MASTER

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

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

Contract No. W-7405-eng-26

AUGUST 1977

MONTHLY HIGHLIGHTS

FOR

OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS

AT

OAK RIDGE NATIONAL LABORATORY

Compiled by
Gordon G. Fee

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Manuscript Completed: September 13, 1977

Date Published - September 1977

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

Prepared by the
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UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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ABSTRACT

Highlights of technical progress during August 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 - Gordon Smith visited Battelle Columbus Laboratory on August 3, and Materials Research Laboratory on August 10 to discuss concepts for demonstrating crack arrest in thick-walled vessels.

Professor C. W. Smith of Virginia Polytechnic Institute and State University visited ORNL on August 9 to discuss plans for the next phase of their subcontract activities and to prepare a research information letter on nozzle corner cracks.

On August 10 and 11, Dr. B.J.L. Darlaston of the UK CEGB visited ORNL to discuss fracture mechanics testing and fracture safety criteria for nuclear pressure vessels.

R. H. Bryan, R. D. Cheverton, J. G. Merkle, and G. D. Whitman participated in the 4th SMIRT Conference in San Francisco, CA, during the period August 15-19.

On August 22, G. D. Whitman attended the Pressure Vessel Study Group meeting at EPRI in Palo Alto, CA.

On August 22, Dr. J. G. Blaue of the Institute for Solid Mechanics in Freiburg, FRG, visited ORNL to discuss the evolution of burst containment requirements for specific reactor sites in Germany and fracture mechanics topics.

On August 24, Mrs. Anderson, Ottosen and Krenk of the Danish Research Establishment at Riso visited ORNL to discuss research concerning the stress analysis and safety of concrete and steel pressure vessels for primary containment.

Dr. G. Markoczy, Manager of the Thermohydraulics Division of the Swiss Federal Institute for Reactor Research, visited ORNL on August 24 to discuss exchange of information under the HSST program.

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Task 4: Irradiation Effects - Lateral expansion and fracture appearance measurements for the unirradiated Charpy V-notch impact tests for the second 4T-CTS irradiation experiment were completed. Lateral expansions on the ductile shelf were 55 to 56 mils for all three welds.

Modifications to the capsule stand, required for relocation of the facility, were completed and the capsule stand reassembled and aligned in the BSR pool. The dosimetry experiment using the dummy capsules was conducted with the reloaded BSR core. Counting of dosimeters is in progress. Some modifications to the experiment control system are in progress and should be completed on schedule.

Assembly of two capsules for the third 4T-CTS irradiation experiment was completed and the third capsule is 95% complete. A safety review of this experiment is scheduled for September 1, 1977.

Task 5: Simulated Service Tests - Additional post-test ultrasonic examinations have been performed on ITV-7B by ORNL and Southwest Research to further characterize the extent of cracking at the flaw prior to cutting this region from the vessel.

Materials investigation and residual stress studies are continuing to evaluate the feasibility of performing a transition temperature test on ITV-8. A total of approximately 30 residual stress measurements have been made on ITV-8 and the prolongation which contains a similar weld repair region. Unexpected low surface values of residual stress were measured in the vessel near the repair weld. It appears the machining operations performed during the vessel fabrication may have resulted in the development of residual compressive stress on the surface. Preparations are being made to remove material from the vessel surface and verify this hypothesis.

We are continuing to examine the fabrication weld in the V-8 prolongation along a radial line approximating 9.5 mm (3/8 in.) from (and parallel to) the ITV-8 prototype repair weld. Static fracture toughness (K_{Icd}) values from precracked Charpy specimens at -32 and -18°C (-25 and 0°F) ranged from 69 to 236 MPa $\sqrt{\text{m}}$ (63 to 215 ksi $\sqrt{\text{in.}}$). The K_{Icd} values at -18°C were all greater than 203 MPa $\sqrt{\text{m}}$ (185 ksi $\sqrt{\text{in.}}$). There was one low value, 69 MPa $\sqrt{\text{m}}$ (63 ksi $\sqrt{\text{in.}}$), at -32°C . The remainder are all greater than 118 MPa $\sqrt{\text{m}}$ (108 ksi $\sqrt{\text{in.}}$). The transition temperature for the fabrication weld is below -18°C and possibly below -32°C (-25°F).

