**August 1976
Monthly Highlights
for
Office of Nuclear Regulatory Research Programs
at
Oak Ridge National Laboratory**

MASTER

OAK RIDGE NATIONAL LABORATORY

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Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy \$4.00 Microfiche \$2.25

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Contract No. W-7405-eng-26

AUGUST 1976
MONTHLY HIGHLIGHTS
FOR
OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS
AT
OAK RIDGE NATIONAL LABORATORY

Compiled by
Gordon G. Fee

Date Published: October 1976

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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PROGRAM TITLE: Heavy-Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

ACTIVITY NUMBER: 40 89 55 10 1 (189 No. B0119)

TECHNICAL HIGHLIGHTS

Task 1: Program Administration - On August 3, G. D. Whitman reviewed the major accomplishments of the HSST Program in a presentation at the Nuclear Power Safety Institute sponsored by Oak Ridge Associated Universities.

R. H. Bryan and P. P. Holz met with the PVRC Task Group on weld repair of pressure vessels on August 4, in Atlanta, Ga., to discuss the V-7A test results and review plans for the repair and retesting of this vessel.

On August 6, a program briefing was held at ORNL for Drs. Knuth, Murley, Tong, and Kelber of RSR.

A review of the HSST program was held at NRC-RSR offices in Silver Spring, Maryland on August 9.

Presentations were made on Intermediate Vessel Testing and the Thermal Shock Project at a Vessel Integrity Review Group Meeting held on August 10.

A presentation was made to Simon E. Bourgin, Deputy Director, NRC, Office of Public Affairs, on August 25.

Task 2: Fracture Mechanics and Analyses - In preparation for writing the topical report on the testing of Intermediate Test Vessels V-5 and V-9, posttest strain tolerance calculations have been made for the two vessels, both of which contained inside nozzle corner cracks.

Strain measurements made on the opposite inside nozzle corners of both vessels indicate that a considerable amount of transverse contraction occurred along the nozzle corner crack fronts. It has been found by previous experience, that under these conditions, the elevation of the fracture toughness above the plane strain value due to transverse contraction can be represented by the Irwin B_{Ic} correction. Using this equation, in conjunction with either LEPM based on strain or the Tangent Modulus Method, results in calculated nozzle corner strains at failure that compare well with experimental data. A rational method for estimating the relationship

between vessel internal pressure and peak nozzle corner strain has also been developed. The result is a straightforward procedure for estimating both the failure pressure and the failure strain for a nozzle corner crack.

Task 4: Irradiation Effects - Reactivity runs and neutron spectrum experiments using the "dummy" capsules were completed and counting of dosimeters is in progress.

During preirradiation testing of one of the specimen capsules at maximum design heater power, several heater leads shorted to ground. Repairs are in progress and measures are being taken to prevent a similar incident in any of the three capsules. The incident will delay start of specimen irradiations about four weeks.

Task 5: Simulated Service Tests - The cut out flawed region of intermediate test vessel V-7A was chilled and broken open to expose the fracture surfaces. Visual examination confirmed measurements and observations of flaw depth made before the material was cut from the vessel. The crack tore 10 to 20 mm in the axial direction beyond the initial crack tip. The length of the rupture at the inside surface of the vessel is about 100 mm. The fracture zone is being examined metallographically.

We have completed a more extensive investigation of the hardness across the fusion line of the ITV-9 prolongation weldment which was used to characterize the repair weld in ITV-7A. In one traverse a total of 85 diamond pyramid hardness tests was made over a distance of 0.85 in. The base metal (BM) hardness ranged from about 192 to 204 DPH. A slight (perhaps 6 to 12 DPH numbers) increase is observed before any microstructural effects could be observed by macroscopic procedures. The hardness noticeably (~25 DPH numbers) increases at the start of the dark etchant band that identifies the heat-affected zone. The maximum hardness recorded in the HAZ is about 291 DPH. The highest hardness in the other two traverses (each was made with 29 indentations) was closer to 270 DPH. The highest hardness values occurred in the region of the HAZ that etched the darkest. The actual location of this area, in respect to the macroscopic indication of the onset of the HAZ varied in the individual hardness traverses. The hardness of the visually (by macroscopic procedures) diluted weld metal was nearly identical to that of the bulk weld metal.

