ORNL/NUREG/TM-189

January 1978
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

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Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy $4.50; Microfiche $3.00

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ORNL/NUREG/TM-189

Contract No. W-7405-eng-26

JANUARY 1978

MONTHLY HIGHLIGHTS

FOR

OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS

AT

OAK RIDGE NATIONAL LABORATORY

Compiled by

Fred R. Mynatt

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Manuscript Completed: February 10, 1978

Date Published - February 1978

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

Prepared by the
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Oak Ridge, Tennessee 37830
Operated by
UNION CARBIDE CORPORATION
For the
DEPARTMENT OF ENERGY

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ABSTRACT

Highlights of technical progress during January 1978 are presented for sixteen 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 10 05

TECHNICAL HIGHLIGHTS

Task 1: Program Administration — A monthly program review was held on January 12 and 13 at ORNL for C. Z. Serpan, Jr.

Task 2: Fracture Mechanics and Analysis — Photoelastic studies have been initiated on inside surface nozzle corner flaws in planes out of the longitudinal plane where all previous studies have been performed. Preliminary results in a plane 90° from the longitudinal plane indicate significant reduction in the stress intensity factor as would be expected. Additional tests at the 45° position are under way.

Task 4: Irradiation Effects — Nine additional Charpy V-notch tests of specimens from the second 4T-CT irradiation experiment were conducted, which completed the Charpy V-notch impact tests for the present. The remaining specimens are being retained for future studies. The results were in line with previous tests of this series since upper shelf (ductile) fracture energies were 52 ft-lbs for weld 61W, 62 ft-lbs for weld 62W, and 42 ft-lbs for weld 63W.

Three point bend tests of unirradiated precracked Charpy specimens were started.

The first rotation of the three capsules comprising the third 4T-CT irradiation experiment was conducted during January and irradiation is continuing.

Task 5: Simulated Service Tests — The cutout containing the ITV-7B flaw was returned from the University of Michigan on January 17. Based on the results of nondestructive examinations conducted at Southwest Research Institute and preliminary confirmation of these results at the University of Michigan, sectioning of the cutout was initiated. Cuts were located to establish the placement of the original electron-beam weld relative to the heat-affected zone and the extent of crack tearing. The preliminary findings of this examination indicate that (1) the electron-beam weld and the initial crack were centered in the heat-affected zone,

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(2) the crack extension occurred adjacent to the heat-affected zone in parent metal and weld metal crossing the heat-affected zone at right angles at various locations, (3) tearing initiated at the center of the flaw, and (4) the extent of tearing on the ends was accurately predicted by the ultrasonic examination. Additional examinations are under way.

An analysis before test for the ITV-8 was transmitted to NRC-RES for approval. The metallographic examination of the precracked Charpy-V specimens from the ITV-8 prolongation is continuing. We have shown a correlation between the microstructure at the tip of the fatigue crack and the apparent fracture toughness level of the weld metal at −46°C (−50°F). Those specimens whose crack tips terminated in the coarse columnar grain region of a weld pass exhibited low fracture toughness, while those that terminated in the fine-grain region of the weld nugget were quite tough. (The fine-grained region is that portion of a previous weld pass that is reaustenitized as a consequent or a subsequent pass.)

The last of the compact tension specimens that had been previously machined from crack arrest model material were tested. The fracture toughnesses that were measured in these tests fell within the scatter band previously obtained from precracked Charpy specimen slow bend tests.

Task 6: Thermal Shock — Development of the coating application technique for coating the inner surface of TSV-F in preparation for an LN_2 thermal shock was completed. A multiple-pass coating of 3M Brand NF-34 was applied to the inner surface of a three-foot length of 10-in. pipe with good axial uniformity and the desired thickness. Following this successful demonstration, TSV-F (21-in.-OD thermal-hydraulic test specimen) was coated in a similar manner with satisfactory results. On January 30, 1978, TSV-F, at an initial temperature of 21°C (70°F), was subjected to an LN_2 quench in the LN_2 thermal shock test facility.

A research information letter (RIL) pertaining to the four completed thermal shock experiments (TSE-1, -2, -3, -4) was prepared and submitted for internal review.

Work was begun on a "significance" report relating to completed thermal shock studies. In this connection, calculations were completed for a parametric study pertaining to the reference calculational model.

PROGRAM TITLE: Fission Product Beta and Gamma Energy Release

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

ACTIVITY NUMBER: 40 10 01 06 2 (189 #B0095)/NRC #60 19 10 04 1

TECHNICAL HIGHLIGHTS

The major activity during January was the preparation of the first drafts of two ORNL/NUREG reports. The first draft of the major report on our ^{239}Pu decay energy-release data was completed, and it will be reviewed internally during February. We anticipate sending the final version out for publication by the end of February. The other report which is being prepared will give tabular and graphic representations of all of our ^{235}U decay beta- and gamma- ray spectra. This report is the one referred to in Reference 61 of ORNL/NUREG-14, and will have the title, "Delayed Beta- and Gamma-Ray Production Due to Thermal-Neutron Fission of ^{235}U, Spectral Distribution for Times After Fission Between 2 and 14000 Sec: Tabular Data."

