**May 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 40-551-71 and 40-552-75**

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MAY 1977

MONTHLY HIGHLIGHTS

FOR

OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS

AT

OAK RIDGE NATIONAL LABORATORY

Compiled by

Gordon G. Fee

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Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
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for the
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MASTER

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ABSTRACT

Highlights of technical progress during May 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 — J. G. Merkle attended a meeting of the ASME Section XI Working Group on Evaluation in Nashville, Tennessee, on May 3.

G. D. Whitman visited HEDL on May 11 to review the status of the Small Specimen Utilization (SSU) program.

Gordon Smith attended a review meeting at Battelle Columbus Laboratory, Columbus, Ohio, on May 10 and 11, to report on HSST crack arrest model studies.

On May 16, J. J. Holms and R. L. Knecht of HEDL and G. D. Whitman attended a meeting at Silver Spring, Maryland, to review the SSU program with C. Z. Serpan and E. K. Lynn.

J. G. Merkle attended a meeting at the NRC offices in Bethesda, Maryland, on May 20, to review work being performed by Prof. P. Paris at Washington University, St. Louis, Missouri, concerning crack stability in the upper shelf range.

A monthly program review was held in Oak Ridge for C. Z. Serpan and E. K. Lynn on May 23 and 24.

A letter was sent to L. J. Chockie, Chairman of the Subcommittee on Nuclear Inservice Inspection, Section XI ASME Boiler and Pressure Vessel Code, transmitting comments on revisions proposed on Section XI weld repair procedures. Mr. Chockie indicated that the comments and suggestions would not be included in the next published revisions of the applicable sections because of timing; however, the comments will be responded to for the next addenda.

Professor C. W. Smith of Virginia Polytechnic Institute and State University visited ORNL on May 31 to review the photoelastic studies on nozzle corner flaws.

Task 2: Fracture Mechanics and Analysis — Editing of the topical report on Intermediate Vessels V-5 and V-9 is completed. Composition and drafting have been started. Following makeup, final checking and printing, the report is scheduled to be issued in late August.

Task 3: Fatigue Crack Growth — The results from the first 4T specimen tested in LWR water environment indicate an effect of the initial applied ΔK on crack growth rate. This test was performed at a frequency of 1 cpm, an R ratio of 0.2, and an initial ΔK of $45 \text{ MN}\cdot\text{m}^{-3/2}$ with a resultant crack growth rate significantly below that obtained with specimens having lower initial loadings. Another test is planned with a 4T specimen beginning at a ΔK of $20 \text{ MN}\cdot\text{m}^{-3/2}$ to determine if lower initial loadings result in higher growth rates.

Task 4: Irradiation Effects — All three of the irradiated capsules of the second 4T-CTS irradiation experiment were disassembled and specimen and dosimeter positions verified this month. The Charpy V-notch and pre-cracked Charpy specimens have been transferred to the testing hot cells.

The hot cell slow bend apparatus has been installed and calibrated in the hot cell.

Most of the unirradiated control Charpy specimens are now on hand.

Preparations for the next 4T-CTS irradiation are continuing.

Task 5: Simulated Service Tests — ITV-7B test preparations are continuing in accordance with the schedule. Instruments and control drawings are essentially complete. The interior seal patch and patch, pressurization tube connector are installed and leak checked, the installation of interior strain gages and thermocouples is complete and the exterior gage layout has started. A test procedure has been prepared and is being reviewed before final issue. The test facility restoration is nearing completion. The high pressure intensifier failed under preoperational check-out and replacement parts have been received to rebuild the unit. The heating system is operational and a mockup of the plate coils is awaiting hydrotest. The strain gage and thermocouple cable junction boxes used in the V-7A test have been installed in the K-25 test cell and the cables have been routed to the instrument trailer.

Battelle Northwest Laboratory (BNL) will be the prime acoustic emission investigator on the V-1B test. P. H. Hutton of BNL visited ORNL on May 25 to inspect the test site and the test vessel.