Task 6: Thermal Shock - The third thermal shock experiment, TSE-3, was conducted on August 4, 1976. Output from the COD and AE instrumentation indicates that crack initiation took place. However, the thermal shock was not quite as severe as desired because of problems with the quick opening valve. The maximum K ratio was 1.1 instead of 1.2. Further interpretation of the experimental results awaits removal of cores from the crack zone.

A beam experiment was conducted with A533 quench-only material for the purpose of investigating the effect of EB-weld residual stress on crack initiation. The beam had a 6-in. by 6-in. cross section and was 30 in. long. The flaw was 2 in. deep and was prepared by machining a groove to within 0.25 in. of the desired crack depth and then applying an EB weld, which cracked upon cooling. The beam was simply supported and loaded at the center. Initiation took place very close to the load predicted in the absence of residual stresses. Further testing is planned.

On the basis of data obtained from TSE-2 and laboratory experiments conducted by others, it is anticipated that warm prestressing will be a very significant factor in the analysis of crack propagation in PWR vessels. Calculations are being made to determine if warm prestressing effects on long axial and continuous circumferential cracks can be studied in the thermal shock test facility. The answer is dependent on the arrest toughness characteristics of the test specimen material. Based on K_{Ia} data deduced from TSE-2, it is tentatively concluded that warm prestressing cannot be studied in the present test facility, even if it were modified to handle the ITV-size test specimens. However, the combination of the large test specimen and a sink temperature of -100°F does permit inclusion of warm prestressing.

PROGRAM TITLE: Fission Product Beta and Gamma Energy Release

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

ACTIVITY NUMBER: 4C 89 55 10 5 (189 No. B0095)

TECHNICAL HIGHLIGHTS

Reduction of data obtained during June and July continued through August. During this period the data yielding the number of fissions, n_f , were completely reduced for all 87 fissile samples which were run. Four gamma rays were used; these were the dominant gamma rays from ^{99}Mo , ^{97}Zr , ^{132}Te , and ^{143}Ce . In addition, the 530-keV gamma ray emanating from ^{133}I was monitored for possible iodine gas loss. None was observed. Data reduction of the gamma-ray energy release measurements of July was initiated. The data for irradiation periods of 1 and 10 sec were reduced to final values; the data for 100 sec irradiation period is nearly finished. However, all gamma-ray data still require an error analysis before completion.

Design of the fission chamber was completed and the fabrication of the system was initiated. The final assembly will be during the first shakedown run. The design allows for rapid (< 5 min) disassembly for nearly immediate counting; however, the first experiments will entail only Ge(Li) measurements at $t_{\text{cool}} \sim 20$ and 70 hours. Optical realignment of the external beam hole HB-1 was completed, and the monitoring chambers mounted in the beam hole were checked out.

PROGRAM TITLE: Fission Product Release from LWR Fuel

PROGRAM MANAGER: A. P. Malinauskas

ACTIVITY NUMBER: 40 89 55 10 8 (189 No. B0127)

TECHNICAL HIGHLIGHTS

Data analysis for Low Burnup Test 1 is incomplete. However, it has been determined that only about 0.0003% of the total cesium inventory in the fuel capsule had been released in the test. Preparations have been completed for Low Burnup Test 2, which will be conducted at 900°C for 2 hr in steam.

A paper entitled "Fission Product Release from Simulated LWR Fuel" has been prepared for presentation at the upcoming CSNI Specialists' Meeting on the Behavior of Water Reactor Fuel Elements Under Accident Conditions.

PROGRAM TITLE: Multirod Burst Tests
PROGRAM MANAGER: Robert H. Chapman
ACTIVITY NUMBER: 40 89 55 10 6 (189 No. B0120)
TECHNICAL HIGHLIGHTS

Efforts to establish a process for reprocessing boron nitride (BN) material have been successful. The lack of acceptable BN material had caused SEMCO to stop production of fuel simulators as reported previously. Small test batches of the BN material, reprocessed by a magnetic separation process, have successfully passed the newly established acceptance criteria based on chemical analysis, distribution of dense particles as determined by radiography, and electrical resistivity as determined from high-temperature conductivity cells.

