ORNL/NUREG/TM-111

March 1977
Monthly Highlights
for
Office of Nuclear Regulatory Research Programs
at
Oak Ridge National Laboratory

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements 40-551-75 and 40-552-75

OAK RIDGE NATIONAL LABORATORY

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Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy $4.00; Microfiche $3.00

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Contract No. W-7405-eng-26

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MONTHLY HIGHLIGHTS

FOR

OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS

AT

OAK RIDGE NATIONAL LABORATORY

Compiled by

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Manuscript Completed: April 15, 1977

Date Published - May 1977

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements 40-551-75 and 40-552-75

Prepared by the
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Oak Ridge, Tennessee 37830
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UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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ABSTRACT

Highlights of technical progress during March 1977 are presented for thirteen separate program activities which comprise the ORNL research program for the Office of Nuclear Regulatory Research's Division of Reactor Safety Research.

PROGRAM TITLE: Heavy Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

ACTIVITY NUMBER: ORNL # 40 89 55 01 1 (189 # B0119)/NRC # 60 19 01 01

TECHNICAL HIGHLIGHTS

Task 1: Program Administration — A meeting was held on March 17 in Silver Spring, MD, to review the status of the fatigue crack growth projects sponsored by NRC-RSR. G. D. Whitman and J. G. Merkle represented the HSST Program.

On March 18, G. D. Whitman and J. G. Merkle attended a presentation made by Prof. Paul Paris on the utilization of R-curve data in structural analysis. The meeting was held in Bethesda, MD.

On March 22, M. Galliani and G. T. Sica of ENEL, and G. C. Angelino of CISE from Italy visited ORNL for a briefing on the HSST Program.

G. C. Smith attended the ASTM Crack Arrest Subcommittee meeting held on March 22, at Norfolk, VA.

P. P. Holz attended a North Carolina State University-sponsored seminar on mechanical and automated welding at Raleigh, N.C., held on March 23 and 24.

Task 3: Fatigue Crack Growth — The 4T environmental chamber was placed in operation with a 4T-CT specimen operating at a low R ratio, $\sim$0.2, to obtain data in a higher ΔK region. The temperature of the chamber is limited to 204°C (400°F) since water leakage occurs above this temperature. The two 2T autoclaves that have been in use to study R ratio and material effects will be devoted to obtaining data using the test matrix developed to evaluate wave form effects.

Task 4: Irradiation Effects — The irradiation of the 4T-CTS capsules was completed on March 23, 1977, and the capsules stored in the BSR pool to permit partial decay of radioactivity before disassembly. Disassembly is scheduled to start April 25, 1977.

We have prepared the Charpy machine for the test program and calibration is complete except for receiving the report on calibration specimen results from the Army Materials Research Agency.

Final details of the disassembly procedures are being worked out with the Hot Cell Operations group.

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Preparations are in progress for the next 4T-CTS irradiation. Design changes have been completed, purchased components are on hand, capsule machining is in progress and about 65% of the specimens have been received.

Charpy V-notch toughness data were obtained for 152.4-mm-thick (6 in.) A533, grade B, steel plate (plate 04BD) that was austenitized at 871°C (1600°F) and quenched in water. The specimens were removed from the quarter thickness location and were oriented transverse to the major rolling direction. The upper shelf C_v toughness was about 81 joules (60 ft-lb). The onset of upper shelf occurred at 135°C (275°F). The 40 joules (30 ft-lb) C_v temperature is about 82°C (180°F). This material was used by HEDL in the small specimen utilization program as a material having properties equivalent to an irradiated quench and temper material.

Task 5: Simulated Service Tests — The typical V-7 type 2.54-cm-wide (1-in.) trapezoidal slot was machined centered about the heat-affected zone of one of the longitudinal edges of the weld repair of vessel 7B. Temporary skid rails were attached to the vessel and machined for parallelism with the slot for transfer and indexed positioning of the vessel into the electron-beam weld chamber.

Various etching solutions were used to properly locate drill and mill centers to form the slot centered about the repair weld's heat-affected zone. Positioning check determinations and appropriate minor tool centering adjustments were made using 0.95-cm-diam (3/8-in.) and 1.27-cm-diam (1/2-in.) cutters at depths of 1/27 cm (1/2 in.), 2/54 cm (1 in.), and 6.35 cm (2 1/2 in.) from the vessel's surface. The heat-affected zone centers for the finished slot are within 0.025 cm to 0.089 cm (0.010 in. to 0.035 in.), both along the trapezoidal slopes and at the 12.7-cm-deep (5-in.) bottom run. The HAZ band width varies from 0.17 to 0.20 cm (0.065 to 0.080 in.). Ten strain gages affixed to the vessel's inner surface and strategically located below the slot served to monitor stress buildup induced by the shop machining operations.

