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GAS-COOLED REACTOR PROGRAMS

HIGH-TEMPERATURE GAS-COOLED REACTOR TECHNOLOGY DEVELOPMENT PROGRAM

ANNUAL PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1987

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AUTHORS/CONTRACTORS

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Gas-Cooled Reactor Programs

HIGH-TEMPERATURE GAS-COOLED REACTOR TECHNOLOGY
DEVELOPMENT PROGRAM ANNUAL PROGRESS REPORT
FOR PERIOD ENDING DECEMBER 31, 1987

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**HIGH-TEMPERATURE GAS-COOLED REACTOR TECHNOLOGY
DEVELOPMENT PROGRAM ANNUAL PROGRESS
REPORT FOR PERIOD ENDING DECEMBER 31, 1987**

FOREWORD

The High-Temperature Gas-Cooled Reactor (HTGR) Program being carried out under the U.S. Department of Energy (DOE) continues to emphasize the development of modular high-temperature gas-cooled reactors (MHTGRs) possessing a high degree of inherent safety. The emphasis at this time is to develop the preliminary design of the reference MHTGR and to develop the associated technology base and licensing infrastructure in support of future reactor deployment. A longer-term objective is to realize the full high-temperature potential of HTGRs in gas turbine and high-temperature, process-heat applications. Examples of the latter include steam reforming of methane, steam gasification of coal, and thermochemical water splitting.

The overall HTGR Program activities are given in a Summary Level Program Plan (SLPP) that summarizes the work scope of the program participants, identifies program milestones, and provides information on coordination among participants. The HTGR Program is carried out under the Division of HTGRs of DOE, with a high degree of coordination being maintained among program participants. In addition to DOE, the present program participants and their key areas include General Atomics (GA), reactor design; Gas-Cooled Reactor Associates (GCRA), utility guidance and review; Oak Ridge National Laboratory (ORNL), lead laboratory for base technology development; Combustion Engineering, Inc. (CE), reactor vessel and heat transfer components; Bechtel National, Inc. (BNI), architect-engineer (AE) for the nuclear island; Stone & Webster Engineering Corporation (SWEC), AE input to licensing activities; and Idaho National Engineering Laboratory (INEL), program review and evaluations. In addition, during the past year, General Electric Company performed work on plant controls and simulation, and SWEC performed work as AE for the balance-of-plant (BOP).

The SLPP is organized into four major elements under a work breakdown structure (WBS) arrangement. The four elements include WBS 1000, HTGR Technology; WBS 5000, MHTGR Design; WBS 6000, Advanced Systems; and WBS 9000, Program Development. Most of the national program effort is under element WBS 5000, with emphasis also being placed on activities under WBS 1000 and WBS 9000. Less emphasis is placed on WBS 6000 activities at this time. Most of the ORNL effort is concentrated under WBS 1000, HTGR Technology Development; the subelements include 1400, 1600, and 1900. WBS 1400 covers technology transfer carried out under international cooperative programs; WBS 1600 covers technology development on HTGR fuels, fission product behavior, graphite, metals, and MHTGR safety; and WBS 1900 covers technology program management activities. Under WBS 5000, ORNL provides reactor internals design support through shielding analyses under subelement WBS 5211. Under WBS 6000, ORNL is engaged in technology development for advanced materials under subelement WBS 6600. Under WBS 9000, ORNL participates in subelements WBS 9100 (program development, planning, and control); WBS 9200 (evaluation of utility/user requirements and design efforts); WBS 9300 (licensing support activities); and WBS 9900 (support for the plant design control office which oversees overall program activities). A summary of ORNL's participation in the national HTGR program, as identified in the SLPP, follows along with an indication of interactions with other program activities. Interactions include work being carried out by other program participants under subcontract to ORNL. In the latter case, only work directly related to ORNL SLPP activities is reported here.

WBS 1000

The work under WBS element 1000 involves HTGR technology development. The subelements under which ORNL work is reported include WBS 1400, technology transfer; WBS 1600, base technology, which covers the determination of basic properties of materials under MHTGR conditions, including fuel performance and fission product behavior (WBS 1601); graphite studies (WBS 1602); metals studies (WBS 1603); safety technology development

(WBS 1604); and WBS 1900, which covers the management of the technology program. Subelements are described in the following paragraphs.

WBS 1400, Technology Transfer: Work includes the coordination of activities for efficient technology transfer of HTGR information from foreign countries to the United States under international cooperation program agreements. The ORNL effort is associated with cooperative technology exchange between the United States and the Federal Republic of Germany (FRG), and the United States and Japan. The work with FRG covers technology information on fuel performance, fission product behavior, graphite studies, metals development, reactor physics evaluation, and MHTGR safety/licensing activities. The work with Japan covers technology information on fuel, graphite, metals, and safety/licensing activities.

WBS 1601, Fuels and Fission Products: This subelement encompasses the effort necessary to develop economic fuel manufacturing processes, to qualify the fuel for reactor use, and to quantify the fission product release that can occur. Work is organized in terms of fuel process development, fuel materials, and fission product performance verification. Fission product retention within the fuel coatings is the key to achieving passive safety in the MHTGR, and additional fuel performance studies are being performed to demonstrate and validate that the expected performance is achieved. Fuel fabrication process development is being done to validate that economic fuel fabrication processes will produce quality coatings. Fission product transport studies are being performed to validate the retention of fission products within the reactor system during all postulated conditions. ORNL is primarily involved in fuel material performance studies and fission product behavior validation.

WBS 1602, Graphite: This subelement covers the work required to qualify graphite materials for use in fuel elements and graphite internal structures. The emphasis of the graphite technology development effort is to obtain physical and mechanical property data under design and licensing conditions in order to provide a basis for reliable component design. Important determinations are irradiation creep effects, oxidation characteristics, fracture mechanics studies, and statistical measurements of graphite properties.

WBS 1603, Materials: The primary effort of this subelement is to obtain metallic-structural-materials data, assemble the structural materials data to a form readily usable by the design team, and support code development activities. This work is pertinent to materials engineering, reactor components, and heat-transport materials; overall, it is most important to reliable and economic component design. Key measurements involve fracture mechanics properties, mechanical properties, radiation effects, environmental and corrosion effects, and weldment properties.

WBS 1604, Safety Technology: This WBS includes specific safety work being carried out under international cooperation agreements. While the coordination of the overall cooperation between the United States and FRG is carried out under WBS 1400, the specific work done with FRG under the Arbeitsgemeinschaft Versuchs Reaktor (AVR) Test Program is under WBS 1604. Other specific work in this WBS covers the general MHTGR safety cooperation activities between the United States and FRG.

WBS 1900, Technology Program Management: Work includes the activities required to manage the HTGR technology program and its planning, coordinate technology development with other program participants, develop quality assurance plans, and review and evaluate technology reports. Because much of the ORNL effort is in base technology development, the overall management of the ORNL HTGR Program is also performed under this WBS. This management function includes overall coordination of meetings with other program participants, coordination of meetings involving international cooperation in HTGR technology development, issuance of technical progress reports, and the myriad activities involved in managing work and meeting schedules efficiently in the national HTGR Program.

WBS 5000

The work under WBS element 5000 focuses on MHTGR design. While ORNL does not carry out specific MHTGR design studies, it does provide design support in the shielding analysis area. ORNL reports shielding analysis under the following WBS subelement.

WBS 5211, Reactor System Design: This subelement covers design of the reactor core, including the design of the reactor internals. ORNL

participates in this work by providing shielding analyses in support of developing the specific core internals design. Emphasis is on shielding analysis of the lower and upper internals of the reactor vessel, with neutron fluences and radiation doses being estimated for various components and regions.

WBS 6000

The work under WBS element 6000 covers the development of high-temperature design criteria and advanced materials technology development. ORNL reports work under the following subelement.

WBS 6600, Advanced System Design Support: Under this subelement, ORNL participates in an American Society of Mechanical Engineers (ASME) task force for high-temperature design criteria and carries out experimental testing of materials for application in advanced HTGRs.

WBS 9000

The work under this WBS element covers general overall program support activities including program coordination, development and maintenance of long-term and annual program plans, development of licensing interfaces, the development and maintenance of consistent economic ground rules, review of general documents and of system design, and maintenance of the Plant Design Control Office (PDCO). ORNL participates in the above activities on a national program basis, as well as on an ORNL program basis. The WBS work elements under which work is reported are WBS 9100, 9200, 9300, and 9900, as summarized below.

WBS 9100, Program Development, Planning, and Control: This subelement covers management of the HTGR Program at the program summary level; included are the development of annual and long-term plans and associated logic networks and subnetworks. ORNL liaison activities with other program participants are carried out in this activity, and the interactions of program milestones are considered in developing work schedules. Development and management of the overall plant-level goals and requirements are also carried out under this WBS.

WBS 9200, Utility/User Requirements and Design Evaluations: Work includes the review of the utility/user requirements document as it is updated and program review activities concerning design and analysis studies.

WBS 9300, Licensing: The overall integration of licensing activities for the HTGR Program is developed under this subelement. This work includes maintenance of the licensing plan; development of criteria, methods, and standards, planning and coordination of licensing submittals to the U.S. Nuclear Regulatory Commission (NRC); support for NRC briefings; and support for revisions to the Preliminary Safety Information Document (PSID) originally sent to NRC in September 1986. ORNL carries out support activities in this WBS by reviewing documents, helping prepare responses to NRC comments, participating in the Safety and Investment Protection Study Group of the HTGR Program, and participating in NRC reviews of the MHTGR.

WBS 9900, Plant Design Control Office (PDCO): Under this subelement, PDCO carries out oversight of the MHTGR design and program activities, issues the SLPP, organizes logic networks for efficient interaction of program participants in carrying out specific work, and performs specific design reviews. ORNL participates in PDCO by assigning a person to oversee the base technology development activities and to provide expertise in the networking of overall activities.

Paul R. Kasten
Technical Director
Gas-Cooled Reactor Program

SUMMARY

1. MANAGEMENT, LICENSING, AND EVALUATION ACTIVITIES (WBS 1900, WBS 9000)

Activities were carried out for management (WBS 1900) of the Oak Ridge National Laboratory (ORNL) High-Temperature Gas-Cooled Reactor (HTGR) Program; this involved planning of activities consistent with the overall budget and overall program requirements, coordination of ORNL activities with those of other program participants, and review and evaluation of technology and overall program reports. The management function includes coordination of meetings with other program participants and coordination of meetings involving international cooperation in HTGR technology development. Technical progress reports were issued, and work and meeting schedules were efficiently maintained. Subcontracts were established for carrying out ORNL responsibilities under the national HTGR Program.

Under WBS 9100, ORNL participated in meetings of the Program Requirements Task Force, developed and updated the detailed ORNL portions of the Summary Level Program Plan (SLPP) and associated networks and work schedules, updated the pertinent portions of the SLPP and networks, and provided a public access file of the modular high-temperature gas-cooled reactor (MHTGR) conceptual design documents to permit appropriate organizations to respond to a Department of Energy (DOE) request for MHTGR plant design work.

ORNL reviewed the utility/user requirements document as it was updated and provided ad hoc review of design studies under WBS 9200.

ORNL participated in the DOE/Nuclear Regulatory Commission (NRC) licensing activities (WBS 9300) as they applied to the MHTGR. MHTGR documents such as the Regulatory Technology Development Plan (RTDP), the Preliminary Safety Information Document (PSID), and the Probabilistic Risk Assessment (PRA) were reviewed. DOE program responses were provided to NRC comments, with ORNL efforts primarily in base and safety technology areas. MHTGR licensing activities between DOE and NRC are continuing.

2. PLANT DESIGN CONTROL OFFICE (WBS 9900)

ORNL was represented in the MHTGR Plant Design Control Office (PDCO) activities through the participation of J. P. Sanders on a half-time basis and by P. R. Kasten as a member of a review team for the reactor system design. In addition to a formal design review of the reactor system, an informal review of the main circulator subsystem was held. Relative to the reactor system review, discussions were held about the basis and uncertainties of the reactor core physics methods.

As a related activity, but not part of the PDCO mission, two papers were prepared and presented at an International Atomic Energy Agency (IAEA) Specialists' Meeting on Circulators for Gas-Cooled Reactors.

Work continued in the revision of the program procedure (HP-10401) that described the application of the integrated approach to the MHTGR. Several drafts were prepared, reviewed, and then revised during the year.

Cognizance was maintained of the overall technology development efforts and the relationship of these tasks to the design needs. A revised RTDP was reviewed. Other revised base technology development plans were also reviewed.

Following the completion of the conceptual design, all major design documents and drawings were reviewed for consistency. These documents were established as the current baseline for the design.

3. INTERNATIONAL TECHNOLOGY TRANSFER (WBS 1400.01)

Progress in international cooperation with the Federal Republic of Germany (FRG) and Japan continued during the year relative to fuels, fission products, graphites, and metals for the HTGR. Of significance in the cooperation was initiation of the irradiated-microsphere gamma analyzer (IMGA) examination of FRG fuel particles from fuel spheres irradiated in the High Flux Reactor (HFR)-Petten and in the Arbeitsgemeinschaft Versuchs Reaktor (AVR). Some of these fuel spheres underwent heatup tests to 1600 and 1800°C following irradiation. Design and fabrication of a capsule (designated HRB-22) for irradiation of Japanese reference fuel was also initiated.

4. FUELS AND FISSION PRODUCT BEHAVIOR (WBS 1601)

The Fuel Materials Development and Fission Product Verification Programs at ORNL are part of a national fuel and fission product development effort being carried out in cooperation with General Atomics (GA). The ORNL fuel and fission product programs provide the planning, implementation, and execution of the base technology necessary to satisfy the technology development needs (TDNs) as expressed in the *Modular HTGR Plant Fuel/Fission Product Technology Development Plan* (F/FP TDP). In the fuels area, this effort includes the irradiation testing, postirradiation examination (PIE), and evaluation of GA-fabricated fuels to validate their quality and performance under both normal operating and accident conditions. In the fission products program, the emphasis is on the acquisition of fundamental data to adequately understand/model the transport of radiological-important fission products within the MHTGR primary circuit materials.

The base technology effort necessary to meet all of the development needs expressed in the F/FP TDP was reviewed, updated, and incorporated into Rev. 1 of this plan. The updated schedule for completion of the TDNs calls for completion of all tasks, including validation, by the end of FY 1993 at a total cost of nearly \$42M.

A critical analysis of the quality-control methods presently available to determine defective and failed fuel particle fractions was carried out.

This analysis points out that the application of current methods is strongly dependent on the quality level of the SiC layer. With the increased MHTGR fuel quality level, current methods may not be sufficient in determining the true quality of the coated particle fuel. An assessment of the feasibility of producing high-quality coated particle fuel compacts and demonstrating equally high in-service fuel performance for the MHTGR was also conducted. This assessment made extensive use of the available FRG experience; in particular, fuel manufacturing methods and quality level, in-service performance data base under normal operating conditions, and fuel performance data base under accident conditions. Overall, the FRG experience has demonstrated convincingly that pebble-bed reactor fuel elements containing high-quality coated particles have outstanding irradiation performance. Present U.S. MHTGR fuel fabrication processes have demonstrated the capability of producing high-quality fuel compacts. As a result, the present U.S. fuel shows promise of achieving a low level of fission product release, indicating feasibility that MHTGR program requirements will be met.

A detailed experimental test plan for irradiation capsule HRB-21 was prepared. This experiment will provide data for the development of the MHTGR in accordance with the F/FP TDP and is part of a cooperative effort between DOE and the Japan Atomic Energy Research Institute (JAERI). Capsule HRB-21 will contain U.S. fuel; and a companion capsule, designated HRB-22, will contain JAERI fuel. Both capsules will undergo irradiation in the High Flux Isotope Reactor, followed by detailed PIE and core conduction cooldown testing.

No significant amount of PIE was performed for the fuels program. The postirradiation gas analyzer, IMGA, and X-ray PIE facilities were for the most part only operated to maintain capability, minor maintenance, and some minor instrumentation replacement. New equipment was added to the scanning electron microscope facility, resulting in a significant improvement in the electron microscopy capabilities. Procurement of a high-temperature furnace component of the Core Conduction Cooldown Test Facility was completed.

A once-through bench-scale flow system employing tracer methods was designed and constructed for the purpose of examining the deposition and lift-off of simulated fission products from reference MHTGR structural materials. Initial operation of the system has confirmed that the techniques and procedures developed give data comparable to existing results. Initial tests are being conducted to investigate the behavior of iodine at low alloy loadings typical of that expected in an HTGR. These will be followed by tests with cesium iodide and cesium oxide to give the approximate ratio of cesium to iodine released by the fuel. Another once-through system has been designed and fabrication begun. This system will be used in fission product lift-off studies on MHTGR structural materials.

A laser-induced fluorescence system obtained from GA was assembled, aligned, and tested along with additional peripheral analysis instrumentation. Calibration of the system for measurement of low concentrations of iodine were initiated. Minor physical modifications and improved data-collection techniques are under way in order to improve iodine detection limits below 10^{-10} atm at MHTGR operating conditions.

Work is under way at the Massachusetts Institute of Technology for the design and construction of a helium circulation loop for investigating the

plate-out and lift-off behavior of fission products and the effects of dust and moisture ingress on fission product release under simulated accident conditions. Work has also been initiated at the Commissariat a L'Energies Atomique (CEA) COMEDIE Loop located in Grenoble, France. The COMEDIE Loop is an engineering-scale facility that will provide integral test data to validate design codes used to predict fission product transport under representative MHTGR conditions. An initial inspection and evaluation of the loop has been completed for the purpose of refurbishment.

5. GRAPHITE (WBS 1602)

Efforts continued on evaluation of reference and alternate graphites for MHTGR internal components and fuel blocks. Destructive (e.g., tensile, brittle-ring, and four-point bend) and nondestructive (e.g., sonic attenuation and radiography) testing was performed on Stackpole 2020, Union Carbide TS1792, Great Lakes Carbon H451, and Japanese IG11 graphites. Correlations were developed between fracture strength and bulk density and fracture strength and stress volume. The relationship of tensile strength to a nondestructive strength parameter (square root of Young's modulus divided by sonic attenuation) was also shown.

A study was initiated to establish a fracture criterion for nuclear graphite. The method of approach is to implement an appropriate probabilistic failure function into an elastic-plastic, finite-element program and to calculate the total probability of failure for a structure subjected to a nonuniform stress distribution.

New information on the creep of graphite grades H451, ATR-2E, and ASR-1RS (the latter two of German manufacture) irradiated in the HFR in Petten suggests that present models of creep need to be modified to include the effects of irradiation strengthening and softening. These data, combined with earlier results from Petten and ORNL, will permit a generalized description of creep for temperatures from 300 to 900°C at fluences approaching the useful life of nuclear graphite. Additional data for several grades of graphite irradiated at 450°C aid considerably in describing physical property effects and dimensional changes during irradiation.

A Bayesian approach to statistics has been employed to evaluate the significance of the two fractured fuel blocks observed during refueling of the Fort St. Vrain reactor.

6. METALS (WBS 1603)

The *Metals Technology Development Plan* for the MHTGR was issued in March. It was developed through the combined efforts of Combustion Engineering (CE), GA, and ORNL.

Efforts began to provide elevated-temperature allowable design stress intensities for SA533B steel, the reactor pressure vessel material chosen for the MHTGR. The first heat of material was procured and tensile and creep property determinations were initiated. A plan for the irradiation testing of this material, plus the SA508 forging steel to be used in vessel construction, was also developed. The plan provides for the determination

of the effects of fluence, neutron energy spectrum, dose rate, and irradiation temperature on fracture toughness.

Study of long-term creep, aging, and decarburization effects in 2.25Cr-1Mo steel continued. Tests were begun in the wetter design composition environment of the MHTGR. Techniques were developed for computer-controlled creep crack-growth experiments and limited testing was performed. The data obtained was analyzed to develop correlations with various fracture mechanics parameters.

7. SAFETY TECHNOLOGY (WBS 1604)

During this past year a Subprogram for Cooperation in the AVR Test Program was established and is being conducted under the US/FRG High-Temperature Reactor Umbrella Agreement. The cooperation involves ORNL and GA of the United States and AVR and Kernforschungsanlage (KFA) of the Federal Republic of Germany. The purpose of the cooperation is to obtain experimental information from the AVR relevant to the performance and safety of modular gas-cooled reactors and to help validate analytical tools used in MHTGR design and licensing by comparison of predicted results with measured data.

8. SHIELDING ANALYSIS (WBS 5211.04)

One-dimensional (1-D) parametric analyses and detailed two-dimensional (2-D) analyses were performed to provide design support to GA for the MHTGR. These analyses focused on the upper portion of the reactor internals design to assist in assuring that component fluences and radiation dose rates do not exceed acceptable levels.

Several tasks were undertaken during the year. The first of these was to determine the degree to which removal of the RSS tube shield jacket assemblies from the upper plenum would increase the neutron fluence at the reactor vessel near the base of the inner control rod housings. The second task was to assist GA in developing an effective neutron radiation shield for this region and involved determining both how much borated graphite shielding would be required and whether other shielding materials might be more effective. Borated graphite was eventually selected as the shielding material and incorporated into the design as a thick plug inside the inner control assembly housing and as a neutron shield bonnet in the region between the upper plenum thermal shield and the reactor vessel. The final task was a completely revised global 2-D shielding analysis for the upper portion of the MHTGR reactor system with the new shielding design.

9. ADVANCED SYSTEM DESIGN SUPPORT (WBS 6600)

As part of the American Society of Mechanical Engineers (ASME) Task Force on Very High Temperature Design (TF-VHTD), isochronous stress-strain curves for times to 100,000 h were developed for Alloy 617 for the temperature range 649 to 982°C. Sets of tensile stress-strain curves for several strain rates were also developed for this temperature range. Both

the isochronous stress-strain curves and the tensile stress-strain curves were submitted to the TF-VHTD as the first step in incorporating these into ASME Code Case N-47.

Work was initiated in support of structural materials for the Very High-Temperature Gas-Cooled Reactor (VHTGR). Materials of initial interest are Alloy 800H and Alloy 617, with emphasis on gas/metal interaction mechanisms and kinetics and thermal stability. Testing equipment originally used in the DOE Advanced HTGR Materials Evaluation and Development Program at General Electric was transferred to ORNL and laboratory space was made available for its installation. Additionally, refurbishment of two small recirculating loops previously used at ORNL began. Finally, experimental efforts for FYs 1988 through 1991 were outlined.

1. MANAGEMENT, LICENSING, AND EVALUATION ACTIVITIES (WBS 1900 and WBS 9000)

P. R. Kasten and J. E. Jones Jr.

1.1 HTGR PROGRAM MANAGEMENT (WBS 1900)

Overall management of the Oak Ridge National Laboratory (ORNL) High-Temperature Gas-Cooled Reactor (HTGR) Program was carried out, including interactions needed with national program participants. ORNL researchers participated in the national HTGR Program planning sessions held in March and June 1987 and coordinated the pertinent activities in preparing and revising the Summary Level Program Plan (SLPP) and the associated logic networks and subnets for both FY 1987 and FY 1988. The overall program networks and subnets were issued by the Plant Design Control Office (PDCO) and utilized in the national program planning sessions to determine the status of work and to update work efforts for each participating organization.

ORNL scientists participated in a number of meetings in which the fuel and fission product technology development plan (TDP) was reviewed by the United States Department of Energy (DOE). In February the overall bases and needs for an experimental program were summarized; in November, General Atomics (GA) presented the detailed justification for the TDP; in December ORNL presented the experimental technology program required to meet the modular high-temperature gas-cooled reactor (MHTGR) design data needs identified in the TDP.

During the year, ORNL personnel reviewed the cost-estimating ground rules and procedures issued by Gas-Cooled Reactor Associates (GCRA) for MHTGR cost evaluation and suggested several changes that were incorporated into the final document; ORNL staff also participated in several review meetings related to the above, including discussions of analytical models for evaluating nuclear fuel cycle costs. ORNL also held several meetings with GA involving fuel and fission product behavior studies and shielding analysis studies.

ORNL participated with GCRA and others in developing estimates for operation and maintenance (O&M) costs and issued a report (DOE-HTGR-87-087) entitled *Cost Estimates for Modular HTGR Operation and Maintenance*.

A proprietary information exchange agreement covering HTGR technology areas was developed between ORNL and GA.

An annual technical progress report (DOE-HTGR-87-017; ORNL-6403) covering the ORNL HTGR Program for CY 1986 was reviewed and issued in July 1987.

Meetings were held with the French Atomic Energy Commission Commissariat a L'Energies Atomique (CEA) in July relative to use of the COMEDIE fission product behavior loop located in the SILOÉ test reactor in Grenoble, France. This work is being carried out by the CEA under a subcontract from ORNL. ORNL was also involved in a review and tour of the COMEDIE test loop and facilities. The loop was found to be in excellent condition, and significant work has been carried out on refurbishment of the loop and associated facilities.

In September, GCRA staff visited ORNL to obtain information on the status of MHTGR technology development and the planned experimental work; the requirements of the SLPP for FY 1988 were also discussed, as was the shielding analyses that ORNL is performing for GA to support the design of the reactor internals. Also in September, ORNL participated in a PDCO-sponsored review of MHTGR reactor physics sensitivity analyses performed by GA. As a result of the review, several changes were incorporated into the final report.

During October, ORNL scientists met with six visitors from the People's Republic of China representing the Ministry of Nuclear Energy, the Office/Department of Industry, and State Science and Technology Commission, industry, and a university. These representatives expressed strong interest in the MHTGR and in a cooperative program involving MHTGR technology development.

ORNL staff participated in and coordinated several meetings associated with international cooperation in MHTGR technology development. In February, participation in a GCRA-coordinated meeting with United Kingdom scientists took place; potential information exchange areas were discussed, and candidate areas were identified. ORNL researchers also participated in

DOE meetings reviewing international cooperation activities and future plans; these meetings were held in March and September and reviewed U.S./Federal Republic of Germany (FRG) activities and U.S./Japan activities under bilateral agreements concerning HTGR technology development.