The hardness traverses across three regions of the ITV-8 prototype repair weld in the ITV-9 prolongation have been completed. The highest value (310 DPH) was obtained in the base metal-repair weld metal heat-affected zones (HAZ). A value of 287 DPH was obtained in the HAZ between the base metal and the vessel fabrication weld. Hardness values across the repair weld-vessel fabrication weld interface did not differ significantly from values obtained from either weld metal and ranged from 186 to 239 DPH. The hardness variation in the weld metal appears to depend upon the variation across individual weld passes.

The refrigeration system at the K-25 test facility has been operated to establish the minimum temperature that could be achieved. It appears the cold loop can be brought down to approximately -30°C .

The DATA TRAK was used to simulate the microstructure in the 3.8-cm-thick (1 1/2-in.) quenched only material used for the crack initiation portions of the crack arrest models. A cooling rate of 9.4K (17R) per second was used. Charpy V-notch (C_v) and precracked C_v specimens were made from the DATA TRAK blanks and tested. The properties obtained from the DATA TRAK specimens are representative of the data obtained with the 3.8 cm quenched only material. This study suggests that the toughness results previously reported for specimens simulating quenched only thick walled cylinders are representative.

Preliminary tests were performed on small three point bend specimens containing electron beam welds. The purpose of these tests was to determine if fast pop-in fracture could be achieved. Analyses were also performed for pop-in events in crack arrest models and ITVs. The results to date suggest that this type of pop-in experiment might be suitable for validating the crack-arrest technology.

The nozzle was burned from the previously tested ITV-9 in order to obtain additional material to further characterize intermediate vessel nozzle materials for planning the ITV-10 test. In addition, the lower circumferential weld was taken from ITV-9 for future destructive examination to confirm and characterize defects detected by acoustic emission during the test and post-test ultrasonic examination.

Task 6: Thermal Shock — Preparations were continued for the cryogenic-quench thermal hydraulic experiment with the 21-OD test specimen (TSV-F).

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 August was preparation for fission-chamber data-taking for thermal-neutron fission of ^{241}Pu . This required considerably stricter precautions for safety reasons than we have encountered before for ^{235}U and ^{239}Pu . A set of procedures was approved, and the ^{241}Pu fission chambers were fabricated. Two chambers have been irradiated and were taken to our low-background counting room for Ge(Li) detector calibration.

PROGRAM TITLE: Fission Product Release from LWR Fuel

PROGRAM MANAGER: A. P. Malinauskas

ACTIVITY NUMBER: ORNL #40 89 55 10 9 (189 #B0127)/NRC #60 19 01 03

TECHNICAL HIGHLIGHTS

Experiments HBU-7 and HBU-8 of the High Burnup Test Series were performed. In these tests the H. B. Robinson fuel rod segments were induction heated and ruptured by internal helium pressure. Both experiments were conducted in a flowing steam atmosphere at 900°C. HBU-7 was held at temperature for only 1 min following rupture, while HBU-8 was maintained at 900°C for 1 hr.

Preliminary analytical results for HBU-7 indicate that 0.024% of the cesium inventory was released in vapor form. Additional cesium was released as part of what appears to be fuel particle aerosol. Based on ^{144}Ce analysis (and assuming no dissociation of cerium from the UO_2 fuel particles), approximately 0.017% of the fuel was found as small particles adhering to the apparatus surfaces. Most of the particles, as well as the vapor-deposited cesium, were found in the quartz furnace tube liner. Approximately 0.97% of the original ^{85}Kr inventory was released.