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 10 04 1

TECHNICAL HIGHLIGHTS:

Preliminary testing and equipment modifications are in process in preparation for tests to be conducted in the temperature range 1300 to 1700°C with H. B. Robinson fuel.

Tests conducted with unirradiated fuel rod segments which were ruptured at 900°C by internal pressurization indicated that temperatures above 1300°C could not be attained using induction heating. Above approximately 1250°C, a rapid axial splitting of the cladding occurs; this results in the loss of induction coupling. The splitting always propagated from the rupture opening.

Testing with unirradiated fuel rod segments in which 1/16-in.-diam holes were drilled through the Zircaloy cladding at the midpoint of the segments was successful, however. One test was conducted at 1500°C for 10 min in steam without loss of induction coupling. Moreover, a fairly consistent temperature profile was maintained throughout the test period. A second test was performed at 1700°C in steam. In this case the loss of induction coupling which was observed was caused by pellet-clad eutectic melting which resulted shortly after 1700°C was reached.

<u>PROGRAM TITLE</u>: Fission Product Transport Tests

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

<u>ACTIVITY NUMBER</u>: ORNL # 40 89 55 11 6 (189 # B0189)/NRC # 60 19 1C 04 1

<u>TECHNICAL HIGHLIGHTS</u>:

The Fission Product Review Group met in Washington D.C. to discuss the Fission Product Transport Test Facility (FPTTF) conceptual design report. As background, presentations by various contractors covering the current status of related experimental and analytical research were made. New data were presented on the radioactive source terms for various hypothetical accidents which indicated that the potential effects of fission product deposition in the primary system of LWRs during a meltdown accident are very much higher than from a loss-of-coolant accident. Consequently, pending NRC concurrence, a more detailed design study of the FPTTF will be undertaken with the emphasis on altering the facility design to accommodate meltdown conditions.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: ORNL # 40 89 55 10 6 (189 # B0120)/NRC # 60 19 10 04 1

TECHNICAL HIGHLIGHTS:

On January 10, R. H. Chapman and F. R. Mynatt visited RSR/NRC offices in Silver Spring, MD, for technical and programmatic guideline discussions.

On January 16-17, E. D. Hindle of the UKAEA Springfields Fuel Laboratory visited ORNL for discussions on the MRBT program and to review Zircaloy oxidation research conducted by the (now completed) ZWOK program.

On January 18-19, R. H. Chapman and J. L. Crowley participated in the Zircaloy Cladding Review Meeting and Experimenters' Workshop at RSR/NRC headquarters in Silver Spring, MD. An informal presentation of MRBT results obtained during the last quarter was made.

Digitized data have been obtained from the enlarged photographs ($\sim$ 5X) of each of the cross sections (about 55) made on the B-1 bundle. The digitized data have been processed to produce accurate strain values for each tube in each cross section. The data were also used to produce computer-drawn profiles of each cross section and to calculate the percentage reduction in coolant channel flow area along the length of the bundle, based on a rod-centered unit cell. The "flare-out" at the tube bursts was treated two ways to obtain reasonable upper and lower limits of the loss in flow area at those cross sections where bursts occurred. The maximum reduction in flow area occurred at elevation 46.7 cm (two bursts but large deformation in several tubes) and was determined to be 58 or 49 percent, depending on how the "flare-out" was handled. Similarly, the reduction in flow area was 57 or 44 percent at elevation 76.5 cm, i.e., the vicinity of four bursts in the test.

Flow tests on the B-2 bundle were completed and the bundle was cast in epoxy for subsequent sectioning and photographing for strain determinations.

In both the B-1 and B-2 tests several tube bursts occurred much nearer the upper end of the heated zone than normally observed in the single rod tests. While these bursts were in general agreement with the pretest infrared characterization scans of the fuel simulators, there may have been an influence of steam flow rate. [The flow rate (Re = 240) and, hence, the thermal entrance region, in the bundle tests was lower than that (Re = 815) used in the single rod tests.] To investigate this possibility, it was decided to use two of the B-2 fuel simulators in single rod tests at both flow rates. One such test (SR-39) was performed this month; however, results are not yet available.

Fabrication of the fuel pin simulators for the B-3 test, using the B-1 fuel simulators, is on schedule, with approximately half of the simulators assembled.

Data obtained in the B-1 test raised questions about the performance and the thermomentry on the heated shroud. These were resolved by performing a series of shroud heating tests (without the bundle being present) in the bundle test facility. It was determined that three of the six B-1 shroud thermocouples provided accurate indications of the temperature and that local shroud overheating probably did not occur.