Relative to U.S./FRG activities, ORNL participated in and/or coordinated meetings involving the Arbeitsgemeinschaft Versuchs Reaktor (AVR) test subprogram, the safety subprogram, the fuel/fission products/graphite subprogram, and the metals/advanced materials subprogram. The AVR test subprogram document was finalized in October; ORNL scientists participated in the drafting of the safety subprogram plan which was finalized about October; a U.S./FRG review and update of the fuel/fission products/graphite subprogram was performed at ORNL in April, while similar action took place for the metals/advanced materials subprogram at ORNL in June. ORNL staff participated in a DOE/FRG meeting at Jülich, FRG, in July at which agreement was reached on the schedule for exchanging technology development plans developed in the United States and FRG and incorporating them into the subprogram plans.

In October, ORNL researchers toured Howden's circulator manufacturing facility near Glasgow, Scotland, and the Torness advanced gas-cooled reactor plant near Edinburgh, Scotland. An ORNL group also attended the International HTGR Conference at Dortmund, FRG, and a meeting of the International Atomic Energy Agency Working Group on Gas-Cooled Reactors held near Dortmund.

Concerning U.S./Japan activities in HTGR-technology development collaboration, meetings were held involving Annexes 2 and 3 under that collaboration [fuel research and development (R&D) and graphite R&D, respectively]. In July, meetings were held at ORNL with Japan Atomic Energy Research Institute (JAERI) to update cooperative efforts in fuel testing and graphite studies. The DOE/JAERI coordinating committees for Annexes 2 and 3 met at ORNL in November and July, respectively, to review progress to date; good progress on the cooperative fuel test program was reported, although there is concern about the delay in startup of the High Flux Isotope Reactor (HFIR) and its effect on test schedules. In the graphite R&D area, three test project plans were approved and included as part of Annex 3 in November.

In accordance with guidance from DOE, a series of subcontracts was initiated to carry out ORNL responsibilities under the Base Technology Plan. These are identified below.

COMEDIE Test Program: A subcontract has been in place during the past year with the CEA in France to perform tests to determine the performance of fission products in a helium loop. This loop is in a reactor at the Center D'Etudes Nucleaires at Grenoble, France. The contract is being extended for CY 1988. Initial objectives are to make specific modifications in the loop in preparation for the fission product experiments.

A subcontract has also been placed with Mr. T. Beresovski. He currently resides in France and serves on a part-time basis as on-site inspector for ORNL to ensure that the quality assurance requirements of the MHTGR program are met at the Grenoble site.

Combustion Engineering, Inc.: A subcontract is being placed with Combustion Engineering, Inc., (CE) to provide continued support of the technology program necessary for the design of the reactor vessel and steam generator of the MHTGR. With respect to the reactor vessel material, CE will provide review and input on the elevated-temperature service and irradiation effects test programs, review the test data, and interface with the American Society of Mechanical Engineers (ASME) Code Committee.

For the steam generator, CE will define an overall design support plan that will integrate data needs, costs, and schedules with the preliminary design program. CE will establish requirements for an air flow test of the steam generator bundle, prepare test specifications and schedule, and initiate the design of the test model. They will also establish requirements for the tube wear protection device test and for the shroud seal test.

Stone & Webster Engineering Corporation: A subcontract is being established with Stone & Webster Engineering Corporation (SWEC) to provide architect-engineering input to the DOE/Nuclear Regulatory Commission (NRC) licensing interactions, to respond to additional NRC questions on the Preliminary Safety Information Document (PSID), and to issue the revised PSID in response to NRC comments.

General Atomics: A subcontract is being placed with GA to perform specific tasks within the MHTGR program relating to technology requirements

described in the fuel/fission products and graphite TDPs. These tasks will address the technology development needs in the areas of fuel process development, fuel materials, qualifications, fission product transport, and graphite characterization. In addition, analyses in support of the U.S./FRG AVR testing program are planned.

Massachusetts Institute of Technology: A subcontract is in place for constructing and operating a loop for studying fission product behavior; the current effort is focused on the design of the gas circulator. The objective of the design is a circulator capable of operating at the same temperature as the loop to prevent preferential deposition of the circulating fission products in the circulator. The circulator will operate over pressures of 0.1 to 8 MPa, temperatures of 200 to 750°C, flow rates of 5 to 20 L/s, and pressure drops of 3 to 37 kPa at 500°C. The design is expected to give a shear ratio of 12 for liftoff studies at 500°C and 8 MPa.

Georgia Institute of Technology: A subcontract is in place for examining the current theories of dust formation from spallation of oxide layers on metal alloys and the behavior of dust in flowing gas streams. Experimental work and modeling studies will be proposed in these areas for future years.

Dust formation from oxide layers depends on many factors including the alloy, the interaction of gas impurities (O_2 , H_2O , etc.) in the formation of the oxide films, mechanical disturbance of the films (e.g., through erosion by gas impingement), and thermal stresses induced by thermal cycling. The characteristics of the dust formed are influenced by the conditions of film formation and spallation and by the mechanical comminution of the particles by impingement on surfaces.

Theories of dust behavior can be largely empirical, such as the shear-ratio model currently used in the HTGR Program, or can be based largely on modeling of the interactions of dust particles with the flowing gas stream and the alloy surface; the prime example of the latter model is the Reeks, Reed, and Hall model developed by the Central Electricity Generating Board (CEGB). Both types of models have advantages and disadvantages. The empirical models do not require the determination of the interaction

parameters, which can be fairly difficult, and may give, in a straight-forward calculation, predictions that are adequate for design and safety analyses. The models do not, however, unambiguously identify the important parameters and are unpredictable for extrapolation beyond the range of the experimental data. The more phenomenological models should give a better understanding of the important factors in dust behavior and should be capable of extrapolation to new operating conditions. These models do, however, require a significant quantity of data to evaluate the parameters describing the interactions of the particles with the alloy surface.

1.2 PROGRAM DEVELOPMENT, PLANNING, AND CONTROL (WBS 9100)

The effort in this area covers management of the national HTGR program at the program summary level. ORNL supports this activity through participation in the Program Requirements Management (PRM) Task Force. Meetings of the PRM Task Force were held at SWEC in March, and at CE in June. The SLPP and associated networks and subnets were reviewed, and work elements were modified in accordance with national program priorities consistent with resources. Detailed efforts were conducted in preparing the ORNL portion of the SLPP and the associated network that provide information on interactions between program participants and the schedule of work. By this process, the ORNL SLPP and networks were revised on a quarterly basis for the first half of the calendar year. The networks were updated on a monthly basis for the first half of the calendar year.

The process of awarding design work for FY 1988 on a Request for Proposal (RFP) basis led to a delay in maintaining the SLPP and networks in the second half of the calendar year. During that time, MHTGR conceptual design documents pertinent to responding to the above RFP were collected at ORNL and made available in a public access file at the ORNL Library from August until the end of the calendar year, at which time the RFP process was closed.

During September, ORNL researchers developed in detail that portion of the SLPP that lists the ORNL efforts for FY 1988; the format was designated by DOE and distributed by GCRA, and it corresponds to the format in this progress report. The ORNL portion of the SLPP was sent to GCRA for incorporation into the final document.

During the year several changes were made to the ORNL portion of the SLPP and the associate networks, as a result of various factors. Change proposals were submitted and approved by DOE in the shielding analysis and technology transfer areas, corresponding to changing the description of milestones and milestone schedules in accordance with program needs.

1.3 UTILITY/USER REQUIREMENTS AND DESIGN EVALUATION (WBS 9200)

Updated portions of the utility/user requirements document were received, reviewed, and found to be acceptable.

From February to May, ORNL scientists participated in several PDCO-sponsored meetings to review the core design, the control systems, and the internals design of the MHTGR. Concerns raised about the uncertainties in reactor physics parameters and measurements are to be addressed during preliminary design of the reactor systems.

ORNL staff also participated in a utility-GCRA review of the MHTGR conceptual design held in November.

1.4 LICENSING (WBS 9300)

Cognizance of licensing activities being carried out by the national MHTGR Program was maintained, and specific participation occurred in selected areas. ORNL scientists reviewed the RTDP as it was developed for submission to the NRC, and the resulting comments were incorporated into the final report. ORNL staff also reviewed the PRA report before submission to NRC early in the year.

ORNL researchers participated in the DOE/NRC meetings concerning MHTGR licensing evaluations and involving NRC review of the PSID submitted in late 1986. In March, Chap. 4 of the PSID was reviewed, covering the reactor design; in April, Chap. 5 of the PSID, covering heat transport systems, was reviewed; in May, Chaps. 6, 9, 10, 11, 12, and 13 of the PSID were reviewed. In June, the RTDP and Chaps. 15 and 17 of the PSID were reviewed. In July, the PRA report, the Emergency Planning Basis Events developed for the MHTGR, and the role of the operator in MHTGR operations were reviewed by the NRC.

These NRC reviews led to comments sent to DOE for response. ORNL staff assisted in the preparation of program responses in selected areas (primarily base technology and safety technology) and reviewed the overall responses.

ORNL also participated in related MHTGR Program activities. In March, the Safety Investment Protection Task Group met and concluded that the term "radionuclide protection function" was more appropriate than use of "safety related" or "important to safety." In June, ORNL personnel assisted in the review of the NRC light-water reactor (LWR) generic safety issues report and the determination of how that report applied to MHTGRs. In November, responses to NRC comments on remaining safety issues were prepared, with emphasis on fuel performance information. The MHTGR licensing interactions between DOE and NRC are continuing, with NRC scheduled to issue a Safety Evaluation Report on the MHTGR in CY 1988.

2. PLANT DESIGN CONTROL OFFICE (WBS 9900)

J. P. Sanders

2.1 INTRODUCTION

During CY 1987, a Plant Design Control Office (PDCO) was maintained at the site of General Atomics (GA) in San Diego, California. Among other activities, it was the responsibility of the PDCO to coordinate the efforts among the various participants in the national HTGR program. These participants were represented in the activities of the PDCO through the assignment of their staff members to full- or part-time responsibilities in the PDCO organization.

ORNL was represented in the PDCO activities through the participation of J. P. Sanders on a half-time basis and by the participation of P. R. Kasten as a member of a review team for the reactor system design and as a member of other special working groups.

As a member of the PDCO organization, it was the responsibility of J. P. Sanders to be cognizant of the networks and activities of the various base technology programs. In particular, the base technology activities were to be continually reviewed relative to the needs of the design efforts. In addition, Sanders coordinated the review of various systems in the design of the reactor.

2.2 REVIEW OF THE MHTGR MAIN CIRCULATOR

An informal review of the MHTGR Main Circulator Subsystem was held by a PDCO review team together with the designers at GA. In particular, the functional requirements of the system were assessed relative to the conceptual design.

The important aspects of the design review included the drive motor, the impeller, and the bearings. At the time of review, a design modification was being considered to change the impeller design so that the axial load due to the reaction with the gas on the vertical-shaft machine would

act counter to the gravitational load. This modification significantly reduced the loads on the thrust bearings.

Further detail specification of the electric motor was reserved as a subcontract for a vendor to be identified in the future.

Activities related to the circulators, but now part of the ongoing review of the circulator design by PDCO, included participation in an International Atomic Energy Agency (IAEA) specialists' meeting on gas-cooled reactor coolant circulator and blower technology. This meeting was held at the GA facilities in San Diego during the period from November 30 to December 2, 1987.

Two papers were prepared and presented at this meeting. One paper¹ outlined the operating experience with gas-bearing circulators in a high-pressure helium loop at ORNL; the second paper² described an unanticipated problem due to erosion of the impeller housing by the high-velocity gas. The gas, which was helium with controlled concentrations of impurities, removed the protective oxide layer from the interior surface of the housing. This housing was fabricated from AISI 410 stainless steel, and it evidently had not been heat treated in a sufficiently oxidizing atmosphere to produce an adherent oxide layer. After the initial layer had been removed, subsequent layers were formed and stripped away. The results were V-shaped grooves that are characteristic of gas erosion in granular beds.

2.3 REVIEW OF THE REACTOR SYSTEM

Early in the year, an informal review of the reactor system for the MHTGR was held by PDCO with the designers at GA. This review included the reactor core, the reactor internals, and the neutron controls subsystems. An internal report presented the information, agreements, and recommendations that resulted.

Additional detailed information about the design of this system was given at the Department of Energy (DOE)/Nuclear Regulatory Commission (NRC) meeting on the reactor design, as presented in Chapter 4 of the Plant Safety Information Document (PSID). This meeting was held in Washington on March 18-19, 1987.

A formal review of the MHTGR reactor system was held by PDCO with the designers at GA in April. A preliminary letter presented a draft version of the major conclusions of the review team. The individual comments of the members of the review team, which included P. R. Kasten from ORNL, were tabulated in a data base management computer program that permitted the comments to be collected by subsystem, by reviewer, or by assigned keywords. Printed output from this correlation was circulated to the review team members for correction or comment. The final report was distributed by letter (PDCO-330-87) in July.

2.4 REVIEW OF REACTOR PHYSICS COMPUTATIONAL PROCEDURES

On June 18, 1987, in conjunction with an NRC review meeting in Bethesda, Maryland, a meeting was held to discuss concerns expressed about the basis for the nuclear physics calculations for the MHTGR core. A letter (PDCO-307-87) was issued that summarized the conclusions and proposed actions developed at that meeting. The major concerns involved documentation requirements of the reactor physics methods, the magnitude of uncertainties in the calculated reactor response characteristics, and the data base used as a source of the microscopic cross sections of the nuclides.

Subsequent correspondence and documentation that addressed the uncertainties in the reactor physics calculations were reviewed.

2.5 PROCEDURE FOR APPLICATION OF THE INTEGRATED APPROACH

A revised draft of the program procedure (HP-10401) that describes the application of the integrated approach to the MHTGR was first completed in April. The remainder of the year was for the review and subsequent revision of the procedure. In August, a revision that described its application to the design task plus its implementation during construction, operation, maintenance, and decommissioning was provided. In September, the application of the integrated approach as the basis for licensing and probabilistic risk assessment was added.

In October, a review of this procedure was held in Washington, and the need for further revision was indicated. During the remainder of the year, additional comments were provided by members of GCRA and PDCO.

2.6 REVIEW OF TECHNOLOGY DEVELOPMENT PLANS

In the continuing task of reviewing the technology development plans, the revised Regulatory Technology Development Plan (RTDP) was reviewed in May in conjunction with revised system design descriptions. Revised sections of "The Fuels and Fission Product Technology Development Plan," "The Metals Technology Development Plan," and "The Graphite Technology Development Plan" were also reviewed relative to the support needed for the design task.

2.7 BASELINE FOR THE CONCEPTUAL DESIGN

Beginning in August, documents that described assigned systems within the conceptual design were reviewed for internal consistency and consistency among documents. This review by PDCO established the "baseline" description of the MHTGR conceptual design.

These review activities continued through September and October. By November, the review of the higher level documents had been completed.

2.8 LOGIC NETWORKS

Draft program logic networks were composed and reviewed for all of the base technology tasks that are currently included in the MHTGR program.

2.9 REFERENCES

1. J. P. Sanders, Uri Gat, and H. C. Young, "Operating Experience with Gas-Bearing Circulators in a High-Pressure Helium Loop," IAEA Specialists' Meeting on Gas-Cooled Reactor Coolant Circulator and Blower Technology, San Diego, California, November 30-December 2, 1987.

2. J. P. Sanders, R. L. Heestand, and H. C. Young, "Gas Erosion of Impeller Housing in the Operation of a High-Temperature, High-Pressure Helium Circulator," IAEA Specialists' Meeting on Gas-Cooled Reactor Coolant

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Circulator and Blower Technology, San Diego, California,
November 30-December 2, 1987.

3. INTERNATIONAL TECHNOLOGY TRANSFER (WBS 1400.01)

P. L. Rittenhouse

3.1 INTRODUCTION - P. L. Rittenhouse

Significant progress in international cooperation with the Federal Republic of Germany (FRG) and Japan continued during the year on many aspects of high-temperature gas-cooled reactor (HTGR) technology. The United States (US)/FRG umbrella agreement covering HTGR technology development has been in place for about 12 years and much mutually beneficial cooperation has taken place during this period. Cooperation between the United States and Japan is much more recent, having been initiated in 1985.

3.1.1 US/FRG Cooperation

Ongoing US/FRG cooperation involves (1) a subprogram on structural materials (both metals and ceramics) and (2) a subprogram covering fuel, fission products, and core and structural graphites. These subprograms have been restructured over the past few years to address modular high-temperature gas-cooled reactor (MHTGR) technology development needs. Areas of cooperation include fuel performance modeling, hydrolysis behavior of fuel particles, fission product retention and transport during postulated accidents, irradiation and oxidation behavior of graphites, mechanical properties and corrosion behavior of reactor materials, and high temperature structural design criteria.

3.1.2 US/Japan Cooperation

This cooperation is being carried out under specific "annexes" to the general cooperative agreement. Oak Ridge National Laboratory (ORNL) is currently involved in Annexes 2, 3, and 4. Annex 2 involves a joint fuel test program performed in the U.S. and has been in progress over the past 2 years. There are three active areas of cooperation on graphite (Annex 3) and four additional tasks under consideration. Four areas of cooperation on metals (Annex 4) have been proposed but have not yet been implemented.

3.2 STRUCTURAL MATERIALS - P. L. Rittenhouse

3.2.1 US/FRG Cooperation on Metals

An annually revised Subprogram Plan for Cooperative Work on High-Temperature Reactor (HTR) Materials provides the framework for performance of cooperative work. Each work task is defined by a Project Work Statement (PWS) describing objectives, activities, schedules, and milestones. Most of the 13 PWSs which have been implemented (6 complete and 7 still active) involve high-temperature mechanical properties and gas-metal interaction behavior.

Since both U.S. and FRG programs on HTGR structural materials are in the process of change, the detailed plans for both countries' new material programs were exchanged. This will provide guidance for the development of new cooperative activities of mutual interest and benefit.

3.2.2 US/Japan Cooperation on Metals

Proposals for collaborative efforts in four areas (creep-fatigue behavior, irradiation and postexposure testing of alloy 800H, gas-metal interactions, and crack propagation) were exchanged and evaluated. Implementation of these activities remains under discussion.

3.3 FUELS AND FISSION PRODUCTS - M. J. Kania

3.3.1 Fuels and Fission Products Cooperation with FRG - M. J. Kania and C. A. Baldwin

Irradiated German fuel from Set 4 under PWS FD-20, scheduled for the irradiated microsphere gamma analyzer (IMGA) examination, arrived at ORNL during February 1987. The irradiated fuels were transferred to the High Radiation Level Examination Laboratory (HRLEL) where the postirradiation examinations (PIE) will take place. Set 4 fuels contained the following:

1. Unbonded particles deconsolidated from experiments:

- HFR-K3/Sphere 1 (16,000 particles),
- HFR-K3/Sphere 3 (16,000 particles), and
- AVR Sphere 76/20 (16,000 particles).

2. Ten intact irradiated AVR spherical fuel elements:

- LEU fueled elements (77/13, 77/14, 77/18, 77/24, 77/25); and
- HEU fueled elements (76/29, 76/31, 77/3, 77/10, 77/20).

The unbonded particles listed in item 1 are the reference FRG low-enriched uranium (LEU) UO_2 TRISO-coated particles that were fabricated to very high quality standards. They represent our first attempt to examine such high quality fuels with the IMGA and postirradiation gas analyzer (PGA) systems. Fuel from experiment HFR-K3 was irradiated to beyond HTR design operating conditions and then subjected to core heatup simulation tests in the hot-cells at Kernforschungsanlage (KFA), Jülich, FRG. Sphere 1 was subjected to 1600°C for 500 h, and sphere 3 was subjected to 1800°C for 100 h. The unbonded particles from Arbeitsgemeinschaft Versuchs Reaktor (AVR) sphere 76/20 were not subjected to a PIE heatup test.

The spherical elements in item 2 will be held in storage at ORNL until the facilities and capabilities are in hand to perform core heatup simulation tests (see Sect. 4.1.3.3). The latter elements will be examined under PWS FD-11.

3.3.2 Fuels and Fission Products Cooperation with Japan - M. J. Kania, B. W. Patton, and K. R. Thoms

Pursuant to Article III-2 of the Implementing Arrangement between the United States Department of Energy (DOE) and the Japan Atomic Energy Research Institute (JAERI) on cooperation in research and development in the area of HTGRs, the parties established a collaborative program for the purpose of conducting a fuel performance test. This program will evaluate the performance of reference fuel compacts containing coated particle fuels that have been fabricated in accordance with the specifications for the U.S. MHTGR concept, and the Japanese High Temperature Engineering Test Reactor (HTTR) concept.

Two separate irradiation capsules, designated HRB-21 and HRB-22, will be employed. Capsule HRB-21 (see Sect. 4.1.2.1) will contain only DOE reference fuel compacts, while capsule HRB-22 will contain only JAERI reference compacts. The DOE fuel will be fabricated by General Atomics (GA) according to MHTGR fuel product specifications, and the JAERI fuel will be fabricated in Japan by Nuclear Fuel Industries in accordance

with HTTR specifications. The experimental capsules will be designed, fabricated, and assembled at ORNL. The irradiation will be performed in the High Flux Isotope Reactor (HFIR) located at ORNL. PIE will be performed both at GA and ORNL, and the core heatup simulation testing (CHST) will take place at ORNL. The analysis of test results would be performed jointly by ORNL, GA, and JAERI, with the final documentation being issued by ORNL and JAERI.

An experimental plan was prepared which describes in detail the capsule design, fuel compact design and loadings, irradiation and surveillance requirements, PIE and CHST test matrices, documentation requirements, and the projected schedule.

3.3.2.1 Objectives

The objectives of this collaborative program are to determine the following for both the DOE and JAERI reference HTGR fuel compacts:

1. the capacity of LEU-coated particle fuels to meet their respective performance goals under normal irradiation conditions, and
2. the capability of irradiated-coated particle fuels to meet specified failure fraction limits under simulated core heatup tests.

Performance goals for the DOE MHTGR fuel specify that the failure fraction on a core average basis for end-of-life (EOL) conditions be $\leq 5.0 \times 10^{-5}$ at the 50% confidence level. This value represents the total expected particle failure level from all sources under normal operating conditions. In the event of a core heatup accident, an additional incremental fraction of $\leq 1.5 \times 10^{-4}$ at the 50% confidence level may be tolerated. At the 95% confidence level, these values are 2.0×10^{-4} and 6.0×10^{-4} , respectively.

For the JAERI HTTR design, fuel performance requirements specify that EOL failure fraction under maximum burnup of 3.8% fissions per initial metal atoms (FIMA) be less than beginning-of-life (BOL) failure fraction plus BOL SiC defective fraction, for which the maximum value is 1.65×10^{-3} with 95% confidence. The failure fraction under the accident conditions is less than 1×10^{-2} with 95% confidence limit.

3.3.2.2 Description of experiment

To accomplish the objectives listed above for both JAERI and DOE fuel concepts, two separate irradiation capsules will be utilized. Specific test requirements that differ significantly are outlined below for the DOE capsule, HRB-21, and the JAERI capsule, HRB-22.

<u>Test requirements</u>	<u>DOE (HRB-21)</u>	<u>JAERI (HRB-22)</u>
1. Operating temperature (volumetric-averaged fuel)	975°C	1250–1300°C
2. Fissile fuel burnup (% FIMA)	≤26	≤10
3. Fertile fuel burnup (% FIMA)	≤4	–
4. Irradiation period	6 HFIR cycles	4 HFIR cycles
5. Fuel compact type	MHTGR reference (cylindrical, 12.4 mm diam × 50 mm long)	HTTR reference (annular, 8 mm ID × 24 mm OD × 39 mm long)
6. Fuel compacts/capsule	24	12
7. Graphite bodies	8, with 3 fuel compacts each	1, with 12 fuel compacts each
8. Graphite type	H451	IG-110

3.3.2.3 Capsule design

The generic capsule design is a doubly contained, single-purged cell with an available cylindrical test volume of about 660 cm³ and with dimensions 40 mm diam × 500 mm long. This test volume is sufficiently large to contain either DOE or JAERI reference fuel compact test specimens surrounded by either a graphite body as in HRB-21, or a single graphite sleeve as in HRB-22. The fuel compacts in capsule HRB-22 will be composed of TRISO-coated UO₂ particles imbedded in graphite matrix and loaded into a JAERI IG-110 graphite sleeve as shown in Fig. 3.1. For both capsules, the

Fig. 3.1. Horizontal section through capsule HRB-22.

fuel graphite components are contained within two Inconel 718 containment tubes designed to independently withstand 6.9 MPa (1000 psi) inside or outside as shown in Fig. 3.2. During HFIR operation, pure helium at 5.5 MPa (800 psi) flows between the primary and secondary Inconel containments. A gas mixture of helium and neon flows between the primary Inconel containment and the graphite body surface. By varying this gas mixture within the primary containment, the radial heat transfer to the surrounding HFIR cooling water system is controlled. Fuel compact operating temperatures are regulated by this method.

3.3.2.4 JAERI fuel compacts, HRB-22

The reference fuel compact for the JAERI HTGR is a molded compact fabricated using only overcoated particles under high pressure. Prior to the compact fabrication step, JAERI TRISO-coated UO_2 particles are overcoated with a 200- μm -thick layer of matrix material made up of natural

Fig. 3.2. Vertical section through capsule HRB-22.

graphite and petroleum graphite powders with the addition of a phenol resin binder. The overcoated particles are then placed into a steel mold at about 100°C and formed into a green annular compact under a pressure of 10 MPa (1450 psi). The particle volume fraction in each compact is maintained at 30 vol % by the addition of inert coated particles when needed. The green compacts are heated in a two-step process to carbonize binder, remove volatiles, and stabilize compact dimensions. Final nominal dimensions of the JAERI compacts for HRB-22 are 24 mm OD × 8 mm ID × 40 mm long. In capsule 22, a total of 12 compacts will be loaded in tandem inside an IG-110 graphite sleeve as shown in Fig. 3.2.

