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ORNL NUCLEAR SAFETY RESEARCH
AND DEVELOPMENT PROGRAM
BIMONTHLY REPORT
FOR NOVEMBER—DECEMBER 1972

Non-LMFBR Programs

Wm. B. Cottrell

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MASTER

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MARCH 1973

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ABSTRACT

The accomplishments during November and December in the research and development program under way at ORNL as part of the U.S. Atomic Energy Commission's Nuclear Safety Program are summarized. Much of this work is in general support of water-cooled power reactor technology, including LOFT and industry programs, and reflects current safety problems, such as blowdown heat transfer studies. Other major projects include experimental investigations of the attachment of nozzles to shells and the implementation of joint AEC-PVRC programs on heavy-section steel technology and nuclear piping, pumps, and valves. Several projects are also conducted in support of the high-temperature gas-cooled reactor (HTGR). Seismic considerations of particular significance in nuclear plant design are evaluated in the seismic program. Several of the above projects are also directly related to another major undertaking, the AEC's standards program, which entails development of engineering safeguards and the establishment of codes and standards for government-owned or -sponsored reactor facilities. The recent activities of the Nuclear Safety Information Center (NSIC and the *Nuclear Safety* journal in behalf of the nuclear community are also discussed. Of particular interest in both of these activities is the emphasis on environmental effects. LMFBR safety studies are reported in a separate document.

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SUMMARY

1. LWR SAFETY STUDIES

1.1 PWR-BDHT Separate Effects Program Scope and Coordination

Program manpower has been adjusted to accommodate the altered budgets; the primary effect has been the postponement of some aspects of facility design and, consequently, of loop construction. The overall cost is a significant delay in test startup, but computer development of alternatives for shortening the schedule are continuing. The System Design Description was reviewed, and it will be necessary to upgrade the invoked quality to the base reference level imposed by AEC-RDT and justify any exceptions thereto. A hold has been placed on the procurement and/or fabrication of system components until this revision to the SDD is completed and accepted.

1.2 Fuel Rod Simulation (Task 1)

Two additional blowdown tests were carried out with ANC heater S/N-8. The blowdowns were from an initial rod power level of 53 kW (80% of full power), a system pressure level of ~ 1500 psia, and a coolant inlet temperature of 550°F. In one test, 10 kW of "decay heat" was applied for the final two-thirds of the blowdown, and in the other the decay heat was zero. Evidences of electrical arcing were noted on the heater surface at the locations of spacing and centering devices following the first run. Alumina spray coating of the spacer pins and unheated rod extension, along with improvement in ground connections, before the second run should have corrected this problem; however, examination of the heater rod has been postponed until completion of additional tests. A gamma densitometer has been obtained which will be used in conjunction with a drag-disk meter in two-phase flow measurement tests.

1.3 Experiment Plans and Evaluations (Task 2)

Computations have been continued using RELAP 3 to model the FCTF with a single heated pin. The calculated blowdown transient was found to be

more rapid than that measured; further, RELAP 3 indicated CHF within the first second of blowdown, while the data showed decreasing cladding temperatures. The first of these discrepancies is believed to be associated with the additional flow resistance deriving from flow through the ragged opening in the rupture disk, and the second appears to have been resolved by modifications to one of the subroutines in the code. Convergence problems with THETA 1-B for low fuel-rod gap conductivities continue; extrapolation and oscillation-damping routines have been introduced to overcome this problem.

1.4 Multirod Transient and Blowdown Heat Transfer Experiments (Task 3)

In consequence of the funding limitations, the THTF design effort will be limited to the test section and the power supply controls; these are currently projected as the critical path items. Final heater element design indicates an Inconel 625 tube with a minimum ID of 0.220 in., minimum wall thickness of 0.013 in., and four tapers. Preliminary butt-weld tests made with machined Inconel 625 tubes 15 mils in wall thickness indicate that satisfactory welds can be made with an ungrooved alumina backup pellet but that a metallic insert may be needed if a slight amount of weld-bed concavity cannot be tolerated. Inspection procedures for tapered tubes are being examined using two stainless steel tubes borrowed from Superior Tube Company. Design based on information from Karl Roesch, Inc., continues to concentrate on the heater rod-rod bundle-flow shroud-pressure housing complex and on modifications to the dc power supply to effect parallel operation of the generators. Heater rod thermometry has been fixed for the first bundle at a maximum of six insulated-sheath couples per rod and no area thermal sensors; fluid temperatures will be obtained in each exit subchannel, at the pressure housing inlets and exits, and along the length of the outermost flow regions in the bundle heated zone. Development of instrument application drawings was continued, and some design changes have resulted from closer study of these drawings.

2. HTGR SAFETY STUDIES

2.1 Reactions of Steam with Graphite

Experimental data are reported for the studies made at 1000°C with H-327 graphite at steam partial pressures of 0.0450 and 0.0700 atm. Comparison of these data with results reported earlier for 0.095 atm steam partial pressure indicates a complicated dependence upon this parameter; this may be partly a manifestation of inhibition by the reaction products.

2.2 Fuel Integrity and Fission Product Release

Further comparisons of cesium diffusion rates in different types of pyrocarbon were made. Better particle characterization is apparently needed to explain observations. Equipment for the study of particle stability at high temperatures and carbon transport in particles was manufactured.

2.3 Fission Products in the Coolant Circuits

The experimental system for studying the desorption of iodine from HTGR coolant circuit materials in a flowing helium-steam atmosphere was rebuilt and tested. Thermochemical calculations of Cr-I-CrI₂ and Ni-I-NiI₂ systems were completed.

Measurements of deposited activities in the primary-coolant circuit of the Peach Bottom HTGR made in May 1972 are reported, in which a significant reduction in the surface concentration of cesium was seen. Analysis of a filter removed from a position downstream of the hot samples was begun, and preparations for postirradiation examination of a Peach Bottom fuel element was completed.

2.4 HTGR System Safety Analysis

Preparation of our HTGR Safety Program Plan continues. In addition, consideration has been given to the form that a unified safety analysis capable of being used to assign program priorities should take.

3. HEAVY-SECTION STEEL TECHNOLOGY PROGRAM

3.1 Program Administration and Procurement (Task A)

During this reporting period four HSST reports were published, and two were approved for publication.

3.2 Material Inspection and Control (Task B)

The fabrication of the remaining 6-in.-thick intermediate test vessels remains a significant procurement item monitored under this task.

3.3 Variability in Plate, Heat-Affected Zones, Welds, and Forgings (Task D)

Series of precracked Charpy specimens continue to be tested as part of the material investigations of the intermediate pressure vessel tests with major emphasis on slow bend testing.

3.4 Fracture Mechanics Investigations (Task E)

The pop-in arrest toughness investigation continues at Materials Research Laboratory. Evaluation of plane-strain fracture theories and the determination of lower-bound fracture toughness of submerged-arc weldment continues at Westinghouse Electric Corporation. An elastic-plastic three-dimensional stress-strain analysis of the nozzle corners of the intermediate test vessels containing a nozzle is being initiated by Marc Analysis Research Corporation.

3.5 Fatigue and Crack Propagation (Task G)

The fatigue crack propagation in a reactor environment, excluding irradiation, in combination with low cycling rates (1 cpm) was seen to be significantly greater than at higher cycling rates (60 to 600 cpm). This work continues. Effects of cycling rates and irradiation are also planned for investigation.

3.6 Irradiation Effects (Task H)

Three capsules each containing two 4T (9.6 by 10 by 4 in. thick) compact-tension specimens are in the Battelle Research Reactor. The target

fluence is 5×10^{19} neutrons/cm², $E > 1$ MeV, irradiated nominally at 550°F. Over 200 smaller specimens are also included in the capsules. The six large specimens consist of two longitudinal specimens, two transverse specimens, and two submerged-arc weld specimens. The work is subcontracted to Westinghouse Electric Corporation.

3.7 Simulated Service Test (Task K)

At ORNL, the third and fourth 6-in.-thick intermediate test vessels were tested at 130 and 75°F respectively. Each contained a fatigue sharpened flaw in an axially oriented submerged-arc weld seam. The fourth vessel also contained a flaw in the base metal. The nominal outside surface circumferential strain away from the flaw was 1.44% in vessel 3 and 0.168% in vessel 4. The difference in behavior is accounted for by the difference of toughness with temperature and flaw size.

3.8 Specific Safety Research (Task L)

The investigation of the extent of ductile pipe rupture continues at Battelle Columbus Laboratories. At ORNL, the lower-bound toughness investigation of piping material is in the preliminary stage.

4. DESIGN AND ANALYSIS OF NOZZLES, PIPES, PUMPS, AND VALVES

4.1 Design Criteria for Piping, Pumps, and Valves

The ORNL Piping Program is part of a cooperative effort between the USAEC and private industry to study the structural behavior of piping system components and to develop basic stress analysis information for use in writing and improving design codes and standards. The ORNL program consists of carefully coordinated analytical and experimental stress analysis studies on piping system components such as straight pipe, lug attachments, elbows, tees, reducers, transition joints, and welds. Stress indices developed for many of these components have been incorporated into the ASME Boiler and Pressure Vessel Code. Close cooperation with private industry is maintained through the Pressure Vessel Research Committee (PVRC) of the Welding Research Council.

Experimental stress analysis tests are currently being conducted on 24-in. ANSI standard B16.9 tees under subcontract at Combustion Engineering. Elastic-response tests and low-cycle fatigue tests have been completed for three tees; the elastic-response tests were completed for the fourth tee, designated T-12, this period. The elastic-response data for all 12 loading conditions on T-12 have been examined at ORNL and accepted from the subcontractor. Based on these data, test loading conditions for the cyclic pressure fatigue test were established. The tee will be pressure cycled between limits of 100 and 1800 psi; failure is expected in 20,000 to 30,000 cycles. A fifth tee (T-16) is currently being instrumented with strain gages for testing later this fiscal year.

The steady-state thermal stress tests being conducted at ORNL on two 12-in. tees were temporarily suspended. The earlier difficulties associated with brittle failures of the weldable strain gages were diagnosed and corrective procedures developed. Temperature profiles and strain levels obtained from one of the tees (T-6) during shakedown tests, however, were not satisfactory. Modifications in the test rig will be necessary before the test work can be resumed. We plan to reactivate the tests next fiscal year.

The elastic analysis phase of the B16.9 tee finite-element computer code development work at the University of California, Berkeley, has been completed. Three programs have been written: one for elastic stress analysis of tees with applied mechanical loadings (i.e., forces, moments, and internal pressure), one for transient and/or steady-state thermal analysis, and one for thermal stress analysis. These programs are now operational at ORNL and are being checked against experimental data obtained from B16.9 tees tested earlier. A fourth computer program, for the elastic-plastic stress analysis of tees, is currently being developed at Berkeley. This program is essentially an extension of the elastic program with provisions for including elastic-plastic constitutive equations. Several basic extensions were completed this period.

The last model (ME-4) in the series of machined elbows being experimentally stress analyzed at ORNL is currently being tested. Complete sets of strain-gage data were obtained for internal pressure, and in-plane bending, out-of-plane bending, and torsional moments applied to one end of the

model. We plan to complete the combined loading cases (internal pressure plus moment) during the next reporting period.

4.2 Experimental and Analytical Investigations of Nozzles

Experimental and analytical studies of single nozzles and clusters of nozzles attached radially and nonradially to flat plates, spherical shells, and cylindrical shells are continuing. Three experimental models in the study of single radial and nonradial nozzles attached to hemispherical shells remain to be tested at the University of Tennessee. Testing of the 7 7/8-in.-OD flush radial nozzle model 807-003 ($d_i/D_i = 0.249$, $D_i/T = 80.35$, and $d_i/t = 40.0$) has been initiated. The final 2 5/8-in.-OD, 22 1/2° nonradial nozzle configuration, model 802-223 ($d_i/D_i = 0.083$, $D_i/T = 80.35$, and $d_i/t = 40.0$), is being partially reinstrumented with strain gages, and the 7 7/8-in.-OD, 45° nonradial nozzle model 807-451 is being fabricated. The equations for the numerical type of analysis for out-of-plane moment loading of nozzles attached to cylindrical shells have been programmed for the computer at ORNL. Further development of this analysis has been deferred to FY 1974 for budgetary reasons.

Data processing of the results of tests conducted at Auburn University on flat plates containing one and two nozzles is nearing completion, and testing of the flat plate having a five-nozzle cluster is in progress. The first stage of the instrumentation scheduled for the Auburn University hemispherical shell with four attached nozzles has been installed, and the model is being prepared for testing.

An analytical study has been conducted at Battelle/Columbus Laboratories to determine whether the widely used axisymmetric methods of analysis could be employed to analyze nonaxisymmetric nozzle-to-cylindrical shell configurations. Although it was initially felt that models of this type having relatively small d/D ratios could probably be analyzed with reasonable accuracy, the study showed that none of the three types of axisymmetric methods investigated was capable of providing reasonably accurate prediction of stresses in nozzle-to-cylindrical shell attachments.

5. RDT STANDARDS PROGRAM

5.1 Program Management Activities

A computer program is being considered which will assist in updating the RDT Standards and handle a number of routine functions of the program. Support to the American National Standards Institute (ANSI) continued.

5.2 Mechanical Equipment and Systems

Work continued in preparation and review of drafts and amendments for ASME Code supplement and air handling standards. One amendment to a Code supplement standard was approved and issued.

5.3 Instrumentation, Controls, and Electrical

One amendment was approved and published, and two new standards and two proposed amendments were prepared and submitted to RDT for approval. ORNL review of three standards was completed, and comments were submitted to the standards writers.

5.4 Programmatic and Procedural Standards

Participation in ANSI N45 committee and subcommittee activities and in the Nuclear Division Committee for Interface of the American Society for Quality Control (ASQC) continued. Two proposed amendments to the cleaning standard were prepared and submitted to RDT for approval, review comments are being incorporated into the transportation standard prior to submittal to RDT, and review of the proposed cask tiedown standard was completed.

5.5 Materials and Fabrication

Amendments to 11 materials and fabrication standards were approved and issued; and 1 proposed amendment, 7 revisions, and 1 new standard were completed and submitted to RDT for approval. Drafts of four new standards were completed and ready for redistribution and reviewed for comments.

6. GENERAL NUCLEAR SAFETY STUDIES

6.1 Seismicity and the Seismic Resistance of Nuclear Power Plants

Work continued under the SW-AJA soils study subcontract with the completion of several significant improvements in the in-situ soil test equipment and the performance of field tests with good results at higher soil strain levels. Fabrication of soil test boxes at Berkeley was completed, and initial instrument calibration tests are under way. Two preliminary requests for proposals have been submitted to AEC-RDT for review and comment.

6.2 Ignition of Charcoal Adsorbers by Fission Product Decay Heat

The installation of apparatus to determine experimentally the consequences of radioactive decay energy of iodine adsorbed on charcoal is nearing completion. The safety analysis reports for the PdI_2 irradiation in the HFIR and the experimental work in the hot cell are being reviewed by the safety review committees. A computer program has been written to calculate the heat balance for the charcoal bed and is being used to determine the rate of heat release from oxidation of charcoal that occurred in our 13 laboratory test ignition runs.

7. NUCLEAR SAFETY INFORMATION

7.1 Nuclear Safety Information Center

The Nuclear Safety Information Center was established by the Commission to collect, evaluate, and disseminate nuclear safety information to governmental agencies, research and educational institutions, and the nuclear industry. The Center's basic computer file now contains references on over 71,300 documents. One new NSIC report has been published and is available for purchase from NTIS. The 1970 Directory of Personnel Involved in the Development of Nuclear Standards in the United States (ORNL-NSIC-75) is currently being revised and updated. During the reporting

period, NSIC responded to 284 inquiries, 145 of which involved the technical staff.

7.2 Technical Progress Review *Nuclear Safety*

The November-December 1972 issue of *Nuclear Safety*, a bimonthly Technical Progress Review sponsored by AEC's Office of Information Services, was distributed in December. The issue carried articles that discussed Ispra's contribution to reactor safety, generation of radiolytic hydrogen after a PWR LOCA, irradiated nuclear fuel equation of state, effect of rupture in a noble-gas adsorption bed, the role of the campus radiation-safety officer, June-July safety-related occurrences, RDT-sponsored R&D project reports, and current activities in licensing, regulation, and other administrative areas.

1. LWR SAFETY STUDIES

1.1 PWR-BDHT SEPARATE EFFECTS PROGRAM SCOPE AND COORDINATION

(AEC Activity 04 60 80 02 1)

H. W. Hoffman C. G. Lawson

Studies within the scope of the PWR Blowdown Heat Transfer Separate Effects Program emphasize programmed LOCA-type depressurizations in large (7×7) arrays of 12-ft heated length electric rods simulating nuclear fuel pins; results will provide a measure of the behavior of rod assemblies experiencing blowdown transients as well as input for testing and upgrading existing analytical procedures for describing such transients. This work is supported by experiments evaluating the effectiveness of electric heaters of several types as simulators for nuclear fuel rods and by appropriate analyses.

Program Status

Program manpower has now adjusted to the tightened fiscal situation. The changes required are manifested primarily in a shift from parallel efforts to sequential efforts. Thus, design on the mechanical and electrical modifications to the Thermal-Hydraulic Test Facility (THTF) is being delayed so as to give preference to the critical items of the test section design (heater rods, flow shroud, electrode and support assemblies, instrumentation, and pressure housing) and the dc power supply parallel-operation controls. Accordingly, other design aspects — pressurizer,

pressure-suppression system, holdup volume, and loop piping — are postponed (probably until early next fiscal year). At the same time, this precludes loop construction. While this procedure does accommodate the lower budget, this is done at the cost of an extended time schedule. Preliminary indications suggest a delay of as much as 12 months in test startup; however, computer evaluations are continuing to examine alternative arrangements that might shorten this schedule.

System Design Description

The first draft of the System Design Description (SDD) for the THTF was reviewed with AEC-RDT in regard to the standards applied. In discussions with the Codes and Standards Branch, it was established that the RDT reference for construction is ASME Boiler and Pressure Vessel Code, Section III, Class 2, as augmented by RDT E-15-2T (the SDD invoked only the ASME Code). Accordingly, the SDD will be reexamined in this light. Lesser quality levels are not ruled out by this position; however, it will be necessary to justify for each individual case the differences between the proposed level and the reference base. Further, the RDT standards require a code stamp for all components of the high-pressure system. This may create some difficulties in outside procurement and is a problem in respect to in-house fabrication. ORNL has initiated procedures for code-stamp qualification, but in the interim, it may be acceptable to comply with the intent of the code.