Vessel 7B was moved to a location just outside the large electron-beam weld chamber on March 30 where preliminary gauss meter checks indicated magnetic field intensities up to about 5 gauss transversely and 8 gauss longitudinally.

Four 5-cm-thick (2-in.) brick shaped A533 material test blocks with premachined cavities to simulate the vessel cavity geometry were also "repair welded" at Tampa to half-bead technique welding standards. These blocks have been ground and etched and will be utilized to establish electron-beam programming input and to verify hydrogen-charge flawing methods. A trapezoidal steel model is also available for final slope set programming checkouts.

The V-8 weld-repaired prolongation was cut into halves; one half was prepared for residual stress measurements by slow speed and slow feed grinding. Wedges containing the repair weld portions were sawed from the other half and are being prepared for material characterization. Strains were measured at 30 locations during the sawing operations to obtain data for adjusting residual stresses measured after cutting.

Site preparations for the V-7B test and some electrical service relocation work were started at the K-25 test facility. Instrumentation plans and test procedures and the required drawing and specification documentation for V-7B are also under way.

The first crack arrest model was tested on March 3 at a test temperature 91°C (196°F) where the Charpy energy of the brittle starter material was 40 joules (30 ft-lb). At approximately 90 MPa (13,000 psi), the crack began to tear slowly. It was necessary to increase the pressure in the vessel in order to extend the crack. The test was terminated when the thin stainless steel liner leaked. At that time, the crack had been driven approximately 0.8 cm (2 in.) in the brittle starter material. The second crack arrest model has been fabricated and is scheduled to be tested in early April. The second test will be identical to the first, except the test temperature will be 0°C (32°F).

Drop weight testing of P-2 specimens has been completed for the high toughness or arrest material used in the model testing. The NDT is −23°C (−10°F) for RT-oriented specimens at the 1/4T and 3/4T depths of the 6-in.-thick plate. Test results with a limited number of specimens indicate that the WT-oriented specimens from the same depths have the same NDT.

The Charpy blanks from HSST plate section 04BE (the source of the low toughness material) have been received and static fracture toughness testing with precracked Charpy specimens has begun.

Task 6: Thermal Shock — Preparation of a topical report covering TSE-3 and TSE-4 was continued, and a copy of *Quick Look Report on LOCA-ECC Thermal Shock Program Test TSE-4* (TSP-1005) was sent to Framatome (France) along with preliminary information regarding our cryogenic quench program.

Cryogenic quench experiments were continued with emphasis on the internal quenching of a 150-mm-pipe specimen insulated on the outside and having a length of 150 mm and a wall thickness of 11 mm. Data obtained thus far have provided good quench-time comparisons but not accurate heat-transfer-coefficient values. Recent improvements in the instrumentation will provide the latter information.

Compact tension and Charpy specimens are being prepared from a section of TSV-1 to further characterize the fracture toughness of the material. Static fracture toughness values will be obtained for 2T and 0.394T compact tension and the precracked Charpy specimens. The fatigue crack tips, where possible, will be at the same depth location for all of the specimens. In addition, the crack tip of one set of precracked Charpy specimens will be located at the same depth as the electron beam induced weld crack used in the vessel test, approximately 16 mm (0.4 in.).

The scanning electron microscopy (SEM) examination of the fracture surface from thermal shock experiment No. 3 has been completed. The crack that resulted as a consequence of the thermal shock propagated predominantly by cleavage mode. There are areas that exhibit a dimple mode which is indicative of ductile fracture; however, these areas represent a small percentage of the total fracture surface.

Task 7: Foreign Research — Reports on subjects of interest were identified from listings in *Nuclear Safety* through Vol. 18, No. 1, January—February 1977, and procedures initiated for obtaining original and English language copies. Copies of "OECD, Nuclear Energy Agency — International Energy Agency, Nuclear Research Index, 1976," were obtained and steps taken to prepare summary information on research projects of interest, using country and category indices.

PROGRAM TITLE: Fission Product Beta and Gamma Energy Release

PROGRAM MANAGERS: R. W. Peelle and J. K. Dickens

ACTIVITY NUMBER: ORNL # 40 89 55 10 5 (189 # B0095)/NRC # 60 19 01 03

TECHNICAL HIGHLIGHTS

A satisfactory binder for the ^{239}Pu samples was found during March. This glue does not contribute a measurable background and remains solid during the longest irradiations required.

