INTERIM REPORT

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Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
Operated by
UNION CARBIDE CORPORATION
for the
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NRC Research and Technical
[redacted] Assistance Report

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INTERIM REPORT

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ABSTRACT

Highlights of technical progress during November 1979 are presented for fourteen separate program activities which comprise the ORNL research program for the Office of Nuclear Regulatory Research's Division of Reactor Safety Research.

PROGRAM TITLE: Advanced Instrumentation for Reflood Studies (AIRS)

PROGRAM MANAGER: B. G. EADS

ACTIVITY NUMBER: ORNL #41 89 55 11 8 (189 #B0413)/NRC #60 19 11 01

A Mod III (with insulators in the side wall) version of the small PKL string probe was tested in steam-water flow. As of this time the results are inconclusive as a leak in a triax cable seal caused some runs to be aborted. The quality of the velocity measurement appeared to be improved over small string probe designs previously tested; however, further testing in steam-water and in the air-water Instrument Development loop is required. These tests are planned for early December.

A series of meetings was held at ORNL with Messrs. J. B. Brand and J. R. Liebert of KWU PKL Facility to discuss the results of the tests of flag probe in the KWU 3x3 rod bundle. These tests were completed in early May 1979 and the meetings were held to discuss the data analysis results of KWU and ORNL and to combine the results to draw conclusions about the tests. In general it may be concluded that the probe survived intact the environmental conditions of the ten tests except that two triax cables apparently had some leakage of moisture through their end seals and this probably affected the data adversely in some of the later tests. The lead wire from one cable failed as an open-circuit during the seventh test. It is thought that this was due to fatigue resulting from the stress of thermal cycling. A design change in subsequent guide tubes has been made in the expectation of eliminating this type of failure. Valid two-phase velocity data appears to have been obtained in at least two tests which have been extensively analyzed. A comparison of velocity determined from dual flash photographs of the in-core flow showed reasonable agreement. The velocities predicted by impedance signal analysis tend to fall at the lower boundary of the range of velocities from the photographs. The reasons for this are highly speculative and do not appear to warrant further investigation based on the relatively small differences.

Probably the most serious concern identified in these tests was the presence of a liquid film on the unheated guide tube in the presence of relatively high void fraction flow in the subchannel. Once a film forms on the guide tube and bridges the gap between the posts of the flags the measurement sensitivity to subchannel phenomena is decreased. The effect of this is most noticeable on the void fraction measurement. At the time of arrival of the quench front or wetting of the guide tube a sudden decrease occurs in the impedance signal. The interpretation of void fraction from the signal once this occurs is more difficult and further work is needed in this area.

A heavy effort is now underway to manufacture parts for in-core guide tube film and impedance probes for SCTF I. Also underway is the manufacture of parts for PKL II upper plenum impedance probes and wall film probes. One problem in this area is that the large effort at this time is saturating the shops of the vendor which machines all the parts and this is slowing down the availability of

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parts needed for continuation of some of the sensor development efforts. At this time no other source of machining has identified.

[redacted]

PROGRAM TITLE: Aerosol Release and Transport from LMFBR Fuel

PROGRAM MANAGER: T. S. Kress

ACTIVITY NUMBER: ORNL # 41 89 55 11 1 (189 #B0121)/NRC # 60 19 20 01

TECHNICAL HIGHLIGHTS

FAST/CRI-III:

Five tests were performed in the FAST facility this month (FAST 35-39). FAST 35 maintained the argon cover gas pressure (above the water) at 2.02 MPA (20 atm). In FAST 36 through 38, the argon cover gas pressure was maintained at 0.025 MPA (0.25 atm). In these three tests, the observed time between the first two pressure pulses (due to bubble oscillations) was ~170 ms. As expected, this was considerably greater than in the tests at the higher argon pressure levels. In FAST 29, the water temperature was raised to 355 K (180 F) whereas in the previous tests, the water had been maintained at room temperature (~298 K (73 F)). FAST 39 is the first test in which a significant amount of fuel aerosol was measured to be transported through the water to the cover gas space. Due to the nearness of the water temperature to saturation in this test, it is possible that significant quantities of water were vaporized forming a water vapor bubble that carried the fuel through the water to the cover gas.

One test was performed in the CRI-III facility (CDV 89). This was a test in an argon atmosphere in which the electrical preheating level was set at 2000 Watts (the normal level used is 1700 Watts) to attempt to ensure full pellet melting before capacitor discharge. In tests with a 1700 Watt preheat, the capacitor discharge current typically rises for ~1 millisecond, then is roughly constant for a few tenths of a millisecond, and then increases up to the time of sample breakup. In CDV 89, the capacitor discharge current continually increased for the start of discharge to the time of sample breakup. This result may indicate that at 1700 Watt preheat levels, some portion of the pellets are not molten.