It has been decided to develop the necessary technology for producing fuel simulators, using BN preforms instead of BN powder, to assure uniformity of quality and, as a result, delivery of simulators on reliable schedules. A sufficient number of simulators will be produced in-house to demonstrate that the technology does, indeed, accomplish these objectives.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: William B. Cottrell

ACTIVITY NUMBER: ORNL # 40 89 55 10 4 (189 # B0126)/NRC # 60 19 10 01 2

TECHNICAL HIGHLIGHTS

During the month of January, the staff of the Nuclear Safety Information Center (a) processed 972 documents, (b) responded to 85 inquiries (of which 41 involved the technical staff), and (c) made 16 computer searches (of which two involved payment totaling $165.18). The RECON system, which now has over 200 remote terminals, reports that the NSIC data file was accessed 176 times during the previous month (see attached table). The bibliography of 185 documents, received from the ACRS the end of November, is in reproduction and another bibliography of 114 documents received mid-January is being processed. During the past month, the NSIC staff received seven visitors, participated in two meetings and one review.

One NSIC report is in composition, Summary Data for All Commercial Nuclear Power Plants (ORNL/NUREG/NSIC-141). Several other NSIC reports are in various stages of preparation, including HTGR Bibliography (ORNL/NUREG/NSIC-128); Reactor Operating Experiences 1975–1977 (ORNL/NUREG/NSIC-144); Bibliography of Reports on NRC Safety Research (ORNL/NUREG/NSIC-145); Reports Distributed Under the NRC Technical Exchange Program (ORNL/NUREG/NSIC-146); Index to *Nuclear Safety*, Vol. 11–18 (ORNL/NUREG/NSIC-147); Bibliography on Common Cause — Common Mode Failures (ORNL/NUREG/NSIC-148); Annotated Bibliography of Licensee Event Reports in Boiling-Water Nuclear Power Plants as Reported in 1977 (ORNL/NUREG/NSIC-149); and Annotated Bibliography of Licensee Event Reports in Pressurized-Water Nuclear Power Plants as Reported in 1977 (ORNL/NUREG/NSIC-150).

In preparing the material for the two Licensee Event Reports (see preceding paragraph), we became aware that we were missing a copy of many (hundreds) events. A copy of some of the missing events was obtained for us by Steve Scott (NRC-DDC) from their files, OMIPC or OI&E. However, most of the missing reports were not available at Headquarters, so we are contacting the Regional Offices for same.

We received twenty-two foreign safety reports (all from FRG) under the NRC-LWR exchange agreement during January and recommended translation of eleven (letters dated January 23 and 24, 1978, to G. L. Bennett). During January we received 46 reports on NRC-sponsored LWR safety projects for distribution to foreign recipients under the Light-Water Reactor Safety Technical Exchange Agreements. Thirty of these reports were accorded full distribution (letters dated January 18 and 20, 1978). However, we received an insufficient number of copies to make full distribution of the remainder of the reports but, per discussion with G. L. Bennett, will make a partial distribution where we have 50% of the necessary copies. On this basis, we are currently making distribution of eight additional LWR reports, but will continue to hold seven reports (including several dating back to October 1977) of which we have ten copies or less. Only one U.S. report was received in January for distribution under the Fast Reactor Safety Exchange Agreement and three reports under the HTGR Exchange Agreement.

NSIC's selective dissemination of information (SDI) is available to paying users (as well as exempt users). During the month of January, we added one paying user and had two renewals for a total of $1264. As of the end of January, the SDI service had a total of 370 users.

All technical articles for *Nuclear Safety* 19(3) were completed on schedule and were mailed to NRC, DOE, and TIC on January 20. The "current events" material (covering events which occurred during November and December) for *Nuclear Safety* 19(2) was completed by mid January (except for "Operating U.S. Power Reactors" which is dependent upon the 'Grey Book') and sent to TIC. All technical articles for *Nuclear Safety* 19(4) have been received, submitted to peer review, and are in various stages of preparation.

When the East Tennessee Chapter of the Society for Technical Communication announced their annual competition last fall, we entered *Nuclear Safety*, Volume 18, in the technical journal category, several

Nuclear Safety articles in the 'technical article category', and two NSIC reports in the 'compilations and bibliography' category. At the awards presentation (January 27), *Nuclear Safety* received the first place award in its category (for the fourth consecutive year), the article by L. S. Tong and G. L. Bennett entitled "NRC Water-Reactor Safety-Research Program" in *Nuclear Safety* 18(1) received the second place award in its category, and the NSIC report by F. A. Heddleson, "Design Data and Safety Features of Commercial Nuclear Power Plants Including Cumulative Index for Volumes I Through VI" (ORNL/NUREG/NSIC-136) received first place, and other report by R. L. Scott and R. B. Gallaher, "Annotated Bibliography of Safety-Related Occurrences in Boiling-Water Nuclear Power Plants as Reported in 1976" (ORNL/NUREG/NSIC-137) received second place, in their category. The winners in their respective categories are automatically entered into the national competition.