The draft of the ORNL/NUREG report was not completed during March as had been anticipated because more time was required to complete the analysis of the external-beam check on absolute normalization. This analysis was completed during March and the documentation has been resumed.

PROGRAM TITLE: Fission Product Release from LWR Fuel

PROGRAM MANAGER: A. P. Malinauskas

ACTIVITY NUMBER: ORNL # 40 89 55 10 8 (189 # B0127)/NRC # 60 19 01 03

TECHNICAL HIGHLIGHTS

The fourth experiment of the High Burnup Fuel Test Series was conducted at 500°C for 20 hours with a flowing steam-argon atmosphere. A 1/16-in. diameter hole was drilled through the cladding at the center of the 12-in. capsule to provide a simulated defect.

Only about $4 \times 10^{-6}\%$ of the total cesium inventory was released during the course of the experiment; as in the previous tests, the cesium vapor was highly reactive with the quartz surface of the furnace tube liner and the quartz fuel rod holder. Most of the cesium deposited on the surfaces near the drilled defect opening.

PROGRAM TITLE: Fission Product Transport Tests

PROGRAM MANAGER: A. P. Malinauskas

ACTIVITY NUMBER: ORNL # 40 89 55 11 6 (189 # B0189)/NRC # 60 19 01 03

TECHNICAL HIGHLIGHTS

Conceptual design of the Fission Product Test Facility is proceeding. Preliminary design of the model steam generator was completed in sufficient detail for cost estimation. The shell of the model steam generator consists of 8-inch pipe and contains 24 3/4-inch diameter tubes, two of which are easily removable.

Health Physics requirements at ORNL which bear on the operation of a moderately sized loop employing tracer levels of Cs-134 have been studied (presumably, equivalent regulations apply at other installations). Preliminary evaluation of the applicable criteria indicates that a maximum input per run of 1 μCi, and a total accumulation of 10 μCi, would be permissible in a well-ventilated room which discharges through a HEPA filter. This low tracer level appears to preclude the use of external scans as a diagnostic method; instead, removable inserts at key locations would be needed to determine deposition.

The completed conceptual design will consist of the following 11 design drawings: (1) Flowsheet and Instrumentation, (2) Model Steam Generator (completed), (3) Model Pump, (4) Hot and Cold Leg Piping, (5) Steam Separator and Condenser, (6) Fission Product Injector, (7) Equipment Layout, (8) Cleanup System, (9) Electrical Diagram, (10) Room Ventilation and Services, and (11) Steam Supply System.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: ORNL # 40 89 55 10 6 (189 # B0120)/NRC # 60 19 01 03

TECHNICAL HIGHLIGHTS

R. H. Chapman and G. Hofmann visited NRC Headquarters in Silver Spring, Md. on April 15, 1977, to participate in a technical review of in-reactor fuel behavior tests performed at INEL.

R. W. McCulloch visited SEMCO on March 10-11, 1977, and GE-Seattle on March 14-15, 1977, to consult on progress with our outstanding orders for fuel simulators for the second and third 4 × 4 bundles, respectively. Nine simulators were received this month from SEMCO and are undergoing acceptance inspection and evaluation. The problem reported last month with swaging appears to have been resolved and production of the remaining portion of the order is underway. Delivery of the simulators for the second bundle should be complete by the end of April, two weeks later than anticipated by Milestone 40080.

Five preproduction simulators fabricated by GE-Seattle failed the insulation resistance test and were not shipped for infrared scanning. The cause for the low resistance was traced to contamination and interruption of the filling and tamping operations. The procedures were modified to perform these operations uninterrupted. In order to avoid further slippage in the delivery schedule, fabrication of the production simulators was initiated without benefit of our evaluation of the preproduction units. As a result of this decision, GE anticipates producing several extra units as replacements for any that might be rejected. Delivery of the simulators is expected by the middle of May, one month later than anticipated by Milestone 40080.

Plans were initiated to purchase an additional ten simulators from SEMCO and an additional five from GE. These additional units will provide a greater selection for each of the next two test arrays as well as simulators for several single rod tests later this year.

Twenty fuel simulators were selected for potential use in fabricating the first 4 × 4 test array. Ten of the simulators have been grooved, seven of these have had plasma sprayed protective coatings of ZrO_2 applied, and the axial temperature distribution of four of the coated simulators has been characterized. The availability of these four simulators permits orderly fabrication of the fuel pin simulators to commence. All the bundle components have been fabricated and assembly will be initiated as soon as the first few fuel pin simulators are ready.

