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WANL-TME-1352
December, 1965

Westinghouse Astronuclear Laboratory

NOTES OF MONTHLY DEVELOPMENT TEST PROGRAM REVIEW MEETING
HELD WITH SNPO AT CLEVELAND ON DECEMBER 9, 1965
(Title Unclassified)

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HELD WITH SNPO AT CLEVELAND ON DECEMBER 9, 1965

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WANL-TME-1352
December, 1965

NOTES OF MONTHLY DEVELOPMENT TEST PROGRAM REVIEW MEETING
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Group-1 Excluded from Automatic
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Notes of Monthly Development Test Program Review Meeting
held with SNPO at Cleveland on December 9, 1965

Attendees:

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The Monthly Development Test Program Review Meeting was held on December 9, 1965, at the SNPO offices in Cleveland.

This was the first meeting in which the order of tests discussed was based on tests for a specific reactor, general development and facilities rather than a sequential description of all of the test series by number. Tables I and II show brief summaries of the tests in each of the reactor classifications. This tabular form summary has been requested in future meetings from both the Fluid Flow and Engineering Mechanics Laboratories.

TABLE I

NOVEMBER PERFORMED

DECEMBER PLANNED

Tests for NRX-A4

HHT-2 - A-2 and A-4 simulation
 HHT-4 - Block from NRX-A3 + 20 minutes
 HHT-5 - Center hex element (2 tests)

HHT-2 - NRX-A4 simulation to be continued

Tests for NRX-A5

HHT-1 - Quality Control
 HHT-2 - Single hole
 Thermocouple tests
 FFL-21 - Support block pressure distribution
 FFL-27 - Instrument port test

HHT-2 - Single holes
 HHT-4 - Skirtless block, thermocouple in tie rod button
 FFL-21 - Finish tests
 FFL-27 - Complete tests on four seal blocks

Tests for Post NRX-A5

FFL-18 - With A-6 scarf joints

FFL-18 - Water tracer tests on scarf joint and hydrogen gas flow tests with ramp clearances

General Development

HHT-6 - Interstitial test 2-inch specimen
 HHT-7 - Rig build with new springs & heater tube test
 HHT-17 - Undercut corrosion

HHT-2 - Direct gas inlet
 HHT-6 - Full length gap test
 2-inch coating tests
 2-inch 6 mil low pressure
 HHT-7 - Test
 FFL-15 - Pressure distribution

Facilities

Pump vaporizers
 WANHES dewar
 Thermocouple furnace

New A-2 furnace

TABLE II

NOVEMBER PERFORMED

DECEMBER PLANNED

Tests for NRX-A4

EML-69 - Support Block

Tests for NRX-A5

EML-69 - Support Block and Washer Strength
 EML-74 - Support System Structural Integrity
 EML-75 - Friction and Wear of ZrC and NbC
 EML-76 - Control Drum Scram Spring and Bimetallic Joint
 EML-88 - Control Drum Bow Tests
 EML-94 - Start Cluster Cocking Test
 and Braze Key to Element Strength

EML-69 - Support Block with Various Washer Designs
 EML-75 - Friction and Wear Between Pyrotile and Element
 EML-76 - New Control Drum Scram Spring Design and Hollow Tie Rods
 EML-86 - Lateral Support System Integrity
 EML-94 - Complete Cluster Cocking Test

Tests for NRX-A6

EML-60 - Radial Stepping Gap
 EML-69 - Skirtless Block
 EML-92 - ZTA and Be Thermal Contact Resistance

EML-69 - Cup and Block Corrosion Tests
 EML-86 - Procurement of Lateral Support System Hardware
 EML-92 - ZTA and Be Thermal Contact Resistance

General Development

EML-61 - Graphite Tie Rods Tensile and Flexure
 EML-70 - Tab Tip Designs

EML-61 - Continue Graphite Tie Rod Flexure
 EML-70 - Skirtless Tab Tip Design, Load Sharing Characteristics of Cluster and Elements

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NRX-A4 TESTS

EML-69 Support Block Mechanical and Thermal Testing

The NRX-A3/A4 block design, consisting of the 20 mil seat radius and coated pyrographite washer, was tested in the constant load thermal cycle used for previous tests. The tests showed that fine cracks occurred for loads of 700 lb or above when tested in the thermal cycle.