Specific questions were raised in regard to the ability of the Graham heat exchangers and the Bingham pump to withstand the required number of temperature and pressure cycles. We are now analyzing this situation and are accumulating such operational experience as is available. Until this evaluation is complete, we have placed a hold on all outside procurement; this primarily affects the heat exchangers since Graham was ready to begin construction.

1.2 FUEL ROD SIMULATION (TASK 1)

(AEC Activity 04 60 80 02 1)

P. P. Holz C. G. Lawson
R. L. Moore

This task considers the extent to which electrical heater rods experiencing blowdown transients simulate nuclear fuel rods in a similar environment. Single heater rods of both direct (ohmic wall heating) and indirect (internal resistance element heating) types will be tested in the existing Forced-Convection Test Facility (FCTF) to determine such factors as time to CHF, cladding-temperature response, heat-flux variation with surface-to-coolant ΔT , and post-CHF heat transfer patterns. Results from the test sets (separate experiments utilizing a direct heater and an indirect heater designed for the same service) will be compared with the behavior predicted for a nuclear fuel pin.

Loop Operation

Two tests were performed with the ANC 5 1/2-ft indirect heaters¹ during this report period; both runs were at 80% of rated power (53 kW), but only one had decay heat in the latter stages of blowdown. Two additional loop-temperature monitors and one additional pressure gage were installed in the system prior to these tests to improve the record of loop behavior prior to and during blowdown.

Test conditions for these runs are given in Table 1.1; data records indicate that pressure and flow reversal occurred during blowdown for both runs. Detailed analysis to extract heat transfer performance is in progress.

Heater Rod Evaluation

ANC heater S/N-8 was examined following completion of the first 80% power run; total exposure had been two blowdowns from a pressure of ~ 1500 psia and a coolant temperature of $\sim 550^\circ\text{F}$. Evidence of electrical arcing on the cladding surface was observed.

Table 1.1. Conditions for Blowdown Tests with ANC Indirect Heaters

Identification		Rod Power		System Pressure	Coolant Inlet Temperature	Mass Flow at Blowdown	Blowdown Time	Remarks
Run	Heater	%	kW	(psia)	(°F)	(gpm, approx)	(sec, approx)	
11-72-1	S/N-3	80	53	1540	550	11.5	20	<i>a</i>
12-72-1	S/N-8	80	53	1590	550	11.5	20	<i>b</i>

^aPower during blowdown = 53 kW for 6 sec without decay heat.

^bPower during blowdown = 53 kW for 6 sec with 10 kW decay heat during final 12 sec; external heater rod thermocouple (upper center) failed mid-way of heatup prior to blowdown.

The S/N-8 heater hangs vertically within a core barrel that is inside of and concentric with the test-section pressure housing. A jam nut centers the rod at the unheated upper end, and a guide disk serves similarly at the unheated lower end. Multiple centering-screw assemblies maintain concentricity in the heated portion; these consist of axially distributed sets of three 1/16-in.-diam stainless spring-steel wires at 120° circumferential spacing. A 1/64-in. clearance between the rod and the jam nut and between the rod and the guide disk allowed for differential radial thermal expansion and for longitudinal "growth" of the heater rod. The upper unheated end of the heater was provided with a welded-on, corrugated stainless steel strap ground which was also welded to the pressure housing. Since arcing had been noted previously with heater SN-25, heater S/N-8 was modified to include a lower-end ground. This consisted of a stainless steel rod that was slip fitted (0.010-in. diametral clearance) into a recess bored into the lower rod extension and was affixed to a Swagelok fitting in the end cap of the pressure housing.

Typical surface defects due to arcing are shown in the quadrant views in Fig. 1.1 of the lower end of the heater rod where this passes through

Fig. 1.1. Surface Defect in Lower Unheated Extension of ANC Heater Rod S/N-8 Due to Electrical Arcing at Surface. Views are 90° circumferential separation.

the 1/16-in.-thick guide plate. Defect depths averaged 0.005 in., and the maximum penetration was 0.015 in. Similar defects, though of lesser magnitude, were noted at all heated-zone centering locations and within the slip-fit zone at the heater's upper end.

To correct this situation, the unheated rod extensions, each heated-zone spring clip, and the jam-nut bore were plasma-spray coated with aluminum oxide to an average thickness of 0.003 in. Hopefully, this non-conductive coating will act as an electrical insulator to prevent further arcing. More positive electrical grounding at the lower end was provided by welding 1/8-in.-diam, stranded, stainless steel wires from the rod end to the core-barrel bottom and the pressure housing.

Following these modifications, blowdown run 12-72-1 was carried out. Although positive examination of the rod surface for possible arcing was not performed (for economic reasons), the reduced noise on the test section thermocouples suggests improvement.

Instrumentation

Consideration is being given to installation of a gamma densitometer and drag disk in the FCTF for the purpose of obtaining two-phase flow data during blowdown. This will not only be useful in interpreting these task 1 data but is imperative for establishing the operational characteristics and calibration procedures for similar instrumentation to be installed in the THTF.

A gamma densitometer consisting of a 200-Ci ^{60}Co source, shielding, collimator, ionization chamber, electrometer, and high-voltage power supply has been "borrowed" from another Oak Ridge facility. A preliminary evaluation of the design drawings and published performance characteristics indicates that this equipment will be adequate for the FCTF application; however, tests will be performed to verify this conclusion.

Drag-disk-type flowmeters to measure the momentum flux (ρV^2) are not available at ORNL and must be purchased; a bidirectional drag disk suitable for the environmental conditions of the FCTF is commercially available from Ramapo Instrument Company.

Reference

1. ORNL Nuclear Safety Research and Development Program Bimonthly Report for September-October 1972, USAEC Report ORNL-TM-4030, Oak Ridge National Laboratory, pp. 5-8.

1.3 EXPERIMENT PLANS AND EVALUATIONS (TASK 2)

(AEC Activities 04 60 80 02 1 and 04 40 03 01 1)

C. G. Lawson	J. P. Sanders
R. F. Bennett	H. W. Hoffman

This task provides design guidance and analytic support for the experimental tasks of the PWR-BDHT Separate Effects Program. Particular attention is to be given to the modeling of the ORNL blowdown facilities and to the interpretation of the data obtained therefrom in the following areas.

1. Modeling the Forced-Convection Test Facility (for single-rod blowdown tests), extracting the test data from blowdown experiments in this facility, and interpreting the results. This will also serve to test the hybrid application to blowdown analysis (see Section 1.5) and indicate future directions for computer expansion and procedure extension.

2. Modeling the Thermal-Hydraulic Test Facility (for multirod blowdown tests) so as to translate reactor LOCA boundary conditions to the experiment and to interpret the resulting data. This effort will depend heavily on existing digital computer programs. Interfaces with the authors and users of the several available codes will assure not only the necessary rapid information exchange for experiment guidance and code development but also will provide the necessary perspective on the role of these studies in the national effort.

Flow-Facility Computer Modeling

Calculations with RELAP 3

Computations have been continued¹ using the model of the Forced-Convection Test Facility (FCTF) with a single heated rod for the RELAP 3 program. This model has been modified to improve the representation of the experimental measurements by the computer response from RELAP 3. The plotting routine has been used to present the computer data, providing a convenient means of comparison of experimental and calculated values.

Two general problem areas were found in obtaining agreement between calculated and measured values.

First, the rate of blowdown, as indicated by the trace of pressure vs time, was more rapid in the calculated transient than in the observed data. Thus, test results showed that the pressure at the top of the test section dropped from 1575 to 200 psia in 12 sec, whereas the program indicated this pressure change in 8 sec of blowdown time. Two sources of this error have been suggested: (1) the ragged openings of the rupture disks may contribute additional resistance to the ejection of fluid; and (2) the initial void volume between the rupture disks and the downstream critical flow orifice, although small in size, will affect the calculated response of the system. Appropriate modifications are presently being incorporated in the calculational model of the FCTF system.

Second, RELAP 3 has indicated the onset of critical heat flux within the first second of blowdown; this resulted in the calculation of rapidly increasing rod cladding temperatures. The experimental data indicate, however, that rod cladding temperatures decrease for approximately the first 12 sec of blowdown. Aerojet Nuclear Corporation² suggested that modifications in a subroutine recently transmitted to the Argonne Code Center should affect this calculation; preliminary results appear to indicate that these changes have eliminated this disagreement between calculated and measured values.

Calculations with THETA 1-B

The calculations using the sample TROJAN input for the THETA 1-B code have not yet been made to converge in the calculation of the initial

temperature profile when the conductivity of the gas filling the gap was reduced to 10% of the thermal conductivity of helium. This conductivity represents the gas filling the gap near end of life when fission product gases have diluted the helium used initially to backfill the rod.

Two relatively direct methods have been introduced to expedite convergence of this trial-and-error calculation: an extrapolation routine and a routine to damp oscillations in the calculations. Additional output statements have been added to the program to follow the progress of this iterative calculation. It was found that the convergence criteria were obtained at most axial positions early in the series of computations; in this sample problem, there are 11 axial nodes. Further, the convergence criteria were satisfied at all axial positions at some point in the series but never at all 11 axial positions during the same trial. Axial conduction of heat in the cladding connects the nodal points and perturbs the calculations at a nodal point even after the convergence criteria have been satisfied at that axial position.

To facilitate convergence, a subroutine incorporating the Aitken Δ^2 process has been used. This method will be extended to the two-dimensional calculation.

The complete THETA 1-B program, but not the CINDA preprocessor or CINDA library, has been recompiled at ORNL, and the image of the resultant hexadecimal machine-language program has been loaded on an IBM 2311 disk pack. This segment has been used successfully to replace the THETA 1-B load module supplied by ANC which had to be transferred to an IBM 2314 disk. The use of the HEX image facilitates modification of the program when operating with the ORNL computer system

A model of the FCTF heater section has been formulated for input to THETA 1-B. For this electrically heated rod, gap conductances have been specified to eliminate the difficulty in evaluation of the initial steady-state temperature distribution. The mass flux, pressure, and inlet enthalpy values obtained from the RELAP 3 computation are used as input for this THETA 1-B calculation. Modifications are in progress for this model, and the present results are considered preliminary.

References

1. ORNL Nuclear Safety Research and Development Program Bimonthly Report for September-October 1972, USAEC Report ORNL-TM-4030, Oak Ridge National Laboratory, pp. 11-13.
 2. W. H. Rettig, Aerojet Nuclear Corporation, personal communication.
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1.4 MULTIROD TRANSIENT AND BLOWDOWN HEAT
TRANSFER EXPERIMENTS (TASK 3)
(AEC Activity 04 60 80 02 1)

R. E. Helms	R. H. Chapman
C. G. Lawson	M. I. Lundin
R. L. Moore	H. W. Hoffman

This task examines experimentally the thermal-hydraulic performance of large bundles of electrically heated rods during LOCA-type depressurization transients. A separate-effects approach is followed in which bundle response to imposed boundary conditions, rather than to flow loop behavior, is observed. Initial emphasis is placed on the time to critical heat flux (CHF), on the CHF magnitude, and on the immediate post-CHF heat transfer (period prior to ECC injection) associated with blowdown transients in 12-ft heated length, 49-rod bundles operating at typical PWR power and pressure level and coolant velocities and enthalpies. The test bundle resembles but, in its limited radial extent, is not a model of a reactor core; nor does the system, by virtue of the decoupling of test bundle and flow loop, model a reactor system. The Thermal-Hydraulic Test Facility (THTF) is being modified to create a large, versatile test bed that satisfies the needs of this task.

Test Facility Modifications

The following sections report the effort to upgrade the capability of the existing THTF so as to accommodate blowdown tests with large multi-rod arrays of electrical heaters. Due to funding restrictions, the design effort during the remainder of this fiscal year will be limited to

the test section and the electrical power supply controls. In consequence, design on the pressurizer, the pressure-suppression system, the holdup assembly, and the loop piping must be delayed until the demands of the more critical test section and power supply designs are satisfied. The current status of these postponed items is given below.

1. Pressurizer. Construction drawings reviewed but not revised.
2. Pressure-suppression system. Layout drawings updated to include a larger recirculation pump heat exchanger and to accommodate the test section relocation.
3. Holdup assembly. Design remains conceptual.
4. Piping design and flexibility analysis. Sufficient to assure no obvious problems.

Test Section

A final geometry for the heater element wall has been selected. This design is based on a minimum tube inside diameter of 0.220 in. and reduces the number of mandrels required in the drawing operation to four (vs five in earlier designs). A significant cost saving is thus realized. The final wall thickness profile is sketched in Fig. 1.2; the resulting axial power profile comparing specified with actual values is shown in Fig. 1.3. For Inconel 625 tubing material and a test-section voltage of 200 V, the test-section total amperage will be 35,280 A (720 A per rod). Other parameters differ from those previously reported,¹ as listed in Table 1.2. Final internal instrumentation and insulating requirements may require some "opening-up" of the bottom ID of the heater; while this will effect some asymmetry in the axial power generation profile, the distortion will not be extreme.

Studies on the feasibility of butt-welding thin-wall Inconel 625 tubes (for the central circumferential join between the two 6-ft tapered sections forming the 12-ft cosine direct heater) were initiated in the ORNL Metals and Ceramics Division.² Results thus far must be considered preliminary, since actual heater-rod tubing was not used in the tests. Thus these results must be verified on commercial tubing and, subsequently, on the actual tubing heat as heat-to-heat composition variations can often affect weldability and even weld penetration.

Fig. 1.2. Axial Wall-Thickness Profile for Inconel 625 Cosine-Power Generation Direct Heater. Tube symmetrical around heater center line; $P_{max}/P_{av} = 1.65$.

Fig. 1.3. Comparison of Specified and Actual Axial Power Profiles for Inconel 625 Heater Rods.

Table 1.2. Design Characteristics of THTF Heater Rod

	Previous Design	Current Design
$P_{\max}/P_{\text{av}}$	1.650	1.650
$P_{\min}/P_{\text{av}}$	0.256	0.272
P_{av} (kW/ft)	12	12
Transverse flux profile	Uniform	Uniform
Voltage at $P_{\max}$ (V)	200	200
Actual heater rod resistance (Ω)	0.2764	0.277
Number of heater rods	49	49
Heater length (in.)	144	144
Heater OD (in.)	0.422	0.422
Minimum heater tube ID (in.)	0.200	0.220
Maximum wall thickness (in.)	0.111	0.101
Minimum wall thickness (in.)	0.0131	0.0130
Number of tapered sections	5	4

All welds to date were made with 6-in. lengths of tubing machined from a single piece of 0.5-in.-thick Inconel 625 plate. The tubes were 0.422 in. OD with a wall thickness of 0.015 in.; tight fitting alumina backup slugs were used beneath each weld. The welds were made using a model K-875 Astro-Arc orbiting-arc welding head. These gas tungsten-arc welds are made with the electrode being rotated around the stationary tube by a copper electrode carrier. A local inert atmosphere is maintained around the weld joint. Thus far, a constant voltage has been used to drive the orbiting-arc head; however, a single rotation speed produces a weld that gradually increases in width due to heat buildup. The welding current was sloped up from a low arc-initiation level, held for 1.25 rotations, and then reduced to the initial value. This procedure reduces the possibility of burn through at arc initiation and eliminates a crack-inducing crater at weld termination.

Welds have been made with the two different joint geometries illustrated in Fig. 1.4: (a) straight butt welds over a grooved or smooth alumina insert and (b) welds with an Inconel 625 T-ring backed by alumina. Tentatively, we conclude that a groove in the backup pellet is not necessary but that a metallic insert may be required if a slight amount of weld-bead concavity cannot be tolerated. All test welds were inspected by fluorescent penetrant and radiographic techniques. We have found occasional, very minute indication with penetrant inspection but nothing of significance. Radiography showed no porosity, but occasional small areas of incomplete penetration were encountered. A typical microstructure is shown in Fig. 1.5; note again that this is the microstructure of

Fig. 1.4. Components for Developmental Butt-Weld Studies with Thin-Wall Inconel 625 Tubing. (a) Alumina backup slug (either grooved or plain), (b) weld insert with matching alumina slug, and (c) tubes joined using the insert in (b).

Fig. 1.5. Cross Section Through Butt Weld Formed by Orbiting-Arc Technique. Wall thickness is 15 mils; grooved alumina insert; vertical line bisecting weld results from matching two photomicrographs.

a tube machined from rolled plate and is not typical of that of drawn tubing.

Two tapered type 347 stainless steel tubes have been borrowed from Superior Tube Company for use in developing inspection procedures for the direct heater tubing. The following techniques are being examined: (1) Vidigage for wall thickness, (2) ultrasonics for flaw and discontinuity detection, (3) infrared scans for heat generation variations, and (4) high-frequency eddy currents for resistivity.

Initial design and assembly drawings for components of the high-voltage assembly (upper electrode) have been received from our Karl Roesch (KR) consultants; a preliminary design of the shroud (flow) tube assembly is due soon. KR has also initiated calculations of the radial inward forces (pinch) acting on the heater rods due to the magnetic field induced by the current flowing through the heater rods.

The bundle top-grid plate serves the dual function of mechanical support for the heater rod bundle and upper electrode. The TD nickel, selected for this duty, is not commercially available in the required thickness; and KR is now investigating a dual-plate design in which one plate has the desired electrical characteristics and the other satisfies the mechanical requirements.

A number of decisions have been made regarding test section instrumentation. Heater-rod internal thermocouples will be 0.040-in.-OD, stainless-steel-sheathed, Chromel-Alumel units with welded junctions electrically isolated from the thermocouple sheath. A braided Refrasil sleeve will further electrically isolate the individual thermocouples in each rod; a maximum of six couples will be used in each heater. No area thermal sensors are to be installed in the heaters for the first bundle. All 64 subchannel fluid thermocouples will be located at the top end of the heated zone; these will be of the same character as the internal couples. Immersion thermocouples (Namac or equivalent) will monitor temperatures at each of the four inlet and two exit lines. We will attempt to incorporate thermocouples just inside of the ceramic shroud liner to measure the fluid temperature in the outermost flow region; three couples would be axially distributed along each of the four boundaries. Temperatures of all pressure-housing flanges and of the bottom O-ring seal assembly

will also be obtained. Fast-response pressure transducers will be located at the test-section inlet and outlet.

