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ORNL/TM-5294

January 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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National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy $4.00; Microfiche $2.25

ORNL/TM-5294

Contract No. W-7405-eng-26

JANUARY 1976

MONTHLY HIGHLIGHTS

FOR

OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS

AT

OAK RIDGE NATIONAL LABORATORY

Compiled by
Gordon G. Fee

FEBRUARY 1976

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

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PROGRAM TITLE: Heavy Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

ACTIVITY NUMBER: 40 89 11 90 1 (189a No. B0119)

TECHNICAL HIGHLIGHTS

Task 1: Program Administration — A midyear program review was held at NRC offices in Germantown, MD on January 8.

On January 14 and 15, G. D. Whitman traveled to the University of Missouri, Columbia, MO, to participate in a review and coordination meeting on fatigue crack growth studies.

G. D. Whitman traveled to Palo Alto, CA, to attend the EPRI Pressure Vessel Study Group meetings on January 21 and 22.

On the afternoon of January 22, G. D. Whitman visited MARC Analysis Corporation offices in Palo Alto, to discuss comments on their draft report on the fracture analysis of nozzle corner cracks.

J. G. Merkle and G. D. Whitman visited HEDL offices in Richland, WA, to review their program on the utilization of small fracture mechanics specimens in fracture toughness determinations.

Task 2: Fracture Mechanics and Analysis — Drawings are being prepared for sectioning and machining specimens from an HSST submerged arc weldment (51B) to determine the effect of an additional 80 hr stress relief at 621, 638, 654 and 671°C (1150, 1180, 1210 and 1240°F). The weldment, during its fabrication, received a stress relief of 40 hr at 621°C (1150°F).

Task 3: Fatigue Crack Growth — Two additional test chambers are being fabricated at Westinghouse Electric Corporation for environmental testing of pressure vessel steels. One chamber will accommodate 2T specimens and the other is designed to contain a 4T specimen.

Task 4: Irradiation Effects — Fabrication of support structures for the BSR, utilities, capsule parts, and instrumentation is on schedule. Assembly of capsules will start as soon as specimens are received in February.

The machining of the composite 2T compact tension specimen with the dovetail joints connecting the three pieces is essentially complete.

Overall the appearance of this specimen is excellent, but the notch tip requires further machining to correct a slight misalignment of the root as it passes through the specimen. A new grinding wheel has been ordered so that this problem can be corrected.

Task 5: Simulated Service Testing — Tentative plans for model tests and intermediate vessel tests for the study of crack propagation and arrest under sustained loading are being studied to determine practicable schedules and budgets. Consideration is being given testing IVT-10 (vessel with nozzle) under hydraulic pressurization at low temperature early in FY 1977.

Analytical studies for propagation and arrest studies show good agreement between ORNL simplified calculation of PWR depressurization rates and RELAP calculations. Analysis also shows that, in cylinders of the dimensions of ITV's, stress wave propagation around the cylinder will have negligible effect on the stress intensity factor of a propagating crack. Model vessel dimensions have been chosen to satisfy the condition that the propagation around the shell be negligible.

Plans for residual stress measurements in the V-9 prolongation (the weld procedure qualification piece for the Section XI repair of V-7) have been made and baseline measurements are being made.

Preparations of ITV-7A for pneumatic testing are continuing. Most outside procurement items have been received. The patch design is still under development. A mockup has been fabricated to check assembly problems. Seal materials and arrangements are being tested. In order to obtain a reliable seal we shall probably have to eliminate all instrumentation on the inside surface of the vessel beneath the prepared flaw (3 UT transducers, several strain gages, ID COD gages, and crack propagation gages). Patch development will delay testing of V-7A by 6 to 8 weeks, if no failures in testing are experienced.