Individual components of the HBU-8 experiment must be examined before an assessment of the effect of additional time at temperature can be made. At this time it appears that the amount of material released with rupture is dominant; the amount of cesium released in 2 hr by diffusion from a predrilled hole in test HBU-2 was only 0.00061%, a factor of 40 less than released in vapor form in test HBU-7. The importance of burst release was also noted in the Implant Test Series.

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 draft design report of the Fission Product Transport Test Facility (FPTTF) has been completed and is being reviewed internally. The draft report should be transmitted to NRC by the end of August. Although a model scale of 1/200 based on flow area was used in the conceptual design work and cost estimates, the relationship of cost to model scale is being investigated.

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

The first 4 x 4 bundle (which was burst last month) has been tested for flow pressure drop at various Reynolds numbers in a water flow loop. The data indicates a distinct pressure change associated with the three burst zones described in the last report. Preliminary raw data indicates that at a Reynolds' number of about 1.2×10^5 in the undeformed section of the Δ ps associated with the 75 cm, the 46 cm, and the 25 cm burst zones are about 1.7 psi, 3.2 psi, and 1.7 psi, respectively. The Δ ps associated with the two center grid spacers at this Reynolds number are about 2.6 psi and 3.0 psi. Bundle 1 will be next cast in epoxy for sectioning.

The assembly of fuel pin simulators for 4 x 4 bundle 2 has reached the halfway point. Bundle 2 is being assembled on an accelerated schedule and will be tested at the same conditions as bundle 1 except with an unheated shroud.

In-house preparation, consisting of grooving, plasma spray coating, and infrared scanning, has begun on the G.E. fuel simulators for the third 4 x 4 bundle. The remaining units of 25 ordered from G.E.-Seattle are expected to be shipped by September 2, 1977. The remainder of the SEMCO order (4 units) was received and accepted.

Three special single-rod burst tests have been completed which are aimed at determination of short-term (50-100 sec) creep-rupture deformation under essentially steady-state temperature. The first two bursts were used sacrificially in the development of the control system necessary to provide the desired test conditions. The system worked very well for the third test (8/29/77) which was ramped at about $14^\circ\text{C}/\text{sec}$ to 760°C . The temperature was controlled very close at 760°C for about 100 sec until burst which occurred at about 5700 kPa. The data are not

processed nor the tube examined as yet. However, preliminary observations indicate the burst area to be in a well instrumented region and there is no unusually large or "sausage" deformation in the remainder of the tube.

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 August the staff of the Nuclear Safety Information Center (a) processed 930 documents, (b) responded to 110 inquiries (of which 52 involved the technical staff), and (c) made 23 computer searches (of which three involved payment totaling \$164.40). A bibliography of 152 documents, received from the ACRS in July, was distributed. No additional documents were received in August. NSIC staff received twelve visitors during June and attended five meetings, two of which *Nuclear Safety* and NSIC literature was distributed.

During the month of August ERDA reported on the usage of the 24 data bases on their RECON System for the twelve-month period ending August 1, 1977. The data from their summary is presented below as Table 1. It is noted that NSIC is the fifth most widely used data base behind only Nuclear Science Abstracts (NSA), Energy Data Base (EDB), Water Resources Abstracts (EPA-WRSIC), and Engineering Index (JEC). Furthermore, our own in-house use of the NSIC data base constitutes less than 20% of the total.

Two NSIC reports were in reproduction at the end of the month; "Annotated Bibliography of Safety-Related Occurrences in Boiling-Water Nuclear Power Plants as Reported in 1976" (ORNL/NUREG/NSIC-137), and "Annotated Bibliography of Safety-Related Occurrences in Pressurized-Water Nuclear Power Plants as Reported in 1976" (ORNL/NUREG/NSIC-138). In addition, drafts have been completed of several other reports including: "Summary Data for All Commercial Nuclear Power Plants" (ORNL/NUREG/NSIC-141); "Reports Distributed Under the NRC Light-Water Reactor Safety Technical Exchange, Vol. III, January-June 1977" (ORNL/NUREG/NSIC-142); and "Bibliography of Reports on Research Sponsored by the NRC Office of Nuclear Regulatory Research, January-June 1977" (ORNL/NUREG/NSIC-143).

