INTERIM REPORT

ORNL/NUREG/TM-253

# August 1978 Monthly Highlights Office of Nuclear Regulatory Research Programs at Oak Ridge National Laboratory

NRC Research and Technical Assistance Report

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

**OAK RIDGE NATIONAL LABORATORY**

**OPERATED BY UNION CARBIDE CORPORATION · FOR THE DEPARTMENT OF ENERGY**

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Prepared for the
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Prepared by the
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## ABSTRACT

Highlights of technical progress during August 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 — J. G. Merkle attended a meeting at NRL in Washington, D.C., on August 28, to review recent results on fatigue crack growth studies performed at Westinghouse and NRL.

An ORNL quality assurance audit of the HSST Program was held on August 29.

Task 2: Fracture Mechanics and Analysis — Photoelastic studies of inside nozzle corner flaws have been performed with flaws located 90° from the vessel vertical axis. These flaws require approximately twice the pressure of flaws along the vessel axis to induce growth and the slightest deviation from the original plane will cause the flaw to move out of that plane. Also, these flaws do not show the flattening in the central region observed in the flaws located in the vessel axis. A new capability has been developed with the photoelastic technique in which through-the-wall flaws have been produced and stress intensities determined for these configurations.

Task 4: Irradiation Effects — In the second 4T-CTS irradiation experiment analyses of lateral load effects in three-point-bend testing using standard and sharp-edged anvils were successfully completed. Correction of slow bend test results for lateral load effects in irradiated precracked Charpy specimens is in progress. Chemical analyses of the three weldments are nearly complete.

The analyses of neutron dosimeters and irradiation temperatures on the third 4T-CTS irradiation experiment are in progress.

HEDL is now performing unloading compliance tests with a computerized system to obtain $J_{Ic}$ and $J_R$ data. This is a part of the planned development of a testing methodology to be used on the irradiated specimens from the last two 4T irradiation series.

Task 5: Simulated Service Tests — The posttest ultrasonic examination of the V-8 flaw was completed and the vessel has been moved to the machine shop for mechanical cutout of the flaw.

Further evaluation of the weld properties of the V-8 vessel are being performed to include the following:

Testing of the precracked Charpy specimens from the V-8 and V-9 prolongation fabrication weld metal has been completed and the static fracture toughness results indicate no significant differences between the welds.

Type P-2 drop weight specimens are being fabricated from the V-9 prolong fabrication weld in order to determine the NDT of the weld metal. These specimens will be tested together with similar size specimens from the V-9 prolongation base metal.

In addition, a limited number of miniature tensile specimens are being fabricated from the V-9 prolongation weld metal. These will be used to obtain the tensile properties over the test range used to evaluate the static fracture toughness of the V-8 and V-9 fabrication welds.

Task 6: Thermal Shock — A conceptual design for a liquid-nitrogen-quench thermal-shock test facility ($LN_2$-TSTF) that could be used for testing the 990-mm-OD cylinders was completed and a preliminary cost analysis prepared. A typical thermal-shock experiment would consist of dunking the cylinder in a tank filled with $LN_2$. The cylinder would be heavily insulated on the ends and outer surface and would be coated with a special material on the inner surface to enhance heat transfer at this surface.

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

A draft of a journal article on our $^{235}U$ fission product decay heat experiment was completed in August.

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:

The apparatus to be used in the High Temperature Test Series of experiments was assembled and tested with an unirradiated fuel rod. Platinum-rhodium thermocouples which were attached to the rod were employed to calibrate the optical pyrometers to be used during the tests with irradiated fuel.

The High-Temperature Test Series will utilize fully irradiated PWR fuel with cladding slightly expanded and containing a drilled-hole defect that will produce fission product diffusional release characteristics similar to those of a pressure-ruptured fuel rod. The test series will measure fission product release in steam in the temperature range 1200-1600°C.

PROGRAM TITLE: Fission Product Transport Tests

PROGRAM MANAGER: A. P. Malinauskas

ACTIVITY NUMBER: ORNL #40 89 55 11 6 (189 #B0189)/NRC #60 19 10 04 1

TECHNICAL HIGHLIGHTS

Project terminated.

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:

R. H. Chapman and J. L. Crowley participated in the Quarterly Cladding Review Meeting and Experimenter's Workshop at Battelle Columbus Laboratories, Columbus, OH, on August 22-23. Preliminary results of the B-3 test and additional data from the B-1 and B-2 tests were presented.