Task 6: Thermal Shock — Quenching asymmetry studies were continued. As a part of this effort the thermal hydraulic test specimen (TSV-F) was reinstrumented with thermocouples on the inner surface to provide a better temperature map and to eliminate transverse leads. Then a test similar to TSE-2 was conducted. Results indicate that transverse leads contributed to the previously observed asymmetry but that asymmetries associated with the pipe elbow just below the test specimen were significant. Based on these data a decision was made to lengthen the vertical section of the inlet pipe before additional thermal shock tests are conducted.

A proposal was submitted to NRC for two additional thermal shock experiments. The first would be a more severe thermal shock to TSV-1 with the present long, axial crack, and the second would involve an additional long, axial crack in the same specimen and the same more-severe thermal shock. Crack extension would be expected in both cases.

Compact tension results have been obtained from the TSV-2 prolongation (1-002) aged for 24 hr at 288°C (550°F) using 0.394T, 1T and 2T specimens from seven depth locations and over a test range of 65.6 to 93.3°C (150 to 200°F). These results are listed in Table 1.

Table 1. Compact tension results from TSV-2 prolongation aged for 24 hr at 200°C (550°F)

Test temperature °C	Test temperature °F	K_{Ic}[a] MPa $\sqrt{m}$	K_{Ic}[a] Ksi $\sqrt{in.}$	Depth	Specimen size
80.0	176	109	99	0.96T	0.395T
80.6	177	108	98	0.76T	0.395T
67.2	153	101	92	0.91T	1T
79.4	175	125	114	0.70T	1T
79.4	175	111	101	0.30T	1T
93.3	200	138	126	0.50T	1T
65.6	150	104	95	0.32T	2T
79.4	175	110	100	0.68T	2T

[a]Calculated by equivalent energy method.

Task 7: Reheat Cracks — Examinations of the heat-affected zone from ITV-4 are continuing.

PROGRAM TITLE: Fission Product Beta and Gamma Energy Release

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

ACTIVITY NUMBER: 40 89 09 50 1 (189a Number B0095)

TECHNICAL HIGHLIGHTS:

The two major activities during January 1976 were (a) calibration of the 10-cm^3 Ge (intrinsic) detector, and (b) preparation of two reports on the interim measurements.

The Ge (intrinsic) detector was originally obtained to be used as the beta-ray detector. However, the two-scintillator system was made to operate with a satisfactory beta response as well as partial identification and rejection of gamma rays, and it was used for the preliminary beta-ray energy-release measurements. In surprising contrast, the 10-cm^3 Ge detector exhibits a complicated response to incident beta rays, and it is now evident that it would be very difficult to ascertain the response for $E_\beta > 1$ MeV, since there are no convenient electron conversion sources having $E_\beta > 1$ MeV. We have now confirmed the statement made in the last monthly report, namely that for energy-release measurements (where detector energy resolution is not the primary consideration) the scintillator system is the preferred system. However, the 10-cm^3 Ge detector has very good energy resolution for monoenergetic beta rays ($\sim$ 4 keV FWHM for 600-keV betas) and also has good resolution and efficiency for low-energy gamma rays. We expect to be able to study selected ^{235}U samples for information on discrete low-energy gamma rays ($E_\gamma < 0.2$ MeV). and possible high-yield conversion electrons (E_β between 0.1 and 1 MeV).

Two reports were prepared during the month. The quarterly report describes the interim beta-ray energy release measurements and has been sent to press. The second report, still in the technical review process, is the topical report planned for publication in FY 1976, which was given the working title, "Interim Results of Fission Product Energy Release from Thermal Fission of ^{235}U," in our 189a. This report will also satisfy our Buff Book Level "C" milestone Mode No. 28030.

We have also initiated the planning of the External Beam Fission Normalization Check, Buff Book Level "C" Mode No. 28027. The major technical problem to be solved is to design a fission chamber having accurately known high efficiency but small dimensions so that uncertainties in the location of the fission fragments will not be a serious problem in gamma counting.

The large volume (80 cm^3) Ge(Li) detector, which will be used to monitor the ^{99}Mo (and possibly other decay product) gamma rays for number number-of-fission determinations, was delivered at the end of the month. Tests to ensure that it meets specifications are in progress, and a suitable lead-shielding "cave" will be fabricated to provide a low-background environment. Efficiency calibration will be initiated, and probably completed, during February.

