ORNL/NUREG/TM-48

July 1976
Monthly Highlights
for
Office of Nuclear Regulatory Research Programs
at
Oak Ridge National Laboratory

MASTER

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION FOR THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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MONTHLY HIGHLIGHTS

FOR

OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS

AT

OAK RIDGE NATIONAL LABORATORY

Compiled by

Gordon G. Fee

AUGUST 1976

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
UNITED STATES
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

PROGRAM TITLE: Heavy Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

ACTIVITY NUMBER: 40 89 55 10 1 (189 #B0119)

TECHNICAL HIGHLIGHTS

Task 1: Program Administration — Dr. M. J. G. Broekhoven of Delft University of Technology, Laboratory for Nuclear Engineering, Rotterdam, Netherlands, visited ORNL on July 1 to review nozzle corner crack analysis.

On July 13, T. J. Dwyer of MARC Analysis Research Corporation, Palo Alto, CA, visited ORNL to review work performed under the HSST program.

P. P. Holz and G. D. Whitman visited Combustion Engineering offices in Chattanooga, TN, on July 15, to review in-service weld repair procedures for possible repair of intermediate vessel V-7A.

The monthly HSST program review was held at ORNL on July 28 and 29, for C. Z. Serpan and E. K. Lynn.

Task 2: Fracture Mechanics and Analysis — Plans have been completed to modify the VPI photoelastic study of nozzle corner cracks at Virginia Polytechnic Institute to include the geometry of a BWR feedwater nozzle.

Task 4: Irradiation Effects — The three capsules for the second 4T-CT irradiation study have been placed in the BSR pool, and instrument, power and gas lines are being connected to the control panels and utilities. A complete operational checkout will be conducted prior to placing the specimen capsules at the irradiation position. The safety calculations indicated that the steam venting system capacity needed to be increased in the event of a water leak into the capsule. The system is being revised to provide adequate venting capacity for a "worse case" accident.

Flux mapping of the new BSR core loading, reactivity runs and neutron spectrum runs using the "dummy" capsules are scheduled for August 2 through 6 and start of the specimen capsule irradiations is scheduled for August 16, 1976.

The adhesively joined 2T composite compact specimen has been machined on the faces to remove a slight amount of misalignment which occurred during bonding. A layer of photoelastic material has been cemented to one face, and three strain gage rosettes are presently being attached to the opposite face near the fatigue crack tip. A mock-up 2T specimen has been machined and is being coated with photoelastic material. This mock-up will be used to check out the experiment for monitoring the composite specimen during testing.

The mechanically joined (dovetailed) 2T composite specimen has been received from the machine shop for fatigue precracking.

Task 5: Simulated Service Tests — Intermediate test vessel V-7A was removed from its test stand and stripped of all instrumentation and other test apparatus. Measurements of flaw and vessel dimensions were made. The ruptured region of the inside surface of the vessel was inspected and a mold was made of the deformed surface for use in making additional measurements. The rupture in the vessel was not detectable on the inside surface with dye penetrant or Zyglow. The region containing the entire flaw was removed from the vessel so that the exact extent of cracking can be determined. The removal of this material is also the first step in preparation of the vessel for weld repair for test V-7B.

Twenty residual stress measurements were taken in the weld repair heat-affected zone of a section cut from the V-9 prolongation. Compressive hoop stresses on the order of one-fourth to one-half the yield stress were observed through the wall thickness.

Static and dynamic fracture toughness results have been obtained from heat-affected-zone (HAZ) specimens from the half bend weld repair in the ITV-9 prolongation. Three V-notched positions were examined — the notch and fatigued crack in the HAZ and in the weld metal (WM) and base plate (BM) immediately adjacent to the HAZ. Identical notch positions were maintained in the static and dynamic test specimens. The values obtained are provided below.

The static fracture toughness values may be compared with the room temperature results from ITV-7 prolongation which ranged from 165 to 247 MPa $\sqrt{m}$ (150 to 225 ksi $\sqrt{in.}$) over the same depth range and the 240 MPa $\sqrt{m}$ (218 ksi $\sqrt{in.}$) value for the weld metal in the shielded metal arc qualification weld for the ITV-7 weld repair. The single dynamic fracture toughness value from the HAZ-BM location seems high and will need to be verified by additional specimens with similar precrack locations.