NSPP:

Only a small amount of data has been received from the analytical laboratory for the third mixed oxide aerosol experiment (No. 305). Estimates from preliminary data indicate that the mass ratio of sodium oxide to uranium oxide may have been in the range 15:1 to 20:1 compared to the target ratio of 10:1. Complete analytical data from this experiment is expected shortly.

CRI-II.

To ensure confidence in the performance of the spiral centrifuge aerosol spectrometer, a primary calibration was made using standard spheres. The test aerosols of monodisperse polystyrene microspheres were introduced after drying in a diffusion type desiccant chamber and neutralizing excess electrical charge in a Kr^{85} ionization chamber. The first few calibration runs at the usual sampling flow rate of 1 liter/min gave an unresolved distribution of signlets, doublets, and triplets; however, following a suggestion by Owen Moss of BNWL, who did the original calibration on this instrument while at LASL, the sample air flow was reduced to 0.5 liters/min and the desired resolution of three or four distinct bands was obtained for each test aerosol. The indicated aerodynamic diameters using the Stöber correction for the doublets and triplets were fitted to the LASL calibration curve with very little deviation although the difference in barometric pressure at the two localities is significant (580 mm vs 740 mm).

PROGRAM TITLE: Heavy-Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

ACTIVITY NUMBER: ORNL #41 89 55 01 1 (189 #B0119)/NRC #60 19 11 5

TECHNICAL HIGHLIGHTS

Task 1: Program Administration — J. G. Merkle attended meetings of ASTM Committee E-24 task groups on crack arrest, crack opening displacement, elastic-plastic fracture technology, precracked Charpy testing and alternative test methods in Pittsburgh, PA, on October 30 and November 1.

On November 5, J. W. Bryson and R. D. Cheverton presented papers at the Seventh Light Water Reactor Safety Research Information Meeting held at NBS in Gaithersburg, MD.

G. D. Whitman attended a VIRG meeting on November 6, at NRC offices in Silver Spring, MD, to review the irradiations testing projects.

On November 9 and 12, Jürgen Föhl of MPA, Stuttgart, visited ORNL to discuss cooperative irradiations work utilizing German steels.

J. G. Merkle attended meetings of the PVRC-MPC Task Group on Reference Toughness, the ASME Section XI Task Group on Reactor Vessel Operating Criteria and the Working Group on Evaluation in Bethesda, MD, on November 12 and 13.

Visitors from North Carolina A&T State University toured the HSST display area on November 30.

Task 2: Fracture Mechanics and Analysis — At the request of Sandia Laboratories, written comments were prepared and transmitted to Washington University Technology Associates on a report prepared by them entitled "Resolution of Unresolved Safety Issue Generic Task A-11, Reactor Vessel Material Toughness, Independent Reviewer Portion." It was recommended that the title be changed to something like "Development of J Integral Analysis Methods for Surface Cracks in Pressure Vessels Carrying Loads in the Nominally Elastic Range."

Work was continued to further clarify the analytical elastic-plastic analysis of notches, and its application to the Tangent Modulus Method of inelastic fracture analysis. The basic relationship, $K_I^2 = EJ$ was shown to hold for notches with finite root radii as well as for cracks, a result previously demonstrated only implicitly by Rice.

Work is continuing on the development of an IBM version of the NOZ-FLAW computer program and the preparation of a user's manual for the program.

This month work was initiated on the determination of K_I distributions for nozzle corner flaws under combined pressure-thermal loadings. Both NOZ-FLAW and BIGIF will be used in conjunction with the ADINA computer code in conducting this work. A finite-element mesh was generated for an uncracked BWR feedwater nozzle configuration and a steady state temperature distribution was determined for conditions simulating normal operating conditions of a BWR. Steady-state data obtained at 97 percent power on November 12, 1975, at the Millstone Nuclear Power Station, Unit 1, were used in the analysis. This temperature distribution will be used to calculate thermal stresses throughout the structure. Subsequent superposition of stresses due to internal pressure loading will then yield a complete thermoelastic solution. Finally, the uncracked stress distribution obtained above will be input to BIGIF and K_I's will be determined for several different flaw sizes.

Task 3: Irradiation Effects — Initial planning was performed to ship all of the remaining compact specimens from the 2nd and 3rd irradiation series experiments to NRL for testing.

A temporary 6-cm-thick steel thermal shield was installed at the BSR and the first capsule of the fourth irradiation series was operated with the BSR at partial and full power. Tests were conducted with several gas mixtures, with and without electrical trim heat. These tests showed that a 4.25-cm-thick thermal shield will allow us to operate the capsule with an adequate temperature control range. The 4.25-cm thermal shield and supports were designed and fabrication is in progress. We plan to conduct a dosimetry experiment using our dummy capsule and the thermal shield to determine neutron spectra and fluxes. Results of this dosimetry experiment will be used in determining irradiation time.

We will use a similar 4.25-cm thermal shield with the capsule to be installed on a second face of the BSR and will conduct a dosimetry experiment for that position. Design of supports for this second position is

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in progress. Use of the thermal shield eliminates the need to revise the capsule design.