Table 1

DATA BASE IDENT.	DATA BASE NAME AND SUPPORTING INSTALLATION IDENT.	NO. OF SESSIONS	NO. OF EXPANDS	NO. OF CITATIONS PRINTED
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	12850	33469	585858
ENG	(EISO) ENERGY DATABASE	262	494	3599
CAS	(CAS) CHEMICAL ABSTRACTS/ENERGY	1029	2497	3235
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	3605	19662	313607
ED2	(TIC-CSD) ENERGY DB PHASE II	145	168	876
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	278	339	3843
HEF	(EISO) HEATED EFFLUENT BIBLIOGRAPHY	93	203	1572
EMI	(EMIC) ENV. MUTAGENS INFO.	712	1034	29245
EDB	(TIC/CSD) FRCA ENERGY DATABASE	16016	39902	617254
ERD	(EISO) ENERGY R&D PROJECTS	920	2412	14204
NBI	(NBIC) NATL BIOMONITORING INV.	136	271	802
DBS	(LLL) DATA BASE SURVEY	237	452	1163
ESI	(EIC) ENV. SCIENCE INDEX	662	1898	16240
EIX	(CSD) ENGINEERING INDEX	1946	6425	44229
MOS	(LLL) ENG. & ENV. DR MODELING SURVEY	243	763	1293
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	1606	3778	119560
MFY	(CSD) METALS ABSTRACTS (METADEX)	563	1843	8142
NRC	(LC) NATIONAL REFERRAL CENTER	439	814	2031
NER	(EIC) NATIONAL ENERGY REFERRAL	331	842	871
RSI	(RSIC) RADIATION SHIELDING INFO.	87	151	2035
EIA	(EIC) ENERGY INFO. ABSTRACTS	226	686	2029
RSC	(RSIC) RADIATION SHIELDING CODES	49	69	91
ESR	(ERDA) FED. ENERGY ENV. & SAFETY RES.	53	104	881
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	42	53	244

We received ten foreign safety reports (three from Germany, one from France and six from Japan) under the NRC exchange agreement during August. Our recommendations concerning the translation of these documents were contained in a letter dated August 12, 1977, to G. L. Bennett. Also in August we received copies of 13 reports on NRC-sponsored LWR safety projects which were distributed (cover letter of August 25) 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 four to six weeks. In addition, in August we received the first reports on NRC-sponsored HTGR safety research to be disseminated under the U.S.-Japanese agreement. This totaled 37 U.S. documents which were distributed under cover letters of August 11, 17, 22 and 25, 1977.

NSIC's selective dissemination of information (SDI) is available to paying users (as well as exempt users). During the month of August we added four exempt users and one paying user (bringing the totals to 380 users, including 60 paying users). However, as noted last month, NTIS -- the NRC billing agent for our nonexempt users -- had not sent renewal notices at the end of the year subscription period, despite assuring us to the contrary. We sent out the renewal notices during the month and expect to discontinue service to those who have not responded by early September.

The recently purchased IBM-3270 computer equipment, consisting basically of a controller (which is hard-wired into the Computing Technology Center at K-25), a printer, and our CTR terminals is now "operational" at NSIC offices. However, the system response times are inordinately long and the UCCND Computer Sciences group is now analyzing and revising their TSO (time shared option) teleprocessing program. Some improvements have already been implemented and others are being evaluated.

All technical articles for *Nuclear Safety* 19(1) are in hand and in various stages of processing for their September submittal to NRC, ERDA and TIC. The NRC and ERDA reviews of articles for *Nuclear Safety* 18(6) were completed and are being discussed with the authors.

