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

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FEBRUARY 1976

MONTHLY HIGHLIGHTS

FOR

OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS

AT

OAK RIDGE NATIONAL LABORATORY

Compiled by

Gordon G. Fee

FEBRUARY 1976

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

PROGRAM TITLE: Heavy-Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

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

TECHNICAL HIGHLIGHTS

Task 1: Program Administration — A Buff Book schedule review was held with C. Z. Serpan and E. K. Lynn on February 2, in Oak Ridge.

The NRC Vessel Integrity Review Group met in Oak Ridge on February 3 and 4, to review the thermal shock and sustained load vessel testing projects.

On February 10, G. D. Whitman and G. G. Fee participated in a review of program commitments in Germantown.

On February 18, J. G. Merkle attended a meeting of the ASME Section XI Task Group on Evaluation held in Chicago.

G. D. Whitman attended a meeting in Germantown on February 19, to plan initial testing on cyclic crack growth studies.

G. D. Whitman and J. G. Merkle visited Virginia Polytechnic Institute and State University on February 24, to review plans for photoelastic studies that will be performed to determine stress intensity factors for surface flaws at nozzle corners.

Task 2: Fracture Mechanics and Analysis — A subcontract has been awarded to Virginia Polytechnic Institute and State University, in the Engineering Science and Mechanics Department, to obtain stress intensity factors in surface flaws at nozzle corner intersections. Stress freezing photoelasticity coupled with a digital computer program known as the Taylor Series Correction Method will be employed to obtain stress intensity factors. A mold has been constructed and the first models are being fabricated.

The specimens from plates 01 and 02 and weldment 51B which will be used to study the influence of stress-relief time and temperature have been incapsulated and are awaiting their post-weld heat treatment.

Task 3: Fatigue Crack Growth — Agreement was reached to utilize one "2T" test chamber at Westinghouse Research and Development Laboratories to start preliminary testing to study the effects of load rise time and hold time on cyclic crack growth at 288°C (550°F) in PWR water. A companion test will also be performed in 100°C (212°F) water to evaluate behavior in low temperature environment.

In the February 18, ASME Section XI meeting, it was agreed to add a note to the fatigue crack growth rate curve in high temperature reactor grade water limiting the applicability of the curve to $0 \leq R \leq 0.25$.

Task 4: Irradiation Effects — A determination of fluences from material samples taken from the 4T-CT specimens previously tested at Westinghouse is as follows:

Specimen No.	Fluence, 10^{19} n/cm^2	
	Average[1]	Center
W58-1 (Weldment)	4.43	3.75
W58-2 (Weldment)	4.79	3.93
W58-3 (RW)	4.17	3.35
02GA441 (WR)	3.93	3.36
02GA442 (WR)	4.11	3.44

[1] Average of three locations; two near surfaces, one near center.

These values agree quite well with the fluences determined from post-irradiation evaluations.

Fabrication of support structures for the BSR, utilities, capsule parts, and instrumentation is nearing completion and is on schedule. Specimens for the irradiation capsules were received during February. However, one 0.5T-CT specimen of the wrong material was received and has been replaced with a correct specimen and the eight precracked Charpy specimens of one of the welds were found to have been overloaded during fatigue precracking. We are precracking eight standard Charpy-V specimens and replacement Charpy-V specimens of this weld are being prepared.

An alternate system for load-line displacement measurement for the compact tension specimens has been devised. This system should provide load-line displacement measurements in both static and dynamic tests. Mounting holes for the system are being tapped in all the compact tension specimens.

All components for utilities, control system, and recording system are on hand except for six flowmeters for the cooling water, the moisture detection instrument, and the data logger (all expected to be delivered March 1976). These items should not delay our schedule.

Preparations for neutron flux and spectrum determinations are in progress with the first BSR flux monitor exposures planned for the week of April 5, 1976. The dummy capsules and monitors have been designed and fabrication is in progress.

It has been our responsibility to relocate the Solid State Division's Thermal Neutron Facility (D_2O Tank). This work was nearly complete but has been temporarily stopped due to a planned change in the facility. Any modifications to the facility should be resolved shortly. This should not affect our schedule since one of our options is to complete the relocation according to original design.