Direct-Current Power Supply

The test bundle heater rods will be fed from the variable voltage dc power supply shown schematically in Fig. 1.6; this diagram includes the modifications previously described.³ In addition, it will be necessary to reverse the polarity on one of the two generator units, since the connection of the equalizer bus requires all generators to have the same polarity.

A more detailed power schematic drawing of the dc system is almost complete, and preliminary control schematic drawings are being prepared showing the interlock and control circuits for parallel operation of the generators. The circuits will permit any number of the generators (i.e., from 1 to 4) to operate in parallel; however, synchronous motor No. 5 (right center of Fig. 1.6) must be running to drive the 25-kW exciter. The 10-kW exciters are used only for nonparallel operation. The bypass resistor switch will remain open during parallel operation so that the generator breaker will provide maximum protection to the generators and heaters. Paralleling of the units is accomplished by closing the disconnect switches with the generator field circuit open. The equalizer switch will be closed only during parallel operation and will open when the generator breaker opens. The test section voltage will be controlled automatically during the power decay period by a solid-state network connected to the exciter field. Manual control is provided for all other operations. Controls for commutator flashover, loss of synchronous motor field, and overspeed are being examined.

The installation drawings for the equalizer bus and switches are almost complete. The bus consists of two 500-MCM cables, and the switches are size 7 contactors with a 1-hr rating of 1200 A. Work on the drawings for the bypass resistor switch installation is being delayed because the switch that was selected is not available with suitable position switches and interlocks. A review of alternative arrangements is presently being made.

Fig. 1.6. Schedule of BDHT dc Power System.

Instrumentation

Work was continued on the high-pressure primary instrument application diagrams, and preliminary application diagrams for the demineralized cooling water and pressure suppression systems were completed. Examination in more detail of THTF operation has necessitated some design changes. Thus the throttling valves for the flow through the primary heat exchangers have been relocated downstream of these units to prevent flashing feed to the heat exchangers. In addition, instrumentation will now be provided to enable heat balances on the test section and on the primary heat exchangers (the electrical power input to the test section will be compared with the shell-side heat gain in the heat exchangers).

Instrument tabulation for all systems continues in parallel with the application diagrams. The criteria for the test-section instrumentation are now becoming firm, and a preliminary sketch of that instrument application diagram has been started.

Quotations were received from four suppliers of analog magnetic tape systems and are being evaluated. Placement of an order is delayed pending final approval of the THTF System Design Description and subsequent relaxation of purchase and component-construction restrictions. Development of final purchase specifications for the computer-based data acquisition system is awaiting AEC (Oak Ridge Operations) approval of the ADP funds.

Updating of drawings for the existing THTF (single-rod burnout) was completed, and a draft of the Measurements Requirements Document was completed and issued for comments.

Building Alterations and Equipment Installation

Expansion of the THTF laboratory area was completed with the installation of the heating and ventilating unit.

Mechanical operations on the Bingham two-stage blowdown pump (such as final grouting of the base, pump-motor alignment, and power supply installation) have been deferred to the beginning of FY 1974 in order to concentrate available FY 1973 funding on the critical facility design aspects.

References

1. ORNL Nuclear Safety Research and Development Program Bimonthly Report for September-October 1972, USAEC Report ORNL-TM-4030, Oak Ridge National Laboratory, pp. 14-18.
 2. A. J. Moorhead, ORNL Metals and Ceramics Division, internal communication.
 3. ORNL Nuclear Safety Research and Development Program Bimonthly Report for July-August 1972, USAEC Report ORNL-TM-3973, Oak Ridge National Laboratory, p. 79.
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1.5 HYBRID COMPUTER HEAT TRANSFER PROGRAM

(AEC Activity 04 40 03 01 1)

R. S. Stone F. H. Clark

The hybrid computer heat transfer program has two objectives: (1) to use the hybrid version of THETA and other hybrid programs as available in support of the BDHT program, particularly at ORNL; and (2) to create a core thermal model on the ORNL hybrid computer, exploiting the capabilities of this machine for easy interaction with the analyst and benefiting from its rapid and stable running characteristics.

Program Status

No report this period.

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2. HTGR SAFETY STUDIES

2.1 REACTIONS OF STEAM WITH GRAPHITE (AEC Activity 04 60 01 10 1)

C. M. Blood A. P. Malinauskas*
H. J. de Nordwall*

The reactions of steam with various graphites are being studied to develop methods for calculating the effects of steam on the core of an HTGR and the rates of hydrogen and carbon monoxide production. Investigations are centered upon the determination of rate constants and the understanding of the reasons why the observed rate of reaction varies with the extent of reaction. The results will be used in the analysis of accidents involving steam ingress.

Data obtained in the course of our experimental program of study of the chemical kinetics characteristic of the steam-graphite reaction are presented in Tables 2.1 to 2.4. The reaction data listed in Table 2.1 represent the results of exposure of an H-327 graphite specimen to a flowing helium-steam atmosphere containing 0.0450 atm partial-pressure steam at 1000°C, and Table 2.2 summarizes the corresponding changes in characteristic geometrical parameters of the graphite due to steam exposure. Similarly, the reaction data in Table 2.3 refer to an H-327 specimen exposed under similar conditions to a stream containing 0.0700 atm partial-pressure steam, whereas Table 2.4 is a listing of the corresponding geometrical parameters. In accordance with our previous practice, corrections

*Part time.

Table 2.1. Reaction of 11.6764-g H-327 Graphite Specimen with 0.0450 atm Steam at 1000°C

Steam Exposure Time (min)	Graphite Removal ^a (mg)	Apparent Reaction Rate (mg/min)	Burnoff ^a (%)
30	28.7	1.17	0.25
70	82.0	1.44	0.70
110	143.9	1.53	1.23
150	204.2	1.53	1.75
200	281.8	1.50	2.41
250	354.2	1.59	3.03
320	473.0	1.78	4.05
420	657.6	1.92	5.63
520	857.8	1.96	7.35
620	1050.0		8.99

^aAdjusted for variations in steam partial pressure and temperature.

Table 2.2. Alterations of Geometrical Features of an H-327 Graphite Sample upon Exposure to 0.0450 atm Steam at 1000°C

Steam Exposure Time (min)	Knudsen Diffusion Coefficient for Helium (cm ² /sec)	Viscous Flow Coefficient for Helium (cm ² atm ⁻¹ sec ⁻¹)	Porosity-Tortuosity Ratio	BET Surface Area (m ²)
0	0.31	5.52	5.16×10^{-3}	1.99
30	0.75	6.11	5.86×10^{-3}	6.06
70				9.51
110	0.94	7.26	7.11×10^{-3}	11.88
150				13.65
200	0.82	8.24	8.92×10^{-3}	16.41
250				17.44
320	1.27	9.16	11.93×10^{-3}	20.04
420				23.20
520	1.21	11.02	18.14×10^{-3}	26.63
620				28.68

Table 2.3. Reaction of 11.6365-g H-327 Graphite Specimen with 0.0700 atm Steam at 1000°C.

Steam Exposure Time (min)	Graphite Removal ^a (mg)	Apparent Reaction Rate (mg/min)	Burnoff ^a (%)
30	33.3	1.45	0.29
68	98.8	2.02	0.85
108	191.1	2.35	1.64
148	286.4	2.50	2.46
198	416.2	2.46	3.58
248	532.0	2.35	4.57
308	674.9	2.33	5.80
400	886.6	2.40	7.62
500	1136.6		9.77

^aAdjusted for variations in steam partial pressure and temperature.

Table 2.4. Alterations of Geometrical Features of an H-327 Graphite Sample upon Exposure to 0.0700 atm Steam at 1000°C

Steam Exposure Time (min)	Knudsen Diffusion Coefficient for Helium (cm ² /sec)	Viscous Flow Coefficient for Helium (cm ² atm ⁻¹ sec ⁻¹)	Porosity-Tortuosity Ratio	BET Surface Area (m ²)
0	0.57	3.53	4.42×10^{-3}	2.68
30	0.62	4.39	5.58×10^{-3}	6.50
68				10.50
108	0.88	5.17	6.98×10^{-3}	13.05
148				17.25
198	0.96	6.29	10.48×10^{-3}	17.96
248				20.78
308	1.22	6.80	13.90×10^{-3}	23.04
400				23.35
500	1.58	7.83	18.95×10^{-3}	28.41

for minor differences in steam concentration and temperature were applied by assuming the rate of reaction to be of first order in steam concentration and characterized by an Arrhenius activation energy of 60 kcal/mole; in no case did the correction exceed 8%. Additional data obtained in this series of experiments have been reported previously.¹⁻³

As noted earlier,³ the development of surface area with graphite removal is sensibly reproducible from specimen to specimen; this is illustrated by the data of Table 2.5, in which (interpolated) values of the surface area of the three samples exposed to steam at 1000°C are listed as a function of burnoff. The results representative of exposure at the lower steam partial pressures do indeed deviate from the 0.0950-atm data (at 800 and 900°C also), but the deviations (about 20%) are not believed to be significant.

Of greater significance is the apparently complicated dependence of the rate of reaction on steam partial pressure, as can be inferred from the data in Table 2.6. These data represent (interpolated) values of the apparent rates of reaction for the specimens exposed to varying steam partial pressure at given specimen surface areas. If the apparent rate of reaction were of first order with respect to steam concentration, the rate values would stand in the same proportion as the steam partial pressures. It is obvious from the data listed in Table 2.6 that they do not.

Table 2.5. Comparative Results of Surface Area Alterations of H-327 Graphite Samples Due to Steam Exposure at 1000°C

Burnoff (%)	Surface Area (m ²) at Steam Partial Pressure of —		
	0.0450 atm	0.0700 atm	0.0950 atm
0.0	1.99	2.68	2.20
1.0	10.85	10.98	9.96
2.0	14.70	14.89	14.98
3.0	17.39	17.59	19.65
4.0	19.91	19.16	23.90
5.0	21.94	21.57	27.18

Table 2.6. Comparative Results of Apparent Rates of Reaction of H-327 Graphite Samples with Steam at 1000°C

Surface Area (m ²)	Apparent Rate of Reaction (mg/min) at Steam Partial Pressure of --		
	0.0450 atm	0.0700 atm	0.0950 atm
6	1.15	1.26	1.56
12	1.53	2.21	1.99
18	1.63	2.46	2.22
22	1.87	2.34	2.34

Part of the discrepancy undoubtedly is due to inhibition by reaction products, a mechanism whose effects were clearly demonstrated in the last report.⁴ This aspect is currently under investigation. Another involves the assumption of negligible in-pore diffusion effects, but it is doubtful whether this facet of the problem could be gainfully pursued before the question of inhibition is resolved.

References

1. C. M. Blood and A. P. Malinauskas, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1971, USAEC Report ORNL-TM-3411, Oak Ridge National Laboratory, pp. 45-47.
2. C. M. Blood and A. P. Malinauskas, ORNL Nuclear Safety Research and Development Program Bimonthly Report for November-December 1971, USAEC Report ORNL-TM-3670, Oak Ridge National Laboratory, pp. 73-76.
3. C. M. Blood and A. P. Malinauskas, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1972, USAEC Report ORNL-TM-3831, Oak Ridge National Laboratory, pp. 65-69.
4. C. M. Blood and A. P. Malinauskas, ORNL Nuclear Safety Research and Development Program Bimonthly Report for September-October 1972, Non-LMFBR Programs, USAEC Report ORNL-TM-4030, Oak Ridge National Laboratory, pp. 25-27.

2.2 FUEL INTEGRITY AND FISSION PRODUCT RELEASE*

(AEC Activity 40 60 10 01 1)

M. T. Morgan	T. B. Lindemer
R. L. Towns	L. L. Fairchild

The goal of this program is to determine the cause and extent of fission product release from irradiated coated-particle fuel bodies at normal and abnormal reactor fuel temperatures. Knowledge of both the factors determining coating integrity and fission product transport mechanisms is necessary for the analysis of fission product release from fuel bodies and ultimately from the reactor core since changes in the fraction of fuel with failed coatings can affect fission product transport substantially. Current emphasis is on describing internal carbon transport mechanisms and failure brought about by excessive temperature in an isothermal environment. Transport of fission products at normal operating temperatures must be considered in accident consequence analysis, since the fission product release from a reactor at the time of an abnormal temperature fluctuation would depend upon the initial fission product distribution.

The information generated will be used in calculating fission product inventories of the core graphite and the coolant circuit and in estimating the changes in those inventories that may occur as a result of accidental temperature changes. To the extent that fission product release from the fuel determines the relatively higher risk inventory of the coolant circuit, these results will also be useful in analyzing the consequences of depressurization accidents and the radiation doses that might be received by maintenance crews.

Comparison of Fission Product Release through
Various Types of Coated Fuel Particles

Additional samples of Biso-coated particles have been annealed at 1600°C to determine cesium release. Diffusion coefficients were calculated from the average time required for the cesium to diffuse through

*Work jointly supported by HTGR Base Program.

the coatings during the postirradiation anneals. These data are plotted in Fig. 2.1 along with other data previously reported.¹ Data numbers in the figure correspond to sample numbers listed in Table 2.7. The new data include the results from anneals of samples 3 through 10 at 1600°C and samples 5 and 8 at 1400°C.

Fig. 2.1. Cesium Diffusion Coefficients in Pyrocarbon Determined from Anneals of Irradiated Coated Particles.

Table 2.7. Preirradiation Data for Biso-Coated Fissile and Fertile Particles Used in Postirradiation Annealing Experiments

Sample	Kernel	Coating Type	Coating Source Gas	Density (g/cc)	BAF or OPTAF	Crystallite Size (Å)
1 ^a	UO ₂	LTI	Propylene	1.83		30
2 ^a	Same batch as 1 above					
3 ^b	ThO ₂	LTI	Propylene	1.99	1.0 ^c	35
4 ^b	ThO ₂	LTIA ^d	Propylene	2.04	1.0 ^e	120
5 ^f	UC ₂	LTI	Propane ^g	1.83	1.06 ^h	
6 ^f	UO ₂	LTI	Propane	1.85	1.00 ^h	
7 ^f	UO ₂	HTI	Methane	1.75		
8 ^f	UC ₂	HTI	Methane	1.82	1.00 ^h	
9 ^b	ThO ₂	HTI	Methane	1.82	1.1 ^e	112
10 ^b	ThO ₂	HTI	Methane	2.00	1.3 ^e	120

^aSamples 1 and 2 were irradiated in the ORR to burnups of 10 and 20% FIMA respectively.

^bAll coated ThO₂ particles were irradiated in the HFIR to a fast fluence of 4×10^{21} neutrons/cm² in the HT-1 capsule. Burnup ~3% FIMA.

^cBAF measured from disk sample.

^dPreirradiation annealed at 1900°C.

^eBAF estimated.

^fThese samples were irradiated in the ORR to a burnup of approximately 10% FIMA in the Cl-28 capsule.

^gThere is no reason to assume that propane yields the same product as propylene.

^hOPTAF.

For annealing, each coated particle is placed in a graphite thimble inserted in a tungsten tube. The tungsten tubes are then inserted into peripheral holes in a graphite crucible which accommodates eight tubes. At 1600°C, two coated particles from each of four samples were annealed at the same time in the eight-tube crucible. Samples 3, 4, 9, and 10 were annealed in one set, while samples 5, 6, 7, and 8 were annealed in another. At 1400°C four coated particles from each of samples 5 and 8 were annealed at the same time in the eight-tube crucible.

In contrast to the wide difference in cesium diffusion coefficients in LTI and HTI pyrocarbon obtained from previous work on samples 1, 2, 3, and 9, the later data show closer agreement for most of the samples. Variations in preirradiation characteristics are not consistent with the diffusion coefficients obtained. Samples 1, 2, 5, 6, 8, and 9 had comparable densities but yielded widely different diffusion coefficients. Sample 7, with HTI pyrocarbon, had a low density and a higher diffusion coefficient, but sample 8, also with HTI pyrocarbon, had a medium density and a similar diffusion coefficient. Samples 9 and 10 were estimated to have slightly higher anisotropy factors and gave the lowest diffusion coefficients. Crystallite size was small for most of the LTI pyrocarbons except for sample 4 and was large in the HTI pyrocarbons. Sample 4 had a high density and a large crystallite size and yet gave a diffusion coefficient similar to other LTI pyrocarbons.

In general, the pattern shows that some HTI pyrocarbons are much better than LTI pyrocarbons in retaining cesium, while others are about the same. The characteristics which influence the diffusion are not clearly defined, and one cannot assume that propane and propylene yield the same products. More detailed postirradiation coating characterization is needed to enable irradiation effects on coating structure to be taken into account. Appropriate measurements are being planned.

In addition to density, crystallite size, and anisotropy, another characteristic of pyrocarbon which may influence diffusion is the relative amount of amorphous and crystalline carbon in the structure. Methods tried in Germany at KFA are being studied for this determination. Koizlik and others² at the Julich Nuclear Research Center Institute for Reactor Materials have developed a semiquantitative method for determining the amount of amorphous carbon in pyrocarbon by a calculation involving the difference in measurements of orientation anisotropy as determined by x-ray diffraction (BAF) and by polarized light reflection (OPTAF). The BAF method depends on the fraction of preferred orientation by crystallites, whereas the OPTAF method, which likewise depends on the fraction of preferred orientation of crystallites, also includes the effect of isotropic reflection from amorphous material.

Other methods tried for the evaluation of the amorphous to crystalline fraction have been (1) the use of ultrasonic energy to drive some of the amorphous carbon out of the pyrocarbon into methyl alcohol and an optical absorption method to measure the carbon dispersed and (2) a cold oxidation of the pyrocarbon plus an electron microscopic examination to determine the structure. Reports on these methods have been received from KFA, Julich. One of these, by Sutterlin et al.,³ is being translated from the German.

Carbon Transport in HTGR Particles in a Temperature Gradient (Amoeba Effect)

During this reporting period, effort was concentrated on the design and construction of equipment. The particle holders for the temperature gradient furnace were designed and fabricated after considerable discussion with O. M. Stansfield, who is doing similar work at GGA. The furnace for this work should be delivered near the end of December. Construction of a bank of miniature high-temperature furnaces for isothermal annealing studies is approaching completion. An alumina muffle for the high-temperature carbon furnace has been made to permit synthesis of kernels such as UO_{2+x} and $U(C,S)_2$ for use in the mass transport studies.

