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The Tower Shielding Facility
Its Glorious Past

F. J. Muckenthaler

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Computational Physics and Engineering Division

THE TOWER SHIELDING FACILITY - ITS GLORIOUS PAST

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In Memoria
To a Stately Lady

I would like to suggest
that we let these words be
the record of our efforts
for all the world to see.

The facility began
from the needs of a plane
to fly without rest
hither, yon, and back again.

Four towers were built
so stately and tall
to raise a reactor
away from it all.

Many shields were hung
between the towers with care
while neutrons and gamma rays
danced through the air.

Measurements were made
both inside shields and out
hoping to provide knowledge
of what shielding was about.

Pages of data were obtained
for changes here and there
that all helped to nurture
the art of shielding with care.

Now the need for the towers
seems long since passed
as we bring to a close
one fine facility at last.

The towers will be razed
the reactor dismantled
to return the good earth
as before it was scrambled.

History has been made
there's no doubt in my mind
its passing will leave
memories so kind.

F. J. (Buzz) Muckenthaler

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LIST OF ACRONYMS

AEC - Atomic Energy Commission
AI - Atomics International
ANL - Argonne National Laboratory
ANP - Aircraft Nuclear Propulsion
ASTR - Aircraft Shield Test Reactor
ATAC - Army Tank Automotive Center
BREN - Bore Reactor Experiment, Nevada
BSR - Bulk Shielding Reactor
CFRMR - Circulating-Fuel Reflector-Moderated Reactor
CRBR - Clinch River Breeder Reactor
DASA - Defense Atomic Support Agency
FFTF - Fast Flux Test Facility
GCFR - Gas-Cooled Fast Reactor
GE - General Electric
HPRR - Health Physics Research Reactor
HTGR - High Temperature Gas-Cooled Reactor
I&C - Instrument and Controls
ISTAF - Idaho Shield Test Air Facility
IVFS - In-Vessel Fuel Storage
JASPER - Japanese-American Shielding Program for Experimental Research
KTAM - Knappen-Tippets-Abbett-McCarthy
LiH - Lithium Hydride
LLFM - Low Level Flux Monitor
LMFBR - Liquid Metal Fast Breeder Reactor
LMR - Liquid-Metal-Cooled Reactor
LSPB - Large Scale Prototypic Breeder Reactor
MIT - Massachuseets Institute of Technology
MTR - Materials Testing Reactor
NACA - National Advisory Committee for Aeronautics
NBS - National Bureau of Standards
ND - Nuclear Data
NEPA - Nuclear Energy for Propulsion of Aircraft
NTA - Nuclear Test Aircraft
ORACLE - Oak Ridge Automatic Computing Logical Engine
ORNL - Oak Ridge National Laboratory
ORELA - Oak Ridge Electron Linear Accelerator
OTAC - Ordinance Tank Automotive Command
PC - Personal Computer
P&E - Plant and Equipment
P&W - Pratt and Whitney Aircraft Company
PNC - Power Reactor and Nuclear Fuel Development Corporation (Japan)
PoBe - Polonium-Beryllium
RRD - Research Reactors Division

ACRONYMS continued.

SM - Spectrum Modifier
SMR - Shield Mockup Reactor
SNAP - Space Nuclear Auxiliary Power
SS - Stainless Steel
STL - Space Technology Laboratory
TLDs - Thermoluminescent Detectors
TORT - Three-Dimensional Oak Ridge Transport Computer Code
TSF - Tower Shielding Facility
TSR - Tower Shielding Reactor
US-NRDL - U.S. Naval Radiological Defense Laboratory
CaO - Calcium Oxide
Na_2CO_3 - Sodium Carbinate
NaCl - Sodium Chloride
K_2CO_3 - Potassium Carbinate
SiO_2 - Silicon Oxide
BaO - Barium Oxide
WARD - Westinghouse Electric Corporation Advanced Reactor Division
TLD - Thermal Luminescent Detectors
HTS - Heat Transport System
FTR - Fast Test Reactor
DEMO - Demonstration Plant
IC - Ion Chamber
GAC - General Atomics Company
PCRV - Prestressed Concrete Reactor Vessel
IHX - Intermediate Heat Exchanger
NIS - Nuclear Instrumentation Systems
NaI - Sodium Iodide

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PREFACE AND ACKNOWLEDGMENTS

The Tower Shielding Facility (TSF) is the only reactor facility in the United States that was designed and built for radiation-shielding studies in which both the reactor source and shield samples could be raised into the air to allow measurements to be made without interference from ground scattering or other spurious effects. A preliminary proposal to build such a facility was presented to the Laboratory management in a central files memo (CF-52-4-85) from E. P. Blizard, C. E. Clifford, J. L. Meem, R. H. Ritchie, and A. Simon in April 1952 with the strong backing of H. A. Bethe. A second proposal was submitted after calculations by A. Simon and R. H. Ritchie showed that radiation scattered from the two tower legs in the design exceeded the upper limit that had been deemed acceptable by E. P. Blizard's group concerned with its design and use. The facility, as constructed, consisted of four towers, a design based on further scattering calculations of A. Simon and R. H. Ritchie. The responsibility for the design rested with the architect-engineering firm of Knappen-Tippetts-Abbett-McCarthy, with construction shared by the Charles Hobson Company and the Karl Koch Erecting Company, Inc. Construction began in 1953, and the first measurements were initiated in 1954 to start the ANP program. Although the ANP project was terminated in 1961, the remarkable versatility and usefulness of the facility extended its lifetime to become a valuable tool in resolving problems for other programs in shielding research. This history of the TSF's activities throughout these years (1953-1992) is based on the many documents that emanated from the work performed by the people involved.

The success of the TSF can only be attributed to a large group of people, both those employed by the ORNL and those from outside vendors, whose efforts may not be fully recognized in these acknowledgments, but it is important that their participation be duly noted. It is with great fear that I do this, however, lest someone's proper recognition be missing, and for that I humbly apologize.

The TSF was first headed by C. E. Clifford, who transferred from his responsibility of managing the Lid Tank Facility. His first technical staff included just four people: T. V. Blosser, L. B. Holland, J. L. Hull, and F. N. Watson, with E. McBee as secretary, all members of the Physics Division, and supported with part-time help from M. K. Hullings and J. Estabrook. Once the reactor became operational and the experimental program began, they were soon joined in mid-1954 by outside vendor personnel: D. L. Gilliland from GE, M. F. Valerino from the NACA, and J. Van Hoomissen from Boeing Airplane Company. These companies were also involved in the ANP program and had considerable interest in the planning of the experiments and results from analysis of the early data. As the use of the Tower progressed into its first year, added support in the form of calculations were rendered by M. D. Pearson on loan from Boeing, S. Auslender, J. B. Dee, J. Smolen, and C. A. Goetz from Pratt and Whitney, F. H. Murray, C. D. Zerby, J. C. Faulkner, and H. E. Hungerford from the Laboratory, and C. L. Storrs, from G.E., who, along with his duties at the Lid Tank Facility, served as contact for the work at the TSF.

That same year, two instrument technicians, E. D. Carroll and G. G. Underwood from the I&C Division, were added to the staff along with J. Money, a technician from the Physics Division. Early in 1955, a second technician, R. Wright, was added to assist J. Money in performance of the experiments. By mid-1955, D. L. Gilliland had returned to his parent company and F. J. Muckenthaler, and F. Sanders had joined the Tower personnel, the latter staying only a brief period on special assignment. Within several months, the addition of I. D. Conner to the staff increased the number of technicians to three. That same year, F. L. Keller had joined the division staff to do analysis of the Tower's experimental data, and he was soon joined by H. C. Woodsum of Pratt and Whitney and D. R. Otis of Consolidated Vultec in that same effort. G. Rausa from the Glenn L. Martin Company arrived at the Tower to aid in both the analysis and experiment. S. Auslender and

J. Dee from Pratt and Whitney, and H. E. Stern from Consolidated Vultec joined Keller in doing the analysis, thus increasing considerably the number of analytical people involved in the ANP Program.

By the first part of 1956, J. Van Hoomissen and M. Valerino had completed their stay at the Tower and returned to their parent companies. As was the case for most guest participants, their assignments were usually for periods of about a year. These vacancies were soon filled by C. R. Fink from the Glenn L. Martin Company and W. J. McCool from Pratt and Whitney who were most interested in the experimental side rather than doing calculations. However, the analytical effort at the Tower was given a boost with the addition of two new hires, V. R. Cain and M. J. Welch, both members of the newly formed Applied Nuclear Physics Division (formerly the Shielding Group within the Physics Division). The analytical work for the Tower experiments performed by people located outside that area was also bolstered with the addition of "on loan" employees: A. J. Futterer from Pratt and Whitney, R. M. Davis from the Glenn L. Martin Company, V. J. Sholund from Lockheed Aircraft, H. B. Hilgeman and L. A. Bowman from the Wright Air Development Center, and W. G. Blessing from Convair, San Diego. During this period, S. K. Penny and D. K. Trubey, both from the Laboratory, were also assigned to special projects within the analytical group.

Late in 1956, a group of about 15 engineers from Pratt and Whitney assembled at the Tower to install and expose a J-57 engine to radiation from the Tower Reactor. Two runs were made in the early part of 1957. Support for the Tower staff was increased by the addition of three technicians, G. M. Argo, C. C. Barringer, and F. E. Richardson. By the middle of the year, F. H. Greene from the Glenn L. Martin Company, W. E. Price from Pratt and Whitney, and W. L. Weiss and H. Clark from GE had joined the Tower work group. W. R. Champion from Lockheed Aircraft Corporation joined the TSF group, arriving with samples of concrete and dirt from the Lockheed area near Marietta, Georgia, to be exposed to neutrons from TSR-I. C. O. Zerby from the Laboratory joined the effort of the Analytical Group to calculate the dose rates from neutron captures in air.

The design of a new reactor for the TSF was initiated in 1957. Nuclear calculations for the reactor were being done by M. E. LaVerne. Fuel element development was being performed by the ORNL Metallurgy Division and the reactor controls were under design by the Reactor Control Group headed by L. C. Oakes. In 1958, safety system measurements were coordinated by J. E. Marks of the same group. Reactor core kinetics were being investigated by R. K. Adams, F. P. Green, and E. R. Mann, also of the I&C Division. The mechanical design of the reactor was done by members of a design group within the Engineering and Mechanical Division under the guidance of C. W. Angel and F. L. Hannon. The cooling system was developed by the Heat Transfer and Physical Properties Section of the Aircraft Reactor Experiment Division. Calculations of the gamma-ray heating in the lead around the core were performed by D. K. Trubey and by W. W. Dunn, who was on loan from the Wright Air Development Center. Critical experiments were being performed on mockups of the TSR-II by D. F. Cronin, J. K. Fox, and L. W. Gilley of the Critical Experiments Group. Calculations of the water flow through the reactor were initiated by J. Lewin. About the same time, the flow patterns were being physically verified by a group under the direction of H. W. Hoffman and W. R. Gambill of the Reactor Projects Division. G. J. Kidd, Jr. developed procedures to calculate the effect of loss of coolant accidents for the reactor. The spherical beam shield, to be used with TSR-II, was being designed by L. Byrnes of GE-ANP in Cincinnati and would be fabricated later at ORNL.

Personnel from Convair, Ft. Worth, headed by B. J. Workman, S. C. Dominey, and F. Malone, arrived at the Tower during the first half of 1958 to perform a series of experiments using their Aircraft Shield Test Reactor as a source. In that same year, O. S. Merrill, on loan from Convair, San Diego, joined the division's analysis group to work with F. L. Keller. D. R. Ward from the Laboratory, transferred to the Tower group to assist in the operation of the reactor and performance of the experiments.

By the middle of 1959, the experiments at the Tower were stopped and dismantling of the reactor had begun. J. Lewin, having been involved in the TSR-II design, joined the Tower staff. It was at this time that C. E. Clifford and F. L. Keller left the Laboratory for "greener" pastures in California. F. J. Muckenthaler replaced W. Zobel as head of the Lid Tank Facility. L. B. Holland was placed in charge of the TSF. T. A. Love and H. A. Todd, both members of the I&C Division, were assigned to the Tower. J. H. Wilson, from Lockheed Aircraft Company, was placed on loan to the Tower, and he became the only non-Laboratory person at the TSF during this transition times, (1959 to 1961), as the others had returned to their parent companies.

In 1960, while the TSR-II reactor system was being assembled at the Tower, G. deSaussure, E. G. Silver, and J. D. Kingston, moved a 300 kV particle accelerator to the Tower to make measurements that would provide confirmation of earlier obtained values for the calibration of the control rods (plates).

In 1961, S. K. Penny took part in the pre-analysis and the performance of the experiment of the Pratt and Whitney divided shield study. With the closing of the ANP Program nationally in June of that year, personnel from outside vendors attached to the program returned to their employers. Use of the TSF was switched to making measurements in support of work by Convair, Ft. Worth, the responsible vendor for the Ordinance Corps Tests for the Army. This effort was led by S. C. Dominey, J. R. Stokes, W. C. Farris, and R. L. Henry from Convair, and W. Riggle, R. R. McGregor, and R. F. Adams from the Army Tank Automotive Center. F. J. Muckenthaler returned to the Tower just to head the Tower's participation in the experiment. Pre-analysis and post-analysis of the in-air measurements that were part of the experiment were performed by R. E. Maerker, who had joined the Laboratory's Neutron Physics Division as F. L. Keller's replacement in the Analysis Group. In 1962, J. J. Manning joined the operating staff at the Tower and C. E. Clifford returned to the Neutron Physics Division (formerly the Applied Neutron Physics Division) in his former capacity as head of the Reactor and Weapons Shielding Group. H. E. Stern and F. H. Clark were placed on assignment to the Analytical Group to aid in the analysis of the data obtained at the Tower from experiments using shields designed to protect against radiation from prompt weapons. That same year, F. J. Muckenthaler rejoined the TSF staff to head its experimental program following participation in the operation BREN (Bore Reactor Experiment, Nevada) experiment at the Nevada Test Site. L. Jung joined him for a brief period to recalibrate some of the Lid Tank detectors. In 1963, V. V. Verbinski, H. A. Todd, and M. S. Bokhari (a United States Agency for International Development Fellow on loan from the Pakistan Atomic Energy Commission), made measurements of the fast-neutron beam spectrum emanating from the TSR-II. At about this same time, R. M. Freestone joined the Analysis Group to do calculations of fast-neutron dose rates behind slabs of water and concrete and K. M. Henry transferred from the Bulk Shielding Facility to bolster the advent of making neutron spectra at the Tower. F. B. K. Kam and J. G. LaTorre, Mathematics Division employees, joined F. H. Clark in doing Monte Carlo calculations for comparison with experimental results from TSF experiments. Through subcontracts, Convair/Ft. Worth employees R. L. French, M. B. Wells, L. G. Mooney, and N. M. Schaeffer performed analyses of TSF experiments performed earlier measuring the radiation penetrating concrete-walled structures and concrete-lined holes. In 1965, W. A. Coleman, a graduate student in the University of Tennessee Nuclear Engineering Department, joined R. E. Maerker in the Monte Carlo calculations of the differential energy-angle albedos determining fast-neutron differential dose rates. Coleman was joined later by his department advisor, P. N. Stevens, who for several years served as a consultant for the Shielding Group. That same year E. A. Straker joined the Analysis Group, doing preliminary studies on the SNAP Program.

In 1966, F. C. Maienschein became division director upon the death of E. P. Blizard and served in that capacity until he retired at the end of 1990. J. C. Courtney, a former graduate fellow, Oak Ridge Associated Universities, participated in the analysis of neutron scattering data from exposure of cylindrical samples to neutrons at the TSF that same year. Also in 1966, W. W. Engle and F. R. Mynatt, on loan from the Computer Technology Center at K-25, joined the Analysis Group working on the development of discrete ordinates methods for use in shielding calculations. They were joined in 1967 by L. R. Williams and W. E. Ford III, also on loan from the K-25 Computer Technology Center. That same year, H. C. Claiborne and M. Solomito from the Laboratory, participated in calculations for the TSF-SNAP reactor program using the discrete ordinates method. W. R. Burrus joined the Reactor and Weapons Shielding Group on a temporary basis to assist in development of the Tower's NE-213 neutron spectrometer and data unfolding program. Efforts in the SNAP Program were further increased by the participation of K. D. Franz and W. E. Ford III from the Computer Technology Center, M. B. Emmett of the Mathematics Division, and R. S. Hubner on loan from Atomics International.

In 1968, C. Y. Fu, a student from the University of Tennessee Graduate Program, participated in calculating the gamma-ray response of the NE-213 scintillator in use at the TSF. That same year, F. A. R. Schmidt, from the Institut für Kernenergetich, Universität Stuttgart, Germany, joined V. R. Cain in calculations of the fast-neutron transmission through lithium hydride slabs.

L. W. Gilley became a TSF staff member in 1969 to assist both in reactor operation and performance of the experimental program. M. L. Gritzner and R. J. Rodgers from the K-25 Computer Technology Center joined F. R. Mynatt in performing calculations of the radiation transport through slabs previously measured at the TSF using the SNAP reactor. R. S. Booth of the Laboratory participated in the experimental measurement of the gamma-ray spectra from thermal neutron absorption in ^{235}U and then performed the analysis. D. H. Wallace from the Computer Technology Center generated calculations of the secondary gamma-ray productions from exposure of slabs to the reactor neutron beam. D. C. Irving joined E. A. Straker in the evaluation of the total cross section for iron. J. W. Paul, from the Plant and Equipment Division, was appointed area engineer for the needs at the TSF.

In 1970, K. J. Yost and J. E. White participated in calculations for comparison with experiment of neutron-energy-dependent capture gamma-ray yields in ^{238}U. R. Q. Wright and G. W. Morrison from the Mathematics Division joined others in the calculation of neutron transport in iron and comparison with a TSF experiment. N. M. Greene, also from the Mathematics Division, participated in calculating the response functions for the Bonner ball neutron detectors to be put in use at the Tower. The following year, J. L. Rathburn, a representative from the Westinghouse Electric Corporation's Advanced Reactor Division, participated in the start of the ORNL-LMFBR shielding program for which experiments were being performed at the TSF.

B. J. McGregor, on assignment from the Australian Atomic Energy Commission, joined the TSF staff in 1972 to participate in the LMFBR experiments. R. L. Childs, a member of the Computer Science Division, joined others in calculations for the benchmark experiment for neutron transport in thick sodium. S. Uchida, on assignment from Hitachi Research Laboratory in Japan, combined with J. Lewin and R. E. Maerker in calculation of energy depositions and dose rates in the new reactor shield proposed for TSR-II. During that same year, V. Rado arrived from Italy to spend 12 months assisting in the experimental program. W. Zobel joined R. E. Maerker doing Tower data analysis. In 1973, D. L. Kirby, D. L. Reed, and W. E. Eldridge, graduate students at the University of Tennessee, spent about a year helping with the experiments. The following year, they were joined by a fellow student, F. J. Slagle. During this same period, T. A. Love joined TSF personnel to initiate development of a method using thermoluminescent detectors (TLDs) to measure gamma-ray heating in reactor shields. Beginning in 1974, W. Yoon, a contract assignee from the University of Tennessee, spent approximately two years at the TSF working on his dissertation in TLD development for a doctoral degree. That same year, E. M. Oblow, W. A. Rhoades, and D. E. Bartine of the Neutron Physics Division, G. Haynes from the University of Tennessee, J. R. Knight and J. V. Pace III from the Computer Science Division, T. M. Sims and J. H. Swanks of the Operations Division became participants in the calculations for the Fast Flux Test Facility (FFTF) studies at the TSF. In 1975, G. G. Simons from the Argonne National Laboratory at Idaho Falls, T. J. Yule from the Argonne National Laboratory in Chicago, and M. J. Driscoll and A. T. Supple from the Massachusetts Institute of Technology participated in a CRBR Experiment at the Tower using TLDs. For this experiment, R. K. Abele and his group from the I&C Division designed and fabricated a high-pressure ion chamber. That year, L. B. Holland and J. L. Hull were transferred from the Neutron Physics Division to the Operations Division that had been assigned responsibility for operation of the Tower Facility. In 1976, V. C. Baker of the University of Tennessee and T. J. Burns of the Laboratory provided calculations for the LMFBR Experiments. A. S. Makarious from the Atomic Energy Establishment of Egypt participated in an experiment at the Tower. J. N. Money left the TSF for a job at Y-12 and C. O. Slater became an employee of the division as part of the Analysis Group working on the TSF experimental data. With the departure of C. E. Clifford to Texas in early 1977, D. E. Bartine took over responsibility for coordinating the Tower's shielding programs, and division personnel at the Tower became the responsibility of R. W. Peelle. In that same year, D. T. Ingersoll joined the division and became an active participant in the shielding program with work on the design of a prototypic GCFR Grid-Plate Shield Confirmation Experiment. At about this time, R. M. Freestone retired and C. C. Webster inherited the job of unfolding the low-energy portion of the neutron spectra obtained at the Tower. It should be noted that throughout the 1970s, G. T. Chapman and G. L. Morgan provided valuable support for the NE-213 spectrometer program at the Tower and the unfolding of its data. Close to this same time, the integrity of fuel cask designs was being tested through a series of drop-tests conducted at the Tower under the supervision of L. B. Shappert and W. D. Box.

During the summer of 1980, J. R. Knight of the Computer Science Division joined C. O. Slater in analysis of the single-cell Neutron Streaming Experiment, and S. N. Cramer was a participant in the analysis of the GCFR Grid-Plate Shield Design Confirmation Experiment. J. O. Johnson co-authored with D. T. Ingersoll the report *A User's Guide for the Revised SPEC-4 Neutron Spectrum Unfolding Code for use at the TSF* that refined that procedure. With this report as a guide, he took over from Webster the unscrambling of the TSF data, only to be relieved of these duties several years later by J. D. Drischler. In 1985, W. A. Rhoades made an analysis of the data obtained from a special experiment run at the Tower to verify the TORT code that he had developed.

Throughout the Japanese-American Shielding Program for Experimental Research (JASPER) conducted at the Tower (beginning in 1986 and ending in 1992), N. Ohtani, K. Chatani, and A. Shono, served as representatives from the Japan Power Reactor and Nuclear Fuel Development Corporation (PNC) and actively participated in the work at the Tower. B. D. Rooney joined the TSF staff in 1986 for several years, leaving in 1991 to return to school. In 1989, J. V. Pace III assumed responsibility for the Reactor Shielding Group, replacing D. T. Ingersoll who became manager of the Nuclear Analysis and Shielding Section. Also in 1986, the Operations Division was changed to the Research Reactors Division (RRD), and operation of the Tower came under the direct supervision of R. L. Stover, facility manager. W. E. Hill became assistant facility manager and an additional reactor operator. With the shutdown of the Health Physics Research Reactor (HPRR), E. G. Bailiff joined the Tower staff as a reactor operator. That same year, calculations were initiated by J. T. Thomas, P. B. Fox, and C. M. Hopper to verify the safety in the methods of shipping and handling of the fuel pins to be used during the JASPER experiments at the Tower. R. R. Spencer transferred from the Oak Ridge Electron Linear Accelerator (ORELA) Group to the TSF staff at the start of 1991 and H. T. Hunter became a staff member in the middle of that same year, transferring from the Physics Division. In 1992, J. A. Bucholz of the Computing and Telecommunications Group made calculations for the analysis of the In-Vessel Fuel Storage (IVFS) Experiment in the JASPER Program. That experiment was followed by the IHX experiment that same year for which J. K. Dickens of the

ORELA Group provided technical support through the counting and determination of the induced activity in sodium foils.

L. S. Abbott served as a member of the Division staff from September 1948 to August 1986 in the capacity as editor of the many reports that were issued by the TSF staff and, for several of the programs, combined the information from the individual reports into summaries as single issues.

Throughout the years from 1954 to 1992, the Tower facility was served by a series of dedicated instrument technicians from the I&C Division, starting with E. D. Carroll and G. G. Underwood as mentioned previously. They hold the record for having served the longest. Others were added or served as replacements for their predecessors who were reassigned, changed jobs, or retired. W. Bird was added to the group as the third man. Their successors were: J. Wallace, R. E. Francis, F. Siler, F. Gallardo, N. Baird, G. Hamilton, C. McAmis, C. O. McNew, C. Littleton, G. Kwiecien, J. Llewellyn, L. Stansbury, and B. VanHorn. Several I&C area engineers were also assigned to help technically in the instrumentation service provided: R. D. Smiddie, H. A. Todd, D. D. Walker, T. Barclay, and J. A. Ray. J. McConnell and J. Todd were generous with their time in providing support for the NE-213 computer system and accompanying electronics.

Support craft from the P&E Division also played a vital and necessary part in the success of the Tower's operation. Of the many different craft groups within the division, only the electricians and millwrights assigned personnel to the Tower originally on a semi-permanent basis, and then only during the early years. The electricians so blessed were James Wallace and Bud Williams, followed later by C. W. Bean, D. L. Sexton, C. Hunt, M. Snow, and W. Haskell. Millwrights with that same distinction were W. G. Cuttrell, E. Clark, W. Bledsoe, D. S. Keller, and J. Arwood. Providing technical support as area engineers were J. R. Gissel, N. Dunwoody, J. W. Paul, G. Kern, H. Grimac, J. Reynolds, and S. Woods. The acknowledgments would not be complete without a special mention of the other crafts who were equally supportive, such as the riggers, iron-workers, machinists, boiler-makers, carpenters, lead-burners, and others, whose many names are beyond listing here but whose very presence and effort made operation of the facility that much easier.

The Tower was also served well by three long-timers, R. L. Clark, L. C. Johnson, and W. T. Martin Jr., from the Health Physics Group, assigned responsibilities for the Tower's personnel radiation welfare.

The Tower would not have been able to function without the secretarial help that many people so amply provided. As was mentioned earlier, E. McBee served for nearly a decade as the first secretary. [She was followed by Virginia Glidewell, J. Penn from 1962 to 1970, and then a mixture of part-time and full-time employees: L. Allen, E. Howe, M. Schulte, S. Raby, A. Alford, and ending with G. Marvin from 1986 to 1993.]

The author is indebted to D. T. Ingersoll under whose inspiration and guidance this report was written. He envisioned the report as not only providing a historical review of the many experiments undertaken within each of the programs served by the facility, but it would also serve to indicate the capability and versatility of the facility.

The content of this report was derived from the work described in reports authored by people who were involved in the TSF programs, people whose participation was indicated earlier in the acknowledgments. These sources were greatly appreciated.

Finally, the author would like to thank Computational Physics and Engineering Division personnel, A. Alford, G. Marvin, C. Householder, and S. Shriner for their assistance in typing the many drafts of the report, T. Henson and A. Taylor for successfully completing the final manuscript.

1.0 INTRODUCTION

In the early 1950s, the shield design for a nuclear-powered aircraft switched from the concept that the shielding should be concentrated around only the reactor to one in which the shielding should be applied to both the reactor and crew compartment, namely, a divided shield. The design for this new concept was proceeding, however, on the basis of calculations which were not only complex but were based on nuclear data pertaining to the single-shield concept. Much effort was being expended exploring different methods whereby reasonably accurate data could be obtained for development of the shielding design. It was soon realized that it would be most appropriate, and even necessary, to be able to test these concepts in full-scale mockups. This was not possible since the Laboratory facilities then in operation were not designed for full-scale divided shield experiments nor could they be easily converted. Thus it was recognized that the quickest and cheapest means for solving this dilemma would be to build a new facility in which full-scale shielding tests could be made.

It was proposed that a tall steel structure be erected that would support the suspension of both the power source and the crew compartment, with variable separation distances up to about 100 ft. The source (reactor) would be contained in a large steel tank filled with water in which the reactor could be selectively positioned. The crew compartment would be represented initially by a square steel tank filled also with water. Measurements would be made as a function of the reactor and detector positions.

This idea was presented to the then Atomic Energy Commission, and after further indepth study, a concept was approved in which the reactor and shield would be supported by a bridge truss atop two tower legs. When the design was completed and calculations repeated for radiation scattering from the towers, it was realized that the scattering from them and the bridge truss in between exceeded the design limits which had been previously established.

Further design considerations produced the tower structures that are still in use today. Four towers were erected in the form of a rectangle from which both reactor and simulated crew shields could be supported while keeping the radiation scattered from these towers below the established limitations. The design permitted the reactor and its shield/coolant to be raised at least 200 ft above the ground between two of the tower legs while providing independent movement of the crew compartment mockups between the other two towers. Spacing between the reactor and crew compartment was variable from about 30 ft to 100 ft.

A control house was built underground adjacent to the towers to house the reactor controls, operating personnel, and data-taking electronics. The first reactor was of the same design as for the Bulk Shielding Reactor, MTR-type fuel elements of a ^{235}U/Aluminum sandwich placed in a rectangular array. It was cooled by a tub of water in which the reactor resided as it was removed from the storage pool and raised into the air. The maximum thermal power at which the reactor could be operated was limited to 500 kW.

As the ANP program progressed, considerable difficulty was encountered in attempting to use the Tower data to design the airplane shield that would contain a different (spherical) reactor source, and it was felt that the only way to be successful was to use a test source for the Tower data whose spectrum was similar to that of the proposed source. This would mean building a new reactor having beryllium instead of water as a reflector.

A reactor designed to meet the new specification was presented to Laboratory management, who made a counter suggestion. Instead of designing a project-specific reactor, design a reactor that was capable of providing different spectra by altering the reactor's shielding. This would allow the source to be tailored to the desired spectra for many different shield studies. Furthermore, the reactor emission should be as isotropic as feasible, which suggested it be spherical in shape. The suggestion was readily accepted by the shielding group and the resulting design became known as TSR-II. It first reached criticality on March 26, 1960, but many changes to the design followed, and it was not until February 19, 1961, that it operated at its approved power limit of 100 kW. It was eleven years later, January 1972, that approval was given to raise the power level to 1 MW, the maximum power it would reach during its lifetime.

During the lifetime of the Tower, two other reactors were also operated on site. In the late 1950s, the ASTR was operated as part of the General Dynamics/Fort Worth experimental program at the TSF. In 1965, a modified version of the SNAP-2 reactor was installed as a source for experiments in both the SNAP and LMFBR programs. The TSR-II was shut down on October 1, 1992, following completion of the Japanese-American program, JASPER, and placed in standby, with dismantling of the reactor presumably scheduled for about 1995.

The TSF proved its usefulness as many different programs were successfully completed. Following the closing of the ANP program in 1961, it became active in work for the Defense Atomic Support Agency (DASA) Space Nuclear Auxiliary Power, Defense Nuclear Agency, Liquid Metal Fast Breeder Reactor Program, the Gas-Cooled and High-Temperature Gas-Cooled Reactor programs, and the Japanese-American Shielding Program of Experimental Research, just to mention a few of the more extensive ones.

The history of the TSF as presented in this report describes the various experiments that were performed using the different reactors. The experiments are categorized as to the programs which they supported and placed in corresponding chapters. The experiments are described in modest detail, along with their purpose when appropriate. Discussion of the results is minimal, but references are given to more extensive topical reports.

2.0 MOTIVATION AND DESIGN

2.1 PREFATORY

In 1946, the United States Air Force awarded to the Fairchild Engine and Airplane Corporation the contract which established them as the responsible organization for directing the NEPA project.[1] One of NEPA's concerns was the design and fabrication of a crew compartment that would provide adequate protection for personnel against neutron and gamma-ray radiation during flight of the aircraft. At about the same time, ORNL was placing into operation a facility, designated the Lid Tank Facility, that used a beam hole through the Graphite Reactor shield as a source of neutrons into which shield samples could be placed and behind which measurements of the penetrating radiation could be made. Since this facility was started to study the feasibility of using mobile reactors for sea vessels, the Laboratory was asked to cooperate in NEPA's needs, and with each year the amount of participation increased, with more and more NEPA personnel becoming actively engaged in activities at ORNL.[2] Because of this participation, the AEC, who had joined together with the Department of Defense and the National Advisory Committee for Aeronautics to promote the development of a nuclear-propelled aircraft, designated ORNL in September of 1949 as the agency for carrying on the AEC's share of the joint technical program. The Laboratory was asked, in general, to study the neutronics aspects of the project. One of the four categories considered by the Laboratory, in which most of the experimental ANP work would fall, was that of reactor shielding.[3] By 1950, there was such a decisive shift in the cooperative effort of combining the NEPA-ORNL working groups that in 1951 the NEPA facilities at K-25 were closed, with most technical groups transferring to ORNL.

However, by late 1949, it was realized that another testing facility[4] was necessary in addition to the Lid Tank Facility to augment the quality of data by extending the range of shield thicknesses and including testing of more diverse mockups. It was felt that this facility, to be called the BSR, should have a source of sufficient strength so that neutron and gamma-ray fluxes were intense enough to make spectroscopy measurements beyond attenuations of 10^8. This increase in source strength, about a factor of 1000 over that of the Lid Tank, would be adequate to provide measurements not only for attenuations estimated for the ANP program, but would also be adequate for those anticipated in submarine reactor studies. The facility[5] was completed near the end of 1950, and the first experiments initiated in early 1951.

2.2 MOTIVATION

By 1952, designs for the nuclear-powered aircraft were switching from the unit shield concept where all the shielding was around the reactor, to the divided shield concept,[6] where separate shielding would be provided for both the crew compartment and the reactor power plant. The divided shield studies were relying on calculations which were not only complex, but were based on data that had been obtained from experiments using a unit shield. Properties of a divided shield had not been tested through physical mockups, and thus calculations were being made based on data with unknown uncertainty. Since the weight and distribution of reactor shields would grossly affect the shape and size of the aircraft, it was imperative that the shield design be accurate and timely. Such data from full scale mockups of divided shields were not possible because the present facilities were not designed to do such experiments. It was decided that full scale mockups provided the surest, fastest, and most economical way to provide the needed data and the only way to make these measurements was through the building of an appropriate facility. This effort resulted in what was to become the TSF.

2.3 INITIAL FACILITY CONCEPTS

Considerable time and effort was expended exploring possible designs to obtain accurate data for verification of calculational methods suitable for development of a divided shield program. As was mentioned earlier, it was not possible to obtain this data from either the Lid Tank or Bulk Shield Facilities since they were not designed for full scale divided shield-type experiments. However, considerable benefits were derived from experiments at the BSR that provided supportive information as to the reactor power necessary and the physical configurations required for clean measurements.

Several proposals[7] were considered including one which was called the "double igloo" or "half space" experiment, in which the level of the ground would be coincident with the horizontal midplane through the shields (Fig. 2.1). In this case, the radiation in the upper half space would be investigated and the measurements multiplied by two. Unfortunately, continued studies showed that an experiment using this concept would not really yield interpretable information since the radiation leaving the reactor would not necessarily be radial, so that some paths intercepted by the ground would not be included in the simplified half-space correction. Furthermore, the radiation scattered from the ground would not only be an important fraction of the total, it would be nearly impossible to measure directly. For a highly asymmetric shield, the ground contribution could have been prohibitively large. A cost estimate was made which turned out to be nearly as great as what it would be for the selected design.

One of the more "far-out" proposals was the consideration of a nearby canyon, over which the shield and reactor would be suspended (Fig. 2.2). The closest canyon that seemed acceptable was outside the confines of the government-owned property in this area. The cost of purchase would have outweighed other savings and the inconvenience of utilities and craft support was not acceptable.

The first tower-type proposal[8] to be seriously considered called for a tall steel bridge-type tower from which the two components of the divided shield could be suspended. The distance between the two components could be varied from 60 to 100 ft. The reactor shield tank would be cylindrical, 12 ft in diameter and approximately 14 ft high with a hemispherical bottom. The crew compartment would be represented by a 5-ft^2 tank of water that was 6 ft deep.

As originally conceived, the tower was to be of steel, bridge-type construction as shown in Fig. 2.3. It was to have two vertical, rectangular-shaped members approximately 20 × 40 × 200 ft high, and a rectangular-shaped bridge approximately 20 × 40 × 200 ft long affixed to the two vertical members at the top to form a completely integrated tower structure. A gantry-type structural extension was to be integrated into the middle of the cross bridge structure and cantilevered out on each side to accommodate the hoist equipment for both the reactor shield tank and the simulated crew compartment.