The fracture toughness of the ITV-7B prototype repair weld in the HAZ and in the weld and base metal adjacent to the HAZ is being determined. The test temperature was varied from -101 to 93.3°C with the major portion of the testing performed at -46°C. The static fracture toughness (K_{Icd}) values were mostly in the 220 to 308 MPa $\sqrt{m}$ range with some values as low as 172 MPa $\sqrt{m}$.

The ITV-8 prototype repair weld was also examined in five radial regions: base metal, BM-WM HAZ, repair WM, repair WM-fabrication WM HAZ, and fabrication WM. The major portion of the K_{Icd} testing was done at -46°C and test results ranged from 44 to 334 MPa $\sqrt{m}$. The lowest K_{Icd} values appear to exist in the original vessel fabrication weld metal 1.3 to 2.5 cm from the repair weld-fabrication weld HAZ. We are continuing to examine the weldments to determine the K_{Icd} characteristics.

The third crack arrest model was tested on May 26, 1977. The model geometry was identical to that used for the first two tests. The test temperature was -21°C. Crack extension of 6 mm was indicated by the first gage of the crack extension ladder at 91 MPa. No further crack extension was indicated as the vessel pressure was increased to the maximum test pressure of 103 MPa. A visual inspection of the crack confirmed the extension. The test of the third model will be continued at a test temperature of approximately -46°C during the week of May 31, 1977. The purpose of the continued test is to determine if fast fracture will occur at that low temperature.

Material characterization has continued for the crack arrest model studies. The static and dynamic fracture toughness of quench-only plate sections 03JZ and 04BE has been determined with precracked Charpy specimens. The dynamic fracture toughness, K_{Idd} , of plate 03JZ from RT oriented specimens does not differ greatly from the static fracture toughness, K_{Icd} , in the transition regime but has a higher shelf value.

The K_{Icd} and K_{Idd} results from 04BE and WT oriented specimens indicate a transition regime from below -73.3 to 10°C with K_{Idd} values of 79 MPa $\sqrt{m}$ at -73.3°C and 123 MPa $\sqrt{m}$ at 20°C. The upper shelf was

approximately 121 MPa $\sqrt{m}$. K_{Idd} values were 22 MPa $\sqrt{m}$ higher than the K_{Icd} values at a given test temperature.

Task 6: Thermal Shock — Efforts were continued on the TSE-3, TSE-4 topical report and with the cryogenic-quench experiments. With regard to the latter, a series of liquid nitrogen quench experiments was conducted with the 150-mm $\times$ 150-mm hollow-cylinder specimen, using initial temperatures of 93, 129, and 149°C. The inner surface was coated with six consecutive applications of 3M-77 (no quench experiments between applications). Heat transfer coefficients as a function of metal surface temperature were derived from the 93 and 129°C tests and were used to calculate transient temperatures for corresponding thermal shock experiments to the 990-mm-OD $\times$ 690-mm-ID cylinders. A fracture mechanics calculation was completed for the 93°C, large-cylinder experiment, and the results indicated warm prestressing for long axial flaws with $a/w \geq 0.5$. Also, for $a/w = 0.5$, $(K_I/K_{Ic})_{max} = 1.8$. Based on this data it appears that a demonstration of warm prestressing under thermal shock conditions may be possible. However, there are several thermal-hydraulic problems, such as axial variations in heat transfer coefficient, that must be investigated before arriving at final conclusions. Such investigations are under way.

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

The major activity during May was preparation for a gamma-ray data-taking session for thermal neutron fission of ^{239}Pu and then taking of the data. Data were obtained for 26 fissile samples, fewer than planned because of the shortened period (four days) allowed us for these measurements. However, the raw data appear to be free of inconsistencies and may prove to be sufficient if subsequent analysis indicates that all of the obtained data may be used.

