multiplexer, signal conditioning differential amplifiers with high common mode rejection, and an analog-to-digital converter.
- A magnetic tape memory transport and associated interface capable of direct access block transfer from the digital controller. The transport will have up to a 60 in./sec, 556 bits/in. rating.
- A fixed head disc memory with up to 100,000 words of storage and block transfer capabilities.
- An up to 500 character cathode-ray tube display system for alpha-numeric character display with associated interface for operation from the digital controller.

APPENDIX BTEST SCHEDULE

The milestones in the schedule (Figure B-1) represent completion of the test number marked in the triangle. Since the accuracy of scheduling must decrease, the scale change from months to quarters is made beyond FY-1970.

FIGURE B-1. Program Test Schedule Thermal Hydraulic Development

APPENDIX CINTERFACES

The major interfaces to the heated fuel assembly model tests in sodium are the water mixing and flow distribution studies. The water tests will provide:

- Flow distribution and mixing information from 7-pin bundle duplicating the geometry of the heated tests. The data, taken over the same Reynolds Number (flow modeling), will be directly used in the analytical model to predict the heated bundle temperature profiles.
- Flow distribution and mixing from 37, 217-pin bundles duplicating the test feature of the 7-pin sodium bundle test, but on an assembly containing enough pins to be directly applicable with the analytical model to predict fuel assembly temperature profiles.
- Mixing data on fabrication prototype fuel assemblies in a radially restrained (tight bundle) and unrestrained condition to obtain basic data on the effect of dispersion of the pins within the duct.

The other interfacing tasks have less direct dependence on the heated tests than the complementary water mixing studies. These interfaces are shown in Figures C-1, C-2, and C-3.

FIGURE C-1. Fuel Assembly Thermal Design

THE TESTING DESCRIBED IN THIS REPORT
SUPPLIES ONLY INCIDENTAL INFORMATION
TO THE EVALUATION OF POWER EXCURSION
ACCIDENTS

FIGURE C-2. Safety Evaluation

THE TESTING DESCRIBED IN THIS PROGRAM DOES NOT SUPPORT THE ESTABLISHMENT OF THESE OPERATING LIMITS

FIGURE C-3. Establishment of Reactor Thermal Hydraulics Operating Limits

APPENDIX DREFERENCES

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