Beneath the suspended reactor shield was to be constructed a concrete-lined reactor handling pool of water (Fig. 2.3) to provide storage for the reactor when not in use and to

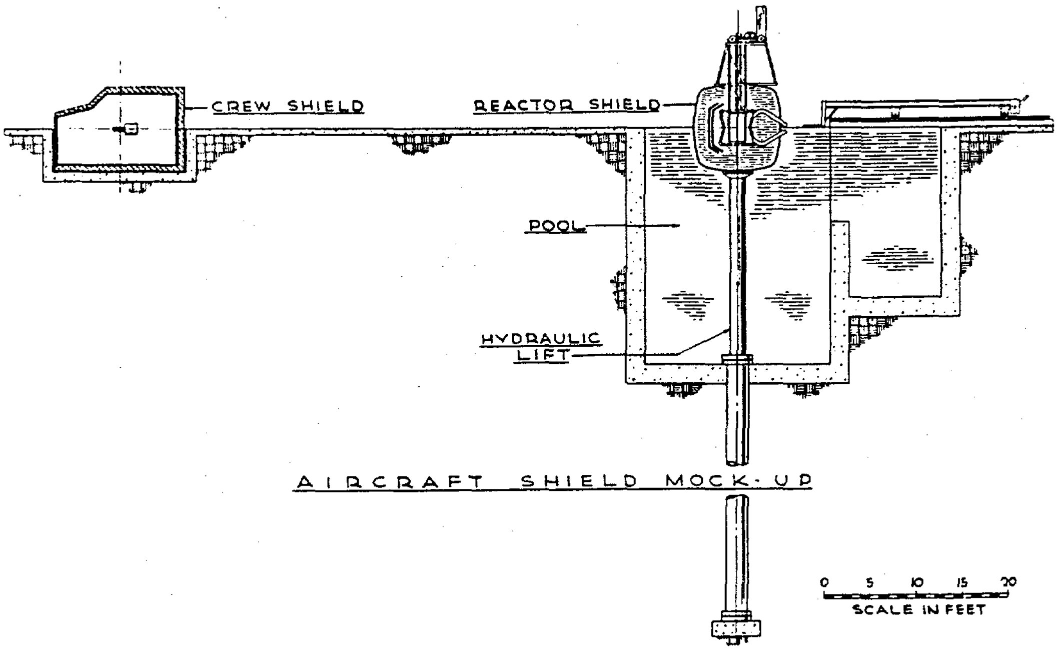


Fig. 2.1 Half Space Proposal for Aircraft-Shield Mockup.

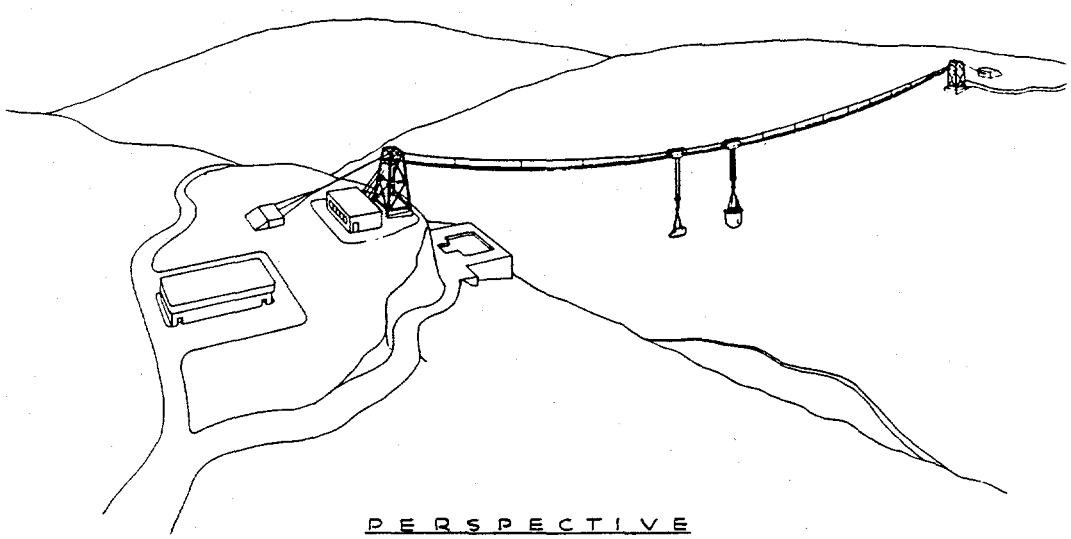


Fig. 2.2 Suspension Proposal for Aircraft-Shield Mockup.

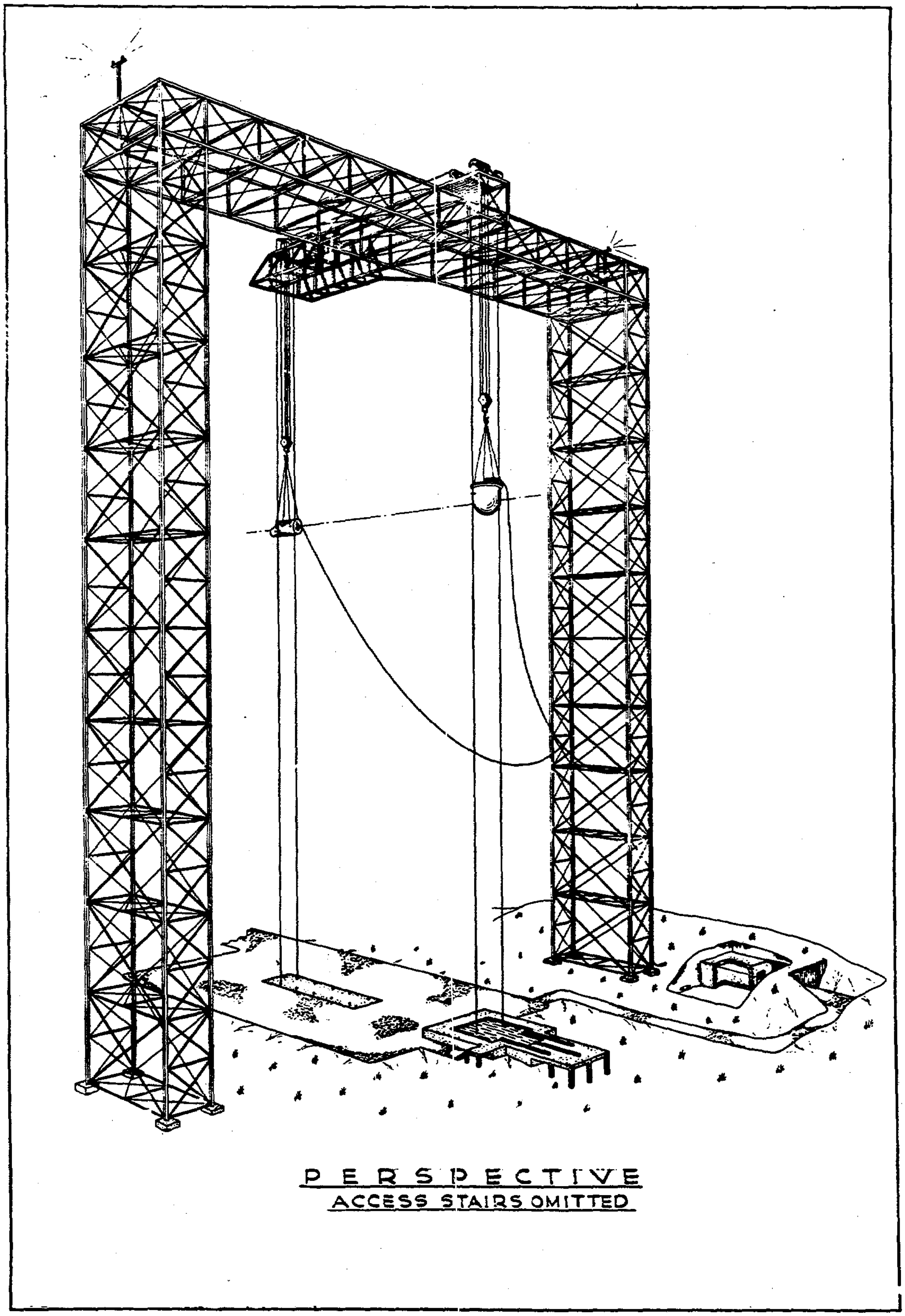


Fig. 2.3 Tower Proposal for Aircraft-Shield Mockup.

permit removal and storage of fuel assemblies. The large part of the pool was to be 20 ft sq and 25 ft deep, with a connecting smaller section approximately 10 ft sq and 18 ft deep.

Removal of the reactor from the shield tank was to be by a loading gantry mounted on rails along the top edge of the small section of the pool.

The reactor shield tank[9] and the crew shield tank were to be guided from the top of the cantilevered tower structure to the ground area by two cables for each shield, and would be raised and lowered by two power winches located on the bridge portion of the tower structure. The tower structure was to be designed so that it would be capable of supporting up to 125 tons if it became necessary.

The reactor assembly was to be moved horizontally in the reactor shield tank by means of a remotely operated gear train mounted on a dolly which would rock on rails affixed to the top of the tank. The water level was to be maintained at 6 ft above the reactor.

Control instruments and motors for operating the control and safety rods, fission chamber, and reactor dolly would be mounted on the reactor superstructure. Flexible cable leads were to be suspended from the reactor and attached to a point approximately 100 ft above the base of one of the tower legs. A reinforced concrete building (Fig. 2.4), approximately 117 × 32 × 12 ft, would be constructed into the side of the hill adjacent to one of the tower legs to provide offices, reactor controls, data collecting, and other services. It would provide adequate radiation protection for operating personnel. At that time it was estimated that it would take ten people to operate the reactor and conduct the experiments. The total cost for design and construction was estimated to be $1,294,000. It was proposed that the Laboratory prepare and furnish design criteria to an architect-engineer and that the work be performed by a construction contractor. Six months, beginning in June 1952, were allotted for design, with construction to begin in January 1953 and be completed by December.

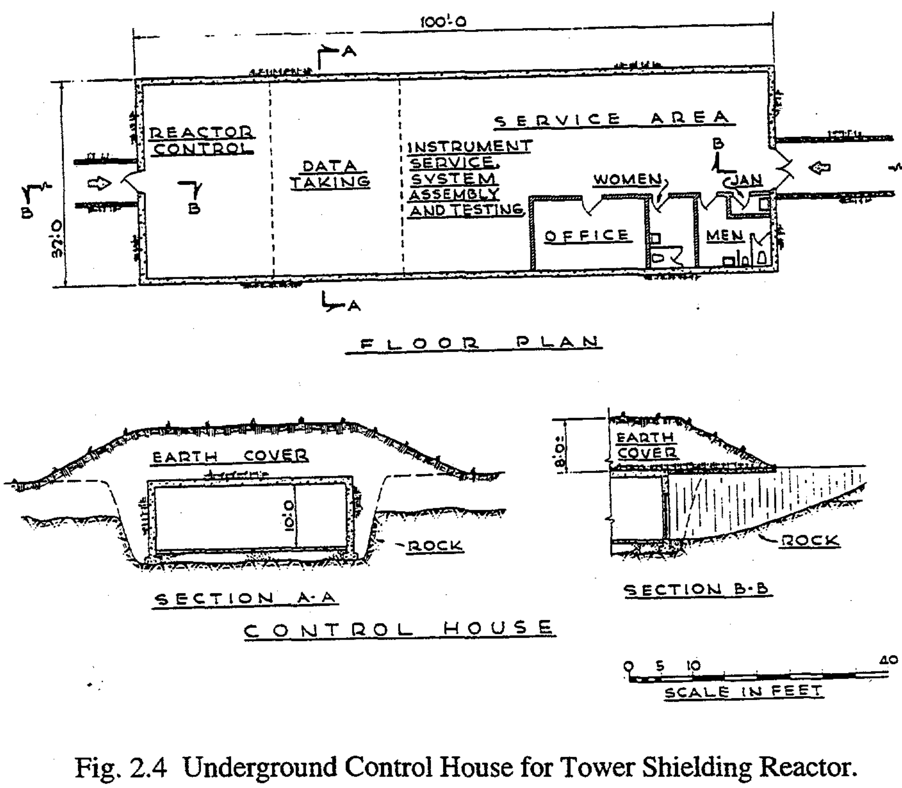


Fig. 2.4 Underground Control House for Tower Shielding Reactor.

2.4 FINAL DESIGN

The TSF concept[10] described above was submitted to the architectural engineering firm KTAM for final design. Their original design was accepted by ORNL in early January. However, before any work could be done, some additional scattering calculations by A. Simon and R. H. Ritchie indicated that the presence of the heavy structural steel in the long overhead members would raise the background from neutron scattering to beyond the limit that had been established as acceptable from earlier design studies. Since the most obvious requirement of the facility was that of reducing the ground-scattered and structure-scattered radiation to a suitable low fraction (recommended to be 15%) of the air-scattered radiation source, changes in design were needed. Many changes were considered before settling upon a scheme following additional calculations and more astute analyses of the background effects.

On the basis of some albedo measurements, it was determined that a height of 200 ft would be necessary to minimize ground scattering effects. Other experiments were done to obtain a measure of scattering from a support structure itself. Based on these studies, the original design was radically altered, eliminating the overhead structure and thereby reducing the weight of the legs by a factor of approximately two. A perspective of the new design is shown in Fig. 2.5. The new structure would consist of four towers 315 ft high, having a 200 ft inside clearance between tower legs in the north-south direction and a 100 ft clearance between tower legs in the east-west direction. The tower columns were to be 9 ft sq and weigh approximately 400 lbs per running foot, with the whole structure guyed at the top. Calculations by ORNL physicists showed that the contribution from the front tower legs would be 5.6%, from the rear tower legs 5.4%, from the upper trusses 1.2%, and from the truss at 100 ft for instrument maintenance was 0.88%, for a total of 13.08%, well within the suggested 15% upper limit.

It was thus concluded after such an elaborate investigation that this facility, as designed, would indeed afford the surest, fastest, and cheapest method for conducting full-scale shielding experiments which would provide reliable data for optimum design studies of the divided-shield concept. The first tests that would be conducted would be a measurement of the radiation leaving the reactor shield, followed eventually by measurements of the flux entering selected mockups of a crew shield design.

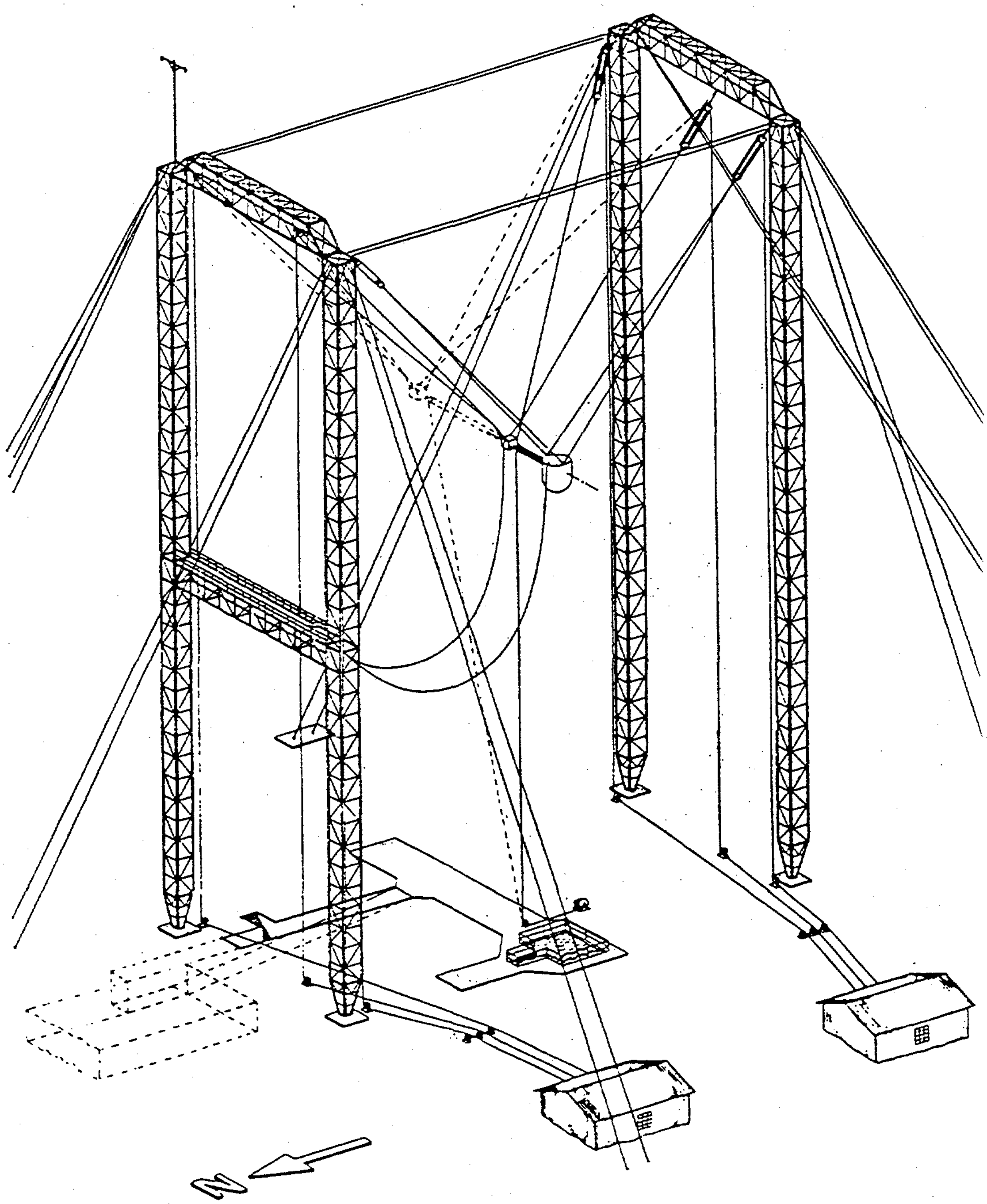


Fig. 2.5 Final Proposal for Tower Shielding Facility.

2.5 REFERENCES

1. W. E. Thompson, "History of ORNL (1943-63)," ORNL/CF-63-8-75, LRD-RC (August 1963).

2. C. E. Clifford, "The ORNL Shield Testing Facility," ORNL-402 (1949).

3. *Quarterly Progress Report for Aircraft Nuclear Propulsion Project and General Reactor Technology,* ORNL-528 (November 30, 1949).

4. W. M. Breazeale and E. P. Blizard, "Proposal for a New Bulk Shield Testing Facility," ORNL/CF-49-12-68 (December 1949).

5. *The New Bulk Shielding Facility at ORNL,* ORNL-991 (January 1951).

6. E. P. Blizard et al., "Tower Shielding Facility Preliminary Proposal Number 181," ORNL/CF-52-3-125 (May 1953).

7. E. P. Blizard et al., "Proposal for a Divided Shield Testing Facility," ORNL/CF-52-4-85 (April 1952).

8. *Oak Ridge National Laboratory Tower Shielding Facility, Proposal Number 181-A,* ORNL/CF-53-1-286 (January 1953).

9. *Quarterly Progress Report for Aircraft Nuclear Propulsion Project,* ORNL-1227 (March 1952).

10. *Quarterly Progress Report for Aircraft Nuclear Propulsion Project,* ORNL-1515 (March 1953).

3.0 CONSTRUCTION OF THE TOWER SHIELDING FACILITY (TSF)[1]

3.1 SCOPE OF WORK

The preliminary design premises were established by physicists and engineers at ORNL, who were also responsible for the final design criteria and the definitive designs. ORNL furnished design criteria for the towers, hoists, and tower foundations; developed the definitive design for the control house, reactor tank pool, reactor with associated instrumentation and shielding, access road, and utilities; procured hoists for installation by a contractor; fabricated and installed the reactor with associated instrumentation and shielding; and performed miscellaneous work to assist the contractors in the completion of the facility.

The AEC made arrangements with the United States Corps of Engineers to perform site exploration work; award contracts for Titles II and III engineering services for the towers, tower foundations, and hoists; award contracts for construction of the tower foundations, control house, hoist house, reactor tank pool, and access road; and award contracts for the fabrication and erection of the tower structure.

3.2 DESCRIPTION OF WORK

3.2.1 Location

The facility is located at an elevation of 1066 ft above sea level on the crest of a knoll which projects from the south side of the main Copper Ridge range, approximately 2.25 miles south-to-southwest of the ORNL plant area (Fig. 3.1). The site has steep-sided valley to the east, southeast, south, and southwest directions. The terrain surrounding the area varies in elevation from about 900 ft above sea level to about 1100 ft at a point 800 ft north of the facility center.

The nearest inhabited area at the time of selection was approximately one mile south and was separated from the facility area by the Clinch River. The site was selected after detailed considerations regarding the topographical surface, accessibility, and nuclear safety. Access to the area was provided by improving 1.75 miles of previously existing road which extended from the White Wing Road to the tower site. The work involved minor relocation of parts of the road along with some widening.

Initially, the site was rough graded to provide the approximate elevations and to permit easier access for the erection equipment and for the storage of material. Figure 3.2 shows a view of the site looking south-southeast from a knoll north of the entry road. Excavation and back-filling were required for the placement of the tower foundations and guy anchors and for construction of the control house, the reactor tank pool, and the hoist house.

The fencing required for this project consisted of installing 9400 linear ft of barbed wire perimeter fencing surrounding the site on a radius of 3000 ft, clearing the right-of-way and installing approximately 1800 linear ft of chain link exclusion fencing on a radius of 600 ft. The installation of fences, guard posts, and lighting were done in accordance with AEC security regulations.

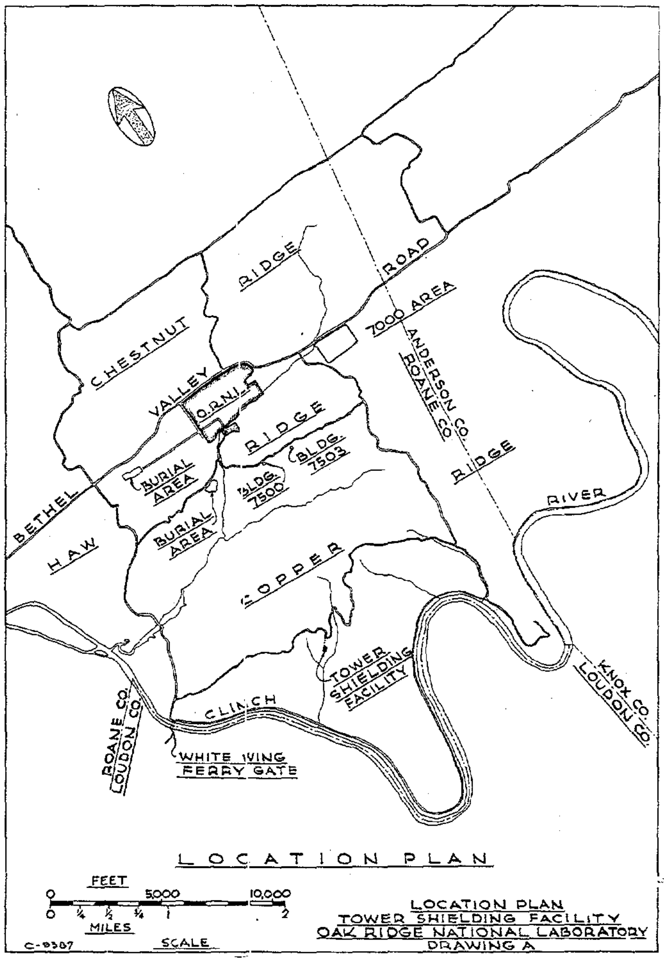


Fig. 3.1 Location Plan for the Tower Shielding Facility.

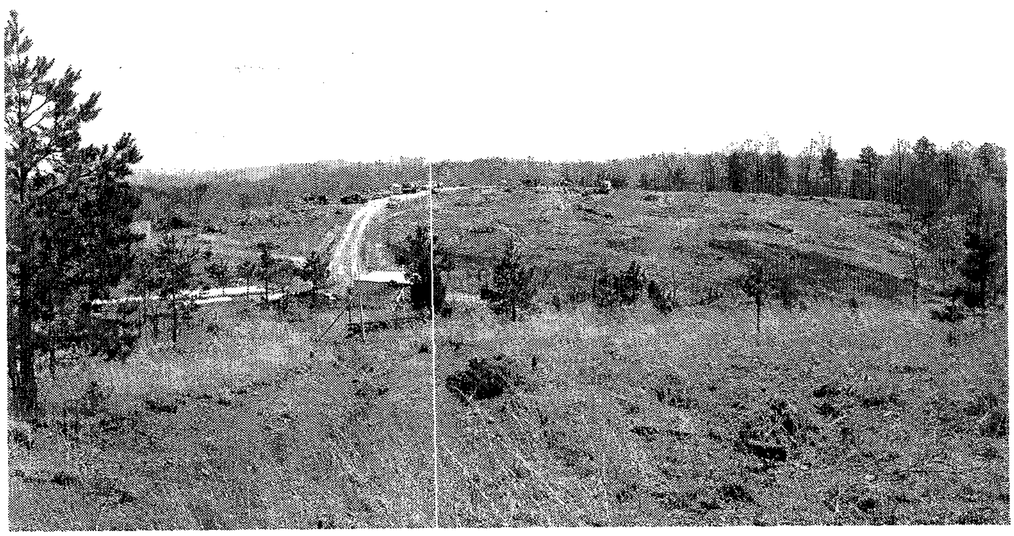


Fig. 3.2 Initial Clearing of the Tower Shielding Facility Area.

Electrical, water, and telephone services were extended along a cleared right-of-way from the 7500 area across Copper Ridge in the X-10 area to the tower site. Electrical services consisted of the installation of a 13.8 kV transmission line and equivalent kV transformer substation at the site with power distribution to planned areas. Water service was provided by a 4-in. galvanized steel water line with accessory equipment, including a concrete surge tank located at the 1250-ft elevation point on Copper Ridge at a point approximately 800 ft from the tower site. A site plan is given in Fig. 3.3.

3.2.2 Facility Description

The tower structure consists of four similar A7 carbon steel towers placed in a 200 × 100 ft rectangle facing due north with the 200-ft dimension forming the longitudinal axis of the facility. Each tower consists of a 9-ft-sq lattice column 315 ft high, terminated on the bottom by an inverted truncated pyramid tied to a concrete base (Fig. 3.4). Both sets of towers, spaced 100 ft apart in the latitudinal direction, are connected on top by a horizontal truss-type bridge. An additional bridge is located between the two northern towers at the 100 ft level to provide access for maintenance. Both levels in the northeast tower are reached via either an elevator or ladders secured inside the tower leg. The two sets of towers in the longitudinal direction are connected at the top by a pair of 2-in.-diam galvanized plow-steel cables. Each cable of the pair was designed to be held in 40,000-lb tension by two similarly constructed pairs of guys which extend from the top of the towers to ground anchors in the north-south direction. All four towers are similarly guyed in the east-west direction by eight inclined guy cables (two per tower). Each of these cables was originally set at 76,000-lb tension and they make an angle of 32° with the towers. Figs. 3.5 and 3.6 show a typical guy anchor under construction and when completed. The eight guy anchors are rectangular concrete blocks of reinforced concrete 20 × 12 × 8 ft thick, with two 10-in.-wide flange H beams embedded in the block for guy cable ties.

The foundation for the four tower legs are square reinforced monolithic concrete footings approximately 12 ft sq by 3 ft thick supporting piers 6 ft sq by 3 ft high. Fig. 3.7 shows one of the tower foundations during construction, while Fig. 3.4 shows the finished support. Embedded in each pier are eight 1-in.-diam bolts and one 3-in.-diam anchor pin.

The control house is a rectangular-shaped, reinforced concrete structure whose outside dimensions are 118 × 33 × 11.5 ft, having 18-in.-thick walls and roof. The structure is completely underground, with 3.5 ft of earth over the roof. Entry is through a partially covered tunnel from the north and a rampway from the area between the tower legs on the south. The building, shown during various phases of construction in Figs. 3.8-3.11, is located on the north side of the four towers, placed equidistant between the northernmost towers. Offices, reactor controls, data equipment, and associated services were located in the building, as shown in Fig. 3.12.

The equipment for hoisting the reactor and shield mockups is located in a structural-steel-framed lean-to-type structure, 135 × 36 × 17 ft with corrugated siding and roof. The building houses four 11,000-lb and two 12,000-lb line-pull hoist drums and motors that can be operated in pairs or singly. Two of the hoists, which can lift 55 t to an altitude of 200 ft, are used to move the reactor in the plane between the two tower legs on the west side of the rectangle. Another pair of hoists can raise a load of 40 t to the same height in a plane 35 ft east of the reactor plane. The third pair can lift 30 t in the plane of the east tower legs or when combined with the second pair of hoists, the load can be positioned from 100 ft to within 35 ft of the reactor plane. A photograph of the building can be seen in the left of center in Fig. 3.13 and a photograph of the cable drums inside the building is shown in Fig. 3.14. The six lifting hoists, purchased from the Whiting Corporation, are all of the single drum,

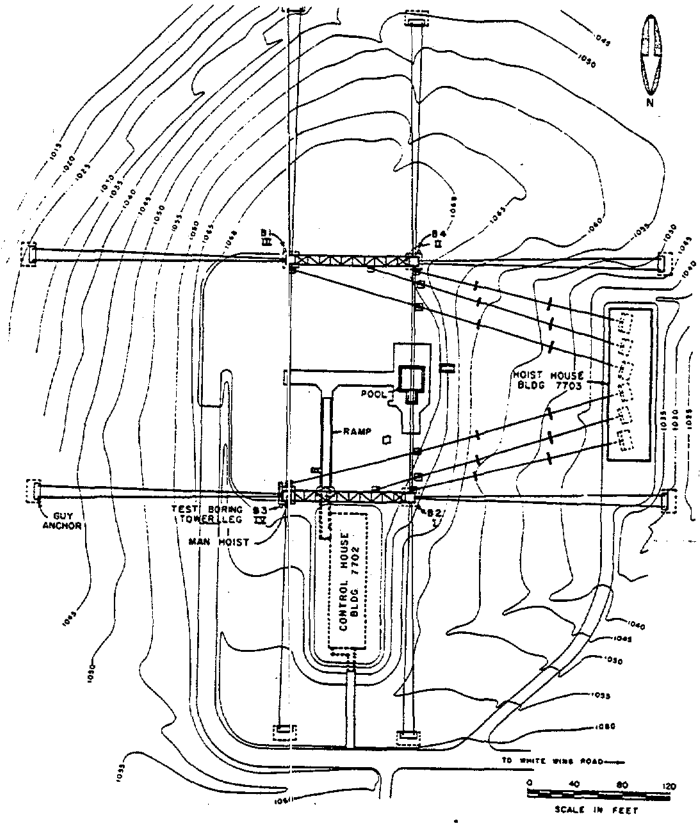


Fig. 3.3 Site Plan for Placement of Elements.

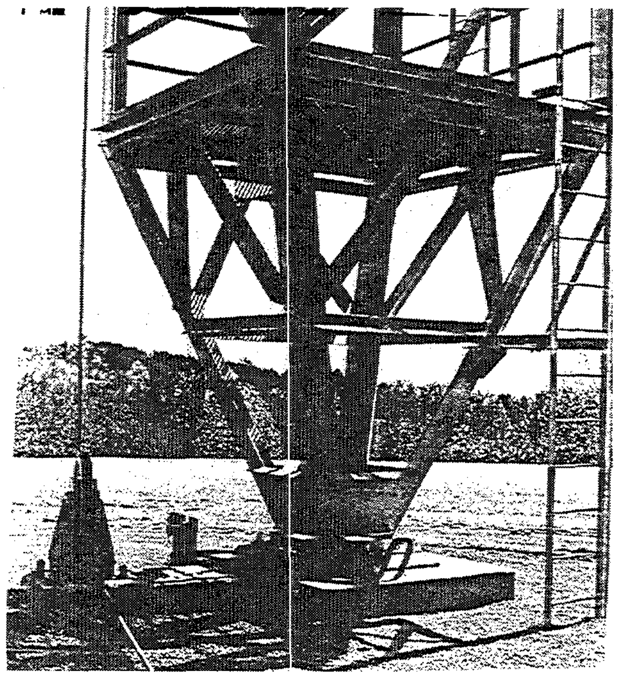


Fig. 3.4 Typical Tower Base.

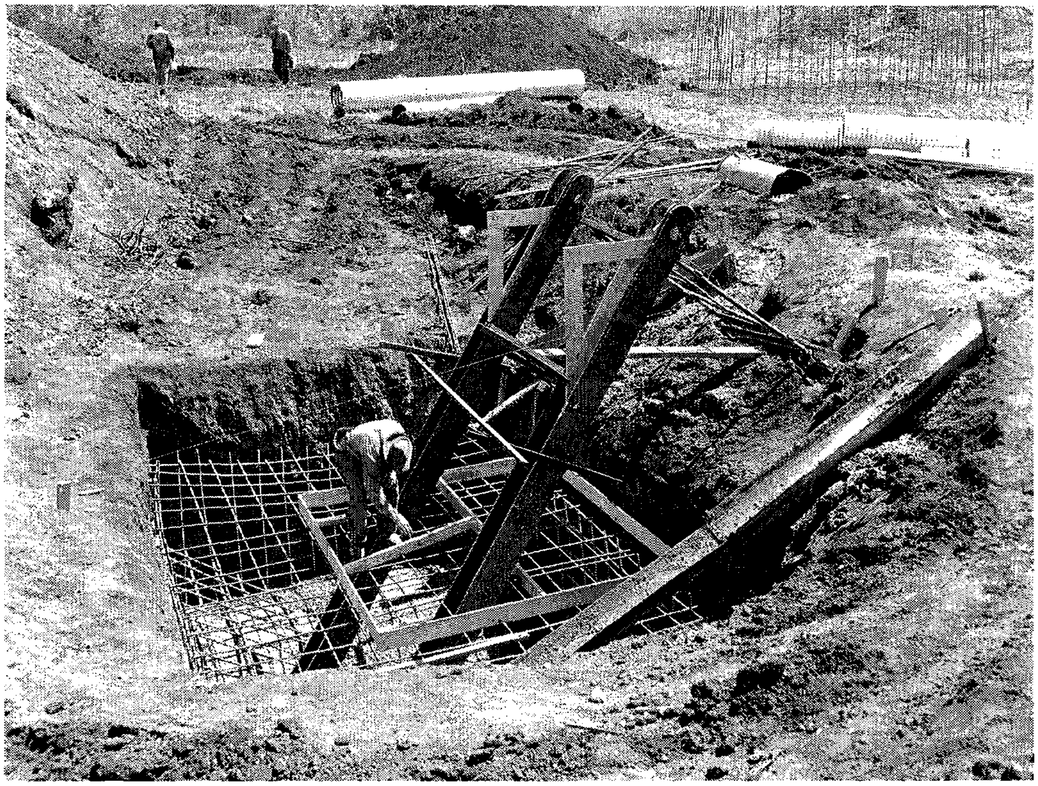


Fig. 3.5 Typical Guy Anchor under Construction.

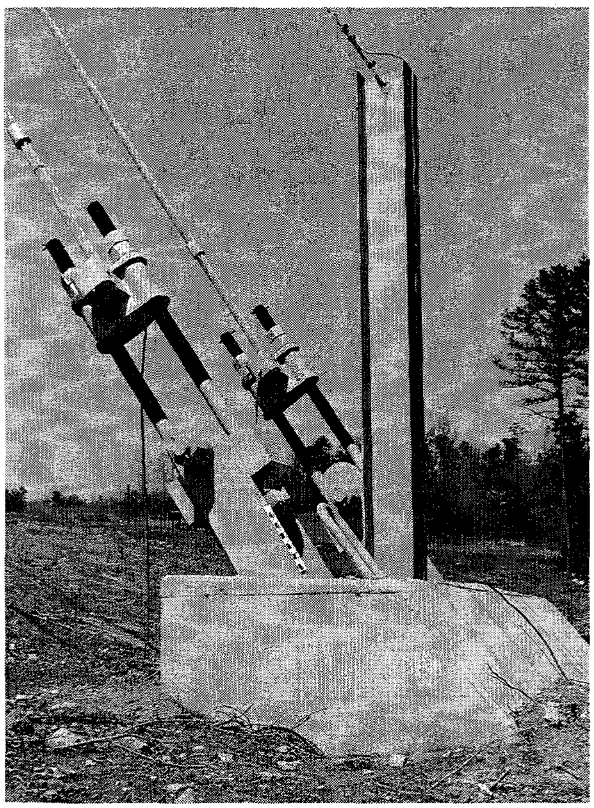


Fig. 3.6 Completed Guy Anchor.