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

Test HBU-6 was performed as part of the High Burnup Test Series at 700°C for five hr in flowing dry air. This was the last scheduled experiment in which a defect was formed by drilling a 0.159 cm dia hole through the cladding. From the resistance to argon flowing through the inlet half of the pellet-to-clad gap space, we calculated the effective radial gap width to be 18 μm . Similar measurements with other H. B. Robinson test segments used in the High Burnup Test Series gave gap widths ranging from 18 to 22 μm . These gap sizes are probably a lower limit, since no correction was made to account for flow resistance occurring at the drilled exit opening.

Some preliminary determinations of fission product release during test HBU-6 have been obtained. Approximately 1.2% of the original inventory of ^{85}Kr was released; 40% of this was released during the warmup and first 30 min at 700°C, whereas the remainder was released at a linear rate over the five hr test period at 700°C.

Approximately $1.0 \times 10^{-3}\%$ (4.7 μg) of the total cesium inventory was released and at least $5.8 \times 10^{-4}\%$ (2.4 μg) of the total ruthenium was released. The release of both cesium and ruthenium was considerably greater in air atmosphere than that observed in a corresponding test in steam.

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

The status of the conceptual design work for the Fission Product Transport Test Facility (FPTTF) is given by the following listing, which includes the titles of the required drawings:

	Percent completion (5-20-77)
Model steam generator	100
Model pump	100
Fission product injector	80
Pipe insert (typical for gamma-scans)	0
System flowsheet (including cleanup system)	50
Instrumentation application diagram	100
Equipment layout and piping	60
Main steam supply system	(Not required)
Wet steam supply	100
Condenser	80
Steam separator	50
Weigh tank	100
Cleanup tank (acid)	10
Electrical power supply	0
Ventilation system	60

We propose the use of either a commercially available package boiler or on-site steam supply for the main steam supply system. Overall, the conceptual design work is about 70% completed.

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

Fourteen additional fuel simulators were received from SEMCO and inspected during May; one additional unit was received and is undergoing inspection. All 14 of the units exhibited acceptable infrared scans; in fact, they represent some of the best produced by SEMCO to date. Four units of the lot were repaired units (rejected earlier by SEMCO on the basis of low insulation resistance) and have low BN density in the "unheated" extensions of the internal element. These are being evaluated to determine if they can be accepted; the other ten units of the lot were accepted. Acceptance of these completes the order for 40 units for the first two 4 x 4 test bundles. The five units presently being evaluated represent partial shipment of the ten units added to the order for future use. Production of the remaining five units on this add-on order is nearly complete; shipment is promised the second week of June.

Sixteen fuel pin simulators were received from G.E. (Seattle) during May. Eleven of these have undergone acceptance inspection and evaluation; seven units exhibited very good infrared scans. The other four exhibited unacceptable scans and were rejected. The remaining five units are presently undergoing acceptance inspection.

W. E. Baucum, one of our fuel simulator development specialists, visited G.E. (Seattle) on May 23-27 to discuss the evaluation of the rejected units and to suggest improvements to prevent recurrence of the problem. Since procedures and production of the total order (original order for 20 units and an add-on order for five units) are in effect, G. E. is of the opinion that it is best to proceed with production and accept the rejection rate that might incur without the change. G.E. promises completion and shipment by June 30 of the original order of 20 units and by July 15 of the add-on order of five units.

On May 20 Mr. Ralph Reed, Plant Manager, G.E. (Seattle) visited ORNL to determine the fuel simulator market potential (at ORNL), to appraise us (in greater detail) of their capabilities in this field, and to discuss potential contractual arrangements for future orders for fuel pin simulators. He stressed their technical resources for engineering and development, their independent quality assurance organization for monitoring quality and compliance with specifications, and their production-line manufacturing capability in accordance with developed procedures. All of these areas are important to the production of acceptable fuel pin simulators on schedule. They apparently do not commit fabrication techniques and schedules to production status until all the details have been developed and demonstrated by engineering development personnel. Thus, all the problems are resolved before production is initiated. This gives them assurance that quality products are produced on schedule. Such an arrangement can lead to higher unit costs but the purchaser has the additional benefit of assured quality products on schedule. Since they have developed successful powder-filling techniques, they are not too interested in the use of preforms. We stressed the advantages of the preform approach, and we believe they would accept preforms as an improvement.