Two fuel pin simulators were assembled and tested as single rod specimens (SR-28 and SR-29); initial conditions identical to those expected for the first bundle test were employed. The data from these tests have not been reduced. The simulators were fabricated as prototypes for the bundle to check the practicalities and to gain familiarity with the assembly procedures and documentation controls specified for the bundle simulators. In general, the procedures worked well.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: William B. Cottrell

ACTIVITY NUMBER: ORNL # 40 89 55 10 4 (189 # B0126)/NRC # 60 19 01 03

TECHNICAL HIGHLIGHTS

During the month of March the staff of the Nuclear Safety Information Center (a) processed 1359 documents, (b) responded to 119 inquiries (of which 61 involved the technical staff), and (c) made 22 computer searches (of which four involved payment totaling $301.30). Although we have always felt that the ERDA policy by which the payments for information services were returned to the ERDA General Fund rather than to the Center which incurred the cost was not in the best interests of information communication, we were surprised to learn that some of NSIC's former paying users now receive the same service directly from the ERDA-Technical Information Center for free. In addition, of the 19 data bases currently included on the ERDA RECON system, NSIC is the fifth most frequently used.

The accumulative bibliography for the ACRS, which contains around 4800 documents, is in reproduction. We have also processed 354 documents received from the ACRS in March and this bibliography should be issued in a month. NSIC staff received 14 visitors during February, participated in eight meetings and prepared five reviews.

Several NSIC reports are in various stages of preparation. The third volume of the five-part "LMFBR Safety: Review of Current Issues and Bibliography of Literature" (ORNL/NUREG/NSIC-131) covering the period 1972–1974 and the fourth volume (ORNL/NUREG/NSIC-132) covering the period 1974–1975, are both in reproduction. Three other reports, ORNL/NUREG/NSIC-134, "Reports Distributed in 1976 Under the NRC Light-Water Reactor Safety Technical Exchange, Vol. 2," ORNL/NUREG/NSIC-133, "Index to *Nuclear Safety*, Vol. 11, No. 1 through Vol. 17, No. 6," and ORNL/NUREG/NSIC-135, "Bibliography of Reports on Research Sponsored by the NRC Office of Nuclear Regulatory Research July–December 1976" are also in reproduction. Another report, ORNL/NUREG/NSIC-136, "Design Data and

Safety Features of Commercial Nuclear Power Plants Including Cumulative Index for Vol. I - VI," is in composition. Work has begun on several other reports, including ORNL/NUREG/NSIC-128, "A Bibliography of LMFBR Safety Literature"; ORNL/NUREG/NSIC-137, "Annotated Bibliography of Safety-Related Occurrences in Boiling-Water Nuclear Power Plants as Reported in 1976"; and ORNL/NUREG/NSIC-138, "Annotated Bibliography of Safety-Related Occurrences in Pressurized-Water Nuclear Power Plants as Reported in 1976."

NSIC's special selective dissemination of information (SDI) is available to paying users (as well as additional non-paying users). During the month of March we added one non-paying user bringing the total SDI users to 379.

We received no foreign safety reports under the NRC exchange agreements during the past month. However, during this period we received copies of 17 reports on NRC-sponsored safety projects which were distributed (March 2 and March 14) to foreign recipients under the Light-Water Reactor Technical Exchange Agreements. The abstracts of these reports, which are being translated into French and German, generally follow the reports by about one month.

The recently purchased IBM-3270 computer equipment consists basically of a controller (which is hard-wired into the Computing Technology Center at K-25), a printer, and four CTR terminals. The controller and two CTRs have been tested by the Systems Engineering Section of the Computer Sciences Division at X-10 so they will be operational with modified programs and are being transshipped to K-25 for installation and further testing. Due to systems teleprocessing hardware and software problems, it is now expected that the new equipment will not be operational at NSIC offices until the latter part of April.

All the technical articles for *Nuclear Safety* 18(4) have been reviewed, edited and submitted to ERDA and NRC and we are awaiting their comments. Manuscripts of the feature sections for *Nuclear Safety* 18(3) were completed and submitted by March, except for reactor operating experience which is obtained from NUREG 0020-2 for January and February.

At the special request of NRC, we are coordinating the technical review of a series of articles on NRC-sponsored decay heat studies. The first review was completed last month. Review of an article by Yarnell and Bendt (second in the series) has been initiated with responses expected early in April.

