INTERIM REPORT

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November 1978 Monthly Highlights Office of Nuclear Regulatory Research Programs Division of Reactor Safety Research 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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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
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ABSTRACT

Highlights of technical progress during November 1978 are presented for fifteen 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 10 1 (189 #B0119)/NRC #60 19 10 5

TECHNICAL HIGHLIGHTS

Task 1: Program Administration — S. K. Iskander presented a paper, "Application of a Dynamic Method of Analysis for Crack Arrest to a Thermal Shock Experiment," at the ASTM Symposium on Crack Arrest Methodology and Applications on November 7. R. D. Cheverton attended this meeting.

G. D. Whitman presented a paper, "Structural Integrity of Weld Repaired Pressure Vessels," on November 8, at the 6th Water Reactor Safety Research Information Meeting. R. H. Bryan attended this meeting on November 8 and 9.

P. P. Holz attended the Welding Symposium of the Materials and Processing Congress in Philadelphia on November 7 and 8. He attended the 6th Water Reactor Safety Research Information Meeting on November 9, and visited the welding facilities of the David Taylor Naval Ship Research and Development Center in Washington D.C., on November 10.

G. D. Whitman attended a meeting of the International Cooperative Group on Cyclic Crack Growth Rate at Bethesda, MD, on November 10.

Mr. Alpo Ranta-Maunus of the Finish Institute of Radiation Protection visited ORNL on November 10.

On November 13, Dr. K. Shibata of the Japan Atomic Energy Research Institute visited ORNL to exchange information on fracture mechanics research and development.

On November 22, G. D. Whitman traveled to NRC offices in Silver Spring, MD, to review plans for the testing of specimens from the 4T irradiations projects and the design of a fourth irradiation experiment.

R. G. Berggren, J. G. Merkle, and G. D. Whitman attended a Vessel Integrity Review Group Meeting at NRC offices in Silver Spring, MD, on November 30.

Two technical reports were issued in October 1978 but not mentioned at that time: *Crack Extension and Arrest Tests of Axially Flawed Steel Model Pressure Vessels*, NUREG/CR-0126, ORNL/NUREG/TM-196, by Smith, Holz, and Stelzman; and *Test of 6-in.-Thick Pressure Vessels. Series 3. Intermediate Test Vessel V-7B*, NUREG/CR-0309, ORNL/NUREG-38, by Bryan et al.

Task 2: Fracture Mechanics and Analysis - Estimates of stable crack growth at maximum load were made for two Charpy thickness compact specimens of A537 steel, using only the load versus front-face clip gage displacement records. The results agree well with R-curve data based on unloading compliance measurements for A533 steel, indicating, as before, the possibility of making reasonable and useful estimates of stable crack growth without auxiliary crack length measurements.

Task 4: Irradiation Effects — Fast neutron fluences and irradiation temperature ranges partitioned between the "forward" and "reverse" periods have been determined for all specimens in the three capsules of the second 4T-CTS irradiation experiment.

Correction of data from the three-point bend tests for lateral load effects was completed for PCC_V specimens from this irradiation. Corrected toughness results from the tests using the sharp-edge anvil are now in good agreement with results from tests using the standard anvil.

Analyses of fast neutron fluxes in the third 4T-CTS irradiation experiment are under way and should be completed in December. The hot cell Charpy tester has been recalibrated.

Planning of the fourth HSST irradiation experiment is continuing. We are designing each irradiation capsule to contain 60 1T compact specimens and 96 Charpy-size (Charpy-V and tensile) specimens. The materials, number of specimens and test plan are designed to yield statistically meaningful results.

Task 5: Simulated Service Tests — The block of material containing the V-8 flaw has been sectioned at several depths from the outside surface to determine the profile of the crack relative to the fabrication weld heat-affected zone. Preliminary SEM examination confirms, at one cross section, the estimated depth of the first pop-in.

Work is progressing somewhat slowly on reporting the pre- and post-test fracture analyses of vessel V-8, due to the necessity for checking the original dimensional measurements and calculations.

We have tested the two 2T compact specimens at 21°C (70°F) which were precracked by H_2 embrittlement of an EB weld. These results are currently being analyzed.

The two mating fatigue precracked 2T compact specimens have been precracked and are ready to be tested.

Task 6: Thermal Shock — Design of the support structure for major LN_2-TSTF components was completed and submitted to Y-12 Engineering for approval. Nearly all test facility components were in the process of being fabricated, and machining of TSC-1 (the test cylinder for thermal shock experiment TSE-5) and the thermocouple thimbles was completed.

In preparation for conducting a three-dimensional fracture mechanics analysis of TSE-2 (semielliptical flaw) an effort was initiated to convert from the present ADINA-based finite-element K_I calculational technique to a recently improved version of the same basic code. The advantage will be a significant reduction in computer time. Also, progress was made in the development of the mesh for the three-dimensional TSE-2 problem.