In-house development of fuel pin simulators based on preforms and computer-controlled, machine-wound heating ribbons continues to show promise. These prototype simulators have been fabricated and evaluated to determine the effect of various parameters. Additional units will be fabricated in June to determine optimum preform density and swaging parameters. Plans are to fabricate six prototype units, using the best set of parameters as a demonstration of the procedures.

Fabrication of the first 4 x 4 test bundle is proceeding rather well; however, since this is the first time many of the fabrication procedures have been used, some delays have been encountered. One fuel pin simulator was rejected due to detachment of two of the internal thermocouples. This unit was placed in storage (for later testing as a single rod assembly) and a replacement unit was assembled. Three assemblies were rejected on the basis of potential alteration of the

Zircaloy metallurgical condition resulting from exposure to excessive temperatures during bakeout of the completed assemblies to improve insulation resistance. (This operation was abandoned for further use.) The internal fuel simulators were recovered from these rejected assemblies and are being used to fabricate new test simulators. As a result of these delays, the burst test on the bundle will be delayed approximately two weeks. We anticipate at this time that the burst test will be conducted in late July or early August.

A flow shroud and modifications to an existing flow loop have been designed to permit flow characterization of the 4 x 4 test bundles; a "dummy" bundle will be used for the pre-burst-test characterization. Fabrication of the test components is underway.

R. H. Chapman and G. Hofmann participated in the Cladding Experimenter's Workshop, the MATPRO Workshop, and the Cladding Review Meeting at Idaho Falls during the period May 16-24. These sessions, held as part of the information exchange with FRG, were also attended by personnel representing the various cladding research programs in the FRG.

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 May the staff of the Nuclear Safety Information Center (a) processed 1188 documents, (b) responded to 76 inquiries (of which 46 involved the technical staff), and (c) made 16 computer searches (of which three involved payment totaling \$144.10). A bibliography of 354 documents, received from the ACRS in March, was completed in April and distributed in May. We are now in receipt of an additional 260 ACRS documents which are being processed. NSIC staff received ten visitors during May, participated in three meetings, and prepared one review.

One NSIC report was issued during the month, the fourth volume of the five-part "LMFBR Safety: Review of Current Issues and Bibliography of Current Literature", ORNL/NUREG/NSIC-132, covering the period 1974-1975. Several other NSIC reports are in various stages of preparation. The report, "Design Data and Safety Features of Commercial Nuclear Power Plants Including Cumulative Index for Vol. I - VI", ORNL/NUREG/NSIC-136, is in reproduction. Work has begun on several other reports, including "A Bibliography of HTGR Safety Literature", ORNL/NUREG/NSIC-128; "Annotated Bibliography of Safety-Related Occurrences in Boiling-Water Nuclear Power Plants as Reported in 1976", ORNL/NUREG/NSIC-137; "Annotated Bibliography of Safety-Related Occurrences in Pressurized-Water Nuclear Power Plants as Reported in 1976", ORNL/NUREG/NSIC-138; "LMFBR Safety Vol. V", ORNL/NUREG/NSIC-139; "Materials Technology Bibliography", ORNL/NUREG/NSIC-140; and "Summary Data for All Commercial Nuclear Power Plants", ORNL/NUREG/NSIC-141.

We received 16 foreign safety reports (all from Federal Republic of Germany) under the NRC exchange agreement during May. Our recommendations concerning the translation of these documents were contained in letters of May 23, 1977, and May 31, 1977, to G. L. Bennett. In addition,

in May we received copies of 41 reports on NRC-sponsored safety projects which were distributed (cover letters of May 13, 26 and 31) 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 were first tested by the Systems Engineering Section of the Computer Sciences Division at X-10 and then sent to K-25 for installation and further testing which was still in progress at the end of the month.