Chemical Characteristics of Fuel and Fission Products

The brief study of the $UO_{2-x}N_{0.75x}$ system is about 40% completed. The justification for this study was given in the previous report.⁴ The results obtained thus far are summarized in the following abstract, which was submitted to the Nuclear Division of American Ceramic Society for presentation at their spring meeting.

"Thermodynamic behavior in the system $UO_{2-x}N_{0.75x}$ ($0.02 \leq x \leq 0.35$) has been investigated from 1250 to 1850°C by reacting UO_2 with carbon at specific gas pressures in the ranges $1 \leq P_{CO} \leq 520$ torr and $18 \leq P_{N_2} \leq 600$ torr. The P_{CO} was held high enough to prevent the formation of $UC_{1-z}N_z$. The thermodynamic analysis was based on the reaction $[UO_2] + 2C + 0.75N_2 \rightleftharpoons [UN_{1.5}] + 2CO$, where the brackets indicate solid solution

in the oxynitride. Free-energy functions from the literature and the experimental parameters were used to determine the activity of $[\text{UN}_{1.5}]$."

Pure UO_2 and $\text{UN}_{1.5}$ were taken as standard states. The activity is dependent upon P_{CO} and P_{N_2} and is approximately 10^{-5} times the mole fraction of $[\text{UN}_{1.5}]$ at all temperatures. Predicted characteristics of the oxynitride as an HTGR fuel are compatibility with pyrocarbon in a temperature gradient, buffering of P_{CO} from fission-released oxygen, and ease of synthesis. However, the oxynitride may be incompatible with silicon carbide.

Design of equipment to be used for determination of gas pressures and compositions in coated particles has been started.

An attempt is being made to identify those chemical systems which could possibly be used to retain the fission products cesium, strontium, and barium within the fuel kernel. This work is currently limited to assessing existing thermodynamic data and equilibrium phase diagrams in various oxide systems. This permits the identification of the compounds in equilibrium phase fields that have cesium, strontium, or barium overpressures low enough to prevent diffusion of these elements from the particle. The Al_2O_3 - SiO_2 -(group IA oxide) ternary systems are being evaluated at present, but the work is not far enough advanced to report specific details at this time.

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2.3 FISSION PRODUCTS IN THE COOLANT CIRCUITS

(AEC Activity 04 60 10 01 1)

F. F. Dyer	M. F. Osborne
J. O. Kolb	S. H. Freid
H. J. de Nordwall*	W. J. Martin

This section is concerned with describing and explaining the distribution and adhesion of fission products in the coolant circuits of HTGRs. It includes the activities associated with the HTGR fission product surveillance program. The results of these studies will be used in the analysis of accidents involving depressurization and coolant composition change, and in testing codes used to predict the inventory and distribution of fission products in the coolant circuit.

Iodine Adsorption and Desorption

Preliminary experiments on iodine desorption from Fe_3O_4 in a flowing helium-steam mixture in a transpiration apparatus produced inconsistent results. Consequently, the experimental technique was reviewed, and a new apparatus was constructed. The horizontal transpiration tube, with the Fe_3O_4 adsorber sample contained in a shallow boat, was replaced by a vertical tube, with the sample dispersed in a thin layer on a fritted quartz plate. This new arrangement should avoid problems associated with re-adsorption and improve the contact between gas and solid. In addition, the $\text{H}_2/\text{H}_2\text{O}$ ratio will be controlled at 10^{-1} to 10^{-2} to assure that the Fe_3O_4 will be neither oxidized nor reduced during the experiment.

Calculations of equilibria in the systems iodine plus iron, chromium, or nickel were completed, at least to the extent permitted by available

*Part time.

data. Equilibrium constants determined by Cerny and Bartovska¹ were used to calculate the partial pressures in the Cr-I-CrI₂(s) system. Figure 2.2 contains the results. The solid curve is the vapor pressure curve for CrI₂, and the dashed lines represent the partial pressure of CrI₂(g) in equilibrium with iodine [I(g)] and chromium metal at pressures below the dissociated pressure of solid chromous iodide at the indicated temperature. Similar data were obtained for CrI₃ and Cr₂I₄ at somewhat lower partial pressures.

Fig. 2.2. Calculated Partial Pressures of CrI₂(g) in Equilibrium with CrI₂(s) and Cr at Various Iodine Pressures and Temperatures.

Since the thermochemical data are relatively sparse for the Cr-I, and especially the Ni-I systems, independent estimates were compared wherever possible. The expressions used to calculate the partial pressures of the various compounds at HTGR coolant temperatures were extrapolations from measurements at higher temperatures in all three systems. When these results were compared with data calculated from the free-energy functions at 500 and 1000°K reported by Brewer et al,² good agreement was obtained for CrI₂; however, the NiI₂ results differed by one to two orders of magnitude over the temperature range 200 to 800°C. Detailed results of these calculations will be reported separately.

Fission Product Surveillance at the Peach Bottom HTGR

Preliminary Examination of Active Particle Filter

The first ORNL active particle filter unit, in the hot valve (HV-201) sample line downstream from the TW-205 diffusion tube sampler, was removed after the Peach Bottom shutdown on November 25, 1972. This filter had operated for approximately three days, during which time 0.26 lb/hr of helium were drawn from the primary-coolant system between the core and the No. 1 steam generator. A replacement active particle filter unit was installed for operation until the next scheduled maintenance shutdown in 1973.

The filter material in the unit removed from service is a 1 7/8-in.-diam sintered silver membrane with holes less than 0.2 μ m in diameter. The filter is shown in Fig. 2.3. Gamma-emitting radionuclides on the filter were measured on December 13, 1972, by accumulating a high-resolution gamma spectrum for 17 hr. The counting efficiency of the detector had been calibrated with standard radionuclide sources purchased from the International Atomic Energy Agency (IAEA). Since the IAEA standards were essentially point sources, it was necessary to correct the filter results for a small bias caused by the filter shape. This correction was found to be 1.6%. After the filter had been analyzed for gamma-emitting radionuclides, it was photographed and cut into four equal pie-shaped pieces. One of the pieces was submitted for analysis for ⁸⁹Sr and ⁹⁰Sr using destructive radiochemical procedures. The weight of total dust deposited

Fig. 2.3. Dust-Covered Silver Membrane Filter Removed from Hot Valve Sampling Line in November 1972.

on the filter was determined by removing and weighing the dust from one of the four pieces. The dust removed was analyzed by high-resolution gamma spectroscopy to obtain the specific activities of gamma emitters (microcuries per gram of dust). These results were then used as "weight tracers" to compute the weight of dust on the total filter. The ^{235}U

was measured by delayed-neutron activation analysis on the weighed portion of dust. Values for all measured quantities are given in Table 2.8.

One interesting feature of these results is that the ratio of ^{137}Cs to ^{134}Cs is 4.05. When this ratio is corrected for decay of both radioisotopes to April 1971 (the time of the 1971 gamma scan of the large ducts), the corrected ratio is 2.43. The corrected ratio is thus significantly different from the ratio (2.97) found on the large ducts in April 1971. It is tempting to speculate that the current ratio partially reflects release of these nuclides from the core rather than the erosion

Table 2.8. Materials Found on Hot Valve Sampling Line Filter^a
Removed in November 1972

Specie Measured	Quantity Found ^b	Specie Measured	Quantity Found ^b
Dust	5 mg	^{106}Ru	1.1×10^{-2e}
^{235}U	<i>c</i>	^{110m}Ag	1.68×10^{-2e}
^{51}Cr	5.2×10^{-2}	^{131}I	1.8×10^{-3f}
^{54}Mn	2.91×10^{-2}	^{134}Cs	8.89×10^{-2}
^{59}Fe	0.125	^{137}Cs	0.359
^{60}Co	5.4×10^{-2}	^{140}Ba	7.5×10^{-3}
^{95}Zr	6.6×10^{-3d}	^{141}Ce	5.7×10^{-3}
^{89}Sr	<i>c</i>	^{144}Ce	2.74×10^{-3}
^{90}Sr	<i>c</i>	^{154}Eu	3.5×10^{-3}
^{103}Ru	3.5×10^{-3}	^{203}Hg	1.1×10^{-2g}

^aValues given are for the total filter. Radionuclide results were corrected to reactor down date of Nov. 25, 1972. The quantity of helium that flowed through filter was 449 lb. However, since the filter was downstream of gaseous diffusion tubes that may have acted as dust and radionuclide sinks, we cannot presently compute concentrations of the various species per unit weight of reactor helium.

^bValues in microcuries unless stated.

^cTo be determined later.

^dBecause of mutual photopeak interference this value is tentative.

^eRadionuclide appears to be present, but values given should be treated as upper limits rather than firm results.

^fUpper limit from high-resolution gamma spectroscopy. No photopeak found.

^gThe 279-keV line in gamma spectrum has been tentatively identified as ^{203}Hg .

of "old" dust from the primary circuit walls. (The uncertainty in the $^{137}\text{Cs}/^{134}\text{Cs}$ ratio is $<\pm 0.2$.) Another interesting feature of these results is that ^{140}Ba was found. An evaluation of this finding in terms of the release of ^{140}Xe from the core will be made at a later date.

Activities Deposited on the Walls of Peach Bottom
Primary Coolant Loop No. 1

Deposited gamma-emitting isotopes were measured in May 1971 and May 1972 during fuel-changing shutdowns. The measurement positions are shown in Figs. 2.4 and 2.5. Revised 1971 results (previously reported)³ and the 1972 results are given in Tables 2.9 and 2.10. Upper limits have been added to the 1971 results for isotopes that were sought but not found. The results indicate that concentrations of many radionuclides decreased significantly from 1971 to 1972. Each set of results was corrected for decay to the time the reactor went down (January 6, 1972, for the 1972 results and April 26, 1971, for the 1971 results). For several of the radionuclides the decrease over the interval of reporting dates (257 days) was significantly larger than can be explained by radioactive decay. This is particularly apparent for the long-lived radionuclides ^{134}Cs and ^{137}Cs , which decreased by a factor of about 3. Other long-lived species also seem to indicate a comparable decrease after radioactive decay is taken into account.

The observed decrease is believed to be real and not just an artifact in the calibration of our detector. Evidently these activities are being removed from the surfaces and not being redeposited.

Another interesting feature is the ratio of ^{137}Cs to ^{134}Cs surface concentration. This ratio is nearly constant for all measured positions within each set of measurements, having a mean value of 3.49 in 1972 and 2.97 in 1971. When the 1972 ratio is corrected for radioactive decay to the 1971 reporting date, the corrected ratio is 2.81 — a value close to the 1971 ratio. (The variation in these ratios was $<\pm 0.2$.) Although this ratio would not be a very sensitive indicator of changes that occur in addition to decay, it indicates that there are no gross replenishments of the cesium isotopes on the large ducts unless the newly deposited isotopes had a $^{137}\text{Cs}/^{139}\text{Cs}$ ratio close to 3.5, the value for "old" cesium.

Table 2.9. Summary of Activities^a Found on Peach Bottom Primary-Circuit Large Ducts, 1971

Position	¹³¹ I	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁴⁰ Ba	^{110m} Ag	⁹⁵ Zr	⁵⁹ Fe	⁶⁰ Co	⁵⁴ Mn	⁶⁵ Zn	¹⁰³ Ru	¹⁰⁶ Ru	¹²⁴ Sb	¹⁵⁴ Eu	Ratio ¹³⁷ / ¹³⁴ Cs
Steam generator exit	0.04 ^b	0.71	2.1	0.39	6.8N3	2.7N2	4.0N3 ^b	1.6N2	1.5N2	9.1N3	2.9N2	0.007 ^b	0.03 ^b	1.3N3	1.1N3 ^b	2.96
Compressor entry	0.09 ^b	0.78	2.4	0.05	6.1N3	2.6N2	7.0N3 ^b	4.6N3	2.0N3	7.6N3	2.5N2	0.015 ^b	0.06 ^b	7.0N4	1.7N3 ^b	3.07
Compressor exit	0.06 ^b	0.73	2.2	0.02	6.3N3	2.2N2	5.0N3 ^b	2.3N3	9.4N4	6.3N3	2.6N2	0.01 ^b	0.05 ^b	6.8N4	1.1N3 ^b	3.01
Compressor exit line near steam genera- tor	0.04 ^b	0.18	0.54	0.06	2.5N3	5.6N3	3.3N3 ^b	1.1N2	7.3N3	3.6N3	5.7N3	0.006 ^b	0.05 ^b	6.5N4	1.0N3 ^b	3.00
Compressor exit vertical duct	0.007 ^b	0.16	0.45	0.07	3.9N3	7.3N2	7.2N3	1.9N2	2.1N2	4.1N3	4.7N3	1.3N3	1.6N2	7.1N4	1.09N3	2.81
															Mean	2.97

^aN2, N3, etc., indicate value given $\times 10^{-2}$, $\times 10^{-3}$, etc.

^bRepresents upper limit; no peak found for this isotope.

Table 2.10. Summary of Activities^a Found on Peach Bottom Primary-Circuit Large Ducts, 1972

Position	¹³¹ I ^b	¹³⁴ Cs	¹³⁷ Cs	¹⁴⁴ Ce	¹⁴⁰ Ba ^b	^{110m} Ag	⁹⁵ Zr	⁵⁹ Fe	⁶⁰ Co	⁵⁴ Mn	⁶⁵ Zn	¹⁰³ Ru	¹⁰⁶ Ru	¹²⁴ Sb	¹⁵⁴ Eu	Ratio ¹³⁷ / ¹³⁴ Cs
Steam generator exit	0.41	1.4	7.8N2			1.3N2		1.1N2	1.0N2	6.3N3	1.7N2					3.41
Compressor entry downstream of HC-203 valve	0.22	0.78	1.6N2			5.2N3	3.6N3 ^c	2.5N3	1.8N3	3.0N3	7.7N3	1.6N2 ^c	2.1N2 ^c	3.2N4	3.5N4 ^c	3.54
Compressor entry by cavity entry	0.23	0.79	1.1N2			7.1N3	2.0N3 ^c	2.4N3	1.3N3	3.3N3	9.4N3	9.2N3 ^c	1.2N2 ^c	3.3N4	1.9N4 ^c	3.43
Compressor exit by cavity entry	0.18	0.62	1.0N2			5.0N3	6.9N3 ^c	2.4N3	8.0N4	1.7N3	4.6N3	2.8N2 ^c	3.4N2 ^c	3.6N4 ^c	6.0N4 ^c	3.44
Compressor exit line near steam genera- tor	0.23	0.81	5.6N2			9.6N3	1.2N2 ^c	2.4N2	2.7N2	6.7N3	1.0N2	4.0N2	4.0N2	1.6N3	1.7N3	3.52
Compressor return before joining con- centric duct	0.13	0.47	3.1N2			3.2N3	6.1N3 ^c	9.4N3	5.9N3	3.1N3	6.1N3	2.1N2 ^c	1.5N2 ^c	7.0N4 ^c	7.8N4 ^c	3.61
															Mean	3.49

^aN2, N3, etc., indicate value given $\times 10^{-2}$, $\times 10^{-3}$, etc.

^bDecay time too long to permit detection or warrant upper limit calculations.

^cRepresents upper limit; no peak found for this isotope.

Fig. 2.4. Positions of Gamma-Activity Determinations in Peach Bottom HTGR No. 1 Loop in May 1971.

Fig. 2.5. Positions of Gamma-Activity Determinations in Peach Bottom HTGR No. 1 Loop in May 1972.

The Ge(Li) gamma detector and collimator were calibrated by counting a full-size cardboard mockup of the ducts and reproducing the relative positions of duct and detector. The mockup is coated uniformly with a radioisotope that emits several gamma rays spanning the energy region of interest. Lanthanum-140 was used for two calibrations. Specimens of known size were cut from the mockup, and the surface concentration of the isotope was measured by standard methods on a detector calibrated with IAEA standard radionuclide samples. The counting efficiency of the collimated detector for the mockup was expressed as

$$EF_0 = \frac{\text{CPS (counts/sec)}}{\text{GPS (photons cm}^{-2} \text{ sec}^{-1})},$$

where the photon emission rate per square centimeter had been determined from the specimens removed from the mockup. The quantity EF_0 was obtained for each gamma ray, and a plot of EF_0 vs energy was prepared.

The count rate for a particular gamma ray from a large duct can then be expressed as

$$\text{CPS} = EF_0 \text{ GPS } \sum e^{-\mu_i X_i},$$

where the summation term accounts for attenuation in the large duct wall and the insulation covering the duct. The photon emission rate (GPS) can thus be calculated from the gamma spectrum after the attenuation factors have been obtained. Disintegration rates are then computed from the photon emission rates. Attenuation factors have been calculated rather than measured. Plans are being made to prepare a full-size steel and insulation mockup of the single-walled ducts and experimentally evaluate the counting efficiencies with the "built-in" attenuation.

This model of the large ducts assumes that the surface concentrations are uniform. The assumption is probably valid enough for the single-walled ducts. However, models that distinguish unambiguously between surface concentration on the inner and outer portion of the concentric hot duct cannot be constructed because we do not know a priori the ratios of the surface concentrations on the two ducts. Nevertheless, one can derive surface concentrations based on models with assumed ratios or activities on one set of surfaces. This has been done, and it appears that

concentrations on the inner duct are unlikely to be significantly larger than those on the outer duct.

Peach Bottom HTGR Fuel Element Examination

Plans and equipment have been prepared for the examination of Peach Bottom element E06-07, which had been irradiated for about 380 days. As part of this preparation a version of the FIPER code developed and used by Gulf General Atomic is being critically examined. The code will be used for (1) interpreting observed fission product distributions, (2) relating them to circuit activities and hence testing the overall fission product transport model, and (3) deriving fission product transport parameters for use in future safety assessments.

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2.4 HTGR SYSTEM SAFETY ANALYSIS

(AEC Activity 04 60 10 01 1)

P. Rubel

The objectives of this task are (1) to provide logical guidance in the planning of the HTGR Safety Program and (2) to assess probable courses and hence endpoints for HTGR accidents. In combination with the descriptive material available from other tasks, one will then be able to formulate the risks associated with an HTGR.

Work on the HTGR Safety Program Plan continues to consume a substantial fraction of available effort. Plans are presently being made to describe potential HTGR accidents as thoroughly as practicable via unified sets of logical mechanistic and probabilistic analyses. In this manner, it is hoped to establish a rational safety perspective which will aid in setting program priorities, in determining needs for additional investigations, and in delineating realistic ranges of the conditions that influence accident mechanisms.