Test specimens are being machined from the tempered segments of the "as-quenched" TSC-1 prolongation. We have received the specimens from the 704 and 677°C (1300 and 1250°F) tempers. The tensile testing for both tempers and the fatigue precracking of the fracture toughness specimens have been completed. Testing of the precracked Charpy-V specimen will begin shortly.

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 04 1

TECHNICAL HIGHLIGHTS:

Activities during November included analyses of the results from the first two tests in the High Temperature Series, HT-1 and HT-2, and preparations for the next experiment. In addition, new instrumentation was installed for temperature recording and for automatic monitoring of ^{85}Kr release.

Test HT-2 was conducted at 1450°C for seven minutes in a flowing steam-helium atmosphere; the cladding had been expanded and defected prior to the release test to simulate a high temperature rupture. Because higher release fractions were experienced in this test (see table below), the components of the experimental apparatus (especially the furnace tube and the thermal gradient tube) were much more radioactive. Consequently, it was necessary to develop equipment and procedures for handling and leaching the apparatus components remotely. Also, to facilitate transfer and direct analysis by gamma spectrometry of apparatus components, a special lead-shielded carrier was fabricated.

Test No.	Temperature (°C)	Time (min)	Release as % of total inventory ^{85}Kr	Release as % of total inventory ^{134}Cs
HBU-11[a]	1200	10	1.3[b]	0.011
HT-1	1300	10	1.07 + ∿0.5[c]	0.112
HT-2	1450	7	5.0 + ∿1.0[c]	4.82

[a]Total test time was 27 minutes; release values adjusted for a 10 minute period.

[b]Includes ^{85}Kr released when this segment was used in a previous test at 900°C, test HBU-7.

[c]Estimated release during cladding expansion.

As shown in the table, 4.82% of the total ^{134}Cs inventory and 6% of the total ^{85}Kr inventory were released during test HT-2. (Analyses for ^{129}I release are still in progress.) About 95% of the released ^{134}Cs deposited in the furnace tube, and an additional 4.2% was collected in the thermal gradient tube; this distribution is similar to that observed in test HT-1.

The present data suggest that a temperature has been attained at which cesium release is influenced by a mechanism other than normal diffusion from either the gap space or the UO_2 matrix. Estimates of release by diffusion from the matrix indicate cesium release in test HT-2 to be only 2.2 times greater than that in HT-1, and only 4.7 times greater than that in HBU-11.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: ORNL #40 89 55 10 6 (189 #BC120)/NRC #60 19 10 04 1

TECHNICAL HIGHLIGHTS

R. H. Chapman and D. O. Hobson attended the Sixth Water Reactor Safety Research Meeting in Gaithersburg, MD, November 6-9; J. L. Crowley attended the meeting on November 6 and 7. A detailed presentation was made at the Workshop on Multirod Bundle Tests on the results obtained from the three multirod bundle tests conducted to date.

Dr. S. Kowasaki of the Japan Atomic Energy Research Institute visited ORNL on November 10 for discussions on Zircaloy cladding deformation in multirod test arrays. The inability to communicate effectively prevented a clear understanding of techniques and results of the JAERI test program.

Messrs. A. Feige, F. Erbacher, and K. Wiehr of the Karlsruhe Nuclear Research Center and G. Cheliotis of the Kraftwerk Union Research Laboratory (Erlangen) visited ORNL on November 16 and 17 for detailed discussions on recent MRBT and REBEKA bundle tests and on future test plans. Results obtained to date in these two programs are in good agreement, considering the differences in experimental techniques.

Fabrication of a flow shroud for the B-3 bundle was completed, and flow tests were conducted on both the B-3 and the reference bundle. Although the data have not been analysed in detail, preliminary analysis indicates the overall (axial) pressure loss of the B-3 bundle was about 75% greater than the reference bundle. (For comparison the B-1 and B-2 pressure losses were 50-60% greater.) This is in substantial agreement with visual observations that deformation in the B-3 array was significantly greater than in the two earlier tests.

The B-3 array is being cast in an epoxy matrix in preparation for sectioning to obtain strain data.

Planning and assembly of the necessary test equipment for demonstrating acceptability of the separate power supply for the 4 × 4 bundle shroud continued. These demonstration tests, now scheduled for January, are necessary to provide operating experience and amplifier gain settings for the feedback control system. The tests will also provide important data for evaluating techniques for using a limited number of external thermocouples in future bundle tests.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: W. B. Cottrell

ACTIVITY NUMBER: ORNL #40 89 55 10 4 (189 #B0126)/NRC #60 19 01 02

TECHNICAL HIGHLIGHTS

During the month of November, the staff of the Nuclear Safety Information Center (a) processed 1128 documents, (b) responded to 59 inquiries (of which 43 involved the technical staff and 9 were for commercial users), and (c) made 12 computer searches. The RECON system, which now has over 200 remote terminals, reports that the NSIC data file was accessed 168 times during the previous month (see attached Table 1). The final cumulative bibliography for the ACRS is being processed. During the past month, the NSIC staff received seven visitors, participated in seven meetings, and has work under way on two evaluations.