A photostress test series was recently completed on the NRX-A3/A4 design. The results of the test showed that a peak longitudinal tension stress of 1.06 times the average bearing stress and a peak circumferential tension stress of .70 times the average bearing stress exists in the seat radius. Another peak stress occurs on the surface of the inner row of flow holes at an angle of 45° below the peak fillet stress. This peak is equal to .58 times the bearing stress. The location and magnitude of these peak stresses explains why a crack that begins at the fillet radius tends to run at a 45° angle to the horizontal as seen in sections of blocks tested in our laboratory.

HHT-2 Single Element Process Development and Evaluation Tests

Single fuel element corrosion tests were made to simulate NRX-A3 and NRX-A4 temperature time histories. The NRX-A4 simulation was almost 90 minutes long and the elements lost about 12 grams.

HHT-4 Single Cluster Development Tests

The main effort on HHT-4 cluster tests was to determine the capability of a support block with a coated bore taken from NRX-A3 which had logged 16 minutes of full power operation as well as approximately 13 minutes of low power operation. The goal was to obtain a total of 40 minutes of full power operation. The attainment of this goal would have meant the testing of this block in the cluster furnace for 24 minutes. In a series of five tests, a total time of 20 minutes of cluster testing has been obtained. While the block is still integral,

its condition was such to preclude additional testing. It is now ready to be sectioned for metallographic studies. Because of its high radiation level attained in the actual reactor test, difficulty will be experienced in sectioning this block. It will have to be done in a hot cell type of operation. The first cycle consisted of running this block for approximately 16 minutes at full power. The cluster was then disassembled and inspected. The next four trials were made with two hollow tie rods rather than the previously solid tie rods and difficulty was experienced. This is evident as only a total of four minutes of operating time in four trials was obtained.

HHT-5 Unfueled Element and Tie Rod Channel Corrosion Simulation

Center Hex Element Tests were made with two sets of the center hex element with all of its hardware. The purpose of these tests is to establish the lack of corrosion on the internal bore of the center hex where it is insulated with pyro graphite tubes held in place by a stainless steel liner which provides the cooling annulus for the tie rod. The first test was made with a slightly tapered tube and as a result the inlet or upstream end became overheated in the course of the 40 minute test. The degree of overheating was such that the spring lost its resilience. The second test was made with a modified tapered heater which closely simulated the reactor temperature distribution. This was a 20+20 test; in other words, there was a planned power interruption between the two 20 minute runs. While this center hex has not yet been sectioned, overall observation indicates that there is practically no corrosion. Even the first one which overtested the inlet, showed very little corrosion with the division lines between the pyro tubes hardly being recognizable.

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NRX-A5 TESTS

EML-69 Support Block Mechanical and Thermal Testing

The NRX-A5 design, consisting of the 60 mil seat radius and the candidate ATJ washer of 310 mil thickness and 70 mil radius, were tested in the constant load thermal cycle. This basic design did not crack when tested for loads under 900 lb. This is about 200 lb higher than results with the NRX-A3/A4 design.

Alternate washer designs were tested to further increase the cracking strength of the NRX-A5 block. One successful method employed a stack of pyrofoil washers placed between the seat and the support washer. The results showed that crack initiation loads could be raised to 1400-1500 lb in the constant load thermal cycle for the NRX-A5 design. The radius of the support washer was also found to affect the cracking strength when pyrofoil is used. The best results were obtained to date with a 90 mil radius support washer. Recent tests have been run to develop the pyrofoil as a seal around the pyrosleeve in the center hole of the support block. Effective sealing is obtained simply by cutting the inner diameter of the pyrofoil to provide a one mil interference on the pyrosleeve.

The NRX-A5 support block has also been tested to ultimate at 4300°R with the 70 mil radius, 310 mil thick, ATJ washer. The design had an ultimate strength of 5800 lb and 6380 lb when pyrofoils were used. Figure 1 shows the results after testing of a specimen with an ultimate strength of 5800 lb. These values are below the NRX-A3 design strength with the difference accounted for by the reduction in bearing area resulting from the seat radius change from 20 mils to 60 mils.

Room temperature tests with the ATJ washers have been unsuccessful because the washers fail under loads over 1650 lb. Similar failures have been found in furnace tests using the constant load thermal cycle. In these tests the washer will fail under 1200 lb when the temperature reaches 2000°R. Current tests are using pyrographite washers with a 70 mil radius to overcome the washer failures. Other washer designs and material combinations are also planned for testing.