RECON DATA BASE ACTIVITY FROM 12-01-77 TO 01-01-78 (21 OPERATING DAYS)

DATA BASE IDENT.	DATA BASE NAME AND SUPPORTING INSTALLATION IDENTIFICATION	NO. OF SESSIONS	NO. OF EXPANDS	NO. OF CITATIONS PRINTED
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	893	1983	29595
RIP	(ERDA) ENERGY RESEARCH IN PROGRESS	166	298	1622
CAS	(CAS) CHEMICAL ABSTRACTS/ENERGY	91	189	141
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	259	1073	20912
GAP	(ERDA) GENERAL AND PRACTICAL INFO.	161	272	1685
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	24	38	63
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	25	45	194
EMI	(EMIC) ENV. MUTAGENS INFO.	111	188	8038
EDB	(TIC/CSD) ERDA ENERGY DATABASE	2061	4569	204760
ERD	(EISO) ENERGY R&D PROJECTS	76	171	417
NBI	(NBIC) NATL. BIOMONITORING INV.	22	27	127
DBS	(LLL) DATA BASE SURVEY	26	43	288
ESI	(EIC) ENV. SCIENCE INDEX	78	192	307
EIX	(CSD) ENGINEERING INDEX	158	436	2307
MOS	(LLL) ENG. & ENV. DB MODELING SURVEY	24	130	72
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	176	323	19433
NRC	(LC) NATIONAL REFERRAL CENTER	52	135	116
NER	(EIC) NATIONAL ENERGY REFERRAL	30	72	667
EXX	ADSEP TEST FILE	22	105	345
RSI	(RSIC) RADIATION SHIELDING INFO.	14	29	211
EIA	(EIC) ENERGY INFO. ABSTRACTS	47	73	30
RSC	(RSIC) RADIATION SHIELDING CODES	7	8	0
ESR	(ERDA) FED. ENERGY ENV. & SAFETY RES.	30	49	6
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	38	24	429

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 10 01 2

TECHNICAL HIGHLIGHTS:

Task 1. FCTF Operations. The forced convection test facility was used primarily as a test bed for determining the performance characteristics of the Auburn International liquid-level probe. (See Task 4.)

Task 2. Analysis. Electric Pin Simulation. The explicit formulation of the inverse parabolic conduction problem was found to be unstable for all time step ranges.

A code which solves a one-dimensional, transient, lumped parameter, implicit formulation of the hyperbolic conduction equation has been generated and debugged. The speed of sound in air was assumed for the wave propagation speed in all regions — there was no significant difference between the parabolic and hyperbolic inverse solutions (that is, with the above assumption).

Development of codes utilizing J. V. Beck's first and second methods for the solution of the inverse conduction problem have been started.

Regression runs have been started to classify the THTF rods for the 150 and 160 series tests. Once the calibration runs are completed, ORINC runs will be made for each test.

Nuclear Pin Simulation. Verification of the back-calculational method is underway. A time-averaging scheme for coolant channel boundary conditions has been developed to contribute to the stability of the hydraulics calculations, and is being tested. The restart module for PINSIM-MOD1 has been completed. Verification of the point reactor kinetics module is underway. Development of the two-group, two-dimensional diffusion equation reactor kinetics for PINSIM-MOD2 continues.

Thermal Hydraulic Simulation. Analysis of tests 100 through 105 for inclusion in the core heat transfer report was concluded. Studies of RELAP's sensitivity to possible errors in experimental data used as boundary conditions were conducted. These studies altered experimental mass flow data by amounts consistent with the standard deviations of the THTF instrumentation and examined the resulting changes in RELAP calculations.

Refinement and quality assurance of the THTF test section model to be used for RELAP calculations for the core heat transfer report was completed. Further analysis of the FCTF awaits receipt of THTF bundle 3 prototype heater rods.

Data Management. A Quick-Look Report for test 166S was generated. Work has begun on the Data Report for this test. Data Reports for tests 103 and 151 have been completed.

Modifications are being made to the existing CHF code to increase the accuracy of times to CHF.

Task 3. THTF Operations. One test was conducted this month: 166R2. Although several minor problems were encountered, the major systems of the facility worked satisfactorily. During the actual blowdown one turbine meter was erroneously set to the wrong signal range and some pertinent data were not obtained. The inboard seal also developed an excessive leak rate during this test and subsequently had to be changed. The Durametallic seal had performed for about 80 hr and 5 blowdowns.