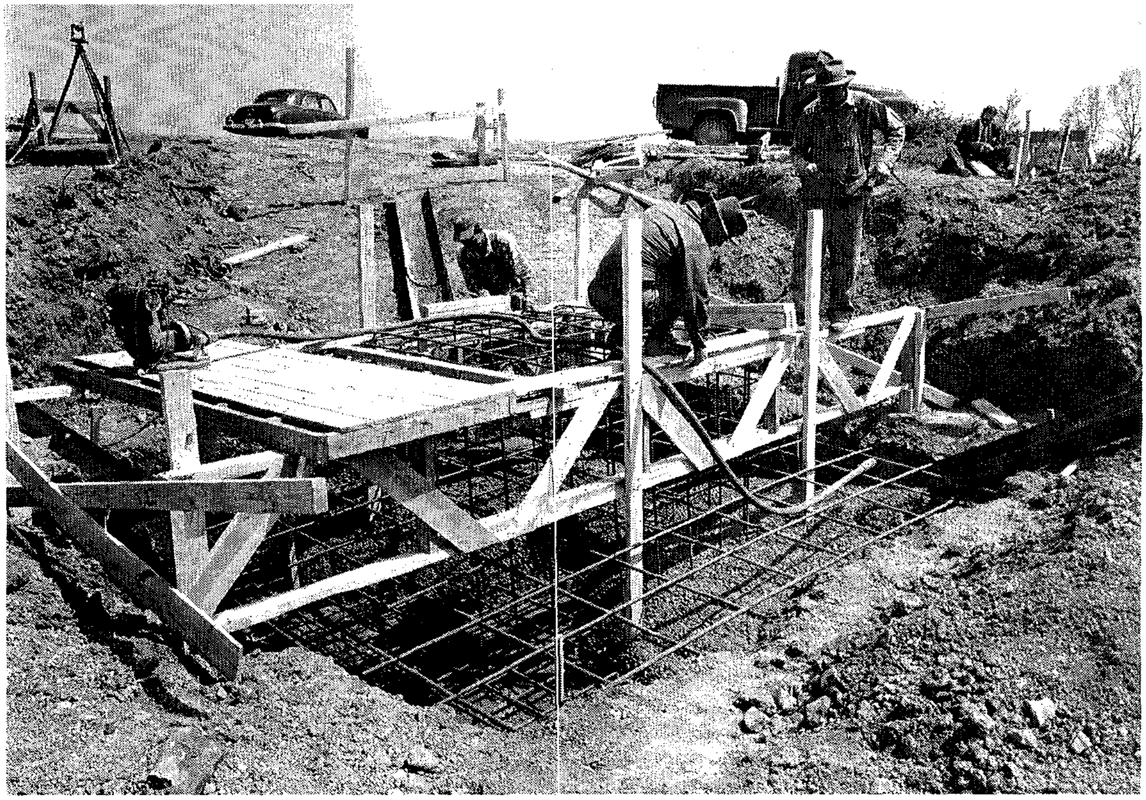


Fig. 3.7 Typical Tower Base under Construction.

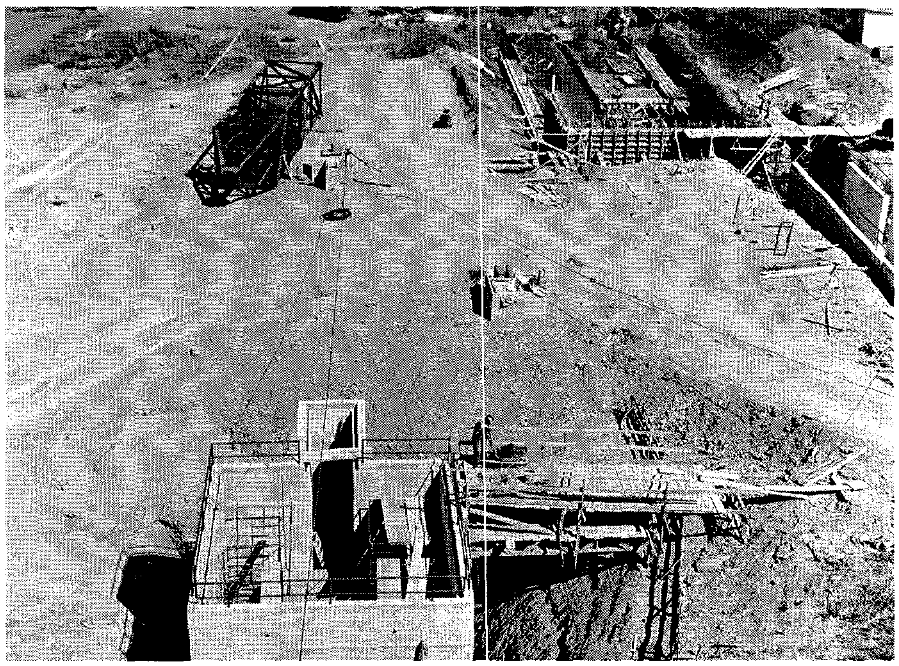


Fig. 3.8 Reactor Concrete Pool and Control House under Construction.

Fig. 3.9 Construction Continues at the Tower Shielding Facility.

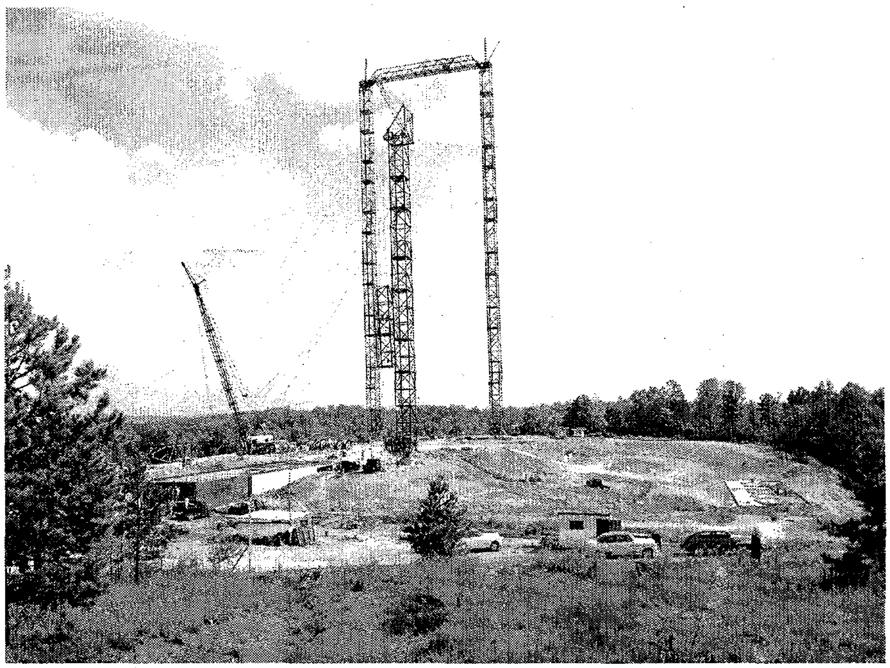


Fig. 3.10 Midway through Tower Legs Construction.

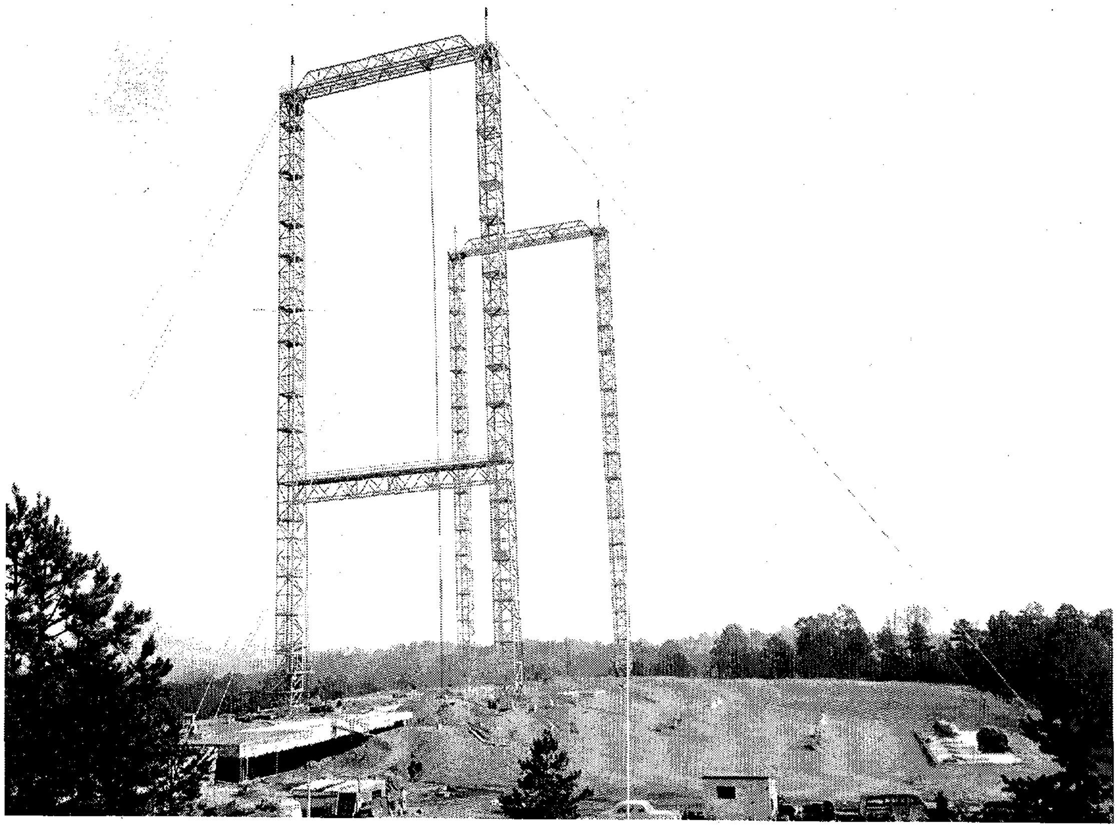


Fig. 3.11 Another Step in Construction.

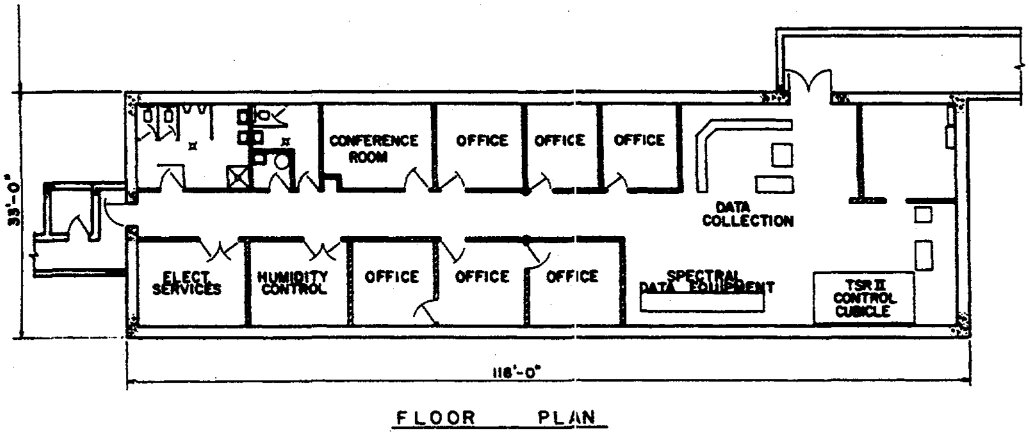


Fig. 3.12 Floor Plan for Control House (1954).

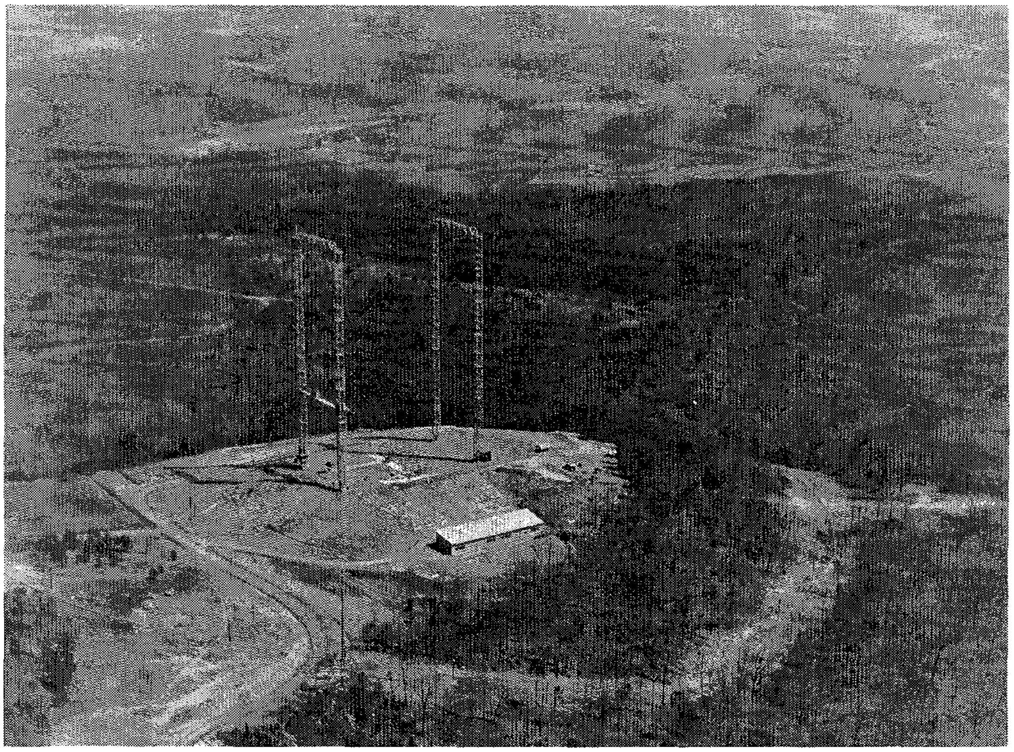


Fig. 3.13 Aerial View of Tower Area.

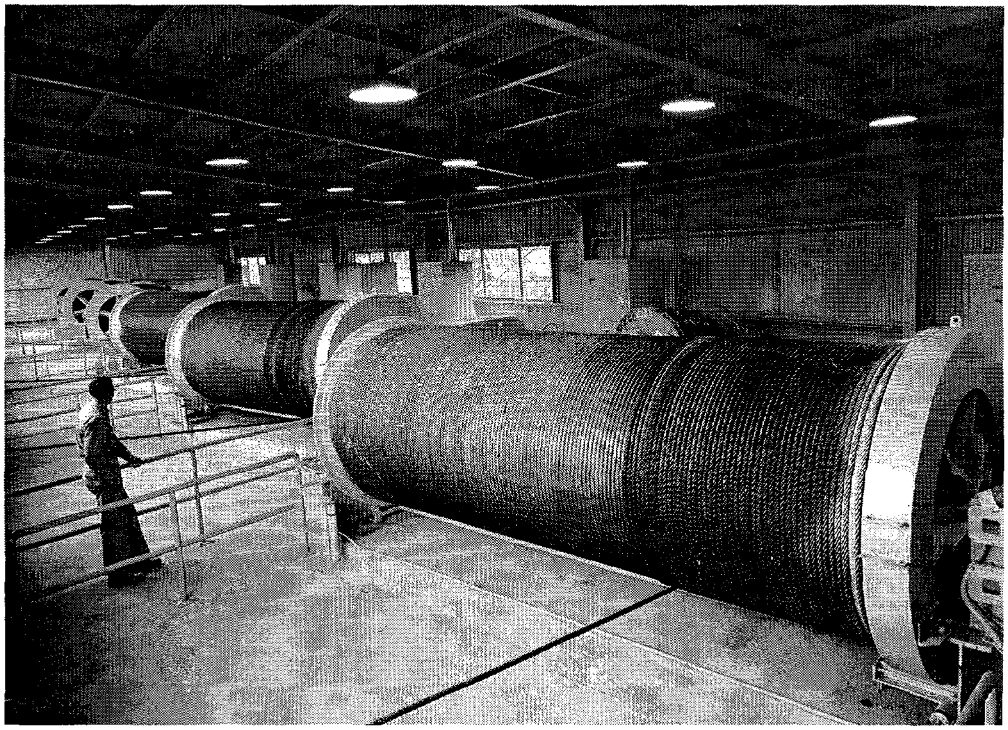


Fig. 3.14 Hoist Control Drums inside Hoist House.

variable speed, single-wrap-type, and have a starting load speed of 16 fpm with a maximum speed of 22 fpm. An eddy-current-controlled brake and two magnetic brakes are incorporated in each unit to ensure smooth and safe operation. In case of power failure, the magnetic brakes automatically lock the hoist in position and may be released only when power is restored or through the use of an emergency dc power source.

Movement of the hoisting cables was performed from either of two locations called "hoist stations." One of these was located alongside the smaller section of the reactor pool to provide maximum visibility of the facility while operating the hoist (Fig. 3.15), and the second station was located at the bottom of the entrance ramp from the reactor "pad" area. This one was shielded from the high radiation levels generated by the reactor. The altitude of the reactor was measured through a stainless steel cable "altimeter" calibrated for specified heights.

Included in the hoisting arrangement was a "constant tension" hoist located about 50 ft northeast of the reactor pool, whose purpose was to provide a means for applying tension to the reactor vessel to prevent it from rotating in the horizontal plane and, when necessary, securing the shield mockup so that its reference point with respect to the reactor shield would not vary when in the air.

A reinforced concrete reactor handling pool, containing a large section 20 ft sq and 25 ft deep with a smaller attached section 4 × 12 × 22 ft deep, lies midway between the two west towers (Fig. 3.15). The larger section, filled with water, was designed to provide a place for storage of the reactor and its 12-ft-diam. containment vessel when not airborne. In addition, the combination of the two sections provides shielding during removal and storage of the fuel assemblies.

The reactor shield tank (Fig. 3.15) was fabricated from 0.25-in. boiler plate shaped and welded to form a 12-ft-diam by 12.5-ft-deep watertight vessel with a hemispherical bottom. On the top of this tank was a structural frame with flanged wheels attached, which rode on the top rim of the tank. This frame could be rotated through 270° by means of a remotely controlled gear train. Two rails supported on this frame permitted the reactor to traverse the tank in all directions.

A copper-clad steel-strand wire was mounted on porcelain insulators (Fig. 3.16) to form a rectangle about 5 ft above the entire tower structure. This same wire extends from the top of the towers to the ground wires at the guy anchors to protect the inclined guys.

The entire tower area, including the guy anchors, is encircled with a No. 1/0 bare copper wire counterpoise buried approximately 18-in below grade. Copper-clad ground rods were driven at each tower leg footing, at each guy anchor, and at intermediate points around the perimeter of the counterpoise. Cross connections, also of No. 1/0 bare copper wire, are buried at right angles to the perimeter counterpoise, crossing near the center of the tower work area. Measurements indicated that the resistance to ground of the whole system was no more than 1-3 ohms.

The design of the hoists and placement of rigging for the hoists was strongly influenced by the magnitude of the objects to be lifted, their movements, limitations on relative displacements and locations with respect to each other (reactor and shield studies) and their relative elevations, and the effect that rigging would have on radiation scattering. Using a recommended system of framing, the hoist capacities depended on the angle between the operating cables and the tower legs. Thus, raising the height of the towers would alter the requirements for capacity of the hoists, affecting a greater cost for the tower, but reducing the cost of the hoists.

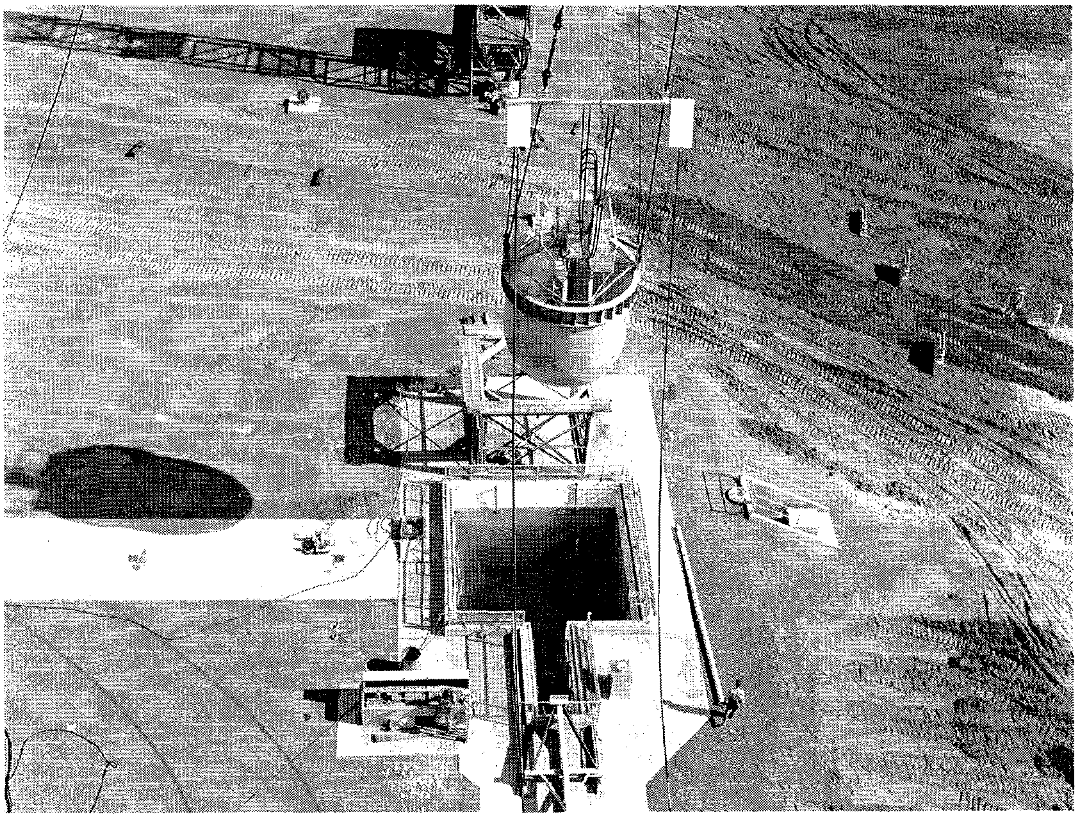


Fig. 3.15 Reactor Shield Tank Suspended over Reactor Storage Pool.

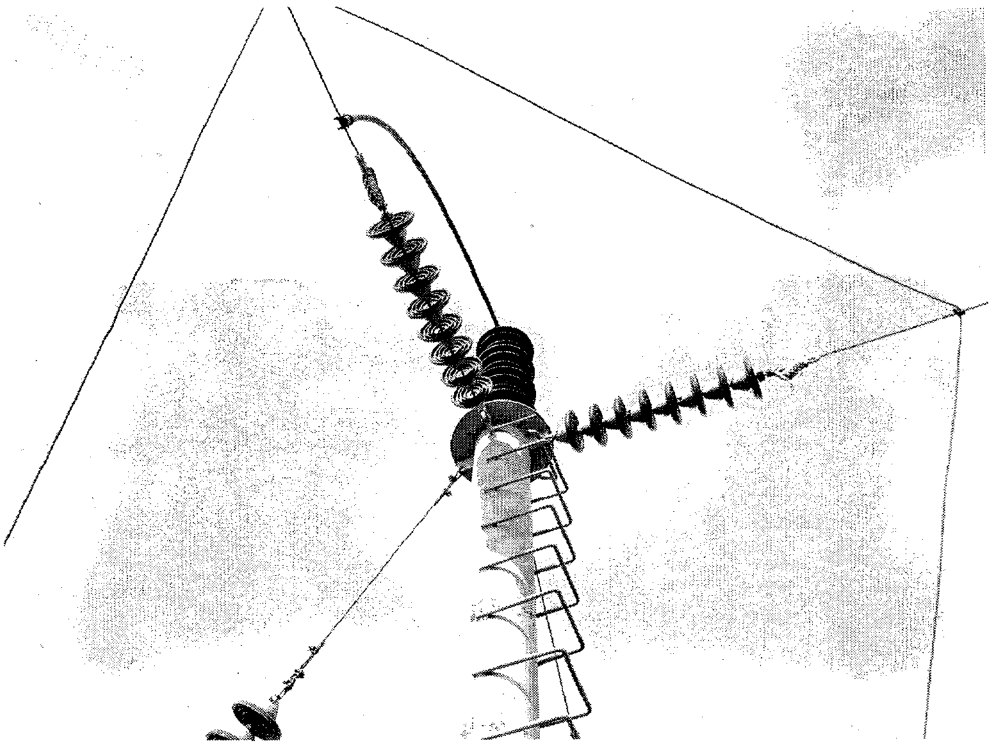


Fig. 3.16 Typical Lightning Protection - Top of Tower Leg.

To help resolve this problem, a model of the towers was built by C. E. Clifford[2] and J. Estabrook, to a scale of approximately 1 in. equal to 40 ft as shown in Fig. 3.17. After much trial and error, the model not only located the position of the sheaves on the towers, but also demonstrated the need for a constant tension hoist to control the distance between the reactor shield tank and the shield sample and provide a smooth operating arrangement for positioning both items. It also prevented rotation of either item with respect to the other. Use of the constant tension hoist provided much more flexibility in the overall operation than was available in the earlier designs.

3.3 CONSTRUCTION RESPONSIBILITIES

Design and fabrication of the project was accomplished through the combined efforts of five different contractors and the AEC. The AEC, in their capacity as administrator, made all arrangements with the United States Corps of Engineers for the site exploration work, reviewed preliminary and final drawing and specifications, performed all the work leading to the selection of architect-engineer and construction contractors, acted as liaison between the Laboratory and prime contractors, and participated in the final inspection of the facility.

ORNL prepared and submitted the preliminary proposal including scope and cost; developed design criteria for the tower foundations, structure, and hoist equipment; design, fabrication, and installation of reactor shield tank, reactor, and contents; prepared working drawings and specifications for control house, reactor pool, access road, utilities, and grounding system; and engineering services for the inspection thereof.

Knappen-Tippetts-Abbett-McCarthy (KTAM) prepared the final design drawings and specifications for the tower structure, tower foundations, and hoisting equipment, and maintained a liaison between the Karl Koch Erecting Company and the Laboratory.

The Charles Hobson Company performed all work related to the site preparation that included site grading, access road, all foundations, reactor tank pool, underground control house, utilities including guard houses and fences, and the electrical grounding system.

The Karl Koch Erecting Company erected the tower columns, installed guy cables, lightning protection system and hoists, erected the hoist house, installed the hoist controls and their cable riggings, installed the personnel lift in the northeast tower leg, and all electrical services located above the tower footings.

The design work was started by the Laboratory in July 1952, with construction beginning March 27, 1953, and completed in February 1954. The entire facility was ready for operation on March 12, 1954 (Fig. 3.18). However, field tests on the completed hoist installation suggested that modifications to the electrical controls should be made to incorporate features that provided greater operational flexibility. These modifications were completed in June 1954. The total cost for construction was $1,997,613.72 (several thousand dollars under the authorized funding of $2,000,000).

3.4 ADDITIONS AND CHANGES

In time, it was realized that the building was not sufficient to house both the crafts and the many staff members involved in the experiments. Thus, in the first half of 1957, an addition[3] was built,

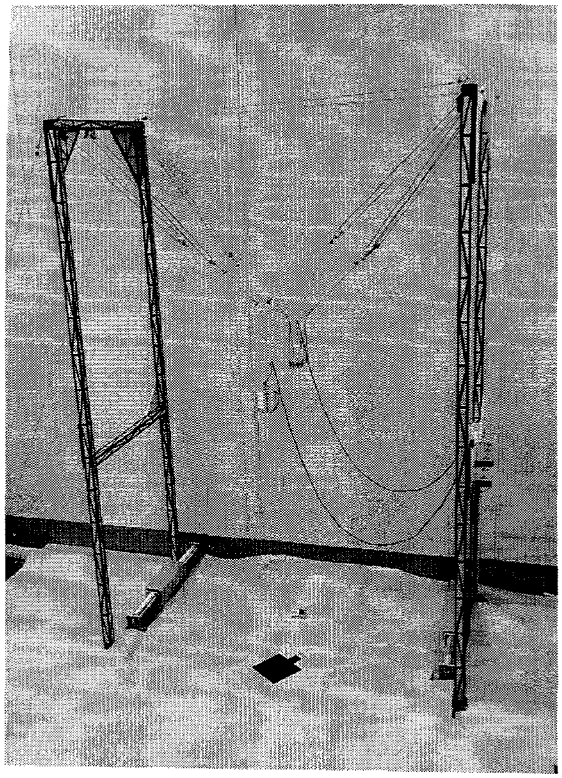


Fig. 3.17 A 1-Inch-Equal-to-40 Foot Model of the Tower

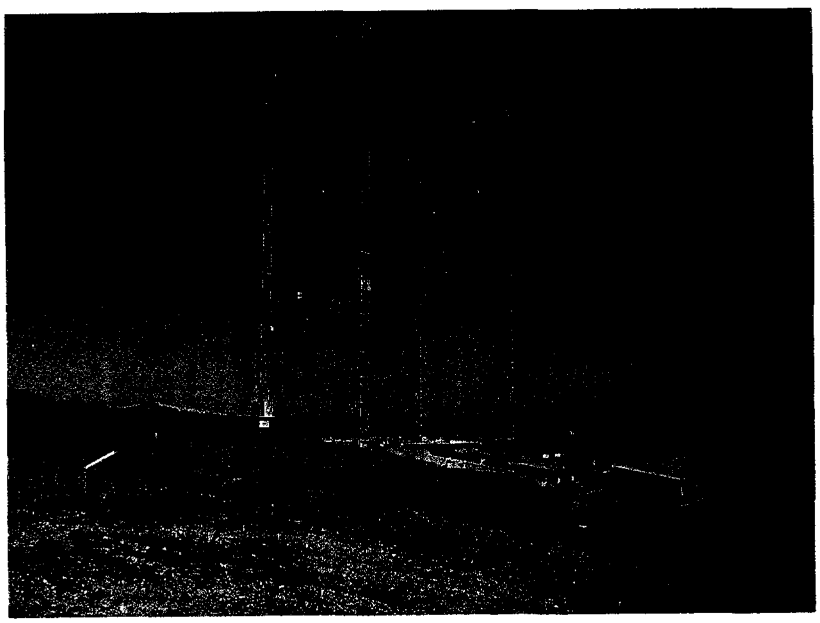


Fig. 3.18 Completed Facility Ready for Occupancy.

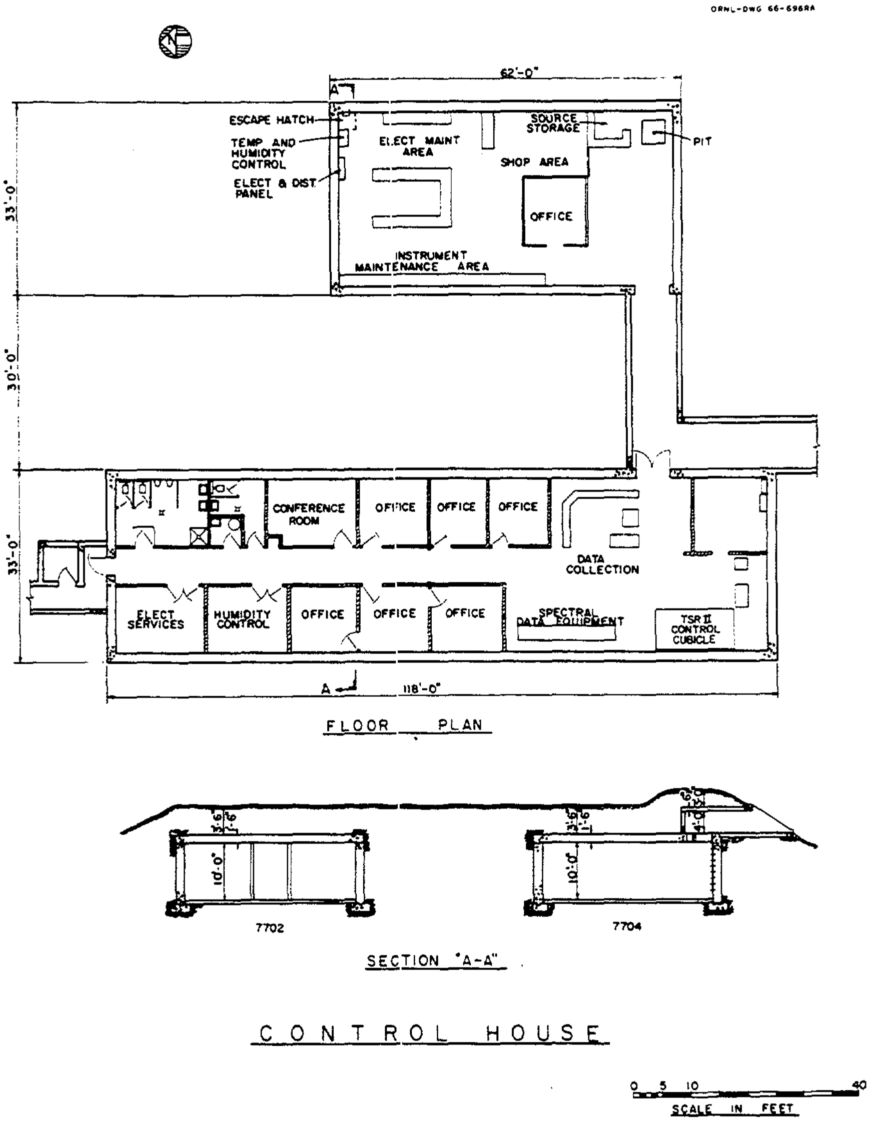


Fig. 3.19 Floor Plan of New Addition and Renovations.

running parallel to the existing one but having only about half the floor space as shown in Fig. 3.19. This area was then occupied by the craft people, and following the conversion of reactor systems TSR-I to TSR-II, the space vacated in the old area was converted to several additional offices and an enlarged data area as shown in that same figure.

Some internal cosmetic changes were made in approximately 1988 that included new tile on the floor and a false ceiling to hide the overhead wiring and pipes in the data-reactor operation area and hallway. The new look is shown in Fig. 3.20. A glass enclosure was then placed around the reactor console to better delineate the reactor control area. These changes are shown in Fig. 3.21. After nearly 40 years of changes, additions and hard use, the facility at the time of its demise still rises with stature above the green grass and trees as shown in Fig. 3.22.

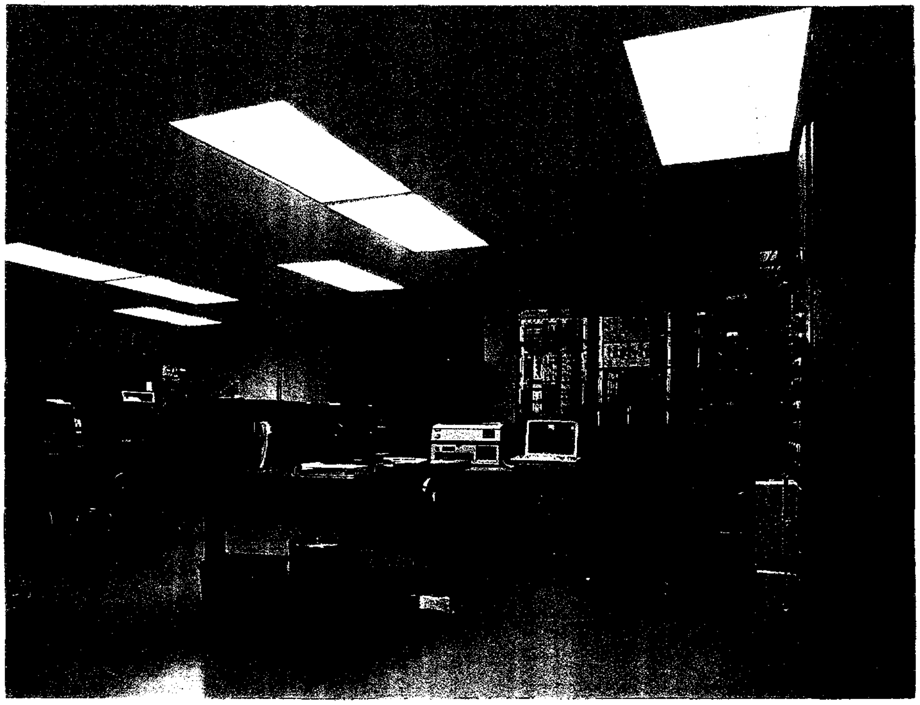


Fig. 3.20 Data-Taking Area (Spectral on the Left, Integral on the Right) after Renovations.

Fig. 3.21 Glass-Enclosed Reactor Console Area.