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ENGINEERING MECHANICS LABORATORY

DATE 11-12-65 EML NO 69

TEST NO _____ RUN NO _____

SPECIMEN A-5 SUPPORT BLOCK A-49086

TEST ULTIMATE STRENGTH TEST

FAILURE 5800 LB.

OTO R. WASHER

0 1 2 3 4 5

FIGURE 1

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EML-74 Reactor Axial Support System Tests

The NRX-A5 reactor support system was tested to determine the maximum thermal stress in the support cone at operating temperature. This stress results from the differential compression that occurs between the support plate and support cone when they are cooled from ambient to operating temperature. The test was performed in the support system test rig. The reactor temperature conditions were approximated by performing the test with the components immersed in liquid nitrogen (see Figure 2). Since the test was performed at 140°R whereas the operating temperature of the components is about 200°R, the measured stresses were higher than those expected during reactor operation and had to be adjusted for the overtest conditions. The adjusted maximum stress was about 50,000 psi. This stress occurred at the position shown in Figure 3. The maximum calculated stress was 35,000 psi. However, this value was based on uniform loading of the cone flanges and did not consider the effects of stress concentration at the flange bolts. Since the maximum measured stress is substantially below the allowable stress for the cone material (120,000 psi) this component should function satisfactorily during A5 reactor testing.

An overload test of the support plate, support cone and associated attachment hardware was performed at room temperature to determine the maximum load that could be carried without permanent set. No stresses above the yield point or excessive deflections were observed at loads up to 200% of the design load. Some elastic separation between the Z ring flange and reflector sectors was observed (0.007 maximum).

EML-75 Friction Test of Reactor Components

Further friction and wear testing of the ZrC filler strip coating originally specified for the NRX-A5 was performed to investigate the severe coating loss reported last month. Tests with dried hydrogen and extensive baking of the furnace components revealed that the coating loss could be eliminated if the

REACTOR SUPPORT SYSTEM
TEST RIG

FIGURE 2

View A-A

FIGURE 3
CRITICAL STRESS AREA - CORE SUPPORT CONE

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water vapor in the test atmosphere could be held to less than 80 ppm. However, when the water vapor reached about 130 ppm, coating damage again occurred. Since there was concern as to whether the extremely low water vapor conditions required to avoid damage might be exceeded during reactor testing, additional tests were performed with other coatings.

Pyrotile specimens and filler strip specimens with NbC paint sprayed coatings were heated to 1950°R in hydrogen and subjected to the relative motions expected during ten reactor test runs of ten minute duration. No damage of the coatings was observed at the conclusion of the test. The average static coefficient of friction was about 0.3. A filler strip specimen with vapor deposited NbC coating was also tested in the same manner. This specimen was cut from a bar which had been coated on all sides and then chemically etched so that only one surface remained coated. No coating damage occurred during this test. The static coefficient of friction was about 0.18. The results of these tests indicate that filler strips with NbC coating should function satisfactorily during the A5 tests.

A photograph of the specimens after testing is shown in Figure 4.

EML-76 Reactor Hardware Static and Dynamic Tests

a. Drum Spring

Tests of the NRX-A5 control drum scram spring were performed to determine the variation of torque with spring angular displacement. This spring is considerably stiffer than the spring used in previous reactors and is designed to meet flight reactor requirements. Two springs made to the same specifications were tested to determine the affects of dimensional tolerances on the spring characteristics.

Curves of the results obtained are shown in Figures 5 and 6. The solid curves in these figures show the results obtained with the parts assembled as supplied. These curves indicate that the winding and unwinding torques were

FIGURE 4

FIGURE 5

FIGURE 6

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somewhat different and the spring rate began to increase after the spring was displaced about 120° . This was attributed to coil rubbing caused by lateral movement of the spring relative to the drum bearing shaft as the spring was wound.

After the initial tests were completed, an examination of the components revealed that the locking tab of the bushing which holds the stationary end of the spring had deformed and permitted the spring to move relative to the drum bearing shaft. A positive friction lock was then incorporated in the bushing to securely hold the spring in the desired position and the tests were repeated. The dotted curves in Figures 5 and 6 obtained in these tests show that this change reduced the variation of spring rate with displacement and brought the tightening and loosening characteristics of the spring into close agreement.