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. Several quench tests were performed on a 3.66-m (12-ft) heater rod shipped to ORNL by EG&G Idaho, Inc. The purpose of the test was to determine the time constants of the heater thermocouples. This heater had undergone life-cycle and elevated-stress testing at Idaho before shipment to ORNL. According to EG&G the heating filament was partially cracked, and the rod exterior no longer met diameter and straightness tolerances after their tests which subjected the heater to water quenching from temperatures of about 1093 K (2000°F). The experimentally measured time constants of thermocouples in the sheath were on the order of 0.5-1.0 sec. Because of the severity of the operating history of this heater at Idaho, there is some uncertainty about whether these results are representative of heaters in Semiscale bundles.

Task 2. Test 163 was conducted in the Thermal-Hydraulic Test Facility (THTF) on August 23, 1977. Test 163 was a 45-rod (4.500 MW) test initiated from a test section inlet temperature of 557.6 K (544°F), a test section inlet volumetric flow of 0.0320 m³/sec (507 gpm), and a test section outlet pressure of 15.886 MN/m² (2304 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.

Task 3. Operation of all systems within the facility were satisfactory during test 163. The internal resistance to ground of one of the four generators was ~100 K Ω , which is below the minimum allowable to operate. An investigation showed this to be in a series coil on the negative side of the generator and not in the armature. The maximum voltage at any point within the coil is ~15 volts. Since a ground fault at this low voltage should not cause damage to the generators, the facility was allowed to operate. It should be noted that the generators had just been cleaned and none of the four had a resistance above the minimum allowable for operation without special permission.

The pressurizer modification was completed prior to test 163. The performance of the system during test 163 indicates that the new piping arrangement and the immersion type pressurizer heaters have improved the stability and responsiveness of the system pressure control.

Construction (phase 1) of the new control room addition continued this month. This phase is just the shell. It is about 99% complete. Phase 2, which includes interior painting, air conditioning, and electrical outlets and lights, should start in September.

During preoperational runs for test 164, both pump seals failed. The seals were manufactured by Borg-Warner. One seal had survived nine blowdowns and the other six blowdowns. These were replaced with seals manufactured by Durametallic Corp.

Task 4. Problems previously experienced with the three-beam gamma densitometers have been corrected. The two defective detectors were fixed and returned to us by Measurements, Inc. They were installed and found to function properly. In addition, two converted photomultiplier tubes salvaged from surplus gamma densitometers are not available as spacers. To date, the following horizontal spool piece tests have been run:

- (1) No screens or ramps
- (2) One 4-20 screen only
- (3) One 1/2-in. ramp and one 4-20 screen, 1 in. apart

The review meeting for the liquid level probe to be supplied by INEL was held on August 15. A brief summary of the design and a description of the electronic inventory was presented. Discussions were held on the actual probe construction to assure compatibility with bundle 2.

Material problems encountered by Auburn, Inc., make it highly unlikely that they will be able to deliver the four liquid level rods for bundle 2 as promised. A review of the status of this contract is to be made shortly.

Task 5. All bundle 2 fuel pin simulators have been received from Watlow and are undergoing testing and evaluation. All tests made thus far have shown the simulators to be acceptable except for the "hot" insulation resistance of some centerline thermocouples. It appears

that six simulators have sufficient numbers of these thermocouples to be considered rejectable if the thermocouples are rejectable.

Task 6. The General Electric Company is still scheduled to ship bundle 3 prototypes by September 16, 1977, but appear to be falling approximately one week behind schedule. RAMA Corp. is now expecting to begin fabrication of the bundle 3 samples during the week of September 6, 1977.

A purchase order for two bundle 3 prototype fuel pin simulators has finally been placed with Watlow. Watlow is planning to use boron nitride preforms in fabricating these units. Delivery is scheduled for the middle of November 1977.