Test temperature °C	(°F)	Notch location	Fracture Toughness Static (K_{Icd}) MPa $\sqrt{m}$	(ksi $\sqrt{in.}$)	Dynamic (K_{Idd}) MPa $\sqrt{m}$	(ksi $\sqrt{in.}$)
21	(70)	HAZ	264	(240)	380	(346)
21	(70)	WM-HAZ	271	(247)	319	(290)
21	(70)	HAZ-BM	316	(288)	459	(418)

Macroscopic examination of fracture path in the specimens used in the slow bend (static) testing indicates that the crack front during the test does not follow a preferred path. We are making a similar examination of the specimens broken in the dynamic testing.

Task 6: Thermal Shock — Preparations for TSE-3 were continued, and the test date was moved from July 28 to August 4 because of repairs required on the data acquisition system.

A parametric analysis concerning depth of propagation of the long axial crack in TSE-3 was conducted. Sets of crack arrest toughness (K_{Ia}) values were selected by shifting the K_{Ic} vs T curve to the right 100, 200 and 300°F. For the three corresponding K_{Ia} curves the estimated points of crack arrest were 20, 30 and 40% of the way through the wall.

PROGRAM TITLE: Fission Product Beta and Gamma Energy Release

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

ACTIVITY NUMBER: 40 89 55 10 5 *(189 No. B0095)*

TECHNICAL HIGHLIGHTS

The final data-taking run for the gamma-ray energy release measurements was completed during the month of July. Seventy-eight sample irradiations were performed; half were ^{235}U samples. The remainder consisted of 19 blank samples for time-dependent background measurements, and Al, CF_2, ^{41}K, ^{36}S, ^{204}Hg, and oxalic acid samples to provide information on energy and irradiation time calibrations. Several Au-Mn samples were also run to provide thermal-neutron flux information. The 39 fissile samples were then counted twice to determine the number of fissions, n_f, using high-resolution Ge detectors, the program being identical to that detailed for our beta-ray measurements discussed in last month's report. These high-resolution data have been analyzed yielding n_f for each of the 39 samples. The average uncertainty in n_f is believed to be < 3%. We anticipate complete data reduction for the gamma-ray energy release data during August. Level "C" Node # 28039, "Conduct Final Experimental Work on ^{235}U Using In-Pile Irradiations," has been completed; this completion is already shown on the most recent Buff Book update.

Design of the fission chamber for the external-beam check is nearing completion, and fabrication is expected during August. Present plans assume that the ORR will operate for 5 days during September* and that we will be ready to test the fission chamber at that time.

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* The Oak Ridge Research Reactor (ORR) was shut down on 31 July until 1 January 1977, with anticipated 6 periods of ∿ 5 days each uptime during the five-month shutdown.

<u>PROGRAM TITLE</u>: Fission Product Release from LWR Fuel

<u>PROGRAM MANAGER</u>: A. P. Malinauskas

<u>ACTIVITY NUMBER</u>: 40 89 55 10 8 (189 No. B0127)

<u>TECHNICAL HIGHLIGHTS</u>

The first experiment of the Low Burn up Fuel Test Series was conducted at 700°C in steam for 5 hrs. We used Capsule No. 013 [T. C. Rowland, "UO_2 Irradiation for ORNL Final Report," GEAP-5642 (June 1968)], which was irradiated to 800 MWd/MT in the GETR in late 1967. A gamma scan performed before the experiment revealed ^{137}Cs activity peaks at pellet interfaces where a much earlier scan showed shorter-lived ^{131}I and ^{140}Ba. The capsule was laser-punctured for fission gas release measurement; a 1/16-in.-diam hole was then drilled to simulate a larger defect. (In experiments which are conducted at 700°C or lower, a hole is drilled through the cladding to simulate a defect, since ductile failure of full thickness cladding by internal pressurization would require unrealistically high internal pressures.) The drilling, loading, 5-hr experiment and unloading operations all proceeded according to plan. Quantitative measurements of released ^{137}Cs and ^{129}I are now in progress.