The purification of 15 lbs of BN material (of a total of 200 lbs to be reprocessed) has been completed by Magnetic Engineering Associates (MEA), of Cambridge, Massachusetts, with additional vacuum-bake out to be performed at Oak Ridge before shipment to SEMCO. SEMCO will then begin fabrication of fuel simulators simultaneous to acceptance testing to be performed on samples of the material at Oak Ridge. Although there is some risk in SEMCO's use of this material before final acceptance, preliminary tests have indicated this risk is small.

An additional 25 lbs of BN material has already been treated by the magnetic separation process with bake out and final acceptance testing remaining to be completed before its use in the manufacture of fuel simulators. The present schedule projects SEMCO beginning production of simulators early in September and completion of the initial 20 simulators (for 4 X 4 bundle No. 1) by December 1. This 3-month delay in the simulator delivery is expected to cause a 2-month delay in the testing of the first 4 X 4 bundle now projected by July 1, 1977.

Sufficient electrical penetration glands for the assembly of fuel pin simulators for 4 X 4 bundle No. 1 have been received and have passed acceptance tests at temperatures and pressures necessary for bundle tests. This special order seal gland has been difficult to obtain because of the requirements imposed by bundle dimensions and test environment. Because of the difficulty in obtaining a satisfactory seal gland for this application, two other manufacturers have been asked to supply alternative versions of this seal gland as backup for test and evaluation.

On August 10-12, R. H. Chapman and G. Hofmann attended the ASTM Conference on Zirconium in the Nuclear Industry which was held in Quebec.

Dr. P. Tarkpea of AB Atomenergi (Studsvik, Sweden) visited ORNL on August 18 to discuss ballooning tests. Dr. Tarkpea is conducting Zircaloy ballooning tests on 30-cm-long tubular specimens under LOCA conditions postulated for BWRs. They employ induction heating primarily, although direct resistance heating of the tube has been used. The tests are conducted in an argon environment and very slow heating rates ($0.5^{\circ}\text{C}/\text{sec}$) are used. These tests start at about 300°C ; fairly fast heating ($\sim 35^{\circ}\text{C}/\text{sec}$) is applied to increase the temperature to 1040°C in 20 sec, then it is reduced to the lower rate ($\sim 0.5^{\circ}\text{C}/\text{sec}$) to produce a temperature of 1204°C at 400 sec. At this time the heating is further reduced (or terminated) to permit cooling of the tube to 1040°C at about 20 min into the transient. Most of the deformation (and failures) occur during the long transient (from 1040 to 1204°C). They have observed very low failure strains, generally 10-20% with a few isolated cases as high as 30-35%.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: William B. Cottrell

ACTIVITY NUMBER: 40 89 55 10 4 (189 No. B0126)

TECHNICAL HIGHLIGHTS

During the month of August the staff of the Nuclear Safety Information Center (a) processed 977 documents, (b) responded to 87 inquiries (of which 47 involved the technical staff), and (c) made 20 computer searches (of which 6 involved payment). A bibliography of 184 reports received from the ACRS during June and July was prepared and distributed. NSIC staff received 9 visitors during August, participated in 2 meetings (including 1 presentation) and prepared 2 reviews. The presentation was a program review for D. F. Knuth, L. S. Tong, C. N. Kelber and T. E. Murley, all of NRC's Division of Reactor Safety Research.

Several NSIC reports are in various stages of preparation. The first volume of the 3-part "Bibliography on LMFBR Safety Literature" is in composition, and work is under way on the other two volumes, as well as ORNL/NUREG/NSIC-128 "A Bibliography of HTGR Safety Literature". Also under way are preparations for an updated bibliography of all reports on NRC subcontract safety studies through June 30, 1976. A draft of this report was submitted to NRC early in August; it will be formally issued as ORNL/NUREG/NSIC-130.

NSIC's special selective dissemination of information (SDI) is available to paying users (as well as additional non-paying users). During the month of August we added 1 free and 1 paying (\$140) users,

bringing the total SDI users to 366, including subscribers who have paid a total of \$13,409 (since October 1975).

During the month we wrote one letter (August 27) to NRC regarding the translation of 4 Japanese and 4 German documents obtained through the NRC exchange program. Also during August the third batch of foreign-language safety documents (consisting of 5 reports which had not previously been translated and distributed) was delivered to the contractor for the preparation of microfiche and distribution of same to the special NRC distribution list. A minor change in format has been made to comply with an ANSI standard now in final stages of approval.