The 12 JAERI fuel compacts were fabricated from a single batch of TRISO-coated UO₂ particles, and the characteristics of these compacts are provided in Table 3.1. The total particle volume fraction was maintained at 29.8 vol % with 23.9 vol % made up from an inert SiC TRISO-coated inert particle batch and 5.9 vol % of the UO₂ particle batch. The mean ²³⁵U content per compact was 0.098 g, of which 4.046% was enriched. The density of each compact was 1.94 Mg/m³ with a matrix density of 1.69 Mg/m³. Quality control performed on five compacts indicated a free uranium

Table 3.1. Characteristics of JAERI compacts in capsule HRB-22

Position HRB-22	JAERI compact	OD (mm)	ID (mm)	Length (mm)	Weight (g)	Total uranium content (g)
(top)						
1	JC-11	24.03	8.06	39.96	31.308	0.097
2	JC-02	24.04	8.06	39.98	31.316	0.098
3	JC-13	24.04	8.06	39.99	31.334	0.097
4	JC-14	24.04	8.07	39.96	31.324	0.099
5	JC-18	24.05	8.04	39.96	31.294	0.099
6	JC-04	24.05	8.06	40.04	31.337	0.098
7	JC-15	24.05	8.04	39.93	31.271	0.098
8	JC-12	24.04	8.06	39.98	31.328	0.097
9	JC-07	24.04	8.06	39.98	31.324	0.098
10	JC-09	24.04	8.07	40.01	31.328	0.097
11	JC-16	24.03	8.06	39.98	31.285	0.097
12	JC-19	24.02	8.06	39.95	31.279	0.098
(bottom)						

fraction that ranged from $5.6\text{E-}5$ to $1.4\text{E-}4$. The SiC defect fraction obtained by a burn/leach procedure yielded values ranging from $2.2\text{E-}4$ to $1.3\text{E-}3$.

3.3.2.5 Irradiation in HFIR

Irradiation of capsules HRB-21 and HRB-22 will be performed in the HFIR located at ORNL. The capsules will be inserted into two recently updated removable beryllium (RB) positions. Capsule HRB-21 will be irradiated for 6 HFIR cycles (about 126 days) and the JAERI capsule HRB-22 will be irradiated for 4 HFIR cycles (about 84 days). Peak burnup for the JAERI UO_2 fuel will not exceed 10% FIMA, while for the U.S. UCO and ThO fuel, the maximum burnups will be about 24% and 3% FIMA, respectively.

The target fuel volumetric mean temperature for capsule HRB-21 is 975°C , and the peak fuel temperature for capsule HRB-22 is 1300°C . In both capsules, the graphite fuel body or graphite sleeve, radial gaps have been specified to maintain this temperature as uniformly as possible over the length of the capsule. Fuel compact temperatures will be inferred from the graphite thermocouple temperature data and the preirradiation design thermal analysis. In-reactor temperature control will be maintained by adjusting the neon and helium concentrations in the sweep gas. After irradiation and completion of the fuel/graphite metrology, a detailed thermal analysis will be performed to obtain in-reactor fuel compact temperatures.

3.3.2.6 Postirradiation examination (PIE)

After completion of their scheduled 4- and 6-cycle irradiation in HFIR, capsules HRB-22 and HRB-21, respectively, will be removed from the reactor and allowed to cool in the pool-side facility for 1 to 3 weeks. The capsule will then be loaded into a special transportation cask and moved to the HRLEL for disassembly and PIE.

The capsule disassembly and PIE will be performed according to a well-defined plan that will provide detailed disassembly procedures for both capsules, a PIE test matrix, and PIE procedures to be used for examination. A tentative PIE test matrix prepared for the fuel compact specimens in each of capsules HRB-21 and HRB-22 is provided in Table 3.2 and will be updated as needed in the PIE plan.

Table 3.2. Tentative PIE test matrix for JAERI annular fuel and DOE cylindrical fuel compacts irradiated in capsules HRB-22 and HRB-21, respectively

Compact ^a number	CHST	TRIGA activation	Leach ^b	Ceramography/ Microprobe	Deconsolidation/ IMGA
<i>JAERI Annular Compacts, HRB-22</i>					
1	X	X			
2		X			
3	X	X			
4		X			
5			X	X	X
6		X			
7			X	X	X
8		X			
9	X	X			
10			X	X	X
11	X	X			
12			X	X	X
<i>DOE Cylindrical Compacts, HRB-21</i>					
1	X	X			
2	X	X			
3	X	X			
4	X	X			
5		X	X	X	X
6		X	X	X	X
7		X	X	X	X
8		X	X	X	X

^aIn capsule HRB-22, number refers to compact location in graphite sleeve from the top of capsule. In capsule HRB-21, number refers to the position of graphite fuel body.

^bJAERI fuel compacts must be deconsolidated prior to leaching step.

3.4 GRAPHITE - W. P. Eatherly

3.4.1 US/FRG Cooperation

Four areas of the collaborative graphite program remain active:

1. tensile creep experiment on H451 in the Petten reactor,
2. comparative oxidation tests on ASR-1RG graphite at ORNL,

3. creep experiment on several graphites involving reversal of stress from compression to tension, and
4. lifetime experiments in HFIR on several FRG graphites derived from new raw materials.

Both of the latter two involve use of the HFIR and thus remain on hold until the reactor is permitted to continue functioning after safety review. The tensile creep experiments at Petten are complete and all data on both the German and U.S. graphites have been supplied to ORNL. The data are currently being analyzed in terms of statistically valid fits to the dimensional and secondary creep data.

3.4.2 US/JAERI Cooperation

US/Japan collaborative work on graphite is defined under Annex 3 of the enabling agreement. At the present time at least three technical areas have been identified under which both data and experiments will be exchanged. These are:

1. Fracture Mechanics: Japanese specimens have been prepared to permit comparison of the Japanese slit disc specimen design with the U.S. compact cylinder type. The work will be performed in the U.S.
2. Multiaxial Stress Failure Criteria: Both parties are examining their respective needs in terms of graphite types and equipment available or to be constructed.
3. Creep: Currently a 400°C experiment is under way in the JAERI Materials Test Reactor. The U.S. has supplied specimens of H451 for test.

It is anticipated that further areas of activity will be defined in the next year.

3.4.3 US/UK Cooperation

Preliminary conversations have been held between technical personnel at GA and ORNL on the one hand, and the Central Electricity Generating Board (CEGB) and United Kingdom Atomic Energy Authority (UKAEA) on the other. Sufficient commonalty of technical interests was identified to indicate a

mutually profitable cooperative exchange can exist. Negotiations at the legal level have been initiated.

4. FUELS AND FISSION PRODUCT BEHAVIOR (WBS 1601)

M. J. Kania and J. C. Mailen

The Modular High-Temperature Gas-Cooled Reactor (MHTGR) Fuel Materials Development Program and the Fission Product Performance Verification Program at Oak Ridge National Laboratory (ORNL) are part of a national fuel and fission product development effort being carried out in cooperation with General Atomics (GA). This chapter reports on the activities conducted under the Fuel Materials Development Program (Sect. 4.1) and the Fission Product Performance Verification Program (Sect. 4.2).

4.1 FUEL MATERIALS DEVELOPMENT (1601.02) - M. J. Kania

The Fuel Materials Development Program is directed toward satisfying the technology development needs (TDNs) as expressed in the *Modular HTGR Plant Fuel/Fission Product Technology Development Plan*¹ (F/FP TDP). Qualitatively, the needs for the Fuel Materials Program are

1. to reduce the uncertainty in the models and physical property data used to predict fission product transport in the MHTGR core under normal and off-normal operating conditions, and
2. to validate fuel performance models used to predict coated particle and fuel compact integrity under normal and off-normal operating conditions.

As lead laboratory for MHTGR Technology Development, ORNL is responsible for planning, implementing, and executing the technology necessary to meet the designer's needs.

The ORNL effort in the United States Department of Energy (DOE) national MHTGR Program is conducted in close cooperation with GA, whose responsibilities include interfacing with core designers and developing the technology for fuel manufacture, process development, and quality control. In addition, the ORNL fuels program maintains cooperative tasks with two international programs: (1) the German Research Center at Jülich

[Kernforschungsanlage (KFA)] and (2) the Japan Atomic Energy Research Institute (JAERI) - Tokai Research Establishment. These cooperative efforts are part of formal agreements between the United States and the governments of the Federal Republic of Germany (FRG) and Japan in the area of high-temperature reactor development. The ORNL efforts directed toward these two international programs in the fuel materials area are presented in Chap. 3.

4.1.1 Technology Planning, Assessments, and Analysis

4.1.1.1 Fuel/fission product technology development plan - M. J. Kania and J. C. Mailen

An extensive review and update of Chap. 5 of the F/FP TDP² was performed for incorporation into Rev. 1 of the F/FP TDP.¹ The updated TDNs and specified data parameters/service conditions, prepared and presented in Chap. 4 of the F/FP TDP by GA, served as the basis for the update. Chapter 5 of this plan presents a detailed summary of the experimental tasks, which conform to one or more of the 27 specific TDNs needed to validate certain assumptions on fission product transport, fuel performance, and process development generated in the MHTGR functional analysis. Specific areas addressed include

1. experimental tasks necessary to formulate fission product transport and behavior correlations and to provide validation of the design methodology used for fission product transport;
2. experimental tasks needed to assess fuel-particle performance correlations and to validate fuel-element and coated-particle performance models;
3. experimental tasks required to develop, produce, and validate production-scale processes for fabricating high-quality, coated-particle fuel compacts;
4. cost schedule for completion of experimental and validation tasks; and
5. time schedule for completion of experimental and validation tasks.

The planned technology approach outlined to meet the needs in the areas of fission product transport and fuel performance specifies a

significant irradiation program to ensure that a sufficient quantity of irradiated fuel specimens are provided. This program includes a total of 12 experimental irradiation capsules. The experimental capsule designation, nominal fuel operating temperature, and applicable TDN number(s) are shown in Table 4.1. These capsules are multiparameter capsules and, in addition to temperature, cover a full range in required fuel burnups and accumulated fast-neutron fluences.

Table 4.1. Capsule designation, nominal fuel temperature, and applicable technology development need for irradiation program planned to meet fission product transport and fuel performance technology development needs

Capsule designation	Applicable technology development need							
	9	10/11	12	14	15	16	17	19
HFR-B1 (950-1250°C)	X	X						X
HRB-19 (1250°C)	X			X	X	X	X	
HRB-20 (1180°C)	X			X	X	X	X	
HRB-21 (975°C)	X			X	X	X	X	
HRB-23 (780°C)	X			X	X	X	X	
HRB-24 (700°C)	X			X	X	X	X	
HRB-25 (1250°C)			X					
HRB-26 (1180°C)		X			X			X
HRB-27 (975°C)	X				X			X
HRB-28 (700°C)	X	X			X			X
HRB-29 (1250°C)	X				X			X
HRB-30 (780°C)		X			X			X

The schedule for completing the TDNs calls for the completion of all tasks, including validation, by the end of FY 1993. Because of the nature of the DOE-funded program, this schedule is subject to external priority and budget constraints. The total cost of the F/FP TDP is estimated to be \$42,504K, of which \$20,954K is allocated to fission product transport

technology development, \$12,275K to fuel materials technology development, and \$9,275K to fuel process technology development.

4.1.1.2 Determination of the fraction of defective and failed fuel particles - B. F. Myers*

The determination of the fraction of defective or failed coated fuel particles is very important in assessing the quality of the particles and, consequently, the acceptability of the fuel compacts containing the particles and their irradiation performance. The methods for determining the defect fraction have been critically analyzed, and a report has been prepared based on this analysis.³ The burn-leach, intrusion, and the gaseous-acid-leach methods are addressed; for the intrusion method, both wetting and nonwetting intrusants are discussed. The burn-leach method is used routinely to determine the fraction of defective or failed SiC layers, and the gaseous-acid-leach method is used to determine the fraction of exposed kernels, as well as heavy-metal contamination in the compact. In both methods, the final step is the extraction of the kernel. These methods are described and quantitatively evaluated.

The burn-leach and the gaseous-acid-leach methods are found to be limited because they gain access to the fuel kernel through the defect of the defective or failed particle. The burn portion of the burn-leach method can aggravate the problem of access through SiC by inducing the growth of oxide layers over defects. The analysis of the intrusion method leads to a characterization of the defects in terms of a distribution of pore sizes through which the interior of the particles are accessible. This characterization is used to further evaluate the methods for determining defect fraction.

Central to this evaluation is the quantitative description of gaseous and liquid transport through pores within the 10- to 10,000-nm radius range. The mechanisms of transport discussed are (1) capillary flow in unfilled pores, (2) viscous flow of fluids, (3) molecular flow of gases, and (4) diffusional transport of fluid components.

*On assignment to ORNL from GA.

The quantitative evaluation of gas transport in the burn-leach and the gaseous-acid-leach methods indicates that only defective or failed particles having pore radii larger than several hundred nanometers are fully detected by these methods within periods of less than 50 h. A similar conclusion is reached for the liquid-leach portion of the burn-leach method. A quantitative evaluation of the intrusion method using mercury as the intrusant is ambiguous in accounting for the detection of pores of the smallest radii associated with observed cumulative defect fractions.

The main conclusions drawn from the analyses of the methods for determining defect fractions are

1. a population of defective or failed SiC layers can be characterized by a distribution of effective pore radii, and
2. currently used methods for detecting defective or failed SiC layers or exposed kernels are limited to layers or particles having pore radii larger than a minimum value.

The distributions peak in the 6- to 8-nm pore radius range, fall to lower frequencies in the 5- to 15-nm range, and have long tails extending to hundreds of nanometers. The question of the importance of the various subranges of pore radius is addressed for (1) the release of fission gas under MHTGR normal operating conditions and (2) the release of fission gas and cesium under accident-simulation conditions. The analysis of fission gas release under normal conditions in the presence of larger (>50 nm) radius pores when particle interior and exterior gas pressures are equalized shows that

1. stable fission gas release from the particles is the same as the release from the kernel; and
2. the steady-state release of radioactive gases depends on the response of the buffer to irradiation
 - a. reaching a maximum value for an undensified buffer, and
 - b. falling to low or even negligible values for a densified buffer and the associated void volume.

The steady-state fission gas release from a small (<25 nm) radius pore is negligible compared with that from a larger radius pore, but given the

distribution of pore radii, the contribution of a collection of small-radius pores to the total steady-state fission gas release cannot be neglected. Under accident-simulation conditions and with good-quality particles, the gas release may be governed by transport through small radius pores, as demonstrated with sphere 1 of the German capsule in reactor experiment R2-K13. For cesium release under these conditions, the small-radius pores may also contribute significantly to release.

The conclusion reached from the analyses of these events and of the burn-leach and gaseous-acid-leach methods of determining defect fractions is that the efficacy of these methods depends on the quality of the SiC layer or fuel particle, which is represented by (1) the cumulative defect fraction (as measured, for example, by the intrusion method) and (2) by the distribution of pore radii. As the defect fraction decreases and the distribution of pore radii narrows, the quality of the SiC layer or the fuel particle increases. However, as the quality increases, the burn-leach and gaseous leach methods become less effective and less useful, and their application in determining defect levels provides inadequate assurance of the true quality of coated fuel particles.

4.1.1.3 Feasibility of high-performance fuel for the MHTGR - M. J. Kania, P. R. Kastén, and D. T. Goodin*

The feasibility of producing high-quality coated-particle fuel compacts and demonstrating equally high in-service fuel performance for the U.S. MHTGR design was assessed. The results were documented in a final report,⁴ a summary of which follows.

The ability of the MHTGR to achieve passive safety features that meet design and user requirements is highly dependent on excellent fuel performance, that is, nearly complete retention of fission products within the coated-particle fuel. In evaluating the feasibility of coated-particle fuel to achieve the requirement goals, use was made of applicable U.S. and FRG experience. In particular, fuel manufacturing methods, in-service fuel performance during normal reactor conditions, and fuel performance during accident-simulation tests were evaluated. Relative to irradiation data, almost exclusive reliance was placed on data made available from the fuel

*On assignment to ORNL from GA.

irradiation program in the FRG. The results of the FRG program have demonstrated that pebble-bed fuel elements containing high-quality coated-particle fuels maintain their coating integrity, retaining essentially all fission products under normal and simulated licensing basis event (LBE) conditions.

If the U.S. reference fuel element were essentially the same as the FRG pebble-bed element relative to the stresses imposed on the coated fuel particles, the FRG data could be directly applied to the MHTGR. However, the stress conditions are not necessarily the same. The FRG fuel element makes use of a carbonaceous overcoating surrounding the individual coated fuel particles and compression of the overcoated particles into a spherical element. The fabrication and irradiation stresses imposed on the coated particles appear to be low in the FRG fuel element, as evidenced by their excellent irradiation performance. The U.S. fuel element contains fuel compacts fabricated by injecting a pitch/graphite mixture into a die containing graphite and coated fuel particles. The pitch is carbonized, leaving a fuel compact having a highly heterogeneous matrix with low overall density and many local voids. Fabrication and irradiation stresses imposed on the coated particles in this process can be high. Because of the fabrication-induced stresses, it has been difficult to meet the draft fresh fuel specifications regarding the uranium contamination level and the number of defective coatings in the fuel compact.

Recently, an improvement in the fabrication process provided a low-density protective pyrocarbon (PyC) coating on each TRISO-coated particle. The resulting overcoated particle is called a "TRISO-P" particle. This new protective coating reduces stresses exerted on the coatings during fabrications of the fuel compact, resulting in a dramatic reduction in the number of failed coatings. Stresses introduced during fuel irradiation are not known at this time, but a test program is under way; initial results are expected during FY 1988-89. The irradiation-induced stresses are not expected to cause significant failures in the high-quality coatings, but it is important that irradiation data be obtained to confirm this result.

The limited amount of U.S. irradiation data available for the UCO/ThO₂ fuel system was obtained on coated fuel particles and fuel compacts fabricated using the old process. The compacts tested previously were

exposed to normal irradiation conditions and contained much higher levels of contamination and defective fuel coatings than were acceptable for the MHTGR. Also, significant numbers of high-quality coatings apparently failed during irradiation testing. The results from testing individually unbonded irradiated coated particles under LBE conditions indicate no correlation of particle failure with prior irradiation history for high-quality coated particles.

Analytical models of fuel performance have been developed both in the U.S. and FRG and predict excellent performance for high-quality coated fuel particles. However, these models do not address the irradiation stresses that can be imposed on coated particles located within a carbonaceous matrix during fuel exposure, and it is doubtful that a particle model can be developed that takes this factor into account. Irradiation data are needed to determine empirically if irradiation stresses cause coating failures.

Overall, the FRG experience has demonstrated convincingly that pebble-bed reactor fuel elements that contain high-quality coated particles (i.e., having a very low fraction of defective coatings and a very low contamination level in the matrix material) have outstanding irradiation performance. The U.S. experience on previous-generation UCO/ThO₂ fuel compacts has indicated that irradiation-induced stresses can occur in such compacts and apparently can cause failure of high-quality coated fuel particles. The fabrication process developed most recently in the U.S. has significantly reduced stresses imposed on the coated fuel particles during the compact injection process. The additional protective PyC layer added to the standard TRISO coating reduces the stresses generated during the fabrication process. As a result, the current U.S. fuel compact shows promise of achieving a very low level of fission product release during reactor exposure. Demonstration that the fuel compacts will indeed have the required performance characteristics in an irradiation environment is a high-priority need, and associated experiments are currently being initiated.

4.1.2 Irradiation Capsule Design, Assembly, and Operation

4.1.2.1 Capsule HRB-21 - M. J. Kania, B. W. Patton, K. R. Thoms, and D. T. Goodin

The HRB-21 irradiation experiment will provide data for the development of the MHTGR in accordance with the F/FP TDP.² The test is a cooperative effort of DOE and JAERI; the participants are ORNL, GA, and the TOKAI Research Establishment. The HRB-21 capsule will contain 24 fuel compacts of U.S. MHTGR reference-quality fuel as well as a variety of encapsulated specimens related to MHTGR fuel performance, fission product transport, and control materials. A companion capsule designated HRB-22 will contain the JAERI fuel. The HRB-21 capsule was designed and will be assembled by ORNL. The fuel compacts, graphite fuel body assemblies, and certain encapsulated specimens were provided by GA. The irradiation will take place in the removable beryllium reflector area of the High Flux Isotope Reactor (HFIR) at ORNL. The performance of the fuel during irradiation will be monitored by on-line fission-gas-release measurements. Postirradiation examinations (PIE) and core conduction cooldown simulation tests will be performed with the HRB-21 fuel compacts.

4.1.2.1.1 Objectives

The overall objectives of the HRB-21 irradiation experiment are (1) to determine the performance of U.S. low-enriched uranium (LEU) coated-particle fuels under irradiation conditions and (2) to provide samples for postirradiation testing to determine the capability of the particles to meet criteria on failure fraction limits under accident conditions. More detailed objectives are

- to provide the first in-reactor test of U.S. fuel meeting MHTGR fuel quality specifications and provide a portion of the data necessary to validate the U.S. reference fuel performance models under normal operating conditions;
- to provide a portion of the samples needed for a series of conduction cooldown simulation tests designed to update, as necessary, and validate the U.S. reference fuel performance models under LBE conditions;

- to provide in-reactor coating-failure data for potential revision of the defective-particle performance models;
- to provide data on partition coefficients and sorptivity isotherms of cesium, strontium, iodine, and silver on MHTGR core materials;
- to provide data on the irradiation performance of both PyC-coated and PyC/SiC-coated particles containing B_4C ; and
- within the auspices of the US/FRG Umbrella Agreement for Cooperation in GCR Development, Fuel Fission Products and Graphite Subprogram, PWS-FD-25, to provide significant fuel performance data for exchange with the FRG in return for the irradiation of the HFR-B1 capsule.

4.1.2.1.2 Capsule design and description

The HRB-21 irradiation capsule consists of a doubly contained, single purged cell in which fuel compacts are surrounded by H-451 graphite bodies (Fig. 4.1). The fuel compacts, three per graphite body, are composed of TRISO-coated UO₂ and ThO₂ particles embedded in graphite matrix material. The eight graphite bodies are contained within two Inconel 718 containment tubes (Fig. 4.2) designed to independently withstand 6.9 MPa (1000 psi) inside or outside. During HFIR operation, pure helium at 5.5 MPa (800 psi) flows between the primary and secondary Inconel containments. A gas mixture of helium and neon flows between the primary Inconel containment and the graphite body surface. By varying this gas mixture within the primary containment, the radial conductance of heat to the surrounding HFIR cooling-water system is controlled and the operating temperature of the fuel compacts may be regulated. The entire capsule is fitted with an outer aluminum liner designed to protect the HFIR beryllium reflector.

The use of Inconel 718 in place of the stainless steel 304 containment used in earlier HRB capsules allows an increase in the linear heat rate from 49 to 65 kW/m (15 to 20 kW/ft). The increased heat rate allows irradiation of a larger number of particles. This is especially important for HRB-21 because large sample sizes are needed to validate the very low failure fraction expectations for MHTGR reference-quality fuel. The HRB-21 fuel loadings for each graphite body are shown in Table 4.2.

Fig. 4.1. Horizontal section through capsule HRB-21.

The design thermal analysis was performed with the HEATING6⁵ code using a three-dimensional model in cylindrical coordinates. The model consisted of a one-sixth mock-up of a graphite holder with insulated boundary conditions on the top, bottom, and planes of symmetry through the body. The graphite dimensions for each fuel body were determined using a two-dimensional heat-transfer analysis. All thermocouples in HRB-21 will be placed within the graphite bodies. A total of 24 junctions will be used to monitor the graphite temperatures in four separate thermocouple array assemblies (TCAT).

Reference Fuel Compacts

The fuel compacts were fabricated and placed within the graphite bodies by GA. Table 4.3 lists the dimensions, weights, and heavy-metal

Fig. 4.2. Vertical section through capsule HRB-21.

Table 4.2. HRB-21 graphite body fuel loadings

Graphite body	Total fuel loading (g)				Fissile particles	Fertile particles
	^{235}U	^{238}U	Total uranium	^{232}Th		
1, 8	0.288	1.173	1.466	7.500	6,700	13,310
2, 7	0.239	0.975	1.214	7.500	5,540	13,310
3, 6	0.176	0.716	0.891	7.500	4,070	13,310
4, 5	0.146	0.594	0.740	7.500	<u>3,380</u>	<u>13,310</u>
Totals (8 bodies) =					39,380	106,480

contents for each of the fuel compacts used in capsule HRB-21. The reference MHTGR fuel compacts are made by a hot-injection process, which consists of injecting a viscous mixture of binder, binder additive, and graphite filler at a temperature of 150 to 200°C into a closely packed array of coated fuel particles and inert H-451 graphite shim particles contained in a cylindrical mold. The nominal injection pressure has been 6.9 MPa (1000 psi) but was decreased for HRB-21 compact manufacture to reduce particle-to-particle interactions and, thus, to decrease the levels of manufacturing-induced coating defects. Petroleum-derived pitch is used as the binder phase, and graphite powder is used for the filler. The as-injected compacts are then heat-treated in incremental temperature steps up to about 1750°C to carbonize the binder phase, drive off volatile gases and impurities, and stabilize the dimensions of the compacts.

The coated fissile and fertile particles used for compact manufacture were TRISO-coated UCO and ThO_2 that were produced in a 240-mm-diam coating furnace at GA.⁶ Companion fuel compact samples were tested before selection of compacts shown in Table 4.3 to ensure that production lots were consistent with MHTGR reference-quality fuel specifications with respect to heavy-metal contamination and manufacturing defect particle fraction. Release to birth (R/B) ratios were determined on each of the unirradiated HRB-21 compacts by a low-flux irradiation in the GA TRIGA reactor.