Two NSIC reports were issued: *Bibliography of Reports on Research Sponsored by the NRC Office of Nuclear Regulatory Research, January–June 1978* (ORNL/NUREG/NSIC-155) and *Reports Distributed Under the NRC Reactor Safety Research Foreign Technical Exchange Program, Vol. V January–June 1978* (ORNL/NUREG/NSIC-156). Three other NSIC reports are in reproduction: *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; and *Index of Microfiched Foreign Reports Distributed Under the NRC Light-Water Reactor Safety Research Foreign Technical Exchange Program 1975–1977* (ORNL/NUREG/NSIC-154). One other report is in composition: *Bibliography on Common Cause – Common Mode Failures* (ORNL/NUREG/NSIC-148). Several other NSIC reports are in various stages of preparation, including *Annotated Bibliography on Stress Analysis and Analytical Techniques Relative to Nuclear Steam Supply System Design* (ORNL/NUREG/NSIC-157) and *Radiation in Perspective* (ORNL/NUREG/NSIC-161).

During the month of October, we received 41 foreign documents (1 French, 6 Japanese, 24 UKAEA and 10 German). All have been submitted for microfiche distribution. The foreign language documents were reviewed for translation (two letters of November 28, 1978, to G. L. Bennett). We would note that 18 of the 24 UK documents were restricted distribution. During the month we distributed 15 reports on NRC-sponsored safety projects to foreign recipients under the Light-Water Reactor Safety Technical Exchange Agreements and 2 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 translation subcontract was recently awarded to a different contractor (who submitted the low bid) and we are still behind in our abstract translations into both French and German.

NSIC's selective dissemination of information (SDI) is available to paying users (as well as exempt users). During the month of November we added 7 exempt and 2 paid users, bringing the SDI service to a total of 390 users.

A data communication translation problem, which had prevented input of document descriptions into the NSIC Computer File starting on October 23, was resolved by November 1. The problem arose during an equipment exchange for diagnostic purposes at the Computer Center but our local problems were eventually attributed to the failure of several printed circuits in our controller.

All technical articles for *Nuclear Safety* 20(2) were completed and mailed to NRC, DOE, and TIC on or before November 22nd. The "current events" material (covering events which occurred during September and October) for *Nuclear Safety* 20(1) was completed by mid November (except for "Operating U.S. Power Reactors" which is dependent upon the 'Grey Book') and sent to TIC. All technical articles for *Nuclear Safety* 20(3) have been received, submitted to peer review, and are in various stages of preparation.

RECON DATA BASE ACTIVITY FROM 10-01-78 TO 11-01-78 (22 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	840	1695	28342
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	216	436	16324
API	(API) AMER. PETROLEUM DATA BASE	19	69	0
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	435	1897	47542
GAP	(DOE) GENERAL AND PRACTICAL INFO.	178	183	6905
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	31	21	183
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	26	58	86
EMI	(EMIC) ENV. MUTAGENS INFO.	245	943	9550
EDB	(TIC) DOE ENERGY DATABASE	2788	5257	149668
ERD	(EISO) ENERGY R&D PROJECTS	80	150	827
NBI	(NBIC) NATL BIOMONITORING INV.	18	14	0
DBS	(LLL) DATA BASE SURVEY	27	36	0
ESI	(EIC) ENV. SCIENCE INDEX	102	252	2270
EIX	(EI) ENGINEERING INDEX	317	891	5548
MOS	(LLL) ENG. & ENV. DB MODELING SURVEY	24	25	14
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	168	257	11080
WRE	(WRSIC) WATER RESOURCE RESEARCH	44	111	909
NRC	(LC) NATIONAL REFERRAL CENTER	43	54	272
NER	(EIC) NATIONAL ENERGY REFERRAL	28	43	11
RSI	(RSIC) RADIATION SHIELDING INFO.	12	13	603
EIA	(EIC) ENERGY INFO. ABSTRACTS	91	144	759
RSC	(RSIC) RADIATION SHIELDING CODES	9	3	7
ESR	(DOE) ENERGY ENVIRONMENT & SAFETY	60	77	870
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	104	207	1386
TUL	(U.TULSA) TULSA DATA BASE	77	150	905
EDA	(DOE/EIA) ENERGY DATA	21	55	1

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: During this month, the first power tests were started on a bundle 3 fuel rod simulator (FRS) which was fabricated at ORNL. The procedures for the tests follow those which were established for FRS's of previous bundles. Thus far the rod has been subjected to three power drop tests and two blowdowns. The loop conditions during the tests were typical of those planned for the THTF. All tests for this rod, designated rod No. 006, should be completed the first week in December.

Although the tests are not complete nor the results final, some observations may be noted. First, 5 of the 16 thermocouples failed when the FRS was first heated. Second, the heating element resistance changed with temperature according to the expected pattern. Third, the leakage current between the heating element and the central thermocouple sheaths is acceptably low. Last, the mechanical aspects appear satisfactory.