Task 4: Thermal Shock — Fracture mechanics calculations were made for TSE-5 using a model that included seven and also nine long axial flaws in addition to the primary flaw. This was done to determine the effect of the secondary cracking on the behavior of the primary flaw. Results of the analysis indicate that the effect is negligible. Based on other evidence it appears that the secondary cracking took place after the third and final event associated with the primary flaw. However, even if this were not the case, it now appears that the primary cracking would be of no consequence with regard to the interpretation of the behavior of the secondary cracking.

Strain gages (Ailtech SG 125) similar to those used as COD gages during TSE-5 have been calibrated in a testing machine (MTS) under simulated TSE-5 conditions, which included submergence in liquid nitrogen and a 10-mm-long unbonded midsection. The displacement vs strain-output curve was essentially linear up to the breaking point, which was ~6% average strain or ~12% local strain (unbonded section). The slope of the curve was considerably different from that deduced from the gage factor. The agreement between calculated COD vs crack depth and measured COD vs crack depth is quite good now.

Calculations were completed in connection with achieving a more severe thermal shock by means of applying heat to the outer surface of the cylindrical test specimen while removing heat from the inner surface. The inner-surface temperature was limited to ~96°C, and the heat flux applied to the outer surface was limited to 31,000 W/m^2 (120 kW for a TSC-1 type cylinder). Results of the analysis show that in terms of the K ratio (K_I/K_{Ic}) the effect of higher thermal stress (more severe thermal shock) is offset by the higher temperature. Thus there is essentially no net benefit from the more severe thermal shock obtained in the manner described above.

Characterization of A508 material in connection with TSE-5 and a possible future thermal shock experiment is continuing. Specimen machining drawings have been prepared, and machining has begun on 24 additional 1T compact specimens to be obtained from two radial segments from prolongation TSP-2. One piece was tempered for 4 h and 675°C; the second for 4 h at 705°C. Both were cooled in still air. The specimens tempered at 705°C will be tested at ORNL and the specimens tempered at 675°C will be tested at Battelle.

The tensile specimens from vessel TSC-1 and prolongation TSP-1 have been tested and the reduction of the data is in progress.

The 2T, 1T, and C_VT compact specimens from prolongation TSP-2 have been received and are being tested. The drop-weight specimens from prolongation TSP-1 have also been received and will be tested shortly.

Two 1T CS (specimen numbers TSP-31 and TSP-32) and one 2T CS (specimen number TSP-1) from prolongation TSP-1 have been tested at 82°C. The 1T CS exhibited fracture toughness (K_{Icd}) values of 209 and 131 MPa $\sqrt{m}$, respectively. The 2T CS that is assumed to be cracked to an a/w depth of 0.508 did not fracture during testing. The load during testing exceeded the recommended capacity of our machine (414 MPa) and testing was stopped. A calculation of the fracture toughness based on an a/w of 0.508 and the maximum load imposed on the specimen (428 MPa) provide a fracture toughness (K_{Icd}) of 176 MPa $\sqrt{m}$. Testing of 1T CS and 2T CS from prolongation TSP-1 is continuing.

Task 5: Simulated Service Tests — Requests for bids for the repair of intermediate test vessel V-8 and the preparation of a low upper-shelf weld in the vessel were issued. The vessel will be tested with the flawed material at upper-shelf temperature for the purpose of studying the application of elastic-plastic fracture mechanics to a thick vessel.

The vessel preparation work to be subcontracted includes the preparation of trial welds, testing specimens to determine that the welding procedures produces the desired properties, preparation of prototypic vessel weldments from which specimens will be taken for characterizing the special weld in V-8, repairing and placing the special weld in V-8, performing material characterization tests, and preparation of a flawing practice weld. Completion of this phase of the V-8A test preparations is expected to take several months.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: S. J. Ball

ACTIVITY NUMBER: ORNL #41 89 55 11 2 (189 #B0122)/NRC #60 19 13 02

TECHNICAL HIGHLIGHTS

Code Development Activities: Development work has continued on the ORTAP-FSV code and its component subsystem simulation subroutines. Sample programs and output runs of the ORTAP code family were requested both by BNL and PSC. Efforts to assemble a complete set of the most recent versions are nearly complete.

The new version of ORTURB, as reported last month, predicted steady state turbine conditions at 100 and 25% power that agreed very well with published heat balances. However, certain feedwater flow transients were identified that might cause ORTURB to calculate pressures improperly. Additionally, the algorithm for calculating the high pressure turbine governing stage shell pressure was identified as needing improvement. Work on the necessary changes is underway.

BLAST calculations were performed using modifications supplied by RWTUV to the matrix inversion technique. Calculated results obtained with this new technique are identical to results obtained with the original BLAST matrix inversion technique and computer time requirements are significantly reduced. The new matrix inversion has been incorporated into ORNL's BLAST. Other modifications to BLAST developed by RWTUV are being studied. Development of input to BLAST for simulation of an oscillation transient was initiated. Measured steam generator response for this transient was supplied to ORNL by FSV.