All manuscript for *Nuclear Safety* 18(4) has been completed and submitted for publication. All technical articles for *Nuclear Safety* 18(5) have completed the peer review procedure and were mailed to NRC and ERDA on May 20. All of the technical articles for *Nuclear Safety* 18(6) have been received and are either being reviewed or revised following review.

At the special request of NRC, we are coordinating the technical review of a series of articles on NRC-sponsored decay heat studies. Review of an article by Dickens, Peelle, et al. of ORNL (third in the series) was initiated following receipt of the article on April 29, 1977. Review of the article was scheduled for completion by the end of May but as of that time only four of 13 reviewers' comments were in hand. However, the missing nine have been contacted by phone and will be forthcoming soon.

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. The prototypical bundle 2 heater rod SN-13 (out of specifications on heater element eccentricity) has survived six blowdown tests in the Forced Convection Test Facility. This heater has been operated for a total of 30.7 hr, with 5.6 hr at full power. During transient critical heat flux conditions, the heater temperatures reached 1436 K (2125°F) in the highest power zone. The resistance of the heating element has remained essentially constant. However, the temperature difference between middle and sheath thermocouples has increased during the testing period and the increase was especially pronounced in the high heat flux zone. This effect contrasts with results of tests of other bundle 2 prototypes which showed nearly constant radial ΔT 's. Since radial ΔT is insensitive to flow rate, system pressure, heat transfer coefficient at the surface, and gap resistance between the heater sheaths, the most plausible explanations for the observed increase in radial ΔT are (1) a change in heater thermal properties; (2) a change in generated heat flux profile; (3) distortion of the heater; and (4) degradation of the thermocouples. Ambient temperature measurement and T/C electrical properties measurements, however, indicate that the T/C's are in good condition. Although the heater is still operable, analysis of the heat transfer data, which is of prime importance, is difficult except for determination of time to critical heat flux. Testing will continue in an effort to define the failure criteria of this heater.

Task 2. Test 160 was conducted in the Thermal-Hydraulic Test Facility (THTF) on May 27, 1977. Test 160 was a 45-rod (rods 19, 24, 39, and 47 were inactive)(4.493 MW) test initiated from a test section inlet temperature of 557.6 K (544°F), a test section inlet volumetric flow of 0.0189 m³/sec (510 gpm), and a test section outlet pressure of 15.624 MN/m² (2337 psig). 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 ~ 2 sec into the transient and then the power was decayed with a time constant of ~ 0.45 sec.

The results of this test will be compared to other THTF tests in the 150 series. Both spatial and temporal shifts in CHF were observed relative to previous tests.

Task 3. The schedule for Test 160 was delayed for one week due to two unrelated problems. First, measurements showed the internal resistance to ground of the four dc power generators was unacceptably low. Cleaning the generators was successful in increasing the resistance to levels at which operation could be safely permitted. Following the cleaning operation, inspection clearly showed that repeated cleaning was having a deleterious effect on generator insulation. Maintenance engineers recommended replacement of the insulation as soon as possible. The insulation is in such condition that continued cleaning may result in a decrease in resistance to ground rather than an increase.

Following generator maintenance a rotor blade failed on one of the spool piece turbine meters requiring replacement of the rotor. Test 160 was successfully completed on May 27, 1977.

Task 4. Preliminary tests of the transient response of the THTF spool piece turbine meters have been conducted in the air-water, two-phase flow facility; the meter was installed in a horizontal run of pipe, and the tests included response to single-phase air and water and two-phase flows. The test procedure included restraining the rotor with the flow at an initial steady-state value. The rotor was then released and the transient signal monitored on the PDP-8 as the rotor accelerated. In general, the rotor accelerated to a steady-state speed within one revolution. Response times were consistent with values reported in the literature and no significant difference was noted in the response of the rotor to single- and two-phase flows.