Parts for the irradiation capsules are being received from the fabricators and assembly should start March 8, 1976. We expect to start irradiation May 5, 1976.

Task 5: Simulated Service Testing — Techniques for measuring residual stresses in the V-9 prolongation, which was used for the Section XI weld repair procedure qualification for ITV-7, are being tested on stress-relieved bars with various surface preparations. The results provide a means of improving techniques and qualifying measurements in the repair weld.

Heater control cabinets and a fast-response trigger for the V-7A data acquisition system have been completed. Modifications to the trigger circuit will be made to accommodate instrumentation changes necessitated by the rupture zone patch.

Qualification weldments for the patch have been made and tested successfully. A system for vessel rupture detection with a leak-proof patch is being tested. Instrumentation and data acquisition system design is complete, except for the incorporation of the rupture detection system.

The precise location of the crack tip by ultrasonics during the test of V-7 provided important information on stable crack growth. Since the rupture zone patch precludes ultrasonic ranging of the crack tip from inside the vessel, a scheme for making equivalent measurements from the outside surface is being tested. The first test indicates that the scheme will probably be successful.

Further analyses of through-cracked vessels have been made to illuminate the relationship of burst pressure (i.e., the pressure required to sustain an axially running crack) and crack length. Burst pressure versus crack length for model vessel 10 (the vessel which modeled the longest flaw) confirms the analysis applied to ITV-7.

Specimens have been prepared from the shielded metal arc weld metal qualification prepared by Combustion Engineering. This weld represents the procedure employed in the repair of vessel ITV-7. Miniature tensile, standard 0.505 diam tensile, Charpy-V, precracked Charpy and 1T compact tension specimens are scheduled for test. Testing has begun and should be completed in March.

Task 6: Thermal Shock – A meeting was held on February 2, at ORNL, with the Thermal Shock Review Committee to discuss results of TSE-1 and -2 and a proposal for TSE-3 and -4. The committee suggested that the effect of residual stresses at the tip of electron-beam-weld-induced flaw be studied experimentally with simple flat specimens. Also, a request was made by the committee to calculate the effect of flaw slot in TSV-1 on temperature distribution and thermal stress. This work is in progress.

The ORNL proposal for TSE-3 was accepted by the committee, but a decision on TSE-4 was deferred.

The crack zone in TSV-2 was trepanned from the test specimen and has been cross sectioned for metallographic studies. Measurements of crack depth at the cross sections indicate that the final crack shape is approximately semielliptical (no tunneling) and that there was essentially no extension at the bottom of the semicircular flaw.

Modifications to the thermal shock test facility for reducing the asymmetry in cooling of the test specimen are still under way.

Finite element calculations were made for a continuous circumferential crack in the reference-calculational-model vessel (typical PWR vessel). The results show that in a high copper vessel the crack-depth range for initiation is 0.01 to 0.49 a/w, and that assuming $K_{Ia} = K_{IR}$, a crack could propagate to a maximum fractional depth of 0.71.

Task 7: Reheat Cracking – Metallographic examination of the heat-affected zone from ITV-4 has been suspended. A report discussing the results to date is in preparation.

PROGRAM TITLE: Fission Product Beta and Gamma Energy Release

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

ACTIVITY NUMBER: 40 89 55 10 5 (189a Number B0095)

TECHNICAL HIGHLIGHTS

During February, a report entitled, "Preliminary Fission Product Energy Release Measurements for Thermal-Neutron Fission of ^{235}U," describing interim results of Fission Product Energy Release was sent to press, satisfying cur Buff Book Level "C" milestone node No. 28030. Based upon guidance received from NRC at the midyear review, we will delay initiating consideration of the ^{239}Pu until after definitive data-taking runs for ^{235}U have been completed. The first of these runs is scheduled for April, and the second for early May. Therefore, we expect to delay completion of Level "C" milestone node No. 28012, "Determine System Changes for ^{239}Pu Reliability and Safety," until May 15, 1976.

As mentioned last month, The large volume Ge(Li) detector was delivered. As soon as it was determined that the detector met specifications, a lead-shielding "cave" was designed. Fabrication was held up for several weeks, however, due to the difficulty in obtaining clean lead. The "cave" was completed reducing the background counting rate to <1 count/sec integrated between 0.05- and 3-MeV photon energy. Absolute efficiency calibrations were initiated; that for source-to-detector distance D of 20 cm was completed. Calibrations for D = 2, 5, and 10 cm are in progress.