PROGRAM TITLE: Fission Product Release from LWR Fuel

PROGRAM MANAGER: A. P. Malinauskas

ACTIVITY NUMBER: 40 89 12 70 1 (189a B0127)

TECHNICAL HIGHLIGHTS:

Implant experiments 4 and 5 were performed at 1100°C (1 hr) and 700°C (2 hr), respectively. The fission product simulants were identical to those used in implant 3 (900°C, 2 hr), viz., CsOH, CsI, and TeO_2. Only a preliminary analysis of these experiments has been completed.

A lengthened induction coil was used with implant 4 to reduce the axial temperature gradient in the fuel rod. This apparently heated the stainless steel ferrule fitting at the inlet end sufficiently to allow leakage of the argon pressurizing gas so that the typical swelling and rupture did not occur. Oxidation of the Zircaloy cladding at 1100°C produced smoke-like deposits in the furnace tube that ranged from black to white in color. The orifice at the entrance to the impactor became partially plugged shortly after reaching 1100°C; this necessitated reduction of the steam-argon flow rate. The cladding became extremely weak and brittle; samples will be examined for hydriding. Migration of iodine was observed within the rod in the direction of the cooler outlet end.

The cladding of implant 5 was thinned in a different manner than that employed in previous tests at 700°C, so that it ruptured with approximately 500 psig internal argon pressure as opposed to the 250 psig burst pressure obtained in the earlier 700°C experiments. A faint smoke-like deposit which was observed on the furnace tube is probably due to the higher rupture pressure. The amount of dark material collected on the impactor stages was also larger than normal for 700°C experiments. Migration of iodine within the fuel rod was minimal. In contrast to implants 3 and 4, there was only partial migration of tellurium to the Zircaloy cladding. At least 80% of the iodine collected in the thermal gradient tube and impactor appeared immediately after rupture.

Samples of particulate material collected in the impactor during implants 1 and 2 (CsI) and implants 3 and 4 ($CsOH + CsI + TeO_2$) were submitted for X-ray diffraction examination. $CsOH \cdot H_2O$ and $CsOH \cdot xH_2O$ were present in all experiments.

The H. B. Robinson high burnup fuel rod segments were received from Battelle Columbus Laboratories. Six segments were selected and sent to the HRLEL (High Radiation Level Examination Laboratory) for detailed gamma scans. The other rod segments were placed in storage.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: 40 89 12 00 1 (189a No. B0120)

TECHNICAL HIGHLIGHTS:

On January 5 and 6, J. S. Bowling of SEMCO visited ORNL to discuss technical and contractural matters related to our option to purchase 40 additional heaters. Due to unforeseen fabrication problems, SEMCO has experienced an unusually high rejection rate on the final acceptance inspection and is unwilling to accept the option at the previously quoted price. As a result of our discussion, SEMCO will continue efforts to resolve the technical problems. After resolution of the fabrication difficulties,they will provide a new production schedule and requote a fixed price for the 40 heaters. One of our technical consultants, D. L. Clark, visited the SEMCO plant during the period January 12 thru January 22 to aid resolution of the fabrication difficulties.

Similarly, on January 14 and 15, D. E. Williams of RAMA visited ORNL to discuss technical and contractual matters related to our outstanding purchase order for 20 heaters. As a result of these conversations, RAMA will continue their effort to produce heaters in accordance with our requirements. Projected delivery schedules are not yet determined.

On January 13 and 14, R. H. Chapman, G. Hofmann, and T. G. Kollie participated in an informal information meeting, held at ANL, on RSR cladding research problems. Preliminary results, obtained from single rod burst tests, were presented.

On January 23, W. E. Baucum, R. H. Chapman, J. L. Crowley, G. G. Fee, and R. E. MacPherson participated in the MRBT Midyear Budget Review at Germantown. A detailed program status report for internal use was issued prior to the review. The report reviews technical difficulties encountered with procurement of heaters and thermocouples and discusses the impact on costs and schedules.