Manuscripts for the technical articles for *Nuclear Safety* 18(1) are in various stages of peer review, revision and editing. ERDA and NRC review, as well as the subsequent revision of the text, of all articles for *Nuclear Safety* 17(6) has been completed.

PROGRAM TITLE: PWR Blowdown Heat Transfer Separate Effects

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: 40 89 55 10 3 (189 No. B0125)

TECHNICAL HIGHLIGHTS

Task 1. As part of the agreements and commitments developed during the June 16-18, 1976, program review, ORNL agreed to determine experimentally the heater rod temperature at which degradation of T/C readings occurs. Starting last month tests were begun in which the heater temperature was allowed to reach higher and higher values during either a steady-state CHF test or a blowdown transient. An indication of impending failure was observed during the last steady-state CHF experiment with bundle 2 prototypical heater W150-100; the data indicated arc or a short between the heating element and a central thermocouple after a central thermocouple indicated a temperature of about 1400 K (2060°F). Severe 60-cycle noise problems appeared throughout the data system. An over-voltage condition ruined 42 field effect transistors in the ANSCAN data acquisition system. A 500 V Megger test after the experiment showed zero resistance from central thermocouple sheath to heating element for four out of five of the central thermocouples. These four thermocouples also showed low ohmmeter resistance readings (on the order of 400 Ω) from chromel wire to thermocouple sheath and alumel wire to thermocouple sheath (compared to usual readings of $\sim 10,000 \Omega$). Heater resistance was 0.38 Ω , a normal value.

After repair of the data system and isolation of the central T/Cs, the heater was operated again at full power and subjected to another blowdown. The heater operated normally throughout the experiment except for indications of an unusually short (3 sec as compared to 8 sec expected) time to CHF for the break area ($\sim 1 \text{ cm}^2$). Heater temperatures above 1400 K occurred but noise on some of the sheath T/Cs prohibited an accurate reading.

The heater was removed from the test section for examination. The stainless steel sheaths and the inconel heating element had melted in the lower third of the middle heat flux zone. The heater actually broke into two pieces at this location during the removal from the test section. The tensile force applied to the heater during the removal process was ~ 28

pounds. Radiographic examination of the entire heater shows low density spots in the heating element in two areas in the highest power zone. The heater broke in the lower of these low-density areas. The hypothesis is that the Inconel melted and refroze with resultant holes in the walls of the Inconel tube resulting in hot spots which finally caused the failure of the heating element. The accompanying failure of the stainless steel sheaths (at the same axial location) is not surprising. Several groups are now examining the heater to obtain additional evidence on the reasons for failure.

The bundle 2 prototypical heater W150-10C failed after 15 blowdowns and 15 steady-state CHF runs.

The heater operated for a total of 59.2 hr, with 26 hr at full power.

Task 2. Quick-Look Reports for THTF Tests 103 and 105 were published. These tests were conducted with core flows approximating those calculated to occur in a 200% LOCA for LWRs.

All the data from Tests 101 through 105 have been processed by the first two sections of the T/C calibration program ORTCAL. The output from ORTCAL is now directly transferred to the inverse code ORINC. Modification of ORINC to accept this data has begun.

Preliminary development of a data reduction program to determine first CHF and CHF propagation has been initiated.

Test 103, conducted in the Thermal-Hydraulic Test Facility (THTF) on August 5, 1976, was a full power (5.975 MW) test initiated from a test section inlet temperature of 558.3 K (545°F), a test section inlet volumetric flow of 0.0271 m³/sec (429 gpm), and a test section outlet pressure of 15.748 MN/m² (2284 psig). System decompression was accomplished by introducing a 40% inlet-60% outlet break. The primary coolant pump was tripped coincident with break initiation, but the electric core was operated at full power for ~2 sec into the transient.

As the test section outlet subcooling was ~13 K (24°F), the outlet break saturated almost instantaneously. The fluid initially between the heat exchangers and the pressurizer has a strong effect on the test section outlet flow causing it to reverse ~4.5 sec into the transient. CHF was observed ~0.4 sec after break initiation and the maximum observed

sheath thermocouple indicated temperature was ~ 977 K (1300°F) 30.48 cm (12 in.) above the axial center of the electric core.