Emphasis during the report period was on the development of two aspects of logical modeling: one to describe the protection systems reliability and emergency response prospects and a second concerning realistic accident event sequences. The systems models are essentially fault trees, but with the addition of structuring to accommodate all special considerations associated with specific contingency conditions. A satisfactory method for handling these additional considerations has been devised.

Accident sequence descriptions tend to branch into unmanageable numbers of event combinations, many of which are similar. Consequently, the modeling objective is to reduce and group these combinations by significant ranges and so achieve a tractable number of discrete cases for further individual study. It appears that both fault and decision trees will be needed, the latter to describe situations with multiple outcomes. A substantial amount of "cut and try" effort has been expended in the attempt to identify an optimal set of logical accident diagrams for future execution. These diagrams will ultimately provide a basis for defining conditions and events to be investigated via plant transient (mechanistic) analyses. Probabilistic evaluations of the logic sequences will also be attempted in due course.

A report¹ reviewing the study conducted for ORNL by UKAEA Systems Reliability Service (SRS) was issued. The study consisted of a reliability analysis of the emergency engine generator systems for the Fort St. Vrain plant, and was described in an earlier progress report.²

References

1. P. Rubel, Reliability Evaluation of the Fort St. Vrain Emergency Engine-Generator System, USAEC Report ORNL-TM-3935, Oak Ridge National Laboratory, December 1972.
2. P. Rubel, HTGR System Safety Analysis, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1972, USAEC Report ORNL-TM-3831, Oak Ridge National Laboratory, p. 85.

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3. HEAVY-SECTION STEEL TECHNOLOGY PROGRAM

3.1 PROGRAM ADMINISTRATION AND PROCUREMENT (TASK A)

(AEC Activity 04 60 80 03 1)

F. J. Witt

The development of quantitative methods and procedures for assessing margins of safety against fracture or break of the primary containment system of the reactor core of civilian water reactor power stations continues as the major technological objective of the Heavy-Section Steel Technology (HSST) program. Current emphases are on the testing of a series of 6-in.-thick flawed pressure vessels, investigations of the effects of fast-neutron irradiation on the fracture toughness of applicable product forms, and studies of environmental effects in combination with low-cycle fatigue.

The activities of the HSST program are carried out under a series of 12 tasks, 4 of which are completed. The activities under the remaining eight tasks are discussed below.

Three technical reports were published,¹⁻³ two are in publication, and several are in preparation. One progress report⁴ was issued, and another is being edited. One subcontract was extended.

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3. S. P. Ying and S. C. Grigory, Tests of 6-Inch-Thick Tensile Specimens, Fifth Technical Summary Report, Acoustic Emission Monitoring of One-Inch-Thick and Six-Inch-Thick Tensile Specimens, HSSTP-TR-24, Southwest Research Institute, San Antonio, Texas, November 1972.
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3.2 MATERIAL INSPECTION AND CONTROL (TASK B)

(AEC Activity 04 60 80 03 1)

The major activity of this task remains the fabrication and inspection monitoring of the intermediate test vessels which are yet to be completed.

3.3 VARIABILITY IN PLATE, HEAT-AFFECTED ZONES, WELDS, AND FORGINGS (TASK D)

(AEC Activity 04 60 80 03 1)

Series of precracked Charpy specimens continue to be tested as part of the material investigations of the intermediate pressure vessel tests. Major emphasis is on slow bend testing.

3.4 FRACTURE MECHANICS INVESTIGATIONS (TASK E)

(AEC Activity 04 60 80 03 1)

Specimens for investigation of the significance of pop-in arrest are being tested at Materials Research Laboratory. Tests for evaluating plane

strain (non-brittle) fracture theories are under way at Westinghouse Electric Corporation as well as the determination of lower bound toughness for submerged-arc welds. The fracture theories evaluation specimen is sketched in Fig. 3.1. The specimen will be tested in thicknesses to 4 in. or greater.

An elastic-plastic three-dimensional stress-strain analysis of the nozzle corner of the nozzled intermediate test vessels is scheduled by Marc Analysis Research Corporation.

3.5 FATIGUE AND CRACK PROPAGATION (TASK G)

(AEC Activity 04 60 80 03 1)

Initial results from the low-cycle (1 cpm) fatigue crack propagation in A533, grade B, class 1 steel under the environmental conditions of PWR and BWR systems, excluding irradiation, are compared with similar results for higher cycling rates in Fig. 3.2. Low cycle results in air are currently being obtained. This test should define whether the significant effects noted in Fig. 3.2 are due to cycling rate alone or to a combination of cycling rate and environment. The latter is expected to be confirmed. This work is in progress at Westinghouse Electric Corporation.

3.6 IRRADIATION EFFECTS (TASK H)

(AEC Activity 04 60 80 03 1)

The effects of low cycling in combination with irradiation are scheduled to be investigated at Hanford Engineering Development Laboratory (HEDL) as a followup to the work described in Ref. 1. Also at HEDL fatigue pre-cracking has been completed on several dozen irradiated specimens in preparation for extensive lower-bound testing.

Three capsules, each containing two 4T (4-in.-thick) compact-tension specimens, are being irradiated in the Battelle Research Reactor under the

Fig. 3.1. Fracture Theories Evaluation Specimen.

Fig. 3.2. Crack Growth Rates for HSST 02 Material at 1 cpm Frequency in PWR and BWR Environments.

direction of Westinghouse Electric Corporation. The target fluence is 5×10^{19} neutrons/cm², $E > 1$ MeV, irradiated nominally at 550°F. A sketch of the capsule geometry is shown in Fig. 3.3. In addition, each of the holes in the specimens contains a combination of from 22 to 26 tensile specimens, 0.394-in.-thick compact-tension specimens, Charpy specimens, and precracked Charpy specimens. These specimens emphasize surface to center variations, directionality, low Charpy impact shelf properties, and forging steels and weldments. Thermocouples to monitor the temperature gradients in the capsules are placed as shown in Fig. 3.4. The six 4T specimens consist of two longitudinally oriented specimens, two transverse specimens, and two submerged-arc specimens.

Reference

1. L. A. James and J. A. Williams, The Effect of Temperature and Neutron Irradiation upon the Fatigue-Crack Propagation Behavior of ASTM A 533 Grade B, Class 1 Steel, HSST Technical Report No. 21, HEDL-TME 72-132, Hanford Engineering Development Laboratory, September 1972.

3.7 SIMULATED SERVICE TEST (TASK K)

(AEC Activity 04 60 80 03 1)

The third (vessel V-3) and fourth (vessel V-4) 6-in.-thick intermediate test vessels were tested in November and December (see Refs. 1 and 2 for a summary report of the behavior of the first two vessels). The third vessel contained a fatigue sharpened flaw slightly over 2 in. deep in an axially oriented submerged-arc weld seam and was tested at 130°F. The dependence of circumferential outside surface strain 180° from the flaw on internal pressure for this vessel is shown in Fig. 3.5. The vessel in the test pit shortly after fracture is shown in Fig. 3.6. The fracture progressed out of the weld seam and into base metal and extended the length of the vessel.

Vessel 4 was tested at 75°F and contained axially oriented flaws in both base metal and weld, 90° apart. The variation with pressure of circumferential outside surface strain 180° from the base metal flaw (90°

Fig. 3.3. 4T Specimen Capsule Geometry.

NOTE:
POSITIONS NUMBERED
11 AND 12 ARE DIRECT
MEASUREMENTS OF SMALL
SPECIMENS PACKED
INSIDE THE HOLES.

Fig. 3.4. Thermocouple Positions Inside Capsule.

Fig. 3.5. Circumferential Outside Surface Strain 180° from Flaw for Vessel V-3 as a Function of Pressure.

Fig. 3.6. Picture of Intermediate Vessel V-3 Shortly After Fracture.

from the weld metal flaw) is shown in Fig. 3.7 and may be compared with Fig. 3.5. Failure occurred at the flaw in the weld and propagated in the weld metal. A picture of the failed vessel is shown in Fig. 3.8. Prior

Fig. 3.7. Circumferential Outside Surface Strain 180° from Flaw in Base Metal for Vessel V-4 as a Function of Pressure.

Fig. 3.8. Picture of Intermediate Vessel V-4 Immediately After Fracture. Note flaw in bottom right hand corner.

to testing, the flaw in the weld was thought to be about 2 1/2 in. deep. However, after failure, the initial flaw depth was found to be 3 in. The increase in flaw depth by about 50% in combination with a decrease in fracture toughness also by about 50% has been shown to account in the main for the difference noted in Figs. 3.5 and 3.7.

Preparation is under way to test intermediate vessel 6. This vessel will feature smaller flaws (1 to 1 1/2 in. deep), higher temperatures (nominally 200°F), and inside as well as outside surface flaws.

References

1. W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for July-August 1972, USAEC Report ORNL-TM-3973, Oak Ridge National Laboratory, pp. 103-108.
2. W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for September-October 1972, Non-LMFBR Programs, USAEC Report ORNL-TM-4030, Oak Ridge National Laboratory, pp. 60-66.

3.8 SPECIFIC SAFETY RESEARCH (TASK L)

(AEC Activity 04 60 80 03 1)

The investigation of the extent of ductile pipe rupture continues at Battelle Columbus Laboratories. At ORNL, the lower-bound toughness investigation of piping material is in its preliminary stages.

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4. DESIGN AND ANALYSIS OF NOZZLES, PIPES, PUMPS, AND VALVES

4.1 DESIGN CRITERIA FOR PIPING, PUMPS, AND VALVES

(AEC Activity 04 60 80 03 1)

W. L. Greenstreet S. E. Moore

Introduction

The ORNL Piping Program¹ is a USAEC-supported program being carried out by the Oak Ridge National Laboratory to study the structural behavior of nuclear reactor piping system components. This program is the U.S. Atomic Energy Commission's portion of a cooperative effort with private industry to develop basic structural response information for critical system piping components, pumps, and valves. The primary objective is to provide information for use in writing and improving design and construction codes and standards for commercial power reactor systems. However, because of the fundamental nature of the investigations a substantial portion of the information is directly applicable to other critical piping systems as well.

The program consists of about 30 carefully coordinated experimental and analytical studies, with many of the individual tasks being carried out at universities and industrial laboratories under subcontract to ORNL. At present the experimental work is confined to tees, branch connections, and elbows, while analytical studies are being carried out for most of the standard piping components including straight pipe and auxiliary attachments, socket-welding and threaded fittings, reducers, weld joints, transition joints, and flanged joints as well as elbows, tees, and branch

connections. The results of these studies are used in evaluating various analytical stress analysis methods and in developing stress indices, flexibility factors, and design rules to be used in the codes and standards.

Close cooperation between the ORNL Piping Program and the industrial code-writing bodies is maintained through the Pressure Vessel Research Committee (PVRC), and an important aspect of our management responsibility to the AEC is active participation in the work of the PVRC and the ASME code committees. The work under the ORNL program as well as other related work is discussed at regular meetings of these groups. No committee meetings were held during the two months of this reporting period.

Following is a brief summary of progress made during the months of November and December 1972.

Survey Reports

We have nothing to report this period.

Structural Analysis of Tees

These studies include experimental and analytical stress analyses of a series of 17 ANSI standard B16.9 tees, experimental and analytical stress analyses of a series of four very carefully made idealized thin-shell tee models, and analytical studies of small-branch connections which meet the fabrication requirements of the ASME Boiler and Pressure Vessel Code, Section III.²

The work on B16.9 tees includes experimental elastic stress analysis and low-cycle fatigue-to-failure tests on the 17 tees listed in Table 4.1, photoelastic stress analysis on one-half-scale models of 4 of the tees (Table 4.1, 7th column), experimental thermal stress analysis on 2 of the tees (8th column), and the development of a finite-element computer code for the elastic stress analysis of B16.9 tees. The photoelastic studies were completed under subcontract at Westinghouse Research Laboratories and reported previously. Experimental stress analysis and low-cycle fatigue tests on five of the tees have also been completed under subcontract at Southwest Research Institute (see Table 4.1, footnote f). Work is

Table 4.1. Experimental Studies for ASA B16.9 Tees

Tee No.	Manufacturer	Nominal Size (in.)	Nominal Wall Thickness (sched)	Material	Type of Fatigue Test ^a	Type of Photoelastic Test ^b	Thermal Stress Test ^c
T-1	I	6 × 6 × 6	40	SS ^d	M		
T-2	I	6 × 6 × 6	160	C ^e	P		
T-3	I	6 × 6 × 6	40	SS	M		
T-4 ^f	I	12 × 12 × 12	80	C	M		
T-5 ^f	II	12 × 12 × 12	80	C	P		
T-6 ^f	III	12 × 12 × 12	80	C	M	M	8A-T
T-7 ^f	II	12 × 12 × 12	160	SS	M	P	
T-8 ^f	II	12 × 12 × 6	40	SS	M	M	8A-T
T-9	II	12 × 12 × 6	160	SS	P	P	
T-10 ^g	III	24 × 24 × 24	40	C	M		
T-11 ^g	III	24 × 24 × 24	160	C	P		
T-12 ^g	III	24 × 24 × 10	40	C	P		
T-13 ^g	III	24 × 24 × 10	160	C	P		
T-14	III	12 × 12 × 6	40	SS	U		
T-15 ^f	I	12 × 12 × 6	40	SS	M		
T-16 ^g	III	24 × 24 × 24	10	SS	M		
T-17 ^g	III	24 × 24 × 10	10	SS	M		

^aThe loading for the fatigue test is designated as M for moment loading, P for internal pressure, and U as yet undecided.

^bPhotoelastic analysis with either internal pressure (p) or out-of-plane bending moment loading on the branch (M).

^cSteady-state temperature distribution thermal stress tests.

^dType 304L stainless steel.

^eA-1C6 grade B carbon steel.

^fTested at Southwest Research Institute (complete).

^gBeing tested at Combustion Engineering at Chattanooga.

currently under way at Combustion Engineering on the 24-in. tees (Table 4.1, footnote *g*), at Oak Ridge National Laboratory on the thermal stress tests, and at the University of California on the finite-element computer program.

Experimental Studies of B16.9 Tees (Combustion Engineering)

Experimental studies being carried out under subcontract at Combustion Engineering include elastic-response tests and low-cycle fatigue tests on the six 24-in. nominal diameter tees — T-10 through T-14, T-16, and T-17 — listed in Table 4.1. The elastic-response tests have now been completed for T-10, T-11, T-12, and T-13, and the low-cycle fatigue tests have been completed for T-10, T-11, and T-13. We plan to complete the low-cycle fatigue test for T-12 and both the elastic-response tests and fatigue test for T-16 this fiscal year. Tee T-17 will be tested next fiscal year.

All the elastic-response data from T-12 have now been carefully examined at ORNL, using diagnostic procedures discussed earlier,³ and accepted from the subcontractor. This tee was instrumented with 230 three-gage strain rosettes (690 individual gage elements) and tested with 12 loading conditions — six moment loads (3 on the run and 3 on the branch), 5 direct force loads (3 on the run and 2 on the branch), and internal pressure load. For all but one of the loadings, fewer than 2.5% of the data (or data from fewer than 17 gage elements) failed to pass the linear diagnostic tests. Data from 26 gage elements (~4%) failed the linearity tests for the direct axial thrust on the branch loading case. In all cases, however, the "bad" data were randomly distributed in location and were most often from regions of low strain. Data from only 3 of the 690 gage elements consistently failed the diagnostic tests. In no case did the maximum stresses occur at or near one of the questionable gage sites.

The low-cycle fatigue test of T-12 will be started during the next reporting period. The test assembly will be loaded with a cyclic internal pressure ranging between pressure limits of 100 and 1800 psi. The maximum pressure of 1800 psi is based on a nominal hoop stress in the run pipe (24 in., sched 40) equal to 90% of the code minimum yield strength for SA-106B steel ($S_y = 35,000$ psi) to eliminate the possibility for plastic ratchetting in the pipe. The elastic-response data indicated that the

maximum stress intensity in the tee would be 2.4 times the nominal pipe stress; its location is at the inside surface in the crotch region of the tee. A through-the-wall crack failure, based on calculations and experience gained from tees T-11 and T-13, tested earlier, is expected in 20,000 to 30,000 cycles. The test should be completed in two to three months.

The test assembly for the stainless steel tee, T-16 (24 × 24 × 24 in., sched 10), is also being prepared for testing this fiscal year. During this period, installation of all the internal surface strain gages was completed and checked. The final closure weld will be made prior to instrumenting the outside surface as soon as T-12 can be moved to a burst pit for the fatigue test.

Thermal Stress Tests of 12-in. B16.9 Tees (Oak Ridge National Laboratory)

The steady-state thermal stress tests being conducted at ORNL on the two 12-in. tees, T-6 and T-8, have been temporarily suspended. We reported previously⁴ that a significant number of strain gages on T-6 had failed during the uniform-temperature strain-gage calibration test. These failures were apparently caused by embrittlement and oxide corrosion of the factory-installed whisker wires joining the active gage element to the lead-wire tabs. Since that time the manufacturer has supplied new gages with whisker wires of a different alloy and suggested a number of changes in our installation techniques in order to minimize the future risk of gage failures.

Subsequently we conducted a number of laboratory-type tests in order to evaluate the performance of the new gages, to establish temperature profiles in the tee for the thermal gradient tests before replacing all the gages, and to check out the overall performance capabilities of the test rig and data handling systems. We replaced six rosettes on the tee with new gages before running the temperature-profile tests and then conducted three different temperature-profile tests on T-6. We also placed several new gages on a test coupon and conducted a 150-hr, 450°F controlled strain-temperature test in a laboratory furnace. The new gages performed satisfactorily during all these tests, thereby indicating that

the gage-failure problem has been solved. Gage readings from the test coupon were stable to about 50 μ in. during the 150-hr furnace test. Temperature profiles and strain levels obtained from the thermal-gradient tests, however, were not satisfactory, and a number of modifications to the test rig will be required before meaningful data can be obtained. We plan to reactivate the tests next fiscal year.