Rust Construction Company finished the installation of automatic closing fire doors on the back side of our control cabinets. The cabinets act as part of the outside wall of the control room. They did not finish phase two of the control room addition. A new estimate is being prepared to finish this work which was stopped because of a lack of funds.

The Westinghouse Company was the successful bidder for rebuilding the spare Allis-Chalmer motor generator set. A purchase order has been issued and some components of the set are expected to be picked up by Westinghouse the second week of February 1978.

Task 4. Two-Phase Flow Studies. A 24-in.-long sample of a new design liquid-level probe developed by Auburn International was subjected to a series of tests including thermal cycling and an isothermal blowdown in the FCTF. After the second thermal cycle (to 450°F and 395 psig) a noticeable steam leak occurred. Repeated thermal cycles and a blowdown from 478°F and 2250 psig resulted in an increased rate of steam leakage. Posttest examination showed that the leakage occurred at a nickel-alumina swaged interface.

Task 5. Bundle Fabrication. Assembly of bundle 2 was completed. Since a date for installation in the test facility has not been set, the unit was sealed with plastic and dust covers in the event extended storage is necessary.

Task 6. Bundle 3. Efforts were continued by four vendors toward fabrication of prototype fuel rod simulators.

Test work was continued to determine the life of fuel rod simulator seal rings as a function of operating temperature and pressure differeential. Preliminary data indicate that the service life of Viton V709-90 is only a few hours at 230 K (450°F).

PROGRAM TITLE: Zircaloy Fuel Cladding Creepdown Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: ORNL # 40 89 55 10 7 (189 # B0124)/NRC # 60 19 10 04 1

TECHNICAL HIGHLIGHTS

The first in-reactor experiment was shipped to ECN-Petten, the Netherlands, on January 19, 1978. After a two-week delay, during which the capsule was lost by the airline carrier, it is being installed in the Dutch reactor. This experiment is a mockup for the creepdown experiments that will be run in FY-1978 and 1979. It will be used to test the heat transfer/temperature calculations and to study the effect of neutron flux and gamma heating on the eddy current system.

A report (Creepdown of Zircaloy Fuel Cladding — Initial Tests, ORNL/NUREG/TM-181) is in the final phase of publication and will be issued this month. It discusses the out-of-reactor creepdown tests conducted to date.

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 20 01

TECHNICAL HIGHLIGHTS

CRI-II:

Optimization of the design of the "flame-spray" plasma arc aerosol generator has continued. With the improved metal powder feed rates, a sustained self-supporting flame using aluminum powder has been obtained. Feed rates, however, are limited by the existing 1/4 in. lead-screw size to about 32 grams of aluminum powder per minute. New equipment on order will increase this by an order of magnitude.

Powdered active uranium has been ordered for use in the NSPP and CRI-II facilities and is being prepared by a hydrogenation-dehydrogenation process at the Y-12 (Union Carbide) metal fabrication facility. At this time, there is no commercial supplier. A vacuum glove-box inert-atmosphere containment is being converted to house the powder feeders for the normally highly pyrophoric uranium. The entire control unit and vacuum dry-box will be operated inside the CRI-II penthouse by remote control. A video camera and a photo diode pyrometer will be used to maintain direct observation of the burning process.

A second generator, which is expected to serve the needs of NSPP for mixed aerosols, should be received this month from the vendor and will be equipped with a triple zone torch to generate Na_2O_x-U_3O_8 aerosols either simultaneously or separately.

NSPP:

During this period, analytical data from Run 201 were obtained. Run 201 was the first of the uranium oxide aerosol test series and utilized the previously untested consummable electrode aerosol generator.

Although this first test was primarily a shakedown run, the full complement of analytical samplers was used.

Approximately 8.9 cm (3.5 in.) of the 2.54 cm (1 in.) diameter uranium electrode was removed by the arc over the 7-minute period of aerosol generation. The major portion of this material fell into the catch pan below the electrode holder as a fine granular black residue which was identified as being predominately U_3O_8. Data from the aerosol mass concentration samplers indicated a maximum aerosol concentration of approximately 0.13 g/m^3 which occurred at about 70 minutes after start of aerosol generation. The concentration dropped to approximately 0.008 g/m^3 at 24 hours after start of generation. Aerosol particle size was measured with Andersen Mark III impactors; the aerodynamic mass median particle diameter increased from about 0.5μ at 17 minutes to 1.8μ at 414 minutes.

Preparations are underway for Run 202 which will be similar to Run 201 but at a higher concentration of uranium oxide aerosol.

FAST/CRI-III:

The CRI-III facility and associated CDV system for use in FAST were made operational this month, and two tests (CDV #32 and #33) were performed. In these experiments, greater than 50kJ was input to the samples during capacitor discharge with energy input times on the order of 15 milliseconds to place the samples in energy states near 4000 Joules/gram. These results are significant improvements over any obtained previously. The resultant aerosol yields were quite large: greater than 3 grams for CDV #32 and greater than 5 grams for CDV #33. The yield for CDV #33 was the largest produced to date in an ORNL CDV experiment. The CRI-III/CDV system is now ready to resume testing the under sodium CDV sample design for the FAST experiments.