FSV Upper Plenum Reverse Flow Plume experiments: The intermediate plume experiment was run at a variety of plume temperature and flow conditions with a fixed plume geometry (nozzle diameter and plume height). The data indicated that the scatter in the resulting Nusselt number calculations will be small enough to determine the validity of Reynolds/Grashof similarity relations for extrapolating from air models to FSV helium conditions.

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

PROGRAM MANAGER: Robert W. McClung

ACTIVITY NUMBER: ORNL 41 89 55 12 1 (189 B0417-8)/NRC 60 19 11 05

TECHNICAL HIGHLIGHTS

We are continuing our task to improve the detection of defects in the presence of other property variations.

We have completed the hardware assembly, programming, and minicomputer control interface for our new automatic positioning device. This device will improve the speed and accuracy with which we can accumulate the data from laboratory specimens that are used to construct the least-squares-fitted functions that are, in turn, used to compute test piece properties. For example, a data-recording task that formerly required a dedicated operator and assisting minicomputer for 8 h can now be completed in 1.5 h with only operator supervision. Possible errors in manual specimen position adjustment are eliminated. (The operation of this device bears no direct relation to the ultimate tube inspection speed that will be obtainable. The limiting factors in that case are the coil length, rates of property computation, and density of interrogations required. With the present system the inspection time will be a few minutes per tube.)

C. V. Dodd and L. D. Chitwood visited the North Anna (Virginia) Nuclear Power Station November 5-7 to observe eddy-current in-service inspection of steam generator tubing. Methods in development at ORNL appear to be compatible with the restrictions imposed by field inspection conditions. Our digital instrument design may offer some advantages in the semiremote operational situation.

C. V. Dodd presented a talk, "Improved Multifrequency Eddy-Current Test and Analysis for In-Service Inspection of Steam Generator Tubes," to the Seventh Water Reactor Safety Research Information Meeting at the National Bureau of Standards, Gaithersburg, Maryland, on November 6. A similar presentation was given at the Nuclear Regulatory Commission, Bethesda, Maryland, on November 7.

At Bethesda, Dr. Dodd reestablished contact with R. Clark of PNL, who agreed to send ORNL some samples of 0.875-in.-diam by 0.050 wall Inconel 600 tubing with chemically induced SCC defects. We expect to have these available for testing during the week of December 10.

PROGRAM TITLE: Instrument Development Loop

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: ORNL #41 89 55 12 5 (189 #B0427)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS

Task 1: Operations — High downflow tests were completed in the single-module air/water loop to obtain calibration data for the tie-plate drag body. The maximum flow rate achieved was 37.3 kg/sec compared to estimated maxima for the UPTF of 42 and 35 kg/sec for the refill and reflood phases respectively. Up to a flow rate of ~18 kg/sec there was no measurable collapsed liquid level which implies that this flow rate is also required to provide a liquid seal at the tie plate. At 36.4 kg/sec water started coming out of the top of the vessel, 1205 mm above the tie plate, i.e., above the bottom of the hot leg location.

At the Tripartite 2D/3D Meetings, October 30-November 2, 1979, it was agreed that the tie-plate drag body was the primary candidate for momentum flux measurements at UCSP region for UPTF and SCTF with dP cell as the backup alternative. In addition, one or more of the following are candidate instruments for the UCSP region: turbine, string probe and heated thermocouple.

Task 2: Construction — Fabrication of the steam-water loop pressure vessel was completed and the vessel was hydrostatically tested at 20 bar. The vessel is scheduled to be installed the first week in December. All procurement is complete except for one heat exchanger and the steam/H_2O separator, both of which are expected during December.

PROGRAM TITLE: Light Water Reactor Pressure Vessel Irradiation Program

PROGRAM MANAGER: F. B. K. Kam

ACTIVITY NUMBER: ORNL 41 89 55 12 0 (189 #B0415) / NRC #60 19 11 05

TECHNICAL HIGHLIGHTS

Task 1: Program Administration - All Charpy-V specimens for the void box capsule have now been received.

F. B. K. Kam and F. W. Stallmann attended the NRC Seventh Water Reactor Safety Research Information Meeting which was held at NBS on Nov. 5-9, 1979, in conjunction with the LWR-PV Irradiation Surveillance Dosimetry Program Meeting. One invited paper was presented at the information meeting.

A poster session on "Nuclear Heating in Thick Iron Slabs in the PSF" was presented at the ANS Winter Meeting by I. Siman-Tov.

Task 2: Benchmark Fields -

A. Dosimetry Measurements

1) PCA Dosimetry Measurements -

a) Fission chamber checks of the core power level and fission chamber traverses in water of the 12/13 configuration for the PCA "Blind Test" were made.