PROGRAM TITLE: PWR Blowdown Heat Transfer Separate Effects

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: ORNL # 40 89 55 10 3 (189 # B0125)/NRC # 60 19 01 03

TECHNICAL HIGHLIGHTS

Task 1. Since Watlow bundle 2 prototypical heater SN-13 did not meet BDHT Program specifications on heater element concentricity, it was diverted to the FCTF for blowdown testing. The purpose of these tests is to determine the effects of eccentricity of the heating element on the behavior of the heater and its thermocouples through all phases of a blowdown. This is the third bundle 2 prototype to be tested in the FCTF.

Task 2. Test 156 was conducted in the Thermal-Hydraulic Test Facility (THTF) on March 10, 1977. Test 156 was a 45-rod (rods 19, 24, 39, and 47 inactive) (3.599 MW) test initiated from a test section inlet temperature of 558.2 K (545°F), a test section inlet volumetric flow of 0.0201 m^3/sec (318 gpm), and a test section outlet pressure of 16.017 MN/m^2 (2323 psig) and outlet temperature of 598 K (617°F). System decompression was accomplished by introducing a 40% inlet-60% outlet break. The total break area was 1/2 that of previous tests and a 1/4-in. orifice was placed in the pressurizer surge line; however, the desired depressurization rate was not obtained because flow from the pressurizer was subcritical for the first six seconds of the transient. The primary coolant pump was tripped coincident with break initiation, but the electric core was operated at full power for ~2 sec into the transient and then the power was decayed with a time constant of ~0.45 sec. Mean time to CHF (from sheath T/C readings) was 1.1 sec with a range of 0.8 to 2.4 sec; 85% of the sheath thermocouples indicated the occurrence of CHF.

Test 157 was conducted in the Thermal-Hydraulic Test Facility (THTF) on March 24, 1977. Test 157 was a 45-rod (rods 19, 24, 39, and 47 inactive) (4.503 MW) test initiated from a test section inlet temperature of 557.6 K (544°F), a test section inlet volumetric flow of 0.0337 m^3/sec (534 gpm), and a test section outlet pressure of 16.161 MN/m^2 (2344 psig) and outlet temperature of 588 K (599°F). System decompression was accomplished by introducing a 40% inlet-60% outlet break. The primary coolant pump was tripped coincident with break initiation, but the electric core was operated at full power for $\sim$2 sec into the transient and then the power was decayed with a time constant of $\sim$0.45 sec. Mean time to CHF (from sheath T/C readings) was 1.0 sec with a range from 0.4 to 3.2 sec; 89% of the sheath thermocouples indicated the occurrence of CHF.

Task 3. During the period in which the last seven blowdowns were conducted in the THTF (11/5/76 to 3/24/77) the resistances of the heater rod assemblies (top lug, heater element, tapered end plug, nickel electrode) have apparently increased with time (about 16% at 1/4 kW/rod and about 3% at 5 kW/rod). Although this increase is within the estimated error band, measurements during approach to power have shown consistent, progressive increases in resistance. Despite their small magnitude, the consistency of the increases has resulted in concern over the remaining life of bundle 1. The subject is now under review by the ORNL staff.

Task 4. Design and procurement is proceeding for inbundle liquid level sensors for THTF Bundle 2. The sensors will be instrumented tubes occupying the location of four control rod guide tubes in the bundle array. Two designs will be considered, both of which operate on the principle of monitoring fluid conductivity. One design is the INEL LOFT liquid level sensor, the other is a proposal of Auburn International, Inc.

Horizontal two-phase flow testing of the THTF instrumented spool piece is continuing in the air-water facility. Calculated mass flows were quite accurate provided woven wire screen flow dispersers were used and the two-phase mixture was not highly stratified. Tests of improved disperser designs are continuing.

Task 5. Plans to ensure the integrity of Bundle 2 centerline thermocouples by increasing the number of thermocouples to five in all those rods originally calling for one or two have been modified somewhat due to recent failures in heater rods already containing five centerline thermocouples and to recent failure analyses made on such heater rods. Those analyses showed that thermocouple failures were principally due to two conditions: (1) the thermocouple had been subjected to excessive shear stresses and elongations in the copper lead-in section caused by excessively high MgO final density, and (2) thermocouples had been subjected to excessive pinch and tensile forces at the MgO to rubber end seal interface at the upper end of the copper lead-in. At the suggestion of ORNL personnel, several changes have been made by Watlow in their fabrication method which have resulted in the survival of 68 out of the last 70 centerline thermocouples. Also, all future centerline thermocouples will be purchased from Watlow since (for their present batch) they appear to have superior elongation properties. Presently, 32 Bundle 2 heater rods have been delivered to ORNL; 27 additional heater rods are needed to complete the bundle and 63 additional heater rods are needed to complete the order.