ANALYTICAL:

The process of debugging and running sample cases for several of the codes useful to the ART program has continued. Thus far, the following programs have been made to operate sample cases successfully:

1. Heat transfer coefficient codes

    a. condensation on droplets
    b. condensation on thick structures
    c. condensation on thin structures
    d. condensation inside a bubble

2. RATE code (J. G. Refling, U. of VA - nonequilibrium evaporation and condensation during a UO_2 fuel expansion)

3. MELCON code (P. L. Garner, U. of VA - condensation of fuel onto above-code structure during an LMFBR HCDA.

The HNPSD code, which deals with primary aerosol particle-size distribution (M. F. Kennedy, U. of VA), is currently being adapted to System 360 installation.

PROGRAM TITLE: Advanced Instrumentation for Reflood Studies (AIRS)

PROGRAM MANAGER: B. G. Eads

ACTIVITY NUMBER: ORNL #40 89 55 21 8(189 #B0143)/NRC #60 19 10 01 2

TECHNICAL HIGHLIGHTS:

The design, procurement and construction of the AIRS Steam-Water Test Stand is proceeding as scheduled. Present plans are to have the Test Stand at a stage to permit some testing of prototype PKL impedance sensors in May 1978.

ORNL received the conceptual design for film thickness and film velocity measurement from Dr. J. C. Chen of Lehigh University. ORNL will now pursue further development of the concept with the aim to make it applicable in the environment of PKL.

Further development of ceramics for ceramic-to-metal insulator/seals has indicated that rosalite and cordierite are difficult to attach to metals. To date a successful attachment has not been made of these materials to any metals tried. An alumina cermet is under development and shows promise from several points of view. Limited samples tested to date show the cermet to have (1) high tolerance to thermal shock, (2) electrical properties comparable to pure alumina, (3) ability to survive in super-heated steam, (4) little loss of weight (0.024%) due to leaching in 190°C water for 96 hours, and (5) machinability (at the cost of several manhours for one piece).

Visits were made to five manufacturers of high-temperature ceramic-to-metal seals to evaluate their capabilities as potential suppliers of seals for PKL sensors. None of the manufacturers contacted have a product which is directly applicable, but some of the techniques they now use will be explored for their potential applicability. Consideration is being given to placing small orders with some of these companies to manufacture a limited quantity of seals on a best efforts basis for evaluation by ORNL.

Very preliminary results have been obtained from the air-water tests of impedance sensors conducted in December 1978. Based on signal amplitude and coherence between two signals the so-called band configuration showed the most promise of the sensors tested. A conductance measurement was made between bands located on two adjacent rods. The configurations which would be more typical of PKL are two bands on the same rod (measurement between bands) or a single band with the measurement made from the band to the sheath of the rod which is at ground potential. Tests of these configurations in air-water will be conducted in early February. Another sensor configuration which showed less promise from the point-of-view of signal quality is the prong configuration; ie., two wire electrodes protruding into the subchannel from insulators on the rod surface. The prong configuration has the advantage over the band configuration in that it is less difficult to fabricate.

Another series of air-water tests of in-bundle impedance sensors was conducted in January. One of the major purposes of this test series was to investigate additional band and prong sensors in an attempt to optimize such parameters as width and spacing of bands and spacing of the prongs. Analysis of data from this series is underway.

The design was completed on one version of electronics for impedance sensors. The design is based on the use of triaxial cable with a driven shield and on the concept of making a two-component (resistive and capacitive) measurement. Bench testing of a breadboard version showed sufficient promise that two prototype units are being fabricated and will be tested in early February with the in-bundle impedance sensors in air-water.

B. G. Eads presented a program status report to NRC at Silver Spring on January 23, 1978.

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 20 02

Technical Highlights

Development of the ORTAP Code for the Fort St. Vrain (FSV) Reactor: Documentation efforts continued for the overall ORTAP code as well as for the BLAST (steam generator) and ORECA (3-D core) component routines. A new solution technique has been developed for calculating the steam holdup line interfaces, which have been the factors which limit the computation time interval. This technique is presently being debugged.

Use of FSV Data for Code Verification: Detailed transient information and plant data logger output for the FSV scram test from 28% power (8/6/77) was received from GA. GA also supplied a correlation for refueling region outlet gas thermocouples' time response characteristics. Initial calculations of the FSV scram test showed excellent agreement between the measured and predicted values of the 37 refueling region gas outlet temperatures. Parameter sensitivity studies are being run to note the effects on the agreement. A plotting code was also written which produces report-quality plots of the results.