Test 105, conducted in the Thermal-Hydraulic Test Facility (THTF) on August 19, 1976, was a full power (5.973 MW) test initiated from a test section inlet temperature of 558.3 K (545°F), a test section inlet volumetric flow of $0.0269 \text{ m}^3/\text{sec}$ (427 gpm), and a test section outlet pressure of 15.534 MN/m^2 (2253 psig). System decompression was accomplished by introducing a 40% inlet-60% outlet break. The primary coolant pump was tripped coincident with break initiation, but the electric core was operated at full power for ~ 2 sec into the transient and then the power was decayed with a time constant of ~ 0.45 sec.

As the test section outlet subcooling was ~ 13 K (24°F), the outlet break saturated almost instantaneously. The fluid initially between the heat exchangers and the pressurizer has a strong effect on the test section outlet flow causing it to stagnate ~ 4.5 sec into the transient. CHF was observed ~ 0.4 sec after break initiation and the maximum observed sheath thermocouple indicated temperature was 1011 K (1360°F) at the axial center of the electric core. A 40% inlet-60% outlet break appeared to provide a reasonable approximation to the desired core flows.

Task 3. Tests 103 and 105 were completed successfully with no major problems; minor problems were some inboard pump seal leakage, failure of one heater on the pressurizer, leakage from the relief valve on the test section outlet and outboard pump bearings running hot. The seal leakage problem was corrected by replacing the inboard seal (which had operated for ~ 300 hr) with a new Borg-Warner seal; bearing temperature was reduced to an acceptable value by providing additional cooling and other problems were corrected by routine maintenance between tests. There continues to be problems with turbine meter signal polarity changes apparently due to interference of the magnetic field with pickoff probe signals. Additional shielding is being provided.

Task 4. The THTF flow monitoring instrumentation spool pieces include target flow meters (drag disks) to measure fluid momentum. The targets are 1.27-cm-diam (0.5-in.) disks 0.89-cm (0.35-in.) thick each of which are supported concentrically with the pipe axis by 0.635-cm-diam (0.25-in.) rods. This design has two major limitations: (1) the target sees a

relatively small sample of the flow field and (2) the drag crises associated with flow past the combined rod-disk bluff body at high Reynolds numbers result in significant variations in drag coefficient with Reynolds number greater than $\sim 10^5$.

As a possible solution to both problems, preliminary tests have been conducted in the air-water two-phase flow facility using a screen target with the drag meter. A circular section of woven wire screen, nearly as large as the pipe cross section, was installed on the support rod. The drag screen sampled most of the pipe cross section so that the instrument output represented more of an area-average momentum. Further, the wire screen should uniformly trip the boundary layer on the support rod resulting in stable drag coefficients over a wide range in Reynolds number. Single-phase calibration in both air and water was linear with fluid momentum over a wider range of momentum than for the present drag disk design calibration. Tests are being conducted to determine the characteristics of the drag screens in steady-state two-phase (air-water) flows.

Task 5. Purified BN has been shipped to Watlow for use in fabricating bundle 2 heater rods. The next 5 prototype heater rods for bundle 2 will be fabricated with reduced final swaging pressure on the outer sheath in an attempt to reduce variation in thermocouple location. Based on information obtained in bundle 1 testing, some bundle 2 thermocouples have been moved to permit more accurate determination of heat-transfer coefficients.

Task 6. A requisition has been written to purchase one heater rod from INEL from those being fabricated for the Semiscale Mod 2 12-ft core. INEL will supply the first spare from their current order. Based on tentative criteria supplied by NRC for bundle 3, a requisition is being written for purchase of prototype heaters. The purchase order will not be placed until final approval is given by NRC.

PROGRAM TITLE: Zircaloy Fuel Cladding Collapse Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: 40 89 55 10 7 (189 No. B0124)

TECHNICAL HIGHLIGHTS

Creepdown testing of Zircaloy fuel cladding has started. Current tests are being used for checkout of the eddy-current monitoring equipment. To date, all strain output is considered tentative and will not be reported pending evaluation and measurement of tested specimens with micrometers. Such measurements will allow further calibration of the instrumentation.