B. LeGrand, a French engineer on assignment to INEL, visited ORNL on January 30 to gain familiarity with the MRBT program.

A preliminary report, describing the thermocouple spot-welding apparatus developed in this program, was prepared for internal use. A formal report on development of the apparatus is scheduled for publication later this year.

All 14 of the SEMCO heaters accepted for use in our test program have been grooved and coated with a thin protective layer of plasma-sprayed ZrO_2. High-temperature infrared scans are being made of the coated heaters to obtain a qualitative measure of the temperature distribution. Effort was continued on scanning heaters for circumferential temperature variations.

Sixteen tantalum sheathed type K and one tantalum sheathed type S thermocouples were received from SEMCO; one of the former and 5 of the latter remain to be delivered on the outstanding purchase order for which delivery was originally scheduled for March 20, 1975. An order with SEMCO for 50 additional thermocouples of the first category was canceled for the mutual benefit of ORNL and the supplier.

Fabrication of bundle parts and construction activities related to the multirod test facility were halted in mid-January due to budget constraints. All materials were tagged and stored until these activities resume in FY 1977.

Simulator PS-19 was fabricated (with SEMCO heater No. 2828005) and tested in a steam environment. The simulator was pressurized with helium to approximately 2590 kPa (376 psi) at 358°C (676°F) and isolated from the supply system. During the transient the pressure increased slowly to about 2820 kPa (409 psi) then decreased rapidly during deformation; rupture occurred at a pressure of approximately 2590 kPa (376 psi) and at a temperature of about 952°C (1745°F).

Simulator PS-18 was fabricated and tested at high temperature in a steam environment. The heater (SEMCO No. 2828007) was grooved for thermocouples and had a plasma-spray coating of ZrO_2 applied to prevent formation of zirconium-based eutectics with the stainless steel heater sheath. Two tantalum sheathed thermocouples were spot-welded to the inside surface of the Zircaloy tube; tantalum wires occupied the other two thermocouple grooves. The simulator was pressurized with helium to 800 kPa (116 psi) at 350°C (662°F) and isolated from the supply system. The pressure increased very slowly during the transient to a maximum of about 860 kPa (125 psi) and then decreased very slowly during deformation until rupture, which occurred at a pressure of 770 kPa (112 psi) and a temperature of 1171°C (2140°F). This was the highest temperature tested to date. The time to rupture (from power on to power off) was approximately 42 sec; this is about two times greater than the previous series of tests at lower temperatures.

Tests on simulators PS-12, PS-17, PS-18, and PS-19 (involving three different heaters) essentially cover the range of pressure included in the test matrix. Table 1 summarizes pertinent results obtained from these tests. Large, generalized ballooning was not observed on any of the simulators tested. This, we believe, results from the lack of temperature uniformity in the simulators and the deformation sensitivity to small temperature variations exhibited by Zircaloy. Since the simulator gas volume and the local temperature variations are not too unlike a nuclear fuel rod, the observed deformation behavior should be representative of that resulting from a fuel rod undergoing a similar heating transient.

Table 1. Preliminary results of single rod burst tests in a steam environment

	PS-18	PS-19	PS-12	PS-17
SEMCO heater number	2828007	2828005	2828006	2828005
Simulator gas volume, cm^3 ($in.^3$)	45.0 (2.75)	37.4 (2.28)	41.0 (2.50)	37.5 (2.28)
Initial temperature, °C (°F)	353 (667)	358 (676)	340 (644)	340 (644)
Initial pressure, kPa (psi)	800 (116)	2590 (376)	6520 (945)	13 270 (1925)
Maximum pressure, kPa (psi)	860 (125)	2820 (409)	6900 (1001)	13 880 (2035)
Burst pressure, kPa (psi)	770 (112)	2590 (376)	6140 (890)	12 130 (1760)
Burst temperature, °C (°F)	1171 (2140)	952 (1746)	898 (1648)	778 (1420)
Rupture strain,[a] %	24	28	18	25
Time to rupture, sec	42.2	27.4	21.8	16.1