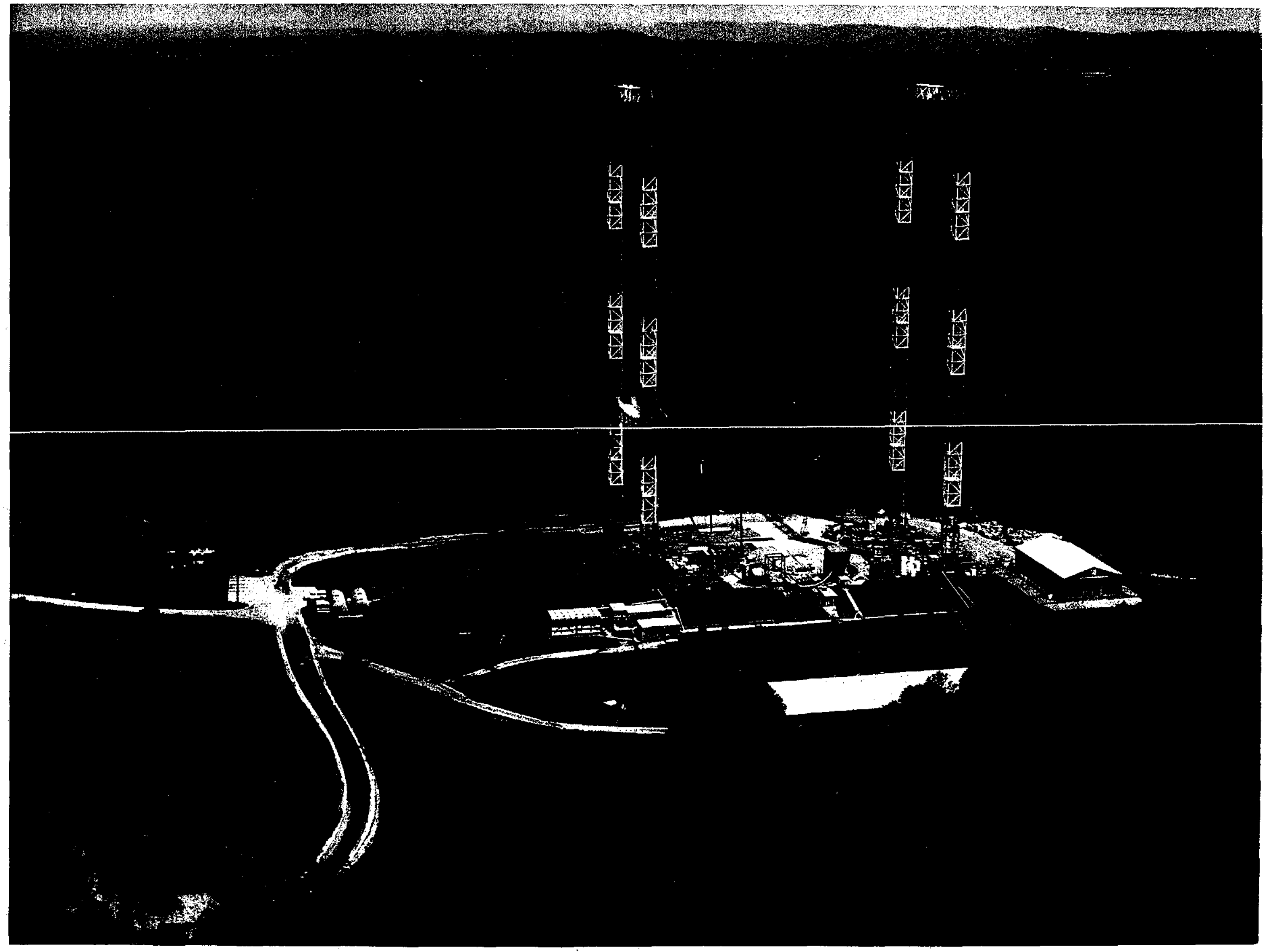


Fig. 3.22 A view of the Tower Shielding Facility as seen looking toward Knoxville with the Clinch River in the background.

3.5 REFERENCES

1. Engineering and Mechanical Division, "Completion Report on the Tower Shielding Facility," ORNL/CF-55-1-165 (February 1955).

2. C. E. Clifford, J. Estabrook, private communication.

3. L. B. Holland, "Tower Shielding Reactor II Design and Operation Report," ORNL/TM-2893, Vol. 1, (February, 1971).

4.0 TOWER SHIELDING REACTORS

4.1 TSR-I DESCRIPTION

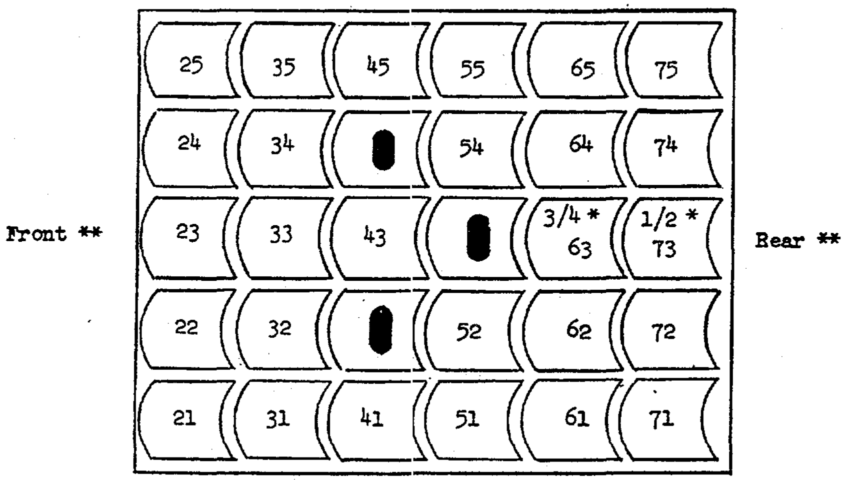
The original TSF reactor closely resembled the BSR, where the fuel for both reactors were slightly modified Materials Testing Reactor (MTR)-type elements. Each element was composed of eighteen fuel plates containing 140 grams of U^{235} surrounded by 2S aluminum, giving an aluminum-to-uranium ratio of about 0.7. The elements were held in the reactor assembly by an aluminum grid plate that accommodated 30 elements in a 5 × 6 array (Fig. 4.1 for a typical array). The reactor was supported 6 ft under water in a 12 ft diam reactor tank containing 40 t of water that acted as the coolant during reactor operation. When not in operation, the reactor and its containment vessel would reside in a ground-level handling pool. The reactor and its support systems were designed[1] so that the combination could be removed from the reactor tank and lowered into specially designed reactor shields. Since these shields usually contained a region of lead around the reactor location, the top sections of the individual fuel elements were modified to include a layer of lead that corresponded to the lead portion of the reactor shield.

The reactor power was controlled by two motor-driven safety rods of boron carbide in an aluminum jacket and one servo-operated control rod. The power level of the reactor was measured by four ion chambers, two of which were gamma-ray compensated, and by one fission chamber required for startup. The shroud, motor drives, and preamplifiers were mounted on a small four wheeled dolly approximately 4 ft long and 1.5 ft wide that was motor driven, as shown in Fig. 4.2. The dolly wheels were mounted on a rotatable track which spanned the reactor tank, allowing the reactor to be moved to any position in the horizontal midplane of the tank.

The reactor was cooled by convective water flow in the reactor tank and by forced flow when used in other shields. The power was automatically regulated to any preset level. The safety system included a one-second period scram as well as a high-level scram and building interlock scrams.

4.1.1 Critical Experiments

The initial critical experiment, completed on March 12, was primarily for the purpose of checking the reactor instrumentation. The TSR fuel loading had been previously checked for criticality in the BSR since the BSR elements were also of the MTR type. The first loading consisted of three half elements (for control purposes) and 23 full elements, making a 5 × 5 array with the additional element in the center of the back face. Tests indicated that this array was critical with a three-quarter element in the front row center. A maximum power of about 50 watts was reached, sufficient to check the safety chambers. Additional critical arrangements using a 5 × 6 loading were then tested to obtain a loading of about 1.5% excess $\Delta k/k$ needed to overcome any loss due to rise in water temperature, xenon poisoning, and fuel burnout while still allowing a ten second period for operation up to 100 kW. However, this power level was soon shown to be inadequate for the experiments, so, on September 22, 1954, the Advisory Committee on reactor safety granted permission to extend the upper power level to 500 kW. As a matter of safety, it was TSF policy to run critical experiments whenever such changes occurred or the reactor was placed in a new reactor shield.



* partial elements

** When the reactor is not in the center of the reactor tank, the face labelled front is nearest the outside of the tank.

Fig. 4.1 TSR-I Reactor Loading No. 1.

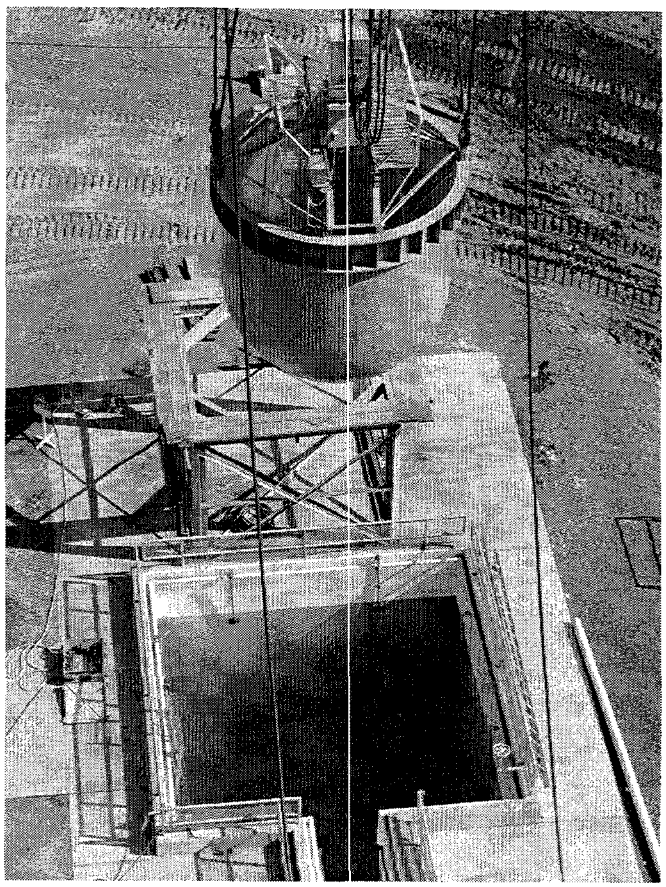


Fig. 4.2 TSR-I Shield Parked Above its Storage Pool.

4.1.2 Power Calibration

The first experiment was to establish a method for the measurement of the reactor power since the data obtained in future experiments would be expressed as a function of power level. To do this, the reactor was loaded with an array of elements whose power had been measured previously at the BSR. The same measurements of the thermal flux using cobalt wires were repeated at the TSF along with selected locations for gold and uranium foils to obtain absolute values. Comparisons were also made using neutron and gamma-ray dose-rate measurements in the water surrounding the reactor.

A procedure was developed and tested[2] for measuring the reactor power using a calorimetric method. Thermocouples were placed throughout the reactor tank to measure the change in water temperature during reactor operation. Initially three different runs were made for starting water temperature at ambient, below, and above ambient, with results agreeing within 1%. Since the calorimetric method was lengthy and not feasible for daily use, it was replaced by thermal flux measurements in the water surrounding the reactor that were normalized to previously correlated calorimetric runs.

4.1.3 Changes

The use of TSR-I was short-lived as the need for definitive-type data to resolve shielding problems became more evident. The initial experiment using the TSR-I began in early 1954 and the final experiment was completed a short five years later at the end of 1958 with the repeat of the earlier measurements for the General Electric 2π experiment. For the next two years, all the effort at the TSF was spent dismantling the TSR-I and replacing it with the new TSR-II system that was being fabricated.

4.2 TSR-II DESCRIPTION

Studies of optimized reactor shield designs are usually made with a full-scale mockup of the shield and a known reactor source. The effectiveness of the actual shield design in its final application would then be predicted by making corrections for the differences in source terms. The power source selected for propulsion of the nuclear-powered aircraft was a CFRMR designed by Pratt and Whitney Aircraft, a source considerably different than the TSR-I swimming-pool-type reactor. It was soon realized that the difference in source terms would make it extremely difficult to make these corrections. Thus, it was recommended to the ORNL shielding group that reactor shield measurements be made with a source whose radiation leakage was similar to that from the CFRMR. That reactor, named the SMR, would be required to mockup all the radiation from the Pratt and Whitney-designed CFRMR except the decay gamma rays and neutrons passing through the heat exchanger. This requirement meant that the SMR should have a beryllium reflector shaped as specified for the CFRMR.

A 5 MW reactor design using fixed fuel closely simulating the CFRMR source was proposed. Every attempt was made to preserve the geometry of the CFRMR. A preliminary design and cost estimate was submitted to Laboratory management, but the concept was turned down. It was the opinion of Alvin Weinberg, then Research Director at the Laboratory, that the real need was that of a more general purpose source, something that would cover problems of importance to shields for any reactor cycle that might present itself. It was left to the shielding group to plan and propose just such a concept.

Experience from past shielding calculations and design studies suggested that several basic types of experiments should be conducted with a new reactor, and this in turn suggested some special requirements as to the type of reactor source needed.[3] Differential-type experiments should be provided for with and without reactor source collimation. It should be possible to alter the source term using modifiers and the direction of the source term should be variable. The reactor system should also allow for studies of complete shield mockups. The power of the reactor should be much greater than 400 kW; 5000 kW was suggested. Most important of all, the reactor shape should be such that its source could be readily calculated. All this led to a suggestion by E. P. Blizard that the reactor should be spherical to provide an isotopic source. C. E. Clifford proposed that the control plates be internal to the core to ensure a minimum perturbation on the leakage flux or spectrum. The fuel plates would still be MTR-type and the reactor would be light-water cooled. The spherical region in the center of the core would contain water and umbrella-shaped boron-loaded plates would serve as control devices. Shields would surround the reactor and be supported so that both the reactor and its shield could be lifted into the air. What was to finally materialize would soon become known as TSR-II (Fig. 4.3).

Considerable effort was spent in the design of this reactor. Calculations using a modified three-group, three-region reactor code (called 2G3R) were performed on the ORACLE and on the UNIVAC at Columbia University to determine the critical mass and control available in various reactor configurations having internal water reflector. People from Pratt & Whitney also made calculations, and the two results were somewhat different. Effective multiplication factors for concentrations of ^{235}U, control of excess reactivity, separations between control plates and fuel elements, and effect of the control plates on fast and thermal flux distributions were all investigated.[4] From calculations it was determined that the internal water reflector should be no smaller than 17.5 inches in diameter. Once this was established, continued calculations showed that the 0.06-mil-thick fuel plates should be separated by 120 mils of water, the core thickness should be about 5.5 in. and contain about 7 kg of ^{235}U. These calculations were later refined to obtain the actual values.

By the end of June 1957, the design drawings for the curved fuel plates were completed and the developmental work for fabrication was initiated.[5] The work was divided into two phases; first, fabrication of the individual plates, and secondly, forming the plates and assembling them into elements. The use of an aluminum alloy (type 6951), which is much harder than pure aluminum (type 1100), made it feasible to use a specific roller process to curve the plates. Three plates were successfully formed and this success led to the fabrication of a dummy core for water flow studies of the proposed cooling system. Studies of the reactor control system were initiated along with other pertinent calculations.

As time passed, more and more effort was being applied to the design of the complete reactor system. A preliminary concept for suspension of the reactor-reactor collimator system that would allow a beam of radiation to sweep both a vertical and horizontal plane was proposed.[6] Development of methods for fabrication of the curved fuel elements was required. Of the 71 different fuel plate sizes needed for the two types of elements, techniques developed to shape 60 of them used depleted uranium as the material. This ultimate success in fabrication is displayed in the two dummy elements made of aluminum shown in Fig. 4.4.

Different groups within the Laboratory participated in the design and fabrication of the reactor system. The engineering design of the TSR-II was performed by the ORNL Engineering Department,

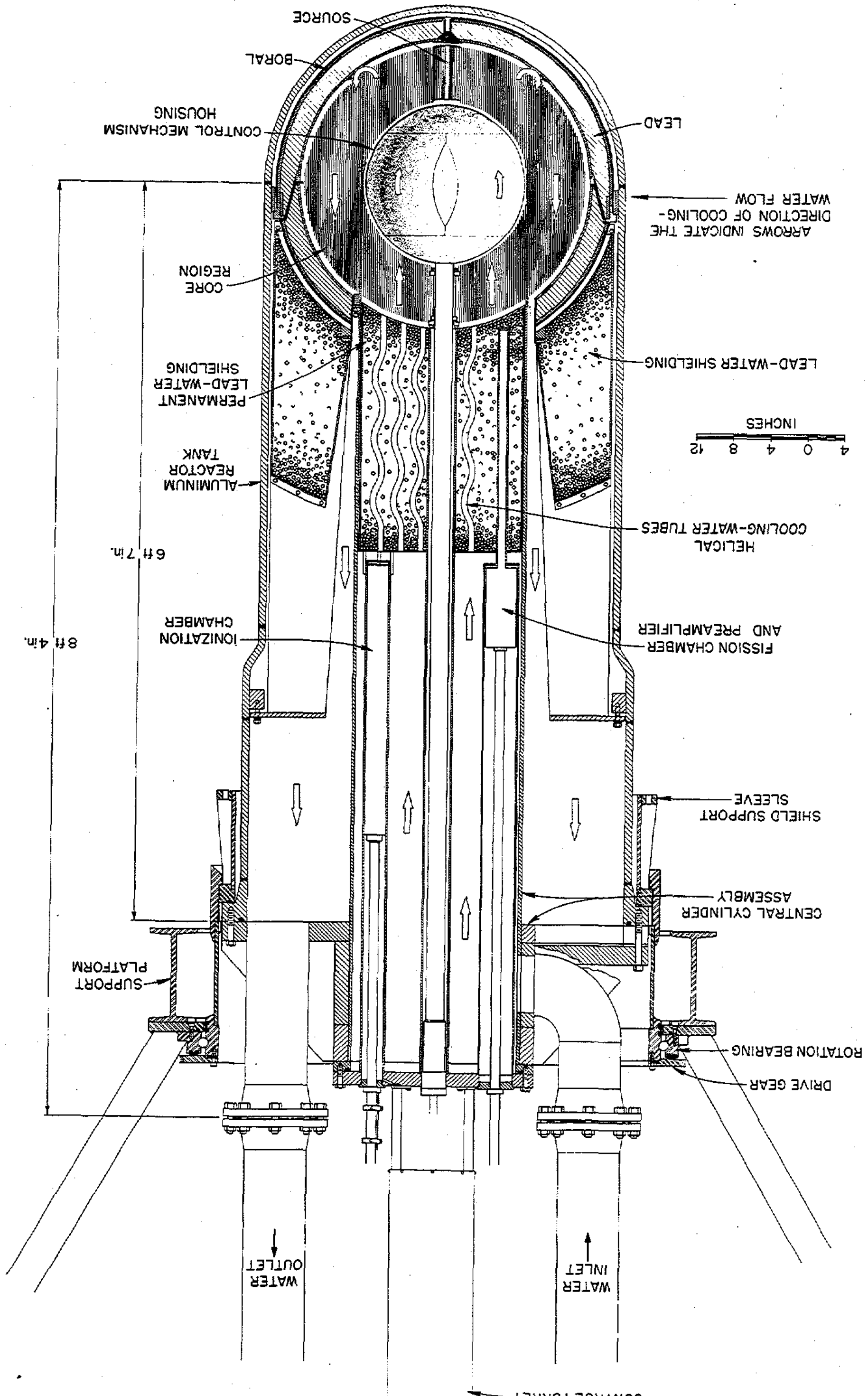


Fig. 4.3 TSR-II Design (Vertical Section).

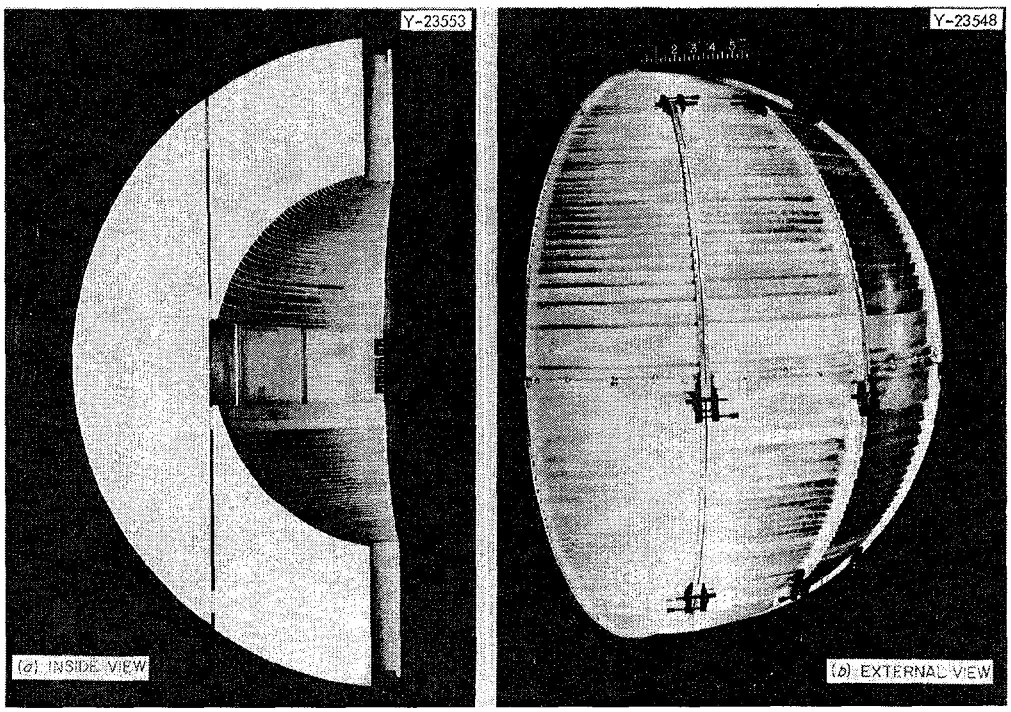


Fig. 4.4 Inside and Outside Views of Dummy Fuel Elements for Proposed TSR-II.

and development of the fuel elements was done by the ORNL Metallurgy Division. The control system was designed by the Reactor Controls group, and the nuclear calculations for the reactor design were done by M. E. LaVerne and his group, with advice in all phases of the work coming from members of the Applied Physics Division Shielding group.

4.2.1 Cooling System

The reactor system was cooled by flowing demineralized water downward through the fuel annulus directly above and below the control ball and returning up through the fuel annulus surrounding the sides of the control ball. Bench-type mockups of the water flow through the core were vigorously studied to obtain the lowest drop in water pressure yet maintain an even distribution of flow through all the channels. The system was designed so that there would be no boiling in the reactor core for a power level of 5 MW. Studies[7] showed that adequate cooling would be provided with the use of specifically located baffle plates and a flow rate of 800 gpm. As the water left the reactor vessel, it passed through a 6-in.-diam hose attached at the 100-ft-level of the tower leg; down the outside of the tower leg to the demineralizer, through cooling radiators where the heat dissipation was controlled, into the 1500-gallon detention tank, from which the 50 horsepower main pump would again force the water up the other tower leg on the opposite side of the reactor to the 100-ft-level and back into the reactor vessel. This flow route is described in Fig. 4.5 as one part of a cooling water display system on the reactor control panel at the operator's control station.

A 5 horsepower battery-operated pump operates in parallel with the main pump. In case of an electric outage, the pump will continue to cool the reactor with a flow of 40 gpm. It has double purpose in that this water flow also serves to prevent the water within the reactor from freezing when it is not being operated. The water passes through two 40-kW heaters for that purpose. This flow diagram is also depicted in Fig. 4.5.

A third pump, called the shim pump, delivers about 20 gpm water flow to the control turret, where it is divided to operate the reactor mechanisms and to cool the control mechanism housing.

4.2.2 Reactor Vessel

The reactor pressure vessel was made of 0.75-in.-thick 5052 aluminum, 8 ft long, having a 37 in. inside diameter at the hemispherical end, and a 40 in. inside diameter at the open end to facilitate removal of the internal shielding, fuel elements, etc., from the vessel. The vessel could be secured to the hoists using a bayonet-type attachment for movement above ground level or it could be placed inside of specially designed shields with both items supported on or above the ground.

4.2.3 Reactor Controls

The final design of the reactor control system was reached after a long development program. The TSR-II fuel elements were fabricated to contain 8.1 kg of ^{235}U, which, calculations indicated, would be sufficient to overcome not only the worth of the control plates in their fully withdrawn position but also the worth of the structural material in and around the core. It was also indicated that this amount would be enough for another 1.6% Δk/k to be controlled by the grid plates (plates formed by layering stainless steel clad aluminum) and 2.0% Δk/k for calculational error. The first critical experiment demonstrated that the core was subcritical by about 0.7% Δk/k even with the control plates removed.[8] Additional critical experiments indicated that large changes in k (as much as 10%) could be gained by replacing the water in the center region with aluminum or voids. Generation of a large void at that location, however, would present a problem from a safety viewpoint. Further tests showed that the severity of the void problem could be reduced by replacing the water with a sphere of aluminum in the internal region while allowing the water to flow between the sphere and the fuel. Continued experimentation showed that the only configuration in which the formation of a void did not produce a positive change in reactivity was when the aluminum sphere was covered with borated plastic and the water thickness between the boron and the fuel was no more than 0.25 in. These same studies also showed that reactor control was more effective using boron than cadmium. It was concluded, therefore, that the TSR-II control plates should be operated inside a 17-in.-diam spherical shell containing aluminum and water and that the originally designed cadmium control grids should be replaced with B_4C plates.

With these changes, the control mechanism housing consisted of an 11-in.-diam central aluminum sphere with four 2-in.-diam aluminum plugs extending from the ball through each of the five B_4C plates shown on the left[a] in Fig. 4.6. The sixth plate, located on top of the ball to be used as the regulating rod, was left without holes. A second set of critical experiments with these changes showed that the excess reactivity realized by replacing most of the water in the control area with aluminum was offset by the boron in the control rods and the lead-boron shield surrounding the fuel annulus, leaving the reactor still subcritical. Criticality was reached by placing fuel-bearing strips totalling 233 g of ^{235}U on the outside surface of the 17-in.-diam aluminum sphere as a cover.

These results led to several changes in the final design of housing for the control mechanism. The diameter of the central aluminum sphere was increased from 11-11.5 in., the aluminum plugs that protruded through the B_4C plates were removed, and the control plate thicknesses were decreased from 1 to 0.5 inches. These changes increased the total regulating plate worth from 0.3 to 0.43% Δk/k. With this arrangement the worth of all the shim-safety plates fully withdrawn was found to be 3.82% Δk/k. A schematic of the control mechanism for the control plate is given in Fig. 4.7.

Following assembly of the reactor at the TSF, the reactor achieved criticality for the first time at 12:33 a.m. on March 26, 1960, and on March 30, the TSR-II instrumentation was first used to operate the reactor. After additional criticality studies were completed, the reactor was disassembled for modifications. Reassembly as[(b)] in Fig. 4.7 was not completed until December 22, when for the first time, it was operated with a full flow of cooling water. It was not until February 19, 1961, that it operated at full (approved) power of 100 kW, beginning a new era at the TSF. Eleven years later, on January 22, 1972, approval was given to operate the reactor at 1 MW, the maximum power achieved during the reactor's lifetime.

Earlier in the discussion on the development of the TSR-II control system, it was noted that to make the original reactor core go critical with a reasonable Δk excess, it was necessary to add lune-shaped, fuel-bearing cover plates around the housing for the controls. With the first control mechanism housing, this loading in the lune-shaped plates amounted to 160 g. With time, these plates were replaced with ones containing 230 g and some time before 1982, it was increased to 300 g.

In 1982, it was realized that to continue extended operation of the reactor at the high power of 1 MW, the excess reactivity in the core (still the initial one) at this time, about 0.94% $\Delta k/k$, would not be adequate to complete an extended shielding program. Because of problems associated with fabrication, the use of still heavier loaded lune plates was not considered. Two choices were proposed: (1) replace the core loading with a new core, or (2) improvise some other approach that would extend the life of the present core. Inspection of the core indicated that the elements were in good condition, and calculations showed that about one-third of the allowable core life was still available. The cost would be high to replace it and the long period of time needed for fabrication of a new core would delay the next experiments, leaving no choice but to implement the second option.

One of the requirements for operation of TSR-II was that four out of the five shim-safety plates be operable at all times. Thus operation of one of the plates could be inhibited and still meet the safety requirements. The desired choice was to modify the plate that would least distort the reactor source term as seen by the shield mockups, namely shim-safety rod number five, i.e., the one on the bottom of the reactor. That boron-loaded plate was replaced with one of 1100 aluminum and then made inoperable. This exchange provided the additional reactivity that was needed.

4.2.4 Power Calibration for TSR-II

The power level of the reactor as indicated by a range switch on a picoammeter channel was established by what was termed a heat power run. For this, the reactor was operated at a steady state in a region near its upper power limitation until an equilibrium condition was reached between the input and output temperatures of the water flowing through the reactor.[9] Many factors played a part in obtaining this, but once it was attained, the flow rate of the water was recorded along with its inlet and outlet temperatures. Simply put, the power in kW was then calculated by multiplying the temperature rise (difference between inlet and outlet) by the rate of flow and by the factor 0.1459.

To monitor the power level on a daily basis, two fission chambers were integrated as part of the reactor shield system by the experimental group and the count rates on these detectors were recorded during the power calibration exercise to establish a counting rate-power level relationship.

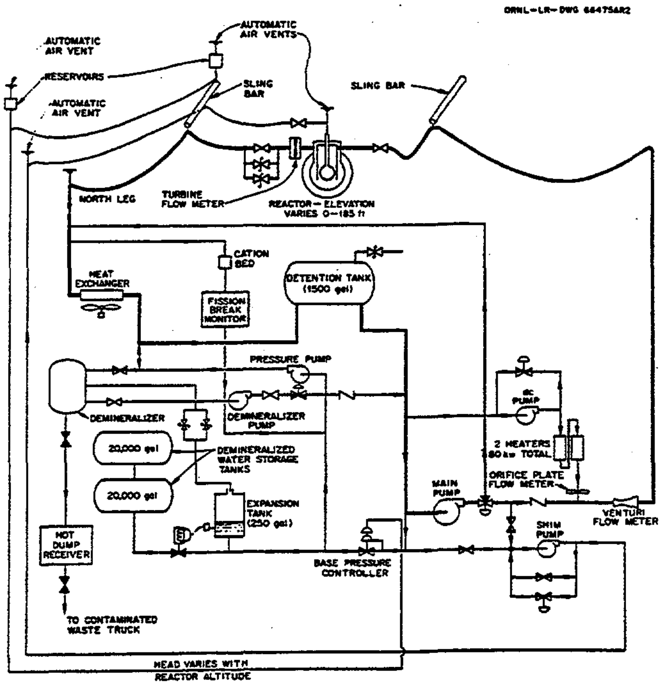


Fig. 4.5 Reactor Cooling System for TSR-II.

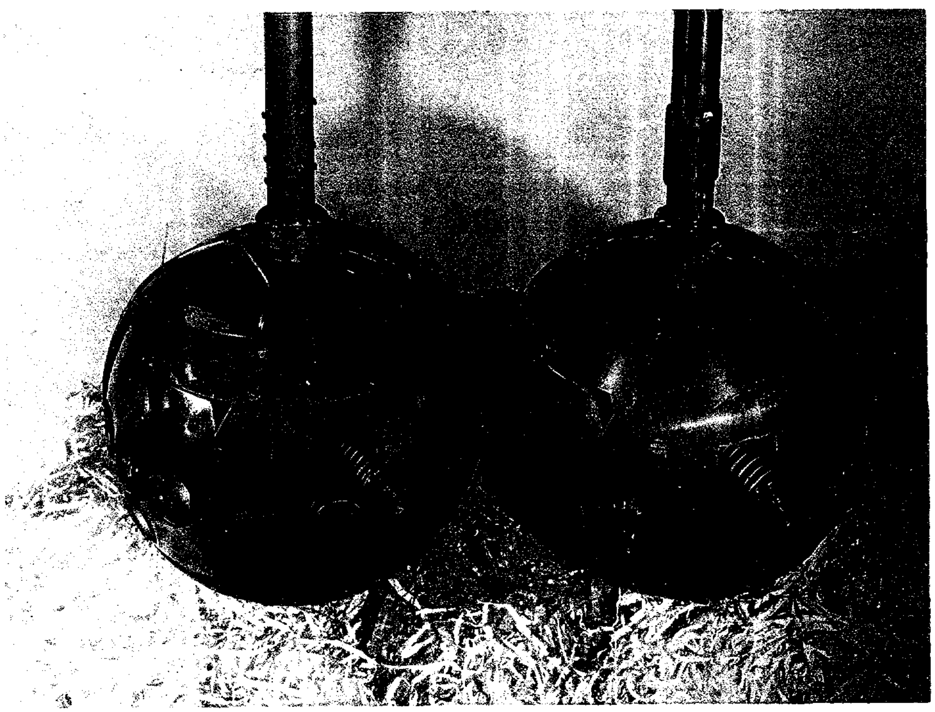


Fig. 4.6 TSR-II control Mechanism Housing: (a) 11-in.-diam Model with Minimum Plus. (b) 11.5-in.-diam Model with Plugs Eliminated.

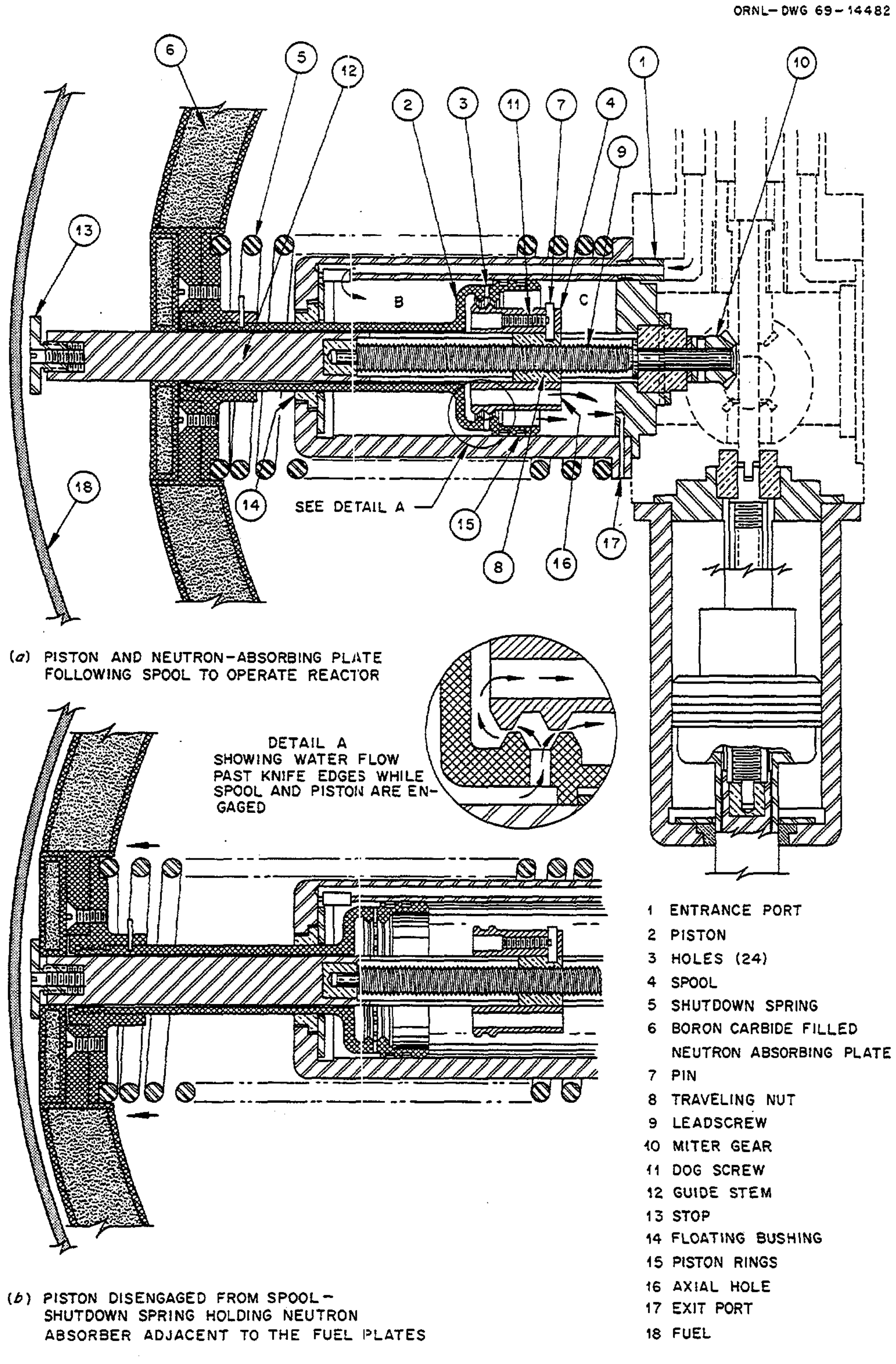


Fig. 4.7 Control Mechanism for TSR-II.