Finite-Element Stress Analysis of B16.9 Tees
(University of California)

Concurrently with the experimental studies of tees, finite-element computer code development is under way at the University of California, Berkeley. The object of this investigation is to develop computer programs for elastic and inelastic (plasticity) stress analysis and for transient heat flow analysis of ASA standard B16.9 tees. Three computer programs have been written and delivered to ORNL: one for elastic stress analysis of tees with applied mechanical loadings, one for transient and/or steady-state thermal analysis, and one for thermal stress analysis. The programs are currently being tested at ORNL against experimental data obtained under other parts of the Piping Program.

A fourth program for elastic-plastic analysis is currently being written and is expected to be completed before the end of FY 1974. The elastic-plastic stress analysis program, identified as B16.9 TJ/EPSA, is to be an extension of the elastic analysis program (B16.9 TJ/SA) by adding constitutive equations for general nonlinear isotropic strain hardening and linear kinematic strain hardening plastic behavior, and by adding the necessary algorithms for solving these equations. Because of the nonlinear stress-strain response which is inherent in elastic-plastic behavior, superposition is not possible. Therefore, geometric symmetry can be taken advantage of only if the loading is also symmetric. To allow for all possible loading conditions, for example, both in-plane and out-of-plane bending, the program will be structured to consider several different configurations: (1) one quarter of the joint, (2) two quarters of the joint with symmetry or antisymmetry about the XY plane (see Fig. 4.1), (3) two quarters of the joint with symmetry about the YZ plane, and (4) all four quarters. Modifications to the elastic program to accommodate these

Fig. 4.1. Assumed Geometry for the Finite-Element Analysis of B16.9 Piping Tees.

features have been essentially completed, and debugging is currently in progress.

Elbows, Curved Pipe, and Miter Bends

Elastic Response of Machined Elbows (Oak Ridge National Laboratory)

The last model (ME-4) in the series of four 10-in. nominal diameter long-radius machined elbows (see Table 4.2) is presently being tested at Oak Ridge. The model is instrumented with 88 three-gage strain rosettes, 16 of which are on the inside surface. Dial gages for measuring deflections and rotations are mounted on the outside surface. Planned loading conditions include internal pressure, applied bending and torsional moments, and combinations of internal pressure and moment loads.

Tests for the individual loading conditions were completed during this reporting period. Strain-gage data were obtained at internal pressure levels of 200, 400, and 600 psi. The maximum pressure was limited

Table 4.2. Machined-Elbow Test Models

Model Designation	Identification	Model Parameters
ME-1	Ideal torus	90° bend, long radius, 10.750-in. nominal OD, uniform wall 0.365 in. thick
ME-2 ^a	Ovaled torus	90° bend, long radius, 10.750-in. nominal OD, uniform wall 0.365 in. thick, flattened 5 to 8% of the diameter ^b
ME-3	Thinned elbow	90° bend, long radius, 10.750-in. nominal OD, bore eccentric 0.093 in., ^c minimum wall thickness 0.272 in., maximum wall thickness 0.458 in.
ME-4	Ovaled and thinned elbow	90° bend, long radius, 10.750-in. nominal OD, bore eccentric 0.093 in., ^c minimum wall thickness 0.272 in., maximum wall thickness 0.458 in., flattened 5 to 8% of the diameter ^b

^aIn addition to the elastic strain-gage tests, this model will be fatigue loaded to failure with a cyclic internal pressure loading.

^bAn attempt will be made to form an elliptical cross section with a major-to-minor diameter ratio between 1.05 and 1.08 and with the major axis lying in the plane of the bend.

^cThe eccentricity of the bore will be away from the center of the bend so that the minimum wall thickness is along the back of the elbow.

to 600 psi to avoid the possibility of plastically yielding the model. Maximum strain levels in this model were approximately twice as high for a given pressure as those obtained in either model ME-1 or ME-3 tested earlier.

In-plane bending, out-of-plane bending, and torsional moment loading tests were also completed with maximum loads of 188,160 in.-lb. The combined loading tests, where the model is first pressurized to a given value and then loaded with one of the applied moments, will be completed during the next reporting period. Because of the high stresses caused by internal pressure in this model, it will be necessary to use smaller loads than used for the previous models.

References

1. S. E. Moore and W. L. Greenstreet, Program Plan for the ORNL Piping Program — Design Criteria for Piping, Pumps, and Valves (July 1, 1970, unpublished).
 2. ASME Boiler and Pressure Vessel Code, Section III-1971, Nuclear Power Plant Components, American Society of Mechanical Engineers, New York 1971.
 3. W. G. Dodge and J. E. Smith, A Diagnostic Procedure for the Evaluation of Strain Data from a Linear Elastic Test, USAEC Report ORNL-TM-3415, Oak Ridge National Laboratory, November 1972.
 4. W. L. Greenstreet and S. E. Moore, Design Criteria for Piping, Pumps, and Valves, ORNL Nuclear Safety Research and Development Program Bi-monthly Report for March-April 1972, USAEC Report ORNL-TM-3831, Oak Ridge National Laboratory, pp. 99-106.
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4.2 EXPERIMENTAL AND ANALYTICAL INVESTIGATIONS OF NOZZLES

(AEC Activity 04 06 70 01 1)

W. L. Greenstreet J. P. Callahan

The distribution of stresses in the region of attachment of nozzles to pressure vessels is of particular interest to pressure vessel designers because of the high stresses produced during normal vessel operation and the complex geometries employed. The ORNL Nozzles Program is assigned the tasks of developing and verifying analytical methods for accurately determining these stresses. Included in the program are studies of single nozzles and multiple nozzle clusters attached radially and nonradially to flat plates, cylindrical shells, and spherical shells, as well as studies of perforated flat plates and spherical shells. Each individual study consists of an analysis development phase and an experimental phase designed to supply reliable data for verifying the corresponding analysis. In addition, such factors as superposition of nozzle loadings, mutual interactions of adjacent openings and nozzles, and interpretation of the resulting information in light of current design practice are under investigation. The

information developed under this program and other technically related programs is evaluated and prepared in forms suitable for use in revising existing design rules or in developing new rules for both industrial and AEC-RDT codes and standards.

These studies are conducted in cooperation with the Pressure Vessel Research Committee (PVRC) of the Welding Research Council. Program personnel actively participate in the PVRC to provide direct communication between the ongoing work and the pressure vessel industry. The PVRC also serves as a means for transmitting information to appropriate code bodies for their consideration. The incorporation of this information into design practice provides added assurance for reliable and safe performance of pressure vessels.

Project Management

As a member of the PVRC, ORNL is assigned the tasks of coordination, review, and evaluation of ongoing research programs for the PVRC Subcommittee on Reinforced Openings and External Loadings. Included in this management function are (1) direction of AEC-sponsored work, including work both at ORNL and under subcontract; (2) conduction of parameter studies in support of code-oriented evaluation and correlation work in progress at Battelle/Columbus Laboratories; (3) preparation of progress summaries and recommendations on PVRC-sponsored work; and (4) solicitation of comments and recommendations from the PVRC Subcommittee concerning AEC-sponsored studies. The next meeting of the Subcommittee on Reinforced Openings and External Loadings will be held in Houston, Texas, on January 30, 1973, at which time recent progress will be discussed.

Single Nozzles in Spherical Shells

Testing of the 7 7/8-in.-OD radial nozzle, model 807-003, has been initiated at the University of Tennessee. This configuration, which was formed by removing the nozzle internal protrusion from model 807-002, has a nozzle-to-vessel inside diameter ratio (d_i/D_i) of 0.249, a vessel diameter-to-thickness ratio (D_i/T) of 80.35, and a nozzle diameter-to-thickness ratio (d_i/t) of 40.0.

Two other models are in various stages of preparation at the University of Tennessee. The nozzle internal protrusion of the 2 5/8-in.-OD, 22 1/2° nonradial nozzle model 802-222 has been removed to form model 802-223, which has a d_i/D_i of 0.083, a D_i/T of 80.35, and a d_i/t of 40.0. The strain gages remaining from the previous tests are being checked, and all faulty or disturbed gages are being replaced.

Fabrication of the 7 7/8-in.-OD, 45° nonradial nozzle model 807-451, the last model of the planned nonradial nozzle series, is in progress. A penetration has been machined in the vessel for the nozzle, and preparations are under way for attaching the nozzle to the vessel by welding. This is the most critical stage of the fabrication process. Only one configuration, consisting of a flush nozzle, is planned for this model, which has a d_i/D_i of 0.236, a D_i/T of 80.35, and a d_i/t of 19.0.

Single Nozzles in Cylindrical Shells

A satisfactory numerical type of analysis for out-of-plane moment loading of nozzle-to-cylindrical shell configurations is under development at ORNL. A computer program of the equations has recently been completed. Trial analyses are being conducted, and the results have been used to partially debug the computer program. As a result of the limited funding available for this effort during the current fiscal year, any further work on this analysis will be deferred until FY 1974. Although some problems remain to be solved with respect to the computer program, it appears to be promising at this time.

Clusters of Nozzles

All data from the tests conducted at Auburn University on the 36 by 36 by 3/8 in. flat plate with a single 2 5/8-in.-OD nozzle attachment have been reduced to final report form by the ORNL computer. Computer processing of the data for both the flat plate with two attached nozzles and the perforated hemispherical vessel is in progress.

At Auburn University, testing of the flat plate with an attached five-nozzle cluster is under way. The model is presently being subjected to

the final series of nozzle bending moment loadings. Once this series is completed, the model will be remounted in the loading frame for application of the plate biaxial loadings, the final loadings planned for this model.

The first stage of the instrumentation scheduled for the Auburn University hemispherical vessel having four 7.00 2-in.-OD nozzles has been installed, and the model is being prepared for testing. Thus far the model has been instrumented with 681 strain gages in the form of three-gage rosettes. When the initial series of tests are completed using the present instrumentation, the data will be used as an aid in determining the full extent of gaging required to provide a complete understanding of the interaction of these relatively large and closely spaced nozzles. Part of the additional strain gaging will be used to provide additional information for evaluating the earlier two-nozzle model test data.

Reinforcement Study

Thus far under this program a number of analytical parameter studies of nozzle-to-spherical shell configurations have been completed, and the results are being used at Battelle/Columbus Laboratories to develop design charts for determining stresses for various loading conditions. The axisymmetric geometry analytical methods employed in these studies are relatively inexpensive and easy to use. However, the majority of the nozzles in pressure vessels are attached to cylindrical shells rather than to the head or to spherical shell portions of the vessels.

The stresses in the region of attachment of nozzles to cylindrical shells can be calculated by finite-element methods. Unfortunately, these methods are costly and time consuming to use. Thus far, a closed-form type of analysis¹ has been developed for internal pressure loadings of fairly simple cylinder-to-cylinder intersections having nozzle-to-vessel diameter ratios (d/D) less than one-third. The development of a numerical type of analysis for the out-of-plane loading case is discussed in an earlier section of this report.

A parameter study was conducted at Battelle/Columbus Laboratories to determine the feasibility of using axisymmetric geometry analysis methods

to provide an estimate of the stresses at nozzle-to-cylindrical shell intersections. The three types of models studied were: (1) a sphere having an equivalent diameter D_e equal to twice the diameter of the cylindrical shell (called the equivalent diameter sphere model); (2) a flat plate model as shown in Fig. 4.2; and (3) a combination of flat plate and spherical shell model as shown in Fig. 4.3 (called the plate/shell model).

The accuracy of these three models in calculating stresses for nozzle to cylindrical shell configurations was evaluated by comparing the results with calculations obtained using Eringen's¹ closed-form type of analysis for selected unreinforced geometries and with results of four photoelastic studies.

For some parameters, the results obtained using the equivalent diameter sphere model were in agreement with those obtained using Eringen's analysis; however, for other parameters the results are unconservative. It was concluded that the equivalent diameter sphere model will at best provide an estimate of maximum stresses, and by its very nature, it is unable to predict stress variations as a function of ϕ , the angular position around the nozzle.

The nomenclature used for the nozzle-to-cylindrical shell configuration is shown in Fig. 4.4.

The flat plate and combined flat plate/sphere models are basically similar, since the edge load along the plate in both cases is assumed to be proportional to $\cos \phi$, and both models are therefore capable of determining stress variations as a function of ϕ . Comparison of results obtained using the sphere/plate model and Eringen's analysis are shown in Table 4.3 for a series of nine models having a nozzle-to-vessel membrane stress ratio s/S of 1.0. The stresses are shown for the longitudinal plane, $\phi = 0^\circ$, and transverse plane, $\phi = 90^\circ$, of each model. The results are in the form of normalized membrane (σ_m) and bending (σ_b) stresses. The nomenclature of Fig. 4.4 is employed as subscripts in representing the directions of the stresses. The results obtained using the plate model were in close agreement with those obtained using the sphere/plate model; therefore only the latter are shown.

Comparisons were also made between results obtained using the sphere/plate analysis and Eringen's analysis with photoelastic test results for

$$N_p = 3/4 ST - 1/4 [\cos (2\phi)] ST \quad ; \quad Q_z = s/S \quad t/2r_p$$

WHERE $S = pD/2T$ AND D IS THE DIAMETER OF THE CYLINDRICAL SHELL

Fig. 4.2. Equilibrating Edge Loads for Flat Plate Model with Axial Pressure in Nozzle.

Fig. 4.4. Nomenclature Used in Eringen's Closed-Form Type of Analysis.

models C-1A, C-2A, and C-3A from Ref. 2 and for model WC-2AY from Ref. 3. In general, the stresses calculated using the sphere/plate model exhibited the same trends as seen in the photoelastic data, but the model did not predict maximum and minimum stresses accurately. Specifically, except for model C-1A as shown in Table 4.4, it did not accurately predict the maximum stresses. For the four photoelastic models, Eringen's solution does give a reasonable estimate of the maximum stresses.

It was concluded from this study that the axisymmetric geometry methods investigated are incapable of accurately predicting the stresses for the nozzle-to-cylindrical shell configurations. This is probably attributable to the observation that stresses at the junction vary significantly as a function of ϕ even in the case of small d/D ratios.

Table 4.3. Comparison of Normalized^a Stresses Calculated by Eringen's Theory with Those Calculated by Sphere/Plate Axisymmetric Geometry Model, $s/S = 1.0$, $D/T = 10$ and 250

$\frac{D}{T}$	$\frac{d}{D}$	Theory	ϕ	$\sigma_{m\phi}$	σ_{mp}	$\sigma_{b\phi}$	σ_{bp}	σ_{mx}	σ_{my}	σ_{bx}	σ_{by}
10	0.01	Eringen	0	2.49	0.00	0.00	0.00	0.22	2.55	1.53	0.06
		Sphere/plate	0	2.52	0.00	0.00	0.00	0.25	2.59	1.70	0.30
		Eringen	$\pi/2$	0.51	0.00	0.01	0.00	0.78	0.74	0.83	0.65
		Sphere/plate	$\pi/2$	0.48	0.00	0.01	0.00	0.75	0.71	0.66	0.40
	0.025	Eringen	0	2.48	0.01	0.02	0.00	0.20	2.54	1.53	0.06
		Sphere/plate	0	2.50	0.00	0.03	0.00	0.25	2.58	1.69	0.30
		Eringen	$\pi/2$	0.52	0.00	0.04	0.00	0.80	0.76	0.84	0.65
		Sphere/plate	$\pi/2$	0.50	0.03	0.00	0.00	0.75	0.72	0.67	0.40
	0.05	Eringen	0	2.48	0.02	0.04	0.00	0.18	2.53	1.57	0.07
		Sphere/plate	0	2.48	0.02	0.06	0.00	0.25	2.56	1.69	0.30
		Eringen	$\pi/2$	0.53	0.00	0.13	0.00	0.82	0.78	0.87	0.66
		Sphere/plate	$\pi/2$	0.52	0.00	0.09	0.00	0.75	0.74	0.71	0.42
	0.10	Eringen	0	2.54	0.04	0.03	0.02	0.15	2.56	2.04	-0.20
		Sphere/plate	0	2.47	0.04	0.16	0.02	0.05	2.55	1.57	0.33
		Eringen	$\pi/2$	0.55	0.02	0.40	0.01	0.85	0.83	1.03	0.73
		Sphere/plate	$\pi/2$	0.59	0.02	0.22	0.00	0.95	0.81	1.11	0.47
	0.25	Eringen	0	2.89	0.13	-0.28	0.15	0.32	2.90	2.42	0.18
		Sphere/plate	0	2.52	0.15	0.20	0.14	0.34	2.57	2.20	0.46
		Eringen	$\pi/2$	0.51	0.11	1.23	0.14	0.69	1.01	2.19	1.27
		Sphere/plate	$\pi/2$	0.82	0.07	0.44	0.10	0.66	0.99	1.68	0.68
250	0.01	Eringen	0	2.54	0.00	0.06	0.00	0.48	2.68	3.02	0.89
		Sphere/plate	0	2.54	0.00	0.08	0.00	0.52	2.70	3.06	0.94
		Eringen	$\pi/2$	0.50	0.00	0.12	0.00	0.52	0.66	-0.55	-0.15
		Sphere/plate	$\pi/2$	0.50	0.00	0.08	0.00	0.48	0.64	-0.58	-0.14
	0.025	Eringen	0	2.73	0.00	0.13	0.00	0.56	2.89	3.50	1.04
		Sphere/plate	0	2.63	0.00	0.44	0.00	0.51	2.78	3.40	0.91
		Eringen	$\pi/2$	0.48	0.00	0.47	0.00	0.43	0.61	-0.29	-0.07
		Sphere/plate	$\pi/2$	0.59	0.00	0.44	0.00	0.49	0.74	-0.22	-0.05
	0.05	Eringen	0	3.27	0.00	0.25	0.01	0.83	3.52	4.95	1.46
		Sphere/plate	0	2.89	0.00	0.47	0.00	0.50	3.04	4.41	1.31
		Eringen	$\pi/2$	0.43	0.00	0.93	0.00	0.16	0.50	0.58	0.20
		Sphere/plate	$\pi/2$	0.87	0.00	0.47	0.00	0.50	1.02	0.81	0.23
	0.07	Eringen	0	3.84	0.02	0.82	0.04	1.00	4.50	8.19	2.43
		Sphere/plate	0	3.13	0.01	0.55	0.02	0.50	3.28	5.30	1.58
		Eringen	$\pi/2$	0.42	0.00	1.25	0.00	-0.02	-0.44	0.23	0.10
		Sphere/plate	$\pi/2$	1.11	0.01	0.55	0.02	0.50	1.26	1.74	0.52

^aAll stresses are divided by $S = PD/2T$.