2) PSF Startup Dosimetry Measurements

a) The ORR-PSF full power (30 MW) dosimetry runs in the dosimetry surveillance and PV simulator capsules have been completed. Foil counting is in progress.

B. Mechanical Design and Installation Activity

1) PSF

a) Revision of drawings reflecting as-built conditions has been completed.

2. IIC Fabrication and Assembly

a) Specimen assemblies were ground with no problems encountered and are ready to be inserted into the simulated surveillance capsule (SSC) and the pressure vessel simulator capsule (PVS).

b) Assembly of the two capsules is continuing.

c) Preparation of the reactor experimental review questionnaire is in progress.

C. Process Control System - The experiment plan and manual for the PVS capsule characterization have been written. In addition, the software for generating this characterization is on-site and ready for implementation in the ORNL DEC-10 system.

The operations manual for the data acquisition routines is in the final revision stage.

Task 3: Neutron Field Characterization - Results from the Monte Carlo calculation need to be improved. Longer runs are not considered a practical option for reducing the variance associated with several detector positions. Several biasing schemes have been evaluated and several remain to be tested. It appears that the PCA Monte Carlo calculations can be successfully biased.

Task 4: Dosimetry and Damage Correlation - Two topics related to this task were discussed at the LWR program meeting at NBS, Nov. 6-8, 1979:

1) Evaluation of the "Blind Test". It was agreed to use both absolute reaction rates and benchmark referenced equivalent fission fluxes for comparison between experimental and calculated data. ORNL CEN-SEK and NBS will process the calculated data in a uniform and consistent manner to obtain a meaningful intercomparison. The deadline for submitting calculations was postponed to February 1, 1980, the results of the evaluation will be presented at a meeting at NBS May 20-22, 1980.

2) Metallurgical Irradiation - The startup dosimetry revealed slightly higher fluxes at the PSF than anticipated. It was agreed to set irradiation times for the metallurgical capsules in such a manner that the minimum fluences >1 MeV at the 1/4 T capsule is $1.6 \cdot 10^{19}$ n cm^{-2}, corresponding to approximately 20 full power months of irradiation.

Analysis of the counting data from the dosimetry in PCA and PSF is continuing. Slight changes in software were necessary to accommodate the changed format in the more recent data.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: ORNL #41 89 55 10 6 (189 #B0120)/NRC #60 19 11 04 1

TECHNICAL HIGHLIGHTS

R. H. Chapman attended the WRSR Information Meeting in Bethesda, MD, November 5–8. This also provided an opportunity to engage in technical discussions with NRC/RSR personnel and foreign researches on matters of mutual interest.

On November 13, Dr. H. E. Rosinger of AECL Whiteshell Nuclear Research Establishment visited ORNL for discussions on cladding deformation research and experimental techniques. He is initiating an experimental investigation, using single rod tests at first, to aid model development and verification appropriate to CANDU reactors.

On November 15 Dr. D. O. Pickman and Mr. C. A. Mann of the UKAEA Springfields Nuclear Power Development Laboratories visited ORNL for discussions of recent MRBT results and to exchange ideas of mutual interest.

On November 30, Dr. J. C. Turnage and Dr. S. P. Schultz of Yankee Atomic Electric Company visited ORNL for discussions on cladding deformation pertinent to licensing issues affecting the Yankee Rowe plant.

Two single rod tests, both with heated shroud, were conducted during this reporting period. This brings the total to 17 tests (14 with and 3 without shroud heating) conducted in this series of scoping tests. (The total and the number of heated shroud tests reported last month for this test series were in error by two and should have been 15 and 12, respectively.) Preliminary evaluation of the two tests conducted this month are consistent with the previous ones, and the comments reported last month remain valid.

Fabrication of the fuel pin simulators for the B-5 (8 × 8) bundle continues to show progress; approximately 75% of the required number have been completed. Fabrication of the other test hardware is in progress.

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PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: R. S. Booth

ACTIVITY NUMBER: ORNL #41 89 55 11 4 (189 #B0191)/NRC #60 19 11 1

TECHNICAL HIGHLIGHTS

Loose-Part Detection Systems. Preparations were made for new impact tests of ~2 weeks duration at the EGCR facility. These tests, which will employ a remotely actuated solenoid device to produce both in-vessel and ex-vessel impacts, will provide additional data with which to (1) test our new amplitude-based impact location scheme, (2) quantify the degrading effects resulting from background noise introduction, and (3) establish with greater precision the merits of low-frequency (dc-7 kHz) and high-frequency (7-50 kHz) information contained in the accelerometer signals.

Substantial improvements to the iterative impact locating algorithm were also made; false convergence to local maxima in the goodness-of-fit surface no longer presents a problem. We also explored further the degree to which signal magnitude in the vicinity of the accelerometer's resonant frequency can be used to characterize impacts.

A comprehensive summary of the theoretical and experimental work on loose-part detection methods performed to date is in preparation. This report is now ~40% complete and is expected to be issued in the latter part of FY80.