<u>PROGRAM TITLE</u>: Multirod Burst Tests

<u>PROGRAM MANAGER</u>: Robert H. Chapman

<u>ACTIVITY NUMBER</u>: 40 89 55 10 6 (189 B0120)

<u>TECHNICAL HIGHLIGHTS</u>

One simulator (SR-18) was tested this month in a steam environment to continue exploration of the effects of gas volume on deformation behavior. Fuel simulator SEMCO 2828005 was used in the test simulator (this particular fuel simulator has been used in a number of tests, and its performance characteristics are known rather well). The room temperature gas volume was 154.6 cm^3 (9.43 $in.^3$). Initial conditions for SR-18 were the same as used in PS-19 [gas volume of 37.4 cm^3 (2.28 $in.^3$)], which is the basis of comparison. The initial pressure was adjusted to 2590 kPa (376 psig); the pressure attained a maximum of 2630 kPa (381 psig) and then decreased to the rupture pressure of 2590 kPa (376 psig) - i.e., the same as the initial value. The rupture temperature was 968°C (1774°F), and the rupture strain was 22 percent. Corresponding data for PS-19 were: initial pressure of 2590 kPa (376 psig); maximum pressure of 2820 kPa (409 psig); rupture pressure of 2590 kPa (376 psig); rupture temperature of 952°C (1746°F); and rupture strain of 28 percent. The results indicate that both the pressure increases during the transient and the rupture strain decreases with increasing gas volume as expected.

Preliminary results from the metallographic examination of PS-18 [burst temperature of 1171°C (2140°F)] indicate considerable variation in the local strain around the tube circumference as evidenced by thinning of the <u>inside</u> surface of the tube wall at positions where cracks were observed in the oxide layer on the <u>outside</u> surface of the tube. Detailed wall thickness measurements are in progress.

As reported in the previous report, a small quantity of HCM Lot 5 boron nitride powder was supplied to SEMCO so that fabrication of the fuel simulators could proceed while repurification of the HCM Lot 345 powder is underway. (The HCM Lot 5 material has larger particles and a broader particle size distribution than the HCM Lot 345 material.) Difficulties attributed to the large particles were encountered in attempts to fabricate three simulators, and fabrication was suspended until the Lot 345 material can be purified. SEMCO will complete fabrication of the three simulators (the heating ribbons are not within concentricity limits) and ship them to us for possible use in tests that require circumferential temperature variations.

Our efforts to purify the Lot 345 material have not been successful. Acceptance criteria have been established for the material, based on chemical analyses, distribution of dense particles as determined by radiography, and electrical resistivity as determined from room temperature and high-temperature conductivity cells. Material purified by a leaching process passed the first two criteria but failed the conductivity criteria. Material processed by three different organizations, each using a magnetic separation technique, also passed the first two criteria but failed the conductivity tests. New conductivity cells are being fabricated to give assurance that the cells are not at fault. Studies are in progress to determine the quantity and distribution of B_2O_3 within the powder, since the resistivity of the compacted powder can be affected by these parameters.

We are unable at this time to project a schedule for purification of the contaminated material. New and more definitive development may be required before the material can be accepted for use in the fabrication of fuel simulators. Until this is accomplished, further attempts by SEMCO to fabricate the 40 fuel simulators will be held in abeyance.

An order is being initiated to purchase 20 fuel simulators from GE-Seattle to replace the previous order with RAMA. (Since RAMA was unable to successfully fabricate fuel simulators in accord with the purchase order requirements, they asked to terminate the contract.) However, GE-Seattle will be unable to proceed until the BN powder becomes available.

Mats of a report, entitled "An Apparatus for Spot-Welding Sheathed Thermocouples to the Inside of Small-Diameter Tubes at Precise Locations" (ORNL/NUREG/TM-33), were forwarded to the printers for reproduction and distribution; this completes Milestone 1(b) of 189 B0120. Final typing of a report on infrared characterization of fuel simulators [Milestone 1(c) of 189 B0120] was completed. As soon as in-house reviews and editing are complete, the report will be printed and distributed.

On July 7 and 8, J. L. Crowley, K. R. Carr, and G. Hofmann visited INEL to further enhance technical communication between INEL's fuel behavior program and our rod burst program. Much useful information was exchanged during the visit.

On July 20 and 21, R. H. Chapman and G. Hofmann visited ANL to participate in an NRC Workshop for Experimenters and Modelers. The need for additional single rod burst tests to provide more detailed information was clearly evident.