Task 5. Watlow is continuing production of bundle 2 fuel pin simulators and eight units are expected to be shipped to ORNL the first week of June 1977. These units contain Watlow supplied centerline thermocouples which have all survived the fuel pin simulator fabrication processes. This shipment will bring the total number of units received to 61 with 34 units remaining to be delivered.

Task 6. CRNL has received the short test sample of bundle 3 fuel pin simulators (FPS) from the General Electric Company. Preliminary analysis has indicated that this unit incorporates all the required features of the bundle 3 FPS design and is generally acceptable with respect to heating element concentricity, boron nitride fill, thermocouple attachment, and infrared scan. However, a high electrical resistance was measured in the heating element to copper lead-in joint, and one of the sheath thermocouples had broken loose from the sheath. General Electric has made design changes to eliminate these problems.

There has been no change in the status of the bundle 3 prototype FPS orders with Sensor Dynamics, RAMA, and Watlow.

Purchasing Division is still negotiating with the Eagle Picher Company to order boron nitride preforms for RAMA.

Task 8. A computer program, OTOCI, has been written to interface between the COBRA and ORINC codes. This code basically generates a tape which contains collated heat flux data from ORINC. Since for the series 100 tests only one-eighth of the bundle needs to be modeled due to symmetry, OTOCI averages the heat fluxes in each of the eight one-eighth sections of the bundle according to the rod position. Also COBRA IV has been modified in order to enable it to read these heat fluxes. Plotting routines are being developed that will plot all parameters of interest either as a function of time or as a function of core axial position for any given time. A copy of REHASH, the CDC plotting package for COBRA IV is being used as a starting point.

PROGRAM TITLE: Zircaloy Fuel Cladding Collapse Studies

PROGRAM MANAGER: D. O. Hobson

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

TECHNICAL HIGHLIGHTS

K. R. Thoms visited ECN-Petten in May to discuss the final design for the mockup capsule and the first pressurized test. He took both assembly and detailed parts drawings for Dutch comments. No substantial changes were called for.

A formal request was made for the Dutch to begin construction of the pressurization system and the capital funding has been approved through ORNL management. A purchase order will be sent to ECN.

Out-of-reactor creepdown testing has resumed.

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

No technical highlights to report this month.

PROGRAM TITLE: Aerosol Release and Transport from LMFBR Fuel

PROGRAM MANAGER: M. H. Fontana

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

TECHNICAL HIGHLIGHTS

CRI-III/CDV:

Eight CDV experiments were conducted this month. Four of these were part of the "upper-limit" source series, and the other four were performance and exploratory tests of potential under sodium designs for use in the FAST experiments. The latter included two tests of the FAST "prototype" design and two using a stainless steel containment rather than the standard quartz. The prototype design did not perform well and the results are discussed in this report under the heading "BUBBLE TRANSPORT/FAST". One of the stainless steel designs was successful and, with some modifications, may serve as an alternate to the prototype design.

The four "upper-limit" series tests included two at low mean energy levels (~350 and ~475 joules/gram above melting) and two at high mean energy levels (both at ~1600 joules/gram above melting). The data for these have not yet been fully assimilated; however, each appeared to meet the objectives of providing additional data points for airborne yield and primary aerosol sizes as functions of the energy level. Additional information was obtained for the transient aerosol behavior by using mass samplers and the Andersen Impactor. For the high energy tests, energy inputs in excess of 30 kJ were observed creating an initial aerosol concentration of ~3 g/M³. The impactor gave a d₅₀ of ~0.2 micron with a standard deviation (σ) of ~4.3 from a sample taken three-four minutes after aerosol formation. Subsequent samples at 12.5 and 45 minutes gave similar mean sizes of ~0.6 micron with σ 's of 1.95 and 2.4, respectively.