Nuclear Industry Standards. In response to a request from the chairman of an ASME subgroup that is preparing an industry standard on PWR core support barrel axial preload monitoring, we prepared four neutron noise signature composite plots from data in our signature

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library and transmitted them to the subgroup chairman for possible incorporation in the proposed standard.

Monitoring Methods to Detect and Quantify Flow-Induced Vibrations of In-Vessel Components. A random access I/O simulation routine has been completed for a 2-D frequency-dependent computer code. The code is now undergoing test on the IBM 370 computer.

Approval to run the code at HEDL has been obtained from DOE. The code will be run at HEDL and the results compared with the IBM 370 version.

Publications and Reports. We presented a paper, "Application of Noise Analysis to Safety-Related Diagnostics and Assessments," at the Division of Reactor Safety Research's Seventh Water Reactor Safety Research Information Meeting at the National Bureau of Standards, November 5-9, 1979.

We presented the following papers at the American Nuclear Society Meeting in San Francisco, November 11-15, 1979:

"Sensitivity of Detecting BWR Control Rod Vibrations Using Neutron Noise"

"Modeling the Local Component of In-Core Neutron Detector Noise in a BWR"

"Monitoring BWR Stability Using Time Series Analysis of Neutron Noise"

We are preparing an invited paper for the American Nuclear Society's Topical Meeting on Thermal Reactor Safety to be held in Knoxville, Tennessee, April 8-11, 1980.

PROGRAM TITLE: NRC Reactor Safety Research Data Repository (RSRDR)

PROGRAM MANAGER: Betty F. Maskewitz

ACTIVITY NUMBER: ORNL #41 89 55 11 9 (189 #B0402)/NRC 60 19 11 01

TECHNICAL HIGHLIGHTS

No technical highlights this month.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: W. B. Cottrell

ACTIVITY NUMBER: ORNL #41 89 55 10 4 (189 #B0126)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS

During the month of November, the staff of the Nuclear Safety Information Center (a) processed 1011 documents, (b) responded to 59 inquiries (of which 38 involved the technical staff and 10 were for commercial users), and (c) made 14 computer searches. The RECON system, which now has over 200 remote terminals, reports that the NSIC data file was accessed 143 times between October 1 to 31 making it the sixth most utilized of the 25 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 8 visitors and participated in 5 meetings.

One NSIC report was issued during October: *Reports Distributed Under the NRC Reactor Safety Research Foreign Technical Exchange Program, Vol. VII* (ORNL/NUREG/NSIC-170). Several other NSIC reports are in various stages of preparation, including *Radiation in Perspective* (ORNL/NUREG/NSIC-161); *Annotated Bibliography on the Safeguards Against Proliferation of Nuclear Materials* (ORNL/NUREG/NSIC-160); *Breeder Reactor Safety: Review of Current Issues and Bibliography of Literature* (ORNL/NUREG/NSIC-166); *Role of Probability in Risk and Safety Analysis* (ORNL/NUREG/NSIC-167); *Annotated Bibliography on Fire and Fire Protection in Nuclear Facilities* (ORNL/NUREG/NSIC-172); *Annotated Bibliography on the Transportation and Handling of Radioactive Materials* (ORNL/NUREG/NSIC-168); *Bibliography of Reports on Research Sponsored By the NRC Office of Nuclear Regulatory Research* (ORNL/NUREG/NSIC-169); and *Description of Selected Incidents Which Have Occurred in Nuclear Facilities* (ORNL/NUREG/NSIC-176).

During the month of November, we received 20 foreign documents (5 German, 9 Japanese and 6 UKAEA). In accordance with the arrangements effective January 1, 1979, a copy of each of these have been sent to Steve Scott for microfiche processing. In addition, the foreign language documents were reviewed for translation (see two letters of November 29, 1979, to G. L. Bennett).

NSIC's selective dissemination of information (SDI) is available to paying users (as well as exempt users). During the month of November we added 8 exempt which, with other withdrawals and renewals, leaves the SDI service at a total of 398 users.

Much of NSIC routine operations are dependent upon the Computer Technology Center at K-25 which houses the IBM equipment which in turn supports our remote consoles. This system was inoperative from November 16 to November 29 before IBM personnel could locate and repair some problems in our console control unit, which may have been induced by some recent changes at CTC.

All technical articles for *Nuclear Safety* 21(2) were completed and mailed to NRC, DOE and TIC on November 21st. The "current events" material (covering events which occurred during September and October) for *Nuclear Safety* 21(1) was completed by the same date (which was about one week late because of November meetings attended by the Editor). All technical articles for *Nuclear Safety* 21(3) have been received, submitted to peer review, and are in various stages of preparation. Final copies of *Nuclear Safety* 20(6) are expected from the printer by the first of next week.