<u>PROGRAM TITLE:</u> Nuclear Safety Information Center

<u>PROGRAM MANAGER:</u> William B. Cottrell

<u>ACTIVITY NUMBER:</u> 40 89 55 10 4 (189 No. B0126)

<u>TECHNICAL HIGHLIGHTS</u>

During the month of July the staff of the Nuclear Safety Information Center (a) processed 1004 documents, (b) responded to 86 inquiries (of which 37 involved the technical staff), and (c) made 16 computer searches (of which 4 involved payment). Design Data Sheets were prepared on Units 1 and 4 of the Washington Public Power Supply System. NSIC staff received 12 visitors during June, participated in 2 meetings (including 1 presentation) and prepared 3 reviews.

Three NSIC reports were distributed in July; these are ORNL/NUREG/NSIC-118, "Siting of Nuclear Facilities, Selections from *Nuclear Safety*", and ORNL/NUREG/NSIC-126 and 127 (Volumes 1 and 2), "Annotated Bibliography of Safety Related Occurrences in Nuclear Power Plants as Reported in 1975". The first volume of the 3-part "Bibliography on LMFBR Safety Literature" is in composition, and work is underway on the other two volumes, as well as ORNL/NUREG/NSIC-128, "A Bibliography of HTGR Safety Literature". Also under way are preparations for an updated bibliography of all reports on NRC subcontract safety studies through June 30, 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 July we added 3 free and 4 paying ($290) users, bringing the total SDI users to 364, including subscribers who have paid a total of $13,359 (since October 1975).

During the month we wrote two letters (July 12 and July 30) to NRC regarding the translation of four Japanese and 12 French documents obtained through the NRC exchange program. During July the second batch of foreign-language safety documents (consisting of 11 reports which had not previously been translated and distributed) was delivered to the contractor for the preparation of microfiche and distribution of same to the special NRC distribution list. Microfiche copies of the 31 reports processed in June were distributed by mid July. A minor change in format will be made to comply with an ANSI standard now in final stages of approval.

Preparation of the sections on "Recent Developments" (nominally May and June) for *Nuclear Safety* 17(5) was completed. Manuscripts for the technical articles for *Nuclear Safety* 17(6) were also completed and submitted to NRC and ERDA for review.

PROGRAM TITLE: PWR Blowdown Heat Transfer Separate Effects

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: 40 89 55 10 3 (189 No. B0125)

TECHNICAL HIGHLIGHTS

Task 1. The bundle 2 prototypical heater W150-100 has survived 11 blowdown tests in the FCTF. In addition, a total of 8 steady-state CHF tests have been performed with this heater. The heater has operated for a total of 47.1 hr with 21.3 hr at full power. At the June 16-17 program review it was agreed that the FCTF heater rod T/C set point would be progressively increased until there was evidence of degradation of T/C readings due to overheating. Maximum heater sheath temperatures during a transient have been increased from normal values of 980°K (∿1300°F) to ∿1230°K (∿1750°F) with no apparent degradation of sheath T/C readings.

The MacBeth-Barnett correlation for a heated annulus predicts with good accuracy DNBR at points on the heater where steady-state CHF was observed for 15.5 MN/m^2 (2250 psia) experiments. The accuracy is not as good at 10.34 MN/m^2 (1500 psia); however, the point of first CHF is predicted in all cases within 21% and, for all 15.51 MN/m^2 (2250 psia) tests, the accuracy at first CHF is within 12%. The prediction at the point of first CHF is emphasized due to the phenomena associated with its occurrence. When the temperature rises due to CHF, it does so at a rapid rate, as high as 167°K (300°F) per second. This escalation in surface temperature causes more boiling at the heater surface, and can cause the CHF phenomenon to "spread" to other parts of the rod.

Further steady-state and transient tests will be carried out with a new test section with a significantly smaller diameter. This new diameter will be applicable to many of the existing CHF correlations for which the present test section is too large. The equivalent diameter of the new test section is 0.535 in (about the same as that of the THTF).