Task 6. The General Electric Company is presently fabricating a short sample of a Bundle 3 heater rod to test their method feasibility. They expect to ship this sample to ORNL for testing by 4/15/77.

Sensor Dynamics, Inc. is presently completing design work and procuring material to fabricate two Bundle 3 prototype heater rods.

The RAMA Corp. has sent in a bid of $12,500 for two Bundle 3 prototype heater rods with delivery time of 16 weeks after receipt of order. UCC Purchasing Division is proceeding to place the order immediately.

Watlow Electric MFg. Co. has sent in a bid of $11,874 with delivery time of 14-16 weeks after receipt of order for two Bundle 3 prototype heater rods but patent agreement problems are still unresolved.

The first prototype heater rod has been shipped from INEL to ORNL for testing in the FCTF. The second is still expected in July 1977.

PROGRAM TITLE: Zircaloy Fuel Cladding Collapse Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: ORNL # 40 89 55 10 7 (189 # B0124)/NRC # 60 19 01 03

TECHNICAL HIGHLIGHTS

The following milestones for the in-reactor creepdown tests were met during this month: No. 47020-Complete Facility Conceptual Design; No. 47025-Establish Interface with ECN; and No. 47038-Order Heaters.

The part drawings for both the mockup and the first pressurized test are complete and undergoing internal review. The assembly drawings are nearing completion.

We have received from ECN, for our review, the instrumentation proposal for the in-reactor experiments. The report is to provide ORNL as well as ECN personnel with a proposal for the instrumentation which will be needed for the performance of the creepdown experiments in the Dutch HFR Reactor. Included in the proposal are descriptions of the experiments (named HOBBIE by the Dutch), instrumentation and piping diagrams, and a breakdown of the responsibilities of ORNL and ECN with respect to the set-up of the instrumentation. We are currently reviewing the proposal.

Difficulties have been encountered with obtaining delivery dates less than 16-18 weeks for the heaters for use in the mockup and pressurized tests. We have obtained promises of shorter (6-8 weeks) times in return for allowing use of welded tubing. The obtainment of these heaters could be a critical path item for completion of the mockup.

It is planned that an ORNL engineer visit the ECN-HFR in May or June to study the space assignments for the experimental facilities and to meet the ECN personnel who will conduct the testing.

Out-of-reactor testing is underway at ORNL. Temperatures and pressures in the autoclave are easily controlled. Some difficulties have been encountered in long-term stability of the eddy-current instrument. This is manifested by calibration changes that take place when an individual coil is lost. Such a loss can be caused by shorting or by an open circuit to the coil. A change of from 2.5×10^{-5} to 1.25×10^{-4} m (0.001-in. to 0.005-in.) in the output can occur with the loss of a coil. This is an unacceptable variation, especially since the system is capable of detecting a change of 2.5×10^{-7} m (0.00001-in.). It is believed that a grounding problem related to the 2 MHz signal is the cause. This is being investigated.

PROGRAM TITLE: Zirconium Metal-Water Oxidation Kinetics

PROGRAM MANAGER: C. J. McHargue

ACTIVITY NUMBER: ORNL # 40 89 55 10 9 (189 # B0128)/NRC # 60 19 01 03

TECHNICAL HIGHLIGHTS

Preliminary steam pressure effects (Milestone 49155, 4/15/77) have been completed, and a test program for further steam pressure effect studies has been developed (Milestone 48190, 4/15/77).

PROGRAM TITLE: Aerosol Release and Transport from LMFBR Fuel

PROGRAM MANAGER: M. H. Fontana

ACTIVITY NUMBER: ORNL # 40 89 12 10 1 (189 # B0121)/NRC # 60 19 02

TECHNICAL HIGHLIGHTS

CRI-III/CDV:

Analytical data from a CDV/CRI-III test with stainless steel filings added to the microsphere packing confirmed an initial yield of 4.2 g/m^3 of mixed aerosol. This is the first test with added stainless steel that has had total energy deposition and yield that compare favorably with pellet-microsphere only samples. Airborne concentration and wall deposition data from this test gave values for the ratio of the diffusion coefficient to the boundary layer thickness of 7×10^{-4} cm/sec two-minutes postshot which decreased to a constant value of $\sim 3 \times 10^{-4}$ cm/sec at 70 minutes. The data appear to demonstrate that rapid coagulation ceases after ~10 minutes; a self-preserving size distribution exists from 10 to 70 minutes; and the average particle size decreases as aerosol concentration becomes small at longer times.