TABLE 1 RECON DATA BASE ACTIVITY FROM 10-01-79 TO 10-31-79
(22 OPERATING DAYS)

DATA BASE IDENT.	DATA BASE NAME AND SUPPORTING INSTALLATION IDENTIFICATION	NO. OF SESSIONS	NO. OF EXPANDS	CITATIONS PRINTED
EDB	(TIC) DOE ENERGY DATABASE	3580	5514	132822
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	661	1142	12765
WRA	(WRSIC) WATER RECOURSES ABSTRACTS	379	881	21559
GAP	(DOE) GENERAL AND PRACTICAL INFO.	178	132	2685
EMI	(EMIC) ENV. MUTAGENS INFO.	168	165	11110
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	143	197	7632
ESI	(EIC) ENV. SCIENCE INDEX	127	226	717
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	89	100	936
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	82	87	1990
EIA	(EIC) ENERGY INFO. ABSTRACTS	56	99	1260
WRE	(WRSIC) WATER RESOURCE RESEARCH	55	130	924
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	53	105	6
EIS	(TIRC) EPIDEMIOLOGY INFO. SYSTEM	33	49	28
PRD	(TIC/NRC) POWER REACTOR DOCKETS	26	38	795
TUL	(U. TULSA) TULSA DATA BASE	22	215	668
RSI	(RSIC) RADIATION SHIELDING INFO.	21	9	1813
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	20	28	248
API	(API) AMER. PETROLEUM DATA BASE	19	55	331
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	19	31	23
CIM	(DOE) CENTRAL INVENTORY OF MODELS	17	23	-
NER	(EIC) NATIONAL ENERGY REFERRAL	17	36	93
NRC	(LC) NATIONAL REFERRAL CENTER	16	25	191
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	11	4	-
NBI	(NBIC) NATL BIOMONITORING INV.	9	4	-
RSC	(RSIC) RADIATION SHIELDING CODES	4	-	-

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PROGRAM TITLE: PWR Blowdown Heat Transfer-Separate Effects

PROGRAM MANAGER: J. D. White

ACTIVITY NUMBER: ORNL #41 89 55 10 3 (189 #B0125)/NRC #60 19 10 01

TECHNICAL HIGHLIGHTS

Task 1: FCTF Testing — Testing has been completed of a fuel rod simulator (FRS) in the FCTF that simulates the upcoming bundle uncovery/ recovery testing in the THTF. Initially, steady-state and power drop tests were conducted that allow characterization of internal heat transfer physical properties of the FRS. Subsequently the rod was heated in air at high surface temperatures for one hour on three separate occasions (surface temperatures reached were 1000, 1300, and 1700°F). Characterization testing was repeated after each hour of heating. Then two boil-off tests were conducted; with no flow in the test section the rod was boiled dry and subsequently reflooded. (Approximately two-thirds or more of the rod appeared to be dry.) Characterization testing was repeated after the second boil-off test. Finally, a blowdown was conducted from a pressure of 2250 psig with a rod power of 144 kW. Examination of sheath and center thermocouple responses revealed no evidence of any degradation of rod integrity nor of any variations in thermal response. The rod did exhibit a noticeable "S" bend when it was temporarily removed from the FCTF after the air heatup tests, but this did not seem to affect the rod's subsequent performance when reinstalled in the facility.

Task 2: Analysis — Electric Pin Analysis. An effort was initiated during the previous month to develop a computer program which could compare heat transfer coefficients determined "experimentally" in the THTF and heat transfer coefficients predicted from existing correlations using the "experimental" local fluid conditions. It was intended to have this code operational by January 1, 1980. To date, code programming has been completed and several subsections have been debugged. Final debugging and checkout will be completed during December.

A HEATING5 R-Z model of a bundle 3 FRS from the THTF test section outlet centerline to the upper extent of the air duct chimney has been developed and debugged. A test case was run similar to the one made by

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K. H. Luk for H. R. Payne in November 1978. Significant discrepancies were noted in the results. These discrepancies were studied and the model was verified. It is estimated that the FRS in the air-cooled section above the upper seal plate will be ∿200°F cooler than predicted by Luk and the upper "O" rings will be 100-175°F cooler than previously predicted.

Development and study of the computational techniques for the one-dimensional solution of the inverse problem with respect to bundle 3 FRSs has been completed. The technique to be used will be a modified formulation of Beck's second method.

Documentation of the multi-dimensional inverse code (ORMDIN) was started. A brief paper describing the technique with applications to FCTF data was prepared for the 1980 ASME convention in San Francisco.

Development of the "transfer function" for bundle 3 multi-dimensional inverse calculations is continuing. The "transfer function" correlates the multi-dimensional effects observed in the FRS with the one-dimensional results of the inverse problem; thus, a user could determine via the "transfer function" an approximation to the multi-dimensional heat transfer in the FRS from a less costly one-dimensional computation. The data base needed for development of the "transfer function" has been completed and development of the regression codes has started.