Mr. E. D. Hindle of the UKAEA Springfields Nuclear Laboratories visited ORNL August 21 for discussions on the B-3 test.

A Quick-look report (internal report ORNL/MRBT-4) on preliminary results of the B-3 test was issued to a limited number of recipients.

Preparations for flow testing the B-3 bundle continued. Deformation and distortion in the bundle were such as to necessitate fabrication of a larger flow shroud in order to perform the flow tests. The flow data will not be available until mid-October. Additional flow characterization tests on the reference (undeformed) bundle were conducted in an attempt to resolve the friction factor dependency on the pitch-to-diameter ratio in the bundle array. The reference bundle must be tested in the new flow shroud also in order to cross correlate the tests in this shroud to the earlier tests.

Development of fabrication procedures and techniques for in-house production of heaters, using BN preforms instead of compacted BN powder, continued. A total of 16 prototype units and 15 preproduction units have been fabricated; it appears that most of the fabrication problems have been eliminated and production of 80 units will be initiated in September.

One of the preproduction units was used to assemble a fuel pin simulator (SR-45) for test and evaluation of the heater. This unit was tested at $\sim$1000°C with good results. Another unit will be tested before production of the 80 units commences.

Development of an improved seal design for the lower end of the bundle simulators was continued with encouraging results. A design concept was selected and proof testing is in progress. If the proof tests are

successful, the gland parts will be modified accordingly for the B-4 simulators.

The programmable power supply on order for use in the heating the shroud in the single rod test facility was received from the manufacturer on schedule. A series of acceptance tests are scheduled for next month. After these tests the unit will be mounted (with necessary controllers and instrumentation) on a moveable dolly so that it can also be used as a power source for heating the 4 × 4 bundle shrouds. This should alleviate the bothersome problems experienced in the B-1 and B-3 tests with adjusting the electrical parameters to cause the shroud heating to be the same as the bundle.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: W. B. Cottrell

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

TECHNICAL HIGHLIGHTS

During the month of August, the staff of the Nuclear Safety Information Center (a) processed 1119 documents, (b) responded to 95 inquiries (of which 58 involved the technical staff and twelve were for commercial users), and (c) made 24 computer searches (of which three involved payment totaling $227.94). The RECON system, which now has over 200 remote terminals, reports that the NSIC data file was accessed 126 times during the previous month (see attached table). The work for the ACRS will be concluded with a cumulative bibliography to be prepared after we receive another batch of documents from them. During the past month, the NSIC staff received eight visitors, participated in one meeting, and has work under way on two evaluations.

One NSIC report is in reproduction: *Breeder Reactor Safety — Review of Current Issues and Bibliography of Literature 1977* (ORNL/NUREG/NSIC-151). Two other reports are in composition: *Reactor Operating Experiences 1975–1977* (ORNL/NUREG/NSIC-144) and *Bibliography of Reports on Research Sponsored by the NRC Office of Nuclear Regulatory Research, January–June 1978* (ORNL/NUREG/NSIC-155). Several other NSIC reports are in various stages of preparation including: *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); *Annotated Bibliography of Licensee Event Reports in Pressurized-Water Nuclear Power Plants as Reported in 1977* (ORNL/NUREG/NSIC-150); *Index of Microfiched Foreign Reports Distributed Under the NRC Light-Water Reactor Safety Research Foreign Technical Exchange Program 1975–1977* (ORNL/NUREG/NSIC-154); and *Reports Distributed Under the NRC Reactor Safety Research Foreign Technical Exchange Program, Vol. V, January–June 1978* (ORNL/NUREG/NSIC-156).

During the month of August, we received 31 foreign documents (9 UKAEA, 5 Japanese, and 17 German). All have been submitted for microfiche distribution. We reviewed the German documents for translation (letters of August 1 and 30, 1978, to G. L. Bennett) and the Japanese documents for translation (letter of August 28, 1978). We would note that four of the nine UK documents were restricted distribution. During the month we distributed 24 reports and four errata on NRC-sponsored safety projects to foreign recipients under the Light-Water Reactor Safety Technical Exchange Agreements and five reports under the Fast Reactor Safety Exchange Agreements. Translations of abstracts of these reports are generally sent to the French and Germans six weeks after the reports are sent. However, the ORNL translations subcontract was recently awarded to a different contractor (who submitted the low bid) and that contractor has been slow in processing material. At this point we are about four months behind.