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

Long-term creepdown testing continues with the eddy-current deformation-monitoring system producing excellent results. We have just completed a 960 hr run at 371°C (700°F) and 15.9 MPa (2300 psig). Radial displacement measurements were taken at 20 surface positions every two hours. The measurements were made with a precision of approximately one μm (0.00004 in.) - that is, changes in displacement of that magnitude could be detected. The accuracy of the readings appears to be excellent also. Three reference liftoff coils showed negligible changes in readout during the test indicating that little drift had occurred from the original absolute calibration.

Fabrication of parts for the in-reactor mockup experiment continues in anticipation of the late fall shipment date.

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

The final report on our Reaction Rate Studies was issued on August 24, 1977, marking the completion of Milestone 48180.

PROGRAM TITLE: Aerosol Release and Transport from LMFBR Fuel

PROGRAM MANAGER: T. S. Kress

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

TECHNICAL HIGHLIGHTS

CRI-III/CDV:

A summary on the status of CDV-CRI-III energy density experiments has been completed. These data will be interpreted and discussed in the "Upper-Limit" source term report now in preparation. In addition, a development report on the capacitor discharge vaporization system is scheduled for draft completion September 30, 1977.

CRI-II:

The first CRI-II fuel aerosol behavior experiment using the tungsten lined hearth was conducted, leading to a maximum aerosol concentration of about 10 grams/m³. In this run (AF-9), the photo-diode pyrometer was mounted on the beehive furnace and focused on the surface of the liquid UO₂. The e.m.f. output was observed to oscillate as the dc arc was rotated; however, a relatively constant average reading of 35 mV (between oscillations) indicated that a theoretical temperature of 4500°C was being maintained on the liquid surface which is somewhat higher than the literature value of 4100°C for the boiling temperature of UO₂. One purpose of these experiments is to compare the primary aerosol sizes for a system that has cooling and nucleation conditions that differ from those in the CRI-III/CDV tests.

A second unit of the modified Andersen 8-stage, 100-jet impactor fabricated out of type 303 (austenitic) stainless steel has been installed to be operated in parallel to our commercial aluminum model. This stainless steel unit will be available later for use with sodium oxide aerosols.

NSPP:

Sodium-oxide Test 105, intended to produce an almost instantaneous aerosol at high concentration, was completed during this period. The test was conducted by rapidly spraying 5 kg of heated sodium into the vessel atmosphere. Sodium droplet size (diameter) was estimated, from spray nozzle manufacturer's data, to be around 480 microns. The pressure rise produced by the burning sodium droplets was 4.55×10^4 Pa (6.6 psi) and maximum gas temperatures at various locations within the vessel ranged from 368 to 477 K (95 to 204°C). Analytical determinations of the sodium oxide content of the various samplers have not been completed.

The NSPP facility is now in shutdown condition to allow the installation of the uranium oxide aerosol generating system. It is planned to perform enough baseline U_3O_8 aerosol tests to characterize the generation technique before initiating the mixed sodium oxide-uranium oxide aerosol tests.

Bubble Transport/FAST:

Major emphasis was placed on installation of the electrical equipment and instrumentation for the CRI-III vessel and the capacitor discharge vaporization (CDV) power supply. The CRI-III vessel was installed and connected to the electrical control cabinet, and approximately 33% of the total electrical and control circuits for the CDV were installed.

Fabrication of the FAST pressure vessel was completed and preparations were made to have the Boiler Code Inspector check the vessel during the first week of September. Overall design of the electrical systems for the FAST facility approached 95% completion and the mechanical design of the eight-stage mass sampler was completed.

Analytical:

The ART analytical team is concentrating on the appropriate modeling required for interpretation of the upper-limit source experimental results. This includes use of our thermal models to establish the energy distributions within the CDV samples after capacitor discharge,

interpretation of cooling rates and nucleation mechanics in terms of our measured aerosol size distributions, and effects of heat exchange and potential non-equilibrium conditions on the measured aerosol yields. Such analyses should be useful in interpreting the CRI-III/CDV results in terms of actual reactor conditions.