FSV Temperature and Power Oscillation Analysis: The ORECA code was used in an initial investigation of the "unexplained" oscillations that have been observed when the FSV reactor power level exceeds ~55%. More information is required before the question can be resolved.

<u>PROGRAM TITLE:</u> Design Criteria for Piping and Nozzles

<u>PROGRAM MANAGER:</u> S. E. Moore

<u>ACTIVITY NUMBER:</u> 40 89 55 10 2 (189 # B0123)/NRC #60 19 10 05

<u>TECHNICAL HIGHLIGHTS:</u>

<u>PVRC and ASME Code Efforts and Licensing Support:</u> Both the ASME Boiler and Pressure Vessel Code Committee and the PVRC met in January. The Code committee held its regular series of working group and subcommittee meetings during the week of January 9, 1978. One item of interest to the Design Criteria Program was passed by the Working Group on Piping (WGPD), i.e., a revision to the Class I piping stress index table (Table NB-3693.2-1, footnote 2) to modify the grinding requirement for use of the stress indices for "flush" welds in the analysis of welded pipe joints. Other items of interest, including several proposed rules revisions on stress indices and flexibility factors for branch connections and flexibility factors for concentric reducers were held over until the March meeting.

The Pressure Vessel Research Committee (PVRC) held its annual out-of-town meeting in New Orleans during the week of January 23, 1978. One item of general interest concerned the scheduled revision of the PVRC long range plan published in September 1975 (WRC Bulletin No. 209). Each subcommittee (S/C) was asked to update each of its assigned problems with a revised problem definition, current status report, and proposed action plan, and to submit potential new problems as needed. The revised long range plan will then be reviewed by the ASME Code Committee for possible additional problem areas. The new long-range plan will be finalized in June. Other PVRC business included reviews of current work and potential work for next fiscal year.

<u>Nozzles in Cylindrical Reactor Pressure Vessels:</u> The major activity under this category was directed toward investigating the effects of placing an ECCS water injection nozzle between and "close" to the primary coolant nozzle in a 4-loop PWR pressure vessel. The MULT-NOZZLE finite element computer is being used for these studies. Several cases were set up with the water injection nozzle located at different distances from

the primary nozzles in an effort to define the effects of interacting stress fields on the maximum stresses. We expect to complete these studies in February.

Other work on the MULT-NOZZLE program included the development of additional post-processing software and improvements in the preprocessing packages for problem set up and checking purposes.

Elbows and Curved Pipe Studies: During January work was done on three reports in this category. One report (ORNL/NUREG-24) was completed and sent to the publishers. This report summarizes and evaluates experimental results from a series of limit-load studies on twenty 6-in. elbows. New and important conclusions are that (1) the limit-load for an elbow depends on the pipe-radius-to-bend-radius ratio r/R as well as the bend characteristic parameter $\lambda = tR/r^2$, and (2) stainless steel elbows have significantly lower collapse loads than corresponding carbon steel elbows.

Editorial review was conducted on two reports: one describing a series of experimental stress analysis studies on four 10-in. diam machined elbows, and one on an analytical study of elbow end effects. Both reports should be ready for publication early in March.

PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: R. S. Booth

ACTIVITY NUMBER: 40 89 55 11 4 (189 #B0191)/NRC #60 19 10 01 2

TECHNICAL HIGHLIGHTS

No highlights to report this month.

PROGRAM TITLE: Improved Eddy Current In Service Inspection for Steam Generator Tubing

PROGRAM MANAGER: Robert W. McClung

ACTIVITY NUMBER: ORNL #40 89 55 12 1 (189 #B0417-8)/NRC #60 19 10 05

TECHNICAL HIGHLIGHTS:

This activity began in January to develop and qualify optimized eddy current examination techniques and instrumentation for steam generator tubing. The effects of diameter variations (e.g. denting), probe wobble, tube supports and conductivity variations will be separated from defect size, depth and wall thickness variations. In the first phase of the investigation we will use our mathematical models to calculate the instrument readings that will be obtained for various sets of test properties as changes are made in the design of the eddy current circuits and probes. We have an interactive computer program that will calculate the magnitudes and phases of the eddy current signals that are produced by an eddy current coil with various layers of cylindrical conductors encircling and inside dual coils. We are modifying the program to work in a batch mode and also calculate the signals due to defects.

Next, we will interface the program to the least squares design program to fit the test properties to the instrument readings on a least squares basis. The optimum coil and test conditions will be determined.

PROGRAM TITLE: Light Water Reactor Pressure Vessel Irradiation Program

PROGRAM MANAGER: F. B. K. Kam

ACTIVITY NUMBER: ORNL 40 10 01 06 1 (189 #B0415)/NRC #60 19 10 05

TECHNICAL HIGHLIGHTS:

Task 1: Program Administration - A monthly program review was held at ORNL on January 12, 1978, for C. Z. Serpan. C. Z. Serpan and F. B. K. Kam agreed that the ORR Pressure Vessel Benchmark Wall Facility should have metallurgical test matrices at the following locations:

1. 3 cm behind the thermal shield;
2. inside the steel PV at the water-PV interface;
3. at the 1/4T location in the PV; and
4. at the 1/2T location in the PV.