4.2.5 Reactor Source Changes

By 1974, most of the experiments were being designed so that the reactor source could remain stationery inside its shield and not be elevated. There was, however, the need for increasing the intensity of the source to improve the quality of some of the experimental data. This need led to the building of what is known affectionately as the Big Beam Shield, a concrete structure that allows exposure of the total core instead of just the center of the core as in the previous shield. However, to present a truly spherical source it was necessary to replace the lead directly above the annular fuel elements with air to match the void shape between the bottom of the pressure vessel and the concrete shield.

4.3 AIRCRAFT SHIELD TEST REACTOR

The ASTR was a portable, enriched uranium thermal reactor with a maximum power of 1 MW.[10,11] It consisted of 32 MTR-type fuel elements arrayed horizontally to form a right circular cylindrical core containing 4800 grams of uranium^{235}. Each element consisted of 18 fuel plates of uranium-aluminum alloy clad in aluminum. The fuel element arrangement is shown in Fig. 4.8. The reactor was cooled by demineralized water that also served as moderator and reflector. The core and moderator section were surrounded by a layer of lead, 7.62 cm on the side and 15.24 cm between reactor and experimental mockups. The lead on the side was followed by two 17.8-cm-thick tanks to contain plain or borated water, or a combination of both. The reactor was controlled by two cadmium-steel safety rods and one dynamic control rod.

The ASTR began operation in 1954 for some ground shielding experiments at Fort Worth, and was used for airborne shielding experiments aboard a modified B-36, the NTA beginning in 1955. An experimental program was conducted at the TSF using the ASTR as a source beginning in early 1958 for about a six-month period.

4.4 SPACE NUCLEAR AUXILIARY POWER REACTOR DESCRIPTION

It was proposed in 1965 that a SNAP2-10A reactor system be installed at the TSF for the purpose of providing experimental data for comparison with Monte Carlo calculations of the radiation intensities transmitted through shadow shields.[12,13] The reactor, a modification of the SNAP-10A, was designed and fabricated by Atomics International, a division of North American Aviation, Inc. The modifications were: (1) a decrease in the reactivity of the core, (2) replacement of two of the four control drives by pneumatically-activated drives which could be scrammed by the action of springs, a feature that was not present in the original design but was added to allow multiple startups and shutdowns of the reactor, and (3) the installation of a NaK-to-air heat exchanger above the reactor. Existing reflector assemblies and fuel elements were utilized to minimize the cost of installation. The maximum power level allowable was 10 kW.

The TSF pad area was modified by building a shelter for the reactor adjacent to the reactor storage pool, consisting of three thick concrete walls for shielding and a thin metal roof cover as shown in Fig. 4.9. One of the walls, seen at the right edge of the figure running parallel to the page, contained a shielding window for adding manipulators if necessary. The fourth wall, opposite the pad

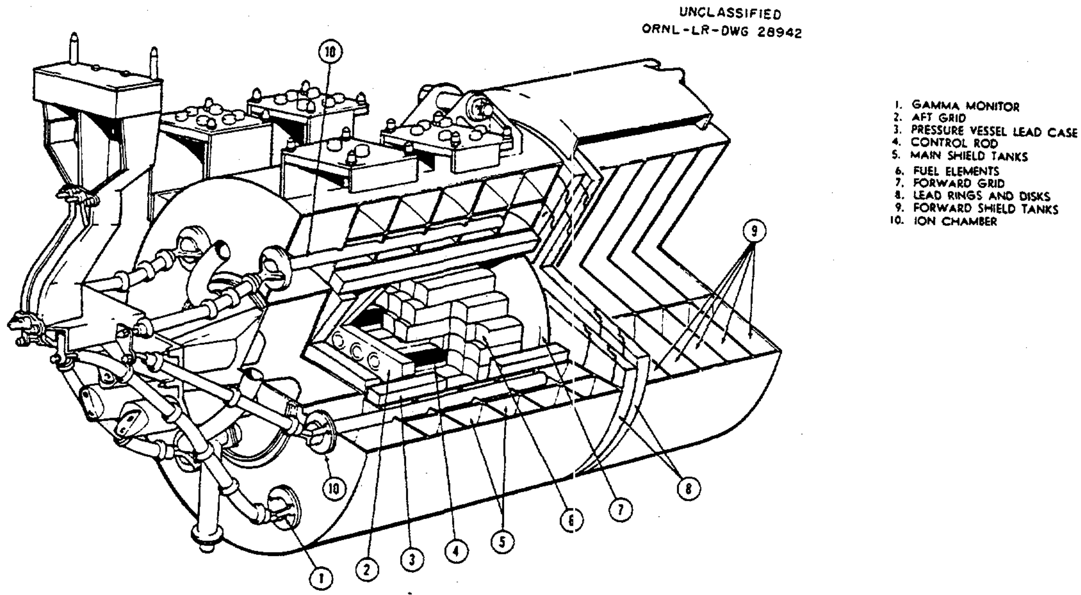


Fig. 4.8 Cutaway View of the Aircraft Shield Test Reactor (ASTR).

Fig. 4.9 SNAP Reactor Suspended Above the Pool Area.

area, was left void so that the reactor could be moved in and out of the building via the boom. The reactor was suspended from the boom that extended out from a vertical column located just outside the northern concrete wall. This arrangement allowed positioning of the reactor as desired for the experiments which was usually over the hole in the concrete covered by a metal plate in the middle of the picture. The boom was designed so that the reactor could be rotated in a given plane and this plane could be lowered and raised as the experiment required.

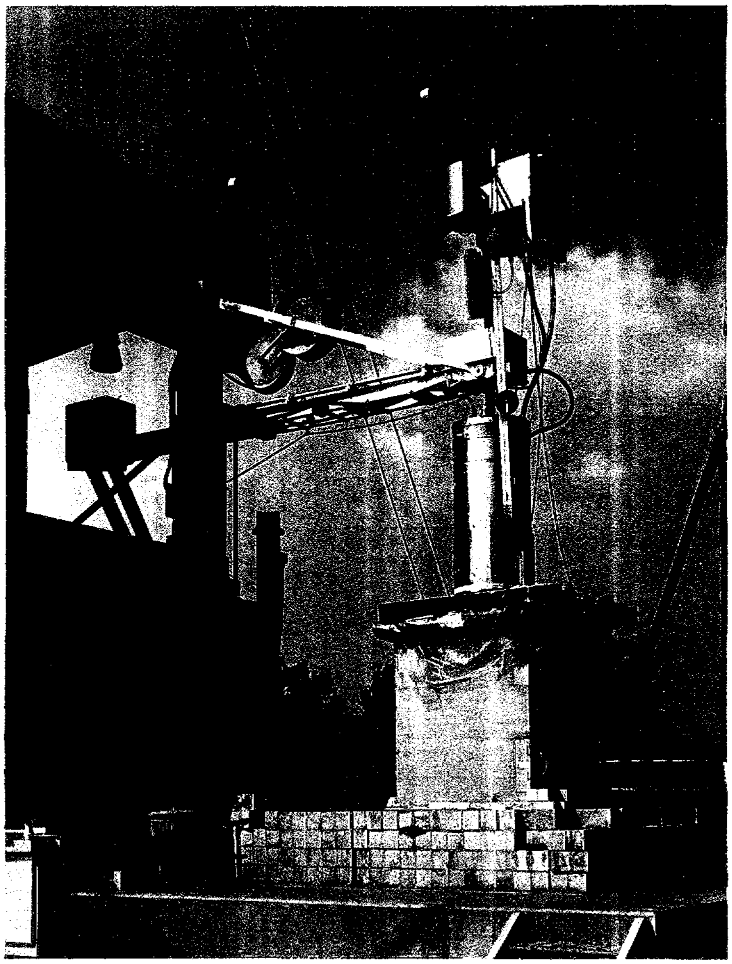


Fig. 4.10 SNAP Reactor above Shield Mockup Residing on the Concrete Cover over the Pool.

The reactor, as displayed in Fig. 4.9, shows the cone-shaped NaK-to-air heat exchanger above the core and controls, the beryllium control pieces in the shutdown mode (extended outward) the control chamber at the side, and the power level monitors placed above the reactor shield cover plate. Figure 4.10 shows the reactor covered with its weather shield and placed in positions above a typical mockup for an experimental measurement.

The reactor[14] was a compact ZrH type with 4.8 kg of highly enriched ^{235}U fuel. The full moderator elements were 1.25-in.-diam by 12.25-in.-long rods densely packed within a 9-in.-diam by 16-in.-long pressure vessel. A beryllium neutron reflector containing four control drums surrounded the cylindrical surface area of the core vessel. The coolant system contained 2.46 gallons of NaK in a sealed loop under an atmosphere of argon. The heat exchanger was directly above the reactor core, lying within the cone formed by the LiH shadow shield.

The reactor went critical on April 7, 1967. A core configuration was established that included a loading of 36 SNAPTRAN V fuel elements and one SS rod with a small quantity of boron carbide in the center position. The relative power distribution of the reactor was determined by scanning individual fuel elements for fission product gammas and the absolute fission rate was determined by uranium foil activation. The last experiment using the SNAP reactor as a source was performed in late 1971 to study the neutron streaming through a design of the grid-plate shield for the Fast Flux Test Facility. The reactor was placed in standby following that experiment and the core was removed from the TSF in 1973.

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5.0 REACTOR SHIELDS

5.1 TSR-I REACTOR TANK

As mentioned previously in Sect. 4.2, the first shield[1,2] for TSR-I was a cylindrical steel tank, 12 ft in diam and having a hemispherical bottom, that supported the reactor to a depth of about 6 ft of water. Movement of the reactor within the tank was possible in the horizontal plane only. The reactor was supported by a motor-driven four-wheel dolly that moved the reactor in a straight line from one side of the tank to the other. This dolly was attached to a frame having flanged wheels that rotated on a track atop the tank wall. Thus it was possible to position the reactor in the same horizontal plane anywhere within the tank. This arrangement was pictured in the previous chapter in Fig. 4.2.

5.2 TSR-I GE ANP-R1

The ANP-R1 reactor shield designed by GE is shown in Fig. 5.1. It was the first compartmentalized reactor shield to be studied at the TSF. The assembly consisted of three sections: (1) a central tank into which TSR-I could be inserted, (2) an aft section containing an annular air duct and hydrogenous shield, and (3) a thicker forward section housing an annular duct, a lead shadow shield, and compartments for water. Each of the compartments,[3] shown in Fig. 5.2, could be flooded with water or drained individually. The shield was 129 in. in length and about 90 in. in diam.

5.3 TSR-I COMPARTMENTALIZED REACTOR SHIELD

The TSR-I compartmentalized reactor shield[4] (Fig. 5.3) was approximately spherical in shape and was built to enclose the core of TSR-I and serve as a neutron shield. The compartments in the tank were welded aluminum construction that could be filled with liquids whose density might be as high as 2.5 and then drained remotely as required. The tank consisted of eight large conical shells containing a total of 44 subshells, some of which were intersected by the cylindrical reactor standpipe. The outer one-foot compartments in each of the eight large cones were used to block out the neutrons from complete sections of the shield beam for differential beam experiments. The 2-in.-thick compartments were used to determine the optimum neutron shield shape by trial and error (filling and emptying).

5.4 ASTR SHIELD

The ASTR[5] was cooled by demineralized water which served as both the moderator and reflector. The core and moderator sections were surrounded by a layer of lead and tanks to contain the water shield, the lead being 3 in. on the side and 6 in. on the front as shown in the previous section. For most measurements, a 0.125-in.-thick boral plate was mounted outside the reactor shield. The ASTR and its shield were suspended from an airframe (Fig. 5.4) that permitted its rotation through 180°. For all configurations, a 6-in.-thick lead shadow shield was suspended in front of the reactor and rotated with it. The airframe itself could be lowered and raised in a vertical plane around a point on the support frane carrying the reactor.

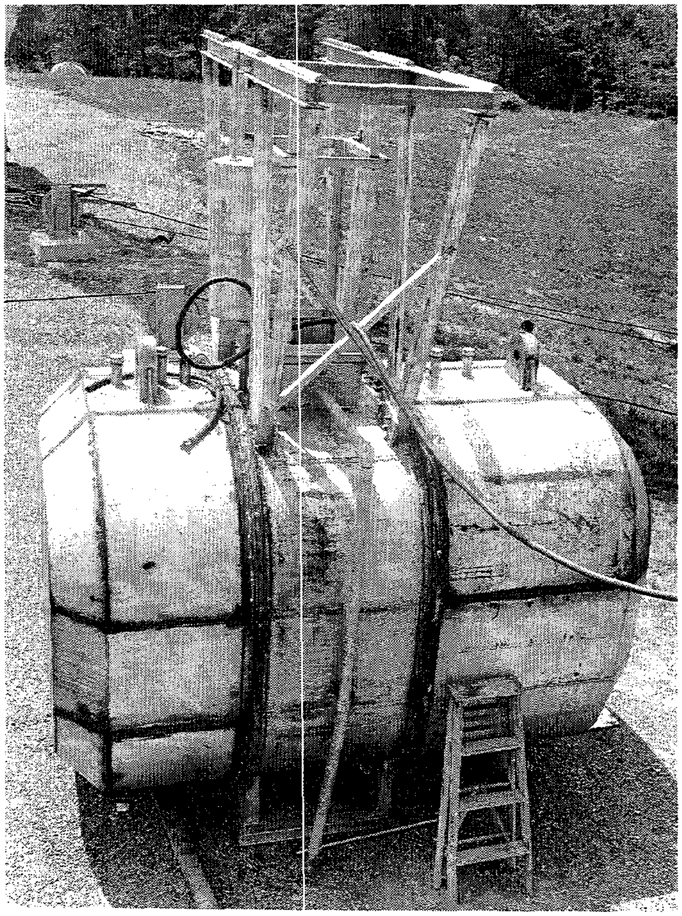

Fig. 5.1 GE-ANP R1-Reactor Shield.

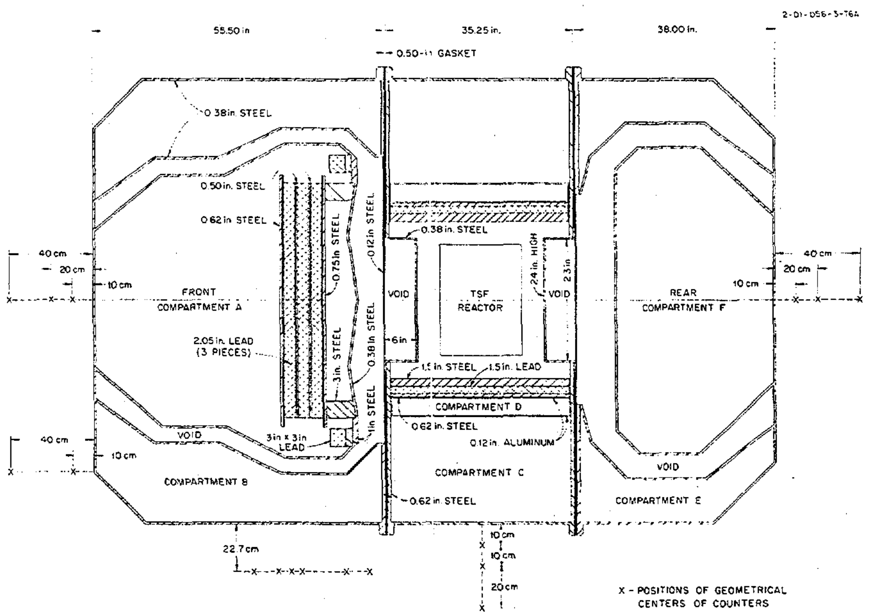

Fig. 5.2 Vertical Section of the Mockup of the GE-ANP R1- Reactor Shield.

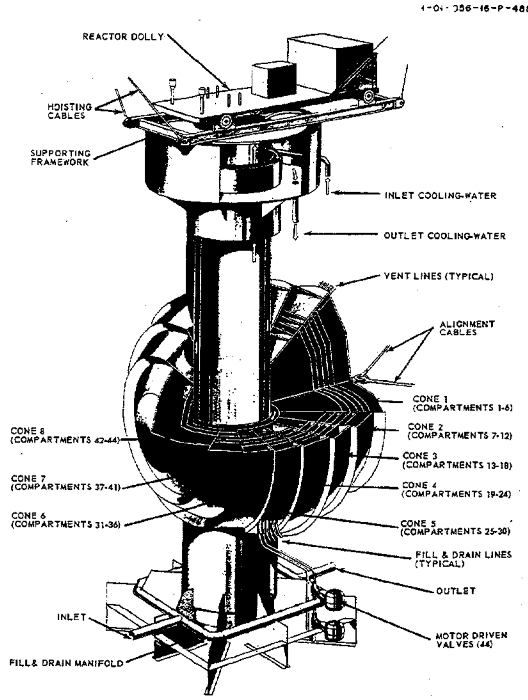

Fig. 5.3 TSR-I Compartmentalized Reactor Shield Tank.

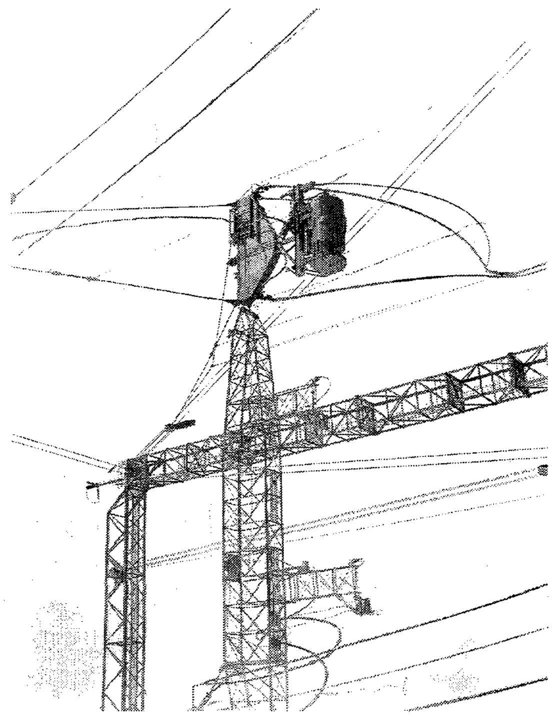

Fig. 5.4 The ASTR Shield (Reactor Included) Suspended in Air at the TSF.

5.5 TSR-II PRATT AND WHITNEY

The first reactor shield mockup using the TSR-II as a source was a shield designed by Pratt and Whitney Aircraft.[6] It was an optimized, highly asymmetric, uranium-lithium hydride shield contained in stainless steel, constructed to provide data for verifications of the calculations for design of an advanced-reactor shield. It consisted of four sections, an inner and three outer sections, as shown in Fig. 5.5. The inner section was 6.43 in. of lithium hydride, followed by a concave-convex lens-shaped shadow shield of depleted uranium. The outer sections were LiH, with the thickest section covering the uranium and the thinnest section directly opposite at 180°. The third section acted as a spacer between the other two sections. The shield was fabricated in sections so that it could be disassembled and the uranium shadow shield removed.

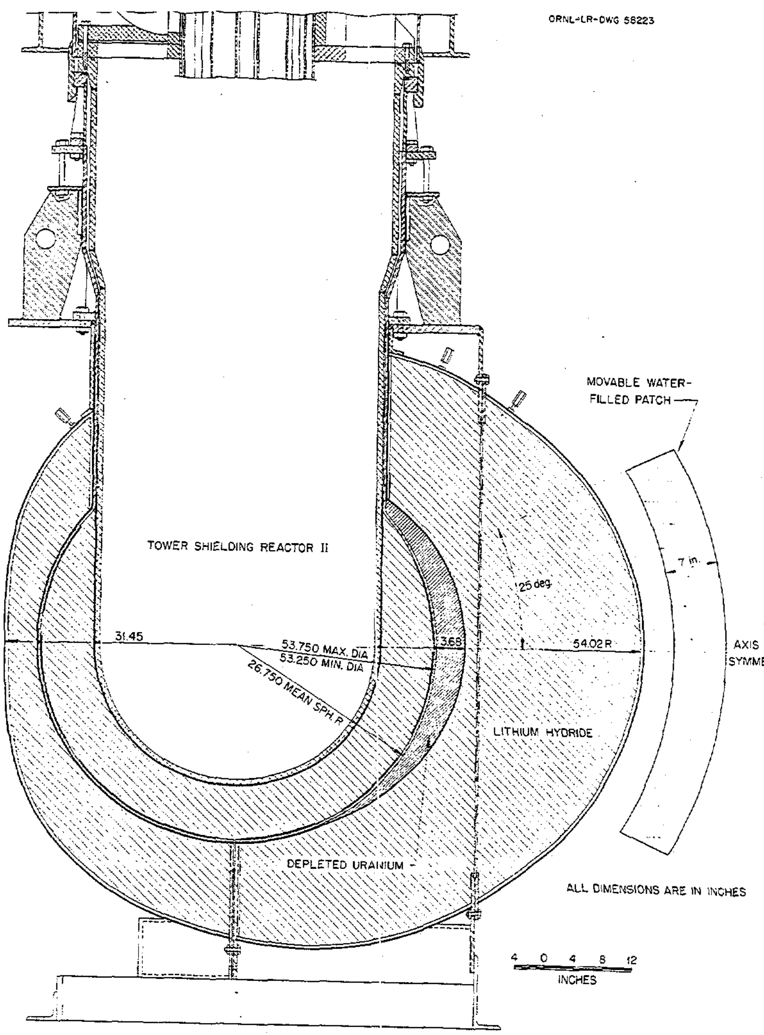

Fig 5.5 Cross-Sectional view of Pratt & Whitney Reactor Shield.

5.6 TSR-II COOL-I AND COOL-II

Two shield configurations, COOL-I and COOL-II,[7] were designed and built so that the COOL-II configuration could be obtained by a single addition of material to the outer surface of COOL-I. The two shields, shown in Fig. 5.6, were designed to differ only in the hardness of their fast neutron leakage spectra, COOL-II providing the harder spectrum. COOL-I consisted of 1.5 in. of lead placed around the reactor pressure vessel. To obtain COOL-II, an additional 3 in. of lead was followed by 0.25 in. boraxy, 7 in. borated water, and 02.5 in. plain water. Leakage properties such as fast-neutron to gamma-ray dose-rate ratio, thermal neutron leakage to gamma-ray flux ratio were calculated to be about the same for the two shields. The two shields were fabricated specifically for converting the TSR-II leakage radiation to resemble that from prompt weapon radiation.

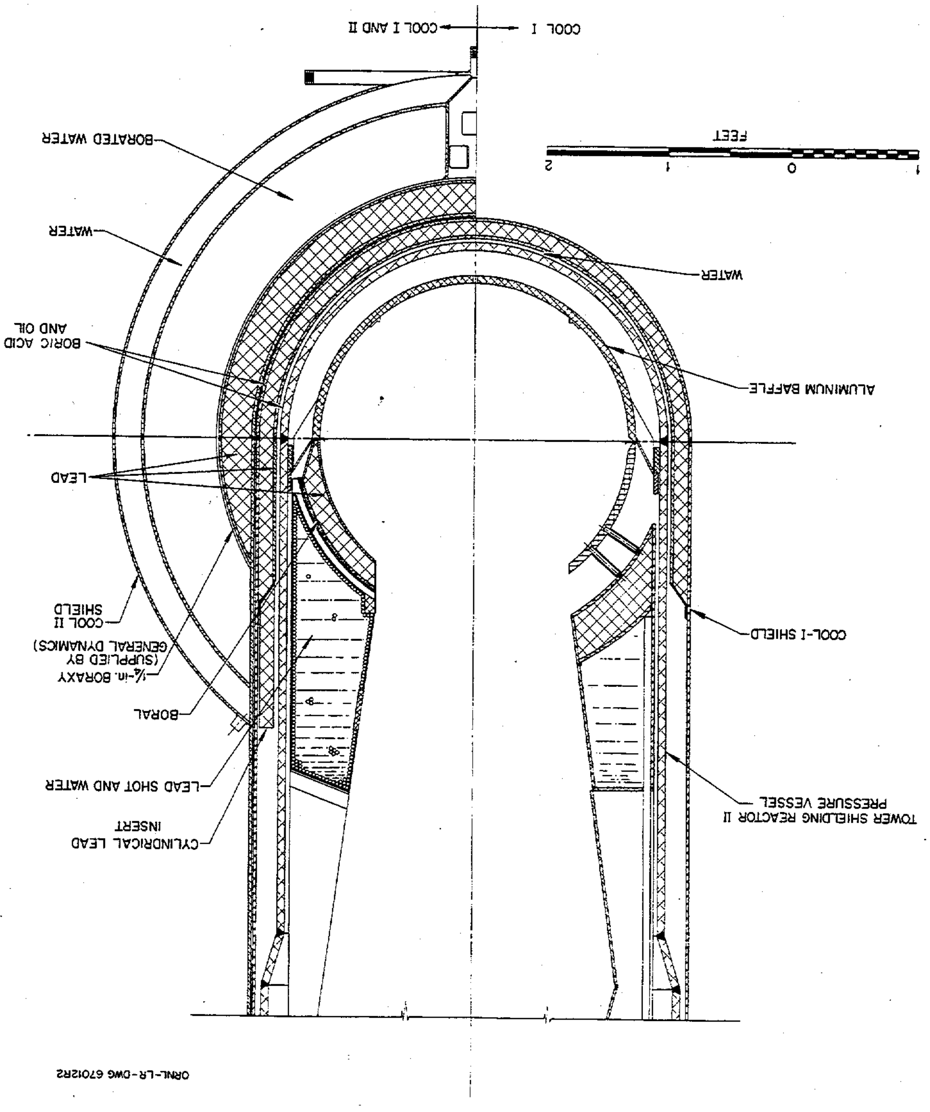

Fig. 5.6 TSR-II Shield Configurations COOL-I and COOL-II.

5.7 TSR-II SPHERICAL SHIELD

The spherical reactor beam shield[8] consisted of a lead-water gamma-ray shield followed by a water neutron shield, all contained in a 344-cm-diam sphere shown in Fig. 5.7. The lead was in the form of Raschig rings which have a packing fraction of 50%. The combined weight of the lead and water in the gamma-ray shield was 32,250 lbs. The neutron shield was 78.7 cm thick and weighed 38,500 lbs. Adding structure weight, the total weight of the beam shield was approximately 40 t.

The shield was designed to fit around the reactor pressure vessel and be supported in the air in the same fashion as the reactor vessel. The sphere and reactor were attached to the lifting hoist cables via the structure seen atop the sphere in Fig. 5.7. The sphere and reactor could then be rotated through 360° in the horizontal plane. The spherical shield was also fitted with two pipe-like extensions, one of which can be seen in the figure as projecting toward the viewer. A support arch, attached to these pipes, provided a means of rotating the sphere and reactor in the vertical plane as well as the horizontal plane.

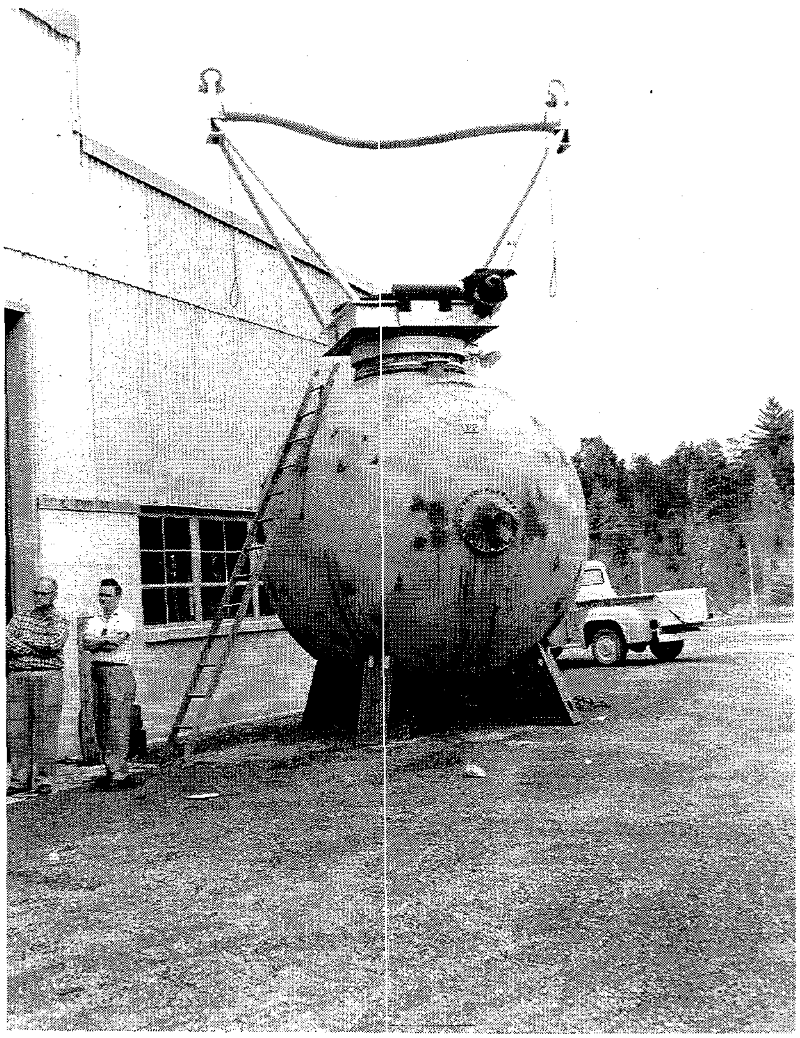

Fig. 5.7 TSR-II Beam Shield and Support for Vertical Rotation.

Two beam holes, spaced 180° apart in the same horizontal plane as the reactor center, extended through the spherical shield. These holes were 38.1 cm (15 in.) in diam at the outer periphery of the shield, but were stepped down to 25.4 cm (10 in.) in diam halfway through the shield. One or both could be plugged or left open. Legs attached to the outside of the sphere allowed the reactor to be ground supported and used to provide a collimated source in the horizontal plane 208.3 cm (82 in.) above the ground. The figure also shows one of the original TSF operating crew, J. L. Hull, in discussion with the fabricating crew during one of many meetings.

For non-airborne-type shield mockups, the TSR-II was initially kept inside the spherical reactor shield and the mockups to be studied were placed against the curved surface of the reactor shield collimator.[9] These slabs were then surrounded on the edges with concrete to help reduce background radiation reaching the detector. The meshing of the spherical vessel and the flat surface of the slabs led to considerable albedo scattering from the mockup which only increased the background component. Originally, small concrete and lithiated-paraffin blocks were stacked astutely around the shield-mockup interface to minimize this scattering. The procedure was tedious, time consuming, and not highly successful. To improve on this technique, in 1971 a concrete interface[10] was poured, in two pieces, around one half of the reactor shield sphere as shown in Fig. 5.8. The vertical face of the concrete away from the reactor shield was flat, to match the shape of the mockup slabs. The cylindrical collimator of the reactor shield was extended through the concrete facing. A lead shutter was placed within the concrete, whose vertical movement removed it from the collimator when experiments were performed but otherwise acted as a personnel shield. The added concrete face increased the source collimator length by about 90 cm.

5.8 TSR-II WATER SHADOW SHIELD

A shadow shield of water[11] contained in an aluminum vessel was fabricated to study the neutron scattering from different sized beryllium samples as a function of their position with respect to the reactor. These samples were to be representative of the control rods for the SNAP reactor. When in flight (operation), these control rods would extend from the reactor core and lie somewhat outside the cone formed by the lithium hydride shadow shield beneath it. To make measurements of the neutrons scattered from these samples, the detector was housed in the long pipe supported by a carriage mounted beneath the shadow shield that moved the pipe to the slope of the side wall of the water shield in a horizontal plane as seen in Fig. 5.9. The TSR-II served as the source as shown in the figure. During the measurements the shield was placed over the pool and the detector pipe submerged in the water to provide the needed collimation.

5.9 TSR-II LARGE BEAM SHIELD

By 1974, most of the experiments were being designed so that the position of the reactor source could always remain stationary inside its shield. There was also the need for an increase in source term intensity and this was not possible with the upper reactor power limited to 1 MW. These criteria led to the design and fabrication of what is now affectionately known as the "Big Beam Shield,"[12] a concrete structure that allowed full exposure of the reactor core to the test mockup instead of just the center of the core as in its spherical shield. This difference can be noted in Fig. 5.10, where the

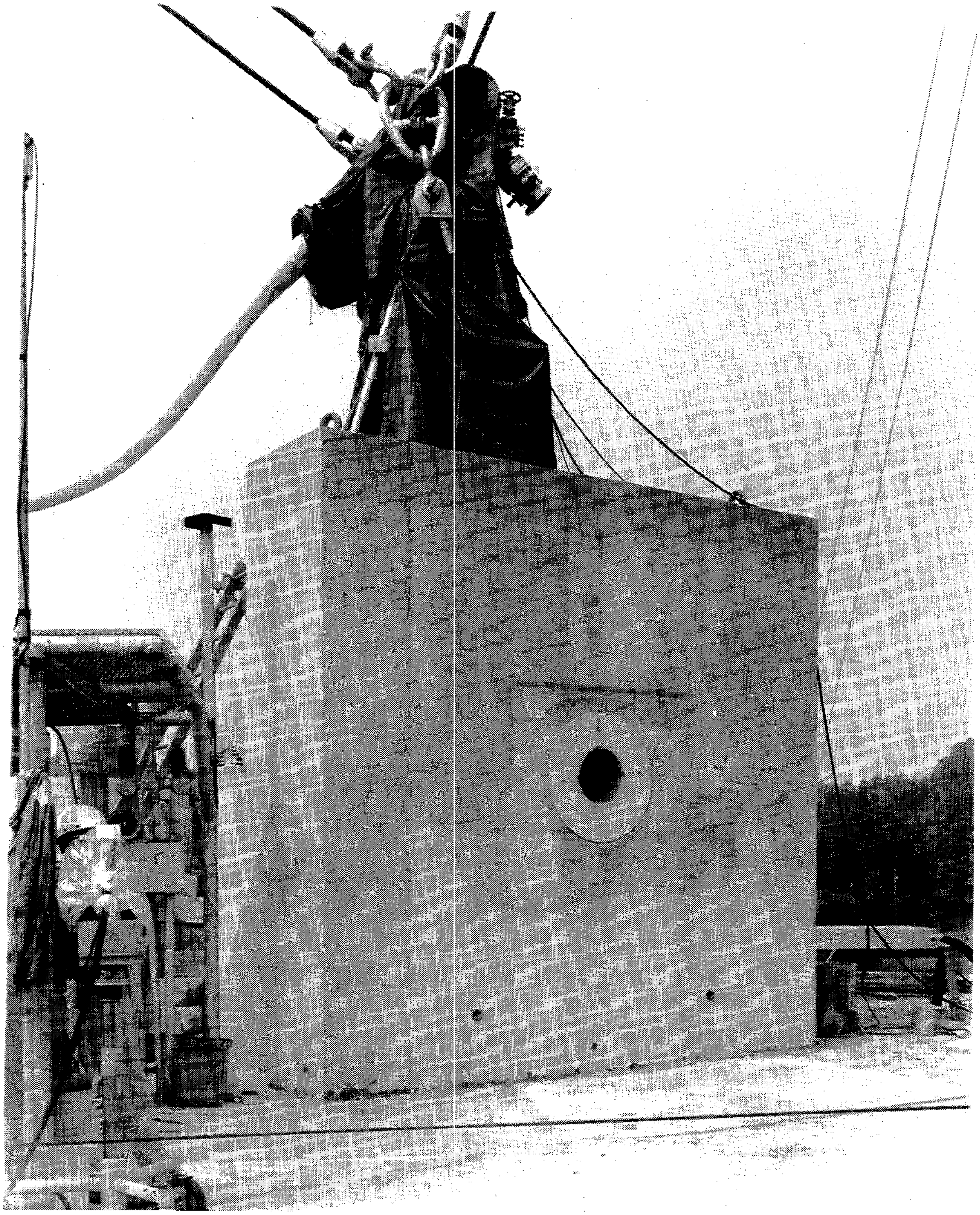

Fig. 5.8 Modified TSR-II Reactor Shield and Beam Collimator.