Table 4.3. Capsule HRB-21 fuel compact dimensions
and the total heavy-metal loading for all
24 compacts to be irradiated in HRB-21

Position from top of capsule	Hole	Outer diameter (mm)			Length (mm)	Weight (g)	Total Uranium (g)	Total Thorium (g)
		Top	Middle	Bottom				
1	A	12.45	12.46	12.30	49.13	10.651	0.516	2.355
	B	12.45	12.43	12.27	49.19	10.666	0.516	2.355
	C	12.46	12.45	12.30	49.22	10.667	0.516	2.355
2	A	12.48	12.47	12.37	49.28	10.655	0.402	2.540
	B	12.44	12.47	12.35	49.30	10.676	0.402	2.540
	C	12.45	12.45	12.35	49.26	10.691	0.402	2.540
3	A	12.48	12.46	12.38	49.28	10.581	0.298	2.540
	B	12.46	12.44	12.39	49.29	10.597	0.298	2.540
	C	12.46	12.43	12.33	49.30	10.591	0.298	2.540
4	A	12.46	12.47	12.31	49.32	10.527	0.248	2.500
	B	12.51	12.46	12.32	49.32	10.482	0.248	2.500
	C	12.48	12.46	12.32	49.32	10.505	0.248	2.500
5	A	12.50	12.46	12.33	49.33	10.522	0.248	2.500
	B	12.49	12.45	12.35	49.33	10.511	0.248	2.500
	C	12.49	12.50	12.34	49.32	10.517	0.248	2.500
6	A	12.44	12.46	12.35	49.35	10.589	0.298	2.540
	B	12.46	12.43	12.37	49.32	10.583	0.298	2.540
	C	12.46	12.45	12.35	49.28	10.574	0.298	2.540
7	A	12.46	12.41	12.37	49.33	10.660	0.402	2.540
	B	12.47	12.43	12.35	49.33	10.676	0.402	2.540
	C	12.46	12.43	12.38	49.26	10.703	0.402	2.540
8	A	12.45	12.44	12.32	49.22	10.655	0.516	2.355
	B	12.45	12.45	12.33	49.22	10.646	0.516	2.355
	C	12.45	12.44	12.31	49.16	10.636	0.516	2.355

Encapsulated Specimens

Each HRB-21 graphite fuel body will include three 2.38-mm-diam holes that will contain encapsulated "piggyback" specimens for study of fission product and B₄C behavior. Specimens to be included are

- samples of unbonded standard fissile and fertile fuel particles selected from the reference fuel fabrication batches,
- UCO fissile particles in which buffer coatings are missing, and
- sorptivity specimens fabricated from oxidized and unoxidized graphite and TRISO-coated inert particles with and without the outer PyC coatings removed.

A general loading arrangement for the encapsulated specimens is shown in Table 4.4.

Table 4.4. HRB-21 encapsulated specimen test matrix

Graphite body	Fuel compact hole		
	A	B	C
1	S ^a	S	Def(U) ^b
2	Std(U) ^c	Def(U)	Def(U)
3	Std(U)	Def(U)	Def(U)
4	S	S	Def(U)
5	S	S	Def(Th) ^d
6	Std(Th) ^e	Def(Th)	Def(Th)
7	Std(Th)	Def(Th)	Def(Th)
8	S	S	Def(Th)

^aS = Sorptivity specimen.

^bDef(U) = Defective fissile particle (UCO).

^cStd(U) = Standard fissile particle (UCO).

^dDef(Th) = Defective fertile particle (ThO₂).

^eStd(Th) = Standard fertile particle (ThO₂).

4.1.2.1.3 Capsule operation

Temperature

Capsule HRB-21 has a target fuel volumetric mean temperature of 975°C. The graphite body radial gaps have been specified to maintain this

temperature as uniformly as possible over the length of the capsule. Fuel compact temperatures will be inferred from the graphite thermocouple temperature data and the preirradiation design thermal analysis. In-reactor temperature control will be maintained by adjusting the neon and helium concentrations in the sweep gas. After irradiation and completion of the fuel/graphite metrology, a detailed thermal analysis will be performed to obtain in-reactor fuel compact temperatures.

No thermocouples will be placed within the fuel compacts themselves. The fuel operating temperature uncertainty during irradiation could be as much as $\pm 125^{\circ}\text{C}$. This uncertainty should be reduced by postirradiation metrology. The average value for the graphite operating temperature is expected to be approximately 650°C .

Fuel Burnup

The HRB-21 fuel fissile and fertile particle burnup estimates have been made using the CACA2 fuel depletion code⁷ for a six-cycle irradiation in the HFIR Removable Beryllium Reflector Irradiation Facility (RB* Facility). The maximum expected end-of-life burnup near the HFIR midplane (positions 4 and 5) is 24.4% FIMA for the fissile fuel and 2.7% for the fertile fuel. The minimum expected end-of-life burnup at the graphite body positions 1 and 8 is 15.5% for fissile and 0.7% for fertile. The expected burnup profiles as a function of irradiation time are shown in Fig. 4.3. Calculations of burnup based on dosimetry data and unbonded particle analysis will be performed during HRB-21 postirradiation examinations.

Fluence

The expected HRB-21 fast fluence exposure values have been calculated using dosimetry data acquired previously.⁸ Because of the modifications to the RB* facility, the fluence predictions are not known precisely and could be in error by up to 10%. Dosimetry located within the HRB-21 graphite bodies will provide measured flux and energy spectrum values. The predicted fast fluence values as a function of capsule position based on prior data are shown in Fig. 4.4.

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Fig. 4.3. Expected fissile and fertile burnup profile in capsule HRB-21.

ORNL-DWG 88-11042

Fig. 4.4. Expected fast fluence in capsule HRB-21.

4.1.2.1.4 Postirradiation examination

After completion of the scheduled six-cycle irradiation in HFIR, capsule HRB-21 will be removed from the reactor and allowed to cool in the poolside facility for one to three weeks. The capsule will then be loaded into a special transportation cask and moved to the High Radiation Level Examination Laboratory (HRLEL) for disassembly and PIE.

The irradiated specimens will be removed from the capsule containments by making circumferential cuts at the top and bottom of the fueled portion of the capsule. At this point, the thermocouple arrays will be removed and stored for further analysis if necessary. The circumferential cut surfaces will be deburred, and the carbon insulators and graphite fuel bodies will be removed by pushing them out onto a clean positioning tray. The insulators and graphite bodies will be visually examined, photographed, and any unusual areas noted and photographed. Dimensional measurements will be made on the insulators and graphite bodies at various locations to permit direct comparison with preirradiation dimensions. These data will be used in the postirradiation thermal analysis.

For each graphite fuel body, the encapsulated specimens - dosimeters, sorptivity specimens, and unbonded particle specimens - will be extracted by removing the screw caps holding them in place. The sorptivity specimens and dosimetry specimens will be stored in properly identified containers to be examined later by high-resolution gamma spectrometry to determine activation product inventories. The standard and defective unbonded particle specimens will be removed similarly and stored in properly identified containers. The unbonded particles will be removed from the niobium encapsulation, and the container gas content will be measured upon opening. After removal, the unbonded particles will be subjected to visual and irradiated-microsphere gamma analysis (IMGA) examinations to establish failure levels.

For a limited number of graphite fuel bodies (tentatively, bodies 5, 6, and 8), the irradiated fuel compacts will be unloaded by removing the screw caps holding them in place at the bottom of the body. The compacts will be visually examined, photographed, and any unusual areas noted and photographed. Diameter and length measurements will be made at several

locations along the compact length to establish radiation-induced shrinkage for use in the postirradiation thermal analysis. After fuel compact removal, the bonding strength of the upper fuel closure plugs (cemented in place at time of fabrication) will be measured by a push-out test using a force gauge.

The fuel compacts from fuel bodies 5, 6, and 8 will be subjected to detailed examinations to establish the fraction of particles having (1) totally failed coatings and (2) failed SiC but intact outer PyC coatings. A tentative PIE test matrix for these compacts follows:

<u>Compact</u>	<u>TRIGA</u>	<u>Metallography</u>	<u>Total failure</u>	<u>IMGA</u>	<u>Reactivation</u>	<u>CCT</u>	<u>Post-CCT examination</u>
1						X	X
2					X	X	X
3						X	X
4					X	X	X
5	X	X	X	X			
6	X	X	X	X			
7					X	X	X
8	X	X	X	X			

The integrity of as-irradiated fuel particles heated in the conduction cooldown tests (CCTs) can also be assessed by the initial release profiles measured as the CCT samples are brought to test temperature (prior to high-temperature-induced failure). For example, adequately low releases (in agreement with model expectations) during both in-reactor operations and during heat-up to 1600°C (and for an expected time thereafter) can be considered a demonstration of the retentive capability of the fuel for MHTGR service. The in-reactor performance of the fuel compacts in positions 1, 2, 3, 4, and 7 will be aided by assessment of the initial CCT release profiles.

A conduction cooldown test facility (CCTF) is being constructed at ORNL to provide simulation of the conditions of MHTGR LBE's and measurement of fuel failure and fission product release. The majority of the HRB-21 graphite bodies will be subjected to postirradiation heating tests in this facility. The CCTs will emphasize the temperature range of 1400 to 1800°C.

A final allocation of samples within the postirradiation test matrix depends on HRB-21 in-reactor fuel performance results. A tentative CCT matrix for HRB-21 samples, which assumes in-reactor performance that meets MHTGR release limits, is given in Table 4.5.

Table 4.5. Preliminary conduction cooldown test matrix for irradiated fuel compacts from capsule HRB-21

Test	Body	Heating temperature (°C)	Average FIMA (%)		No. of particles	
			Fissile	Fertile	Fissile	Fertile
1	1	1600	15.5	0.7	6,700	13,310
2	2	1400	20.2	1.5	5,540	13,310
3	7	1800	20.2	1.5	5,540	13,310
4	3	1600	23.0	2.3	4,070	13,310
5	4	1600	24.4	2.7	<u>3,380</u>	<u>13,310</u>
			Total =		25,230	66,550

For planning purposes, the CCT times are estimated to be 1000 h for tests at temperatures of less than or equal to 1600°C and 500 h for tests at temperatures greater than 1600°C. The actual test times will be determined by the test engineer in consultation with the data analyst, based on the observed fission product release in each test, the furnace capabilities, and the data needs for model validation purposes.

4.1.2.1.5 Schedule

The schedule for capsule HRB-21 is dependent to a large extent on the availability of the HFIR. Plans call for pre-startup preparations to be completed such that irradiation of HRB-21 can begin in July 1988. The irradiation is planned for six full cycles at a full-power rating of 85 MW for HFIR.

4.1.2.2 Irradiation experiment HFR-B1 - B. F. Myers

Irradiation experiment HFR-B1 is being conducted in the High Flux Reactor (HFR) located at the Joint Research Centre (JRC), Petten, The Netherlands. This experiment is devoted primarily to the measurement of

fission product transport as it occurs in core materials of the MHTGR.⁹ The experiment consists of three capsules:

1. Capsule A is designed to measure metallic fission product transport partition coefficients in graphite. It is operated in dry helium at 900°C and will achieve an accumulated fast fluence of 6.3×10^{25} neutrons/m².
2. Capsule B is designed to measure fission gas release as a function of temperature. It is operated over a temperature range of 950 to 1200°C in dry helium and will achieve a fast fluence of 1.0×10^{25} neutrons/m².
3. Capsule C is designed to measure the effects of hydrolysis of graphite matrix material and exposed UCO fissile fuel. It is operated over a temperature range of 800 to 1050°C and will achieve a fast neutron fluence of about 7.7×10^{25} neutrons/m². Water vapor is injected into this capsule to facilitate hydrolysis of the core material contained in the capsule.

The experiment conducted in Capsule C is interactive, requiring adjustment of water vapor level and temperature in subsequent testing based on the results of foregoing testing.

From June 14 to 30, 1987, a U.S. representative was sent from ORNL to the JRC in Petten to participate in and provide guidance for the first water vapor injection tests. During HFR irradiation cycle 87.05, two water vapor injection tests were performed in Capsule C. In the first test, a nominal concentration of water vapor of 150 ppmv (parts per million by volume) existed in the sweep gas for a period of 6.2 h. In the second test, the nominal concentration was 60 ppmv for a period of 95.2 h. Fuel operating temperatures during these injections were about 1015°C at the fuel centerline and about 915°C at the fuel compact surface. Four phenomena were monitored during these tests: (1) the quantity of water vapor injected into the capsule, (2) the quantity of water vapor discharged from the capsule, (3) the gas species and their concentrations resulting from hydrolysis, and (4) the release of Kr and Xe fission gas isotopes.

A prompt analysis of these experiments was performed to assess the performance of the system, to understand qualitatively the chemical and

physical processes involved, and to recommend a course for future tests. This analysis indicated that

- both carbonaceous material and exposed UCO fuel kernels were being hydrolyzed;
- the hydrolysis of the carbonaceous material was clearly dominant (the exposed UCO fuel was hydrolyzed to the extent of 0.04% of the UC_2 portion);
- the molar quantities of H_2 , CO , and H_2O were equal (within scatter of data), as expected from the hydrolysis of carbonaceous material (and UC_2 fuel);
- water vapor was discharged from the capsule only after about 4×10^{-4} mol of H_2O had been injected into the capsule, and at this point the concentrations of H_2 and CO decreased significantly while that of CO_2 rose;
- the release of fission gases increased when water vapor was injected, continued briefly thereafter, and eventually declined to the pre-hydrolysis value;
- the surface of the carbonaceous material accessible to the H_2O was limited either by structural features or by a relatively high reactivity of water; and
- in addition to adsorption of H_2O leading to reaction, water adsorbed nonreactively, as was indicated by the desorption of H_2O after the termination of the water vapor injection.

The hydrolysis experiment in Capsule C of HFR-B1 is very informative and has already provided new and useful results. To exploit the full potential of Capsule C will require a significant effort in the future. Details of the experimental results from the first two water injections were provided in an unpublished memorandum (unpublished results of B. F. Myers, August 1987).¹⁰

In addition to the experimental results described in the preceding discussion, the visit to Petten uncovered several shortcomings in the operation of the HFR-B1 experiment. These have been discussed in detail with the Petten staff, and steps have been taken to solve these problems.

4.1.3 Equipment Development and Maintenance

The IMGA, postirradiation gas analyzer (PGA), X-radiography facility, scanning electron microscope (SEM), and new CCTF are important systems

integrated into the PIE of irradiated MHTGR fuels at ORNL. Periodic upgrading and maintenance of these facilities are necessary so that their reliability and capabilities remain consistent with current requirements.

No significant amount of PIE was performed in the HRLEL for the Fuel Materials Development Program. For the most part, PIE facilities such as the PGA and the X-ray facility were not operated at all; the IMGA system was operated only about 20% of the available time, and little or no PIE work was involved. New equipment added to the SEM facility has resulted in a significant improvement in the electron microscopy capabilities at the HRLEL. Procurement of the high-temperature furnace component of the CCTF was completed, and furnace modifications were designed and initiated to accommodate the needs of the core conduction cooldown testing program.

4.1.3.1 The irradiated microsphere gamma analyzer - M. J. Kania and C. A. Baldwin

The availability of the IMGA system during this report period was significantly improved with the procurement and installation of a new data storage device. The two original 5-megabyte (Mb) PERTEK disk drives purchased along with the ND-6700 Computer-Based Multichannel Analyzer System were replaced with a fixed 30-Mb Winchester Drive and a removable 10-Mb cartridge system. Before the new system was installed, down times caused by hard-disk crashes on the ND-6700 system amounted to about 25% of the available user time. The new system has virtually eliminated down time resulting from equipment failure and, at the same time, has increased the data storage capacity by a factor of four.

Recalibration of the IMGA system high-resolution germanium detector and its associated high-speed counting electronics was completed for several source-to-detector positions. Excellent efficiency vs photon energy calibrations were obtained using National Institute of Standards and Technology (NIST) and Amersham calibrated sources. A linear relationship was established between the analog-to-digital converter (ADC) channel and full-energy peak full width at half maximum (FWHM) with photon energy indicating that the IMGA electronics system was quite stable.

**4.1.3.2 Scanning electron microscope/microprobe - M. J. Kania,
H. M. Henson, and L. G. Shrader**

During this report period, accessory equipment procurement and installation was completed for the JEOL JSM 840A research-grade SEM located at the HRLEL. Two wavelength dispersive spectrometers (WDS) were added to the system to provide electron microprobe capabilities. Also, a Tracor-Northern model 5700 image processing and enhancement system was obtained and installed. The acquisition of the new equipment completes the first stage in upgrading electron microscopy capabilities for use in the Fuel Materials Development Program. The new system is now fully operational, and operator training is under way on non-radioactive materials. Figure 4.5 is a photograph of the JSM 840A operator console and the SEM column. Modifications to the system to incorporate shielding to permit examination of highly radioactive materials will be accomplished in the second stage of the upgrading process, which will occur over the next two to three years.

As a result of completing stage 1 of electron microscopy capabilities upgrade, a definitive decision was made to abandon the MAC 450 Shielded Electron Microprobe. The MAC 450 system was installed on the second level of the HRLEL in the late 1960s for the purpose of performing electron microprobe examinations on highly radioactive fuel materials. For the past ten years, it has been operated only sporadically because of nonexistent support and lack of available maintenance. Disassembly of the system is in process, and all noncontaminated electronic components have been removed and disposed of properly. The contaminated components will be placed in shielded containers and removed to the proper burial facilities. The new JEOL JSM 840A system will be moved to this area during FY 1988.

**4.1.3.3 Core conduction cooldown test facility - M. J. Kania,
D. T. Goodin, S. D. Clinton, J. C. Mailen,
M. S. Osborne, and M. E. Peters, Jr.**

A remotely operated facility is being prepared at ORNL to simulate MHTGR core conduction cooldown accidents. The CCTF will permit the in situ detection and quantification of key radionuclides released from irradiated fuel compacts at temperatures up to 1800°C. During this report period, procurement was completed on the acquisition of a Model 1000-3560-FP20

Fig. 4.5. JSM 840A operator console and SEM column.

Graphite Element High Temperature Furnace. The furnace has a cylindrical heating zone of about 89 mm diam (3.5 in.) by 152 mm long (6.0 in.) and is rated for a maximum temperature of 2650°C in helium. The entire system, including the graphite furnace, a power supply system, and advanced control and monitoring instrumentation, was delivered in August 1988. A second furnace was also procured as a backup device.

In September 1988, assembly and checkout of the high-temperature furnace was initiated. Modifications to the vertical furnace to facilitate loading and unloading of fuel compacts were designed and begun. Elevator assemblies are to be used to permit automated insertion and removal of the water-cooled cold-finger device from the top of the furnace into the hot zone. Similarly, a second elevator assembly will permit automated specimen loading and unloading from the bottom of the furnace. Both the design of the cold-finger apparatus, which contains the removable fission product deposition surface, and the fuel specimen holder were modified to accommodate this.

Procurement was also initiated for a gamma counting table and high-resolution gamma spectrometry system to be used as diagnostic equipment to quantitatively determine fission product release during the conduction cooldown simulation tests.

Completion of the CCTF is planned for mid FY 1988, and installation and checkout in a hot-cell environment is planned by the end of December 1988. The first use of the facility in the DOE-funded program will be with the irradiated fuel from irradiation capsule HRB-21. Completion of the irradiation of capsule HRB-21 is planned for December 1988, and the core conduction cooldown simulation will follow in the spring of 1989.

4.2 FISSION PRODUCT PERFORMANCE VERIFICATION (WBS 1601.03) - J. C. Mailen

4.2.1 Bench-Scale Adsorption/Desorption System - S. D. Clinton, J. H. Wilson, C. M. Simmons, and T. V. Dinsmore

A bench-scale system using tracer methods is being used to examine the deposition and lift-off of simulated fission products (SFP) from alloys

typical of those that will be used in the MHTGR. During CY 1987, the tests have examined the behavior of iodine.

A bench-scale unit was designed and constructed for the SFP sorption/desorption experiments. A schematic drawing of the apparatus is shown in Fig. 4.6. Gases, mixtures of helium with low levels of impurities, such as hydrogen, carbon monoxide, and carbon dioxide, are supplied to the system from standard high-pressure cylinders. Flow rates are set using shielded rotameters that can withstand ~8 atm. One helium gas stream is passed through a hot copper oxide bed to convert the hydrogen in the mixture to water. A reaction bed containing calcium carbide is provided for determining the amount of water in the gas stream by passing the gas through the bed and measuring the acetylene that is produced by the reaction of water and calcium carbide.

ORNL-DWG 87-169R3

Fig. 4.6. Once-through apparatus for iodine plate-out and lift-off.

The feed gas, with the desired levels of impurities, is apportioned to two test-furnace assemblies, which are located in a hood, as indicated in Fig. 4.6. Each furnace assembly consists of a 19.1-mm-OD quartz tube mounted in a clam-shell heater. Each heater has three independent windings to allow separate temperature zones and/or temperature gradients to be maintained. The quartz tubes contain, sequentially, a solid quartz cylinder, the iodine source, the alloy specimens to be used for plate-out measurements, and another solid quartz cylinder. The solid quartz cylinders are 2.54 cm long and occupy ~90% of the cross-sectional area; this helps minimize the amount of iodine transported from the tube by diffusion. The gas stream passes through the quartz tube and, after contacting the alloy specimen, exits through charcoal traps that collect any gaseous fission products. A gamma detector is mounted on a track under the furnace assemblies. Activities of radioactive tracers at various locations (e.g., on the iodine source, alloy specimen, and in the charcoal traps) can be determined on-line by gamma counting with this mobile detector.

Two methods for introducing the tracer-labeled iodine into the furnace assembly have been examined. In the initial tests to be reported here, tracer-labeled palladium iodide was prepared by precipitation, according to the method of Osborne et al.¹¹ This method was suitable for iodine loading on the metal alloy specimens of $\sim 0.3 \mu\text{g I/cm}^2$, but because of difficulties in precipitating and collecting tiny amounts of palladium iodide, was not useful for the much lower loadings expected in the MHTGR. A second method has recently been developed in which the iodine tracer (iodine as sodium iodide), without added carrier, is added to a small volume of nitric acid containing nitrite. The iodine is oxidized to the elemental form, volatilized into a stream of helium, and deposited on a small amount of palladium powder or wire at 150°C. The powder or wire is then loaded into the furnace assembly and heated to $\sim 500^\circ\text{C}$ to desorb the iodine. This technique allows transfer of $\sim 0.01 \mu\text{g}$ of iodine and can be used to study loadings on the alloy as low as $10^{-6} \mu\text{g I/cm}^2$ (more than 5 orders of magnitude lower than achieved by the first method) and can also be used for external monitoring of iodine concentration on the alloy specimen and the charcoal traps.

In an experiment, the iodine is desorbed from the iodine source and deposited on the alloy strip. The temperature of the iodine source is then reduced to 100 to 200°C, at which temperature no additional iodine is desorbed and the charcoal traps are replaced. A known, slow flow of helium is then passed over the alloy specimen to desorb iodine, and the equilibrium concentration in the gas phase is determined by the amount of iodine collected by the charcoal traps. This procedure can be repeated several times with a loaded alloy strip to obtain additional data. The data obtained to date,¹² using the precipitated PdI₂ source has confirmed that these techniques can give data comparable with that obtained earlier by Osborne et al.¹¹ The three data points obtained with the PdI₂ source have been added to a figure reproduced from Osborne et al.¹¹ (see Fig. 4.7).

Future iodine studies will use iodine adsorbed on palladium metal and will examine the low alloy loadings of more interest to the MHTGR program. In later tests, a mixture of cesium iodide and cesium oxide will be used to represent the approximate ratio of cesium to iodine released by the fuel. In some tests, the iodine source will be contained in a graphite capsule to more closely simulate fission product release from HTGR fuel.

4.2.2 Bench-Scale Lift-Off System - S. D. Clinton, J. H. Wilson, C. M. Simmons, and T. V. Dinsmore

The system to be used in the lift-off studies is shown in Fig. 4.8. Gas from a standard, high-pressure helium cylinder passes through a high-flow pressure regulator, a preheater, the lift-off test section, a gas-cooler, a filter capable of absolute retention of particles $\geq 0.2 \mu\text{m}$, an orifice, and a Hastings flowmeter. The preheater and the lift-off test section are contained within a Marshall furnace; the preheater has been sized to ensure that the maximum gas flow to be used will be heated to essentially the temperature of the furnace up to 700°C, the maximum lift-off temperature to be examined. The gas flows necessary to obtain shear ratios (relative to the nominal reactor flow conditions) of 0.5 to 5 range from 200 to 850 L/min (STP). The actual flow rate is monitored by the flowmeter and is set by the size of the orifice. The maximum pressure in the system as currently designed is limited to 690 kPa by the silver membrane filter (Poretics Corp., Livermore, California 94550). The iodine

Fig. 4.7. Comparison of measurements of iodine sorption at 400°C (from Osborne et al.). The figure also contains data from Milstead and Neill. Sources: M. F. Osborne, R. B. Briggs, and R. P. Wichner, *Iodine Sorption on Low-Chromium Alloy Steel*, ORNL/TM-7755 (January 1982); C. E. Milstead, W. E. Bell, and J. H. Norman, "Deposition of Iodine on Low Chromium-Alloy Steel," *Nucl. Appl. Technol.* 7, 361-66 (1968); and F. H. Neill, *Adsorption and Desorption of Iodine on Mild Steel*, ORNL/TM-2763 (1970).

or other fission product will be preloaded within the lift-off test section before the lift-off test; this procedure will be described later. Because the capability to load fission products from a high-velocity gas stream is lacking, it will be necessary to pretreat the test section by exposure to gas flows up to a shear ratio of 1.0. Any material lifted off during this low-velocity period will be collected and examined but represents material that would not have been deposited on the alloy under normal reactor operations. After this period of the test, the filter will be replaced. A series of additional lift-off tests will then be made at increasing shear ratios. In each test, a properly sized orifice will be installed and the

Fig. 4.8. Lift-off system.

gas flow will be increased by increasing the pressure at the pressure regulator until a maximum pressure of ~600 kPa is obtained at the exit of the lift-off test section. This flow and pressure will then be maintained for a period necessary for obtaining an accurate flow rate. The quantities of lift-off obtained will be primarily determined by the iodine found on the silver membrane filter and on a backup iodine trap made of silver plated York mesh. The lift-off test section will be gamma-scanned before and after a test to determine the distribution of iodine within the test section, but the lift-off is not expected to be large enough to yield an accurate determination of lift-off by this method. The cooler will also be gamma-scanned after the test. The silver membrane filter will be both gamma-scanned and examined electron microscopically to determine the size, shape, and metallic composition of any dust collected.

As pointed out previously, the fission product of interest must be plated on the inside of the test section before the lift-off tests are made. The lift-off test section will be a piece of tubing of dimensions 0.635 cm ID, 1.33 cm OD, and ~30.5 cm long. The external surface will be plated with gold to prevent chemisorption of the fission products on the exterior during deposition and to facilitate any necessary cleaning of the

exterior. The tube will be inserted into the deposition test system, and a clean gas flow over the exterior surface will limit exposure to the fission product. Fission product vapor will be generated by heating the source and allowing the vapor to deposit in the interior of the lift-off test tube. If possible, the deposition will be limited to the first 20 cm of the tube and the tube will be installed in the lift-off system such that the gas will enter the end of the tube without plate-out. This is desirable to minimize the lift-off of activity near the entrance of the test section where a stable flow pattern has not been established. After the plate-out is complete, the tube will be removed from the system but a flow of helium will be maintained through the tube to prevent exposure to air. At this time, the exterior of the tube will be cleaned and surveyed to ensure that no significant quantity of fission product is on the exterior surface. After gamma scanning, the tube will be attached to the lift-off test system; again, continuous helium flow will be used to prevent air contact.