Two styles of thin-window sample holders for beta-ray measurements were tested, and for both styles several blank (i. e., no sample) holders were successfully sent to the reactor and returned to our counting area without mechanical failure. One style involves mounting 5 mg/cm^2 mylar using a slow-drying adhesive; the other involves using a special self-sticking tape of 8 mg/cm^2. Both styles increase "background" counting rates by moderate and comparable amounts. Further testing is scheduled for March.

PROGRAM TITLE: Fission Product Release from LWR Fuel

PROGRAM MANAGER: A. P. Malinauskas

ACTIVITY NUMBER: 40 89 55 10 8 (189a B0127)

TECHNICAL HIGHLIGHTS:

Two additional implant tests were completed. Implant Test 6 was a 20-hr experiment which was performed at 500°C, and Implant Test 7 was a 5-hr test at 700°C. The same types and quantities of fission product simulants (CsOH, CsI, and TeO_2) as were used in Implant Tests 3-5 were employed, but Implant Test 7 was conducted in dry air rather than the usual 80% steam-20% argon atmosphere. In addition, a 1.6-mm diameter hole was drilled in the center of both rods, which is approximately the average rupture opening that had been obtained in the previous runs by applying internal argon pressure. This method of simulated rupture permits a more controlled study of the transport of the vapor species exiting the fuel rod.

Only a small release of the simulants from the fuel rod occurred in Implant Test 6 (0.5% Te, 0.8% Cs, and 0.6% I), and approximately half the released iodine was of the elemental form. Elemental iodine accumulation in the impactor-silver screen collector was monitored and found to be essentially linear during the 20 hour run. Approximately 45% of the tellurium left the pellet interfaces and deposited on the inner cladding surface.

In Implant Test 7, conducted in dry air, swelling (probably due to the formation of higher fuel oxides) occurred at the hole location. The external surface of the fuel rod was pinkish in color, rather than the typical black (ZrO_2) seen in the previous implant tests which used steam atmospheres.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: Robert H. Chapman

ACTIVITY NUMBER: 40 89 55 10 6 (189a No. B0120)

TECHNICAL HIGHLIGHTS:

Two prototype heaters were received from SEMCO and evaluated during February. Fabrication of these heaters incorporated techniques and procedures developed by SEMCO and D. L. Clark (consultant to ORNL) during Clark's visit of January 12–22 to SEMCO's production facilities. Infrared characterization of the two heaters showed one to be of exceptional quality and the other to be unacceptable. This was disappointing, since considerable attention was devoted to fabrication of the heaters. Further examination of the heaters revealed evidence that the unacceptable heater had been swaged first over its full length, followed by a second swage over half the length. A third prototype has been received and is undergoing evaluation.

Six heaters remain to be delivered by SEMCO on the original order for 20 heaters; these are in production and are promised for delivery in early March. Based on the experience gained in the fabrication and acceptance inspection of these six heaters, SEMCO will prepare a quotation for the 40 heaters on option. Until that is accomplished, cost and delivery of these heaters (for the first two 4 X 4 bundles) cannot be established.

One tantalum sheathed type S and three tantalum sheathed type K thermocouples were received from SEMCO. The shipment completes and provides two extra thermocouples (a total of 47) of the latter type and leaves four of the former type to be shipped. Two thermocouples of each type were calibrated to 1350°C; the two type K thermocouples were within 0.5°C of each other and within ±0.375% of the NBS standard up to 1100°C. Both read low by about 12°C (~0.9%) at 1350°C. The performance above 1100°C appears to be affected by electrical shunting. The quality of the two type S thermocouples was rather poor; at 1000°C one was low by 10°C and the other by 20°C. Both exhibited large (–65 and –165°C) decalibration errors on cooling from 1350 to 1000°C. Based on this information, each type S thermocouple would require calibration before use in a high-temperature test. The performance of the tantalum sheathed type K thermocouples in the calibration experiment indicates they are acceptable for use in MRBT test at temperatures up to 1350°C; this alieviates the need to procure type S thermocouples.