Further testing will be performed to measure the maximum stresses developed in the spring and establish whether torque relaxation or spring failure will occur under repeated cycling at operating temperature conditions.

b. Hollow Tie Rods

The hollow tie rods for use with thermocouples in NRX-A5 have not been received. The scheduled date for delivery to Engineering Mechanics is December 15. Four of the twelve tie rods which we expect to receive will be tested for tensile strength and elongation at failure. The remaining eight rods will be tested for fatigue life. These rods were made with material which was vacuum melted and is expected to have improved elongation characteristics compared with the characteristics of the material which was used to make the hollow tie rods which were rejected at the time of the NRX-A4 assembly.

c. Instrument Port

Pressure and mechanical impact tests were conducted on the instrumentation port assembly which is planned for use in the A5 reactor. This design includes a bimetallic, brazed joint and two pressure welds. The brazed joint showed no adverse effect due to the pressure and impact tests.

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The ability of these welded and brazed joints to maintain their seal and structural strength through thermal cycling and mechanical vibration will be demonstrated during the month of December.

EML-86 NRX-A5 Lateral Support System Component Test

The test to evaluate the integrity of the NRX-A5 lateral support system will be conducted during the month of December. During the past period all test specimens and rig parts were received. Many of the test rig parts have been assembled. This assembly includes a nine inch section of the lateral support system with A5 seals, plungers, springs and associated hardware. The core will be simulated with a solid slug of graphite.

EML-88 Insulated Control Drum Bow Test

Control drum bow tests were performed with an NRX-A5 type control drum which incorporates a cooling slot diametrically opposite the control plate slot. Helium gas flows were used to simulate the startup hydrogen flow ramps expected during reactor testing. Three tests were performed with an initial drum temperature of 590°R and flow ramps of 1 , 2 , and 3 lb/sec² to determine the effects of flow ramp. Two additional tests were performed with a flow ramp of 1.0 lb/sec² and initial drum temperatures of 660 and 700°R to determine the effects of initial drum temperature. The results of these tests shown in Figure 7 indicate that the bow during any of the anticipated A5 startup conditions should be very low (0.009 - 0.015 inches).

EML-94 Buffer Periphery Testing

The objective of this test is to evaluate the capability of the buffer periphery design. This series of tests will measure the integrity of the filler

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Control Drum Bow Tab Results

FIGURE 7

strip ledges and keys as well as the tilting effect which will occur on the irregular support blocks.

During the month of November a series of single key and element brazed joints were tested and shown to be capable of carrying loads between 130 and 170 lb. The maximum load which is expected in service is about 70 lb.

The test to evaluate the complete assembly including the core band will be conducted during the month of December.

The testing to measure the tilting effects of the unbalanced support load is being conducted in the rig shown in Figure 8 . Three blocks have been tested; two other blocks will be tested next month. The maximum tilting observed to date was measured in the test of block J5. The dial indicators used to measure this tilting are shown in Figure 9. (note at hole location J9 the deflection was about 3/32 in.)

HHT-2 Single Element Process Development and Evaluation Tests

A large number of tests on single hole (Code 3) peripheral non-fueled elements were performed. These were one hole elements radiantly heated. Four radiant heater tube and shield combinations, progressively increase the improvement in the axial temperature distribution. In the last step of this evolution, the simulation is practically perfect (Figure 10). These tests have evolved a quality assurance test for single holed elements. In addition, two sets of thermocouple tests were made on 32 and 45 inch thermocouple stations. All of these tests were made in the A-2 furnace configuration modified for low flows and using a radiant heater. This is the configuration that the furnace has had for the past six weeks.

FFL-21 Tie Rod Hardware Flow Tests

This is basically a flow test made with air as the working medium at simulated Reynolds Numbers to that expected in the actual reactor support block. The purpose of this test is to determine the pressure difference that exists between the inner most row of coolant flow tubes and the pressure at the corner of the

counterbore where cracks are anticipated to start. The general conclusion from this test is that as long as any flow exists in the fuel element flow channels, a definite pressure drop is available; that is, the pressure in the coolant flow channel is higher than that in the pressure tap at the counterbore fillet location.

FFL-27 NRX-A5 Instrumentation Port Qualification Tests

Leakage tests have been made on two blocks and thermal gradient tests have also been made on two blocks. Four more will be tested for seal leakage.

Future Work Planned for December Under NRX-A5

1. Two more batches of Code 3 single hole elements will be tested and support block pressure distribution will be continued and completed.
2. The construction of a cluster with a skirtless block and a tie rod button thermocouple will be started. Testing will be either late December or early January.
3. Four more instrumentation port blocks will be tested.