The main portions of the lift-off system are in place, and initial testing will begin in April 1988.

4.2.3 Laser-Induced Fluorescence System - S. A. Sherrow

The laser-induced fluorescence (LIF) system obtained from GA was assembled, aligned, and tested, and additional peripheral equipment was installed. The Nd:YAG/dye laser system was found to exceed performance specifications for a new system. The system is located in a newly refurbished laser laboratory and has been approved for operation by the ORNL Laser Safety Committee.

Calibration of the LIF system for measurement of low concentrations of iodine has begun. The LIF system has given an iodine fluorescence spectrum for iodine at a partial pressure of $\sim 10^{-10}$ atm at room temperature. Minor physical modifications to the system and increasing the data accumulation time will allow the determination of iodine concentrations significantly below this level. Procedures have been developed for determining fluorescence vs iodine concentration at elevated temperatures. A quartz fluorescence cell is being fabricated for this work.

4.3 UNIVERSITY SUBCONTRACTS

4.3.1 Massachusetts Institute of Technology - L. M. Lidsky, D. D. Lanning, and J. L. Martin

Massachusetts Institute of Technology is designing and will construct a helium circulation loop (DABLE loop) for studying the behavior of fission products. Studies with the DABLE loop will include the plate-out and lift-off behavior of fission products and the effects of dust and moisture ingress on fission product release during tests at simulated accident conditions. The basic designs of the loop and on-line detection system have been completed, and the current effort is focused on the design of the gas circulator. The objective is to design a circulator capable of operating at the same temperature as the loop to prevent preferential deposition of the circulating fission products in the circulator. The circulator will operate over pressures of 0.1 to 8 MPa, temperatures of 200 to 750°C, flow rates of 5 to 20 L/s, and pressure drops of 3 to 37 kPa at 500°C. The design is expected to give a shear-ratio of 12 for lift-off studies at 500°C and 8 MPa. The current expectation is that the circulator will have centrifugal impellers of approximately 130 mm diam operating at a maximum speed of ~20,000 rpm. After development of the design through testing of rotors machined from aluminum, it is anticipated that the final rotor will be cast from one of the nickel-based superalloys, probably one of the René or Udimet alloys. The seal is to be a labyrinth-type gas seal.

It is expected that the introduction of fission products into the loop can be modeled on the methods developed during bench-scale studies at ORNL. Additional planning is necessary to design for the peripheral systems needed for gas analysis/purification and accident-condition simulations.

4.3.2 Georgia Institute of Technology - A. Schneider, B. Dewald, and A. Levin

Studies at Georgia Institute of Technology are examining the current theories of dust formation from spallation of oxide layers on metal alloys and the behavior of dust in flowing gas streams. Experimental work and modeling studies will be proposed in these areas for future years.

Dust formation from oxide layers depends on many factors, including the alloy used, the interaction of gas impurities (O_2 , H_2O , etc.) in the formation of the oxide films, mechanical disturbance of the films (e.g., through erosion by gas impingement), and thermal stresses induced by thermal cycling. The characteristics of the dust formed are influenced by the conditions of film formation and spallation and by the mechanical comminution of the particles by impingement on surfaces.

Theories of dust behavior can be largely empirical, such as the shear-ratio model currently used in the HTGR program, or can be based largely on modeling of the interactions of dust particles with the flowing gas stream and the alloy surface; the prime example of the latter is the Reeks, Reed, and Hall model.¹³ Each type has advantages and disadvantages. The empirical models do not require the determination of the interaction parameters, which can be fairly difficult, and in a straightforward calculation, may yield predictions that are adequate for design and safety analyses. They do not, however, unambiguously identify the important parameters and are of unknown predictability for extrapolation beyond the range of the experimental data. The more phenomenological models should give a better understanding of the important factors in dust behavior and should be capable of extrapolation to new operating conditions. They do, however, require a significant quantity of data to evaluate the parameters describing the interactions of the particles with the alloy surface. A particular advantage of the Reeks, Reed, and Hall model¹³ is that it has been validated by the injection of tracer-labeled dust particles into the R3 Hinkley Point 'B' Gas Cooled Reactor.

4.4 COMEDIE LOOP STUDIES

The Commissariat a l'Energies Atomique (CEA) COMEDIE Loop is a unique in-reactor facility in the SILOÉ Materials Test Reactor located in Grenoble, France. The facility was designed specifically to characterize (1) the release, migration, and deposition of radionuclides in an HTGR under normal operating conditions and (2) the subsequent response to rapid depressurization events. The primary purpose of the COMEDIE facility is to provide engineering-scale, integral test data to validate the methodology used to predict fission product transport under representative MHTGR

conditions. Work during the past two decades has shown that differential laboratory tests, or out-of-reactor loop tests, are not sufficient; in-reactor testing is required, especially with respect to representative fission product plate-out and lift-off.

The minimum test program in COMEDIE will consist of three in-reactor experiments. Test 1 will characterize plate-out and lift-off under "clean" or "dust free" operating conditions; Test 2 will determine the effects of "dust," or circulating particulate matter, on plate-out and subsequent lift-off; and Test 3 will determine the effects on plate-out and lift-off of a large water ingress before depressurization.

The MHTGR Program requirements dictate that the new data base required to validate the methodology for fission product release be completed one year prior to submission to the Final Standard Safety Analysis Report (i.e., September 1993).

The initial inspection and evaluation of the loop was completed during CY 1987. The objectives of the loop refurbishment are to form a qualified team to operate the loop, review the existing archives from the last operating period (SR1 test), examine the installation and draw up an equipment inventory, check the condition of the existing equipment, draw up a list of the equipment to be replaced or modified, and replace or refurbish the various portions of the loop. These objectives have been completed except for some additional equipment replacement/refurbishment, the inspection of the interior of the loop, and final leak-testing of the loop. Significant replacements of instrumentation have been made and plans are in place to install a gas sampling and analysis system in the control room. The gas circulator has been extensively tested and found to operate up to specifications.

During CY 1988, refurbishment of the COMEDIE loop and blowdown system will continue and will include procurement of required instrumentation as identified during checkout, installation and testing of instrumentation including the computer data accumulation systems, testing of the blowdown system, and fabrication of heat exchanger bundle closure for the depressurization system. The quality assurance plan for the COMEDIE test program will be approved by the U.S. participants and will be issued. The designs of the in-pile heater and for the fuel and reflector loading will be

completed. The heat exchanger design will be completed after mock-up tests. Mock-up tests will be performed on the heat-exchanger bundle closure, cartridge filter (efficiency for iodine removal from the gas phase), and the depressurization train operation (cartridge filter plus depressurization valve). It will not be possible to complete the refurbishment of the COMEDIE loop during CY 1988 because leak-checking cannot be done while the SILOE reactor is being reworked. The pool containment is being modified by the installation of a cylindrical stainless steel liner. The liner installation will be completed in June 1988, the fuel elements will be replaced during July and August, and a new cycle of reactor operation will begin during September. Leak testing of the COMEDIE loop will be done in January 1989, and any leaks found will take days to weeks to repair.

4.5 REFERENCES

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5. GRAPHITE (WBS 1602)

W. P. Eatherly

5.1 GRAPHITE MATERIALS - INTERNALS (WBS 1602.1)

5.1.1 Oxidation Studies - R. A. Strehlow

Stackpole 2020 Graphite has been selected as the material of choice for several applications as a permanent component for modular high-temperature gas-cooled reactors (MHTGRs). Five shipments received during the past several years represent successive attempts by the manufacturer to improve the quality of this material for this nuclear application.

These five shipments were (1) "off-the-shelf" 2020, three billets; (2 and 3) improved, nuclear quality, low-iron content, Lots 1 and 2, three billets; (4) gas purified, four billets; and (5) nuclear quality, five billets.

The first shipment was used in the Core Support Performance Test.¹ Chemical analysis data were obtained on the first three shipments² as well as radiographic examinations. Found in the off-the-shelf material were high levels of iron that are objectionable from the standpoint of oxidation catalysis. This observation led to the subsequent efforts by the manufacturer to improve the material's quality.

The presence of the impurities that catalyzed oxidation of the graphite was of prime concern. Radiographic examination and ash studies of the first shipments also showed, however, the presence of particulates that might impair the strength of the graphite. The improvement program undertaken by the manufacturer had as an additional objective the elimination or minimization of these agglomerated high-Z materials.

The fourth shipment of purified material was stored pending the final result of the improvement program.

A lot that had been intended to be the final iteration of this production sequence was made. However, the manufacturer found it to be of

inadequate purity and withheld it. The final shipment, Lot 5, was received during this reporting period and was submitted for machining.

Lot 5 consisted of five billets numbered 666, 672, 687, 690, and 707. These billets were to be cut for analytical samples, oxidation test specimens, slabs for radiography, and remnants for archival storage.

The experimental plan requires from each of the billets analytical specimens that are sufficient in number to map the impurity profiles longitudinally, radially, and azimuthally in each of the billets. Regions of the billet proximate to the analytical samples were to be used to provide specimens for oxidation studies as well as radiography to assess the inclusions. The cutting plan for the chemical analysis specimens is similar to that for the earlier studies.¹

The experimental plan for this material is to obtain (1) chemical analysis data for each billet, (2) selected comparison oxidation rates, and (3) radiographs of slabs to assess inclusions.

Because strength impairment due to the inclusions has not yet been clearly demonstrated, the comparative oxidation rates and supporting chemistry comprise the primary basis for acceptability of this material for the purpose. The minimization of inclusion size would be expected to provide additional assurance of a well-controlled process for the manufacture of this product.

5.1.2 Destructive and Nondestructive Assay of Union Carbide Grade TS1792 - C. R. Kennedy

The use of nondestructive evaluation (NDE) for graphite has been demonstrated to be effective in estimating the mean strength and variability for fine-grained aerospace and an experimental coarser-grained grade H451I. The general procedures of evaluation are similar; however, the results of NDE must be calibrated for each specific type of material. Therefore, a destructive testing series was required to calibrate the NDE of large billets of the fine-grained grades for potential use as core support and permanent reflector material. The first of this series is an examination of grade TS1792, an experimental grade furnished by Union Carbide Corporation.

Three billets of this material were examined and found to have fairly high sonic attenuation measurements compared with other competitive grades.

The higher attenuation suggests that the material may have a larger background defect size and thus have lower strength. Also more seriously, the higher attenuation will reduce the sensitivity of the NDE for detection of disparate flaws in the billets. One of the three billets was found to have obvious flaws (cracks) and also had large regions where the sonic signal was very high attenuated, yielding regions where the strength estimates were quite variable. An end of the slab from this billet, Z865D, was cut off and examined in more detail by both NDE and destructive testing. This testing was done to demonstrate the credibility of NDE both qualitatively as well as quantitatively with mechanical strength.

The end of slab Z865D was further examined sonically in detail using a 2.25-MHz frequency crystal to increase sensitivity for flaw or defect detection. One additional flaw was found beyond the obvious cracks; however, there were large areas where no back-face signal was obtained because of high attenuation. These areas are shown in Fig. 5.1. The block was further cut into smaller sections (A, B, C, D, E, F, and H, also seen in Fig. 5.1) for further study. These sections were radiographed, and then destructive test specimens were machined from the regions having high variability (toward the ends) and regions having low variability (toward the center). The radiographs of these sections clearly demonstrated that the signal loss was in part caused by strong flow lines in the structure. This has been observed in other molded grades with varying effects on the mechanical properties. In some cases, severe losses in properties were observed, and in others almost no effect was seen. The major concern in this type of structure is that it renders the material uninspectable because of the severe attenuation; as a result, the quality of the graphite cannot be ensured by NDE.

Mechanical property specimens were removed from the sections with and without the flow lines for evaluation. Brittle-ring specimens with 38.1- and 17.8-mm outside diameters and four-point bend specimens $3.2 \times 6.4 \times 44.5$ mm were used to obtain the effect of stress volume on the strength of the material. The larger 38.1-mm-diam brittle-ring specimen did allow a controlled fracture so that the remnant of the broken ring could be used as a C-ring specimen as well. The smaller ring broke completely, and C-ring testing was not possible. The result of the testing

Fig. 5.1. Sketch of the end of billet Z865D showing the regions of no-back-face signal from sonic testing.

is shown in Table 5.1, giving the averages and standard deviations of the specimens from both regions relative to the orientation in the billet. In both groups the billet approaches a molded texture with the axis of symmetry in the long direction. This is in agreement with the electrical resistivity results obtained from eddy-current testing shown in Table 5.1, indicating the same type of preferred orientation. The results of the mechanical testing are also shown in Figs. 5.2 through 5.4 as a function of stress volume. Figure 5.2 shows the strength differences caused by the preferred orientation of the material from the ends of the block, and Fig. 5.3 shows the material from the center of the block. In both cases, the strength-stress volume slopes, which are the reciprocal of the Weibull m constant describing the variability in strength, are the same for both orientations. A comparison of the strength of the material from the two regions is shown in Fig. 5.4, clearly demonstrating the differences in the slope m and the variability of the two regions' strength. This can also be seen in the comparison of the coefficients of variability (COV) given in Table 5.1. There is an excellent agreement in the actually observed COV from the mechanical testing and the COV calculated from NDE and the Weibull m values for material from both regions in the block. This points to the problem in comparing the strength of a material by a comparison of the means from a single-specimen-size test series. It is very clear that as the specimen volume increases, the difference in strength between the two regions becomes greater because of the differences in variability. This is an excellent test series demonstrating the need for evaluating the stress-volume effect and variability in mechanical testing.

5.1.3 Tensile Testing of Grades 2020 and IG11 Graphite - C. R. Kennedy

Tensile testing of grades 2020 and IG11 graphite has been completed. Fifteen specimens in both the width and axial directions from three positions from three billets of each grade have been tested. Comparisons of the tensile strength averages with average density of each position are shown in Figs. 5.5 and 5.6. An exponential relationship can be written to describe the results, as has been previously shown, for other grades and for the effect of oxidation in reducing the density of the graphites.

Table 5.1. Results of mechanical testing specimens
from graphite grade TS1792, billet Z865D

	Number of specimens	$\bar{x}$ (MPa)	s (MPa)	COV (%)
<i>Specimens from higher attenuating region</i>				
B and D specimens				
17.8-mm brittle-ring	34	40.51	2.10	5.18
38.1-mm brittle-ring	18	37.04	1.90	5.13
38.1-mm C-ring	18	32.92	2.62	7.96
F specimens				
17.8-mm brittle-ring	36	43.09	2.26	5.25
38.1-mm brittle-ring	18	38.65	2.77	7.17
38.1-mm C-ring	18	35.37	3.42	9.67
Bend specimens				
Axial (B, D)	23	29.13	2.32	7.96
Face (B, D, F)	43	35.49	2.71	7.64
Side (F)	24	36.86	5.56	15.08
			Average	7.89
			COV from $m = 13.3$	9.00
			COV from NDE	9.50
<i>Specimens from lower attenuating region</i>				
B and D specimens				
17.8-mm brittle-ring	36	40.90	1.86	4.55
38.1-mm brittle-ring	18	38.86	1.31	3.37
38.1-mm C-ring	18	35.99	1.26	3.50
H specimens				
17.8-mm brittle-ring	35	42.54	2.00	4.70
38.1-mm brittle-ring	18	40.95	1.55	3.77
38.1-mm C-ring	18	37.80	2.62	6.92
Bend specimens				
Axial (B, D)	24	35.03	1.17	3.34
Face (B, D, H)	42	37.92	1.58	4.17
Side (H)	22	39.03	2.74	7.02
			Average	4.59
			COV from $m = 23.0$	5.50
			COV from NDE	6.50

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Fig. 5.2 The results of testing specimens from the end region of billet Z865D.

ORNL-DWG 87-8250

Fig. 5.3. The results of testing specimens from the central region of billet Z865D.

ORNL-DWG 87-8248

Fig. 5.4. A comparison of the results from the end and central regions of billet Z865D.

ORNL-DWG 87-19216

Fig. 5.5. Tensile strength of grade 2020 as a function of density.

Fig. 5.6. Tensile strength of grade IG11 as a function of density.

While these grades are fairly isotropic, each demonstrates a slightly molded texture with the width direction stronger than the axial direction.

Each localized position was reexamined by nondestructive sonic testing on a finer scale than used on the larger billets for comparison. The results are given in Fig. 5.7, comparing the average tensile strengths to the nondestructive strength parameter (square root of Young's modulus divided by the sonic attenuation). There is quite a bit of scatter in the results, but it is clear that this parameter can be used to ensure a minimum strength in graphite components with a fairly high confidence level.

The statistical analysis of these results is a two-factor problem that is complicated by the possible interaction between billets and positions. The problem is under study to demonstrate the relevance of estimating the strength of billets by sampling from a single position in the billet. This will also yield the relevance of estimating the strength of other billets from the same lot of material.

Fig. 5.7. The tensile strength of grades 2020 and IG11 compared to the nondestructive strength parameter $(E/\alpha)^{0.5}$.

5.1.4 Mechanical Property Testing - C. R. Kennedy

Tensile testing of Grade S2020 is in progress with results for billets 2 and 3 completed for comparison with the brittle-ring test results. A demonstration of the homogeneity in variances is one of the major goals in this study. Unless the strength variance in these billets can be shown to have homogeneity, the design strength must be lowered considerably due to the uncertainty in the strength distribution. The NDE test results indicated that homogeneity may not exist within billets or between billets; however, the sample volumes were fairly large. Thus, in the destructive test series, the sample volumes were as small as practical to obtain statistically relevant results. Testing a variety of specimens with stress-volumes that vary over several orders of magnitude was used to obtain an estimate of the variance without having to test a very large number of specimens with the same stress volume. This approach permits good estimates of the variance with fewer specimens and reduces the overall sample volume. It is very important in the evaluation of variance that the

sampled volume be homogeneous with the same statistical characteristics or comparisons cannot be made. Billets 2, 3, and 19 were chosen because of indicated variability from the NDE test results.

The destructive test results are shown for billets 2, 3, and 19 in Figs. 5.8 through 5.10 where the average strength (each data point represents from 10 to 30 tests) for each specimen size is plotted against the specimen stress volume. In each case the specimens taken from the end of the billet had a lower mean strength than the specimens taken from the middle of the billet; however, the variance (as indicated by the slope of the fracture strength-stress volume curve) for each group of specimens is almost identical. This result is a very clear indication that even though the mean strength in the billets may vary over 10%, the variance or coefficient of variation is constant. This approach will permit a rational estimation of design strength with an appropriate consideration of the stress volume of the component part compared with test coupons used to measure the strength of the material.

ORNL-DWG 87-17161

Fig. 5.8. Effect of stress volume on fracture stress of S2020 graphite, billet 2.

Fig. 5.9. Effect of stress volume on fracture stress of S2020 graphite, billet 3.

Fig. 5.10. Effect of stress volume on fracture stress of S2020 graphite, billet 19.

Fracture-toughness specimens from each location in each billet from two grades (S2020 and IG11) were also evaluated in this last period. Five specimens each, orientated with the fracture surface normal to the axial and to the radial direction, at all locations (top, edge, and middle) were tested and the results compared. The results of fracture-toughness testing are given in Tables 5.2 and 5.3. The fracture toughness in both directions and in each location of all the billets were exactly the same as has been

Table 5.2. Grade 2020 fracture-toughness test results
(MPa $\cdot\sqrt{m}$)

Billet No.	Section	Average	Std. dev. ^a	S _a ^b	S _e ^c	F ^{d,e}	χ^2 ^{f,g}
2B	Top	1.232	0.066 (10)	0.00	0.046	0.494 (0.613)	7.000 (0.030)
	Edge	1.251	0.030 (10)				
	Middle	1.249	0.033 (10)				
	All	1.244	0.045 (30)				
3B	Top	1.255	0.030 (10)	0.00	0.035	0.547 (0.586)	1.360 (0.507)
	Edge	1.270	0.045 (7)				
	Middle	1.253	0.033 (10)				
	All	1.258	0.035 (27)				
19C	Top	1.268	0.036 (10)	0.011	0.034	4.182 (0.026)	0.162 (0.922)
	Edge	1.229	0.032 (10)				
	Middle	1.230	0.035 (10)				
	All	1.234	0.038 (30)				
All		1.248	0.040 (87)	0.003	0.039	1.441 (0.193)	11.331 (0.183)

^aNumbers in parentheses are the number of specimens tested.

^bBetween billet standard deviation.

^cWithin billet standard deviation.

^dVariance ratio.

^eValue in parentheses is probability $F >$ value calculated for the group.

^fBartlett's test of homogeneity of variance.

^gValue in parentheses is probability that Chi-square $>$ value calculated for the group.

Table 5.3. Grade IG11 fracture-toughness test results
(MPa $\cdot\sqrt{m}$)

Billet No.	Section	Average	Std. dev. ^a	S _a ^b	S _e ^c	F ^{d,e}	x ^{2f,g}
8C	Top	1.401	0.053 (10)				
	Edge	1.416	0.040 (10)				
	Middle	1.384	0.061 (10)				
	All	1.400	0.052 (30)	0.000	0.052	0.925 (0.409)	1.489 (0.475)
14A	Top	1.369	0.063 (10)				
	Edge	1.420	0.041 (10)				
	Middle	1.425	0.063 (10)				
	All	1.405	0.061 (30)	0.014	0.057	2.964 (0.069)	1.833 (0.380)
16A	Top	1.381	0.029 (10)				
	Edge	1.371	0.061 (10)				
	Middle	1.333	0.040 (10)				
	All	1.365	0.054 (30)	0.013	0.052	2.722 (0.084)	8.848 (0.012)
All		1.390	0.058 (90)	0.007	0.054	2.892 (0.007)	12.490 (0.131)

^aNumbers in parentheses are the number of specimens tested.

^bBetween billet standard deviation.

^cWithin billet standard deviation.

^dVariance ratio.

^eValue in parentheses is probability F > value calculated for the group.

^fBartlett's test of homogeneity of variance.

^gValue in parentheses is probability that Chi-square > value calculated for the group.

observed in the past. These results again give strong evidence that the variability in fracture strength is a result of the flaw size distributions in the material. Differences in the mean value of strength are results of defect size differences, in particular where the flaw size has a preferred orientation.

5.2 GRAPHITE MATERIALS - CORE (WBS 1602.2)

5.2.1 Fracture Mechanics of Graphite - S.-J. Chang, C. R. Luttrell, and J. R. Drake

The objective of this study is to establish a fracture criterion for nuclear graphite based on the general principles of fracture mechanics. In view of the microstructural differences between graphite and metals where the fracture mechanics finds its most applications, the development of the fracture mechanics for graphite will be concentrated on resolving the following two properties: (1) the strain softening effect in regions near the crack tip and (2) the probabilistic nature of failure as a result of brittle fracture. The fracture criterion will contain both ingredients in its final form.

The study consists of several experiments that will be discussed in the future. At present, we shall report mainly on the development of the method of analysis and its application to a possible diametric compression specimen.

5.2.1.1 Elastic-plastic, finite-element program

A special purpose elastic-plastic, finite-element program has been developed in its preliminary form. In its present form, the program can be used to analyze any two-dimensional or axisymmetric structures and to calculate the mode I stress distribution for general elastic-plastic solids. The program has been made compatible with the commercial PATRAN preprocessor and postprocessor. The finite-element grid can be prepared by invoking a few simple commands on screen interactively. With this preprocessor, the laborious work for generating and labeling the grids can be avoided. After calculation, the postprocessor can readily plot the numerical results.

As an example, Fig. 5.11 shows a diametric compression specimen that has been analyzed using the finite-element grid. Results of an elastic-plastic disc of this specimen are shown in Figs. 5.12 and 5.13.

5.2.1.2 Probabilistic considerations

The goal of this study is to implement an appropriate probabilistic failure function into the finite-element program and to calculate the

Fig. 5.11. Finite-element grid generated from PATRAN preprocessor and the enlarged displacement field. The diametric specimen is a circular shape with a central cut and is subjected to a concentrated load at an axis of symmetry passing through the crack plane.

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-0.498	= A
-1.40	= B
-2.31	= C
-3.21	= D
-4.12	= E
-5.02	= F
-5.92	= G
-6.83	= H
-7.73	= I
-8.64	= J
-9.54	= K
-10.4	= L
-11.4	= M
-12.3	= N
-13.2	= O

Fig. 5.12. σ_{xx} stress contours generated from PATRAN postprocessor.

ORNL-DWG 89-2120

Fig. 5.13. σ_{yy} stress contours generated from PATRAN postprocessor.

total probability of failure for a structure that is subjected to a nonuniform stress distribution. The fracture criterion can be interpreted as the total probability of failure of the structure. At present, we are in the process of implementing a general probability function into the finite-element code. As a separate yet related effort, we are also working on an asymptotic formulation at the crack-tip region with the probabilistic failure effect.

5.2.2 Stress Volume Effects Studies on H451 Graphite - G. T. Yahr and R. L. Battiste

Twenty-six large volume American Society for Testing and Materials (ASTM) 25.4-mm-diam (1-in.) tensile specimens³ from a single billet of H451 graphite were tested. Results are summarized in Table 5.4. In addition to the failure data given in Table 5.4, complete stress-strain curves and lateral strain vs axial strain curves were obtained. Although the mean strength and strain to failure were in the same range as has been reported by other investigators, these specimens exhibited a very high variability in strength and strain to failure. The coefficient of variation was >25% for both. Small tensile specimens were machined from the grip area of the large specimens and will be tested to determine volume size effects in this graphite.

The fracture stress and strain of each specimen is plotted in Fig. 5.14. The fracture points of specimens 3 and 18 do not appear to belong with the other data. All of the other specimens had approximately the same ratio of stress-to-strain at failure. For no apparent reason, these two specimens had a lower ratio of stress-to-strain at failure.

The histogram of strength values in Fig. 5.15 indicates a bimodal strength distribution with most of the data in the lower strength group.