Task 2. Pretest predictions for tests 103 and 104 were prepared and quick-look reports were published for tests 102 and 104. RELAP4 THTF model development continued with improvements made in pump performance curves and pressurizer response. The T/C calibration program was successfully linked to the STAT program and THTF bundle No. 1 calibration was initiated. Interface programs necessary for utilization of the calibration data in the inverse code ORINC were defined and their programming begun. Work continues on the large plotting program necessary for production of the experimental data report. Test 104 was conducted in the Thermal-Hydraulic Test Facility (THTF) on July 8, 1976. This test was a full power (5.975 MW) test initiated from a test section inlet temperature of 559.7°K (548°F), a test section inlet volumetric flow of 0.0271 m^3/s (429 gpm), and a test section outlet pressure of 15.603 MN/m^2 (2263 psig). System decompression was accomplished by introducing a 50% inlet-50% outlet break. 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.

As the test section outlet subcooling was 13°K (24°F), the outlet break saturated almost instantaneously. The pressurizer discharge was sufficient to maintain negative core flow for ~5 sec and to cause fluid to bypass the outlet break and enter the test section. The test section inlet vertical spool piece saturated ~1.5 sec after the break following the arrival of the hot fluid from the core. CHF was observed ~0.4 sec after break initiation and the maximum observed temperature was ~938.7°K (1230°F) in the axial center of the electric core.

Task 3. Matrix test 104 was completed on 7/8/76. All systems appeared to function satisfactorily except that the signal from three of the four ion chamber gamma densitometers saturated ~15 sec into the transient. A clamp power supply in the data acquisition system also failed at the same time causing loss of certain signals, in particular the signal from the horizontal inlet turbine meter. Apparently a voltage surge of unknown (at this time) origin was responsible for this behavior. Also, the signal from the vertical inlet turbine meter was not reliable before 2 sec into the transient apparently because of some interaction of the instrument sensor and the heater rod magnetic field. This will be corrected before the next test. Since the critical portion of the data was recorded before the failures, it is not necessary to repeat this test. The primary pump was operated for ~34 hours during this period with no apparent seal problems. The generators which supply DC power to the THTF continue to be a problem because the resistance to ground can be maintained at barely acceptable levels only by intensive maintenance operations.

Task 4. In order to improve our understanding of the operation of drag disks in the THTF instrumented spool pieces, an oscilloscope was used to study the drag disk signal components during steady state flow tests in the THTF and the air-water two-phase flow facility. The drag targets of the THTF instruments were 1.27-cm-diam (0.5 in) disks supported by 0.63-cm-diam (0.25 in) rods.

THTF tests were conducted with water flows of 0.0063 to 0.032 m^3/sec (100 to 500 gpm). Oscilloscope pictures of the drag disk signals showed that the signal oscillated at a fixed frequency of 58Hz over the entire range of water flows studied. While the frequency was invarient with flow rate, the signal amplitude (relative to the mean) increased with flow rate. Results of single-phase water tests in the air-water facility were similar with a frequency of 48Hz predominating. In two-phase flow in the bubble and froth flow regimes a lower frequency of 2Hz was superimposed on the 48Hz signal and in annular mist flow a high frequency (417Hz) component was superimposed on the signal.

Since the standard frequency for vortex shedding varies directly with fluid velocity and inversely with characteristic length, the observed frequencies of 58 and 48Hz which were constant for a given drag disk were apparently the natural resonance frequencies of the two vibrating members. The Strouhal frequency was the same order of magnitude (30 to 60Hz depending on exact target size and water flow rate) for all drag disks studied in the air-water loop and in the THTF calibration test. The results will be used as a basis for the modification of the drag disk design so that the drag disk natural frequency will be significantly higher than the vortex shedding frequency. This will result in member vibration with much lower amplitude (reduced signal variance) at the Strouhal frequency.

Task 5. Further negotiations were conducted with Watlov Electric Manufacturing Co in an attempt to expedite bundle 2 fuel pin simulation production. Watlow had previously threatened to cancel the order unless the infrared scanning specifications were loosened. Since BDHT program personnel had agreed that program objectives for bundle 2 could be achieved using fuel pin simulators whose infrared scans were of the same quality as the four simulators already supplied by Watlow, ORNL agreed to Watlow's demand that the infrared scan tolerance be loosened to accept the first 4 simulators. It was further agreed that Watlow will produce 10 additional simulators which will be accepted by ORNL if they meet the infrared scan quality of the first 4 simulators and all other specifications. If these 10 simulators are acceptable, production of the bundle 2 simulators will proceed; otherwise, further negotiation will be required. Watlow will also attempt to reduce the eccentricity between heating element and sheath by 0.005 cm (0.002 in) in future simulators. Watlow is now awaiting boron nitride from ORNL to begin production of the 10 simulators.