NSIC's selective dissemination of information (SDI) is available to paying users (as well as exempt users). During the month of August we added three exempt users and one paying user, bringing the SDI service to a total of 375 users.

The regular bimonthly *Nuclear Safety* staff meeting was held August 2nd. Minutes of that meeting and a tentative outline for the next several issues of *Nuclear Safety* have since been disseminated. Copies of *Nuclear Safety* 19(4) were not received in time for critique at the August 2nd staff meeting. *Nuclear Safety* 19(5) was completed on schedule. The technical content of *Nuclear Safety* 19(6) is in final composition at TIC, but that issue awaits the July—August "current events" material (submission deadline - September 15th). All technical articles for *Nuclear Safety* 20(1) have been peer reviewed and are being edited for submission to NRC, DOE, and TIC.

TABLE 1 RECON DATA BASE ACTIVITY FROM 07-01-78 TO 08-01-78
(20 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 | 679 | 1425 | 21305 |
| RIP | (ERDA) ENERGY RESEARCH IN PROGRESS | 125 | 224 | 1704 |
| WRA | (WRSIC) WATER RESOURCES ABSTRACTS | 261 | 1592 | 26013 |
| GAP | (ERDA) GENERAL AND PRACTICAL INFO. | 159 | 226 | 1424 |
| NSR | (NDP) NUCLEAR STRUCTURE REFERENCE | 19 | 84 | 0 |
| ARF | (EMIC/ETIC) AGENT REGISTRY FILE | 11 | 10 | 1 |
| EMI | (EMIC) ENV. MUTAGENS INFO. | 105 | 352 | 3234 |
| EDB | (TIC/CSD) ERDA ENERGY DATABASE | 2092 | 4364 | 100303 |
| ERD | (EISO) ENERGY R&D PROJECTS | 51 | 103 | 181 |
| NBI | (NBIC) NATL BIOMONITORING INV. | 5 | 6 | 0 |
| DBS | (LLL) DATA BASE SURVEY | 19 | 18 | 210 |
| ESI | (EIC) ENV. SCIENCE INDEX | 65 | 127 | 888 |
| EIX | (CSD) ENGINEERING INDEX | 268 | 691 | 5335 |
| MCS | (LLL) ENG. & ENV. DB MODELING SURVEY | 14 | 13 | 116 |
| NSC | (NSIC) NUCLEAR SAFETY INFO. CENTER | 126 | 239 | 12127 |
| WRE | (WRSIC) WATER RESOURCE RESEARCH | 7 | 52 | 0 |
| NRC | (LC) NATIONAL REFERRAL CENTER | 43 | 123 | 265 |
| NER | (EIC) NATIONAL ENERGY REFERRAL | 12 | 43 | 11 |
| RSI | (RSIC) RADIATION SHIELDING INFO. | 13 | 3 | 1174 |
| EIA | (EIC) ENERGY INFO. ABSTRACTS | 57 | 92 | 747 |
| RSC | (RSIC) RADIATION SHIELDING CODES | 5 | 46 | 385 |
| ESR | (ERDA) FED. ENERGY ENV. & SAFETY RES. | 20 | 10 | 11 |
| ETI | (ETIC) ENVIRONMENTAL TERATOLOGY | 57 | 76 | 2257 |

PROGRAM TITLE: PWR Blowdown Heat Transfer-Separate Effects

PROGRAM MANAGER: J. D. White

ACTIVITY NUMBER: ORNL #40 89 55 10 3 (189 #B0125)/NRC #60 19 10 01 2

TECHNICAL HIGHLIGHTS

Task 1: FCTF Operations — Pressed pellets of materials which are possible candidates for use as insulators for bands on impedance probes were tested in the single-rod loop (FCTF). Impedance probes are instruments which are sensitive to the presence/absence of water in a rod bundle. The tests lasted for eight hours each; operating conditions were 561 K (550°F) and 15.5 $MN/m^2$ (2250 psi). The purpose of the tests was to determine the survivability of the materials in hot circulating water. The materials were boron nitride, Lavite, Macor, and Mylnoy (trade names). All of the materials suffered at least 4% weight losses and were judged unacceptable for use in the THTF. Other materials will be tested later.

Task 2: Analysis - Data Management — The critical heat flux (CHF) computer code which searches the rod surface temperatures has been debugged. The code will be run for all power tests to date.