The specimens tended to fail at or near the ends of the straight gage section. The fracture locations are shown in a bar graph in Fig. 5.16. The location of the straight gage section is indicated in the figure and is divided into four equal sections by the grid lines. Only seven specimens failed in the central half of the gage section. Six specimens failed outside the gage section, and four failed right at the end of the gage section. There was no apparent undercut at the ends of the gage sections. These data suggest that the ASTM 25.4-mm (1-in.) tensile specimen design

Table 5.4. Data from 25.4-mm-diam (1-in.) tensile specimens
from a single billet of H451 graphite

Slab	Specimen number	Fracture load (lb)	Axial strain (%)	Failure length (in.)	Stress (psi)	Stress (MPa)	Elastic modulus (1,000,000 psi)	Elastic modulus (1,000 MPa)
23	1	2160	0.266	5.2	2750	19.0	1.54	10.6
23	2	1470	0.188	5.1	1871	12.9	1.48	10.2
23	3	1415	0.284	5.3	1801	12.4	1.43	9.8
23	4	1425	0.197	9.3	1814	12.5	1.38	9.5
23	5	1595	0.187	8.6	2030	14.0	1.33	9.2
23	6	2385	0.294	9.2	3036	20.9	1.54	10.6
34	7	2515	0.320	9.2	3202	22.1	1.54	10.6
34	8	1440	0.183	8.0	1833	12.6	1.41	9.7
34	9	1125	0.135	6.4	1432	9.9	1.46	10.1
34	10	1400	0.198	8.8	1782	12.3	1.47	10.1
34	11	1330	0.163	5.3	1693	11.7	1.32	9.1
34	12	1870	0.255	5.5	2381	16.4	1.44	10.0
34	13	2490	0.325	5.1	3170	21.9	1.50	10.4
45	14	2345	0.291	8.6	2985	20.6	1.53	10.6
45	15	1675	0.218	5.2	2132	14.7	1.44	9.9
45	16	1195	0.145	9.4	1521	10.5	1.50	10.4
45	17	1190	0.160	6.2	1515	10.4	1.46	10.1
45	18	1275	0.166	8.5	1623	11.2	1.44	9.9
45	19	2125	0.288	5.1	2705	18.7	1.54	10.6
45	20	2475	0.310	9.0	3151	21.7	1.53	10.6
56	21	2240	0.268	8.9	2852	19.7	1.52	10.5
56	22	1585	0.198	7.3	2018	13.9	1.50	10.3
56	23	1275	0.188	8.0	1623	11.2	1.39	9.6
56	24	1410	0.179	7.6	1795	12.4	1.31	9.1
56	25	1749	0.223	5.0	2226	15.4	1.48	10.2
56	26	2440	0.311	6.5	3106	21.4	1.57	10.8
Mean		1754	0.2285		2233	15.40	1.463	10.1
S.D.		481	0.0602		612	4.23	0.072	0.5
Min.		1125	0.135		1432	9.9	1.31	9.1
Max.		2515	0.325		3202	22.1	1.57	10.8

ORNL-DWG 89-2121

Fig. 5.14. Fracture stress and strain values from 25.4-mm-diam (1-in.) H451 tensile specimens.

ORNL-DWG 89-2122

Fig. 5.15. Histogram of 25.4-mm-diam (1-in.) H451 tensile strength data.

ORNL-DWG 89-2123

Fig. 5.16. Fracture locations in the 25.4-mm-diam (1-in.) tensile specimens.

should be reexamined to ensure that the transition to the gage section is gradual enough.

5.2.3 Tensile Testing - C. R. Kennedy

An automated tensile testing system has been added to our Materials Testing Services testing machine that will aid considerably in the large tensile testing program in progress. The tensile testing procedures and calibration are documented with the stress-strain data for each individual test and stored on a micro disc for future reference or confirmation. Each tensile test is analyzed automatically to give the fracture stress and strain and the second-order coefficients fitting the stress-strain data by a regression analysis; this data is recorded separately on a second micro disc that can be used to perform a statistical analysis on the fracture

data. A separate analog recording of each test is also made to monitor the test progress and to furnish a redundant set of data for insurance.

Grips, as specified by ASTM C749, have been made to test 6.35- and 9.53-mm-diam (0.250- and 0.375-in.) tensile specimens. The bending moment is virtually eliminated in these grips by the addition of a four-link welded chain on both ends of the grips. A test series of grades H451 and PGX graphite with both 6.35- and 9.53-mm-diam (0.250- and 0.375-in.) has been run to check the grips and the data acquisition system. An equal number of companion specimens have been sent to the Y-12 Plant for comparative testing.

The results of the preliminary testing are given in Table 5.5, comparing the average strengths of 6.35- and 9.53-mm-diam (0.250- and 0.375-in.) specimens of both H451 and PGX graphite. Several characteristics of the data are quickly observed. The variability of the extruded grade H451 is considerably higher than that of the molded grade PGX, although the mean strength of PGX is much lower. The variability in extruded grades is generally found to be greater than in molded grades; however, an additional problem is the lack of homogeneity in blocks of extruded graphites. The sample that is tested here is a random sample across the billet and therefore does not truly represent a good indication of the variability expected from a single sample volume in a billet. The problem of handling multiple distributions truly describing the variability in material for design purposes is still under study; however, it is clear

Table 5.5. Results of tensile testing representative samples of grades H451 and PGX graphite

Grade	Specimen diameter [mm (in.)]	Number	Fracture strength ^a		Fracture strain (%)
			(MPa)	(psi)	
H451	6.35 (0.250)	18	17.66 [3.50]	2562 [507]	0.247 (0.050)
	9.53 (0.375)	19	17.91 [3.45]	2597 [500]	0.263 (0.058)
PGX	6.35 (0.250)	19	9.99 [1.15]	1449 [167]	0.195 (0.035)
	9.53 (0.375)	20	11.11 [0.88]	1611 [128]	0.226 (0.024)

^aNumbers in brackets are 1 standard deviation.

that unless the variability can be better defined, the design stresses in extruded grades like H451 will be overly conservative.

The effects of stress volume and defect size are also demonstrated in these data. Generally, as the specimen stress volume is reduced, the probability of finding a critical flaw is reduced and the average strength is higher. The difference in stress volume between the 6.35- and 9.53-mm-diam (0.250- and 0.375-in.) specimens is probably too small to demonstrate the increased strength with these few specimens. However, the reduced strength of the 6.35-mm-diam (0.250-in.) PGX graphite is significant and does demonstrate the effect of minimum cross section compared with the critical defect size in the graphite. Grade PGX graphite, with an average strength of two-thirds that of grade H451, would have a critical defect size about two and one-half that of H451 if the fracture energies are similar. Grade H451 has a critical defect size of about 1 mm, giving grade PGX a critical defect size of 2.5 mm. It is understandable that grade PGX 6.35-mm (0.250-in.) specimens would not have a stable stress distribution when the flaw size is more than one-third of the diameter and the strength is reduced, while grade H451 is not affected. This clearly illustrates the need to demonstrate that tensile data is not adversely affected by specimen size.

5.2.4 Comparison Tensile Tests - R. L. Battiste and G. T. Yahr

As noted in the previous section, companion ASTM C749 tensile specimens of grades H451 and PGX graphite with both 6.35- and 9.53-mm-diam (0.250- and 0.375-in.) were tested in two different laboratories. The same ASTM C749 grips were used in both laboratories. The results that we obtained are given in Table 5.6. Comparison with Table 5.5 shows excellent agreement between the two laboratories.

5.2.5 Petten Creep Results - C. R. Kennedy

Recent creep results received from Petten on the irradiation creep of graphite grades H451, ATR-2E, and ASR-1RS at 900°C have been very informative in resolving the differences between the Oak Ridge Research Reactor (ORR) data and the High Flux Reactor (HFR) data. The apparent differences between our ORR compression creep data and the HFR tensile creep data

Table 5.6. Second laboratory results of tensile tests of grades H451 and PGX graphite

Grade	Specimen diameter [mm (in.)]	Number	Fracture strength ^a	
			(MPa)	(psi)
H451	6.35 (0.250)	20	17.98 [3.85]	2608 [560]
	9.53 (0.375)	19	18.71 [3.08]	2713 [446]
PGX	6.35 (0.250)	19	10.49 [1.08]	1521 [156]
	9.53 (0.375)	19	10.66 [1.15]	1545 [166]

^aNumbers in brackets are 1 standard deviation.

appear to have been a result of data scatter that was resolved by continuing the irradiation experiment to higher fluences. The high-fluence Petten results are extremely significant in clarifying the overall creep behavior and the effect of irradiation temperature. These new data indicate that present models of creep will have to be modified as suggested by M. R. Cundy to include the effects of irradiation strengthening and softening. There does not appear to be a constant second stage of creep but a slowly decreasing creep rate until the graphite begins to degrade by loss of density and structural integrity. This weakening then results in an accelerating creep rate. These changes are probably not a result of creep deformation but an actual changing of the material's resistance to creep as a result of the microstructural changes caused by irradiation damage. This additional information, combined with earlier Petten and Oak Ridge National Laboratory (ORNL) data, will permit a generalized description of creep to be made from 300 to 900°C up to fluences approaching the useful life of the graphite. The effect of this reexamination with the current data on present stress calculations will be small; however, the calculated stresses will be larger. Because of the larger data base now available, a significantly increased confidence now exists in the quantification of irradiation creep behavior of graphite under steady loads. A major question of the effect of stress reversal to be expected in service must still be resolved.

5.2.6 Irradiation of H451 Graphite at 450°C - C. R. Kennedy

In the HFIR United Nuclear has sponsored an irradiation experiment that contains graphite grades TSX (the present grade in the N-Reactor), S0818, TS1507, and H451 (the last three are possible replacement grades). The temperature of the irradiation was 450°C, and the maximum fluence was 3×10^{22} neutrons/cm² ($E > 50$ keV). This experiment has yielded the only low-temperature data for H451 and similar graphites to meaningful fluences. This experiment will aid considerably in describing physical property effects and dimensional change results with increasing confidence levels to the lower temperatures of more interest to the MHTGR.

The dimensional changes at 450°C for H451 are given in Fig. 5.17, which compares the axial and radial directions with the volume changes in the graphite. The indication is that, as expected, the life expectancy of H451 at 450°C is a very high fluence exceeding 4×10^{22} neutrons/cm² ($E > 50$ keV). A comparison of the radial dimensional changes of the four graphites is given in Fig. 5.18 and the axial dimensional changes in Fig. 5.19. The growth instability of the more anisotropic grade TSX (similar to H327) compared with the isotropic grades is quite evident. Also note that the behavior of grade TS1507 is almost indistinguishable from that of H451, while grade S0818 yields a more isotropic behavior with less dimensional change, but also has a shorter useful life because of a more rapid volume expansion. The microstructures of TS1507 and H451 are similar, while S0818 is evidently made up of large, spherical-shaped particles of fluid coke. The similarity in structure and behavior of H451 and TS1507 does imply that a competitive material to H451 could be quickly qualified. A few carefully selected experiments would be all that is needed to demonstrate that the H451 data base is equally applicable to both grades.

5.3 GRAPHITE TECHNOLOGY (WBS 1602.3)

5.3.1 A Bayesian Approach to the Fort St. Vrain Fractured Fuel Block Occurrence, Part 8 - W. P. Eatherly

We now conclude our study of this problem and in so doing obtain at last the solution. The earlier proposed approach of coupling $n_{11} + n_2 = 2$

ORNL-DWG 87-10949

Fig. 5.17. The dimensional changes of H451 graphite at 450°C.

ORNL-DWG 87-10965

Fig. 5.18. A comparison of four graphites at 450°C.

Fig. 5.19. A comparison of four graphites at 450°C.

as a necessary condition for the calculations of the relative probabilities of Cases I and III against II is erroneous. Our concern is rather that the value $n = 2$ should occur at all against repeated values $n = 0$.

To anticipate the results, we will find at finite risk that we can reject first Case I then Case III as statistically unreasonable, albeit at much more information than we now possess. To do so we will use that branch of statistics called order theory. For discrete distributions, this reduces to a particularly simple argument.

First, arrange the observations x_i in order, that is, for m values of x_i ,

$$x_1 \leq x_2 \leq x_3 \leq \dots \leq x_m$$

Choosing any value of x , we partition this sequence at some value $i = j$ such that

$$x_1 \leq x_2 \leq \dots \leq x_j \leq x$$

$$x_m \geq x_{m-1} \geq \dots \geq x_{j+1} > x$$

with the lower partition containing x , the upper excluding x . The probability that exactly j values of x_i are $\leq x$ is

$$[F(x)]^j ,$$

and that $m - j$ values exceed x are

$$[1 - F(x)]^{m-j} ,$$

where F is the cumulative distribution function. Allowing for permutations (binomial coefficient) and denoting the probability j values of x_i , given a sampling of size m , will lie at or below x by

$$Pr(j \ x_i \leq x) ,$$

we have

$$Pr(j \ x_i \leq x) = \binom{m}{j} [F(x)]^j [1 - F(x)]^{m-j} . \quad (1)$$

Taking this equation one step further, the probability j or fewer x_i are $\leq x$ becomes

$$Pr(\leq j \ x_i \leq x) = \sum_{k=0}^j \binom{m}{k} [F(x)]^k [1 - F(x)]^{m-k} . \quad (2)$$

Clearly, if $j = m$, then this probability becomes unity, that is, we are certain that over the range $0 \leq j \leq m$ all possible unordered sequences will occur for any and every value of x permitted. Equations (1) and (2) are no more than the binomial distribution we have seen so frequently in this analysis, first implicitly in Part 1.⁴

To produce numerical results from Eq. (1) or (2), we take the Poisson distribution

$$F(x) = \sum_{n=0}^x f(x) = \sum_{n=0}^x \frac{1}{n!} \mu^n e^{-\mu} \quad (3)$$

introduced in Part 4,⁵ and for the true mean μ we will take the Bayesian most probable, or equivalently, the Maximum Likelihood estimator $\hat{\mu}$, thus identicalizing the estimator with the underlying parameter. With this caveat, we provide in Table 5.7 the values for Eq. (1) with $m = 4$ and

Table 5.7. Probabilities that exactly j values of x_i in the observed set $\{x_i\}$ ($i = 1, 2, \dots, m$) are equal to or less than the value x , at $m = 4$ and 18 for Cases I and III, that is, an ordered sequence of the form $\{0, \dots, 0, 2\}$

Calculations are for a mean value μ certain and given by $\hat{\mu} = 2/m$

j	$x = 0$	$x \leq 1$	$x \leq 2$	$x \leq 3$	$x \leq 4$
$m = 4, \mu = 1/2$					
4	0.135	0.685	0.944	0.993	0.999
3	0.351	0.272	0.055	0.007	0.001
2	0.342	0.040	0.001	0.000	0.000
1	0.148	0.003	0.000		
0	0.024	0.000			
$m = 18, \mu = 1/9$					
18	0.135	0.902	0.996	1.000	
17	0.286	0.094	0.004	0.000	
16	0.286	0.005	0.000		
15	0.179	0.000			
14	0.080				
13	0.026				
12	0.007				
11	0.001				
10	0.000				

$m = 18$ for Cases I and III ($\hat{\mu} = 2/m$) and in Table 5.8 for Case II ($\mu = \hat{\mu} = 1/m$).

Note the structure of these tables carefully, because both the logic and nomenclature are somewhat formidable on first glance. Take the columns of Table 5.7 for $m = 4$ as an example. The entry in each case is a specific value of Eq. (1), namely,

$$Pr(j \text{ } x_i \leq x) = \binom{4}{j} [F(x)]^j [1 - F(x)]^{4-j}, \quad (4)$$

and scanning down the column, we simply see the values for all possible j 's, that is, $j = 4, 3, 2, 1$, and 0. From Eq. (2), the sum of each column must be unity. Missing entries are zero or rather values of probability $< 1 \times 10^{-3}$ when rounded. Reading across the table for $j = 4$, the values

Table 5.8. Probabilities that exactly j values of x_i are equal to or less than the value x at $m = 4$ and 18 for Case II, that is, an ordered sequence of the form $\{0, \dots, 0, 1\}$

j	$x = 0$	$x \leq 1$	$x \leq 2$	$x \leq 3$
$m = 4, \mu = 1/4$				
4	0.368	0.898	0.991	1.000
3	0.418	0.098	0.009	0.000
2	0.178	0.004	0.000	
1	0.034	0.000		
0	0.002			
$m = 18, \mu = 1/18$				
18	0.368	0.974	1.000	
17	0.378	0.026	0.000	
16	0.184	0.000		
15	0.056			
14	0.012			
13	0.002			
12	0.000			

increase steadily to unity, at least in the sense that any number > 0.9995 is rounded to 1.000. For this first row, all possible values of the x_i are constrained by the fact $j = 4$, that constrain being the column heading. Thus we must find, on repeated measurements in sets of four, 13.5% of the time all $x_i = 0$, 68.5% of the time all x_i will be 1 or less, and so forth, until we reach the limit that "always" all x_i will be 5 or less. In other words, any sequence containing an observed value of 6 or more is vanishingly probable (i.e., "totally" impossible).

In the second row, we find the probabilities decreasing as we go across the table. But now the constraint on one value of x has been lifted. The first entry, 0.351, states that the probability for the permuted sequences $(0, 0, 0, n)$, n being any number > 0 . The second entry pertains to the permutations of (x, x, x, n) with $x \leq 1$ but n any number > 1 . Clearly, with $\mu = 1/2$, the probability of this second sequence is less than the first. The final entry for the permutations of (x, x, x, n) with $x \leq 5$ requires $n > 5$, and we already know from the first row a value

of $x \geq 6$ is improbably small. The last row simply states the probability of finding no values zero in a set of four observations is only 2.4%, finding no values < 1 is virtually improbable (0.01%), and finding no values < 2 is totally improbable (to the limit of our rounded values).

We summarize and emphasize the peculiar aspect of Table 5.7 for $m = 4$ (and the remainder of Tables 5.7 and 5.8), which appears strange because the first row $j = 4$ is inclusive; the second and following rows are exclusive. Each move to the right in the row for $j = 4$ includes the previous values to the left and must reach an ultimate value 1. Each move to the right for $j < 4$ excludes one or more possibilities for the x_i , and as these excluded values for x_i are forced upward by our choice for x , the probabilities must reach an ultimate value of zero.

We now turn to the result of looking at a number of tables such as Table 5.7 or 5.8 for a collection of selected m -values spanning $2 \leq m \leq 48$. We confine ourselves to the inclusive case $j = m$, that is, the first row of the two example Tables 5.7 and 5.8. We are, of course, discarding much information for simplicity, but the cardinal results are contained in this first row. Table 5.9 presents the results for Cases I and III (sequences containing all zeros except for one value two for the x_i) and Table 5.10

Table 5.9. Restrictions on the occurrence of values of all x_i as a function of the sample size m for Cases I and III, for which the observed sequence of sample observations includes only zeros except for a singular value two

Here, $\mu = \hat{\mu} = 2/m$ by hypothesis

m	All $x_i = 0$	All $x_i \leq 1$	All $x_i \leq 2$	All $x_i \leq 3$
2	0.135	0.541	0.846	0.962
3	0.135	0.627	0.912	0.986
4	0.135	0.685	0.944	0.993
5	0.135	0.728	0.961	0.996
6	0.135	0.760	0.971	0.998
12	0.135	0.861	0.992	1.000
18	0.135	0.902	0.996	
24	0.135	0.924	0.998	
36	0.135	0.948	0.999	
48	0.135	0.961	1.000	

Table 5.10. Restrictions on the occurrence of values of all x_i as in Table 5.9 but for Case II: observed sequence of the form all zeros except for the singular value one

Here, $\mu = \hat{\mu} = 1/m$ by hypothesis

m	All $x_i = 0$	All $x_i \leq 1$	All $x_i \leq 2$	All $x_i \leq 3$
2	0.368	0.828	0.971	0.997
3	0.368	0.872	0.986	0.999
4	0.368	0.898	0.991	1.000
5	0.368	0.915	0.994	
6	0.368	0.928	0.996	
12	0.368	0.961	0.999	
18	0.368	0.974	1.000	
24	0.368	0.980		
36	0.368	0.986		
48	0.368	0.990		

for Case II (sequences containing all zeros except for a singular value one). We consider in detail only Table 5.9 with similar conclusions following from Table 5.10.

In Table 5.9, the probability that we observe all $x_i = 0$ is insensitive to the value of m , a consequence of the fact that we update $\hat{\mu}$ for each additional value of x_i , specifically assumed to be 0 for $i > 2$. Our interest, therefore, is confined to the columns for greater upper limits on x_i . Consider, as we did above, the values for $m = 4$. If we were to observe 100 reactors, we would now estimate that 99.3% of them would exhibit < 4 fractured billets in any one of the four observations per reactor, a statement which can be rephrased as "we are 99.3% confident < 4 fractured billets will be observed in any reload segment when four observations are made on that type of segment at a future date." The limit to four observations exists because we are using moving estimates as the number of observations increases. Also at $m = 4$, we can say we are 94.4% confident that all future observations will be ≤ 2 .

Scanning down the table, we see at $m = 18$ that we are about 90% confident all values will be ≤ 1 . More importantly, we can throw out the already observed value 2 as nonstatistical at 10% risk. At $m = 36$, we can

throw the value out at about 5% risk. If we are willing to make either decision, then Cases I and III are deemed statistically improbable and only Case II (Table 5.10) survives as tenable. On the basis of the simplified analysis based on the assumption $\mu = \hat{\mu}$, we cannot throw out Case II in any event, but Case II also gives us very small probabilities for fractured blocks, only a 2% probability of finding a broken block in the entire core in its equilibrium configuration after 18 observations.

This completes our purely statistical analysis of the observation that two fuel blocks were observed to be fractured on removing the second reload segment. At the present time we are better than 95% confident in any case that no more than three fractured blocks will ever be observed for a segment at time τ_1 (for which $m = 3$) or at time τ_2 ($m = 2$). If no blocks are observed to be fractured on removing the fourth reload segment, our confidence in this statement exceeds 99%. If we continue to observe no fractured blocks, by reload segment 18 we can conclude that the observation of two fractured blocks in one time interval is so unlikely we can reject the observation as arising from a nonstatistical event. Our confidence in this increases substantially as further observations of no fractured blocks occur. Thus, our objectives in undertaking this study have at least in part been met.

The use of the Bayesian approach has provided us with considerably more information than a Maximum Likelihood analysis, even though we accept the most probable value of μ (i.e., $\hat{\mu}$) as the estimator. We have at each stage of observation the distribution function for μ and, if we desired, can thus make various estimates of the reliability of $\hat{\mu}$. We have produced 95% coverage limits as an example. It is also clear that the Bayesian approach brings out emphatically the relative slowness at which the confidence levels increase after the first ten or so observations have been made. Finally, although we can reject eventually the occurrence of two blocks breaking within the same time interval, the case where one broke in the first interval and a second in the second interval will not be rejectable on statistical grounds. Other arguments, such as the state of the reactor in the second time interval, will have to be used to totally reject the observation of any fracturing of fuel blocks.

5.4 REFERENCES

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6. METALS (WBS 1603)

P. L. Rittenhouse

6.1 MATERIALS ENGINEERING (WBS 1603.01) - P. L. Rittenhouse and J. P. Strizak

Combustion Engineering (CE), General Atomics (GA), and Oak Ridge National Laboratory (ORNL) collaborated in the preparation of the *Metals Technology Development Plan*¹ issued in March. The plan addresses requirements and assumptions related to metallic component designs and the technology programs needed to satisfy data needs identified by designers in their requirements-assumptions analyses. The work described in Sects. 6.2 and 6.3 responds directly to the plan.

6.2 REACTOR VESSEL MATERIALS (WBS 1603.03)

6.2.1 Elevated Temperature Properties of SA533 Grade B, Class 1 Pressure Vessel Steel - H. E. McCoy

Section III of the *ASME Boiler and Pressure Vessel Code* does not provide allowable design stress intensities for SA533B steel at temperatures above 371°C (700°F). However, the temperature of the modular high-temperature gas-cooled reactor (MHTGR) reactor vessel is expected to exceed 371°C during Service Level C and D events. The maximum temperature predicted is 473°C (884°F), but with a time above 371°C (700°F) of no more than 400 h. Despite these relatively undemanding conditions, application of the material under ASME Code rules will require that data be available to demonstrate negligible plastic deformation during these events and to permit extension of the Sect. III allowables to temperatures above 371°C (700°F).

The first lot of material was procured. The material contained a submerged-arc weld and the associated heat-affected zone. The welding and heat treating were performed by CE of Chattanooga, Tennessee. Duplicate tensile tests have been run at two strain rates over the temperature range

of 25 to 538°C. The strength and ductility of the base metal, weld metal, and heat-affected zone are quite similar. Creep tests were run on this same material at 371, 427, 482, and 538°C (700, 800, 900, and 1000°F). Tests were run at several stresses, and the data were correlated parametrically.

One further heat treatment was performed at ORNL to place the material in the lowest strength condition allowed by the Code. Tempering for 30 h reduced the room-temperature ultimate tensile strength to about 572 MPa (83 ksi), near the minimum value of 552 MPa (80 ksi) allowed by the Code. Stress-temperature correlations were developed for the material in this heat-treatment condition based on 0.1 and 1.0% strain in 1000 h.

Because of the known variation of mechanical properties between different heats of steel, test data will be collected from at least three heats of material. These data will be used to determine the allowable stress limits over the temperature range of 371 to 538°C (700 to 1000°F).

6.2.2 Irradiation Effects Testing - R. K. Nanstad

The effects of neutron irradiation on the steel reactor vessel for the MHTGR are not well known, primarily because the operating temperatures are low [121 to 210°C (250 to 410°F)] compared with that for commercial light-water reactors (LWRs) [~288°C (550°F)]. Design Data Need (DDN) M.11.06.01, "Nil-Ductility Transition Temperature Shift for Reactor Vessel Materials Irradiated at Low Temperatures,"² describes the need for a program to determine irradiation effects for the SA533 Grade B Class 1 plate, SA508 Class 3 forging, and welds used for the vessel shell, vessel closure head, and vessel flange. Table 6.1 summarizes the operating conditions given in the DDN.