[a]Based on the perimeter of the tube outside surface from lip to lip of the failed zone.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: William B. Cottrell

ACTIVITY NUMBER: 40 89 12 60 1 (189aNo. B0126)

TECHNICAL HIGHLIGHTS:

During the month of January the staff of the Nuclear Safety Information Center (a) processed 963 documents, (b) responded to 68 inquiries (of which 54 involved the technical staff), and (c) made 33 computer searches (of which 4 involved payment). Design Data Sheets were prepared on the Fort St. Vrain Nuclear Generating Station (an HTGR) and on the Fort Calhoun Station, Unit 2. In addition, an indexed bibliography of the 204 ACRS reports received in December 1975 was prepared and issued, as was also an accumulative bibliography covering the months of October through December 1975.

Two NSIC reports, ORNL-NSIC-120 "Annotated Bibliography of Hydrogen Considerations in Light-Water-Power Reactors" and ORNL-NSIC-121 "Reactor Operating Experiences 1972—1974," are in reproduction and should be distributed early in February. Two other reports, ORNL-NSIC-123 "Nuclear Power: Accident Probabilities, Risks and Benefits; A Bibliography" and ORNL-NSIC-124 "Index to *Nuclear Safety*" (Volume 11—Volume 16), are also in reproduction and should be distributed in February. Another report, ORNL-NSIC-118 "Siting of Nuclear Facilities, Sections from *Nuclear Safety*," is in composition. Work is underway on several other reports, including "A Bibliography of LMFBR Safety" which because of its size will be prepared in two volumes.

During most of the past four months, J. R. Buchanan, Assistant Director of NSIC, has been on loan to the National Academy of Sciences Study Group on "Study on Risk Associated with Nuclear Power." The NSIC computer file and document collection have been of valuable assistance in the study. Although the NAS study covers the entire fuel cycle, Buchanan's role was concerned with the assessment of radioactive releases from nuclear power plants under both normal operating and accident conditions. The study is now being reviewed and should be released this spring.

R. A. Hartfield and R. G. Maranaka of NRC visited the Center to explore the possibility of our assisting NRC in two tasks; (1) the preparation of the 1974 and 1975 report on Nuclear Power Plant Operating Experience, and (2) the reclassification of pre-1975 reportable occurrences (back to 1969) according to 1975 standards. Personnel to undertake the first task have been earmarked but the latter is still being assessed.

During January we were also visited by P. Lochak and R. Goldsmith of the Society Internationale de Technologie (SIT). They have been marketing products (i.e., searches) from NSIC takes in Europe and were interested in contracting NSIC to undertake additional studies for them. The work being discussed, i.e., increased effort on nuclear safeguards and the nuclear fuel cycle, would enhance the NSIC file for general usage.

As of the end of September, notices were sent to all former SDI (Selective Dissemination of Information) recipients advising them of the availability of SDI under the recently implemented cost recovery policy. As expected, the immediate response was greater from those who are entitled to this service for free (i.e., primarily NRC and ERDA contractors) than from others who have to pay. During the month of January, 3 of the former and 14 of the latter (totaling $4136) were added to the SDI distribution list and requests are still coming in steadily. These 17 latest users increase our total number of SDI recipients to 362, including 52 subscribers who have paid a total of $11,973.

During the month we have received 3 Japanese nuclear safety reports and 1 German report to review in order to assess the merits of an English translation.

Nuclear Safety 17(1) is at the printers with distribution expected momentarily. All material for *Nuclear Safety* 17(2) have been submitted to TIC (except the Summary of Operating Reactors for which we are awaiting the December "Operating Units Status Report" from NRC). All technical articles for *Nuclear Safety* 17(3) have been completed and submitted for NRC and ERDA review. Manuscripts for most of the articles for *Nuclear Safety* 17(4) are now in hand.