The magnetic separation process of Magnetic Engineering Assoc. (MEA), Cambridge, Mass. appears to have produced boron nitride samples of acceptable purity for use in fuel pin simulators. These samples have passed x-ray radiographic, chemical, and conductivity cell analysis. One additional conductivity cell is presently being tested to complete the analysis of this material. If it is acceptable, MEA will purify large quantities of BN which will be supplied to Watlow for bundle 2 fuel pin simulators.

Task 6. ORNL is awaiting final specifications from NRC on bundle 3 fuel pin simulators. When this information is received, purchase orders will be placed for prototype simulators.

PROGRAM TITLE: Zircaloy Fuel Cladding Collapse Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: 40 89 55 10 7 (189 No. B0124)

TECHNICAL HIGHLIGHTS

The preliminary design for the in-reactor creepdown and collapse experiment was subjected to a detailed heat transfer calculation in which arbitrary gamma-heating levels of 4, 6, and 8 watts/gram were assumed. In each case the specimen was found to overheat by substantial amounts. This overheating resulted from the use of the double-containment design which produced a gas gap between two low-thermal-conductivity stainless steel walls. The primary and secondary containment vessels were designed to ASME Boiler and Pressure Vessel Code, Section III, specifications for allowable stresses.

The following alternative is presently being explored: a single aluminum pressure vessel that would contain the specimen and eddy-current measuring devices with a minimum of gas volume available as a potential energy source. The specimen chamber would be pressurized through a capillary tube to further reduce the available gas volume. Replacing the two stainless steel pressure vessels, separated by a gas gap, with a single wall of aluminum that has a thermal conductivity approximately 10 to 12 times greater than the stainless steel should greatly decrease the entire experiment temperature. Calculations are presently being made to determine the wall thickness of aluminum necessary to contain the pressure and to determine the temperature of the aluminum and the specimen.

PROGRAM TITLE: Zirconium Metal-Water Oxidation Kinetics

PROGRAM MANAGER: C. J. McHargue

ACTIVITY NUMBER: 40 89 55 10 9 (189 Number B0128)

TECHNICAL HIGHLIGHTS

As indicated in the previous monthly report, our final report on the diffusivity of oxygen in β-Zircaloy was issued during July, thus completing Milestone 43106.

PROGRAM TITLE: Aerosol Release and Transport from LMFBR Fuel

PROGRAM MANAGER: M. H. Fontana

ACTIVITY NUMBER: 40 89 12 10 1 (189 Number B0121)

TECHNICAL HIGHLIGHTS

CRI-III/CDV Facility:

The ORNL capacitor discharge vaporization (CDV) system is nearing completion and a first test is expected next month.

Cold Proof Tests in CRI-II:

The last in the U_3O_8 cold proof tests was completed in CRI-II with two demonstrations of the importance of thermophoretic effects. At 120°C vessel wall temperature the wall plateout to floor settling fraction increased by a factor of about 20 over that observed at room temperature. In the last run a heat source was placed in the center of the tank to heat the gas phase about 20°C above the wall temperature and comparable enhanced plateout was observed. Following this run, the RF uranium oxidation equipment was dismantled and the new arc furnace was set in place in the cubicle beside the CRI-II vessel.

Bubble Transport:

Three instrument and control drawings were prepared and are ready to issue for comments. A purchase requisition was prepared for the quick response pressure transmitters to be used in the pressure vessel.

The test stand has been fabricated and delivered. Installation of the stand will occur after the CPFF contractor completes the site preparation.

Work has been started on the electrical control cabinets and fabrication of the small fill and drain tank.

NSPP:

Fabrication and installation of the in-cell piping is approximately 70% complete. Installation of the instrument tubing in the control room from the panel to the tubing junction box in the control room is complete. Cleaning of the instrument tubing penetrations in the experimental cell is 60% complete. Installation of instrument tubing inside the experimental cell is 30% complete.