Task 2: Analysis. Nuclear Pin Simulation. The nuclear pin material properties module (MATPRO-09) used by PINSIM has been replaced by MATPRO-10, an updated version developed at INEL.

Development has begun on subroutines required to give PINSIM the capability of bounding pin heat transfer calculations on both nuclear and electric pin models by local fluid conditions calculated by a system hydraulics code such as RELAP or TRAC. These conditions would be supplied to PINSIM on a coolant conditions data tape which will be identical in both content and format to the coolant conditions data tape read by FRAP-T4. PINSIM will retain the capability of calculating local fluid conditions between user-supplied inlet and outlet plenum conditions.

A study of predicted time to critical heat flux (CHF) and post-DNB heat transfer using PINSIM-MOD1 has begun. The study will contrast the predicted behavior of typical nuclear and electric pin models in similar hydraulic environments. A portion of the study will use the MOD1 CHF correlation recently developed by Keith Condie of EG&G.

Thermal Hydraulic Analysis. Preliminary analysis with RELAP4 indicates that THTF MOD2 should not produce choking due to the new downcomer design. Analysis in this area is continuing. Evaluation of the predictive capability of RELAP4/MOD6 is continuing. It appears that MOD6 overpredicts upper core rod surface temperatures for THTF test 105 as did MOD5. Preliminary analysis of THTF test 156 indicates that use of an orifice in the pressurizer line significantly affects rod surface temperatures. Analysis of this test is continuing. Evaluation of the optimal bounding method for COBRA in local fluid condition determination is continuing. The capability to allow smoothing and filtering of experimental data used in local fluid condition determination is under development by CSD support. Arrangements were made with LASL to obtain TRAC-PIA in December. The version to be received will be a CDC version which will require conversion to IBM format.

Data Management. Work has been completed on the Data Report for THTF test 164R.

Code development has begun for data reduction of data acquired by the PDP-11 of the THTF. The format of the raw data tape to be processed by the analysis group has been defined.

Electric Pin Simulation. Sensitivity studies of ORINC are continuing. Sensitivity studies of ORNL's version of GE's HCODE style inverse method and a J. V. Beck inverse method are continuing.

Evaluation of prototypical bundle 3 heaters in the FCTF has started. The evaluation of the heaters includes determination of the contact resistance between the sheath thermocouples and the outer sheath and determination of the effective thermal properties of the insulators.

Studies to develop a "transfer function" for BDHT fuel pin simulators which will correlate the flux through the thermocouple with the actual mean flux are continuing.

Task 3: THTF Operations. The system remained shutdown during the report period to work on installation of the refurbished MG set (about 10% complete) and the instrumented spool pieces (about 80% complete). The turbine meters and drag disks for these spool pieces have been received.

Design for the addition of five instrumented spool pieces (one each in the three main heat exchangers, one in the pump bypass, and one in the pressurizer exit) is approximately 90% complete. All signal conditioning electronics for these additional spool pieces has been received and the installation has begun. Completion of the hardware installation is expected the first part of January.

The upgraded computer-based data acquisition system (DAS) has been installed and has been operational in the new auxiliary control room since August 29, 1978. Design for the interconnection wiring between the DAS and field instruments is complete. All cables and interconnecting racks have been fabricated, and installation of the cables has begun.

The package of software necessary for loop operation and the recording of data is approximately 90% complete. The software design has been "frozen" for the shakedown tests and the first blowdown tests. At that point, the package will be evaluated by the operating and analytical team and any recommended changes or enhancements will be incorporated.

Task 4: An uncertainty analysis of the two-phase mass flow calculation was begun and a method of analysis has been defined and preliminary blowdown uncertainties have been obtained. Current analysis involves all three homogeneous models and preliminary results from an error propagation analysis of these models indicate steady-state mass flux uncertainties of less than 3% of the steady-state values. The data analyzed thus far has been for test 177 during the steady flow, preblowdown full-power portion of the test. A technique to extend this analysis into the transient is undergoing testing.

Task 5: Bundle 3. An analysis reported by K. H. Luk indicates that we can use Viton O-rings in the upper and lower seal plates.

Thermocouple rods. The design is over 90% complete. A procedure for brazing the thermocouples into the pressure sealing bulkhead in the rod has been developed and documented. Final tests of this procedure are being completed; no real problems are anticipated. Fabrication will be initiated in early December.

Impedance probes. Development of the fabrication methods and procedures for metal-cermet joints in the assembly is being conducted by the Metals and Ceramics Division and the Instrumentation and Controls Division. If the stainless steel bellows are received as scheduled on December 13, all critical material purchases will be complete. The cermet insulators are being fabricated and the first of four insulators is being machined. Final shop fabrication will begin January 2, 1979.