PROGRAM TITLE: PWR Blowdown Heat Transfer-Separate Effects

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: 40 89 12 50 1 (189a No. B0125)

TECHNICAL HIGHLIGHTS:

Task 1. The second production heater rod (from the lot used to assemble THTF bundle 1) has undergone 10 on-off cycles and 3 blowdowns in the FCTF with 18 total hours of powered operation. As with the first production heater rod tested in the FCTF, this heater rod also gave results which showed that, at the same power, pressure, fluid temperature and flow rate, the sheath thermocouple readings increased with each blowdown. These results, together with special calibration tests, are being used in the verification of the ORTCAL calibration code which will be used in the reduction of THTF heater rod results.

The FCTF modifications for blowdowns from 2250 psia are nearly complete. The loop was hydrotested at 4300 psia at room temperature. New heavy-duty rupture disks were installed in the buffer zone. The high pressure dP cells for measuring Venturi flow and pressurizer liquid level are currently being calibrated; other tests are being conducted to determine the temperature dependence of drag disk output.

Task 2. Commitment Nos. 5 and 6 of the NRC Dry Run were fulfilled. The heater rod thermocouple calibration program (ORTCAL) has been used to process the data from the FCTF tests of the second production heater. The output from ORTCAL will be used to develop a dynamic gap model for the THTF heater rods.

A new version (5) of RELAP4 has been installed at ORNL. Preliminary verification of the installation has been completed. Continuing improvement in the RELAP4 model of the THTF loop has been sought in the area of pump-pressurizer interaction and break flow. Key parameters have been identified. A special FOCAL program was written for the data system to obtain data during cold and hot calibration runs. The FORTRAN plot code was modified to print units on scales along the vertical axis. The code is being modified to also print SI units next to the British units.

Task 3. Reconditioning of the insulation on two of the four generators providing power for the THTF has proceeded on schedule. However, excessive moisture problems with the other two generators has required special efforts to dry them out. This may delay completion of the reconditioning of the generators until the first week in February. Meanwhile cold and hot calibration runs are being made with the THTF to improve the error bands on the spool piece flow measurement instruments. In addition, checkout and fine tuning of all THTF instruments and control systems is continuing to insure that all systems will be operational when generator reconditioning is completed.

Task 4. The THTF gamma densitometers utilize photomultiplier tubes. The photomultipliers have the short response times necessary to monitor transient, two-phase densities, but they are sensitive to magnetic fields and temperature change and exhibit significant signal drift. A radiation detection assembly utilizing an ionization chamber was developed for testing in the air-water two-phase flow facility. The ionization chamber has no known sensitivity to magnetic fields and signal stability is significantly better than that of the photomultiplier tubes. In preliminary transient tests, the response time of the ionization chamber assembly was at least as small as the specifications for the THTF densitometers (<16 msec). Tests in the air-water facility showed similar responses of the photomultiplier tube and ionization chamber to steady-state two-phase flows over a variety of flow regimes and the ionization chamber showed essentially no signal drift. Instrument design is proceeding to utilize ionization chambers in all THTF gamma densitometers.

Task 5. As was reported previously, Watlow Electric Manufacturing Co. is attempting to eliminate the sheath gap problem in the THTF heaters. They have observed some shifts in ΔT (core to sheath T/C output) and found that stress relief heat treatments do not solve the problem.

They have recently fabricated a test heater with a copper instead of stainless steel inner sheath and have observed no ΔT shifts during tests. ORNL is presently evaluating the use of a copper inner sheath and the possible procurement of a prototype heater with this modification for testing.

All the necessary materials have been supplied to Watlow for the two pre-production heaters of bundle 2 and they expect to begin fabrication ∿2/9/76. The first heater will be used to determine the active component assembly resistance values for each heat flux zone so that the length of the zones may be properly calculated to give the desired heat flux profile.