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FIGURE 8

Seven Cluster Test Rig

FIGURE 9

Dial Gage Arrangement for J5 Support Block
 Cocking Test

Comparison of Experimentally Obtained
Axial Temperature with Analytically
Predicted Distributions for Peripheral Elements

FIGURE 10

NRX-A6 TESTS

EML-60 Core Effective Gap Test

A series of tests were performed to determine the size of the radial step between the adjacent filler strips in the NRX-A6 design. This information is necessary to predict the leakage between the filler strips and seal segments. Gaps were measured at lateral bundling pressures of four and eight psi using rubber bladders to bundle the core. Duplicate tests were performed using simulated seal segments under the bladders to determine their effect in "flattening" out the gaps. Dial gage measurements were taken on both sides of each filler strip at three axial core levels.

It was found that the gap between adjacent filler strips varied from 0.1 to 4.0 mils. The average gap is approximately 1.0 mil. No apparent flattening effect was indicated in the results of the tests using the simulated seal segments. The data summarizing the test work is shown in Figure 11.

EML-69 Support Block Mechanical and Thermal Testing

The NRX-A6 support block design has been tested for crack initiation load and ultimate strength. The crack initiation tests showed that the design will crack in the seat radius under loads of 1500-1600 lb when tested in the constant load thermal cycle test. In these tests the corrosion cup and ATJ washer will crack but no damage is done to the NRX-A6 support block.

In the ultimate tests it was found that the design will fail at loads of 2100-2560 lb, at room temperature, and at 4340 lb at 4300°R. In all cases the outer cup and washer shattered without damage to the support block as shown in Figure 12.

Tests are currently underway using coated corrosion cups to determine any change in strength as a result of coating the thin ATJ cups.

FIGURE 11

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SIGNATURE *R. Schaff*

DATE 10-29-65 CURVE NO. 13

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FIGURE 12

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EML-86 NRX-A6 Lateral Support System Component Test

Detailed drawings of the components and a partial length reflector section were completed during the month of November. Procurement of this hardware is taking place at this time. Layouts of the test fixtures and assemblies are complete. Details of these assemblies will be completed in December. Figure 13 shows the schematic of a test rig which will be used to conduct these tests. The objective of the test is to evaluate the A6 lateral support system which is significantly different from the previous reactors. We will obtain a hysteresis curve to show the force vs displacement characteristics of the A6 design. This test fixture also includes the capability of measuring the leakage through the peripheral seals while the tests are being conducted i.e. we will be able to measure the flow through the circumference seals in the lateral support system while we are displacing the core relative to the outer reflector and the lateral support system. Wear characteristics of the components in the design will also be evaluated in this test program.

EML-92 Thermal Contact Resistance Test

The initial tests to determine the thermal conductance coefficients for the NRX-A6 lateral support system materials were carried out during this period. Figure 14 shows the thermal gradient vs distance for heat flow through ZTA graphite and beryllium separated with a 1/10 inch of graphite foil. These data were taken in a vacuum with 100 psi contact pressure. Tests will be conducted during December at other contact pressures and at other temperature levels. We also plan to measure this thermal conductance in a helium atmosphere.

FFL-18 Single Seal NRX-A5 Tests

During November, tests with the A-5 type scarf joint shown on Figure 15 were made. This is a support system flow test in which various types of scarf joints are being tested for their leakage behavior. Tests have been made with

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and without scarf joints in the seal section. Verification of the flow behavior was obtained by cementing the scarf joint. The cemented scarf joint checked the seal without the joint to within 3 percent. Tests were also made with sharp corners on the joint and with ramp clearances of zero and 2 mils. Additional ramp clearances and flow visualization tests are planned in December. Flow visualization tests using water, dye tracers and transparent sides on the flow model will be made.

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NRX-A6 LATERAL SUPPORT SYSTEM TESTS

VIBRATION TEST

STATIC & SEAL LEAKAGE TESTS

FIGURE 13

FIGURE 14
Thermal Contact Resistance

Seal Segment Scarf Joint Geometry

FIGURE 15

GENERAL DEVELOPMENT TESTS

EML-61 Tie Rod Fatigue Properties

Additional graphite tie rod tests were conducted during the past month. Two tensile strength tests which provided ultimate load results of 2300 and 4000 lbs.

We also measured the flexural fatigue strength of the material and found that it would endure several millions of cycles without a failure. The maximum stress induced in the specimens during these flexural cycles was in excess of 20,000 psi. The specimens used were taken from the rods which have been previously loaded to failure in the tensile test.