A small number of collapse tests were run with specimens that had pre-existing ovalities -- up to 0.0016-in. (0.038 mm) difference in major and minor diameters. Collapse pressures for these specimens ranged from 6 to 12% lower than the pressures measured for specimens with less than 0.0006-in. (0.015 mm) ovality. ASTM B353-71 specifications for the tubing used in these studies allowed up to 0.004-in. (0.10 mm) ovality. It is expected that a correlation between ovality and collapse pressure can be obtained with a small number of additional tests.

PROGRAM TITLE: Zirconium Metal-Water Oxidation Kinetics

PROGRAM MANAGER: C. J. McHargue

ACTIVITY NUMBER: 40 89 55 10 9 (189 No. B0128)

TECHNICAL HIGHLIGHTS

The apparatus for studying the effect of steam pressure on the oxidation rate of Zircaloy has been assembled and is being debugged. The schedule for the construction of this apparatus was delayed by the late release of capital equipment funds in FY 1976, and there is some doubt that the scoping tests of steam pressure effects can be completed by Oct. 1, 1976 (Milestone No. 43102).

PROGRAM TITLE: Aerosol Release and Transport from LMFBR Fuel

PROGRAM MANAGER: M. H. Fontana

ACTIVITY NUMBER: 40 89 12 10 1 (189 No. B0121)

TECHNICAL HIGHLIGHTS

CRI-III/CDV:

Progress toward the completion and checkout of the ORNL high voltage capacitor bank for fuel vaporization (the CDV system) has been significant. All of the major coaxial power leads and switch gear have been connected leaving only the panel wiring connections between the control and data collection system and the bank and test assembly to be completed. Additional instrumentation will continue to be added from time to time as received.

Cold Proof Tests:

The U_3O_8 cold proof tests which were completed last month have served to demonstrate a significant discrepancy between several methods of particle size classification resulting from high velocity fragmentation of the large agglomerates of the primary particles. This effect (fragmentation) was noted first when comparing the apparent agglomerate size calculated from the Stokes settling velocity with the values obtained from the two types of cascade impactors and the particle electrical mobility analyzer. Two of the instruments depend upon sonic velocity jets and it is in these that we obtained relatively small MMD values of 0.2 to 0.3 μm while the low velocity impactor tended to confirm the Stokes behavior values of 1.0 to 1.5 μm MMD.

Bubble Transport:

All pipe, bar stock and plate necessary for component fabrication and facility construction were received and are now being ultrasonically inspected. Delivery of the required tubing is now estimated at November 1, 1976.

Drawings and calculations submitted by the pressure vessel vendor were reviewed and our comments transmitted to the vendor.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: J. P. Sanders

ACTIVITY NUMBER: 40 89 55 11 2 (189 No. B0122)

TECHNICAL HIGHLIGHTS

General: Development of the ORTAP code for the Fort St. Vrain (FSV) reactor was continued. Comparisons were made between ORNL (BLAST) and the University of Tennessee (LAP) steam generator code outputs.

Development of Overall NSS Simulation (ORTAP): The circulator-turbine subroutine (ORCIRT) was modified to include the latest FSV design data. Provisions were also made for operation in the "coastdown mode," where the requested RPM is less than the RPM resulting from zero turbine steam flow.

Analysis of System Transients Using ORTAP: Analysis of a slow design basis depressurization transient assuming a reactor trip due to low primary system pressure was completed up to the time at which the turbine trip occurs (approximately two minutes after the reactor trip). A hot channel core analysis using results from ORTAP was performed with the CORTAP¹ code. Also, a hot channel analysis was performed for the transient resulting from a rod withdrawal accident.

¹J. C. Cleveland, "CORTAP: A Coupled Neutron Kinetics-Heat Transfer Digital Computer Program for the Dynamic Simulation of the High Temperature Gas-Cooled Reactor Core," ORNL/NUREG/TM-39, (in preparation).

PROGRAM TITLE: Design Criteria for Piping and Nozzles

PROGRAM MANAGER: S. E. Moore

ACTIVITY NUMBER: 40 89 55 10 2 (189 No. B0123)

TECHNICAL HIGHLIGHTS

Code Rules Development: Proposed ASME Boiler and Pressure Vessel Code (BPVC) revisions for the analysis and qualification of flanged piping joints in Class 1, Class 2, and Class 3 nuclear piping systems, were distributed to members of the Code Committee Working Group on Piping (UGPD) along with the technical backup material. These rules revisions are scheduled for consideration at the regular September meetings of the ASME BPVC Committee during "Code week" (September 13 to 17, 1976).