At present, 290 single-diameter [0.0508-cm (0.020-in.)] thermocouples have been received from the Claude S. Gordon Co. of which 57 have been supplied to Watlow for the bundle 2 pre-production heaters. Claude S. Gordon Co. is scheduled to ship the balance of their order on 2/9/76.

Kaman Sciences has supplied 97 dual-diameter thermocouples and plans to supply the balance of their order as single-diameter [0.0508-cm (0.020-in.)]. They are scheduled to ship these by 3/31/76.

PROGRAM TITLE: Zircaloy Fuel Cladding Collapse Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: 40 89 12 40 1 (189a Number B0124)

TECHNICAL HIGHLIGHTS:

No technical highlights to report.

PROGRAM TITLE: Zirconium Metal—Water Oxidation Kinetics

PROGRAM MANAGER: C. J. McHargue

ACTIVITY NUMBER: 40 89 12 80 1(189a Number B0128)

TECHNICAL HIGHLIGHTS:

No technical highlights to report.

PROGRAM TITLE: Aerosol Release and Transport from LMFBR Fuel

PROGRAM MANAGER: M. H. Fontana

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

TECHNICAL HIGHLIGHTS:

CDV Development:

Six successful capacitor discharge vaporization (CDV) tests were completed in January using an ORNL designed and assembled CDV firing system to control all sequencing including preheater startup, camera start, and ignitron firing. The new firing and control system apparently eliminated recurring malfunctions in the previously used firing control equipment which had caused preheater damage and loss of data. The preliminary determination of yields indicates that these tests ranged from 0.4 to 2.4 g/m^3.

Additional diagnostic data obtained for the first time in these tests included reliable transducer shock pressures of 5.9 psi. An additional test device, a peizoelectric accelerometer, was attached to the retractable tungsten electrode in one test. The intent of this device was to provide data that could be interpreted in terms of the rate of pressure rise in the assembly and the peak pressure attained. Preliminary examination of the trace showed a ramp rate faster than expected and suggests that the peak pressure may be obtained in this manner.

CRI-III Facility:

In the construction effort on the Aerosol Pressure Vessel (CRI-III), all materials have been received and accepted by the Quality Assurance inspection program, and much of the welding and finishing has been completed. The final inspection and pressure testing before delivery to the CDV site will occur about February 15 and will result in the attachment of the ASME Boiler Code Approval name plate.

Electrical components purchased for the capacitor bank are beginning to arrive, and no holdup of the test program by the extended time for fabrication of the CRI-III is foreseen.

NSPP:

The NSPP staff was moved into renovated office space at the NSPP site (Building 7500) to closely supervise the reactivation activities now under way.

The existing instrumentation and control panels were removed and placed in the ORNL instrument shops where they will be checked, overhauled, and reorganized to conform to current requirements.

Removal of unnecessary equipment from the vessel cell was started.

Bubble transport:

The low pressure water test vessel was used to create Freon-22 bubbles loaded with various size solid particulates including lead shot, sand, and powdered talc. High speed motion pictures were obtained to be developed and examined later.

Design of the under sodium facility and components continues. The Quality Assurance Program Plan has been prepared and issued. Drawings of the Na sump tanks have been approved and issued. The designs of the sampling devices and fuel pellet vaporizer assembly are being reviewed.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: J. P. Sanders

ACTIVITY NUMBER: 40 89 12 20 1 (189a Number B0122)

TECHNICAL HIGHLIGHTS:

General: The BLAST, ORTAP, And ORECA codes were developed further, and a variety of shutdown transients were analyzed. Work on the Fort St. Vrain (FSV) tests and analyses was continued under the University of Tennessee subcontract.

Reheater and Steam Generator Model Development: The FSV reheater-steam generator module has been modeled with the BLAST code, and steady-state conditions are being determined. This is being done in preparation for the comparison of BLAST results with dynamic test data from FSV.

Development of the System NSS Simulation Code (ORTAP): The helium circulator simulation SSTHC,* obtained from Brookhaven National Laboratory, is being coupled with the other component models in ORTAP. Coupling between the helium side of the steam generator and the circulator has been completed. Inclusion of the circulator simulation will complete the initial version of ORTAP.