Task 6: Bundle 3 Fuel Rod Simulators. Three additional full-length fuel rod simulators (FRS) prototype units for bundle 3 have been fabricated by ORNL in the FRS Development Laboratory. The first of these units (FRS #6) is undergoing tests and the other two units (FRS #7 and 8) are available for similar tests. All three units have full complements of thermocouples (16 each).

Following receipt of the small diameter tube 2-mm-OD (0.078-in.) by 1.7-mm-ID (0.068-in.) for the center thermocouples, an additional prototype unit (FRS #9) has been fabricated and will be available for operational tests by December 4, 1978. The reduced wall thickness of this tubing will reduce the thermocouple elongation significantly (by approximately 2.5%) which might eliminate an apparent brittle fracture of some of the inner thermocouples. If stress relieving or annealing is required, it can be provided either on the initial material or as in-process fabrication step.

Problems in the shipment of acceptable sheath-type 0.5-mm-OD (0.020-in.) type K thermocouples have been overcome by placing an emergency purchase order (for fabrication of the measuring junction area only) with a small thermocouple vendor. Present schedules call for deliveries to be made to meet all fabrication schedules.

At present, fabrication of five FRS units per week is scheduled and there is no apparent problem with this schedule for the first twenty FRS units. These will be of a revised type having only six thermocouples at 60° spacing in the outer sheath. Three of these six will be at 120° spacing at the mid-plane point of the FRS heated section and one each will be at three other locations at 60-cm (23.5-in.) spacings. A center thermocouple will be at each axial location.

Boron nitride (BN) back-filled thermocouples 0.5-mm-OD (0.020-in.) are being developed to replace the magnesium oxide filled thermocouples presently being used. This will be for the thermocouples on the inside surface of the FRS sheath only. BN powder will be located only over the last 3 mm (1/8 in.) of the unit. Initial evaluation of four of these thermocouples show time responses for insulated junctions to be even faster than for standard grounded thermocouples of the same size. It is felt that the time response ($\sim$21 milliseconds) is an excellent indication of the high density and high thermal conductivity of the boron-nitride insulation in the vicinity of the measuring junction. Proper design of this junction area will result in significant decreases in the perturbations in the thermal profile of the FRS especially during transient tests.

Tooling and fixturing are on order to allow for fabrication of FRS units with twelve thermocouples located at 30° inside the FRS sheath. This equipment and the fabricated parts will be received in time to permit fabrication of the entire sixty units on schedule by the end of February 1979. Other revisions to thermocouple layouts such as placing the full complement within a few inches of a grid spacer are being accommodated also without affecting the schedule.

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 10 01

TECHNICAL HIGHLIGHTS

FAST/CRI-III:

Two more of the scheduled FAST/argon tests were completed at standard temperature and pressure conditions (FAST 6 and 7). Following these, the FAST vessel was prepared for the next four tests to be performed at elevated pressure (2.02 MPa, or 20 atm).

The CRI-III vessel was equipped with the fast-response pressure transducer planned for use in the FAST tests. This will be used in CRI-III as both a test of the transducer and to record the shock pressures produced in the FAST vaporizer underwater tests.

NSPP:

A sodium oxide aerosol test, No. 107, was performed during this period. The sodium oxide aerosol was produced by injecting 10 kg of heated sodium over a period of 6.5 minutes through a single spray nozzle which was centrally located at an elevation of 13 ft from the bottom of the vessel; the direction of spraying was downward. A peak pressure rise of 6.3 psi was produced by the sodium spray fire. Peak vessel atmosphere temperatures ranged from 207°C (at an elevation of 4.4 ft), to 370°C (at an elevation of 9 ft), and to 416°C (at an elevation of 13.6 ft). At five minutes after termination of the sodium spray, the pressure had decreased 5.9 psi from the peak pressure of 6.3 psi, and the vessel atmosphere temperatures had decreased into the 75-100°C range. Analytical data from this test will not be available until the next reporting period.

CRI-II:

The initial test of the 180 cm Stöber-LASL spiral aerosol centrifuge, obtained on loan from the Los Alamos Scientific Laboratory, was successfully conducted at the CRI-II facility with the aid of a visiting consultant from the lending institution. Starting with a maximum initial U_3O_8 aerosol concentration of about 40 g/m^3, generated by the metal-oxygen plasma torch, three analyses were performed over a time interval of about four hours. The results, by visual observations of the collection foils, showed that the size distribution did not change much over this interval confirming our earlier cascade impactor data.

Incomplete analytical results for the initial centrifuge sample, taken at maximum concentration, give an apparent agglomerate AMMD size almost a factor of two greater than that obtained from the Andersen inertial impactor. This is essentially in agreement with our usual "Stokes" diameters calculated from fallout rates. These have consistently exceeded the measured impactor diameters. In order to optimize the test of the spiral centrifuge with mixed sodium oxide-uranium oxide, the original inlet and collection surfaces have been replaced with stainless steel parts which also aid in the analytical process.

Extra length cables for the NSPP flame spray aerosol generator torch have been received. These will enable us to connect the present CRI-II power supply and the NSPP generator to the CRI-II vessel for a few trail aerosol generation tests. We expect that this torch will almost double the burning rate of the CRI-II torch and should give initial U_3O_8 concentrations above 60 g/m^3.