Table 4.4. Comparison of Calculated Normalized Stresses with Experimental Photoelastic Results of Nozzle-to-Cylindrical Shell Attachments

Model No.	s/S	d/D	D/T	Photoelastic Study (Maximum Stress)	Analytical Studies	
					Sphere/Plate	Closed Form
C-1A	1.04	0.05	11.9	2.45	2.41	2.50
C-2A	0.97	0.13	12.1	2.80	2.31	2.66
C-3A	1.01	0.20	11.9	3.00	2.38	2.91
WC-2AY	0.49	0.12	59.4	2.96	2.38	3.14

References

1. A. C. Eringen et al., Stress Concentrations in Two Normally Intersecting Cylindrical Shells Subject to Internal Pressure, Welding Research Council Bulletin No. 139, April 1969.
2. C. E. Taylor and N. C. Lind, Photoelastic Study of Stresses near Openings in Pressure Vessels, Welding Research Council Bulletin No. 113, April 1966.
3. M. M. Leven, Photoelastic Determination of Stresses in a Reinforced Steam Charging Connection with a Thermal Sleeve, Westinghouse Research Laboratories Report 69-907-PHOTO-R1, Jan. 14, 1969.

5. RDT STANDARDS PROGRAM

5.1 PROGRAM MANAGEMENT ACTIVITIES

(AEC Activity 04 60 80 03 1)

M. Bender	P. C. Keeble
W. F. Ferguson	R. A. Schmidt

The prime responsibility for managing the organization, preparation, and distribution of engineering standards was assigned to the Oak Ridge National Laboratory by the AEC Division of Reactor Development and Technology. ORNL is the responsible standards office for RDT standards applicable to general nuclear technology and specifically to all systems except liquid-metal-cooled systems. Those standards prepared exclusively for liquid-metal-cooled systems are under the purview of the Liquid Metal Engineering Center (LMEC), which is the responsible standards office for them.

A program that employs a computer-stored information file with selective recall capabilities is being developed to provide for routine updating of portions of the RDT standards such as references included in the standards. The program is planned to be sufficiently flexible to cover certain existing routine functions, and a flowsheet is being prepared to show the suggested sequential operational steps, manpower requirements, and estimated costs for installation and operation.

Support to American National Standards Institute (ANSI)

The ORNL Standards Group continued to provide support to the ANSI Nuclear Technical Advisory Board (NTAB) by completing preparation of the

Handbook for ANSI NTAB Area Managers and issuing the handbook to the area managers.

Additional support was provided through participation in NTAB Priorities Committee and Executive Committee meetings in November and December and through participation in a meeting with the ANSI Standards Coordinator to discuss the proposed ANSI status report form. The first issue of the status report to be prepared on the new form, which has been approved by ANSI, will be prepared by ORNL.

Summary of Other Activities

During this period, 7 approved standards and 17 amendments to approved standards were issued, and 4 draft standards (1 of which will supersede a previous issue) and 6 proposed amendments to approved standards were submitted to RDT for approval. RDT numbers and titles were assigned for 8 new standards, the number for 1 standard was changed, the title for 1 standard was revised, and 2 proposed standards were withdrawn from the program.

A total of 258 tentative standards are currently available for use, and 16 draft standards, revisions, and amendments prepared by ORNL are awaiting approval by RDT. The mailing lists, which include AEC installations and contractors, codes and standards organizations, architect-engineers, consultants, nuclear components manufacturers, and electric utilities, now contain 411 addressees who receive 791 copies of the standards and 589 addressees who receive 1192 copies of the Index of RDT Standards.

In response to telephone and memo requests, 2508 copies of RDT publications were mailed to 158 organizations, 14 of which were added to the distribution list to receive future issues of the standards or the bi-monthly index.

The Bimonthly Program Management Letter No. 5 was prepared and submitted to RDT, and the Bimonthly Status Report and Index of RDT Standards were prepared and issued.

The following approved RDT standards were issued.

Number	Title	Standards Writer
C 7-2T	Thermocouple Material, Iron-Constantan, Mineral-Oxide Insulated, Sheathed, Amendment 1 (October 1972)	ORNL
C 8-4T	Electrical Continuity-Type Liquid Metal Leak Detector (October 1972)	LMEC
C 17-8T	Annunciations for Control Systems (October 1972)	ANL
E 2-3T	Reactor Vessel for Liquid Metal Service (October 1972)	HEDL
E 6-5T	Collapsible-Rotor, Roller-Nut Control Rod Drive Mechanism for Sodium Service, Amendment 1 (December 1972)	HEDL*
E 6-19T	Core Radial Reflector for Sodium Cooled Reactors (August 1972)	HEDL*
E 6-34T	Fabrication of Core Component Pot for Liquid Metal Service (March 1972)	HEDL
E 10-3T	Tank - Liquid Metal Service, Amendment 2 (November 1972)	LMEC
E 15-2T	Requirements for Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III), Amendment 5 (November 1972)	ORNL
F 3-37T	Special Requirements for Metal Products, Amendment 2 (November 1972)	ORNL [†]
F 6-11T	Fabrication and Installation of Piping Sub-assemblies for Liquid Metal Service, Amendment 1 (November 1972)	LMEC
F6-14T	Brazing Fabrication Requirements, Amendment 1 (December 1972)	ORNL [†]
M 3-2T	Stainless and Alloy Steel Seamless Tubes (ASME SA-213 with Additional Requirements), Amendment 2 (October 1972)	ORNL
M 3-3T	Austenitic Stainless Steel Seamless Pipe (ASME SA-376 with Additional Requirements), Amendment 2 (December 1972)	ORNL
M 3-5T	Austenitic Stainless Steel Welded Tubing (ASME SA-249 with Additional Requirements), Amendment 1 (October 1972)	ORNL

*Transferred to ARD by RDT 6-72; negotiations are under way for implementation.

[†]To be transferred to HEDL.

Number	Title	Standards Writer
M 3-7T	Austenitic Stainless Steel Welded Pipe, Large Diameter (ASME SA-358 with Additional Requirements), Amendment 1 (October 1972)	ORNL
M 3-11T	Carbon and Low Alloy Steel Welded Pipe (ASME SA-155 with Additional Requirements), Amendment 1 (October 1972)	ORNL
M 3-16T	Carbon and Alloy Steel Pipe (ASME SA-333 with Additional Requirements), Amendment 1 (November 1972)	ORNL
M 3-17T	Nickel-Molybdenum-Chromium Alloy Welded Pipe (ASME SA-358 with Additional Requirements) (October 1972)	ORNL
M 4-1T	Carbon Steel Castings (ASME SA-216 with Additional Requirements), Amendment 1 (November 1972)	ORNL
M 4-4T	Alloy Steel Castings (ASME SA-217 with Additional Requirements), Amendment 1 (November 1972)	ORNL
M 12-2T	Calcium Silicate Block and Pipe Thermal Insulation (ASTM C533 with Additional Requirements), Amendment 3 (November 1972)	ORNL
M 12-3T	Mineral Fiber Hydraulic-Setting Thermal Insulating and Finishing Cement (ASTM C449 with Additional Requirements), Amendment 2 (November 1972)	ORNL
M 14-2T	Fuel and Control Assembly Tag Gas (October 1972)	HEDL

The following documents prepared by ORNL were submitted to RDT for approval.

C 6-2	Differential Pressure Transmitter, Pneumatic Output Signal (November 1972 draft)
C 7-4T	Thermocouple Material, Copper-Constantan, Magnesium-Oxide Insulated, Sheathed, Amendment (11-28-72 draft)
C 7-5	Thermocouple Material, Chromel-P and Alumel, Solid Conductor (Bare, Fiberglass Insulated, and Sheathed over Fiberglass Insulation) (October 1972 draft)
C 7-15T	Thermocouple Connectors and Thermocouple Connector Panels, Amendment (11-28-72 draft)
F 3-6T	Nondestructive Examination, Revision (November 1972 draft)

- *F 5-1T Cleaning and Cleanliness Requirements for Nuclear Components, Amendment (11-17-72 draft)
- *F 6-5T Welding Qualifications (Supplement to ASME Boiler and Pressure Vessel Code Section IX), Amendment (11-10-72 draft)
- *F 6-14T Brazing Fabrication Requirements, Amendment (11-27-72 draft)
- M 3-3T Austenitic Stainless Steel Seamless Pipe (ASME SA-376 with Additional Requirements), Amendment (11-27-72 draft)
- M 3-27 Seamless and Welded Small Diameter Austenitic Stainless Steel Tubing (ASMT A632 with Additional Requirements) (October 1972 draft)

Numbers and titles for the following new standards were assigned.

		Standards Writer
C 13-2	Computer, On-Line Reactivity	GE Breeder Reactor Division
[†] E 6-10	Temperature and Liquid Level Control Monitor, Port Plug	HEDL
E 6-11	Simulated Core Assembly for Nuclear Reactors	HEDL
E 6-12	Simulated Test Position Assembly for Use in Nuclear Reactor Core	HEDL
E 10-8	Steel Containment Vessel	ORNL
F 1-3	Preparation of Incident Reports	RDT
F 5-8	Cleaning and Restoration of Systems Following Liquid Metal Service or Contamination	LMEC
F 9-3	Energy Dependent Damage Functions	HEDL

RDT E 2-4, Shield Plugs and Closure Caps for Vacant Penetrations - Sodium Cooled Reactor Vessel Head, was renumbered from E 2-1. Proposed standards M 1-12, Aluminum and Aluminum Alloy Welding Electrodes (ASME SFA-5.3 with Additional Requirements), and M 1-14, Gold-Nickel Brazing Filler Metal (ASME SFB-5.8 with Additional Requirements), were withdrawn; and the title for P 2-6 was changed to Undervoltage Relay for Plant Protection System Use.

*To be transferred to HEDL.

[†]Title changed.

5.2 MECHANICAL EQUIPMENT AND SYSTEMS

(AEC Activity 04 60 80 03 1)

H. G. Arnold	J. H. Evans
C. A. Burchsted	W. R. Gall
W. A. Bush	R. W. Dehoney

Nuclear Code Supplements

Amendment 5 to RDT E 15-2T, Requirements for Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III), was approved and issued. A revision of E 15-2 which will incorporate all approved amendments is being prepared.

Air Handling

An analytical comparison of RDT M 16-1T, Gas-Phase Adsorbents for Trapping Radioactive Iodine and Iodine Components, and the proposed ASTM standard for radioactive testing of activated carbon was completed, and comments were submitted to the ASTM committee.

Review of draft standard E 9-7, Fans, Blowers, and Compressors for Dry Gas Circulation, was completed by ORNL, and comments were correlated and transmitted to HEDL for evaluation.

A meeting of the NTAB Management Area 11 was attended at the Ginna Reactor in Rochester, New York, to continue planning of the program for on-site fuel handling standards. A preliminary scope statement for Section III of the on-site fuel handling project description was prepared following the meeting and submitted to the members of the planning group for review.

A meeting was held with a representative of AEC Regulatory to review the second draft of the proposed AEC criteria for engineered-safeguard air-cleaning systems.

5.3 INSTRUMENTATION, CONTROLS, AND ELECTRICAL

(AEC Activity 04 60 80 03 1)

J. L. Anderson	T. L. Hudson
K. R. Carr	C. S. Lisser
S. J. Ditto	J. A. Russell

Amendment 1 to RDT C 7-2T, Thermocouple Material, Iron-Constantan, Mineral-Oxide Insulated, Sheathed, was approved and issued for use.

The following standards and amendments were completed and submitted to RDT for approval.

- C 6-2 Differential Pressure Transmitter, Pneumatic Output Signal
- C 7-4T Thermocouple Material, Copper-Constantan, Magnesium-Oxide Insulated, Sheathed, Amendment (11-28-72 draft)
- C 7-5 Thermocouple Material, Chromel-P and Alumel, Solid Conductor (Bare, Fiberglass Insulated, and Sheathed over Fiberglass Insulation)
- C 7-15T Thermocouple Connectors and Thermocouple Connector Panels, Amendment (11-28-72 draft)

ORNL review of the following draft standards was completed, and comments were forwarded to the standards writers.

- P 4-4 Electric Heaters — Simulated Pressurized Water Reactor Fuel Pins
- C 10-5 Eddy-Current Flowmeter Power Supply and Signal Conditioning Electronics
- C 17-5 Metal-Sheathed, Mineral-Insulated Cable Bulk Material

A revised draft of C 7-17, Platinum Resistance Thermometer, was submitted to HEDL for review. Comments are now being incorporated into the final draft of the standard prior to submittal to RDT for approval.

A meeting of the ASTM E20.04 committee was attended. Of particular interest to the Standards Program were plans for converting the following RDT standards into ASTM standards.

RDT Standards C 7-2T, Thermocouple Material, Iron-Constantan, Mineral-Oxide Insulated, Sheathed, and C 7-4T, Thermocouple Material, Copper-Constantan, Magnesium-Oxide Insulated, Sheathed, have been factored into a draft of Specification for Sheathed Base-Metal Thermocouple Material. This is a general specification with optional testing and inspection and quality assurance requirements which is scheduled to be submitted to the committee for ballot by the end of the fiscal year.

Standards C 7-1T, Thermocouple Material, Iron and Constantan, Solid Conductor (Bare, Fiberglass Insulated, and Sheathed over Fiberglass Insulation), and C 7-3T, Thermocouple Material, Copper and Constantan, Solid Conductor (Bare, Fiberglass Insulated, and Sheathed over Fiberglass Insulation), are to be combined into a specification entitled, Fiberglass Insulated Base-Metal Thermocouple. This will be a specialized standard, mainly for reactor use, and the decision of ASTM to issue it is not yet definite.

The proposed ASTM standard which will be based on E 5-2T, Electric Heater and Connector Assembly for Pressurizer for Pressurized Water Reactors, will not include the heater connector. It is planned to expand the committee membership to include representatives from three suppliers of pressurized-water reactors, heater suppliers, and utilities groups. A draft of the standard is scheduled to be completed by the end of the fiscal year.

5.4 PROGRAMMATIC AND PROCEDURAL STANDARDS

(AEC Activity 04 60 80 03 1)

J. W. Anderson	F. L. Hannon
H. G. Arnold	R. M. Fuller
C. A. Burchsted	R. M. Holmes
G. L. Copeland	T. W. Pickel
W. C. T. Stoddart	

Quality Assurance

The following activities were performed in support of the ANSI N45 subcommittee and work groups.

1. A flow chart was prepared to show activities involved in the development and approval of the N45.2 and supporting standards from the time the need for the standard is recognized through final approval and publication.

2. The status report on development of the N45.2 and supporting standards was revised and submitted to the NTAB Priorities Committee and the N45 Subcommittee. Quarterly publication of this report is planned.
3. The index of ANSI N45.2 and supporting standards was submitted to the chairman of the subcommittee, approved, and distributed to the members of the subcommittee.
4. A preliminary draft of the Manual of Operating Procedures for the N45.2 Subcommittee and Work Groups was presented for review in a meeting of the Area Management Advisory Group, revised to incorporate comments received during the review meeting, and distributed to the Subcommittee for review and comments.
5. The comments received on the N45.2.8 mechanical equipment standard were distributed to the work group prior to the December meeting of the work group. A revised draft of the standard was prepared in accordance with resolutions and comments resulting from the meeting and was distributed to the Subcommittee for concurrence.
6. Drafts of the N45.2.9 (records) standard and the N45.2.10 (definitions) standard were reviewed, and comments were prepared and submitted to the subcommittee and work group chairmen.
7. A meeting of the ANSI Joint Steering Committee for Nuclear Power Standards was attended.
8. A meeting of the N45.2.1 subcommittee was attended to make final changes to the proposed standard on Construction Phase Cleaning and Cleanliness Control and to plan the ongoing program of the subcommittee.

Arrangements were made for the forthcoming meeting of the ASQC Subcommittee on Interface. The completed matrix comparison of quality assurance standards, which was submitted to Westinghouse Electric Corporation for publication, has been distributed to the Nuclear Power Technical Committee for review and comments.

High-Temperature Design

The third meeting of the Task Force preparing the new RDT High-Temperature Design Standard was held November 20, 1972. At this meeting the ORNL representative presented (1) drafts of Chapters 2 and 3 of Volume II,

for which ORNL has the lead responsibility; (2) an analytical study performed to evaluate proposed ratchetting design criteria and recommended criteria based on this study; (3) information prepared on weld criteria, crack propagation, and irradiation effects; and (4) comments on the draft of Volume I of the standard, which was distributed at the previous meeting.

A task group consisting of six people representing RDT, Westinghouse Advanced Reactors Division (ARD), ORNL, General Electric (GE, Atomics International (AI), and Westinghouse was appointed to complete the final draft of Volume I. At a meeting of this task group in December, the ratchetting design criteria proposed by ORNL were adopted for inclusion in Volume I and have been rewritten in the appropriate form.

The next meeting of the Task Force is planned for mid-January 1973, at which time it hopes to consider the "final" draft of Volume I and a complete draft of Volume II.

Construction, Installation, and Transportation

Review comments received on draft standard F 8-7, Transportation of Critical Components and Equipment, have been incorporated, and the final draft is being prepared for submittal to RDT for approval.

The ORNL review of F 8-11, Tiedown for Transport of Containers Greater than One Ton, was completed, and comments were prepared and submitted to Savannah River Operations Office.

Review comments were submitted to the ANSI N14 task groups on the Administrative Guide for Control of Radioactive Contamination and Exposure to Radiation During the Packaging, Transport and Unpacking of Radioactive Materials, and Administrative Guide for Obtaining Department of Transportation Permits for Radioactive Materials Shipments.

Testing

Amendment 2 to RDT F 3-37T, Special Requirements for Metal Products, which was prepared to update the standard to the 1972 Summer Addenda to the ASME Code, was approved and published.

The proposed ANSI N14 standard for leak testing of radioactive shipping packages was reviewed, and comments were submitted to the committee chairman.

Air Cleaning

Two proposed amendments to F 5-1T, Cleaning and Cleanliness Requirements for Nuclear Components, were prepared and submitted for RDT approval during this period. It was suggested that the second amendment, which was prepared in cooperation with the RDT Components Branch, be combined with the previously submitted amendment proposed by HEDL.

5.5 MATERIALS AND FABRICATION

(AEC Activity 04 60 80 03 1)

R. M. Fuller F. L. Hannon
O. A. Kelly

The following amendments which were prepared to update the standards to the 1972 Summer Addenda to the Code have been approved by RDT and issued for use.