Data reduction of the steady-state files for test 175 has been completed. Production of a calibrated engineering units tape for this test is in the final stage.

Electric Pin Simulation — The rough draft for the documentation of ORTCAL has been completed, reviewed, and sent to the reports office.

Documentation of the surface heat flux perturbation studies has continued.

Necessary program modifications required for the sensitivity studies of the ORINC calculations to such variables as the radial dimensions, thermocouple signal, etc. have been completed. A test matrix for the necessary computer runs has been set up. Debugging of the modified version of ORINC has started.

Nuclear Pin Simulation — The power program developed by PINSIM-MOD1 was used in THTF test 175; results indicate reasonably good agreement

between actual and anticipated generator response, although differences among individual generator responses were noted.

Updated FRAP-T4 models for nuclear pin gas gap pressure and pellet-cladding deformation were obtained from INEL; interface routines between these models and PINSIM are being developed. Development of the two-dimensional, two-group reactor kinetics module is essentially complete; integration of this module into PINSIM has begun.

A rough draft of the documentation for PINSIM-MOD1 has been completed and is being reviewed.

Thermal Hydraulic Simulation — The data evaluation report for test series 2 was written. The results of THTF test 175 have been studied and new orifice sizes have been determined for test 176 (both are part of the preparation for the INEL Code Verification Test). Study of COBRA for use in determination of local fluid conditions in the THTF has begun.

Task 3: THTF Operations — Test 175 was conducted this month. This was the second precursor to the INEL independent verification test 177. Blowdown was initiated from the same loop conditions as in test 174. System decompression was changed by reducing the outlet break area from 60 to 20%. Bundle power decay during blowdown was planned to be controlled by the power programmer. The power was tripped at about 9 sec after blowdown initiation by the thermocouple high temperature limit safety circuit. The primary objectives of determining fluid flows and heat transfer during the first part of the transient were met.

For this safety circuit to trip the power, two thermocouple outputs must be over the temperature limit. Posttest investigation showed that the two thermocouples responsible for the trip had evidenced erratic behavior in previous tests and were clearly in error during this test. As a result, all of the thermocouples on each rod were inspected. A total of 26 were found defective and their output was removed from the safety circuit.

Final acceptance tests of the refurbished MG set was conducted this month in Atlanta. The results were satisfactory and the set will be shipped the last week of the month. The unit will be stored in Building 9204-3. Installation is scheduled for about Oct. 15, 1978. Refurbishment proceedings for the second set have been initiated.

Engineering was completed this month on the five additional instrumented spool pieces. The weldments will be available for installation during the generator changeout.

Hardware installation into the control room addition continued this month. The instrumentation wireways were finished. Installation of the new computer is scheduled for the last day of this month.

Task 4: Three additional (larger) drag targets were installed in the THTF piping spool pieces, and water calibration runs were made at temperatures ranging between 310 to 490 K (100 to 420°F). The results were similar to those reported earlier for the larger target in the horizontal outlet spool piece where the data were linear over the calibration range with very little scatter, essentially independent of temperature. All four drag body flowmeters will contain the larger targets for tests 176 and 177.

TUFASE, a computer program which aids in model studies of the three-beam densitometer has been completed. Testing of various models can now begin.

All drawings required for construction of the additional THTF two-phase flow instrumentation sites planned for FY-78 have been issued. Modifications to the air-water loop to allow testing of the new instrumentation are essentially complete. I&C has apparently solved the problems associated with the ORNL production turbine meter electronics.

Task 5: Bundle 3 — The Babcock and Wilcox Company's grid spacer design was approved and a sole source requisition is being processed by the Purchasing Division. This grid spacer will provide a rod-to-rod pitch of 1.27 cm (0.501 in.).

Initial tests of the Kalrez O-rings that are, or were, intended as FRS seals in the bundle 3 test section disclosed possible difficulties, i.e., leakage. The supplier of the elastomer, DuPont, has been consulted. Additional tests are being designed. The sealing problem is amplified by the closeness of adjacent rods (above), that prevents the use of O-rings having a larger diameter cross-section. Decreasing the FRS outside diameter in the length that is used to contain the seals will prevent individual FRS replacement. Until additional tests are run no final conclusion or remedy by design alterations will be suggested.