ANALYTICAL:

The CSMP code has been modified to ensure an improved material balance during the bubble rise process. As a consequence, the program runs much more rapidly so that cases which formerly ran extremely slowly, or would produce no useful results at all, now converge quickly. Comparisons have been made of two bases of estimating radiation heat losses. Checks of possible error due to use of the ideal gas law as the equation of state for UO_2 show that using more complicated relationships are not justified at this point. Work on an interim report describing the work done so far on FAST experiment modeling continues.

We have converted our version of the HAARM-3 code to double precision arithmetic and have increased the time steps in the stirred settling model to allow its use without excessive computer time.

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

Further survivability tests have been conducted on the ceramic-to-metal seal used in the PKL guide tube probes. A typical seal was mounted in a short section of tube and subjected to an external pressure (in helium) of 40 bars at a temperature of 600°C. The inside of the tube remained at atmospheric pressure. The seal suffered no observable damage except that the leakage increased in proportion to the pressure differential as would be expected since the seal had a slight initial leak.

A series of corrosion tests are being planned for one of the PKL prototype guide tube probes. These tests are to verify the operability of the probes in a carbon steel environment which has been subjected to the corrosion prevention treatment being proposed for the Upper Plenum Test Facility (UPTF). The required chemicals and inhibitors have been obtained. The probe will be installed in a carbon steel vessel which will then be given the proposed UPTF treatment and operated under simulated UPTF conditions for a period of time.

Design concepts for the PKL upper plenum instruments and the core wall film probes are being finalized. A prototype string probe using high temperature materials is expected to be ready for air water testing very soon. Fabrication of the first high temperature film probe module prototype will begin in December. The film probe calibration chamber is essentially completed. Piping and installation will be completed in January 79.

Materials development related to the fabrication of a band probe is making some progress. The required cermet cylinders manufactured last month are being machined for incorporation into a prototype. A bellows type thermal expansion joint has been designed and is being manufactured by a vendor.

Important meetings attended include the 2D/3D Coordination Meeting at Silver Spring, MD on November 10 through November 14. A preliminary design meeting for Slab Core I instrumentation between ORNL and JAERI personnel was held at ORNL. Design concepts and constraints were discussed and progress was made toward achieving suitable sensor configurations. Conceptual designs will now be prepared and reviewed at a meeting in late January in Japan.

During this month, as in the month of October, considerable effort was devoted to the preparation of cost estimates for various instrument configurations and quantities. The scope of instrument supply by ORNL for the various test facilities is still under discussion.

Alternative techniques of signal processing for transit time measurement are being investigated. The tentative conclusion is that frequency-domain (Fast Fourier Transform, FFT) processing has several advantages over the time-domain (adaptive cross correlation) method. The chief advantages of FFT are a reduced sampling rate and a predictable statistical error band.

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 BLAST and ORTAP for the Fort St. Vrain Reactor: The routine used in both BLAST and ORTAP to perform an initial steady state search for the steam generator helium and water side nodal pressures, enthalpies and temperatures and for tube nodal temperatures was modified to achieve convergence with less accurate initial guesses at nodal conditions.

Development of the FLODIS Code: The iteration loops that had been consuming the largest amount of computer time were rewritten to improve iteration efficiency. This improvement showed a significant decrease in computation time for a sample transient.

Miscellaneous: A paper entitled "Investigations of Postulated Accident Sequences for the Fort St. Vrain HTGR" was presented at the second JAEB-USNRC HTGR Safety Seminar held at Tokyo-Fuji Japan (Nov. 22-26). The gas cooled reactor related experimental facilities at the National Institute of Metals (Tsukuba), MHI (Nagasaki and Kobe), IHI (Yokahama) and Engineering Research Associates (Tokyo) were visited and goals and operations were discussed with Japanese scientists and engineers.

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: Moment Loading Parameter Studies for Isolated RPV Nozzles — During the past month the first draft of the moment loading parameter study report was completed. The report includes a description of the CØRTES finite-element computer code analysis, results of comparison studies of three experimental models for validation purposes, and summary tables of maximum stresses and displacements. Complete listings of the calculated stresses for each of the six moment loadings for all 25 isolated nozzle models are printed on microfiche for inclusion on the back cover of the report.

Task 2: Stress Indices and Flexibility Factors for RPV Nozzles and Small Branch Connections — Tabulated summaries of the maximum stresses, maximum stress intensities, and maximum displacements from the 25 model moment loading parameter study were transmitted to E. C. Rodabaugh at Battelle-Columbus Laboratories for use in his development of stress index and flexibility factor formulas. These new formulas will be compared with those currently in the Code as a basis for recommended changes.

Task 3: Documentation and Release of the MULT-NØZZLE Computer Program — In-house review of the subcontractor's draft report documenting the development of extensions to the MULT-NØZZLE program for analyzing RPVs with two or three closely spaced nozzles under force and moment loadings has been completed. The report is currently being prepared for publication.