Topical Reports: A report - ORNL/NUREG-2 entitled *Stress Analysis of Perforated Flat Plates Under In-Plane Loadings* by J. P. Callahan and J. W. Bryson was issued this month. The report describes the stress analyses of a series of flat plates with one, two, three, and five closely-spaced unreinforced penetrations and comparisons of the analytical results with experimental results obtained previously. The report is the third and final report in a series describing experimental and analytical stress analysis studies of flat plates with unreinforced holes and attached nozzles conducted under the ORNL Design Criteria Program. Flat plate models were used in the studies to simulate the behavior of small diameter penetrations and nozzles in spherical heads and cylindrical pressure vessels. The information contained in these reports will be used to support future evaluations of design analysis methods, design criteria, and Code rules for reactor pressure vessels.

Nozzles in Cylindrical Shells Studies: We completed the operational checkout of the elastic-plastic finite-element computer code CORTES-EP, at the Oak Ridge computer facilities. The program has been transmitted to the Argonne Code Center; however, several less than adequate features were identified during the check-out phase. The two most troublesome are (1) the Code's inability to accurately calculate stresses at the element surfaces (stresses are calculated at positions inside the solid

elements), and (2) a high ratio of wall clock time to CPU time. The first makes it difficult to compare analytical results with experimental results, and the second adversely affects the total job turn-around time. We plan to improve the program in these two areas as time and money permit.

The finite element computer program MULT-NOZZLE being developed for us by Mechanics Research Inc. (MRI) is still not operating satisfactorily. Although we have analyzed the required demonstration problems and have obtained reasonable results for steel models, the calculations are sensitive to the value used for Poisson's ratio ν . Since the only good experimental data for closely-spaced nozzles in cylindrical pressure vessels were obtained from a photoelastic model, which had a value for ν of 0.485 rather than the value of $\nu = 0.3$ commonly used for steel, validation of the computer program essentially requires that we be able to compute stresses accurately using the higher value for ν . We believe that we have identified the specific problem and are in the process of revising the program. Our current best estimate for completing the internal pressure analysis for two models with two closely-spaced nozzles is the end of October. In the meantime, however, we have started work on the second phase development to add force and moment load analysis capability to the program. This should put us in a better position to meet our longer range commitments when the first problem is solved.

ANSI B16.9 Tee Studies: The subcontract with Combustion Engineering Inc., under which we had conducted experimental stress analysis and fatigue tests of ANSI B16.9 tees and elbows, was closed. The untested models for the 12-in. tees T-5 and T-9 and for the 24-in. tee T-17 were returned to ORNL along with most of the laboratory equipment that was on loan to CE for conducting the tests. This equipment will be used to support other programs at ORNL.

PROGRAM TITLE: Dose Conversion Factors for Inhalation of Radionuclides

PROGRAM MANAGER: J. W. Poston

ACTIVITY NUMBER: 40 10 01 06 1 (199 No. B0400)

TECHNICAL HIGHLIGHTS

The section is engaged in generating data for the NRC similar to material furnished previously for inclusion in the report, Reactor Safety Study, WASH-1400, Chapter D, Appendix VI on Internal Dosimetry. The previous data consisted of metabolic models together with dose-conversion factors estimated at twelve time intervals ranging from 2 days to 50 years. These estimates were made for the inhalation of 1 micron size particle of the radionuclides involved in the safety study. The new data consists of dose conversion factors for the same radionuclides based on the same metabolic models but for aerosols of two different sizes, namely, 0.2 and 10 microns. The output for the 48-50 radionuclides considered will be quite voluminous. Thus, in addition to the hard copy showing in detail dose vs type of decay, a tape of just the total doses has been requested. To date, two tapes and several mailings of hard copy have been sent containing dose estimates for a total of 42 radionuclides for the 10 micron size aerosol. Data for the remaining radionuclides is much more complicated to generate because the metabolic parameters, being from the work of Marshall in ICRP Publication 19, are not of standard form. After completion of the computations for the 10 micron size, runs will be started for the aerosol of 0.2 micron. This project will be completed by September 30, 1976.