Task 6: The failure of the second SEMCO fuel rod simulator (FRS) prototype was due to a short circuit between the lead-in section and the sheath at the point where two of the sheath thermocouples were crossed. This had reduced the BN insulation thickness from approximately 0.762 mm (0.030 in.) to less than 0.381 mm (0.015 in.)(actual measurement of this gap was impractical). As previously covered (July period), there were many other areas of poor quality plus preoperational open-circuits in the three remaining pairs of thermocouples on the sheath inner surface. None of the prototype FRS units received from any of the four vendors were acceptable.

The Fuel Rod Simulator (FRS) Development Laboratory fabrication component has been completed sufficiently to permit fabrication of the approximately 1.5-m (5-ft) long units such as those for the MRBT program. Fill fixtures and other items for the BDHT program are approximately 75% complete with completion scheduled during September 1978. In-house fabrication of prototype units for the BDHT-THTF bundle 3 will begin in October 1978 following receipt of the required materials.

Refurbishing of an adjacent room in 9201-3 is underway to provide a facility for infrared scanning tests and for processing boron nitride (BN) powder and BN preforms. The first of the grooved BN preforms for the bundle 3 outer annulus have been fabricated. Inspection showed excellent control and uniformity of all dimensions with all surfaces smooth (as viewed at 8X) and with a uniform density of 2 gm $cm^{-3}$ (90% of theoretical density). Following purification treatment at temperatutes up to 980°C in ammonia, all results were excellent. Both ID and OD were within 0.0127 mm (0.0005 in.) of design and the BN density was 83.6% (compared to the design parameter of 83%). The handling strength was excellent.

PROGRAM TITLE: Zircaloy Fuel Cladding Collapse Studies

PROGRAM MANAGER: D. O. Hobson

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

TECHNICAL HIGHLIGHTS:

Report not available.

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

FAST/CRI-III:

Two vacuum tests (CDV 55 and 57) were attempted in the CRI-III vessel. Internal arcing caused CDV 55 to be unsuccessful, however, CDV 57 was completely successful, marking the first successful application of the CDV technique in a vacuum ($\sim$200 μm Hg pressure). Two FAST vaporizer tests under water in CRI-III (CDV 54 and 58) were also attempted. In the first, CDV 54, energy input was large but the steel tube did not rupture. In CDV 58, the tube did rupture making it the first successful under-water test of the CDV system.

One FAST vaporizer, with a long pellet stack ($\sim$11 cm), was also tested in argon in CRI-III; the CDV energy input and initial aerosol yield ($\sim$5.7 grams) were the largest achieved to date in a CDV argon test.

NSPP:

During this period, analytical data were received for Run 302, the second mixed oxide aerosol experiment. The $U_3O_8$ aerosol was generated for approximately 44 minutes with the consumable electrode to produce a maximum concentration of about 1 g/m$^3$. At 93 minutes after start of the uranium oxide aerosol generation, the sodium oxide aerosol was generated with a pool fire for about a 2 minute period. At the time of mixing, the $U_3O_8$ concentration was estimated to be around 0.5 g/m$^3$ and the sodium oxide aerosol concentration was about 2 g/m$^3$ producing a $Na_2O$ to $U_3O_8$ mass ratio of 4 to 1. From this point in the experiment, the rate of removal of $U_3O_8$ from the vessel atmosphere increased and was similar to that for the $Na_2O$ aerosol

indicating that the two aerosols were co-agglomerating. This aerosol behavior was essentially the same as that observed during the first mixed oxide aerosol test (Run 301) in which the aerosol mass ratio was about 30 to 1.

CRI-II:

The vacuum glove box and metal powder feed system for the plasma-torch aerosol generator was completed. On August 21, the first test of this generator using uranium powder was conducted. The test was highly successful with a 1 minute burn at an estimated feed rate of 100 g/min producing an aerosol concentration of $\sim$17 g/m$^3$ for a total of $\sim$75 grams airborne in the CRI-II vessel. The aerosol-exhibited behavior is similar to that previously observed for $U_3O_8$ aerosols produced by alternate methods. Later analyses of the aerosol confirmed that it was indeed $U_3O_8$.

The NSPP is now being prepared for installation of this type of generator for conducting the mixed aerosol tests at appropriate concentrations. A five minute burn at a feed rate of 200 g/min should produce about 20 g/m$^3$ in the NSPP if the conversion efficiency is the same as observed in the above test.

ANALYTICAL:

The NSPP data for Runs 103 and 104 have been analyzed in "separate effects" manner in an attempt to see if the measured fallout rates are conforming to the well-stirred Stokes law models in the aerosol behavior codes. Good agreement was obtained in the later portion of the transients but not in the very early period immediately after the pool burning had ceased. This may indicate that the aerosol had not yet become well mixed in the vessel at these early times.