Task 4: Parameter Studies and Proposed Code Rules for Closely Spaced Nozzles — During November, further checkout of the MULT-NØZZLE program and its postprocessors was conducted at the UCCND Computer Sciences Division to familiarize personnel with operation of the program. In addition, two models with D/T = 20 and d/D = 0.16 were analyzed for internal pressure loading. One of the models had a single reinforced nozzle while the other had two reinforced nozzles in a longitudinal plane of the vessel spaced as closely as permitted under current Code rules. The results indicated a considerable amount of interaction between the two closely spaced nozzles.

PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: R. S. Booth

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

TECHNICAL HIGHLIGHTS

There is no 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

A series of experimental measurements has been performed to verify the calculated results for the 7/8 in. diam Inconel 600 steam generator tubes. The same set of properties was not run in both cases. (It is easy to add thickness to a tube sample on the computer, but is hard to do experimentally.) The two sets of properties were close enough, however, to get a good indication. The only two properties that we will examine in detail are wall thickness and defect size. They are shown in the table below:

Table 1. Summary of Experimental and Calculated Results for Measurement of Properties of 7/8 in. Diameter Inconel Tubing

Property	Calculated Wall Thickness (in.)	Measured Wall Thickness (in.)	Calculated Depth of 0.125 diam Hole (in.)	Measured Depth of 0.125 diam Hole (in.)
Range	0.025–0.055	0.035–0.051	0.000–0.0153	0.000–0.021
Fit (RMS Variation)	0.0002	0.0002	0.0049	0.0051
Drift (RMS Variation)	0.0003	0.0002	0.0034	0.0029

While the agreement is very good between the calculated and experimental property determinations, there are some additional improvements that must be made.

The defect size error is much greater for the OD defects than the ID defects. The calculated readings and experimental measurements need to be re-run with more weight given to the OD defects.

The calculated readings can not include the edge effect when the tube support is at the edge of the coil. This can and will be included in the experimental measurements. (Without this included, there was a 0.003 in. change in the thickness reading as the support was moved by the coil.)

The program, LSQENC, has been modified to allow different properties to be omitted from the data set. We will use this to determine which property variations are causing the most errors and try to get better fit on these properties.

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 - F. B. K. Kam attended the Light Water Reactor Pressure Vessel (LWR-PV) Dosimetry Surveillance Program meeting at NBS November 7, 1978. The comments, commitments and agreements of this meeting have been distributed by HEDL. Not included in the minutes was an agreement signed by W. N. McElroy and F. B. K. Kam to request that all experimenters make an earnest effort to comply with a procedure for estimating uncertainties associated with PCA experiments.

Task 2: Benchmark Fields (Controlled Environments) -

A. LWR Dosimetry Pressure Vessel Benchmark Field (PCA) - The design, fabrication, and installation of an experimental rig for IRT's proton-recoil measurements sponsored by EPRI were completed on time. Stanley Friesenhahn and T. R. Dittrich of IRT completed their experiments on schedule.

The photofission experiment was cancelled by EPRI because of funding.

The fission chamber, radiometric, and SSTR measurements by Mol and HEDL are continuing on schedule.

B. LWR Metallurgical PV Benchmark Field (PSF) -

1. Because proposed changes in the PSF configuration, suggested at the November 7, 1978 LWR-PV Program meeting, have not been decided, revisions of drawings will await final specification.

2. Process Control System (PCS) - Hardware design for the PCS has been completed. Technical specification of the equipment will be initiated December 1 with a tentative completion date of December 31.

3. Instrumented Irradiation Capsule (IIC) - Changes in the location of the Surveillance Specimen Capsule (4 cm closer to the core) has increased the neutronic heat estimate to where it is deemed advisable to measure the heat generation. This change has also raised the question of possible bulk boiling in the vicinity of the thermal shield. As a consequence, it has been decided to build a simple gamma heat measurement experiment which will also incorporate a full-sized thermal shield to permit evaluation of not only the gamma heat but also what, if any, bulk boiling occurs. The design for this heat measurement test has been formulated and we hope to start parts procurement by December 1, 1978.

   The prototype heater and cooler plate assembly test parts are on hand and assembly is in progress.

   The final modifications to the drawings for the PVS and SSC capsules to accommodate the position change are essentially completed, and some procurement of long term items has begun.

Task 3: Neutron Field Characterization - Transport Methods - Table 1 of the previous Monthly Highlight is included to reflect corrections in the locations and buckling values. One-dimensional calculations of five of the seven different core configurations agreed upon at the November 7, 1978 LWR-PV meeting at NBS have been completed. Calculations of all configurations will be completed in December of 1978.

Task 4: Dosimetry and Damage Correlation Analysis - Foil dosimetry experiments in the PCA Benchmark Field are being analyzed and compared to results from neutron transport calculations.