A third sodium burning test was made in the sampler test rig. The wall aerosol sampler was used for the first time in this test. Minor modifications and adjustments of the samplers are being made on the basis of the test results.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: J. P. Sanders

ACTIVITY NUMBER: 40 89 55 11 2 (189 Number B0122)

Technical Highlights

General: Development of the ORTAP code for the Fort St. Vrain (FSV) reactor was continued. Discussions were held with GAC/PSC/NRC personnel about plans for acquiring FSV transient data for use in code verification.

Development of Overall NSS Simulation (ORTAP): The plant control system portion of ORTAP was expanded to include a flow-dependent time constant for the temperature measurement; this time constant is used in determining the main steam temperature signal for the circulator speed controller and the reheat steam temperature signal for the neutron flux controller. Also, calculation of system pressure from helium mass and temperature at various points in the primary system was incorporated into ORTAP.

The overall system transient resulting from a rod withdrawal was analyzed. This transient was analyzed both for the case of a reactor trip at 140% power and for a reactor trip due to an increase in measured reheat steam temperature of 75°F above its rated value. Analysis of a slow design basis depressurization transient, assuming a reactor trip due to low primary system pressure, was initiated.

Code Verification: In a meeting with GAC, PSC, and NRC representatives, it was agreed that steady-state and transient data from FSV would be made available to ORNL as soon as possible after the 25 and 40% power runs are made.

Code Implementation: Modification of the second version of the GAC-supplied TAP code for proper execution on the IBM 360 system is proceeding cautiously. As modifications produce system errors, the errors are being eliminated by the inclusion of alternate branch statements that will duplicate the system response of the UNIVAC computer. At the same time, segments of both the TAP and RECA program are being examined to evaluate the logical operations represented by the FORTRAN statements. The meeting with GAC was helpful in that areas of coding that represent specific operations were indicated.

PROGRAM TITLE: Design Criteria for Piping and Nozzles

PROGRAM MANAGER: S. E. Moore

ACTIVITY NUMBER: 40 89 55 10 2 (189 No. B0123)

TECHNICAL HIGHLIGHTS

Code Rules Development: We met with a special task force of the ASME Working Group on Piping (ASME WGPD) in Chicago on July 8, 1976, to draft new Code rules for the design of Class 1, Class 2, and Class 3 flanged piping joints. The proposed new rules are based on studies conducted under the ORNL Design Criteria program which we presented to the WGPD in March for their review and comment. These new rules clarify the intent of the Code and considerably simplify the work required to show conformance for a given design. If accepted by the full Code Committee, the new rules will completely replace the present rules in Subsections NB, NC, and ND of Section III of the ASME Boiler and Pressure Vessel Code.

Topical Reports: The final draft of the report *Evaluation of the Bolting and Flanges of ANSI B16.5 Flanged Joints — ASME Part A Design Rules* was completed with the addition of proposed design rules for Class 1, 2, and 3 flanged piping joints discussed above. The draft has been returned to BMI for publication as an ORNL subcontractor's report.

Cylindrical Shell Studies: We have completed the draft of a report describing the results of a parametric finite element stress analysis study for isolated nozzles in cylindrical pressure vessels with internal pressure loading. The study consisted of 25 models, of which six were unreinforced, fourteen were reinforced using the so-called ASME "standard" reinforcement design, and five were reinforced using the ASME 30°-pad reinforcement. We also plan to analyze these same models for moment loadings applied to the nozzle and to the ends of the run. Results from the studies will be used to examine stress indices and spacing requirements for nozzles in reactor pressure vessels and piping.

The finite element computer program MULT-NOZZLE developed for us by Mechanics Research Inc. (MRI) has been used to analyze the required demonstration problems and reasonable results have been obtained.

However, we recently discovered the existance of a precision problem in the University of California SAP routine used to set up and solve the stiffness equations. Corrections to the program were made, but we have not yet examined the results.

ANSI B16.9 Tee Studies: The summary and interpretive report on the elastic response and fatigue tests of five 12-in.-diam ANSI B16.9 tees has been completed, reviewed and edited. We are currently preparing the report for publication. We also received the draft report for similar studies on five 24-in.-diam tees from the subcontractor — Mechanics Research, Inc.

PROGRAM TITLE: Dose Conversion Factors for Inhalation of Radio Nuclides

PROGRAM MANAGER: J. W. Poston

ACTIVITY NUMBER: ERDA 40 10 01 06 1

TECHNICAL HIGHLIGHTS

No technical highlights to report this month.