Work has continued to consolidate the source term models and to use PAD to put bounds on our calculation of the energy distribution in the CDV experiments.

PRESENTATIONS:

G. W. Parker, G. E. Creek, and A. L. Sutton, Jr., Agglomeration Characteristics of Fast Reactor HCDA Aerosols, given at 15th DOE Nuclear Air Cleaning Conference, August 7-10, 1978, Boston, Massachusetts.

PROGRAM TITLE: Advanced Instrumentation for Reflood Studies (AIRS)

PROGRAM MANAGER: B. G. Eads

ACTIVITY NUMBER: ORNL # 40 89 55 21 8 (189 # B0413) NRC #60 19 10 01 2

Fabrication of the PKL guide tube impedance probes is approximately 50% complete. The first four completed probes have been transported to the steam test loop for bundle assembly. Probes 5, 6, and 7 are in fabrication and should be completed during the first week of September. Several difficulties have been experienced during this initial fabrication period. Most of these problems will be eliminated or greatly reduced in the building of subsequent probes as fabrication techniques improve with learning and experience.

PKL personnel have agreed to a revised delivery date of October 15, 1978 for the first ten probes. The first five will be tested in the steam water loop in September while the remaining probes are being fabricated. The latter probes will be subjected to normal quality assurance tests during fabrication but will be shipped without testing in the steam-water loop.

The development program for PKL upper plenum instruments is concentrating on two major areas; finding a machinable ceramic which will withstand the less severe environment, and air water testing of prototype versions of the string probe.

The use of a machinable ceramic would offer a lower cost alternative to the alumina-platinum cermet which is very difficult to machine. Materials such as boron nitride, Macor[1], and Lavite[2] have been tested and have exhibited excessive leaching in steam. Another material, tantalum tungstate, was previously reported[3] to have good thermal shock and steam leaching resistance. It is also somewhat softer and easier to machine than the alumina-platinum cermet, but its brazing properties are unknown. It is planned to obtain more of this material for testing in the near future.

Several different prototype designs of the string probe have been fabricated using some of the above materials. These probes will be tested in the air-water loop in order to finalize the configuration for

PKL.

Work is also continuing with the alumina-platinum cermet in the hopes of producing larger sizes suitable for use in the band probes. A solid cylinder about 1 1/2 inches long and 0.6 inches in diameter has been successfully hot pressed and is now being machined. Another attempt will be made to press a hollow cylinder using the tantalum cans as reported last month. The cans have been prepared and shipped to a commercial vendor for the hot pressing operation since the hot press previously used (located in the Y-12 Plant) is down for repairs.

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[1] Trade name - Corning Glass Company

[2] Trade name - 3M Company (American Lava)

[3] B. G. Eads et al., Advanced Instrumentation for Reflood Studies Program Quarterly Progress Report, October 1—December 31, 1977, ORNL/NUREG/TM-202 (in publication).

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: S. J. Ball

ACTIVITY NUM' :: 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: A complete and consistant set of Fortran subroutines was compiled, collated, and documented for an ORTAP sample transient. Copies requested by West Germany (RWTUV) and Japan (IHI) will be sent soon. J. C. Conklin attended a short course entitled "Power Plant Simulation," given at the University of California at Los Angeles (UCLA) on August 21-25, 1978. Dynamic computer modeling of transient heat transfer and fluid flow was examined in detail for both NSSS and BOP components. Emphasis was placed on obtaining fast, stable and accurate numerical solutions to the appropriate equations governing the individual plant component. Control system theory and components were also studied so that the total dynamic response of a power plant can be modeled and predicted.

Assistance to NRC on FSV Licensing Questions: Followup analyses are being made on FSV bypass flow questions and the FSV oscillation problem data.

Use of FSV Data for Code Verification: Data was received from GA on a FSV 30% power scram of 7/23/77, and it is being compared with ORECA code predictions as before. Information on refueling region orifice settings for all 4 scram tests and on an updated model for region outlet thermocouple response was also received from GA and reviewed.

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 10 05

## TECHNICAL HIGHLIGHTS

Task 1: Code Efforts and Licensing Support — There was no activity under this category during August.