Experiments with a pellet stack in a quartz tube (no thermal packing) were also conducted. If good aerosol yields can be obtained from this configuration, multipin assembly tests and real fuel tests would be less difficult to conduct. Initial preheater experiments indicated that suitable sample resistance conditions could be obtained to allow an energy input rate of 0.1 to 0.6 MW/g from the capacitor banks. In the first two experiments, carried through the capacitor discharge phase, arcing occurred without appreciable energy deposition or sample disassembly. After fuel pin modification, a successful firing resulted in 0.68 $\mu g/cm^3$ of aerosol. About 12 kJ were deposited in the sample at 0.17 MW/g.

CRI-II:

UO_2 fuel aerosol characterization studies continued in the CRI-II facility with runs AF-3 and AF-4 from the arc vaporization furnace. In these tests, a thermal precipitator for deposition on microscope grids has been added. Comparisons are being made between the thermal precipitation samples and those deposited by the electrostatic method.

Efforts to improve the yield of UO_2 vapor from the arc furnace are continuing. The charge of UO_2 pellets has been increased so that the total bed thickness of the combined melt and solid UO_2 reduces the heat transfer rate and provides a higher total temperature gradient.

The addition of a tungsten liner, loosely fitted to a hearth bowl, will further reduce the heat transfer and will provide higher electrical conductivity than the thick UO_2 bed thereby permitting higher current flow from the arc.

Bubble Transport/FAST:

Work continued on the in-house fabrication of the FAST pressure vessel in the X-10 shops. The manufacturing plan was completed and is ready for review and approval. The plate to be used for the vessel shell has been rolled and is ready to be seam welded. Approximately 90% of all the material needed for the pressure vessel is now on hand.

Drawings showing the electrical equipment, conduit, cable trays and control cabinets were issued for review and comments.

Purchase orders were issued for sample cylinders to be used with the mass sampler, for the electrical wiring needed for the CRI-III and CDV installation in Building 9201-3, and for a three-phase transformer required for the CDV installation.

NSPP:

During this reporting period, NSPP activities included 1) reduction of experimental data from Sodium Pool Fire Test 101, 2) preparations for Sodium Pool Fire Test 102, and 3) conduct of Test 102 at the end of this period.

Experiment 102 was conducted by introducing 5 kg (11 lbs) of sodium at 540°C (1000°F) into the burn pan also at 540°C (1000°F); initial vessel pressure was approxiamtely 14.7 psia; the vessel atmosphere was normal air; and the relative humidity varied from 48% near the bottom of the vessel to 73% near the top of the vessel. Qualitative observations indicate a pressure rise of about one psia and the vessel atmosphere temperature increased about 40°C (104°F) due to the sodium burn. Data reduction is continuing on this test.

Fabrication of the uranium aerosol generating system was completed in the shops. The various parts have been assembled on a test stand, and the initial check of mechanical function was accomplished. Electrical checkout will be performed upon installation of the system onsite.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: S. J. Ball

ACTIVITY NUMBER: ORNL # 40 89 55 11 2 (189 # B0122)/NRC # 60 19 03

TECHNICAL HIGHLIGHTS

Development of the ORTAP code for the Fort St. Vrain (FSV) Reactor: Several more refinements and improvements were added to the main steam bypass system and circulator turbine subroutines. Variations on the rod withdrawal accident simulation were run to note the sensitivity of the results to various model assumptions. A paper entitled "ORTAP: A Simulator of High Temperature Gas-Cooled Reactor Nuclear Steam Supply System Dynamics" was accepted for the 1977 Summer Computer Simulation Conference in July. A paper entitled "Simulation of the Response of the Fort St. Vrain High Temperature Gas-Cooled Reactor System to a Postulated Rod Withdrawal Accident" was accepted for presentation at the ANS Thermal Reactor Safety Meeting in Sun Valley, Idaho, this summer.

PROGRAM TITLE: Design Criteria for Piping and Nozzles

PROGRAM MANAGER: S. E. Moore

ACTIVITY NUMBER: ORNL # 40 89 55 10 2 (189 # B0123)/NRC # 60 19 01 01

TECHNICAL HIGHLIGHTS

PVRC and ASME Code Efforts: The ASME Boiler and Pressure Vessel Code Committee held its regular series of working group and subcommittee meetings during the week of March 1, 1977. Two items of interest to the Design Criteria Program were considered by the Working Group on Piping (WGPD). One was a proposed Code revision which would restrict the use of the present stress indices for branch connections to values of the dimensionless parameter $\rho = d/\sqrt{DT} \leq 0.8$. This is a rather severe restriction for piping design and the proposed revision included provisions for altering the design calculations so as to make the indices acceptable (see ORNL/SUB/2913-2). A more appropriate solution, however, would be to revise the Code indices to include the parameter ρ in the formulations. WGPD, therefore, elected not to accept the proposed revision but to wait until we can prepare revised stress indices based on results from our analytical parameter studies on nozzles in cylindrical vessels. We expect to present revised stress indices for branch connections some time after the June meeting.