Core Simulation for Emergency Cooling Analysis: A series of LOMLC and DBDA runs for investigating steam-graphite corrosion in the 3000-MW(t) GASSAR plant was completed using the ORECA code. An automatic plotting routine was written which produces both time-varying axial temperature profiles and other parameters chosen vs time. Studies of FSV LOMLC accidents from 40% power (during the initial startup sequence) were run with ORECA to assist with plant licensing questions.

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*P. L. Versteegen and D. A. Sargis, "SSTHC: A Program to Evaluate the Series Steam Turbine Helium Circulator Performance Under Dynamic Conditions," Science Applications, La Jolla, California (Apr. 18, 1975).

PROGRAM TITLE: Design Criteria for Piping and Nozzles

PROGRAM MANAGER: S. E. Moore

ACTIVITY NUMBER: 40 89 12 30 1 (189a No. B0123)

TECHNICAL HIGHLIGHTS:

Messrs. S. E. Moore, G. D. Whitman, and G. G. Fee of ORNL met with Messrs. E. K. Lynn, C. Z. Serpan, Jr., and Dr. L. S. Tong of RSR and invited guests at NRC Headquarters for the mid-year program review. We discussed the program accomplishments for the first half of fiscal year 1976, expected accomplishments for the second half, and projected plans for FY-1977.

The Pressure Vessel Research Committee held its annual out-of-town meeting in San Diego, California, on January 26-28. We attended meetings of the S/C on Reinforced Openings and External Loadings, the S/C on Piping, Pumps, and Valves, the Task Group on Characterization of Plastic Behavior of Structures, and the PVRC Design Division.

Code Rules Development: The ASME Working Group on Piping Design (WGPD) held their regular meeting in New York on January 6, 1976. The proposed design rules and stress indices for socket-welding fittings that we had prepared earlier were returned for further work as expected. We were asked to revise the proposal to satisfy the objections of the ASME Committee and to resubmit it to WCPD at the next meeting in March.

A number of Code revisions, originally prepared under this program, were published by the ASME in the Winter 1975 Addenda, ASME Boiler and Pressure Vessel Code, Section III — Division 1, Subsection NB. Included in the Addenda are a new subparagraph NB-3685.4 on the classification of stresses for elbows, a revision to NB-3682 on the definitions of flexibility factors, a revision to NB-3687.4 on the flexibility of ANSI B16.9 tees or branch connections, and new stress indices for ANSI B16.9 butt-welding reducers. These revisions became effective Jan. 1, 1976.

Topical Reports: One report in this category was issued: ORNL-5035, *FLANGE: A Computer Program for the Analysis of Flanged Joints with Ring-Type Gaskets*. This report documents the computer program and includes user instructions and a detailed discussion of the theoretical basis of the program. The report also includes a complete analysis and discussion of several sample problems. The program will be made available to the technical community through the Argonne Code Center.

We completed preparation of a draft report on the evaluation of ASME Part A (ANSI B16.5) flanges and bolting for piping joints. This report recognizes the most recent (pending) revisions to the ASME Boiler and Pressure Vessel Code, and also presents a number of additional recommendations.

We also started preparation of a summary report on the experimental stress analysis of a hemispherical shell-radial nozzle model tested earlier at the University of Tennessee under subcontract.

Cylindrical Shell Studies: Two projects in this category are currently underway: (1) the development of a finite element computer program for the stress analysis of closely spaced nozzles in cylindrical pressure vessels and (2) a finite element parameter study of the stresses at isolated nozzles in cylindrical pressure vessels. Excellent progress is being made on both studies.

ANSI B16.9 Tee Studies: Work is continuing on the preparation of a summary report on the elastic response tests of five 24-in. ips ANSI B16.9 tees tested earlier at Combustion Engineering, Inc. We are also preparing a computer based experimental data set for an empirical study on combined loads which we plan to begin in April.

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