- F 6-14T Brazing Fabrication Requirements, Amendment 1 (December 1972)
- M 3-2T Stainless and Alloy Steel Seamless Tubes (ASME SA-213 with Additional Requirements), Amendment 2 (October 1972)
- M 3-3T Austenitic Stainless Steel Seamless Pipe (ASME SA-376 with Additional Requirements), Amendment 2 (December 1972)
- M 3-5T Austenitic Stainless Steel Welded Tubing (ASME SA-249 with Additional Requirements), Amendment 1 (October 1972)
- M 3-7T Austenitic Stainless Steel Welded Pipe, Large Diameter (ASME SA-358 with Additional Requirements), Amendment 1 (October 1972)
- M 3-11T Carbon and Low Alloy Steel Welded Pipe (ASME SA-155 with Additional Requirements), Amendment 1 (October 1972)
- M 3-16T Carbon and Alloy Steel Pipe (ASME SA-333 with Additional Requirements), Amendment 1 (November 1972)

- M 4-1T Carbon Steel Castings (ASME SA-216 with Additional Requirements), Amendment 1 (November 1972)
- M 4-4T Alloy Steel Castings (ASME SA-217 with Additional Requirements), Amendment 1 (November 1972)

In addition to the above, Amendment 3 (November 1972) to RDT M 12-2T, Calcium Silicate Block and Pipe Thermal Insulation (ASTM C533 with Additional Requirements), and Amendment 2 (November 1972) to RDT M 12-3T, Mineral Fiber Hydraulic-Setting Thermal Insulating and Finishing Cement (ASTM C449 with Additional Requirements), were also issued for use following approval by RDT. These amendments were required to correct discrepancies resulting from the recently published revision of RDT M 12-2T, Test Requirements for Thermal Insulating Materials for Use on Austenitic Stainless Steel.

RDT comments were incorporated into new standard M 3-17T, Nickel-Molybdenum-Chromium Alloy Welded Pipe (ASME SA-358 with Additional Requirements), prior to publication.

The proposed amendment to F 6-5T, Welding Qualifications, previously submitted to RDT for approval was redrafted to incorporate RDT comments and the request for an amendment from ARD and was resubmitted for approval.

The following standards were revised to conform to the 1972 Summer Addenda to the Code and to incorporate appropriate comments, and drafts of these revisions are ready for submittal to RDT for approval.

- M 2-3T Carbon and Alloy Steel Welding Fittings (ASME SA-234 with Additional Requirements)
- M 3-9T Nickel-Iron-Chromium Alloy Seamless Pipe and Tubing (ASME SB-407 with Additional Requirements)
- M 3-10T Nickel-Molybdenum-Chromium Alloy Seamless Pipe and Tubes (ASME SB-167 with Additional Requirements)
- M 5-4T Nickel-Chromium-Iron Alloy Plate, Sheet, and Strip (ASME SB-168 with Additional Requirements)
- M 6-3T Alloy Steel Bolting Material for High Temperature Service (ASME SA-193 with Additional Requirements)
- M 6-6T High-Strength, High-Temperature Bolting Materials (ASME SA-453 with Additional Requirements)
- M 7-4T Nickel-Chromium-Iron Alloy Rod and Bar (ASME SB-166 with Additional Requirements)

RDT M 3-27, Seamless and Welded Small Diameter Austenitic Stainless Steel Tubing (ASTM A632 with Additional Requirements), was revised to incorporate review comments and submitted to RDT for approval.

The drafts of the following new materials standards have been revised and will be redistributed for review and comments.

M 1-13	Aluminum and Aluminum Alloy Welding Rods and Bare Electrodes
(ASME SFA-5.10 with Additional Requirements)
M 2-10	Aluminum Alloy Welding Fittings (ASTM B361 with Additional
Requirements)
M 2-13	Aluminum Alloy Forgings (ASME SB-247 with Additional Re-
quirements)
M 3-14	Aluminum Alloy Seamless Tubes (ASME SB-234 with Additional
Requirements)

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6. GENERAL NUCLEAR SAFETY STUDIES

6.1 SEISMICITY AND THE SEISMIC RESISTANCE OF NUCLEAR POWER PLANTS

(AEC Activity 04 60 40 01 1)

J. Foster G. C. Robinson
 G. D. Whitman

The AEC seismic research program has involved an extensive effort to identify the full range of the seismic effects that require consideration in the siting, design, and construction of nuclear power plants. Many of the areas in which methods of prediction, design, and analysis are being studied are applicable to engineering structures in general and are not of exclusive interest to the AEC. The AEC has undertaken to support seismic research work in those areas in which nuclear power plant siting and design will benefit most directly and seeks to make full use of research supported elsewhere as useful results become available.

The major efforts under the seismic program during this period have consisted of work on three tasks under the soil study subcontract with Shannon & Wilson-Agbabian Jacobsen Associates (SW-AJA). These efforts relate to the development of an in-situ test probe for determining dynamic soil shear moduli, studies of cohesionless soils using shake tables to clarify conditions under which soils liquefy, and the preparation of a report of recommended procedures for evaluation of the seismic response of soil deposits at engineering sites. In addition to this subcontract work, preliminary drafts of two requests for proposals (RFP) have been prepared and submitted to the AEC for review and comment.

Additional modifications have been made to the trailer-mounted in-situ soil test system. The 3-in.-diam probe drive rod has been replaced by a 5-in.-diam drive to overcome the bowing which caused unequal tensile and compressive forces to be delivered down the hole to the soil test probe. An instrumented 1-ft length of 5-in. drive tube has been fabricated with four strain gages located at 90° intervals around the inside surface to measure the vertical oscillating forces delivered to the soil by the test probe. The trailer platform supports were improved by adding a support column from the ground to the trailer frame under the probe drive rod bearing.

Tests have been conducted with these improvements, and good data have been obtained at higher strain levels than were previously possible. Two improvements have been made in the sensing and data-recording equipment: more sensitive geophones have been installed in the sensing holes, and a dual-trace oscilloscope system has been assembled to permit more accurate data collection and phase shift determinations. The overall system is now working well and producing good data records. Testing at the present Fort Lawton site is almost completed, and plans are being made to move the trailer to a more sandy test site at Fort Lawton to conduct further tests of the apparatus in a different soil type.

The analytical work in support of the in-situ test probe development has examined the influence of probe anchor design on pullout resistance. The soil stress patterns associated with two- and four-segment anchors were calculated, and results to date indicate a slight advantage for the four-segment anchor in resisting pullout under heavy vibratory loads. It is important that anchor design and performance be understood as in-situ tests are performed at higher soil loads and strain levels. This analytical support work is continuing.

Fabrication work on the large and small soil test boxes has been completed at the Richmond Field Station of the University of California at Berkeley. These boxes have been built to contain sand soil samples for testing under dynamic conditions. Some difficulties were encountered in depositing controlled soil samples in the test boxes using the sand spreading machine, but these problems have been solved. The sealing membrane

installation over the soil specimen has been tested and found to be satisfactory. Present work is devoted largely to techniques of instrument placement in the soil test specimens, and initial test runs have been conducted for the purpose of instrument calibration. These tests are near completion and will be followed by the planned series of instrumented soil specimen tests.

Work on the preparation of a report of recommended procedures for evaluation of the seismic response of soil deposits at nuclear plant sites is continuing. It is presently planned that this report will consist of four parts — Introduction, Scope of Investigations and Studies, Analysis of Ground Response, and Recommended Procedure for Evaluating Ground Response. Parts I and II have been completed and are undergoing internal review by the subcontractor. Parts III and IV require some analytical studies of soil column dynamic responses, and these studies are in progress.

Two RFP's have been prepared in preliminary form and submitted to AEC-RDT for review and comments. These were prepared in response to AEC direction. One relates to the conduct of an investigation to identify and, to the extent possible, eliminate the apparent wide variability in the results obtained from the standard penetration test (SPT) which is widely used to estimate the properties of subsurface soils at engineering sites. The second outlines a comparative study of the most commonly used structural modeling and analytical methods for estimating the dynamic response of nuclear power plant structures and systems to earthquakes.

6.2 IGNITION OF CHARCOAL ADSORBERS BY
FISSION PRODUCT DECAY HEAT
(AEC Activity 01 03 00 00)

R. A. Lorenz W. J. Martin
 H. Hagao*

An experiment to demonstrate the consequences of iodine decay heating of a charcoal bed is being assembled and tested for safe operation. The objective of this work is to provide information to enable confident prediction of the desorption of iodine, and possible ignition of the charcoal, at abnormally low flow rates. The installation of apparatus to determine experimentally the consequences of radioactive decay energy of iodine adsorbed on charcoal is nearing completion. The large charcoal adsorbers for the hot-cell ventilation system have been installed, and the required testing has been scheduled. The experiments will ultimately use several thousand curies of ^{130}I to provide radioactive decay energy in a charcoal bed of approximately 28 cm^3 volume.

The series of experiments in the hot cell will begin with low radioactivity and increase by factors of 10. In addition to the question of ignition, we intend to determine the effect of radiation on the release of iodine from charcoal at 120 and at 150°C . Since decay heating is negligible at low activity levels, and since we will not have the supplementary charcoal bed heater in the hot-cell installation, we are planning to adjust the bed temperature for these experiments by using the inlet heaters on the tubing and flange below the test bed assembly. The schedule of operation for the first hot-cell experiment is given in Table 6.1.

Since the only heating within the charcoal bed itself will be from the radioactivity of ^{130}I , we must know the activity accurately. We irradiated four ampuls containing 7.6 to 13.8 mg of PdI_2 in the pneumatic tube of the ORR in order to verify the activation cross section and our methods and consistency of preparation, assay, and calculation. The thermal-neutron flux was determined by means of Co-Al flux monitor wire to be 2.46×10^{13} neutrons/ $\text{cm}^{-2}\text{ sec}^{-1}$ at the irradiation position. The

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Table 6.1. Schedule of Operations — Hot-Cell Experiment No. 1

Time from Opening of Capsule (hr)	Operation	Bed Air Velocity at 25°C (fpm)	Bed Temperature (°C)
0-1	Decompose PdI_2 at 680°C	30	100
1-3	Raise inlet temperature to 120°C; collect desorbed iodine on trap 1	30	120
3-4	Reduce air flow to 4 fpm; begin collecting desorbed iodine on trap 2; measure temperatures ac- curately	4	120
4-12	Raise inlet temperature to 150°C; continue collecting desorbed io- dine on trap 2	4	150

^{130}I specific activity for a 5-min irradiation was found to be 244 ± 4 Ci/mg of PdI_2 by gamma-ray spectrometry. This value agrees very well with our calculated value of 242 Ci/mg PdI_2 using the measured flux, 12.3-hr half-life, 29.41-b activation cross section, and 86.1% ^{129}I isotopic concentration of the NaI stock solution from which the PdI_2 was prepared.

A computer program has been written to calculate the heat balance across the charcoal test bed. In the heat balance the heat input terms include electric heat [only for experiments 1 to 13 (Ref. 1), which used a calibrated electric heater in simulation of decay heat], radiation decay heat, and heat from oxidation of charcoal. Heat output terms include heat carried by the air flow, heat lost from the walls of the apparatus by conduction, and heat content of the charcoal, thermocouples in the bed, and the inner walls of the quartz Dewar flask. Data from the first 13 runs will be used to verify our previous calculations of heat loss from the apparatus, and we will calculate the charcoal oxidation rate for each type of charcoal as a function of temperature.

The rate of heat release resulting from the oxidation of charcoal at moderate temperatures must be known in order to predict the temperature rise in charcoal beds in accident situations that include abnormally low

air flow through the charcoal. If the charcoal becomes heated to a moderate temperature level as a result of a combination of very high ambient temperature and heat from the decay of radioactive iodine, then a very low rate of air flow (above zero but perhaps less than 1 or 2 fpm) will merely supply oxygen and therefore contribute to additional heating, but high rates of air flow (perhaps above 10 fpm) will provide a cooling action. The critical charcoal temperature and air flow range over which increasing temperature could occur would depend upon the chemical reactivity of the charcoal (which is affected by the moisture content of the air) and the heat loss rate from the adsorber system. A similar situation was analyzed by Bratzler² for a nonnuclear installation, and two examples of the effect of oxidation heat were given in a recent progress report.³

The final safety analysis report for the PdI_2 irradiation in the HFIR has been approved by the reactor operations staff and by the experiment review committee. The final safety analysis report for the hot-cell portion of the experiment is being reviewed by the Radioactive Operations Committee of the Laboratory. Approval has been given for the low-radioactivity experiments.

References

1. R. A. Lorenz and W. J. Martin, Ignition of Adsorbers by Fission Product Decay Heat, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May-June 1972, USAEC Report ORNL-TM-3900, pp. 122-125, Oak Ridge National Laboratory.
2. K. Bratzler, Die Selbstentzündung des Kohlenstoffs als verfahrenstechnisches Problem, *Chemie-Ing.-Techn.*, 28(8/9): 569-576 (1956); being translated as ORNL-tr-2568, The Spontaneous Ignition of Carbon as a Process Problem.
3. R. A. Lorenz and W. J. Martin, Ignition of Charcoal Adsorbers by Fission Product Decay Heat, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May-June 1972, USAEC Report ORNL-TM-3973, pp. 142-145, Oak Ridge National Laboratory.

7. NUCLEAR SAFETY INFORMATION

7.1 NUCLEAR SAFETY INFORMATION CENTER

(AEC Activity 04 60 70 01 1)

J. R. Buchanan Wm. B. Cottrell

The Nuclear Safety Information Center was established in 1963 by the USAEC Division of Reactor Development and Technology to collect, evaluate, and disseminate nuclear safety information to governmental agencies, research and educational institutions, and the nuclear industry. The Center's basic computer file now contains references on over 71,300 documents.

Computer Programming and Processing

The remaining NSIC programs implemented in F-level COBOL have been converted to ANS COBOL.

The KWIC list of key words and their respective codes have been modified for easier readability.

SDI Program

The SDI (selective dissemination of information) program distributes biweekly descriptions of all documents known to be of particular interest to the recipient. During this period 32 new participants were added to the program and 20 were removed, for a total of 2097 participants. Twenty-three existing profiles were revised.

PPIF Program

The PPIF (program and project information file) program is a complete file of nuclear safety research and development projects, which is updated triannually and the information distributed to recipients through a modification of the SDI program. During this period 3 new participants were added to the program, for a total of 280 PPIF users.

Reports Status

The following report has been published and is now available for purchase from the National Technical Information Service, U.S. Department of Commerce, Springfield, Virginia 22151:

<u>Report No.</u>	<u>Title and Authors</u>	<u>Price</u>
ORNL-NSIC-104	Operation of the Nuclear Safety Information Center, CY-1971, J. R. Buchanan and W. B. Cottrell	\$3.00

Nuclear Standards Activities

The 1970 Directory of Personnel Involved in the Development of Nuclear Standards in the United States (ORNL-NSIC-75) is presently being revised and updated. Information regarding the nuclear standards activities of the employees of some 500 companies or organizations has been entered into the computer file and is presently being compared for duplication with parallel data supplied by standards-generating organizations. It is anticipated that the final data will be entered by the end of January, with publication of the revised directory in March.

The Compilation of Nuclear Standards, Part I: U.S. Activities, is presently under revision to be published as the 1972 edition. Data are now being collected to provide for revision of Part II: National (Excluding U.S.) and International Activities - 1972. Publication of both parts is scheduled for early 1973.

Summary of Information Activities

During the report period, the following requests were received and answered by NSIC personnel:

Accessions added to the computer file	1019
Technical inquiries (letter or telephone)	145
Requests for NSIC reports and other routine inquiries	139
Additions to NSIC reports announcement list	2
Visits for consultation or use of reference material	16
Additions to SDI program (net)	12
Additions to PPIF program (net)	3

7.2 TECHNICAL PROGRESS REVIEW *NUCLEAR SAFETY*

(AEC Activity 04 60 80 01 1)

J. P. Blakely Wm. B. Cottrell

Nuclear Safety, a Technical Progress Review, is prepared and edited at ORNL on a bimonthly schedule under contract with the AEC Office of Information Services. The November-December 1972 issue (Vol. 13, No. 6, 84 pages) contained articles by authors from Ispra, UKAEA, Harvard, Purdue, ORNL, and the *Nuclear Safety* staff. The issue also contained the Vol. 13 topical index, generated by the Technical Information Center.

The lead article dealt with Ispra's contribution to reactor safety as reported by Jack Randles of the Ispra Scientific Directorate. Euratom's Joint Research Centre at Ispra has had an extensive program in reactor-safety research and development for several years. In the fast reactor safety field, it included work on the interaction between uranium dioxide and sodium and on the dynamic loading of liquid-metal-cooled fast breeder reactor containment. Research in support of water-reactor safety covers depressurization and emergency-core-cooling-system studies. A more generalized research effort is under way in the areas of reactor dynamics, reliability theory, early failure detection, and fracture mechanics and pipe rupture.

Other articles dealt with radiolytic hydrogen generation after loss-of-coolant accidents in water-cooled reactors (by H. E. Zittel, ORNL Analytical Chemistry Division); some preliminary considerations relating to an equation of state for irradiated nuclear fuel (by A. J. Brook, UKAEA Safety and Reliability Directorate); effect of rupture in a pressurized noble-gas adsorption bed (by D. W. Underhill, Harvard University); the emerging role of the campus radiation-safety officer (by P. L. Ziemer, Purdue University); safety-related occurrences reported in June and July 1972 (by W. R. Casto, *Nuclear Safety* staff); summaries of RDT-sponsored R&D projects (compiled by Myrtle Sheldon, *Nuclear Safety* staff); and general administrative activities, action on reactor and other projects undergoing regulatory review or consideration, and proposed AEC rule changes as of Aug. 15, 1972 (all compiled by J. P. Blakely, *Nuclear Safety* staff). The Editor's Page was devoted to a description of the NSIC SDI program, and the recent NSIC siting bibliography was pictured and captioned on the inside front cover.

Photo-ready copy for *Nuclear Safety*, 14(1), January-February 1973, was scheduled to be sent to the printer in mid-January for distribution in mid-February. Draft material for the March-April issue of *Nuclear Safety*, 14(2), was distributed for external review in early January. Subsequent issues of *Nuclear Safety* through November-December 1973 are in various stages of development. Potential contributors are encouraged to submit articles to the editors for consideration.