Hermetic end seals have been applied to 55 of the 60 stainless steel sheathed type S thermocouples purchased for use with exposed junctions on the exterior of the Zircaloy tubes in the 4 X 4 temperature mapping experiment. The remaining end seals will be applied in early March and the thermocouples stored for use when needed.

Progress was made on evaluation of errors introduced by cold working the sheathed and bare wire thermocouples during fabrication of the test assemblies.

Three single rod simulators were fabricated and tested during this month; this initiates the first group of 15 single rod tests and includes all the features planned for these simulators. It has been decided to delay use of a heated shroud until the second group of single rod tests, since additional development will be required. Since the data have not been analyzed, preliminary results are not yet available.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: William B. Cottrell

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

TECHNICAL HIGHLIGHTS:

During the month of February the staff of the Nuclear Safety Information Center (a) processed 923 documents, (b) responded to 101 inquiries (of which 59 involved the technical staff), and (c) made 37 computer searches (of which 4 involved payment). Design Data Sheets were prepared on Peach Bottom Unit 1 (an HTGR) and on the two-unit Fulton Plant (also HTGR - but recently cancelled). In addition, an indexed bibliography of the 137 ACRS reports received in January 1976 was prepared and is in reproduction. NSIC staff received 9 visitors during February, including Norman Sarry of NRC. NSIC staff participated in two meetings.

Two NSIC reports, ORNL-NSIC-120 "Annotated Bibliography of Hydrogen Considerations in Light-Water-Power Reactors" and ORNL-NSIC-121 "Reactor Operating Experiences 1972–1974," were distributed during the month. Two other reports, ORNL-NSIC-123 "Nuclear Power: Accident Probabilities, Risks and Benefits; A Bibliography" and ORNL-NSIC-124 "Index to *Nuclear Safety*" (Volume 11—Volume 16), are in reproduction and should be distributed early in March. Another report, ORNL-NSIC-118 "Siting of Nuclear Facilities, Sections from *Nuclear Safety*," is in composition. Work is underway on several other reports, including "A Bibliography of LMFBR Safety" which because of its size will be prepared in two volumes, and Volume V of NSIC-55 "Design Data and Safety Features of Commercial Nuclear Power Plants."

In response to a request from NRC, NSIC has submitted a proposal to assist NRC in two tasks; (1) the preparation of the 1974 and 1975 report on Nuclear Power Plant Operating Experience, and (2) the reclassification of pre-1975 reportable occurrences (back to 1969) according to 1975 standards.

NSIC's special selective dissemination of information (SDI) is available to paying users (as well as additional non-paying users). During the month of February we added one free and one paying ($515) user, bringing the total SDI users to 360, including 51 subscribers who have paid a total of $12,488 (since October 1975).

During the month we wrote two letters (February 24 and 29) to NRC regarding the translation of 5 German documents (of these, 1 was in English, 2 were recommended for translation, 1 not and 1 maybe). The 3 Japanese reports received in January are still being evaluated.

Nuclear Safety 17(1) was distributed by TIC in mid-February. Composition of *Nuclear Safety* 17(2) was virtually complete and was expected to be delivered to the printer early in March. All technical articles for *Nuclear Safety* 17(3) have completed their ERDA and NRC review; the current events material (covering January and February) for that issue is being prepared. Manuscript for all technical articles for *Nuclear Safety* 17(4) are in hand and in various stages of processing.

PROGRAM TITLE: PWR Blowdown Heat Transfer-Separate Effects

PROGRAM MANAGER: D. G. Thomas

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

TECHNICAL HIGHLIGHTS:

Task 1. The first blowdown in the FCTF from 2250 psi occurred during this period. The operating parameters were: test section inlet temperature - 550°F, test section outlet temperature - 632°F, heater power - 122 kW. The break area was 0.0051 ft^2; time to first indication of critical heat flux was 5.2 sec. The boiling crisis first occurred in the downstream end of the highest heat flux zone of the heater. The time to critical heat flux and the location of the first boiling crisis were similar to those occurring in earlier 1500 psi blowdowns with the same break area.