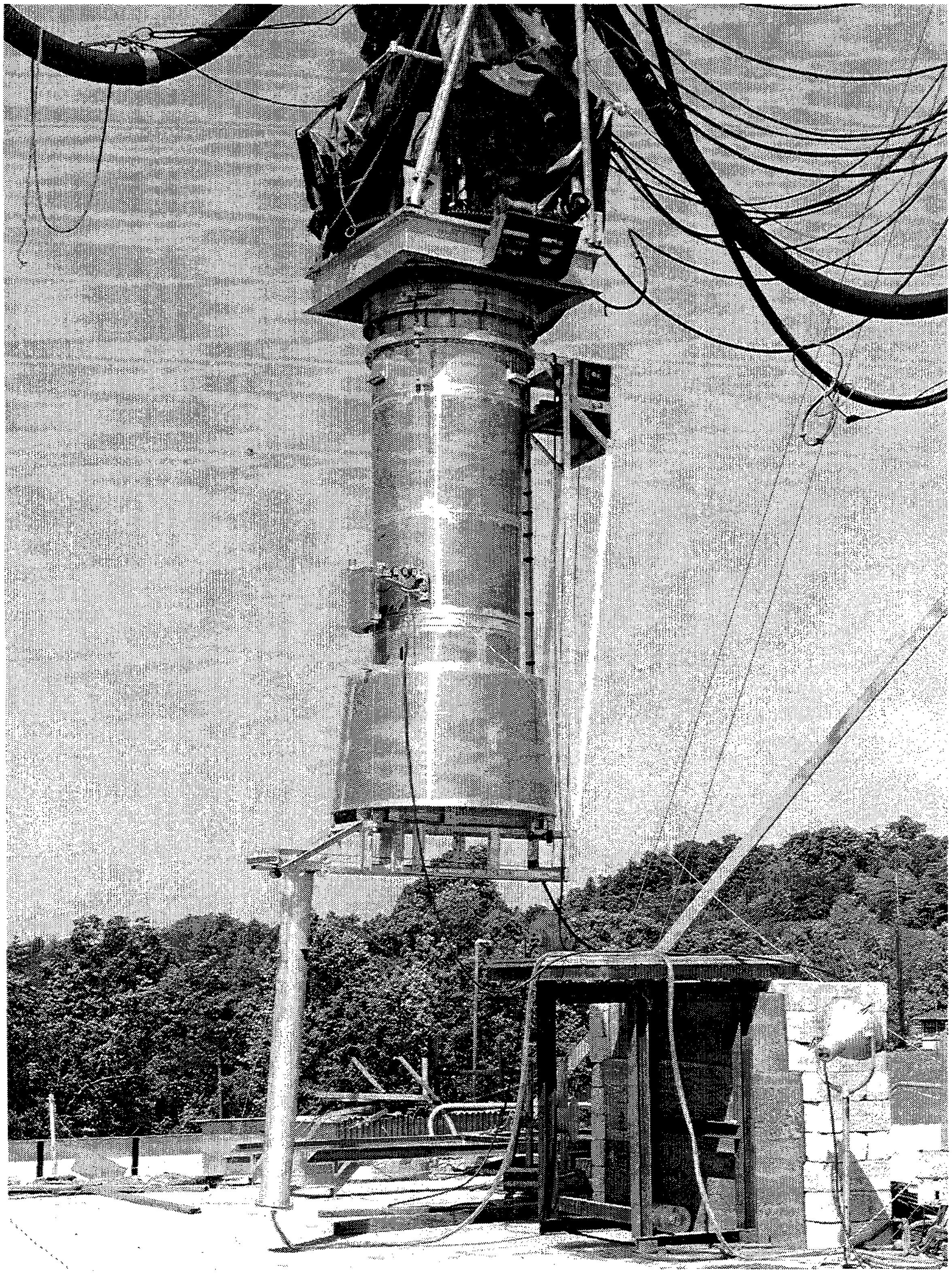

Fig. 5.9 TSR-II Water Shadow Shield Mockup of a SNAP Reactor Lithium Hydride Shield.

circumference of the spherical shield is superimposed on the new Big Beam Shield. This figure does not show, however, the concrete structure at the end of the spherical shield which makes the comparison even more dramatic. Movement of the shield samples much closer to the reactor and exposure of the sample to the full reactor core increased the intensity of the source term by a factor of 200 at the test mockup.

Placement of the reactor in this shield left a void area between the bottom of the pressure vessel and the concrete in the shield. It was necessary to duplicate this void spacing above the core by removing some lead and replacing it with appropriately shaped vessels of air. This change made the reactor source appear to be spherical to the mockups.

The interior of the concrete structure around the pressure vessel was lined with stainless steel and water to protect the concrete from the heat generated by the reactor's radiation. The opening in the shield that allows exposure of the reactor to the mockup was covered by a lead slab when the reactor was shutdown and a concrete slab with an opening as shown in Fig. 5.11 during performance of the experiment. The concrete slab with collimator and the location of the shield on the pad area is shown in the picture in Fig. 5.12.

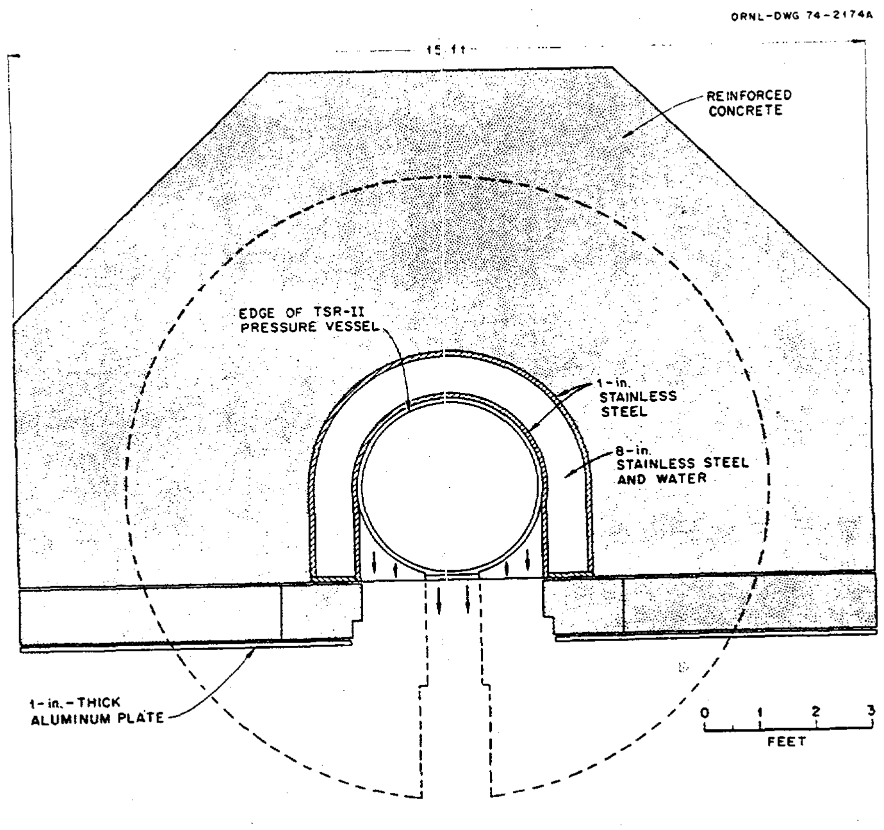

Fig. 5.10 Schematic of TSR-II Large Beam Shield through Horizontal Midplane.

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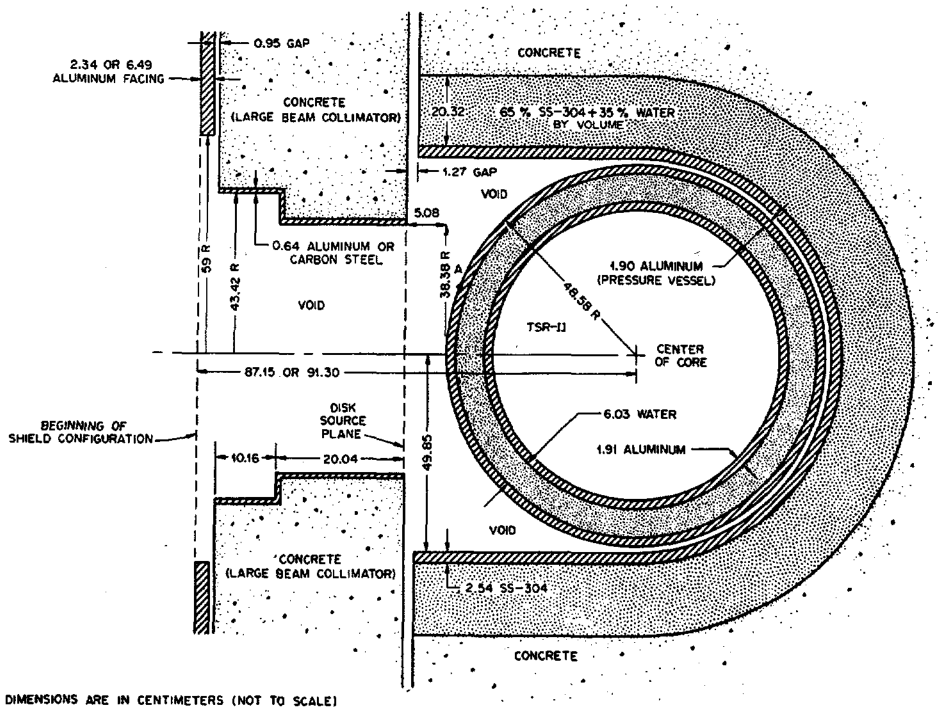

Fig. 5.11 Top view of Reactor and Large Beam Collimator Geometry.

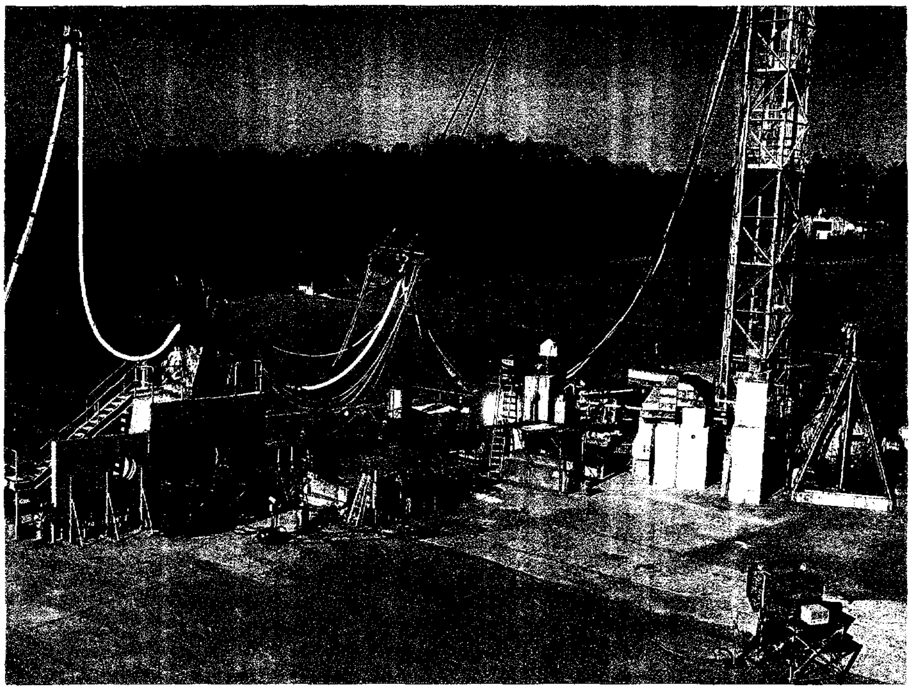

Fig. 5.12 A View of the TSR-II Positioned in its New Large Beam Shield.

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6.0 REACTOR INSTRUMENTATION

6.1 TSR-I

The TSR-I reactor was controlled by two motor-driven safety rods, each consisting of boron carbide in an aluminum jacket, and one servo-operated control rod.[1] The power level of the reactor was measured by four ion chambers, two of which were gamma-ray compensated, and by one fission chamber that was required for startup. All of the instrumentation was contained within the aluminum shroud positioned above the reactor lattice.

The length of each electrical interconnection between the reactor control chambers and the instrumentation in the Control Building was about 400 ft. These cables passed through a junction box attached to the hoist cable that supported the reactor shield and on to a second junction box located on the maintenance truss between two of the towers at the 100 ft level. This location permitted movement of the reactor from ground level up to a height of 200 ft without restrictions from the cables. The instrument cables then ran down one of the tower legs into the Control Building. The long length of these cables required that the preamplifiers be located at the reactor to provide adequate signal voltage to the amplifiers in the Control Building. These cables were then terminated at the Control Console, with signals fed to the various readout devices such as sigma amplifiers and Brown recorders positioned on a vertical panel within easy surveillance by the operator when seated at the console. The system was designed by E. P. Epler and L. C. Oakes of the Instrumentation and Controls Division, with guidance from C. E. Clifford.

6.2 TSR-II

The instrumentation system for control of the TSR-II was designed with the same basic philosophy in mind as was used for the TSR-I.[2] The neutron flux from source level to full power was measured by an ORNL-designed fission chamber and assorted electronic equipment whose output was presented in logarithmic form on a recorder. The chamber position was variable so that its usefulness could be extended to provide another five decades of count-rate information. The reactor power level was measured by a compensated ion chamber whose signal was amplified by a six decade log N amplifier which covered the range from 10 W to 1 MW. The reactor period was determined by differentiating the log N ion chamber signal. The reactor power level was controlled by a compensated ion chamber whose current was modified by a picoammeter, which in turn drove a servo amplifier that regulated the reactor power at a pre-determined level. The reactor safety system was composed of three uncompensated ion chambers whose current was modified by three independent safety amplifiers which drove an auctioneering bus to scram the reactor in the event the reactor power exceeded 150% of the maximum allowable power. All of these controls were mounted on the outside of a walk-in cubicle as seen in Fig. 6.1. Reactor power levels, startup period, and neutron flux outputs from the fission chamber were printed continuously on the Brown recorders.

A display of the coolant water system for the reactor was also displayed on the panel, with recorders maintaining a record of the rate of flow. This same display also contained off-on switches

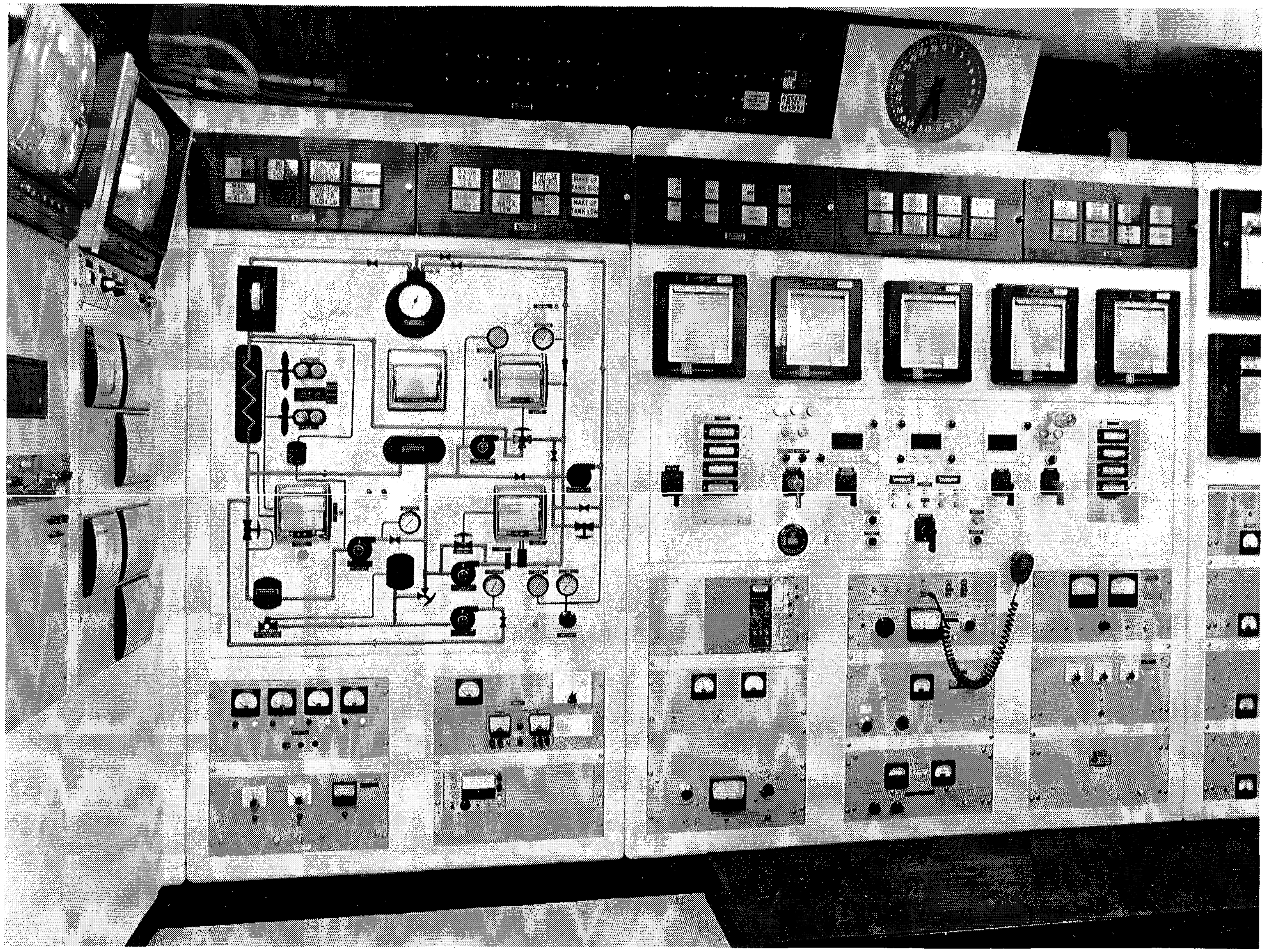

Fig. 6.1 TSR-II Control Console

for the motors that drove the pumps along with controls for regulating the rate of flow and water temperature.

Signals from closed circuit TV cameras mounted on the cross walks between the towers provided a scan of the outside area during reactor operation. These signals were displayed on the receivers also mounted in cabinets to the left of the coolant system panel. Radar was added to provide information about the weather within a 25 mile radius of the towers. This information was needed so that the reactor could be lowered for safety before thunderstorms or hazardous weather conditions materialized.

6.3 TSF-SNAP

The nuclear design of the TSF-SNAP reactor was identical to the SNAP-2 and SNAP-10 systems.[3] The nuclear instrumentation for reactor control and safety systems included a startup channel, log N channel, two level safety channels, and a linear flux channel. The startup channel consisted of a fission chamber preamplifier, and associated linear amplifiers whose output fed a log count-rate meter that provided the power level and reactor period data on two strip chart recorders. The fission chamber was mounted on a boom that could be displaced from the core at high power levels. The log N channel included a compensated ionization chamber, a log N amplifier, period amplifier, and composite amplifier, whose range extended down about six decades from full power. Both the period and level outputs were displayed on strip chart recorders. The chamber was attached to the reactor boom about seven feet from the core. Two non-compensated ionization chambers supplied the scram signals to the level safety channels that provided information to monitor the neutron flux level over the upper two decades of power and provide level scram signals to the magnet amplifiers controlling the air supply to the coarse control drums. Each channel contained a preamplifier and safety amplifier whose output was displayed on a meter. The linear flux channel was the compensated half of a dual ionization chamber whose output was fed to a picoammeter with range change capabilities. That output was displayed on a strip chart recorder and was used as an input to the automatic flux controller that was used to position one of the fine control drums when in the servo (automatic) mode of operation.

The control console, pictured in Fig. 6.2, was rejuvenated from the TSR-I days. The control instrumentation was located on the panel facing the console, both of which were in the immediate area of the TSR-II controls.

6.4 ASTR-CONVAIR/FORT WORTH

Controls for operation of the ASTR were provided by Convair/Fort Worth, and were essentially unchanged at the TSF. Where needed, the TSF electrical service was adapted for their use. The ASTR did, however, use the TSF aerial cables that ran from the sling boxes at the reactor to the control room. Description of their chambers or mode of operation are no longer available.

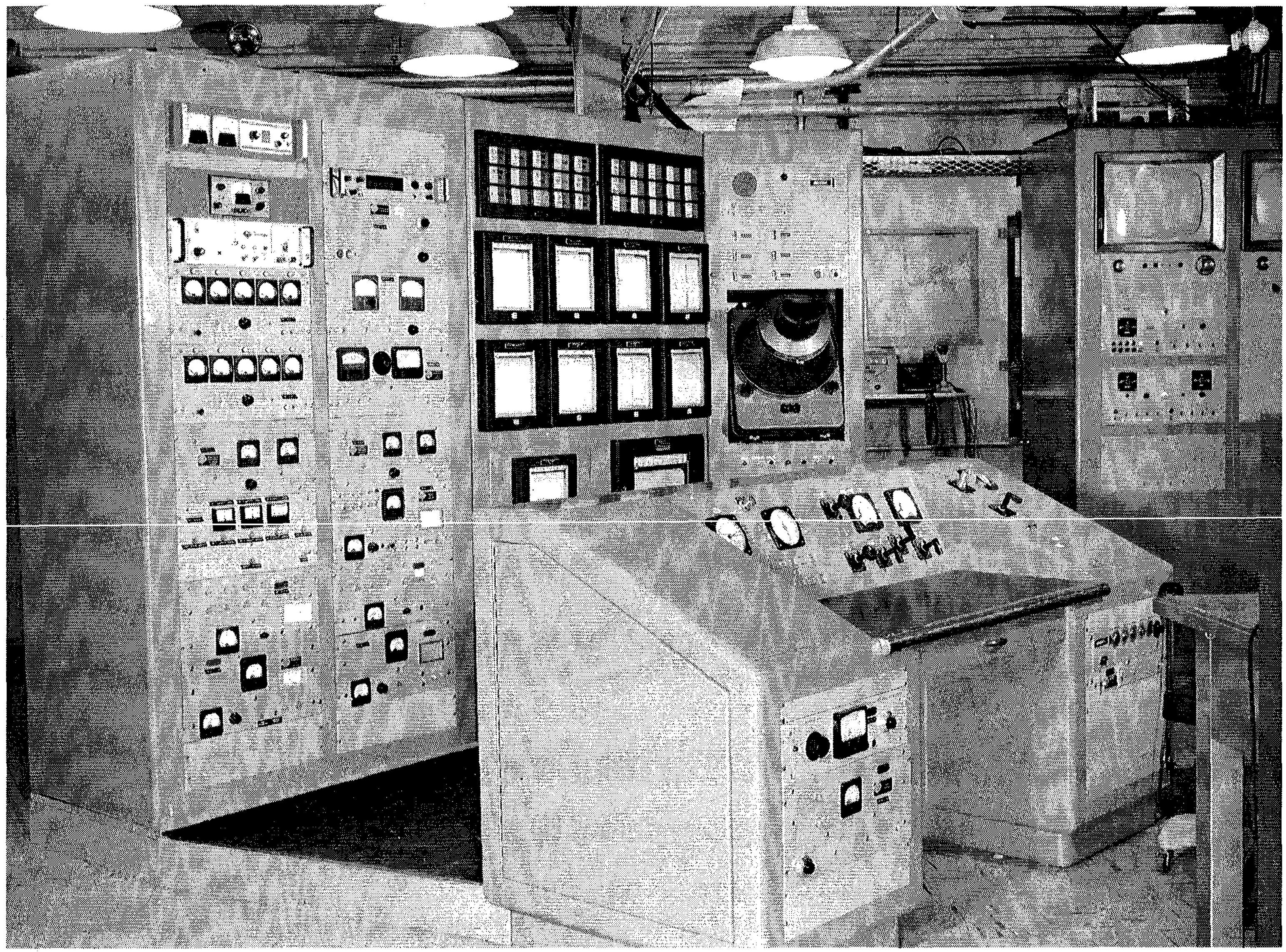

Fig. 6.2 SNAP Reactor Control Console.

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7.0 EXPERIMENTAL INSTRUMENTATION

7.1 INTRODUCTION

Considerable development work was required by the ORNL Instruments Division to produce a variety of radiation detection instruments needed for the initial experimental measurements at the TSF. Since measurements of the incident neutron and gamma-ray radiation on crew compartments were to be made in the open air (rain or shine) as well as in water, the detector housings were designed to be waterproof and kept at a positive gas pressure. Radiation levels outside the control house required that the detectors be positioned remotely during the experiments. All this necessitated the use of aerial cables that ran from the detectors and positioning devices to the truss between the tower legs as described earlier for the reactor control cables. Preamps were located adjacent to the detectors to prevent excessive loss of signal over the long lines. All of the detectors, except the ion chambers and anthracene scintillators, which were used for gamma-ray detection, were proportional counters designed to measure the thermal flux and fast neutron dose.

The original group of detectors consisted of boron trifluoride-filled (BF_3) thermal neutron counters, Hurst-type fast neutron dosimeters, (so named after its developer, G. S. Hurst of the Laboratory) fission counters, graphite-lined gamma-ray ionization chambers, anthracene crystal scintillators, and indium and gold foils. The detector covers were usually made of lucite, and there was about 6 in. of lucite between the detector and preamp to prevent radiation streaming paths through the preamp area when submerged in water. These detectors are described in Sects. 7.2-7.7.

With time, improvement in the development of analytical methods for calculating reactor shields called for greater refinement in the accuracy of the experimental data. This led to the gradual shift from integral dose-rate measurements to differential neutron and gamma-ray spectra measurements. The Hornyak button detector system was developed to provide a more sensitive neutron dosimeter, but its usefulness became more important in that its sensitive area could be considerably reduced to give spatial definition to neutron profiles across small voids. The need for development of both neutron and gamma-ray spectrometers became very important, the first being a 3-in.-diam sodium iodide (NaI) crystal to measure the gamma-ray spectra. This was soon followed by a 5-in.-diam crystal that gave much better resolution. Neutron spectrometers using 2-in.-diam NE-213 liquid scintillators were introduced in the early 1960s as useful devices to measure neutron energies above 1 MeV. Benjamin-type hydrogen-filled proportional counters were soon added to measure the neutron energy from about 10 keV to 1 MeV. The energy range from thermal to 10 keV was covered by a boron-filter spectrometer developed by T. V. Blosser. At the start of the 1970s, spherical Bonner balls of different polyethylene thicknesses were developed at the TSF to provide integral neutron measurements against which calculational methods could be easily compared.

Considerable interest in the development of methods for calculation of the heat deposition occurring in reactor shields inspired the development of an experimental program to make energy measurements using small passive *in-situ* type devices called thermoluminescent detectors. All of these contemporary detectors are discussed in detail in Sects. 7.8-7.14.

7.2 THERMAL NEUTRON DETECTOR (BF_3)

The first thermal neutron detector used at the TSF was a brass-walled cylindrical chamber 17 cm long and 5.2 cm in diam with a central wire running the length of the detector. The chamber was filled with boron trifluoride (BF_3) gas to a pressure of one atmosphere.[1] The thermal neutron sensitivity of the counter was calibrated against the thermal flux measurements obtained with indium foils. Thus, bare minus cadmium-covered count rates of the BF_3 counter could be easily converted from counts/min to thermal neutron flux nv_{th}.

Pulses generated within the BF_3 detectors were amplified about a factor of 15 in the preamps and passed through about 500 ft of coaxial cable to A-1 amplifiers in the control house. The output was then fed to scalers and/or through count-rate meters to recorders. The detector assembly consisted of either one or a combination of three chambers (Fig. 7.1), the latter being used only as a means of increasing the count rate. Gas pressure was maintained between the brass chambers and the lucite casing that enclosed the detector and electronics to prevent leakage or moisture.

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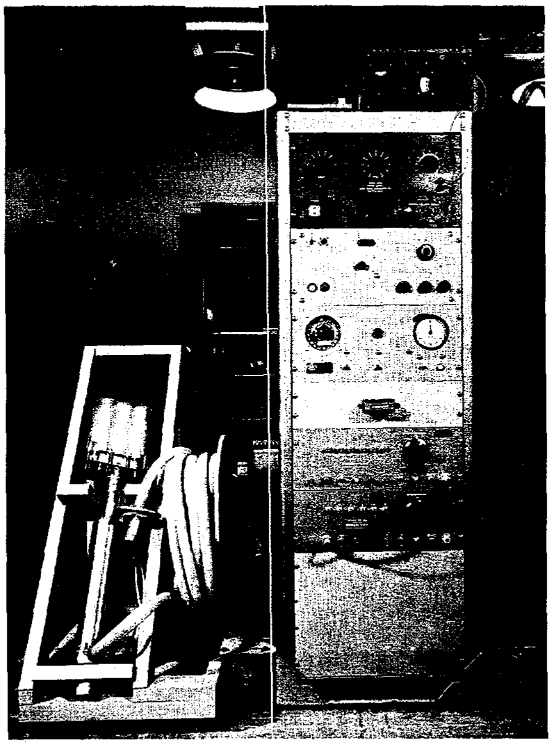

Fig. 7.1 Triplet BF_3 Chamber and Associated Electronic Panel.

7.3 HURST-TYPE DOSIMETER

The Hurst-type[2] fast neutron dosimeter consisted of a brass-walled cylindrical chamber that was divided into three smaller chambers containing a common central wire (Fig. 7.2). The three chambers were lined with polyethylene and filled with ethylene gas to an absolute pressure of one atmosphere.

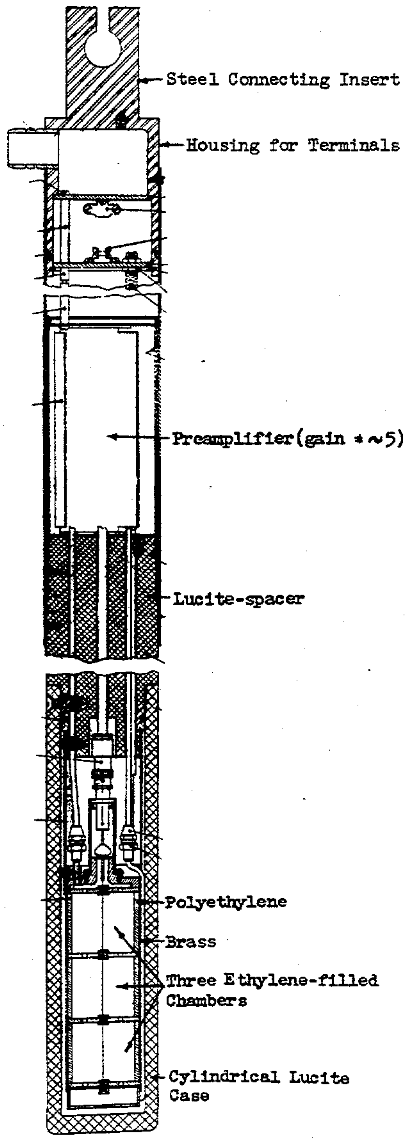

Fig. 7.2 Fast Neutron Dosimeter.

As with the BF_3 detectors, the outside housing was made of lucite and the space between the housing and detector-electronics was kept under gas pressure to keep water out. Pulses from the proportional counter were amplified by about a factor of 5 in the preamp and sent through the cables to an A-1 amplifier where the amplitude was increased by about a factor of 1000. These pulses were sorted using a device with seven discrimination channels. The sorted pulses were fed to the inputs at intermediate scaler stages of two binary scalers and the weighted sum as recorded by the scalers was proportional to the fast neutron dose rate.[3]

The fast neutron dosimeter was calibrated using a known polonium-beryllium (PoBe) source strength previously calibrated at the National Bureau of Standards (NBS). The dose rate, in mrep/h, was calculated using the known PoBe energy spectrum, the single collision energy deposition as calculated by Hurst and Ritchie, and the NBS calibration. The triple-barrel dosimeter, built to increase detector sensitivity, consisted of three of the single dosimeter tubes connected in parallel and enclosed in a single plastic housing.

7.4 FISSION CHAMBER

The fission counter was a proportional counter consisting of uranium (^{235}U) sputtered on a nickel plate encased in a brass cylinder filled with argon to a pressure of 6 psig. The sensitive element and its accompanying preamp were housed in containers similar to those previously shown for the neutron dosimeter and BF_3 detectors. The preamps were of the "tube type" as were all the preamps at the TSF prior to 1960. Response of the detector to thermal neutrons was compared to data obtained with indium foils. Pulses from the detector were counted directly using similar electronics as described for the BF_3 counter.

With the transition to the TSR-II, it was felt that some changes were necessary in detector designs. A big problem was the maintenance of the vacuum tubes in the preamps. The problems with vacuum tube preamps were resolved by replacing them with transistors. The following is provided by the author:

> In 1959, I asked Floyd Glass of the Instrumentation and Controls Division if he would consider converting the preamps from tubes to transistors, a new device that was just then entering the market, but which had no history of use in pulse type applications. His reply was that it wouldn't work because transistors weren't fast enough and nobody knew very much about them. Hence, he turned down my request. I persisted, giving him a $3000 work order and suggested if it didn't work, there was very little lost. Reluctantly, he gave in, still denying its possibilities. I said, "Fine, I'll call you back in two weeks." His last words were: "Don't get your hopes up!"
>
> Before the two weeks were up, Floyd called, his voice in high spirits, saying, "I think it is going to work," and it did. To my knowledge, that was the first time transistors had been used successfully in pulse work anywhere, certainly here at the Laboratory. The same preamps were still in use in those fission chamber detectors as pictured in Fig. 7.3 when the TSF shut down in 1992.

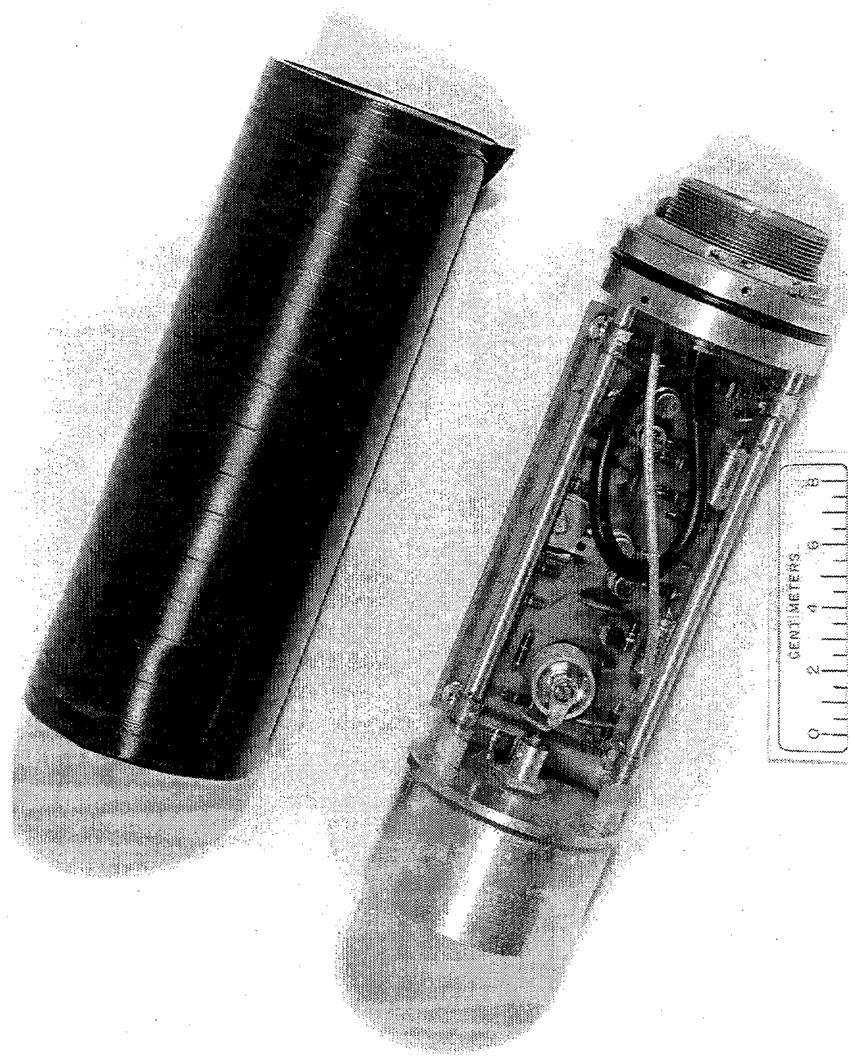

Fig. 7.3 Single Plate Fission Chamber in use after 1959.