Work is being considered on the third HSST experiment and is near completion.

Table 1. Comparison of Measured and Calculated $^{238}U(n, f)$ Profiles as a Function of Assumed Buckling[a]

Location	Experiment	ANISN Model			XSDRNPM Model	
		No Buckling	65 cm × 38 cm	40 cm × 30 cm	No Buckling	65 cm × 38 cm
1" before TS	1.0	1.0	1.0	1.0	1.0	1.0
3.38 cm behind TS	0.10	0.15	0.125	0.11	0.12	0.108
1" before PVS	0.093	0.12	0.10	0.090	0.102	0.085
1/4-T in PVS	0.023	0.045	0.033	0.027	0.031	0.021
1/2-T in PVS	0.010	0.018	0.013	0.010	0.014	0.009
3/4-T in PVS	0.0037	0.0076	0.0048	0.0036	0.0057	0.0032

[a] It should be mentioned that the measurements are still preliminary and corrections may be necessary.

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 02 2

TECHNICAL HIGHLIGHTS

At NRC request, the name of the program is now Reactor Safety Research Data Repository (RSRDR) and subsequent reports will carry the new title. Formal public relations material and a news release publicize the RSRDR program and the available data.

A brochure describing the RSRDR was designed, printed, and made available at the recent annual WRSR information meeting. This brochure provides a means of contacting the RSRDR for additional information. A report, prepared on the current inventory of available test data in the RSRDR, includes a copy of the abstract from the experimental data report for each available test. The document was made available at the information meeting and will be sent to each person requesting additional information.

PROGRAM TITLE: Safety Related Operator Actions

PROGRAM MANAGER: T. F. Bott and P. M. Haas

ACTIVITY NUMBER: ORNL #40 10 01 06 01 (189 #B0421-8)/NRC #60 19 10 01 2

TECHNICAL HIGHLIGHTS

A site visit to Peach Bottom Atomic Power Station was completed this month. Examination of normal plant records on reportable events that have occurred there tends to confirm the preliminary conclusion based on the Zion and Connecticut Yankee visits. That is, it will be a tedious, labor intensive task to extract from normal plant records information that is much more detailed than is already available in NRC docket files. In the case of Peach Bottom, there is some further information on operator response times available in internal reports called "plant upset reports." This information will be included in our summary of initial data collected during this preliminary study. There may be similar plant documents available at other plants which have applicable information.

Additional operator-opinion survey forms were completed at Peach Bottom. Because there were no relief operators available, only a few surveys could be completed during our visit, but operating supervisors agreed to administer the survey to operators and forward results to ORNL. Since Peach Bottom was the first BWR to be visited, the form was revised to accommodate specific BWR events of interest.

A detailed literature survey has been initiated by human factors consultants at the University of Tennessee to gather all available information on studies of human behavior under stress that might be directly applicable to nuclear operator behavior under severe accident conditions. Particular emphasis is being placed on field studies, rather than laboratory experiments, especially for jobs such as airline controllers or coal-field power plant operators that are likely to be similar to nuclear operators.

Information has been retrieved regarding a previous job analysis of the task of nuclear operator using the Position Analysis Questionnaire (PAQ). The study, performed at the Center for Nuclear Studies at Memphis State University, included approximately 370 operators at a number of nuclear plants (including Zion), and may have information applicable to the N660 standard. If available results of previous studies such as this can be shown to be applicable, the need for additional PAQ surveys may be reduced or eliminated.

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:

A report on the HOBBIE-1 in-reactor creepdown results is almost completed, the HOBBIE-2 results are being analyzed, the HOBBIE-3 test is running in the HFR reactor in Holland, and HOBBIE-4 is being assembled. Out-of-reactor tests are running concurrently to provide baseline data for comparison with the in-reactor data.

A decision was made to lower all test temperatures to 343°C (650°F) from 371°C (700°F) in order to separate flux effects from thermally activated creep effects. The dividing line for this creep phenomenon is, unfortunately, just at the operating temperature of the reactor fuel cladding.

We are also modifying HOBBIE-4 in order to decrease the temperature gradient that exists due to thermal convection loops in the helium around the specimen. We found temperature differences of 25°C in HOBBIE-1 along the specimen length. This will be discussed in the above mentioned report.

Finally, a trip was made by K. R. Thoms and L. D. Chitwood to ECN-Petten to repair leaks in the HOBBIE-3 instrumentation lead wires. The trip was initiated by a preplanned contingency agreement between NRC and ORNL in which assistance could be provided if an emergency arose. During pre-startup checks it was discovered that the HOBBIE-3 capsule had developed leaks through the insulation of the sheathed lead wires, and one of the eddy-current instruments was inoperative. The leaks in the lead wires were sealed with epoxy, the inoperative instrument was repaired and the backup instrument, which developed problems while they were there, was also repaired. After all repairs had been made and tests were performed to assure that the repairs did not affect the calibration of the instrumentation, the capsule was installed in the reactor and the creepdown test was successfully started.

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