A secondary objective of these tests was to measure the electrical resistance of the UO₂ sample throughout the pre-heat and capacitor discharge periods. This may provide an indirect measure of the mean

transient temperature to be inferred from theoretical or empirical correlations of electrical resistance and temperature. Such measurements of electrical resistance were obtained and the value measured at the end of the pre-heat period compared well with the "known" value for molten UO_2 at atmospheric pressure.

BUBBLE TRANSPORT/FAST:

Two performance tests of the prototype FAST CDV unit were performed in the CRI-III Facility. The prototype design is similar to the standard CDV units used in CRI-III, except it has extra provisions for accommodating the higher external pressures expected in the FAST experiments and has an added alumina containment to isolate the sample from sodium. In the first test, the alumina fractured during the pre-heat cycle. This did not happen in the second test; however, early arcing in both tests limited the amount of energy deposited in the sample to about 3 kJ. Modifications planned for the next performance tests include stress relieving of the alumina, replacing the alumina with stainless steel and adjustments in tolerances that may influence potential arc paths.

The FAST Facility is still on the construction schedule established after it became necessary to fabricate the vessel in-house. The primary vessel shell and bottom head are now complete and are undergoing final inspection. Design of the single stage mass sampler is complete, and design of an eight stage sampler is 90% complete.

NSPP:

The third sodium pool fire, single component, aerosol test (designated Run 103) was completed. This is one of the initial series of single component sodium oxide aerosol tests being conducted at progressively higher levels of initial concentration to provide a base for comparison with later two-component mixed aerosol tests to provide data appropriate for validating scale features of aerosol behavioral models when related to experiments in different size vessels.

The previous test (Run 102) had introduced 5 kg of sodium to create a maximum airborne concentration of $\sim 11 \text{ g/M}^3$. Not all the test

objectives were achieved. The system of cascade impactors for aerosol size measurement did not function properly, and the test was conducted in a moist atmosphere. Consequently, Run 103 was intended to duplicate Run 102, except in a dry atmosphere. This would provide an early comparative test between "moist" and "dry" behavior as well as provide the aerosol size measurements missed in Run 102.

All the analytical data for Run 103 have not yet been processed. It is apparent, however, that in spite of a concerted effort and extensive drying procedures, a significant amount of moisture must have also been present during Run 103 even though the measured relative humidity was only 6% at the start of the experiment. A thorough post-test examination of the vessel, the attached process piping, and the spray system revealed several places where small, but significant, amounts of water had been trapped and apparently not removed by the air blowing and vacuum drying procedures. It is postulated that the sodium fire must have heated and driven this trapped water into the vessel during the run. Steps are being taken to eliminate all potential water traps and to ensure that the level of moisture can be controlled in future experiments.

All other test objectives of Run 103 appear to have been achieved, including the transient size distribution measurements. The preliminary data indicate that Run 103 duplicated Run 102 in all respects.

ANALYTICAL:

An outline of the important elements requiring quantification for source term assessment was made and transmitted to NRC.

The simplified thermal model for attenuation of the HCDA bubble, completed earlier, requires the use of three separate condensation heat transfer coefficients - to the bubble interface, to internal structure, and to entrained sodium droplets. A separate model developed to determine the coefficient for condensation at the interface was completed and submitted for publication in the open literature.

A preliminary model was outlined for condensation onto internal structure, and work started on a computer solution which will provide the second of the three required coefficients in the simplified model.

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: Investigations of rod pair withdrawal accidents were continued. The code was modified to account for the 150 sec rod bank insertion time on a scram and for several variations of the minimum assumed scram reactivity available. Previously, a "standard" GAC supplied scram power vs time curve had been used. The modification resulted in higher peak fuel temperatures. The sensitivity of the results to various control system parameter variations has also been studied. The circulator-turbine model (ORCIRT) was modified so that the turbine will operate on wet steam. Simulated FSV plant operation at 25% power has been achieved. However, further trimming of plant control parameters is continuing in order to achieve stable conditions at the low power. A modification to the feedwater heater subroutine has resulted in a 30% decrease in computer running time for certain transients.