EML-70 Blockless Support Systems

The tab tip of the candidate blockless cluster design has been tested under tension loads to simulate conditions caused by differential expansion. Two of the specimens had ultimate strengths of 158 lb each with failures occurring in the elements rather than at the joint. Two other specimens with obviously poor bond joints failed at 89 lb and 43 lb. All of the specimens used P-514 graphite cement and were graphitized for these tests.

The candidate design of the corrosion cup for the skirtless tab tip was tested under non-uniform as well as uniform loading conditions. The uncoated corrosion cup went as high as 2400 lb in the uniform loading condition, 1450 lb with three elements taking the load and 1000 lb with two of the elements taking the load. In all of these tests the cups failed without damaging the tab tips or the elements.

A single test of a coated corrosion cup was completed for the case where two elements were taking the load. The coated cup failed at 775 lb as compared to 1000 lb for the similar case with an uncoated cup. This test will be repeated when additional coated cups are available.

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The compressive strength of single hole and 13 hole center elements was also tested for the case where the center element would carry all of the cluster load. The single hole element went as high as 1865-2550 lb and the 13 hole element went to 1500-1570 lb before failure. These tests show that the center element can sustain all of the reactor loads if required to do so.

Assembly of a bundled cluster arrangement is in progress to determine the load-sharing between clusters and elements as axial displacements occur. This test will be invaluable in establishing new criteria for blockless clusters and in analysis of current designs.

HHT-6 Interstitial Corrosion Tests

Interstitial corrosion tests on the two inch long, one inch wide, six mil high gaps were made with the following material: one side coated with niobium carbide, one side exposed graphite, tested in hydrogen for 30 minutes at 3500°F, coating removed over one third of area. One side coated with niobium carbide, one side with zirconium carbide did not spall or chip as had samples of this coating prepared for other activities.

HHT-7 Filler Strip Corrosion Tests

The status of the filler strip corrosion test model was reviewed. The model is presently completely finished and being installed. Testing is expected to begin on or about December 17. This model is a hot buffer configuration with selected filler strip coated with zirconium carbide. Several new features are included such as metallic springs to provide seal ring pressure, modified high performance radiant heater, and a sealed and impregnated outer casing.

HHT-17 Undercut Corrosion Tests

The small scale test to simulate fuel end elements (HHT-17) has had several initial tests for reproducibility. The initial tests will use 5 and 10 mil gaps between the hexagonal test specimen and its surrounding hexagonal confining space. Undercuts of 10 and fifteen mils are presently planned. Axial flow perturbations

by additional gas injection locations to study the vector effects of the gas flow will be evaluated.

Future work during December consists of additional tests on fuel elements for direct gas injection at the upstream end (HHT-2), 52 inch annular gap corrosion tests (HHT-6) and the assembly of the interstitial pressure distribution test (FFL-15).

Facilities

The following facilities were covered in the discussion: pump vaporizers, WANHES dewar, thermocouple furnace and the new A-2 experimental furnace. The pump vaporizers are in this status. The WANHES vaporizer has been repaired by the Linde people for all except one minor valve leak. This is expected to be accomplished during the week of December 13. The large dewar of approximately 8000 gallons for WANHES has been specified and is now in the procurement cycle. The thermocouple furnace has been received and is currently being installed in Test Cell 3. The new A-2 furnace has been received at Large but has not yet been delivered to the flow laboratory. Once received this furnace in its initial build will be very carefully instrumented and will replace the present furnace in the test cell which will be rebuilt and serve as a standby furnace for either Waltz Mill or Large.

General Discussion

Mr. Wilke asked the question whether or not an HHT-5 center hex test is desirable with NRX-A5 hardware since the NRX-A5 hardware differs only slightly from the NRX-A4 hardware. The necessity for this test will be reviewed. Mr. Wilke also brought up the point that the problem of pinholing has not been emphasized as much as it should. Several test possibilities were discussed. One of which is to plug the outer 12 holes of the single element, use reduced flow on the inner seven holes and heat with a radiant heater setup so that the pinholing problem is completely divorced from the electric heating problem. This will probably be done in the very near future in the A-2 furnace while it still has the radiant heater configuration.

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Larry Larson of Aerojet General Corporation (AGC) suggested that the planned resonance survey test to be conducted on XE-1 be performed on XECF. Mr. Rowan of WANL felt that this could be done and he agreed that by so doing the information would be of greater value since it could be available earlier in the program. WANL personnel will be in contact with AGC personnel to further define the information which must be provided on this subject during the next few months.

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