The primary data base available for irradiation effects on pressure vessel steels comes from LWR technology. As discussed in the DDN, the LWR data base used in the development of Nuclear Regulatory Commission (NRC) *Regulatory Guide 1.99* includes irradiation temperatures from 274 to 302°C (525 to 575°F) under fast neutron (>1-MeV) fluxes in the range 10^{10} to 10^{13} neutrons/cm²·s. Moreover, the DDN also points out that the spectrum for fast (>1-MeV), epithermal (0.4-eV < E < 1-MeV), and thermal (<0.4-eV) neutrons in LWRs is about 10, 25, and 65%, respectively; whereas, the

Table 6.1. Summary of operating conditions for MHTGR reactor vessel components

Component	Materials ^a	Metal temperature [°C (°F)]	Design equivalent fast fluence (neutrons/cm ²)	Corrected ^b equivalent fast fluence (neutrons/cm ²)
Beltline shell	Plate, weld	201-210 (394-410)	2.8×10^{16}	2.8×10^{17}
Upper flange	Forging, weld	173 (343)	6.2×10^{16}	1.2×10^{18}
Lower flange	Forging, weld	177 (350)	3.2×10^{16}	6.4×10^{17}
Closure head	Plate, weld	155-169 (311-336)	3.7×10^{16}	7.4×10^{17}
Closure head	Forging, weld	123-131 (254-267)	7.3×10^{15}	1.4×10^{17}

^aExistence of welds in each location was assumed.

^bCorrected for uncertainties using factors of 10 for the beltline and 20 for the head.

high-temperature gas-cooled reactor (HTGR) spectrum has a much greater proportion of epithermal neutrons as shown in Table 6.2. The DDN accounts for the effects of neutrons with energies below 1.0 MeV through the use of weighting factors³ (Table 6.2) to calculate an equivalent fast neutron flux for each vessel component. There is substantial uncertainty, however, that such a procedure is adequate, especially at low irradiation temperatures. Additionally, the recent observations from surveillance tests on the High-Flux Isotope Reactor (HFIR) at Oak Ridge⁴ have identified a possible irradiation rate effect resulting from long-term exposure to low fluxes (10^8 to 10^9 neutrons/cm²·s) at low temperatures [$\sim 49^\circ\text{C}$ (120°F)]. Possible effects of lower energy neutrons under those conditions have been questioned but are unknown.⁵

Table 6.2 Comparison of weighting factors and neutron spectra for HTGR and LWR

Neutron energies	Weighting factor	HTGR spectrum (%)		LWR spectrum (%)
		Upper flange	Closure head	
$E > 1.0 \text{ MeV}$	1.0	2	8	10
$0.1 \text{ MeV} < E < 1.0 \text{ MeV}$	0.5	21	30	25
$0.4 \text{ eV} < E < 0.1 \text{ MeV}$	0.0025	67	62	
$E < 0.4 \text{ eV}$	0.07	10	0.2	65

Information regarding the effects of irradiations much below 288°C (550°F) is sparse and most data were obtained with "older" steels like SA212 Grade B. Figure 6.1⁶ shows some of the results often quoted regarding effects of irradiation temperature. For the equivalent fast fluences given in the DDN for the MHTGR vessel (7.3×10^{15} to 6.2×10^{16} neutrons/cm²), the effects would be expected to be very low. Using the uncertainties given in an earlier DDN (factors of 10 for the beltline and 20 for the vessel head), the fluences range from about 1.4×10^{17} to 1.2×10^{18} neutrons/cm². For comparison, the SA212 Grade B steel from the

Fig. 6.1. Effect of irradiation temperature on CVN transition temperature increase of ASTM A212 Grade B steel, heat A-2056. Source: R. G. Berrgren, W. J. Stelzman, and T. N. Jones, "Radiation Effects on Pressure Vessel Steels," *Solid State Division Progress Report for Period Ending January 1967*, ORNL-4097 (April 1967).

surveillance program in the HFIR vessel showed a transition temperature shift of 42°C (75°F) at a fast fluence of $\sim 1.3 \times 10^{17}$ neutrons/cm² at 49°C (120°F). That compares to a shift of 56°C (100°F) for the same steel irradiated in a test reactor to a fast fluence of 2.5×10^{18} neutrons/cm². Thus, including uncertainties for the MHTGR, the design fluences and temperatures are in the range where embrittlement may be of concern.

Because the MHTGR incorporates a steel pressure vessel, the requirements of "Title 10," *Code of Federal Regulations*, "Part 50" (10 CFR 50), Appendix G⁷ will likely be applied by the NRC. The 10 CFR 50 requirements

refer to the ASME Code, Sect. III, Div. 1,⁸ for determination of the "reference temperature," RT_{NDT} . It also refers to an "adjusted reference temperature" determined by adding to RT_{NDT} the temperature shift, measured at the 41-J (30-ft·lb) level, in the average Charpy curve for the irradiated material relative to that for the unirradiated material.

Additionally, the determination of Charpy upper-shelf energy is required to demonstrate that each material has an upper-shelf energy level no less than 102-J (75-ft·lb) initially and maintains an upper-shelf energy of no less than 68-J (50-ft·lb) throughout the life of the vessel. These requirements apply to base materials, welds, and weld heat-affected zones.

Determination of the RT_{NDT} requires the use of drop-weight testing and Charpy V-notch (CVN) impact testing of material obtained from the product forms in the orientations as specified in the ASME Code. Drop-weight specimens are not, however, required for determining the adjusted reference temperature. Tensile specimens are necessary for material characterization and are also advisable in the irradiated condition to assess the changes in material strength and for comparison with the CVN results.

6.2.2.1 Proposed irradiation plan

The proposed plan to address the issue of the effects of neutron irradiation on MHTGR reactor pressure vessel components incorporates ORNL characterization of the unirradiated materials using drop-weight, CVN, and tensile tests, irradiation of CVN and tensile specimens in the University of Buffalo Reactor (UBR), and testing of the irradiated specimens at ORNL. Material proposed for inclusion are two heats of SA533 Grade B Class 1 plate, one with low copper (~0.05) and one with high copper (~0.15); this will allow for an assessment of the significance of copper in the temperature range of interest. Also included will be an SA508 Class 3 forging and a submerged-arc weld with relatively high copper. Characterization of materials will include standard tensile, CVN, and drop-weight testing of the SA533 Grade B Class 1 plates, SA508 Class 3 forging, and a submerged-arc weld; CVN testing of the weld heat-affected zones will also be conducted for comparison but no irradiations are planned. Full Charpy curves will be obtained with tests conducted to emphasize determination of

the 41-J (30-ft-lb) transition temperature, the upper-shelf energy level, and the RT_{NDT} . In addition to those mechanical tests, hardness testing, metallography, and chemical analyses will be performed. Also, limited aging at 121 and 204°C (250 and 400°F) will be conducted and tests performed to allow separation of aging effects from the irradiation effects.

Details of the irradiation plan were not yet fully defined at the end of this reporting period. It is clear, however, that irradiations will be conducted to examine the four materials over a range of irradiation temperatures and equivalent fast fluences. The conditions selected should provide sufficient information to allow for some interpolation (not extrapolation) if design operating conditions change. The temperature range of 121 to 204°C (250 to 400°F) will encompass the current operating conditions, and irradiations at 288°C (550°F) will provide a reference to LWR conditions for comparison with the extensive data base from both test reactors and surveillance programs. Equivalent fast fluences will range from 10^{16} to 10^{18} neutrons/cm² to encompass the anticipated operating conditions with relatively large uncertainties in predictions.

Uncertainties about the effects of relatively low-energy epithermal neutrons (<0.1 MeV) and a low flux at low temperatures will necessitate separate irradiations with a tailored neutron spectrum (to match that anticipated for the MHTGR) for a long time (at a low flux), for example, 3 years.

The current plan envisions the use of the UBR for the irradiations. Through Materials Engineering Associates, the NRC has characterized the neutronics at various positions in the reactor, a necessary set of information for reliable determination of equivalent fast fluences and for spectral tailoring experiments. All specimen testing will be conducted in the hot cells at ORNL. Tensile tests will be conducted over the MHTGR operating temperature range, and CVN tests will be conducted to obtain full transition curves with emphasis on determination of the 41-J (30-ft-lb) shifts and changes in the upper-shelf energies.

6.3 HEAT TRANSPORT MATERIALS (WBS 1603.04)

6.3.1 Creep, Aging, and Decarburization of 2.25Cr-1Mo Steel - H. E. McCoy

The 2.25Cr-1Mo steel tubes in the economizer/evaporator/primary superheater section of the MHTGR steam generator will be installed in a heat-treatment condition that produces a metastable microstructure. The MHTGR-helium environment may either add or remove carbon depending on the temperature and the exact gas composition. Aging will also coarsen the microstructure, and these changes may alter both strength and fracture properties.

Previous studies have shown that long aging has reduced the creep strength, likely caused by coarsening of the fine bainitic microstructure.⁹⁻¹¹ Aging in environments that remove carbon also reduced the creep strength. Aging or testing in HTGR-He was carburizing at 482 and 538°C (900 and 1000°F), decarburizing at 649°C (1200°F), and nearly inert at 593°C (1100°F). Some of the creep tests have exceeded 70,000 h, and the environmental interactions are continuing to be followed.

Some tests are running in the wetter "design composition" gas of the MHTGR. The longest of these tests has reached 8000 h, and the creep properties seem within the range observed previously for the drier HTGR composition. Two samples were destructively examined after 2000 h of testing in the "design composition" test gas, and the carbon changes were similar to those noted previously in the more dilute composition.

6.3.2 Fracture Mechanics Evaluations - R. K. Nanstad, D. J. Alexander, R. L. Swain, and R. W. Swindeman

The material selected for tubing in the cooler portions of the MHTGR steam generator is 2.25Cr-1Mo. The design operating temperature is in the range of 371 to 538°C (700 to 1000°F), where temper embrittlement may occur with a concomitant degradation in the fracture toughness of the material. Data needs identified for design include fracture toughness, fatigue crack-growth rate (FCGR), and creep crack-growth rate (CCGR). The description of the test material [a 25.4-mm-thick (1-in.) plate in the annealed condition] and results of basic properties characterization tests (i.e., tensile, Charpy impact, and creep) were provided in the annual report for 1985.¹²

In the annual report for 1986,¹³ a detailed description of the computer interface and dc-potential drop (PD) technique for conducting the fracture toughness and FCGR tests was presented. That description included a schematic diagram for the computer-interactive test system and calibration curves for determining crack extension measurements during the tests. Additionally, graphical presentations of FCGR data and J-integral resistance curves, obtained using the dc-PD technique, were included. Results of exploratory CCGR tests were excluded from the 1986 report, but are presented here.

Regarding the computer-interactive dc-PD system, the primary activities during this reporting period included experimentation with variations in placement of current leads and potential probes and refinement of the calibration expressions for PD voltage as a function of crack length in the compact specimens.

6.3.2.1 Creep crack-growth testing

Exploratory testing is under way to develop test procedures and analytical methods for CCGR evaluations. Two specimen sizes are being examined, both of which are of the modified compact specimen (CS) design. The larger specimen is the nominal 1TCS [25.4-mm-thick (1-in.) compact specimen] design in the plane section with a thickness of 12.3 mm (0.5 in.). The smaller specimen is the nominal 1/2T design.

Before CCGR testing, the samples were precracked at room temperature using a 10-Hz sinusoidal wave with a minimum-to-maximum load ratio of 0.1. After precracking, the ratio of crack length to width (a/W) was ~0.45. PD leads were attached to the specimens at locations identified by McGowan and Nanstad,¹⁴ and their calibration curve for relating crack length to dc-PD is being examined as part of the experimental methods development. Tests were performed in a 12/1 lever arm creep machine capable of 24-kN (6000-lb) loads.

The first test used a specimen without side grooves, while subsequent tests incorporated specimens with side grooves equal to 10% of the specimen thickness on each side. Side grooves allow for relatively straight crack-front extension during the test. Following completion of each test, the specimen is cooled in liquid nitrogen and broken open; the creep crack

extension is measured using a nine-point averaging technique with a digital microscope. The fatigue crack length is also measured at that time. Scanning electron fractography was conducted on one specimen and showed the differences in fatigue morphology between the fatigue precrack, the creep crack growth, and the posttest cleavage area as further confirmation of the occurrence of crack extension caused by creep mechanisms.

Four tests have been conducted at net section stresses ranging from 34.6 to 58.4 MPa. One test was conducted at 43.4 MPa, the 100,000-h design stress intensity for annealed 2.25Cr-1Mo steel in Code Case N47-22 of the *ASME Boiler and Pressure Vessel Code*. In most tests, the crack growth was limited to short distances to provide data to evaluate the correlation of the PD measurements with crack extension. This limited the data on individual specimens to one or two values of crack extension per unit time (da/dt) to plot against the parameter characterizing crack growth.

Many parameters have been proposed to correlate CCGR data. Some of these are stress intensity factor (K), J-integral, transient value of creep crack growth (C_t), net section stress, and reference stress. For preliminary calculations, K and C_t have been used here. The da/dt data have been plotted against these parameters in Figs. 6.2 and 6.3, respectively. Comparisons made with other results¹⁵⁻¹⁷ are included on the plots. Tensile and creep test results were used to examine the applicability of those characterizing parameters. Figure 6.4 shows a plot of engineering stress vs plastic strain from the tensile test at 538°C (1000°F). The relationship shows that the plastic component of J is not contributing to the deflection rate for the relatively low stress intensities used for the creep crack-growth tests. Figure 6.5 is a plot of engineering stress vs minimum creep rate from the creep test results to determine the stress sensitivity of creep, characterized by the stress exponent, needed for the C_t evaluation.

Scanning electron microscope fractography of the creep crack-growth specimen fracture surfaces has been performed and reveals the ductile-dimple characteristics typical of ductile crack growth but also exhibits the more elongated and smeared nature of the features typical of very slow growth.

Fig. 6.2. CCGR vs stress intensity K of annealed 2.25Cr-1Mo steel plate at 538°C (1000°F) compared to that for two other Cr-Mo steels. Source: T. Iwadata, J. Watanabe, and Y. Tanaka, p. 30 in *Residual Life Assessment, Nondestructive Examination, and Nuclear Heat Exchanger Materials*, PVP-Vol. 98-1, American Society of Mechanical Engineers, New York, 1985; and K. Ohji, K. Ogura, S. Kubo, and Y. Katada, p. 9 in *Engineering Aspects of Creep*, Vol. 2, Institute of Mechanical Engineers, London, 1980.

Fig. 6.3. CCGR vs transient value of CCGR C_t of annealed $2.25\text{Cr}-1\text{Mo}$ steel plate at 538°C (1000°F) compared with a Cr-Mo-V steel. Source: A. Saxena and J. D. Landes, *Characterization of Creep Crack Growth of Metals*, paper 83-1D7-EVFLA-P1, Westinghouse Research and Development Center, Pittsburgh, June 1983.

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Fig. 6.4. Engineering stress vs plastic strain from tensile test at 538°C (1000°F) of annealed 2.25Cr-1Mo steel plate.

Fig. 6.5. Engineering stress vs minimum creep rate from creep tests of annealed 2.25Cr-1Mo steel. Stress exponent of creep at 538°C (1000°F) was determined to be 5.24 and used for the evaluation of C_t .

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7. SAFETY TECHNOLOGY (WBS 1604)

J. C. Cleveland

7.1 INTRODUCTION

During this past year a Subprogram for Cooperation in the Arbeitsgemeinschaft Versuchs Reaktor (AVR) Test Program was established and is being conducted under the United States/Federal Republic of Germany (US/FRG) High-Temperature Reactor Umbrella Agreement. The cooperation involves Oak Ridge National Laboratory (ORNL) and General Atomics (GA) of the United States and AVR and Kernforschungsanlage (KFA) of the FRG.

The test program is providing experimental information beneficial to the U.S. High-Temperature Gas-Cooled Reactor (HTGR) Program. AVR test data are being utilized to help validate computational methods used for modular high-temperature gas-cooled reactor (MHTGR) design and licensing. In the FRG program, the AVR is being used experimentally to investigate performance and safety features pertinent to MHTGRs.

7.2 SUBPROGRAM FOR COOPERATION IN AVR TEST PROGRAM

The individual U.S. and FRG gas-cooled reactor programs have many similarities in objectives and in experimental data and performance validation requirements in spite of some significant differences in the specific reactor designs.

While the AVR is a relatively low-power-level reactor [46 MW(t), 15 MW(e)], it possesses many features also incorporated into MHTGR [low power density, low enriched uranium (LEU) fuel; coated fuel particles, and capability for passive removal of afterheat through the reactor vessel wall]. It therefore permits experimental investigations and demonstrations of important features pertinent to MHTGR performance and safety.

The purpose of the cooperation is to obtain experimental information from the AVR relevant to the performance and safety of MHTGR and to compare measured results with predictions of analytical tools. The scope of the

cooperation involves examining the behavior of the AVR under various planned test conditions pertinent to modular gas-cooled reactor performance. AVR test data will be utilized to help validate computational methods used in the areas of reactor physics, fission and activation product transport, and plant transient response and understand modular gas-cooled reactor safety behavior (e.g., fission product behavior, dust behavior, and thermofluid dynamic behavior).

The following specific areas are included in the scope:

- reactor physics measurements and methods validation,
- transient response measurements and methods validation, and
- fission and activation product behavior and methods validation.

The AVR is conducting experimental measurements of reactor physics parameters such as reactivity, reactivity coefficients, and control rod worth. These measurements are being carried out over a range of temperatures and xenon concentrations and with increasing LEU fueling component. Plutonium and higher actinide buildup in LEU fuel is also being experimentally determined. These data provide a basis for further validation of present gas-cooled reactor physics calculational tools for computing physics parameters used in safety analyses.

Data on the behavior of fission products in the AVR are generic information useful to gas-cooled reactors. AVR operating data can provide a basis for comparison with out-of-pile tests and can allow comparison with predictions by models used for modular gas-cooled reactor design and licensing. An important AVR experiment involves determining the activity released from the reactor core and circuit during a partial depressurization.

Other tests at the AVR will examine reactor behavior under transient and accident conditions including core flow and power changes and simulation of a loss-of-coolant accident (LOCA) without auxiliary cooling system operation. For such an accident, the U.S. and the FRG MHTGRs both rely on passive-inherent safety features similar to those at the AVR. These tests will demonstrate in an actual reactor that inherent characteristics can acceptably limit LOCA consequences.

Five Project Work Statements (PWSs) covering eight AVR tests have been developed within this Subprogram. Table 7.1 lists the PWS titles and the related AVR test numbers.

Table 7.1. PWS titles, status, and related AVR test numbers

PWS title	PWS No.	Related AVR test No.	Status
Measurement and computation of reactor physics parameters for AVR	AVR-1	HTA-1, HTA-3 and part of HTA-8	In progress
Measurement and computation of plutonium buildup in selected LEU elements of AVR	AVR-4	HTA-4	In progress
Reactor dynamics testing and analysis	AVR-2	HTA-2	In progress
LOCA simulation test and analysis	AVR-5*	HTA-5 and HTA-8	In progress
Examination of fission product release with depressurization	AVR-6*	HTA-6	In progress

*These PWSs relate to certain objectives identified in the Safety Subprogram.

ORNL work on Subprogram milestones was initiated in November with the first effort involving pretest predictions for the LOCA simulation test.

7.3 ANALYSES OF CONTROL ROD TEMPERATURES DURING THE LOCA SIMULATION TEST

The LOCA simulation test, planned for May 1988, will simulate a LOCA occurring at full power with a gas outlet temperature in the 800 to 850°C range. The reactor will first be shut down and depressurized to 1 bar by

normal procedures. It will then be taken critical and heated to normal operational temperatures with forced circulation and nuclear power. The accident will be simulated by stopping forced circulation and matching the time dependence of accident levels of decay power by varying the nuclear power generation.

The analyses at ORNL determined the temperatures that would be experienced by the control rods if the reactor were to scram during the test. These analyses were performed jointly by A. Bergerfurth, Group Leader for Thermofluid Dynamics, AVR, and J. C. Cleveland, ORNL. The core and control rods were modeled with the ORNL HEATING-6 transient heat transfer code. A three-dimensional model was used to determine the time-dependent temperatures at the control rod channel locations. Temperatures at these locations were used as time-dependent boundary conditions for a two-dimensional model representing the control rods following scram. Results showed that the LOCA test could be conducted at conditions simulating an accident occurring at full power, without excessive control rod temperatures if a scram occurs at any time during the test.

8. SHIELDING ANALYSIS (WBS 5211.04)

J. A. Bucholz

8.1 INTRODUCTION AND OVERVIEW

This chapter summarizes the reactor shielding work performed for the Modular High-Temperature Gas-Cooled Reactor (MHTGR) Program during CY 1987. The one-dimensional (1-D) parametric analyses and detailed two-dimensional (2-D) analyses described below were performed to provide design support to General Atomics (GA) and to assist in specifying the reactor internals design so that component fluences and radiation dose rates do not exceed acceptable values.

The upper portion of the reactor system was the sole focus of several analytical studies in 1987. The initial global 2-D analysis performed in 1986¹ specifically accounted for radiation streaming through the twelve 95.3-mm-diam (3.75-in.) reserve shutdown system (RSS) penetrations, but assumed that six 521-mm-diam $\times$ 762-mm-high (20.5-in. $\times$ 30-in.) shield jacket assemblies would be located in the upper plenum to provide additional localized shielding for each pair of RSS tubes entering the upper plenum. By late 1986, GA had decided that these RSS tube shield jacket assemblies would no longer be included in the reference design. The first task of 1987 was, therefore, to determine the degree to which their removal would increase the already high neutron fluences at the reactor vessel near the base of the inner control rod assembly housings. The second task was to assist GA in developing an effective neutron radiation shield for this region. Two series of 1-D shielding analyses were performed to determine how much borated graphite shielding would be required to reduce the neutron fluences in this region to acceptable levels. An auxiliary series of 1-D calculations were performed to determine if any of nine other proposed shielding materials might be more effective. Borated graphite was eventually chosen, and three major design changes were made to provide the necessary neutron shielding in the upper plenum: (1) lowering the RSS

hoppers inside the inner control assembly housings, (2) placing a thick borated graphite plug inside the inner control assembly housing, and (3) placing a thick borated graphite neutron shield bonnet in the region between the upper plenum thermal shield and the reactor vessel. The final task for 1987 was to perform a completely revised global 2-D shielding analysis for the upper portion of the MHTGR reactor system with the neutron shield bonnet located inside the reactor vessel.

8.2 REVIEW OF THE INITIAL 2-D ANALYSIS OF THE UPPER INTERNALS¹

Figure 8.1 shows the 2-D DOT² model used for the initial upper internals analysis. The upper axial reflector was 1189 mm (46.8 in.) thick. Distributed about the inner reflector, as shown in Fig. 8.2, are twelve 95.3-mm-diam (3.75-in.) penetrations for the RSS. They run the entire length of the core and penetrate through the upper axial reflector and the upper plenum shield cans. Pairs of RSS tubes were drawn in a goosenecklike fashion through the RSS tube shield jacket assemblies illustrated in Fig. 8.3. To obtain conservative worst-case results on the operating deck, the inner control rod assembly housings were treated as being completely void in the initial 2-D analysis.

Because of the intense, highly directional, streaming source introduced in the upper plenum by the RSS penetrations, a complete analysis of each case required several calculational steps. First a global 2-D calculation was performed for the upper portion of the core of the MHTGR reactor system using a biased S10 quadrature with 100 discrete angular directions. This 2-D model included all major components from the midplane of the reactor up to, and including, the concrete floor of the operating deck. In this first calculational step, however, the RSS penetrations in the core and axial reflector were intentionally ignored, as were the RSS tube shield jacket assemblies in the upper plenum. Second, a special high-order DOT calculation was performed to determine the angular-dependent streaming through a typical RSS penetration. The special quadrature used in this calculation had 280 of its 315 discrete angular directions in the upward hemisphere, with most of them close to the z axis. Radially, this calculational model encompassed a single RSS penetration and all the surrounding media within 328 mm (12.9 in.) of the centerline. Axially, it

Fig. 8.1. Upper portion of MHTGR reactor system as modeled in global 2-D shielding analyses.

Fig. 8.2. Top view of annular MHTGR core, showing location of control rod channels, RSS tube penetrations, and RSS tube shield jacket assemblies.

included the upper reflector, the upper plenum shield cans, and an idealized model of the RSS tube shield jacket assembly. The previously calculated scalar flux at the interface between the core and the upper axial reflector was used as an isotropic boundary flux condition on the lower surface. The upward-directed, spatially dependent, multigroup angular flux was then calculated along the upper surface that included the 102-mm-diam (4-in.) hole in the center. Third, a similar calculation was performed using the same boundary conditions, spatial mesh, and angular quadrature. In this calculation, however, the RSS penetration and the shield jacket assembly were removed from the model. Next, using the results of steps 2 and 3, the excess angular flux (F_2-F_3) was calculated

Fig. 8.3. Side view of RSS tube shield jacket assembly.

and averaged over the upper surface. This "excess angular flux" is the flux above and beyond that which would normally be present if the RSS penetration and shield jacket assembly did not exist. Lastly, a second global 2-D calculation was performed for the upper portion of the core and the MHTGR reactor system. In this calculation, the volumetric sources within the core were set equal to 0 while an upward-directed, angular-dependent internal boundary source was applied across a 483-mm-wide (19 in.) flat annular ring located 864 mm (34 in.) above the upper plenum

shield cans, directly below the 521-mm-diam (20.5-in.) inner control assembly housing penetrations. Given the excess angular flux corresponding to a single RSS penetration, the number of RSS penetrations (12), and the area of the annular ring, the upward-directed, angular-dependent internal boundary source was properly normalized to conserve the number of "excess" neutrons entering the upper plenum. Results of this global 2-D calculation were then combined with those of the first global 2-D calculation to obtain the composite flux solution in the upper portion of the MHTGR reactor system.

Streaming through the RSS penetrations was found to be extremely important, and the angular flux entering the upper plenum was very forward-directed.¹ This was especially true of the neutrons at full power and the gamma radiation at and after shutdown. Not including gamma radiation from plate-out sources or activated steel components, the direct gamma radiation dose rates atop the center of the reactor vessel would be 52.4 and 29.1 mrem/h at 1 and 10 d after shutdown. Because of the highly collimated nature of the radiation streaming through the housing(s), the gamma dose rate at the centerline of each housing may be $3.7\text{E}+4$ mrem/h 1 d after shutdown, and $1.9\text{E}+4$ mrem/h 10 d after shutdown. (Note, however, that components later located inside the lower portion of the housing would attenuate some of this radiation while at the same time scattering much of it into other regions above the reactor.) Total and thermal neutron fluences on the inner surface of the reactor vessel were $9.7\text{E}+17$ and $2.6\text{E}+17$ neutrons/cm² near the centerline. The total and thermal fluences around the base of the 521-mm-diam (20.5-in.) penetration for the RSS/control rod assembly housing were essentially the same, while at the centerline of the housing penetration they were considerably higher (i.e., $3.3\text{E}+18$ and $1.9\text{E}+18$ neutrons/cm², respectively). The effective fast fluence* at these locations was $1.1\text{E}+17$, $1.3\text{E}+17$, and $4.0\text{E}+17$ neutrons/cm².