Thermal-Hydraulic Analysis. Work during the month of November was centered on the pretest analysis for the Bundle Uncovery Tests (# 3.02.10A) and the Upflow Film Boiling Test (# 3.03.6B). Pretest analysis for the Bundle Uncovery Tests has been completed and a posttest analytical plan has been submitted to the NRC. Work on a THTF MOD 2 test section model using RELAP4 M6 is continuing. The test section model will be used in preliminary local fluid condition calculations.

RELAP4 M5 U2 and RELAP4 M6 U4 THTF loop model runs have been made for tests 177, 166S, and 167R. These runs will be used in ascertaining the predictive capabilities of the two codes. A comparison of hydraulic predictions for test 177 has been completed and a heat transfer comparison has begun.

Nuclear Pin Simulation Analysis. Development of PINSIM MOD2 continues: the transient calculational routines have been debugged, the local fluid condition interface is operational and has been verified, and output enhancement is underway. Development of PINSIM MOD2 problem models to facilitate the posttest analysis of THTF Test Series 4 has begun.

An investigation of alternate power programming calculational techniques has been initiated. This effort is intended to contrast our current back-calculational technique with other available methods, thereby illustrating relative advantages and disadvantages of each method.

Compilation of the report on the posttest nuclear pin simulation analysis of THTF Test 105 is nearly complete; the report should be finished early in December.

Data Management. Debugging of the THTF MOD2 (Bundle 3) Data Reduction Code (DACREP) is in its final stages. An addition to the code will be made in the future to process test section fill test data.

Data reduction has begun for Tests 178 through 181. The new Data Report Plot Code, which is in its final stages of development, will be used for these and all future tests.

Data reduction has been completed for all FCTF tests to date.

Task 3: THTF Operations — Isothermal blowdown, Test # 3.01.5A was completed. Preparations are being completed for power drop testing (such tests allow characterization of key material properties within the fuel rod simulators). These preparations include dc power connections to the bundle 3 heater rods and installation of a magnetic field suppression system. Work is in progress on modifications for the quasi-steady-state bundle uncovery tests, Test # 3.03.10A. Scheduled test dates have slipped two days due to a rupture in the Grayloc seals located at the test section bottom. New seals are now being installed.

Task 4: Two-Phase Instrument Development — Acquisition and calibration of instruments for the bundle uncovery tests are underway. The drag disk in the 2-in. outlet spool piece is being calibrated in the AIRS steam-water test facility. A full-flow four-bladed target is being used. Low range differential pressure cells have been acquired for test section water inventory measurements during bundle uncovery.

Work is now in progress to complete mass flow calculations for Tests 178-181 using the mass flow code AMICON. Work is also being carried out on estimating transient uncertainties for the spool piece mass flow instruments for eventual inclusion in the mass flow code AMICON.

Preliminary work has also begun on interpreting data from INEL's liquid level probe for determining collapsed liquid level during the bundle uncovery tests.

PROGRAM TITLE: Safety Related Operator Actions

PROGRAM MANAGER: P. M. Haas

ACTIVITY NUMBER: ORNL #40 10 01 06 01 (189 #B0421-8)NRC #60 19 11 01

TECHNICAL HIGHLIGHTS

The results of the simulator survey conducted by Memphis State University Center for Nuclear Studies (MSU/CNS) were presented to NRC during a meeting November 14 at Bethesda. The revised draft of the written report has not yet been completed by MSU/CNS. In addition to a summary of the current capabilities of simulators, and their use in the nuclear industry, and recommendations for more effective use of simulators, results include a proposed procedure for systematically determining which malfunctions should be simulated as part of training for emergency/abnormal events. The procedure could be used to evaluate existing simulators on a site-by-site basis and to help evaluate or develop standards or guidelines for simulator design.

Planning for field data collection and simulator experiments continued. Several candidates for events/operator actions to be included as part of the correlation development were identified, and selective searches of NRC docket information on those events were initiated. The events selected have to include operator action which is likely to be measureable in both the field situation (real events which have occurred) and on the simulator. In addition, there should be enough occurrences in the historic (field) data base to provide a reasonable statistical basis for estimated response times or error rates.

PROGRAM TITLE: Zircaloy Fuel Cladding Creepdown Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: ORNL # 41 89 55 11 7 (189 # B0124)/NRC # 60 19 11 04 1

TECHNICAL HIGHLIGHTS

The HOBBIE-6 test was completed successfully after an extended test time. The test was run as scheduled at 343°C and 17.2 MPa external pressure. However, instead of the test being terminated, it was depressurized in 1.4 MPa steps to zero pressure differential across the wall and then repressurized internally to 6.9 MPa in the same step increments. The resultant elastic strains and the reverse creep strains during the half cycle of reactor operation under internal pressurization were recorded by the eddy-current probes. Data have not yet been received from ECN-Petten.

The test procedure outlined above is similar to the ones planned for HOBBIE-7, -8, and -9 and will give needed information about cladding creep as a function of internal fission gas pressure buildup during reactor operation.

HOBBIE-7 was installed in the HFR reactor at ECN-Petten on November 27, 1979, and is running.

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