7.5 IONIZATION CHAMBER

The 900-cc ionization chamber had graphite walls and a graphite central electrode (Fig. 7.4), the chamber being filled to a 6 psig pressure of argon gas and surrounded by a lucite housing.[1] A modified two-tube electrometer (ORNL-Q-826)[4] was used to measure the ionization current produced within the chamber. A potentiometer in series with the grid resistor was used to maintain balance between the electrometer tubes as indicated by a galvanometer in the plate circuit. The potentiometer gave the voltage drop across the grid resistor from which the ionization current was obtained. The chamber was calibrated against the known strength of a Co^{60} source.

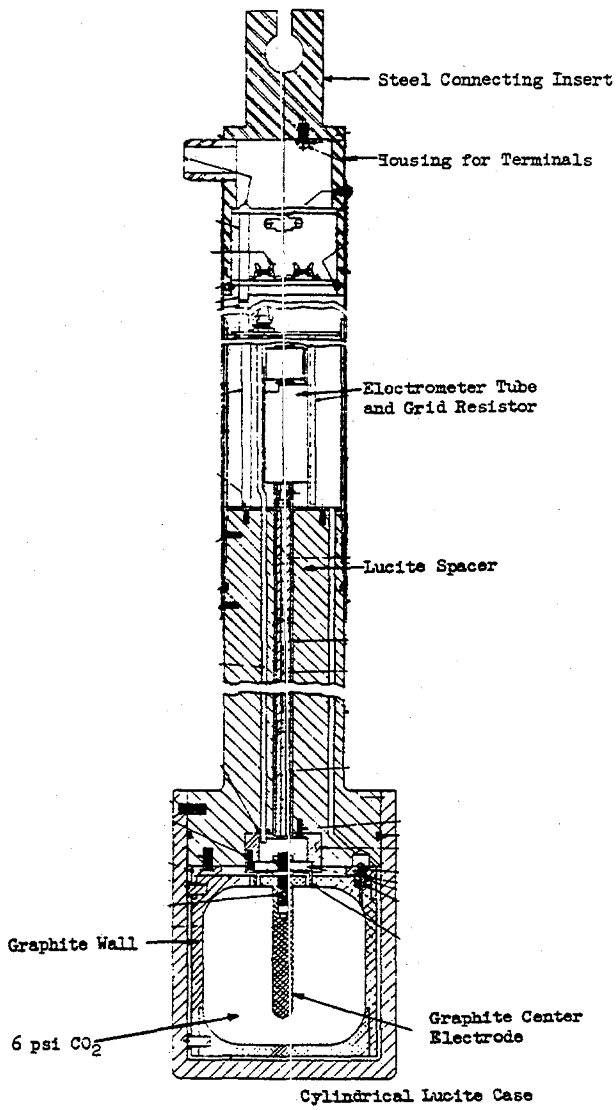

Fig. 7.4 900-cc Gamma-Ray Ion Chamber.

7.6 ANTHRACENE SCINTILLATOR

The gamma-ray scintillation detector used for dosimeter measurements consisted of a 1-in. right cylinder of anthracene imbedded in plastic and mounted to a RCA 5819 photomultiplier tube using a lucite light pipe (Fig. 7.5). The unit was enclosed in an aluminum can and pressurized with carbon dioxide gas to keep the detector dry against the weather. The current flow from the photomultiplier tube from events occurring in the scintillator was fed into an electrometer system similar to that used for the ion chamber. Calibration was identical to that described for the ionization chamber.

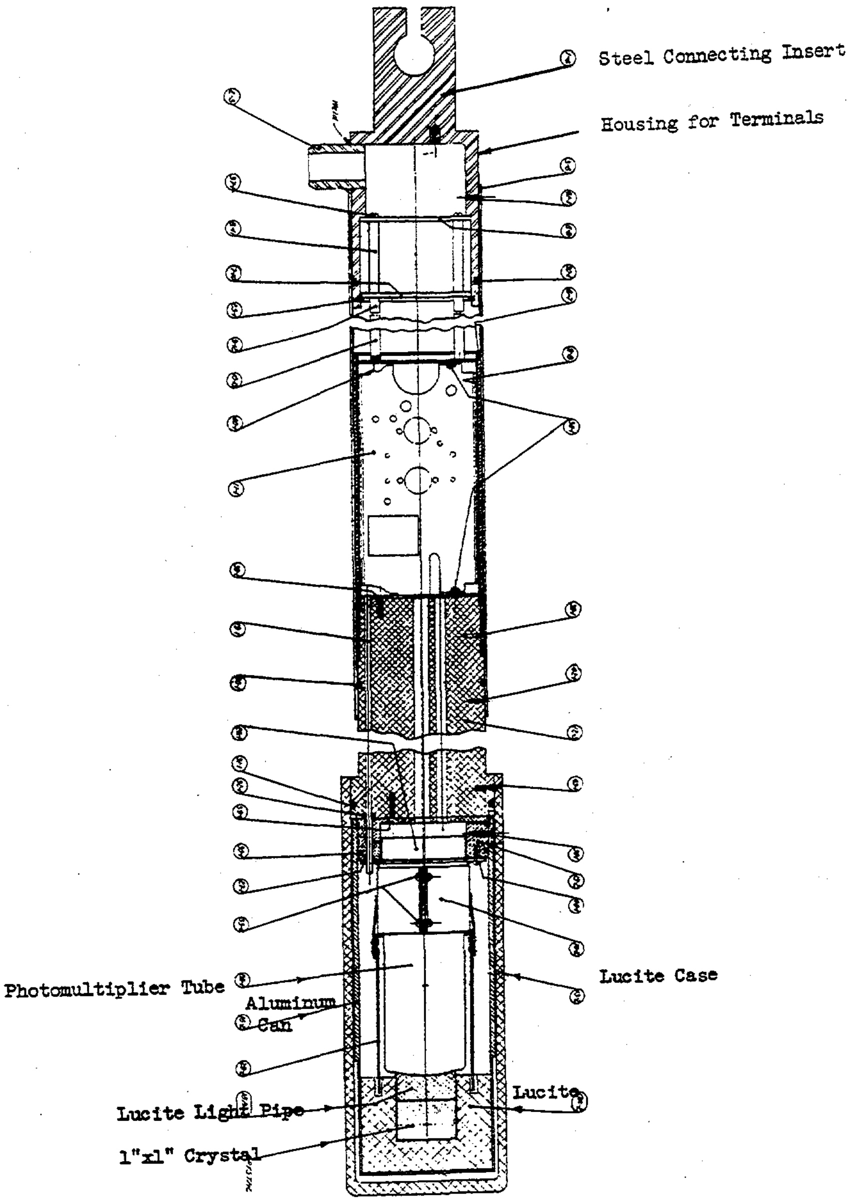

Fig. 7.5 Anthracene Crystal Scintillation Detector.

7.7 HORNYAK BUTTON

In 1956, an effort was initiated to develop a new detector, called the Hornyak button after its originator, W. F. Hornyak,[5] for measuring fast neutron dose. This development was deemed necessary because measurements using the standard Hurst-type dosimeter were being made in high intensity gamma-ray fields and that dosimeter was responding to the gamma rays. The button consisted of a homogeneous mixture of fine lucite and zinc sulfide (ZnS) powders pressed into cylinders and heat-treated at 375° for several hours. These were fabricated for the TSF by R. K. Abele's group in the Instrumentation and Controls Division. Once fabricated, the buttons were machined to the desired diameter and thickness as needed. Light reflection from the surface of the button was controlled by enclosing it in a light reflector or, as in the case for use at the TSF, coated with a layer of white paint.

The button was mounted on a photomultiplier tube and the pulses fed through the preamp and amplifier to a multichannel analyzer or to a scaler. Comparison measurements against the Hurst-type dosimeter indicated that for a 1.6-mm-thick (~ 0.060 in.) button, the responses of the two detectors were very similar except for energies below about 200 keV, where the dosimeter response was about 50% higher. Even this difference was reconcilable by being selective in the size of the ZnS particles, namely, for those 3 to 5 microns in diameter. In high gamma fields, 100 r/h and greater, counts from neutrons in this energy region were overshadowed by the gamma-ray counts, making it necessary to bias either of the detectors to higher neutron energies where the response ratio between detectors was known to be one. Since the Hornyak button was more sensitive, it was used when the extra sensitivity was needed and/or the gamma-ray intensity was quite large.

Its use had even greater importance when the button diameter was reduced from 5.08 cm to 0.635 cm, while keeping its thickness the same. This size of button provided an ideal detector for measuring the spatial distribution of neutron streaming through small voids in mockups. That size button no longer provided a measurement of absolute neutron dose rate; its usefulness was its ability to define the structure of the neutron intensity changes in mockups being studied. The button, large or small, was calibrated using a Co^{60} source to set the energy bias and, for that bias, the count rate from a known intensity neutron source was still recorded in units of dose rate. For use in high gamma-ray fields, the energy bias would be raised and the same calibration technique used.

7.8 SODIUM IODIDE CRYSTAL

Gamma-ray spectra were first obtained at the TSF in the early part of 1956 using a 3-in.-high × 3-in.-diam NaI crystal and a portable 3 channel analyzer. Window widths of 1 MeV were established to cover the spectrum from 0 to 12 MeV while widths of 0.2 MeV were used to obtain finer details of the spectrum for the energy span from 0 to 2.4 MeV. The pulse-height data were converted to energy spectra using a series of single-energy response functions whose window widths were the same as those used to obtain the pulse-height data. The following year, the energy resolution was improved by replacing the 3 channel analyzer with a 20 channel analyzer for measurement of the decay spectra from elements within the J-57 engine following its exposure at the TSF. During 1958, the resolution was increased further to several hundred channels when spectra were obtained during the ASTR experiment at the TSF. This resolution was also used during the measurement of the air capture spectra later that year.

It was not until the later part of 1966 that gamma-ray spectra were again measured at the TSF using the 3-in. NaI crystal, this time as part of the DASA program. This experiment included the repeat of a series of measurements using single-energy gamma-ray sources, both from neutron capture and radioactive decay, to generate both relative and absolute response functions so that spectra obtained for shield samples in the beam could be properly analyzed.

Shortly after that experiment in late 1967, the quality of the gamma-ray spectral measurements was significantly upgraded with the replacement of the 3-in.-diam NaI crystal with one 5-in.-diam by 5-in.-high (Fig. 7.6) purchased from Harshaw Chemical Company. To provide weather protection and

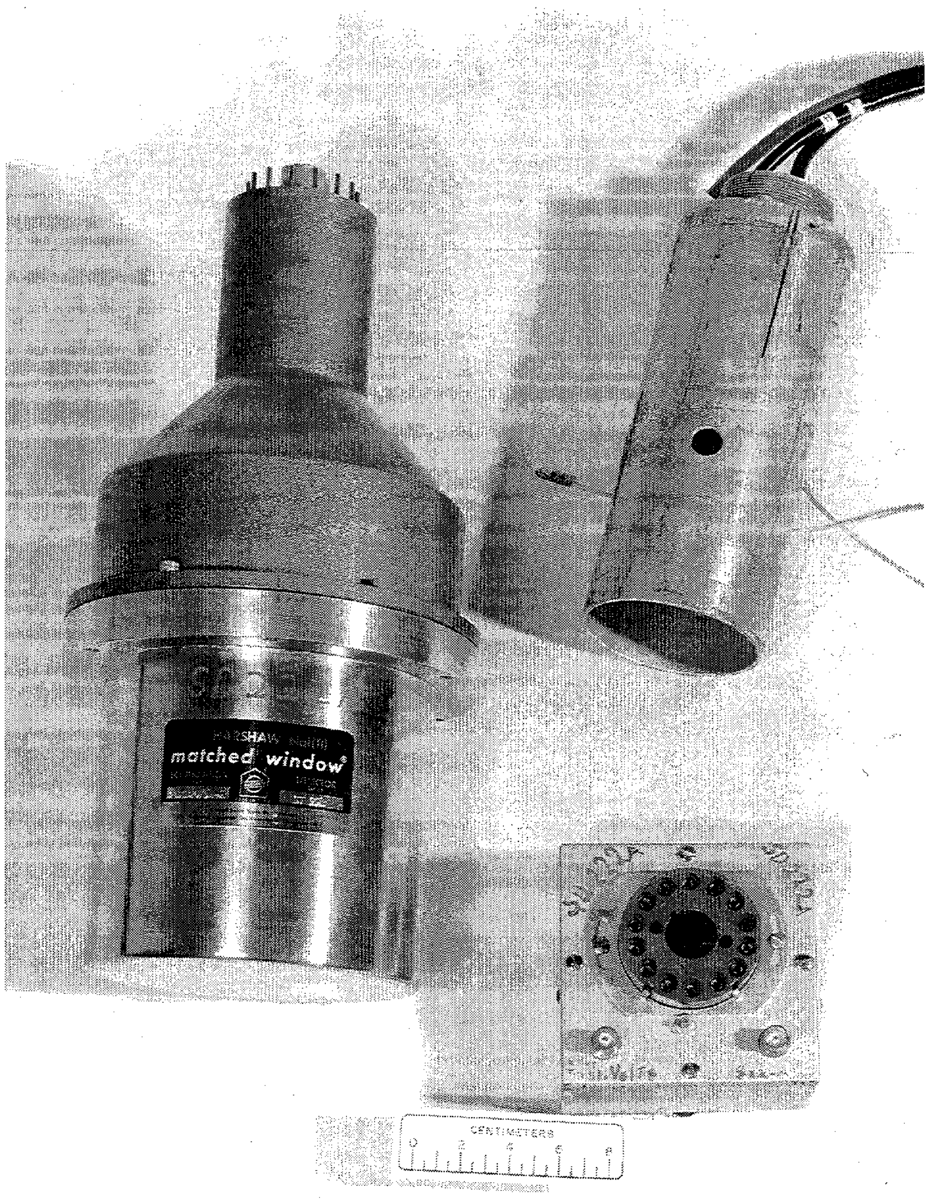

Fig. 7.6 5-in. Sodium Iodide Gamma Ray Spectrometer.

good temperature control, along with good collimation, the 11-ft-diam lead-water shield that was fabricated in the late 1950s for use in the ANP program was adapted to accept the 5-in.-diam crystal. A series of special lead irises were fabricated to minimize gamma-ray scattering from the collimator to the crystal. Response functions were generated for this detector system using single-energy sources and adapted by R. E. Maerker[6] for "unscrambling" the data collected using a 512-channel Nuclear Data (ND) analyzer, until 1971, when the responses were regenerated using a newly purchased 1024-channel ND-2200 analyzer. In late 1986, an IBM compatible PC (personal computer) replaced the 2200 system.

7.9 NE-213 SPECTROMETER

In 1963, the first neutron spectral measurements were made at the TSF using the liquid, organic scintillator NE-213 enclosed in a glass bottle and mounted on a 2-in.-diam photomultiplier tube (Fig. 7.7). Two designs of the bottle envelope were fabricated, one with and one without an added reservoir that maintained a full complement of the liquid in the bottle when the detector was placed in a horizontal position.

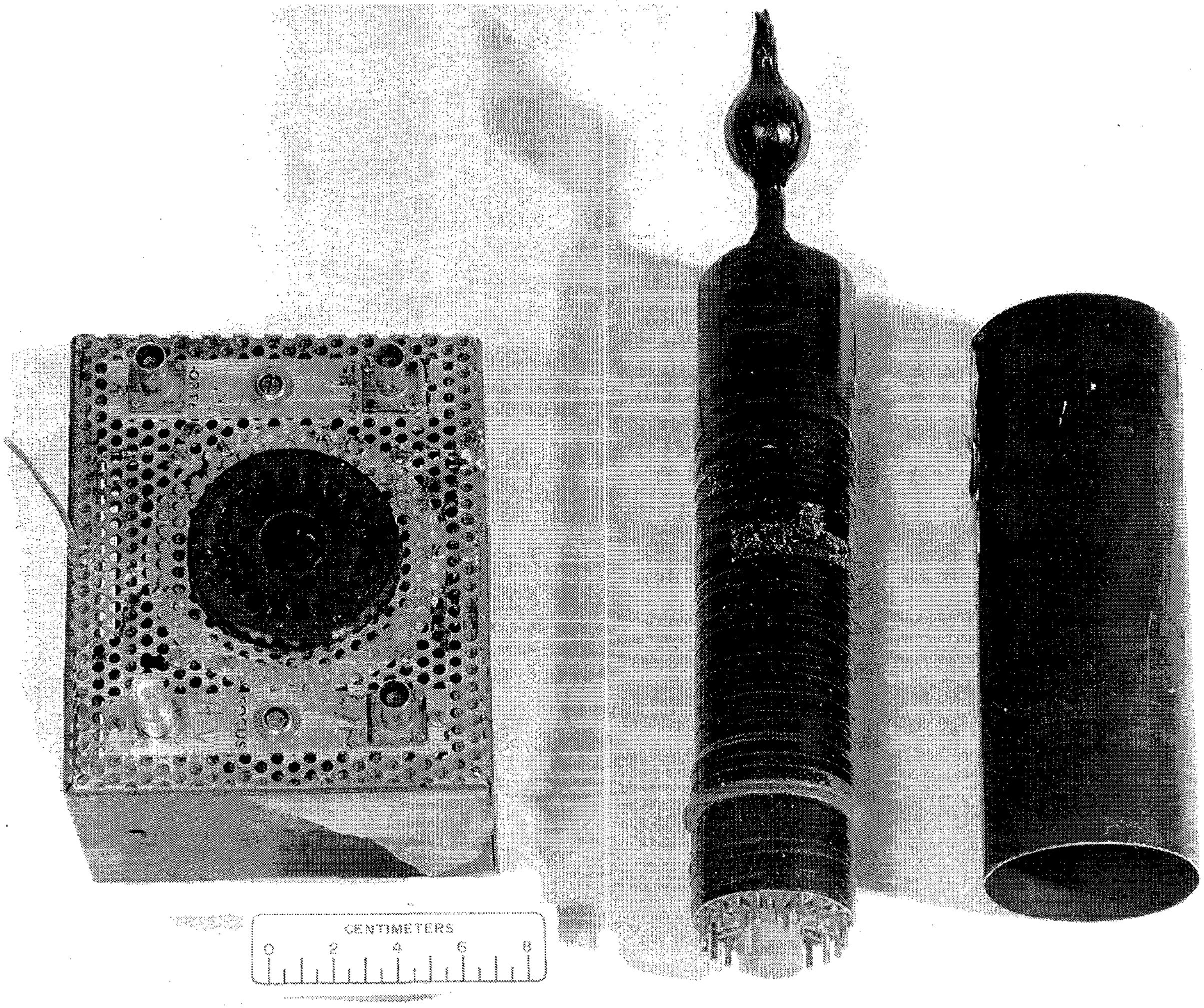

Fig. 7.7 Glass Bottle NE-213 with Reservoir.

The NE-213 liquid, purchased from Enterprises LTD in Canada, contained 5.5×10^{22} hydrogen atoms per gram of fluid and had a density of 0.876 gm/cc. Light pulses generated from both neutron and gamma-ray activity were recorded by the electronic system and they were separated and stored in separate halves of the analyzer through the use of a modified Forte-type pulse shape discrimination circuit.

So that the full energy spectra of both the neutrons and gamma rays could be obtained with a limited storage capacity, the measurements were made originally in two energy bites that differed in gain by a factor of ten. Thus two separate runs were needed. It was not until several years later that a second 512-channel nuclear data analyzer became available, allowing the two energy bites to be stored simultaneously. Output data from the analyzers were punched onto black tape and stored for future unscrambling when such a method became available.

At the time of these original measurements, there was no procedure for converting the pulse height data to spectra. It was not until about three years after our first measurements that W. R. Burrus was successful in developing the code FERDOR that provided transformation of the pulse height data into neutron spectra. To unscramble the TSF data, it was necessary to combine the black tape data for both of the energy bites.

The NE-213 liquid has good pulse-shape discrimination capabilities[7] because it contains an enhanced emission of delayed light. Being a liquid, it is also isotropic in response to neutrons. For the TSF work, the glass envelope was surrounded on top and sides by a shiny aluminum foil as a light reflector. A light pipe was placed between the glass envelope and the photomultiplier tube to provide good light passage through the curved face of the photomultiplier.

In 1977, a Nuclear Data 4420 computer system replaced the two 512-channel analyzers and the peripheral electronics was also changed from the Forte-type discrimination circuit to that of a cross-over pick-off method for differentiating between gamma rays and neutrons. At the same time, the glass bottle NE-213 scintillator was replaced by one with an aluminum housing designed and built at ORNL. In 1987, the spectrometry system was again updated briefly by bringing on line a PDP-11 computer and program previously used by G. T. Chapman to operate the data acquisition system. Another updating occurred in 1990 by replacing the PDP-11 with an IBM PS/2 Model 80-111 personal computer[8] to control the data handling and analyze each event. The system was adapted to use a commercially-available interface card, MC-D10-32F, purchased from National Instruments Corporation.

7.10 BENJAMIN-TYPE PROPORTIONAL COUNTERS

Benjamin-type hydrogen-filled proportional counters were fabricated at ORNL in 1969 for measurements of the neutron energy in the region below that covered by the NE-213 spectrometer. The detectors were modeled after the British version[9] but were somewhat larger in diameter, namely 5.08 cm (Fig. 7.8). Instead of adding a trace of helium to the hydrogen fill gas, as the British did for calibration purposes, the TSF detectors had an alpha source placed at the middle of the center wire. The energy deposited by the alpha source within the detector was calibrated against known neutron energies from a Van de Graaff generator. Three detectors, each filled to different pressures of 1, 3, and 10 atmospheres, were needed to cover the energy range from about 1.5 MeV down to about 50 keV, since the energy range for each detector was somewhat limited.

Years later, new 1-, 3-, and 10-atmosphere detectors were built with the alpha source placed in a collimator attached to the detector's surface instead of on the wire, decreasing the time needed for calibration and greatly improving the resolution of the alpha peak.

A fourth detector, filled with a gas mixture of 85% hydrogen, 10% methane, and 5% nitrogen to a pressure of 3 atmospheres, was developed later by K. M. Henry and was used to measure

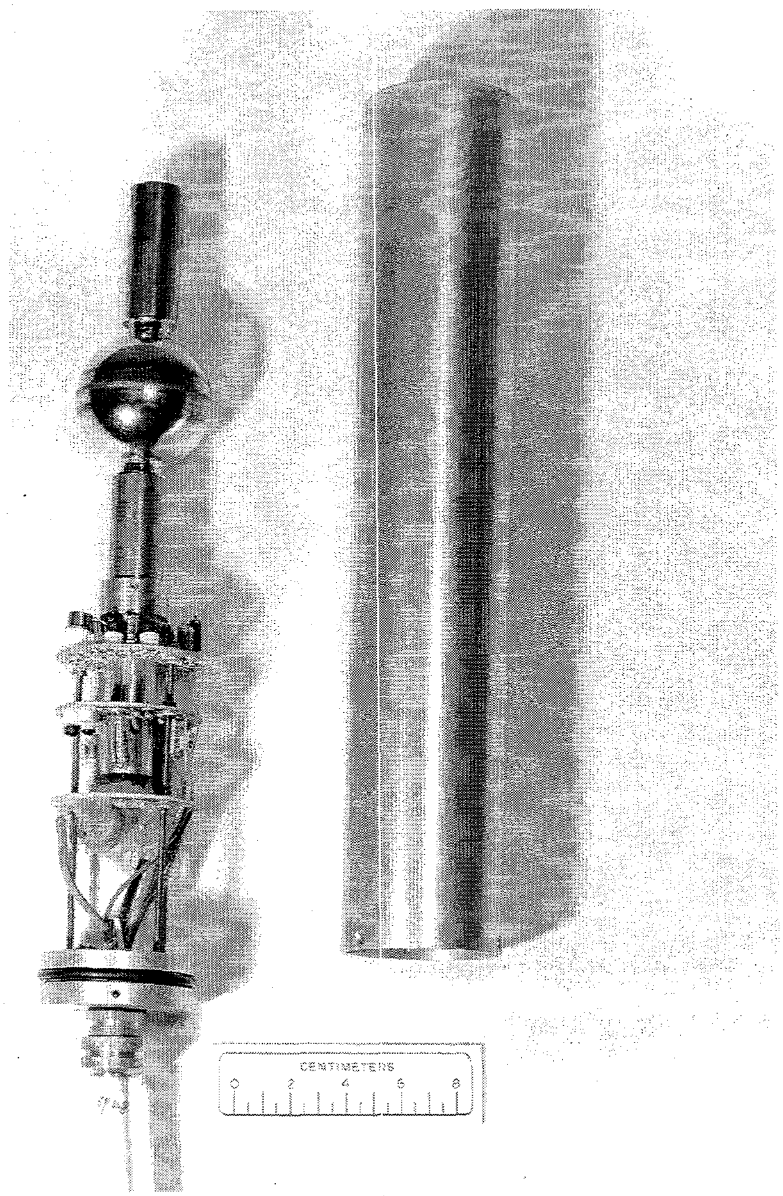

Fig. 7.8 Hydrogen-filled Proton Recoil Bejamin-type Detector.

neutron spectra in the region from about 5 keV to 150 keV using gamma-ray discrimination. To do this, pulses from the detector were fed into the computer system where they were accumulated as a function of specific ionization (rate of rise divided by ionization) versus ionization. To separate the neutron and gamma-ray contributions in this lower energy region, it was necessary to make individual hand plots for each energy bin of data accumulated so that the gamma-ray portion of the plot could be extrapolated as seen in Fig. 7.9, and then subtracted from the neutron portion of the plot before submitting this difference, the neutron portion, for conversion to spectra. Though the technique was successful, it was time consuming, and as the analytical skills improved, the use of the neutron data obtained in this energy region lost favor and use of the detector was eliminated.

Pulse-height data obtained with the hydrogen counters was converted to spectra through the use of the SPEC-4 unscrambling code originally developed by P. W. Benjamin, C. D. Kernshall, and A. Brickstock[10] from the United Kingdom's Atomic Weapons Research Establishment. This approach was modified for TSF use by R. M. Freestone Jr.,[11] who added special editing and some graphical output routines. His work was later improved on by J. O. Johnson and D. T. Ingersoll[12] and published in a form that detailed the user input requirements.

7.11 BONNER BALLS

The concept of the Bonner ball detector to be used as a neutron detector at the TSF was based on the work of T. W. Bonner[13] and his associates and T. V. Blosser[14] of the then Neutron Physics Division. Bonner used a Li^6I (Eu) scintillation crystal surrounded by spheres of polyethylene and cadmium, to measure thermal flux generated from fast flux slowing down in the polyethylene. This principle was maintained by Blosser in the development of a low-energy (thermal to approximately 5 keV) neutron spectrometer, but the detection device was switched from Li^6I (Eu) to a spherical copper shell filled with BF_3 gas where the boron was enriched to 96% ^{10}B. The unique feature in this design was the replacement of the usual center wire within the sphere by a double loop of a 2-mil-thick tungsten wire, with the two loops at right angles to each other that served as the ion collector. This double loop was supported by a single, much thicker wire to the high voltage source and the preamp. This design and construction was the handiwork of R. K. Abele and W. T. Clay of the Instrumentation and Controls Division.

Development of the Bonner ball detector system at the TSF began in 1970 using Blosser's approach. The detector size was reduced from 3 to 2 in. and it was surrounded by a series of polyethylene spheres whose diameters ranged from 3 in. to 12 in., each enclosed with form-fitting cadmium whose thicknesses were considered black to incident thermal neutrons (Fig. 7.10).

The Bonner balls were calibrated[15] against Sb-Be and ^{252}Cf sources of known strength. The result of the calibrations yielded the overall counting efficiency of each ball, which when multiplied by the calculated response functions for each ball using ANISN, yielded the measured counting rate in counts incident neutron/cm^2. Experimentally measured count rates were predicted analytically by folding a calculated neutron spectrum with the Bonner ball response functions. The initial measurements using this detector were started in 1971.

With the advance in the development of calculational methods in the 1960s, along with improvements in the state of the art for computers, the type of useful measurements obtained at the TSF changed from those of dose rates to that of flux and spectra. Thus, with the development of the NE-213, hydrogen-filled proton-recoil detectors, and Bonner balls the measurements with the neutron dosimeters in shielding studies at the TSF became essentially passe.

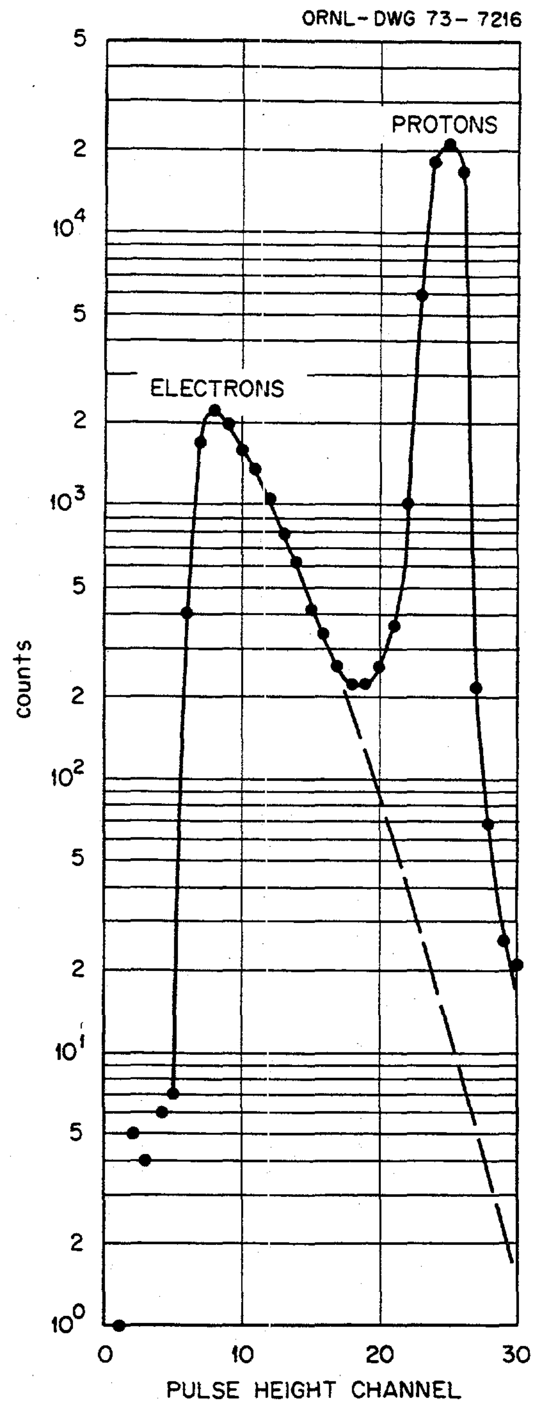

Fig. 7.9 Typical Specific Ionization Curve.

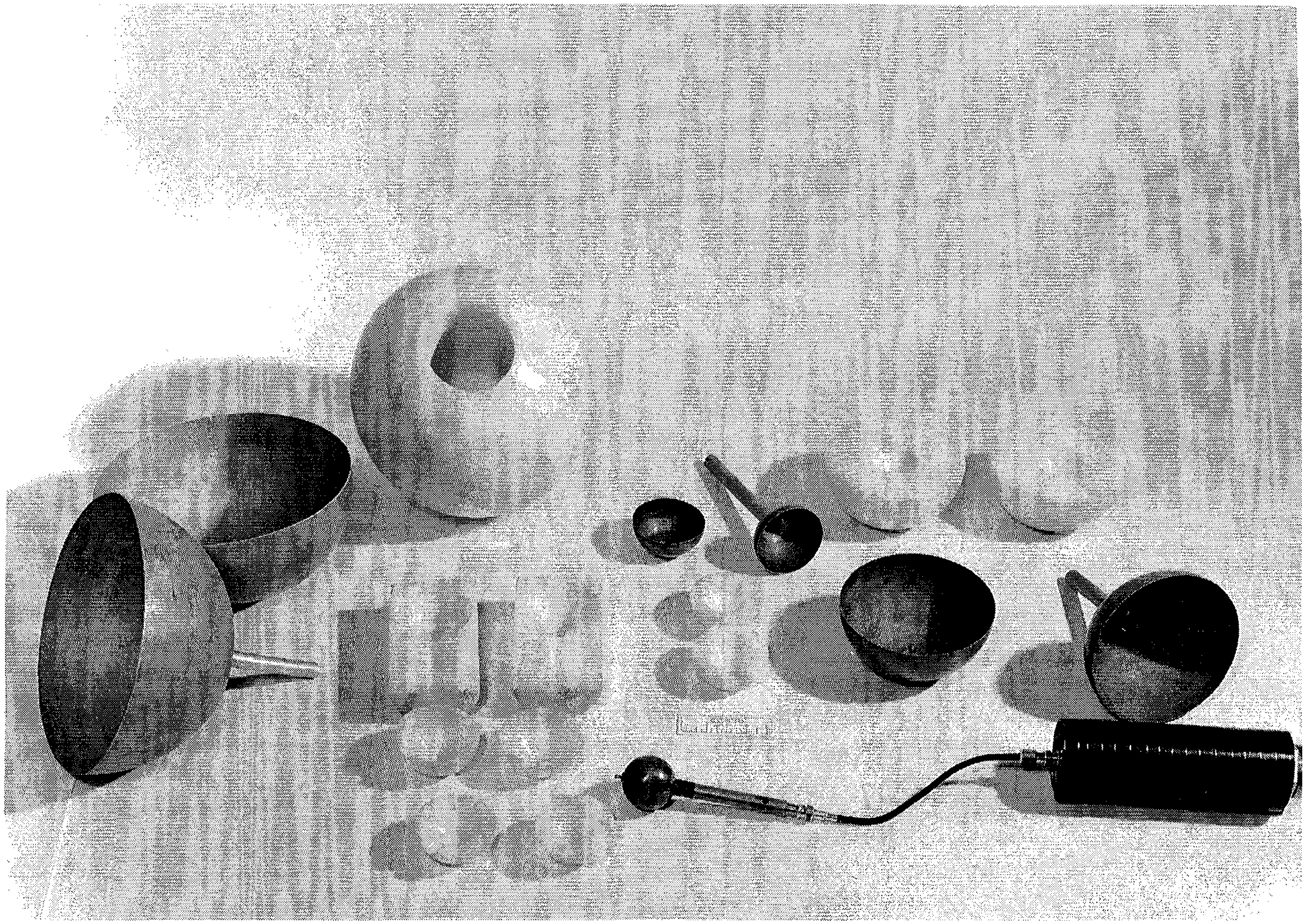

Fig. 7.10 Thermal Neutron Sensitive BF_3 filled Bonner Ball Dectector System.

7.12 THERMOLUMINESCENT DETECTORS

Advances in the capability of calculating gamma-ray spectra and the need for determination of the gamma-ray energy deposited within the reactor shields changed the experimental approach from making ionization chamber measurements behind the mockups to measurements of the *in-situ* type within the shields. This required the use of small detectors and it was suggested that thermoluminescent material would fill this need. Thus, in the early 1970s, an investigation into the usefulness and validity of the thermoluminescent detectors (TLDs) in shielding measurements was initiated at the TSF. Measurements were first made using TLDs in powder form, exposing it to known gamma-ray sources. The readouts were inconsistent, certainly not good enough to provide the desired accuracy against which the calculations could be compared. The powder was soon replaced by a solid chip as the development of TLDs improved (Fig. 7.11). The chips chosen were CaF_2 and ^7LiF because they provided good gamma-ray sensitivity, broad, linear dose ranges, and low neutron sensitivity. The chips were 0.125 in. sq. and about 35 mils thick, and when exposed, were placed in small iron capsules whose material was similar to the material in the reactor shields. These capsules are shown in Fig. 7.11.

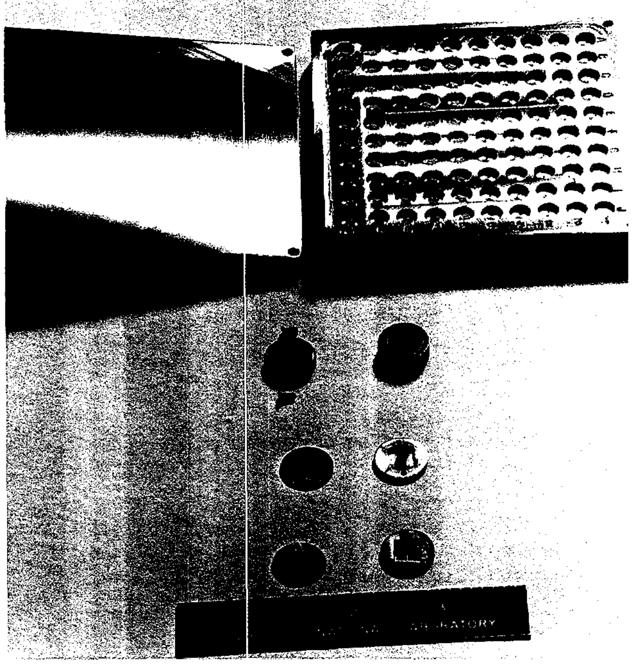

Fig. 7.11 ORNL-CaF_2 TLDs, Storage Bins, and Stainless Steel Sleeves.