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:
A paper on rod withdrawal accidents postulated for the Fort St. Vrain Reactor was presented at the ANS Topical Meeting on Thermal Reactor Safety in Sun Valley, Idaho. A model was developed for predicting maximum temperatures of the PCRV thermal barrier cover plates in the core inlet plenum following a loss-of-forced-convection accident.

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: There was no activity under this category during August.

Miscellaneous Studies: The first draft of a report describing the development of modified stress indices and flexibility factors for piping elbows to include the restraining effects of attached pipe was completed this month. In this study a series of finite element analyses were conducted for elbow-pipe models over a range of the dimensional parameters $0.01 \leq \lambda = tR/r^2 \sqrt{1-\nu^2} \leq 1.0$ and $R/r = 2.0, 3.0$ where t , r , and R are the wall thickness, mean radius of the pipe, and bend radius of the elbow respectively, for several values of the bend angle θ . The set of maximum stress indices were then correlated as the basis for modified stress index and flexibility formulas for Code use. We expect to finalize the report next month; publication will be early in October.

Nozzles in Cylindrical Reactor Pressure Vessels: The first draft of a report describing the extension and validation of the computer program MULT-NOZZLE with three closely-spaced nozzles loaded with internal pressure was completed at the subcontractor's offices in Santa Monica, California. The draft report is undergoing internal review at MRI before being submitted to ORNL for our examination. The most significant modification to the program is the inclusion of an option in the mesh generator module FEMG to automatically develop a mesh layout for three nozzles in addition to the one and two nozzle options currently in the program. A user's manual for MULT-NOZZLE is also being prepared which will provide instructions for setting up and running any of the currently available options.

Work was continued on developing post processing procedures for analyzing results from the MULT-NOZZLE computer program. Processors currently being developed are oriented at reorganizing the output data, which is stored on a 9-track tape, for tabular presentations and for input

to various available plotting routines. We are also investigating the feasibility of using a hidden line remover program developed at Livermore for the display of three-dimensional finite element mesh layouts. These capabilities are needed in order to efficiently conduct the parameter studies needed to examine the validity of current design criteria.

PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: R. 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 focused on the generation of cross sections to be entered into the TASK and VART computer programs. We have decided to employ the NRC 218-group microscopic cross section set in this work rather than the Hansen and Roach set specified earlier because the NRC set will give a better treatment of the significant amount of neutron upscattering in the thermal energy region that is characteristic of the well-moderated situations that we have chosen to model. As a result of this change, the sequence of computer codes for generating microscopic cross sections (see *Noise Diagnostics for Safety Assessment Quarterly Progress Report for January-March 1977*, ORNL/NUREG/TM-112, pg. 15) will now be NITAWL, XSDRNPM, AXMIX; this sequence of operations will collapse the detailed cross sections into a four-energy-group structure.

Subroutines for entering cross sections and TASK-generated fluxes into VART are now being written.

Loose-Parts Monitoring Systems. Procurement of necessary apparatus and instrumentation is proceeding somewhat more slowly than anticipated. Only a portion of the accelerometer/preamplifier order that was expected in early August was received, with the remaining items backordered. Also, no commercial source of supply could be found for a standardized impacting apparatus meeting our requirements, so this will have to be designed and built in-house. An initial impact test fixture (surface plate) was obtained however, and necessary modifications are in progress. Likewise, a suitable commercial four-channel transient signal capture instrument was identified, and a purchase request for this major capital item is now being processed. A more detailed picture of equipment needs and the status of our acquisition activities will be presented in the quarterly progress report for July-September.

Surveillance and Monitoring by Noise Analysis. New subtasks, dealing with pressure boundary crack and leak detection in LWRs and stability monitoring in BWRs by means of noise techniques, are being considered for inclusion in our FY 1978 research proposal.