Task 2: Miscellaneous Studies — A draft report on the development of stress indices for girth welded joints in Class 1 nuclear piping was completed. The report contains historical background material and technical justification for 63 B, C, and K type stress indices for primary, secondary, and peak stresses including indices to account for the effects of radial weld shrinkage; tapered wall transitions; and mismatch in the diameters or wall thicknesses of the abutting pipes. The report also includes recommendations for specific changes in the stress indices and footnotes of Table NB-3682.2-1 of the Code; as well as suggestions for the incorporation of permissible limits on radial shrinkage of girth-butt welds into the Fabrication Section of the Code (NB-4000).

Task 3: Nozzles in Cylindrical Reactor Pressure Vessels — One report, ORNL/NUREG-18/V2 entitled *Stress Analysis of Cylindrical Pressure Vessels with Closely-Spaced Nozzles by the Finite-Element Method, Vol. 2. Vessels with Two Nozzles Under External Force and Moment Loadings*, by F. K. W. Tso and R. A. Weed of Systems Development Corporation was published. This report documents the development and capabilities of the finite-element computer code MULT-NOZZLE for analyzing 2-nozzle cylindrical vessels. The report also includes results from qualification/demonstration analyses of a thin-walled experimental model with an isolated nozzle (ORNL Model No. 3) that was loaded with orthoginal sets of forces and moments on the ends of the nozzle; and a thick-walled cylindrical vessel with two closely-spaced nozzles. The latter was loaded with internal pressure and with a set of combined loadings on the ends of the nozzles. Results from these analyses demonstrate the program capabilities but do not serve as complete validation for the program because of the absence of suitable experimental benchmark data for comparison.

PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: R. S. Booth

ACTIVITY NUMBER: 40 10 01 06 1 (189 #B0191) #60 19 10 01 2

TECHNICAL HIGHLIGHTS

Monitoring Methods to Detect and Quantify Flow-Induced Vibrations of In-Vessel Components. Previously described difficulties with some of the digital computer codes used in this work have now been resolved and we are well into the "production running" phase of our BWR boiling channel study. During July and August the *forward* TASK calculations were completed, as was a parametric study using the final numerical noise model (including fluctuations in slip and velocity). We are now proceeding with *adjoint* TASK calculations for the A, B, C, and D detector positions; these should be completed by early September.

We next plan to study the changes in in-core detector responses that would be produced by anomalies, such as flow blockages and flow mismatches among the four fuel bundles surrounding an in-core instrument tube. Also, the effects introduced by pressure fluctuations, hitherto ignored in the boiling channel noise model, will be investigated.

Loose-Part Detection Systems. Work during this reporting period focused on three areas: (1) characteristics of nonperpendicular impacts on a large, flat plate, (2) relative merits of impact location by arrival time and signal magnitude methods, respectively, and (3) preparations for impact tests of increased realism at the Experimental Gas Cooled Reactor (EGCR) site. These topics, as well as results from earlier comparisons of the signal transmission properties of various popular accelerometer mounting methods, will be discussed at the September 8 Noise Diagnostics Review Group meeting.

Baseline PWR and BWR Noise. We compared baseline signatures from six PWRs and four BWRs with signatures from two plants having known in-core anomalies. Results from this comparison are shown in the third-quarter technical progress report.

IMPORTANT MEETINGS

W. S. Farmer, NRC:RSR, met with ORNL staff on August 8 and 9 to discuss program accomplishments during FY78 and to provide guidance in planning our FY79 work scope.

PROGRAM TITLE: Impro ' Eddy Current In Service Inspection for Steam Generato .ubing

PROGRAM MANAGER: Robert W. McClung

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

## TECHNICAL HIGHLIGHTS

We have received three additional sizes of Inconel 600 tubing, purchased from Westinghouse, and standards are being machined from these.

Development is continuing on the new microcomputer. Tubing has been completed on the "bread board" version and a printed circuit board is being laid out for ultimate installation in multi-frequency eddy-current instrumentation.

On a separate program a report is being prepared on a modular three frequency eddy current instrument. This instrument will be used for early tests of the steam generator inspections.

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 - Plans are underway for the LWR-PV Irradiation Surveillance Dosimetry Program meeting to be held at ORNL September 12–14, 1978. The agenda and registration information have been sent to expected participants of the meeting.

Free-field spectrometry, SSTR, and emulsion measurements have been completed as well as the in-situ gamma and neutron spectrometry scoping measurements. Although all the experiments have been completed on schedule, a significant amount of reactor operation was required on the evening shift to accomplish that schedule.