The second production heater rod has undergone 14 on-off cycles and 4 blowdowns in the FCTF with 29.3 total hours of powered operation. A second special calibration test to provide data for development of the thermocouple calibration code, ORTCAL, was run.

Task 2. Analysis continues on the pump-pressurizer interaction for the RELAP4 loop model. The pump coastdown performance and pressurizer level have come under close examination. Great improvement has been made in the hydraulic response to the model and new measurements are scheduled to verify the analytical findings.

Three dynamic gap models are under evaluation for the THTF heater rods. The data from the FCTF has been processed by ORTCAL and is being used as the basis for the dynamic gap model development. The current results are encouraging. The effective thermal conductivity for the BN insulator in the heater rods is being calculated. This is the first step in a four part sequence necessary to develop ORSIN and provide fundamental information for ORINC (inverse codes for calculating heater rod skin temperature, heat transfer coefficients, heat fluxes, etc.).

The FORTRAN plot code THPLOT was successfully modified to provide both SI and British unit labels. All that remains to be modified is the data deck for this code to incorporate the instruments which are to be published in the Quick Loop Report. Initial studies are underway to determine what is needed to provide calibrated drag disk readings and mass flow rates. The FOCAL code FOPLOG (heated rod operator's log) has undergone preliminary shakedown during loop calibration runs conducted recently.

Task 3. During the second hot instrument calibration run, the insulation resistance of several bundle thermocouples was measured at different loop temperatures. Measurements were made (1) from the thermocouple sheath to building ground and (2) from the thermocouple positive lead to building ground, on five center and seven sheath thermocouples. Normally, the sheaths of the center thermocouples are strapped to ground. During these tests the ground strap was disconnected; however, the sheaths of a group of center thermocouples were still electrically connected and so the readings represented several thermocouple sheaths tied in parallel. A 50 V dc megger with a minimum reading of 90 K ohm was used. Insulation resistance decreased significantly as the loop temperature increased. Center thermocouple insulation resistance sheath-to-ground (essentially sheath to heater element) was $\sim$18 M ohm at ambient, $\sim$200 K ohm at 400°F, and <90 K ohm at 600°F. Thermocouple insulation resistance element-to-sheath varied widely, from $\sim$3 M ohm at ambient and $\sim$150 K ohm at 600°F, to $\sim$970 M ohm at ambient and $\sim$420 M ohm at 600°F. Further evaluation of these and previous ambient measurements will be required to assess the effects on bundle temperature measurements and to determine if bakeout of heaters and thermocouples will be necessary.

A significant effort will be required prior to test 101 to tune the process control loops for operation with rod power. Primarily, this involves the automatic control loops associated with the four heat exchangers. Heater rod power is required to establish a sufficient loop temperature for tuning and to permit the insertion of loop perturbations to test control action.

The checkout and final verification of several items are awaiting the completion of the refurbishing of the dc generators. The primary items are (1) final checkout of the generator voltage, generator current, and rod current isolation amplifiers, and (2) final verification of the safety interlocks related to bundle power.

Refurbishing of the dc generators was completed on 2/23/76. Measured resistances of the generators at various stages in the refurbishing were (500 volt megger):

Generator number	Original 12/75	After cleaning and wrapping	After final varnishing
9	0 Ω	0.58 M Ω	1.08 M Ω
10	10 K Ω	1.18 M Ω	3.5 M Ω
11	10 K Ω	0.72 M Ω	1.8 M Ω
12	70 K Ω	1.32 M Ω	3.0 M Ω

The increase in resistance is sufficient to permit operation of the THTF and crowbar checkout will begin on 3/1/76.

Task 4. Calibration of THTF two-phase flow monitoring spool pieces has continued and computer codes are being prepared for the calculation of mass flow rates from THTF instrument signals. THTF drag disk calibration was conducted over a temperature range from ambient to ~600°F (315°C) using the pump as an energy source. The test showed slight variations from previous tests in instrument zero shift, but no significant change in the effects of drag crisis. Codes being developed include the liquid phase calibration for conversion of instrument signals to engineering units, effect of operating temperature on instrument "zero shift" and drag crisis of drag disks, calculation of mass flow and quality by a two-velocity (drift flux) model, and the results of the steady-state two-phase (air-water) calibration of the instruments.