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 annual out-of-town meeting in Nashville, Tennessee, during the week of May 2-6, 1977. At these meetings Code rules modifications for the design analysis of Class 2 and Class 3 flanged piping joints (ORNL/Sub/2913-3) which were passed earlier by WGPD, were presented to the Subcommittee on Nuclear Power and were rejected. The rejection was based on arguments that special exemptions to the proposed rules are needed for small diameter, low temperature water lines, and that the proposed rules essentially prohibit the use of low strength bolting materials. The proposed rules will be returned to WGPD for further consideration.

The Pressure Vessel Research Committee held its spring meetings in New York during the week of May 16, 1977. At these meetings the S/C on Piping, Pumps, and Valves and the Design Division accepted and approved our report ORNL/Sub/2913-5, *Dimensional Control of Buttwelding Pipe Fittings for Nuclear Power Plant Class 1 Piping Systems*. This report will be forwarded by PVRC to the ASME Boiler and Pressure Vessel Code Committee, the Manufacturers Standards Society (MSS), and to the American National Standards Institute (ANSI) for use in their Codes and Standards. PVRC also discussed projects and budget proposals for fiscal year 1978.

Miscellaneous Studies: A study for the development of stress indices and flexibility factors for piping elbows to account for the restraining effects of attached pipe (end effects) was resurrected this month for completion. Computer runs from a finite element stress analysis parameter study, consisting of about 100 cases, were completed some time ago and the results were stored for later analysis and correlation. We plan to use these data, along with additional information from the literature, in the development of more accurate stress index and flexibility factor formulas for use in the ASME Code design qualification analysis procedures.

Nozzles in Cylindrical Reactor Pressure Vessels: A paper entitled *Validation of the Finite Element Stress Analysis Computer Program CORTES-SA*

for Analyzing Piping Tees and Pressure Vessel Nozzles for presentation at the ASME Petroleum and Mechanical Engineering Conference, Houston, Texas, Sept. 18-22, 1977, was completed. This paper presents comparative results between finite element calculations and experimental data for six isolated nozzle models and discusses the validation process for the corrected version of CORTES-SA.

Using the new version of CORTES-SA, we began rerunning the 25 model isolated nozzle parameter study, and completed analyses for 18 of the models. We plan to complete the parameter study and to begin preparation of the final report in June.

The automatic mesh generator for phase 3 for the MULT-NOZZLE finite element computer program for analyzing three closely-spaced cylindrical pressure vessel nozzles was completed. Data input preparation was also completed for the validation-demonstration models. We plan to analyze these models during the second week of June when MRI personnel from Los Angeles can be present.

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

Monitoring Methods to Detect and Quantify Flow-Induced Vibrations of In-Vessel Components. Work on the identification and characterization of noise-equivalent sources in BWRs is proceeding on schedule (see *Noise Diagnostics for Safety Assessment Quarterly Progress Report, January - March, 1977, ORNL/NUREG/TM-112*).

Loose-Parts Monitoring Systems. An investigative program for achieving specific goals in the four general work areas identified (see *Noise Diagnostics for Safety Assessment Informal Progress Report, April, 1977*) has been formulated. We have a need for six equipment items not presently on hand: (1) a two-channel (or more) transient signal capture device, (2) a generator of complex waveforms, (3) a standardized impacting device, (4) accelerometers of various types, (5) preamplifiers of various types, and (6) a test fixture. Detailed technical specifications for this equipment and possible sources of supply are now being investigated.

Surveillance and Monitoring by Noise Analysis. As a part of our study of the applicability of statistical algorithms for the automated surveillance of reactor protection systems, we attended a meeting of the Mechanical Failures Prevention Group in Chicago, May 17-19, 1977.

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