The gamma spectrometry measurements, the SSTR, and emulsion measurements were accomplished by HEDL experimenters. The proton recoil measurements were made by J. W. Rogers of INEL. A scoping experiment in the void box was also completed by J. W. Rogers with his proton recoil counters.

Dr. W. Mannhart of Physikalisch-Technische Bundesanstalt (PTB), West Germany, visited the PCA benchmark facility. He stated that if there was a need, he would be interested in the dosimetry measurement program.

An information packet is being prepared with a description of the PCA. The purpose of this packet is to provide interested participants with sufficient information to perform the PCA neutronics calculations.

Task 2: Pool Critical Assembly Pressure Vessel Wall benchmark facility (PCA-PVF) - ORNL Engineering still has not reported results of the analysis of the structural characteristics of the experiment rig and the pool floor with the lead-filled thermal shield in place. Continuation of design of the flux mapping device is awaiting information regarding the geometry of the miniature fission chambers.

Fabrication of all components with the exception of the lead-filled thermal shield is complete. Fabrication of this component is on schedule and should be ready by September 15, 1978.

Task 3: Oak Ridge Research Reactor Pressure Vessel Wall Mockup Facility (ORR-PVF) -

Design Support Calculations - Additional neutron transport calculations were performed during the month of August to accommodate recent changes in the location of the metallurgical specimens in the ORR-PVF. These changes were necessary because the neutron fluence levels on the metallurgical specimens were increased by a factor of four at the request of NRC. The neutron transport calculations performed satisfied two objectives: 1) a detailed evaluation of the validity of the neutronics model used for the design support calculations, and 2) a determination of the fast flux magnitude in the pressure vessel simulator with the ORR operating at 30 MW. The results indicate that the 15 group cross section set is quite acceptable. However, there is some concern with regard to how well the one-dimensional (or two-dimensional) model accounts for three-dimensional leakage effects. The preliminary conclusion is that the one-dimensional model for an infinite slab reactor is adequate. This approximation is realistic since the pressure vessel simulator is sufficiently close to the core that it still acts like a plane source; thus, there is essentially no geometric attenuation. Based on the above model (i.e. one-dimensional with zero transverse leakage) the flux above 0.82 MeV at the 1/4-T position in the pressure vessel (with the ORR at 30 MW) is estimated to be approximately $1 \times 10^{12}$ n/cm$^2$/sec for the present design configuration.

Facility Design and Fabrication - Detailed design of the ORR facility is continuing. Assembly and layout work on the drive mechanism are almost complete. Design of the dosimetry capsule for the pressure vessel simulator is continuing. Current efforts are directed toward verifying the nuclear equivalence of a proposed concept with the design of the actual instrumented irradiation capsule. Two-dimensional transport theory is being used to evaluate the two designs.

Instrumented Irradiation Capsule (IIC) - Computational heat transfer model V, which represents all of the IIC in two-dimensional XZ geometry, indicates that the required temperature range may be obtained by a total power of 12 kW including gamma heating and heat generated in the six heater assemblies, or by substituting Neon for the Helium as the filler

gas and reducing the load on the heaters in the event that the gamma heat is much lower than expected.

Results of a crude calculation of the temperature rate of change show that a rate of 1°F/min per kW power is an upper limit on the temperature rate of change in the capsules. This result indicates slow temperature response to variations in the heat generated in the system.

Further analysis on different methods of cooling the surveillance specimens capsule have to be looked at. The location of the thermal shield and this capsule as well as the narrow clearances above and behind the capsule may complicate the design.

The drawings of the different capsule's components have been reviewed. Locations of water lines, gas lines and electric instrumentation and heater connections have been determined so that the detailed design may be finalized.

A special container box for the prototype assemblies is being designed. This box will allow testing the performance of the different assemblies as well as the heat transfer properties of the gas gaps. The prototype cooler plate assembly was satisfactorily machined.

The technical memo on the gamma-heating computation is in the review process.

Process Control System (PCS) - Alternate system designs are still under evaluation.

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

Six test data tapes containing measurement data and project computed data were received from the NRC/RSR Data Bank and copied to MDR storage. These data tapes represent the following Semiscale tests:

| | |
|---|---|
| S-06-3 (Short) | S-06-4 (Long) |
| S-06-3 (Long) | S-06-5 (Short) |
| S-06-4 (Short) | S-06-5 (Long) |

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