It was further agreed that C. Z. Serpan will decide on the specimen configuration within each of the metallurgical test matrix and transmit this information to F. B. K. Kam by the end of January, 1978.

A collaborative program between Mol, Belgium and ORNL to compare transport calculational results relating to the pressure vessel wall benchmark facilities has begun. George Minsart from Mol, will be at ORNL February 20-23 to compare the results of 1-D and 2-D calculations reported in section 3 below. George will also be here to discuss the use of ORNL's cross section package AMPX and the sensitivity analysis codes available at RSIC. This collaborative program is part of NRC's effort to establish a closer working relationship with several European countries for information exchange in the LWR program.

It is anticipated that Albert Fabry, and G. and S. DeLeeuw from Mol, will be active participants in the experimental dosimetry program. Tentative schedules for their arrival at ORNL in the summer of 1978 have been discussed with Albert Fabry by telephone.

Task 2: Pool Critical Assembly (PCA) Pressure Vessel Wall Benchmark Facility - The engineering drawings through the check-print stage has been completed for the facility. Verification of the structural characteristics of the stand and support components is currently underway by engineering. The drawings are expected to be checked and approved for construction by the next monthly report. A safety analysis review has begun, and the procurement of materials initiated.

Task 3: Oak Ridge Research Reactor (ORR) Pressure Vessel Wall Benchmark Facility - Several one-dimensional (1-D) and two-dimensional (2-D) transport calculations have been completed in support of the design. Based on the results of the two dimensional calculations, it was concluded that a 27" X 27" X 8.85" slab is sufficiently large to simulate an infinite slab as far as the metallurgical irradiation volume was concerned. The results and summary will be presented in a quarterly report to NRC for the period October 1, 1977 to December 31, 1977. The results and analysis of the 1-D calculations are expected to be completed during the second quarter of FY 78.

A heat transfer analysis has been done on a preliminary configuration of the metallurgical test matrices in the instrumented irradiation capsule. This analysis indicated that a more accurate knowledge of the gamma heating in the capsule is needed. Therefore, a gamma heating calculation using a 1-D transport code is currently being prepared. Because of the possible excessive gamma heating, the design concepts include the investigation of heating and cooling capabilities to ensure the required temperature control in the capsule. Provisions will be made in the instrumented capsule to accommodate passive dosimetry sensors.

The dosimetry capsule for the initial flux characterization will be designed to duplicate the dosimetry capsule in the PCA facility. Therefore, this phase of the program will be initiated after the completion of the PCA dosimetry capsule.

PROGRAM TITLE: NRC Measured Data Repository (MDR)

PROGRAM MANAGER: Betty F. Maskewitz

ACTIVITY NUMBER: ORNL #40 89 55 11 9 (189 #B0402)/NRC #60 19 10 01 2

TECHNICAL HIGHLIGHTS

Activities during the first quarter (October-December) focused on the selection and training of technical personnel, and formulating procedures for receiving, storing, and retrieving the expected data.

A visit was made to the ORNL Thermal-Hydraulic Test Facility (THTF), and discussions were held with staff members of the PWR Blowdown Heat Transfer Separate Effects Program with emphasis on the computer processing of test results and the transfer of data to the NRC/RSR Data Bank at INEL.

ORNL MDR personnel visited EG&G Idaho, Inc. for a review of the NRC/RSR Data Bank Program at INEL with emphasis on the Data Bank Processing System and to define procedures for efficient data transfer between the INEL Data Bank and the ORNL Repository. A tour of the LOFT Facility and of the INEL Computer Facilities was included in the visit.

MAJOR PROBLEM

Effective computing technology transfer requires that the transmittal language be independent of the hardware on which the data/technology is written for transmittal. In the discussions with EG&G personnel, plans to transmit the measured data to the ORNL Repository in CDC-dependent binary language (SIDU format) were outlined. ORNL MDR personnel are dependent on using IBM hardware for processing the data for storage and retrieval, therefore this exchange mode is currently impractical. An attempt by MDR personnel to influence the selection of a compatible exchange mode had little success.

An attempt during the visit to convert the CDC binary to IBM binary, using the EG&G computers and CDC software, failed. More than a month later (January 30, 1978), an IBM binary tape containing measured data and project computed data from INEL Semiscale Test S-02-9, was received.

The MDR computer specialist is making a study of the feasibility of devising methods for solving the problem of reading the non-standard INEL imposed SIDU format data tapes. Any mass transfer of data in this format will depend on overcoming this problem.