WGPD also discussed a proposed new standard, MSS-SP-XX, for the manufacture of buttwelding fittings for Class 1 Nuclear Piping applications. Mr. H. H. George, Chairman of the Manufacturers Standards Society (MSS) Committee No. 113, Wrought Welding Fittings, which is responsible for the proposed new standard, attended the meeting to discuss our needs and comments. Most of the discussion centered around the ORNL report *Dimensional Control of Buttwelding Pipe Fittings for Nuclear Power Plant Class 1 Piping Systems*, ORNL/Sub/2913-5 that was published in December. As a result of the discussion, Mr. George agreed to revise the proposed standard to include our recommendations, subject to confirmation by his committee.

On March 29, 1977, E. C. Rodabaugh and I met with Mr. George and Mr. Benson of Tube Turns to review a revised version of MSS-SP-XX. This new revision incorporates essentially all the recommendations of WGPD and

ORNL/Sub/2913-5. Mr. George expects to send the draft standard out for committee review early in April.

Miscellaneous Studies: A study on the development of flexibility factors for small branch connections with external loadings was completed and a report, ORNL/Sub/2913-6, was published this month. For some time we have felt that the subject of piping flexibility, particularly for components other than elbows, has been inadequately represented in the ASME Code. In this study we undertook to develop a more rational flexibility analysis method for small branch connections in piping systems, and to develop a consistent set of instructions for Code use. The recommendations section of the report contains proposed revisions to Code paragraphs NB-3682, NB-3687.2 and NB-3687.5.

Nozzles in Cylindrical Reactor Pressure Vessels: During the past month we have been conducting validation studies for the corrected computer program CORTES-SA. One additional error was discovered and corrected, and five nozzle-cylindrical vessel models were reanalyzed. The comparisons between the new analytical results and the experimental results show excellent agreement.

We have also been working on the third phase of the closely-spaced nozzle program MULT-NOZZLE. Programming for the mesh generator portion, to enable the program to lay out finite element idealizations for three closely-spaced nozzles, was essentially completed by MRI in Los Angeles. We plan to bring the mesh generator to Oak Ridge in April to install it on the UCCND computer system, and to check it out with the aid of our in-house graphics.

PROGRAM TITLE: **Noise Diagnostics for Safety Assessment**

PROGRAM MANAGER: Ray S. Booth

ACTIVITY NUMBER: ORNL # 40 89 55 11 4 (189 # B0191)/NRC # 60 19 01 03

TECHNICAL HIGHLIGHTS

In-Core Monitoring Methods. Neutron noise data collected from all operable local power-range monitor (LPRM) strings at Browns Ferry Units 1, 2, and 3 were further scrutinized for evidences of LPRM impacting with surrounding fuel channel boxes. Selected data were made available to the TVA for their information and comment.

Work continued on the identification and characterization of noise-equivalent sources in BWRs. This calculational effort, which constitutes the first step towards our development of an overall methodology for predicting the response of an in-core neutron detector to various postulated fuel and control rod flow-induced motions, will be described in greater detail in the forthcoming *Noise Diagnostics for Safety Assessment Quarterly Progress Report, January-March, 1977.*

Loose-Parts Monitoring Systems. The first of four planned on-site information gathering tours of loose-parts monitoring system (LPMS) vendors and selected utilities was accomplished. The four are: Babcock & Wilcox (LPMS vendor) on March 17, Oconee station (B&W reactor, B&W LPMS) on March 22, St. Lucie station (CE reactor, CE/Columbia Research LPMS) on March 23, and Crystal River station (B&W reactor, B&W LPMS) on March 24. Much Valuable technical information and a clearer understanding of the reasons for the nonuniform application experience that has been reported within the nuclear industry were derived from these initial visits. Telephone reports were made to both NRC:DES and NRC:DOR. Arrangements are now being made with the assistance of the licensing project managers for the second tour, which will focus on the remaining major LPMS vendor, Atomics International.

Surveillance and Monitoring by Noise Analysis. Overall objectives and signal requirements for a medium-duration, on-line demonstration of an automated performance anomaly detection system were formulated, and initial contacts were made with two utility companies that were anticipated to be receptive to a demonstration proposal.