Task 5. Watlow Electric Manufacturing Co. began the fabrication of THTF bundle 2 pre-production fuel pin simulators on 2/16/76.

Investigations of the possible advantages of using a copper instead of stainless steel inner sheath on the THTF fuel pin simulator are continuing. A new heater design is being developed with Watlow for possible use in future bundles which includes a continuous wound coil and a single heater sheath with thermocouples spot-welded to the inner surface of the sheath. RAMA Corp. has agreed to fabricate a test unit of a design similar to that of the ANC semi-scale heater but with a different coil to meet ORNL power supply requirements, completion is expected by ~3/19/76. ORNL will evaluate this unit and, if indicated, will order a full scale prototype for further investigations aimed at qualifying RAMA to bid on bundle 3 fuel pin simulator order.

The Claude Gordon Co. is presently scheduled to ship 400 thermocouples for bundle 2 fuel pin simulators on 3/8/76 with the balance due on 3/31/76. This schedule, if met, will supply sufficient thermocouples to Watlow to meet their production schedule. Kaman Sciences has shipped 150 thermocouples to date for bundle 3 simulators and is scheduled to ship 386 on 2/27/76 with the balance due on 3/31/76.

PROGRAM TITLE: Zircaloy Fuel Cladding Collapse Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: 40 89 55 10 7 (189a Number B0124)

TECHNICAL HIGHLIGHTS:

The results of the collapse-study test matrix have been published as ORNL-TM-5279, *Quarterly Progress Report on the Creepdown and Collapse of Zircaloy Fuel Cladding Program Sponsored by the NRC Division of Reactor Safety Research for October–December 1975.*

Further testing is being conducted to examine the effects of annealing on the collapse properties of the cladding. It has been found that room-temperature hardness is an additional parameter that must be evaluated. We are examining the interactions among the variables both statistically and graphically to form a model for predicting collapse behavior.

Both dimensional and x-ray texture characterizations are being performed on a representative sample of the tubing.

PROGRAM TITLE: Zirconium Metal-Water Oxidation Kinetics

PROGRAM MANAGER: C. J. McHargue

ACTIVITY NUMBER: 40 89 55 10 9 (189a Number B0128)

TECHNICAL HIGHLIGHTS:

The Thermometry Error Analysis report was issued on February 12, 1976, thus completing the requirements for Milestone 43090 (Levels B and C).

<u>PROGRAM TITLE:</u> Aerosol Release and Transport from LMFBR Fuel

<u>PROGRAM MANAGER:</u> M. H. Fontana

<u>ACTIVITY NUMBER:</u> 40 89 55 11 1 (189a Number B0121)

<u>TECHNICAL HIGHLIGHTS</u>:

<u>CDV Development</u>:

Five capacitor discharge vaporazation (CDV) test assemblies were prepared for the next series of experiments to be carried out at the Arnold Engineering Development Center (AEDC). The developmental tests at AEDC will be completed and a draft report on the CDV development will be prepared by July 1976.

<u>CRI-III Facility</u>:

The ORNL CDV system for use in the upper limit source term tests is being assembled. A shipment consisting of 45, 2500 V capacitors, for construction of the first two 60 kJ banks at ORNL has been received and installation begun. Capacitors and ignitrons for future banks will be ordered as needed.

Orders for the transient power control and recording instrumentation and the Hycam movie camera are being prepared. Requests for proposals for a high-speed, high-temperature pyrometer have been solicited without much success. One company, Ircon, will probably reply with a modified version of an earlier NBS photodiode device which would require local calibration.

The electric-arc hearth furnace is nearing initial operation. The first practice run will use a metallic charge for safety review. The CRI-III aerosol vessel has been completed, but the attached discharge chamber which houses the fuel assembly and viewing ports will not be completed for a few weeks.

NSPP:

Reconditioning of the NSPP facility is continuing on schedule. Recalibration of the panel board instruments to be reused is about 80% complete and reassembly of the panels is under way.

Requisitions have been issued for the hot-wire anemometer in-vessel flow instrumentation, and bids have been received for the instrument supply air compressor and air dryer. Price and delivery quotations on all these items appear satisfactory and compatible with our schedule.