*The effective fast fluence is an artificial quantity used to relate vessel damage in a high-temperature gas-cooled reactor (HTGR) system to that in light-water reactor (LWR) systems, which typically have softer neutron spectra. The effective fast flux is defined as the actual flux above 0.9 MeV, plus half the actual flux between 0.1 and 0.9 MeV, plus 0.0025 times the actual flux between 3.05 eV and 0.1 MeV, plus 0.07 times the actual thermal flux below 3.05 eV. Fluences are based on the reactor operating at full power, 80% of the time, for 40 years.

Without the RSS penetrations, the thermal flux in the upper plenum would be about 2 orders of magnitude lower, while the fast and epithermal fluxes would be about 1 order of magnitude lower.

8.3 REVISED 2-D ANALYSES WITHOUT THE RSS TUBE SHIELD JACKET ASSEMBLIES³

By late 1986, GA had decided that the RSS tube shield jacket assemblies shown in Fig. 8.3 would no longer be included in the reference design. The first task of 1987 was, therefore, to determine the degree to which their removal would increase the already high neutron fluences at the reactor vessel near the base of the inner control assembly housings. The revised shielding analyses were performed in exactly the same manner as those described in Sect. 8.2 except that (1) the shield jacket assemblies were removed from the high-order streaming analysis for a single RSS penetration and (2) the internal boundary source used to represent the neutron streaming into the plenum through the 12 RSS penetrations was now introduced in the global 2-D analysis at a point directly atop the upper plenum shield cans rather than 864 mm (34 in.) above the shield cans.

Total and thermal neutron fluences on the inner surface of the reactor vessel near the centerline and around the base of the inner control assembly housing were found to be 2.8 to $3.1\text{E}+18$ neutrons/cm² and 1.4 to $1.6\text{E}+18$ neutrons/cm², respectively, while those at the centerline of the housing penetrations were now $1.1\text{E}+19$ and $7.3\text{E}+18$ neutrons/cm². The effective fast fluences at these locations were $2.9\text{E}+17$, $4.5\text{E}+17$, and $1.2\text{E}+18$ neutrons/cm². Relative to the earlier analysis, the removal of the RSS tube shield jacket assemblies appeared to increase the maximum neutron fluence in the vessel by a factor of 3.2 or 3.3. This same increase was also typical of the gamma dose rates at 1 d after shutdown. Further from the base of the inner control assembly housing, the increase was somewhat less. Near the outer edge of the outer control assembly housing, for example, the total fluence increased from $6.9\text{E}+17$ to $8.3\text{E}+17$ neutrons/cm², while the effective fast fluence there increased from $8.5\text{E}+16$ to $8.9\text{E}+16$ neutrons/cm².

8.4 EFFORTS TO IMPROVE THE SHIELDING CONFIGURATION IN THE UPPER PLENUM

8.4.1 Simple 1-D Studies^{4,5}

The first series of 1-D shielding calculations was performed to determine the necessary thickness of a neutron shield "bonnet" to be made of borated graphite and placed between the upper plenum thermal shield and the reactor vessel.⁴ Moreover, 24 such calculations were performed with the thickness of the loose-packed borated graphite varying from 50.8 to 1219 mm (2 to 48 in.). One set of calculations was performed with an isotropically impinging surface source, while a second set of calculations was performed using a monodirectional source impinging on the surface of the shield. Using the more realistic monodirectional source caused the estimated shield thickness to be significantly greater than when the isotropic condition was used. To reduce the effective fast fluence down to a level comparable to that found around the beltline of the vessel ($26.E+16$ neutrons/cm²) would require 894 mm (35.2 in.) of loose-packed borated graphite particles having an average density of 0.89 g/mL, or 18.0 in. of solid borated graphite at 1.74 g/mL. A slightly higher effective fast fluence of $5.0E+16$ neutrons/cm² would require 389 mm (15.3 in.) of borated graphite at 1.74 g/mL.

At GA's request, a second series of 1-D calculations was performed to identify other shielding materials that might be more effective than the borated graphite at reducing the effective fast fluence.⁵ Nine alternate materials were considered. Carbon steel, stainless steel, and depleted UO₂ were found to be less effective (per unit thickness) than the borated graphite at 1.74 g/mL; magnesium oxide and ceramic alumina were found to be just marginally better; and zirconium hydride, lithium hydride, zirconium hydroxide, and beryllium oxide were all found to be significantly better per unit thickness. Because of other material properties, only zirconium hydride and beryllium oxide appeared to be potentially viable substitutes. Ultimately, GA deemed borated graphite to be the preferred material.

8.4.2 Design Changes to Reduce Vessel Fluences in the Upper Plenum

Three major design changes were ultimately made to reduce the vessel fluence in the upper plenum and inside the inner control assembly housings:

- (1) lowering the RSS hoppers inside the inner control assembly housings such that the bottoms of the hoppers would be inside the reactor vessel, level with the upper plenum thermal shield;
- (2) placing thick borated graphite plugs inside the bottom of the inner control assembly housings to fill the large void spaces between the RSS hoppers and the housing;
- (3) placing a 381-mm-thick (15-in.) borated graphite neutron shield in the region between the upper plenum thermal shield and the reactor vessel. Because this would be atop the thermal shield, it would not interfere with refueling operations.

To accommodate the neutron shield bonnet (item 3), the overall length of the reactor vessel had to be increased by 559 mm (22 in.).

8.5 GLOBAL 2-D ANALYSES WITH THE NEUTRON SHIELD BONNET LOCATED INSIDE THE UPPER PORTION OF THE REACTOR VESSEL⁶

Figure 8.4 shows the current 2-D RZ DOT-IV model of the upper portion of the MHTGR reactor system. It now includes a tapered, 381-mm-thick (15-in.), borated graphite neutron shield bonnet; a reactor vessel that is 559 mm (22 in.) longer than that in the original design; 12 new 527-mm-diam (20.75-in.) penetrations below the outer control assembly housings; no housing support structure; a 102-mm-thick (4-in.) steel plate above the reactor vessel in lieu of the original 305-mm-thick (12-in.) steel plate; and a significant amount of neutron shielding material inside the inner control assembly housings containing the RSS hoppers. The model also includes a 914-mm-thick (3-ft), 2134-mm-wide (7-ft) concrete ledge around the top of the reactor vault; an annular, 1219-mm-thick (4-ft), concrete shield surrounding the control rod assembly housings above the reactor vessel; and a floating steel seal ring around the top of the reactor vessel, attached to the adjoining concrete ledge by means of a fairly thick steel mounting bracket. With 100 discrete directions in the angular quadrature, 172 mesh intervals in the radial direction, 339 mesh intervals in the axial direction, 15 neutron groups, and 18 gamma energy groups, each global 2-D DOT calculation required ~9 h of CPU time on the Oak Ridge National Laboratory (ORNL) CRAY-XMP computer.

Fig. 8.4. Actual DOT-IV model of upper portion of MHTGR reactor system with neutron shield bonnet located inside reactor vessel.

Following the procedure described in Sect. 8.2, the coupled neutron/gamma problem was first solved using just the volumetric source in the core, with no RSS penetrations through the upper axial reflector or the upper plenum shield cans. The coupled neutron/gamma problem was then solved a second time with an internal boundary source in the upper plenum representing the neutron and gamma radiation streaming through the RSS penetrations (cf. Sect. 8.3). In this second global 2-D calculation, the volumetric source term in the core was set to zero. The actual flux distribution throughout the system was then obtained by combining (i.e., superimposing) the two sets of results.

Figure 8.5 shows the combined results for the effective fast fluence on the inner surface of the reactor vessel. Note that it now diminishes from $7.4\text{E}+15$ neutrons/cm² near the centerline to $2.5\text{E}+15$ neutrons/cm² near the outer edge of the outer control assembly housings, and then increases to $2.5\text{E}+16$ neutrons/cm² near the in-service inspection port, and $6.2\text{E}+16$ neutrons/cm² by the top of the vessel flange. Across from the top section of the core it is $2.8\text{E}+16$ neutrons/cm². While the effective fast fluence is the primary quantity of interest for vessel designers, also note that the total fluence may be significantly greater than the effective fast fluence, depending on location. More detailed results may be found in ref. 6.

The presence of the neutron shield bonnet and the other shielding materials inside the inner control assembly housing have also been found to significantly reduce the direct gamma dose rate along the centerline of the housing and in the region above the reactor vessel. Along the centerline of the system, the gamma dose rate 1 d after shutdown now varies from 3.2 mrem/h above the vessel to 0.5 mrem/h at the bottom of the 102-mm-thick (4-in.) steel plate gamma shield. Along the centerline of the inner control assembly housing, the gamma dose rate varies from a high of 87 mrem/h near the outer surface of the vessel to a low of 8.7 mrem/h near the bottom of the 102-mm-thick (4-in.) steel plate gamma shield. Without the RSS hoppers and the other shielding materials inside the inner control assembly housings, the corresponding gamma dose rates 1 d after shutdown would have been 400 to 4000 times greater. (As before, note that these dose rates are based solely on radiation from fission products in the core

Fig. 8.5. Effective fast fluence (neutrons/cm²) along inner surface of the reactor vessel.

and do not include radiation from plate-out source terms or activated metallic components.)

While the neutron shield bonnet and the other shielding materials inside the inner control assembly housing reduced the effective fast fluence at the apex of the reactor vessel by nearly a factor of 40, several points should be noted: (1) local effects that are impossible to model in a global 2-D calculation may cause the actual vessel fluences directly above an RSS penetration to be higher than the values cited here; (2) the present analysis, like earlier analyses, does not account for possible streaming through the 6 inner control rod channels or the 24 outer control rod channels; and (3) the neutron shield bonnet does nothing to reduce the fluence on the vessel close to the flange, which is $6.2\text{E}+16$ neutrons/cm² with the bonnet, and $6.7\text{E}+16$ neutrons/cm² without the bonnet.

8.6 EFFECT OF BLIND HOLES AROUND THE CONTROL RODS ON THE EFFECTIVE FAST FLUENCE AT THE BELTLINE OF THE REACTOR VESSEL

In June, GA informed ORNL of a potential need to reduce the effective density of the graphite in the removable side reflector by 35% to enhance the worth of the control rods located in the reflector. This would be accomplished through the use of blind holes in the graphite blocks. While the portion of the removable side reflector that would have to be modified had not been established, ORNL was asked to estimate what effect that would have on the effective fast fluence at the beltline of the vessel. One-dimensional calculations have since shown that such a density reduction throughout the two annular rings comprising the removable side reflector ($R = 175.16$ to $R = 242.60$ cm) would increase the effective fast fluence at the beltline of the reactor vessel by a factor of 9.6. This result is in good agreement with what one would estimate through graphical analysis of Fig. 2.14 in ref. 7. Indeed, such an analysis would also suggest that reducing the density in just the first ring would raise the effective fast fluence at the vessel beltline by a factor of 3.2, while reducing the density in just the first half of the first ring would increase it by a factor of 1.8. This matter will be examined in more detail if and when the actual design of the reflector is modified.

8.7 REFERENCES

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9. ADVANCED SYSTEM DESIGN SUPPORT (WBS 6600)

P. L. Rittenhouse

9.1 NUCLEAR HEAT SYSTEM DESIGN CRITERIA (WBS 6600.01) - S. -J. Chang

The objective of this task is to extend the existing high-temperature design methodology to the particular alloys and conditions of interest in the very high-temperature gas-cooled reactor (VHTGR). As described in previous annual reports, this is being accomplished in concert with the Task Force on Very High Temperature Design (TF-VHTD) established in March 1983 within the American Society of Mechanical Engineers (ASME) Code organization. The task force is to formulate design rules and procedures for Class 1 components of helium-cooled nuclear reactor plants operating at temperatures to 950°C (1742°F). One of the most important primary needs is to obtain valid mathematical representations of inelastic materials behavior for the purpose of design calculations. The key material of interest is Inconel 617. At present, it is anticipated that the new rules will be an extension of the current high-temperature Code Case N-47. To include Inconel 617 in Case N-47, transient stress-strain curves and isochronous stress-strain curves must be available. These curves must represent average Inconel 617 behavior.

As described in the last annual progress report, we have completed the task of developing these curves at temperatures from 760 to 982°C (1400 to 1800°F) based on data supplied by General Electric (GE) for an average heat of Inconel 617 (heat XX63A8UK). The results were submitted to the Code Committee for approval. In response to the suggestions from the Committee, we extended the curves to temperatures between 649 and 982°C (1200 and 1800°F) and revised the calculation to include a temperature-dependent Young's modulus. The revised curves and material constants are described below.

9.1.1.1 Deformation of Inconel 617 Modeled by Unified Constitutive Equations with a Softening Function

As described in the last annual progress report, after introducing a softening function $\Phi(\epsilon_i)$, the unified equations have been applied to represent the inelastic deformation of Inconel 617 at temperatures up to 950°C (1742°F).¹ Using unified equations, the experimentally observed stress drop after yielding for constant strain-rate tensile test and accelerated creep in constant-load creep tests were both modeled effectively.

In the unified theory used, the three basic equations describe the deformation behavior of Inconel 617 with revised material constants.

Total Strain Equation:

$$\epsilon = \frac{\sigma}{E} + \epsilon_i \quad , \quad (1)$$

where ϵ_i is the inelastic strain.

Flow Equation:

$$\dot{\epsilon}_i = 1.704 \times 10^{-5} \times 10^\delta \times \Phi(\epsilon_i) \left[(\sigma - \alpha)^2 - K^2 \right]^{\frac{n-1}{2}} (\sigma - \alpha) \quad . \quad (2)$$

Growth Equation:

$$\dot{\alpha} = H \dot{\epsilon}_i / \alpha^{1.5} - R \alpha^{1.5} \quad . \quad (3)$$

In Eqs. (1)-(3),

$$H = 10^m \quad ,$$

$$\Phi(\epsilon_i) = 1.0 + 3.16 \times 10^3 \epsilon_i^{0.35} \quad ,$$

$$R = R_1 \times 10^{-6} \Phi(\epsilon_i) \quad ,$$

$$R_1 = 0.46 \times 10^{m-\delta} \quad ,$$

$$n = \begin{cases} n_0 + 0.08 (\sigma - \sigma_0) \quad , & \sigma > \sigma_0 \\ n_0 \quad , & \sigma \leq \sigma_0 \end{cases} \quad ,$$

and E , m , n_0 , δ , σ_0 , and K are given in Table 9.1. In these equations, stress σ and back-stress α are in kips per square inch, time is in hours, and ϵ is in percent. Two specific functions $\Phi(\epsilon_i)$ and n are used in the

Table 9.1. Material constants

Temperature [°C (°F)]	E	m	n_0	δ	σ_0	K
649 (1200)	246.000	3.800	5.000	5.000	10.000	10.000
677 (1250)	242.844	3.353	4.750	4.486	12.500	9.250
704 (1300)	239.625	3.000	4.500	4.000	15.000	8.000
732 (1350)	236.344	2.743	4.250	3.542	17.500	6.250
760 (1400)	233.000	2.585	4.000	3.113	20.000	4.000
788 (1450)	229.594	2.286	3.875	2.568	20.625	3.369
816 (1500)	226.125	2.009	3.750	2.067	22.500	2.825
843 (1550)	222.594	1.756	3.625	1.611	25.625	2.369
871 (1600)	219.000	1.526	3.500	1.200	30.000	2.000
899 (1650)	215.344	1.319	3.375	0.833	35.625	1.719
927 (1700)	211.625	1.135	3.250	0.511	42.500	1.525
954 (1750)	207.844	0.974	3.125	0.234	50.625	1.419
982 (1800)	204.000	0.836	3.000	0.001	60.000	1.400

unified equations for Inconel 617. Function $\Phi(\epsilon_1)$ is introduced to accommodate the stress drop after yielding and the accelerated creep. The exponent n for plastic flow is a function of temperature as well as the applied stress.

Using these equations, a set of isochronous curves for temperatures between 649 and 982°C (1200 and 1800°F) was plotted and submitted to the TF-VHTD. A set of tensile stress-strain curves for different strain rates was also submitted covering the same temperature range. The isochronous curves will be incorporated directly into Case N-47, while the tensile curves will be used to develop allowable stresses and yield stress values. The isochronous stress-strain curves and tensile curves are shown in Figs. 9.1 to 9.14.

9.2 ADVANCED SYSTEMS DESIGN SUPPORT (WBS 6600.03) – O. F. Kimball

9.2.1 Introduction

Research in support of the Advanced High-Temperature Gas-Cooled Reactor (HTGR) Program involves the evaluation and development of materials with adequate strength, environmental compatibility, and thermal stability to provide essential safety margins for service from 100,000 to 300,000 h

ORNL-DWG 89-2100

INC 617 ISOCHRONOUS CURVES

TEMP(°F) - 1800. TIME(H) - 0.1 TO 100000.0

Fig. 9.1. Isochronous stress-strain curves at 982°C (1800°F).

ORNL-DWG 89-2101

INC 617 ISOCHRONOUS CURVES

TEMP(°F) - 1700. TIME(H) - 0.1 TO 100000.0

Fig. 9.2. Isochronous stress-strain curves at 927°C (1700°F).

ORNL-DWG 89-2102

INC 617 ISOCHRONOUS CURVES

TEMP(°F) - 1600. TIME(H) - 0.1 TO 100000.0

Fig. 9.3. Isochronous stress-strain curves at 871°C (1600°F).

ORNL-DWG 89-2103

INC 617 ISOCHRONOUS CURVES

TEMP(°F) - 1500. TIME(H) - 0.1 TO 100000.0

Fig. 9.4. Isochronous stress-strain curves at 816°C (1500°F).

ORNL-DWG 89-2104

INC 617 ISOCHRONOUS CURVES

TEMP(DEG F) - 1400. TIME(H) - 0.1 TO 100000.0

Fig. 9.5. Isochronous stress-strain curves at 760°C (1400°F).

ORNL-DWG 89-2105

INC 617 ISOCHRONOUS CURVES

TEMP(DEG F) - 1300. TIME(H) - 0.1 TO 100000.0

Fig. 9.6. Isochronous stress-strain curves at 704°C (1300°F).

ORNL-DWG 89-2106

INC 617 ISOCHRONOUS CURVES

TEMP(°F) - 1200. TIME(H) - 0.1 TO 100000.0

Fig. 9.7. Isochronous stress-strain curves at 649°C (1200°F).

ORNL-DWG 89-2107

INCONEL 617

TEMP(°F) - 1800. EOOT(1/M) - 5.0×10^{-4} , 5.0×10^{-3} , 5.0×10^{-2}

Fig. 9.8. Tensile stress-strain curves at 982°C (1800°F).

ORNL-DWG 89-2108

INCONEL 617

TEMP(°F) - 1700. $\dot{\epsilon}$ (/M) - 5.0×10^{-4} , 5.0×10^{-3} , 5.0×10^{-2}

Fig. 9.9. Tensile stress-strain curves at 927°C (1700°F).

ORNL-DWG 89-2109

INCONEL 617

TEMP(°F) - 1600. $\dot{\epsilon}$ (/M) - 5.0×10^{-4} , 5.0×10^{-3} , 5.0×10^{-2}

Fig. 9.10. Tensile stress-strain curves at 871°C (1600°F).

ORNL-DWG 89-2110

INCONEL 617

TEMP(DEG F) - 1500. EDOIT(/M) - 5.0×10^{-4} , 5.0×10^{-3} , 5.0×10^{-2}

Fig. 9.11. Tensile stress-strain curves at 816°C (1500°F).

ORNL-DWG 89-2111

INCONEL 617

TEMP(DEG F) - 1400. EDOIT(/M) - 5.0×10^{-4} , 5.0×10^{-3} , 5.0×10^{-2}

Fig. 9.12. Tensile stress-strain curves at 760°C (1400°F).

ORNL-DWG 89-2112

INCONEL 617

TEMP(°F) - 1300. EDOT(1/M) - 5.0×10^{-4} , 5.0×10^{-3} , 5.0×10^{-2}

Fig. 9.13. Tensile stress-strain curves at 704°C (1300°F).

ORNL-DWG 89-2113

INCONEL 617

TEMP(°F) - 1200. EDOT(1/M) - 5.0×10^{-4} , 5.0×10^{-3} , 5.0×10^{-2}

Fig. 9.14. Tensile stress-strain curves at 649°C (1200°F).

at temperatures of 1000°C (1832°F) and above. Emphasis now is placed on determining the environmental compatibility of high-strength, wrought, nickel-base alloys, particularly Inconel 617, that would be used for nuclear process heat components (e.g., steam reformers, heat exchangers) for maximum service temperatures to 950°C (1742°F). Equipment is being developed and testing started to determine the range of simulated HTGR (SHTGR) helium chemistries (gas chemistry simulating that of the HTGR primary coolant) within which carburization or decarburization and subsurface oxidation are minimized. These reactions can produce substantial property degradations during service at elevated temperatures. Future work will emphasize determination of mechanisms of gas/metal interactions and will address the environmental compatibility, thermal stability, and mechanical properties of other nickel-base alloys, iron-base alloys (e.g., Alloy 800H), development alloys, ceramics, and composite materials.

The specific, near-term objectives of the gas corrosion and thermal stability studies tasks are to

- determine, for existing candidate alloys, primary coolant chemistry ranges (as a function of temperature) within which environmental compatibility is maximized;
- determine basic oxidation and carburization/decarburization mechanisms within and outside the optimum gas chemistry range; and
- demonstrate and map thermal stability of candidate alloys over temperature ranges of interest for advanced HTGRs.

9.2.2 Program and Facilities Development

Efforts on this program at Oak Ridge National Laboratory (ORNL) began in the second quarter of the calendar year and were focused on program planning and transferring the Department of Energy (DOE)-owned (Advanced HTGR program) equipment, materials, and program records from the General Electric (GE) laboratory in Schenectady, New York, to ORNL. This government-owned SHTGR helium test equipment had been used in the DOE-GE Advanced HTGR Materials Evaluation and Development Program in GE's High-Temperature Reactor Materials Testing Laboratory (HTRMTL). (The DOE-GE program was discontinued at the end of FY 1986.)

Near the end of July, most of the DOE-owned equipment and material was received at ORNL. The equipment consisted of 2 recirculating SHTGR helium loops and 1 once-through SHTGR helium loop with all associated systems for gas-impurity additions and measurement and control equipment; 6 large aging and corrosion furnaces and associated ceramic and metallic retorts; 6 multispecimen creep rigs (6 stringers per rig) and furnaces; 2 fatigue environmental chambers; and 45 single-specimen creep retorts (23 ceramic, 22 metallic). Materials included a number of nickel-base alloys (plate, bar, and sheet) and several thousand test specimens. The remaining government-owned equipment, material, test specimens, and records was received at ORNL in September. Included with this shipment were 45 creep testing units originally purchased by GE for the Advanced HTGR Materials Program. These units were purchased by DOE San Francisco Office for their remaining value (cost less depreciation), as part of the closeout costs.

By the end of October, laboratory space had been made available for the gas corrosion and thermal stability equipment. By the end of December, six large 5-zone furnaces and associated retorts, one recirculating SHTGR helium gas supply and analysis system, and ancillary equipment had been moved into one laboratory. The once-through SHTGR helium supply and analysis system and five horizontal furnaces and associated equipment were installed in a second laboratory.

During 1988 this equipment will be returned to operating condition. This involves installing the required power supplies; running electrical, water, and SHTGR helium gas supply systems to the retorts; checking for leaks in the gas supply system; and checking out and debugging all the systems. Work will also begin on returning to operating condition two small loops constructed earlier at ORNL but not operated for about 5 years. These are partly glass systems utilizing high-vacuum gas addition valves and other high-vacuum components in recirculating loops. One loop utilizes a quartz retort, the other a mullite retort.

9.2.3 Experimental Efforts Planned in 1988 Through 1991

Work planned for the Advanced HTGR Materials Program for 1988 to 1991 is outlined in the following sections. Emphasis has been placed on the gas corrosion studies part of the programs, because this is fundamental to

understanding the performance of material in the primary circuit of advanced HTGR systems.

9.2.3.1 Work plan for 1988

- Complete installation, leak checking, and checkout of gas corrosion equipment (small loops).
- Define gas chemistry ranges for minimal interactions with Inconel 617 and alloy 800H.
- Determine the critical temperatures (T_c) for 2.25Cr-1Mo, alloy 800H, Hastelloy X, Inconel 617) in several relevant simulated HTGR environments. Above T_c either rapid carburization or rapid decarburization occurs and, therefore, protective corrosion behavior is not possible.
- Map the thermal stability studies of advanced HTGR alloys such as chemistry-modified Inconel 617, Haynes 230, and Inconel 618.

9.2.3.2 Work plan for 1989

- Complete installation, leak checking, and checkout of gas corrosion equipment (large recirculating SHTGR helium loop, 6 large furnaces with 12 small and 2 large retorts).
- Define gas chemistry ranges for minimal interactions with Inconel 617 and alloy 800H.
- Develop basic gas/material interaction mechanisms for material interactions with simulated HTGR helium.
- Continue thermal stability studies of advanced HTGR alloys
 - Initiate exposures (650 to 1000°C) on material from a new heat of chemistry-modified Inconel 617.
 - Start evaluation of developmental alloys [ORNL, Federal Republic of Germany (FRG), and Japanese].

9.2.3.3 Work plan for 1990 and 1991

- Extend development of basic gas/material interaction mechanisms and models to include other materials such as intermetallics, ceramics, and composites.
- Complete thermal stability studies started in prior years.
- Initiate air creep-rupture tests of advanced alloys including U.S., FRG, and Japanese developmental materials.

- Perform creep-fatigue tests to assess optimum creep-fatigue interaction model.
- Begin mechanical properties and environmental compatibility testing of advanced materials (e.g., metal matrix composites, ceramics, and carbon/carbon composites).

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