Bubble Transport:

Design of the sodium test facility continues. The fuel pellet vaporizer, assembly, the second fill and drain tank and view port design were completed and requests issued for material procurement and fabrication. The test vessel design was also finalized and a request prepared for procurement of the vessel. Structural drawings for the facility were reviewed and approved for issue.

Work has continued on developing a preliminary bubble transport and source attenuation model.

The program, PAD, was obtained from Los Alamos and is being converted to operate on the ORNL computer system.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: J. P. Sanders

ACTIVITY NUMBER: 40 89 55 11 2 (189a Number B0122)

TECHNICAL HIGHLIGHTS:

General: Further work was done on the development of the BLAST, ORTAP, and ORECA codes. Progress was made under the University of Tennessee (UT) subcontract in the areas of steam generator model development and preparation for the Fort St. Vrain reactor dynamics tests.

Reheater and Steam Generator Model Development (BLAST): The water equation of state routines were expanded to provide more accuracy in the pressure range from 18.616 MPa (2700 psi) to 21.374 MPa (3100 psi). A tape of the BLAST Fortran was prepared and is being transmitted to LASL for use in verification of a simpler model of the steam generator used in the CHAP code. Documentation of BLAST is complete and is going through an internal laboratory review. Transient calculations for small perturbations of the FSV steam generator from 100% power steady state conditions were made in order to compare the results of BLAST to results obtained by the University of Tennessee with General Atomic's LAP* computer program.

Core Emergency Cooling Analyses: The TM report on the ORECA code was completed for delivery to the reports department. A code for simulating the dynamics of the core auxiliary cooling tower (CACT) and associated piping was completed and debugged.

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*C. K. Tang and C. F. Lehner, "Linear Analysis Program Documentation," General Atomic Co., GA-A13684, Nov. 1975.

PROGRAM TITLE: Design Criteria for Piping and Nozzles

PROGRAM MANAGER: S. E. Moore

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

TECHNICAL HIGHLIGHTS:

Code Rules Development: The draft of a report entitled *Appropriate Nominal Stresses for Use with ASME Code Stress Indices for Stresses Due to Pressure at Nozzles in Pressure Vessels* was written to point out a number of conflicting statements in the Code concerning the use of stress indices for the design analysis of pressure vessel nozzles. A case history review indicates that considerable confusion concerning the subject exists, and as a result what we believe to be inappropriate changes to the Code rules have been made (June 1975). Hopefully, the suggestions given in this report will be used by ASME to correct some long standing errors, and to prevent other errors of this type from being made.

Topical Reports: We completed preparation of a draft report on the experimental stress analysis of a series of four 10-in. ips machined long-radius piping elbows under internal pressure and external force and moment loadings, and combinations of pressure and moment loadings. Data obtained from these tests will be used in conjunction with results from analytical studies in the evaluation of current design rules.

Spherical Shell Studies: A draft report on elastic stresses at reinforced nozzles in spherical shells under internal pressure and external moment loadings was completed. This report presents a complete summary of all presently planned work on the subject of stresses at isolated nozzles in spherical shells, and gives an evaluation of current design rules based on the available data.

Cylindrical Shell Studies: Two projects in this category are currently underway: (1) the development of a finite element computer program for the stress analysis of closely-spaced nozzles in cylindrical shells, and (2) a finite element parameter study of the stresses at isolated nozzles in cylindrical shells. Under the first item, the overall computer program specification and the mesh generator portion of the program have been completed. We are currently working on modifications to the main frame to add the 8 to 20 variable node isoparametric element to the library, and to incorporate a more efficient matrix solver. We expect to complete these modifications during the next month.

Under the second item, about 20 of the 27 models in this study have been analyzed using the computer program CORTES-SA. These include six unreinforced nozzle models and fourteen models with the so-called standard reinforcement. We are currently analyzing the "pad-reinforcement" models and preparing summary material for the report.

<u>ANSI B16.9 Tee Studies</u>. Work is continuing on the preparation of a summary report on the elastic response of a series of five 24-in. ips ANSI B16.9 tees tested earlier at Combustion Engineering, Inc. We are also preparing the report on the elastic response tests of a series of five 12-in. ips ANSI B16.9 tees that were tested at Southwest Research Institute for publication.