Development of the TLD as a useful tool in measuring the gamma-ray energy deposited within a shield became the task of Woo Yong Yoon, a graduate student from the University of Tennessee, who undertook this challenge for his doctoral thesis. A technique was established and demonstrated that not only produced results which repeated within ±3%, but they also agreed, within those limits, with measurements from an iron equivalent high-pressure ion chamber designed and fabricated just for this work.

7.13 IRON-EQUIVALENT IONIZATION CHAMBER

Figure 7.12 shows a sketch of an ion chamber developed by R. K. Abele's group in the Oak Ridge Instrumentation and Controls Division to make direct measurements (*in-situ* type) of the gamma-ray energy deposited in specified reactor shields. The chamber was approximately 1 in. in diam and 7 in. long, having an active volume of about 12 cm^3 between two stainless steel cylinders. The chamber was filled to approximately 1500 psi of 33.4% by weight krypton and 66.6% by weight argon. This mixture and pressure were chosen to give an energy absorption coefficient that closely matched the energy absorption coefficient for iron. The high pressure was chosen to give needed sensitivity. To make a measurement, the chamber was placed in a cavity within a 5.08-cm-thick stainless steel (SS) slab so that induced voids between slabs would be kept to a minimum and the closeness of the SS around the chamber would be similar to that for the TLDs.

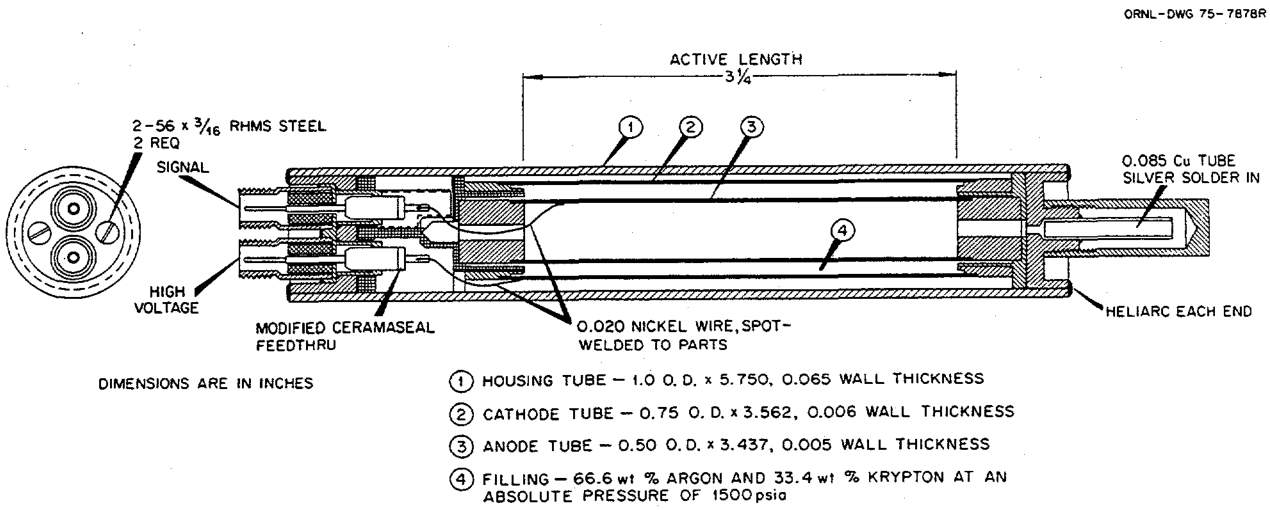

Fig. 7.12 High-Pressure Iron-Equivalent Ionization Chamber.

7.14 DETECTOR PLACEMENT SYSTEMS

Much of the data taken at the TSF was in the form of detector traverses where continuous information was plotted or where many individual points were recorded without shutting down the reactor. To do this, a detector was mounted on a traversing mechanism whose traversing position was identified by reading the output of a calibrated selsyn drive system. Positioning of detectors within ±1 mm was desired, but this was not always possible due to the backlash in the traverser-selsyn relationship. With the development of television cameras whose vidicon could be equipped with a cross hair, the selectivity and reliability in positioning was much improved. The target for the camera was a meter stick mounted on the traversing mechanism that moved with the detector. The detector location was then noted using a stationary camera equipped with the cross hair. An example of this technique is shown in Fig. 7.13. This method could indicate the location of the detector to within one-half millimeter or less.

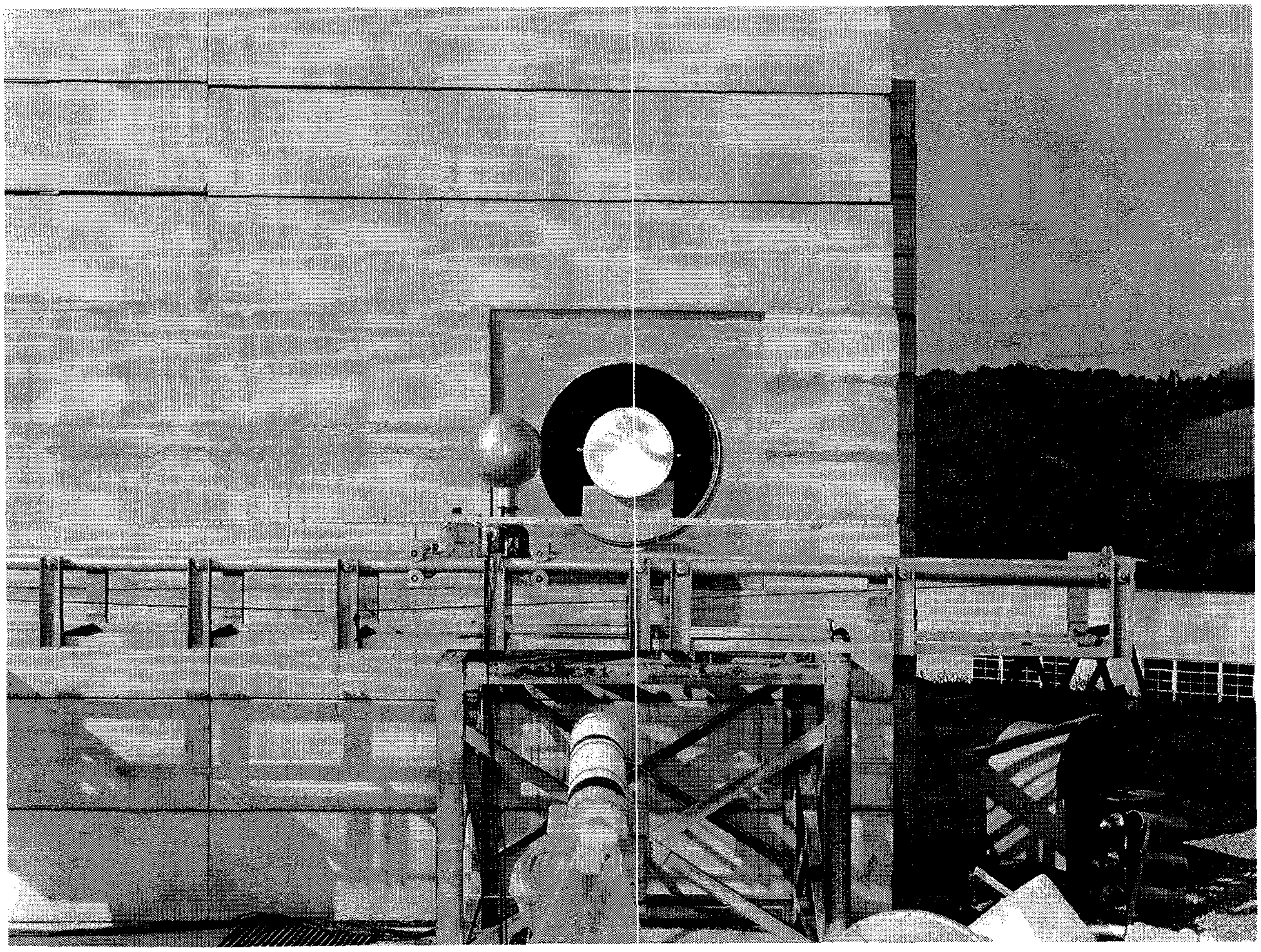

Fig. 7.13 Camera-Meter Stick Traverser Combination for Remote Positioning of Detectors.

7.15 REFERENCES

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8.0 EXPERIMENTAL PROGRAMS

8.1 Overview of Experimental Programs

The purpose of the TSF was to enhance the capability of the U.S. to successfully complete the design and fabrication of a nuclear powered airplane using a divided shield concept. The data obtained from the various experiments were intended to provide a verification of the calculational methods developed by personnel from the many vendors involved in the overall ANP program.

The program began with a reactor source whose design and behavior was well known from the use of a similar reactor in the Bulk Shielding Facility. With it the program was very successful, demonstrating its versatility by not only providing a useful source for experiments while contained in its water tank, but also providing versatility by its capability to be inserted into specially designed reactor shields from which data could be obtained to aid in the study of advanced shield design. In conjunction with this versatility it was possible to obtain the data needed to generate the initial analytical programs used to calculate the radiation penetrating the various crew compartment designs.

As these analytical programs became more and more refined, it was realized that it would be very difficult to apply the data obtained using TSR-I as a source to design studies for the aircraft where the source would be a circulating-fuel, reflector-moderated reactor. With this problem in mind the shielding analysis section at ORNL, in collaboration with the outside vendors, proposed a reactor concept that developed into TSR-II.

The usefulness of the newly accepted TSR-II to the ANP program was short lived, as only one experiment was completed before the entire ANP program nationwide was canceled effective June 30, 1961. That experiment was a divided shield mockup study perceived by P&W in which the TSR-II was surrounded by an optimized, highly asymmetric, uranium-lithium hydride shield of their design, with the detector located in a compartmentalized crew shield.[1]

That experiment, however, was actually the second experiment planned to be run using the TSR-II. The first one was to have been a "beam differential" type experiment designed to obtain information of a basic nature. Both the reactor and detector would have been positioned inside separate, but almost identical, spherical shields with well defined collimators. However, fabrication of the detector shield was not completed in time, and rather than delay the use of the reactor, the P&W experiment prevailed. With the cancellation of the ANP program on June 30, 1961, that beam differential experiment no longer had any priority and was canceled.

The loss of the ANP support was not "deadly" to the TSF, as other programs had already become aware of its usefulness . The TSR-II was used in January 1961 to start studies comparing the attenuation of radiation by ordinary armored vehicles supplied by the British to the attenuation obtained from their proposed designs, and to test specific slab mockups for verification of theoretical studies toward the optimization of the design. This program resulted from an agreement between the United Kingdom and the U.S. to measure the attenuation characteristics of specific military equipment and research models of mutual interest. The U.S. vehicles were included through an agreement between the USAEC and the OTAC from Detroit Arsenal. The experiment included two British mockups of fighting vehicles, a U.S. M-48 armored tank, and a radiologically protected pod designed by Convair/Fort Worth in cooperation with OTAC.[2] Participants were from the U.S. ATAC, General Dynamics in Forth Worth, and ORNL. The work was interrupted, however, after the first month, but was completed later in the year.

Included in the program was an extensive "free field" mapping of the radiation from the TSR-II enclosed in two "spectrum modifier" type shields called Cool-I and Cool-II. Measurements were made as a function of reactor height, vehicle location, separation distance, and detector height above the ground.

Early in 1959, a nuclear radiation panel, appointed by the STL, had made an evaluation of the radiation shielding effectiveness of the covers being designed for missile silos then under construction. Their study indicated the need for further investigation to be sure that not only the personnel and equipment in the silos were adequately protected but that the amount of concrete in the covers should be minimized to reduce costs and increase ease of operation.

The TSF was proposed as the likely place because of area terrain and the reactor source that could be adapted to simulate that of a weapon. The fact that the source would be steady state, not burst, was a drawback; however, a steady state source would provide more reliable data. It was also recognized that the data would provide a basis for generating and verifying calculational methods devised for computing the neutron and gamma-ray dose rates inside the silos even under burst conditions. Thus, a specific series of measurements were made for the military and office of Civil Defense in 1961-62, the first of many experiments to be performed for the DASA.[3]

Other programs soon followed. Work was initiated in direct support of the SNAP program, where reactors were to be used for power in space.[4] In reality, the work was to be a continuation of the ANP work with the emphasis now shifted toward high-energy neutrons where angular and energy distributions were of prime importance. During this time a short program was conducted for the US-NRDL who were interested in radiation penetration through steel structures exposed to a weapons source.

In 1965,[5] it was proposed that the TSF install a SNAP reactor the following year to provide experimental data for comparison with Monte Carlo calculations of the radiation intensities transmitted through shadow shields. The installed reactor was a modified SNAP-10A designed and fabricated by Atomics International, a Division of North American Aviation, Inc. The SNAP reactor went critical at the TSF on April 7, 1967, with a loading of 36 SNAPTRAN V fuel elements and one SS rod containing a small portion of B_4C.[6]

In 1969, the TSF became involved for the first time in work for the LMFBR[8] with support for development of the Benjamin-type neutron spectrometer that measured spectra between about 10 keV and 1.5 MeV. A brief experiment was performed by placing samples of sodium, iron, and iron rods in sodium, all of which were surrounded by oil, beneath the SNAP reactor while measuring the transmitted neutron spectra. The following year about 30% of the work load at the TSF was for studies of neutron spectra transmitted through thick iron shields in support of evaluation of iron cross section point data sets. This effort expanded to a significant portion of the TSF schedule with the work for the analytical grid plate shield studies and the neutron streaming through the iron shield covering the top of the reactor tank at the FFTF.

The majority of the measurements, however, in this period were concerned with the SNAP shielding program, beginning with dose profiles beneath the SNAP reactor to check the accuracy of predicted spatial distributions of the neutron environment. This was followed by spectral measurements through typical shield materials considered for the SNAP program. For this work the reactor was placed over the maintenance pool, that had been drained, and the detector-collimator placed at the bottom of the pool. Shield samples such as lead, ^{238}U, tungsten powder, hevimet, lithium hydride, and laminated slabs of these materials were placed beneath the reactor between it and the detectors. The DASA program during this period was limited to the measurement of gamma-ray spectra from shield samples using the TSR-II as a source.

By 1971, the TSF effort in the SNAP program had been greatly reduced, being limited to a few measurements of gamma-ray production in shields with combinations of heavy metal such as iron, tungsten, and ^{238}U combined with lithium hydride (LiH). The majority of the Tower time was concentrated on mockups of shields connected with the design of the FFTF.[8] By 1974 the FFTF work was completed, and the effort in the LMFBR program was switched to work for the CRBR program.

It was also during this time that initial proposals were made for the ORNL shielding group to become involved in shielding studies on the demonstration model of the GCFR system proposed by the General Atomic Corporation. The group was asked to look at the program, define problem areas with respect to calculations, and prescribe experiments needed to verify the calculational methods. By early 1975 the shielding group had developed a program that was to include eight specific experiments at the TSF. Six of those experiments had been completed before the GCFR program was canceled in 1980.

The experimental capabilities at the TSF were greatly improved by addition of a new reactor shield in 1975. The collimator was much larger in diameter and much shorter in length than the previous shield. Placement of the reactor in its new shield increased the source strength incident on the experimental mockup by a factor of 200. The change also altered the description of the source term from a previously considered point source to a disk source whose energy and angular distribution varies as a function of location on the disk.

The first measurements with the new collimator were to fully characterize the new TSR-II source term including a determination of the neutron beam profile, its magnitude, and energy spectrum. This was followed by measurements for the upper axial shield experiment for the CRBR. The new source intensity was severely tested with measurements behind a mockup of 18 inches stainless steel, 15 ft sodium, and 35 in. iron. This program was then interrupted for two experiments of a preliminary nature for the GCFR program, where measurements were made of the streaming through a single-cell mockup and through a lattice of GCFR-type fuel pins,[9] and for several months of TLD development work to bring on line a detector much needed for future CRBR experiments.

A busy year of CRBR work followed, with measurements for the determination of the reliability of the LLFM in a high gamma field, a study of the radial blanket mockups, a measurement of the neutron streaming in a CRBR prototypic coolant pipe chaseway, and ending with comparison measurements of the neutron attenuation through SS and inconel as a radial blanket shield.

By late 1977, the GCFR program was reinstated with measurements for the Grid Plate Shield Design Confirmation Experiment, followed by the measurements for the Radial Blanket configuration, the Shield Heterogeneity mockup, and the Exit Shield Experiments in the Plenum Shield Design. These experiments ended the GCFR work as the program was terminated in the latter part of 1980.

During the years 1981-82, there was very little experimental activity at the TSF. However, during the last half of 1981 and first half of 1982, the Bonner ball system was taken to the FFTF at Hanford Engineering Development Laboratory near Richland, Washington, to measure the neutron fluxes above the reactor vessel head.[10] The purpose was two fold: (1) to test the effectiveness of the shield protecting the top head compartment; and (2) to provide data that would show how accurately the neutron fluxes had been predicted by ORNL calculations.

In the fall of 1982, the same detector system was taken to Los Alamos to participate in measurement of the leakage fluxes from a critical mockup of the Hiroshima bomb Little Boy.[11] The measurements were part of a larger program to provide benchmark data for testing the methods used to calculate the radiation released from the bomb.

Experiments were resumed at the TSF in early 1983 with participation in two HTGR shielding studies. Phase I, completed in the first half of 1983, covered measurements for the HTGR Bottom Reflector and Core Support Neutron Streaming Experiment.[12] In this phase the first four of eight segments that comprise the full experimental mockup were used. Phase II of the program followed, with the last four segments added to the first four. Again the measurements provided data against which the calculational methods used to predict the radiation transported through the bottom reflector and core support structure of General Atomics current HTGR design were validated.

In the latter half of 1984, a concrete structure simulating a small, single-story concrete block house exposed to the radiation from the atom bomb explosions over Hiroshima and Nagasaki[13] was modeled at the TSF. Measurements were made for successive variations in the house structure such as introducing windows, adding an inner wall, and moving a support pillar. The data were to provide validation of the discrete ordinates TORT being developed at ORNL.

The latter half of 1985 was spent making measurements for the Alternate Shielding Materials Experiment[14] to provide a benchmark study of neutrons penetrating proposed boron carbide and SS shield configurations for advanced liquid-metal-cooled reactor systems. With the advancements in LSPB and LMR concepts, the shield designers were interested in developing lighter material shields to reduce costs and aid in seismic response.

In March 1986, the TSF began participation in a cooperative experimental program between the Japanese PNC and the United States Department of Energy.[15] The program, entitled "JASPER," for Japanese-American Shielding Program of Experimental Research, was designed to meet the needs of both participants. Eight experiments were planned, but only the first two, the Radial Shield Attenuation and the Fission Gas Plenum studies, were completed before operation of the TSR-II was halted by the Department of Energy's Oak Ridge Operations at the end of March 1987 to undergo a thorough inspection of the reactor's safety, both physical and operational. One year later, March 26, 1988, DOE ordered the TSF staff to be closed permanently.

After considerable negotiation and effort, the shutdown order was rescinded and the TSF was prepared for restart. On December 7, 1989, the facility was once again given permission to operate the reactor and complete the JASPER program. On December 11, 1989, the reactor went to "token" power at 1420 hours to make a spectral measurement of the SM mockup. From December 1989, through March 1990, preliminary-type measurements were made with and without the SM in the beam before the reactor was shut down again for major repairs. It was not until July 27, 1990, that reactor operation was started in a productive manner. The JASPER experimental program resumed with measurements of several Axial Shield mockups and continued, with only mild interruptions, until the JASPER program was completed in September 1992. On October 1 of that year, the reactor was again officially shut down by DOE-Headquarters, placed in standby, with orders to prepare for dismantling the reactor core by 1995.

8.2 REFERENCES

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9.0 AIRCRAFT NUCLEAR PROPULSION EXPERIMENTS

9.1 INTRODUCTION

The Aircraft Nuclear Propulsion (ANP) shielding program was initiated with measurements of the ground-scattered component as a function of reactor and detector heights above the ground. These measurements were followed by studies of various configurations that involved differential-type measurements as well as integral mockups. Several reactor shields and crew compartment shields were designed and fabricated by different vendors so that detailed studies could be made of the angular distribution of radiation both leaving the reactor source and entering the crew shield. Such studies were used to verify calculational methods that were being developed concurrently with the experiments and to identify future needs in shield designs. The descriptions of the experiment are quite general and no results are given.

9.2 BACKGROUND RADIATION MEASUREMENTS

Since the facility was built to provide a means for performing experiments with a divided shield free from an excessive background of ground-scattered radiation, it was fitting and proper that the first shield-type experiment performed should demonstrate that the facility did indeed provide that capability.[1] Therefore, measurements of the fast neutron flux as a function of reactor-detector altitude were necessary to determine the scattered radiation component at the maximum height (200 ft). The geometry for the experiment is shown in Fig. 9.1. The TSR-I was placed in the reactor tank so that its vertical center was located 45 cm from the wall of the tank at an angle of 30° with respect to the reactor-detector shield axis. Measurements of the thermal neutron distribution were made in the detector tank, which was essentially a 5-ft cube of water separated from the reactor tank by a distance of 64 ft. Traverses were made with a BF_3 detector along one half the horizontal axis perpendicular to the reactor tank/detector tank axis while varying the altitude of both tanks simultaneously in discrete steps from ground level to 195 ft above ground. Results from this series of measurements indicated that the ground-scattering component was about 2% of the total scattered neutrons at both the 150- and 195-ft elevations.

In addition to the air- and ground-scattering experiments, there were some experimental investigations into multiple scattering of neutrons in air and from the ground.[2] A technique similar to that described above was used for those measurements, the difference being the location of the reactor and detector within their respective tanks. Results indicated that the intensity of the multiple-scattered neutrons was higher than had been expected and indicated a greater importance than had been previously considered in divided shield designs.

9.3 GE-ANP-R1 SHIELD

During September 1954, the first experiments using a specially designed reactor shield tank were initiated.[3] This design, known as the General Electric (GE) ANPR-1 shield, see Figs. 9.2 and 9.3, contained voids and individual components that permitted some control over the measurements that provided information toward design of the ultimate shield. The first measurements were made with

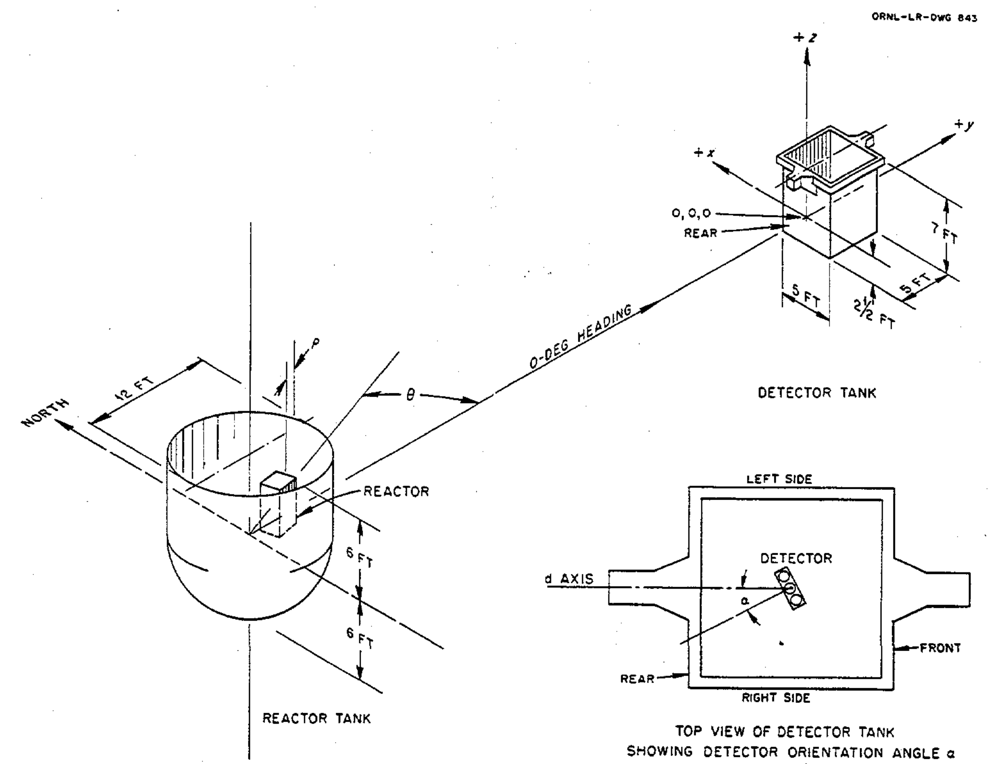

Fig. 9.1 Coordinate System for Reactor and Detector Positions within their Respective Tanks - Tower Shielding Facility.

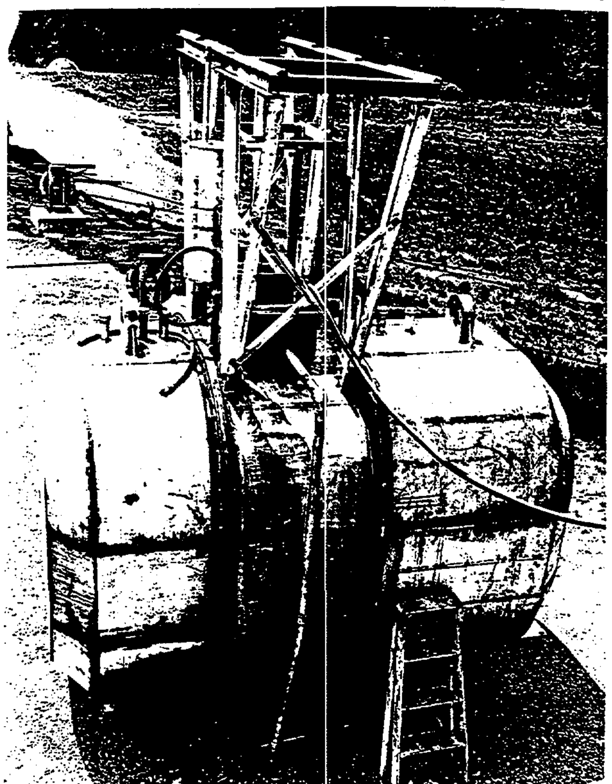

Fig. 9.2 GE-ANP R-1 Reactor Shield.

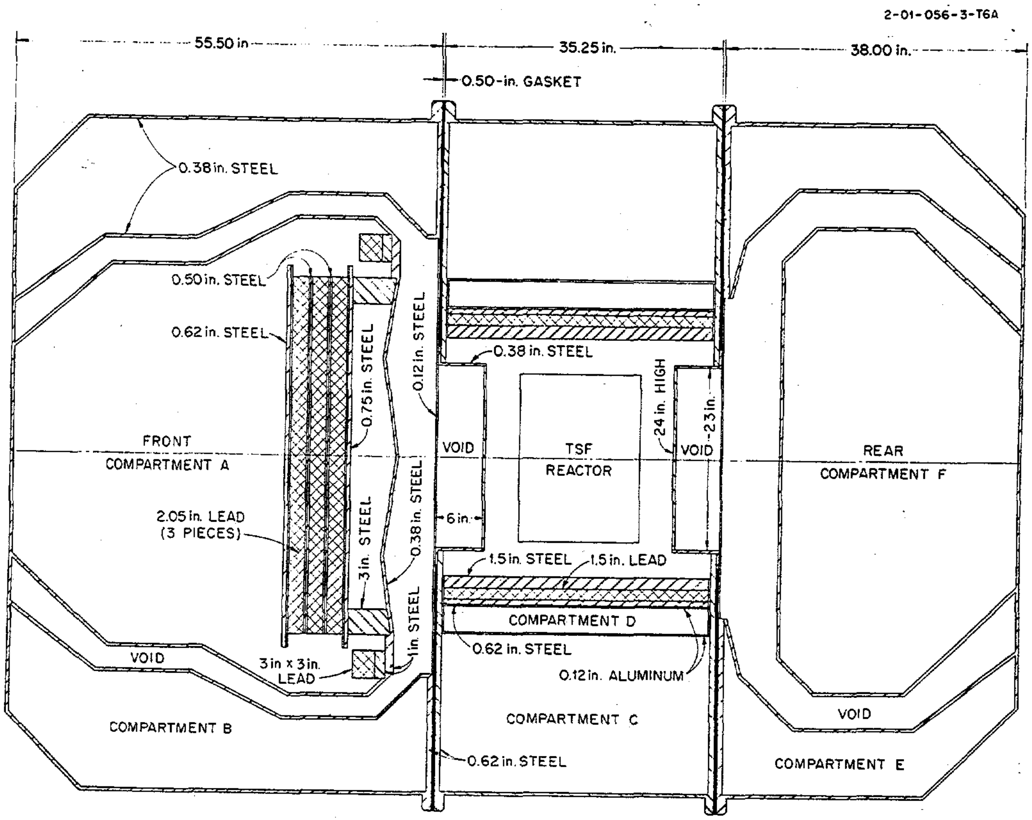

Fig. 9.3 Vertical Section of the Mockup of the GE-ANP R-1 Reactor Shield.

the GE ANPR-1 shield located in the TSF reactor handling pool using a TSR fuel element array of 5 x 7 that provided a symmetrical power distribution throughout the reactor. This combination of reactor and shield had been tested previously in the pool at the Bulk Shielding Facility and similar measurements were being repeated at the TSF so that comparisons could be made as a check on the reliability of the data. In general, agreement was good, but in specifics there were some differences attributed to the variations in the actual mockups. The reactor in its ANP-R1 shield was then combined with the 5-ft-sq detector tank used in the previous experiment to once again measure the neutron contribution from ground scattering at discrete elevations up to 200 ft. Measurements indicated that in going from 150 ft to 200 ft, there was still about a 6% change in count rate. Analysis of the experimental data indicated that at 200 ft there would be about a 5% contribution from ground-scattered neutrons for this particular reactor shield design. A series of gamma-ray measurements were also included for this reactor shield.

9.3.1 Detector Tank

The first group of experiments using the GE reactor shield and detector tank combination in air were completed in March 1955, with the measurement of the gamma-ray dose rates along the axes of the detector tank.[4] The arrangement between reactor shield and detector tank was the same as for the earlier neutron measurements. A 64-ft separation, except that 5 in. of lead was installed 1 ft from the rear face (reactor side) of the detector tank to simulate the gamma-ray shielding included in a crew compartment design. Gamma-ray dose rates were measured with both plain and borated water in the detector tank to obtain information on the dose rates from gamma rays incident on the soon-to-be-studied GE-designed crew compartment and of the attenuation characteristics of the hydrogen material within the crew compartment. The gamma-ray measurements in the detector tank provided information for determination of gamma-ray relaxation lengths for both plain and borated water.

9.4 REACTOR TANK - GE-ANP CREW SHIELD

During the second quarter of 1955, the TSF program was concerned more with performance of differential-type measurements than those for specific mockups.[5] Measurements were made of the neutron dose rates in the detector tank and in the GE-ANP crew shield as a function of angle of radiation emission from the reactor tank and its water thickness (Fig. 9.1). The GE-ANP crew compartment consisted of a cylindrical void surrounded by compartments of water (sometimes borated) and partially lined with lead (Fig. 9.4). These studies were intended to provide data for development of calculational methods useful in predicting radiation dose levels in other shield designs.

9.5 GE-ANP-R1 - GE-ANP CREW SHIELD

The reactor was then placed inside the GE-ANP-R1 shield and the differential-type measurements continued using the GE-designed crew compartment as the detector shield. These experiments came to a close in the last months of 1955 with (1) measurements of the fast neutron dose rates in the crew compartment, (2) measurements of the fast neutron and gamma-ray dose rates in air around the reactor shield with and without the shadow shield, and (3) measurements of the thermal flux outside the reactor shield. It was during these measurements that a 5-in.-diam Hornyak button fabricated at the Laboratory was introduced as a fast neutron dosimeter. Its sensitivity proved to be about three times that of the gas-filled dosimeter, a welcomed increase because of the low count rates. Development of the detector as a dosimeter was being pursued by TSF personnel using a Van Der Graaff generator as a neutron source. It was also during these runs that an anthracene crystal was used for the first time as a gamma-ray dosimeter.

Also at this time, the GE shielding group became interested in an aircraft design where the reactor and engines would be supported in the wings of the aircraft, off line from the axis of the fuselage.[6] Additional experiments were requested to obtain some indication of the changes in the neutron and gamma-ray radiation reaching the crew compartment as a function of the lateral displacement of the reactor in its GE-ANP-R1 shield. Measurements were made inside the crew shield for a lateral reactor displacement up to 60 ft.

9.6 GAMMA-RAY SPECTRAL MEASUREMENTS

Previous measurements had indicated a significant contribution to the gamma rays measured inside the crew compartment from a source other than those generated in the reactor or its shield. Preliminary-type gamma-ray dose-rate measurements were initiated in early 1956[7] to study what was believed to be a gamma-ray source due to neutron capture in air. The beginning measurements were made in air as a function of distance from the reactor positioned inside the reactor tank located

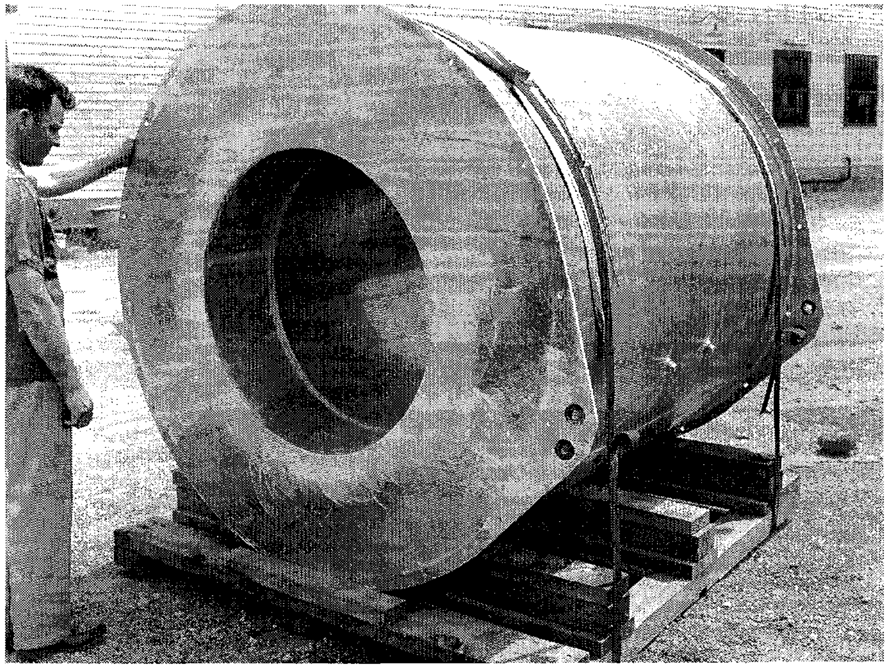

Fig. 9.4 Crew-Compartment Section (Ends Removed) of the GE-ANP R-1 Divided-Shield Mockup.

195 ft above the ground. This was followed by a series of differential-type measurements in which gamma-ray dose rates incident on and inside the crew compartment were studied as a function of water thickness and angle of radiation emission from the reactor shield.

This work was continued during the second quarter of 1956, with measurements of the gamma-ray dose rate inside the crew compartment as a function of the crew compartment shield composition and thickness. As the experiments progressed it became more and more apparent that for analytical methods to properly interpret the many dose rate measurements of the gamma rays leaving the reactor shield it was necessary to have knowledge of the gamma-ray spectra. Since such spectra could not be obtained by simple calculational efforts using dose-rate data, it was necessary that the information be gathered through direct measurements. To accommodate this need, a 3-in. right cylinder NaI crystal was used to obtain a measure of the gamma-ray energies incident using a portable, three-channel analyzer.[8] Window widths were on the order of 1 MeV over an energy span from 0 to 12 MeV, and reduced to 0.2 MeV for a look at the lower energy gamma region from 0 to 2.4 MeV. The data were converted to a spectrum by G. J. Rauss of the Glenn L. Martin Company using a series of successive subtractions of histograms from discrete energy gammas with energy width equal to that used in the measurements, starting with the highest energy gamma ray. Corrections for crystal efficiency and peak-to-total spectrum count rates provided the resulting energy spectrum. A typical experimental arrangement is shown in Fig. 9.5. Spectra were obtained of the decay gammas from the reactor, the gamma rays from fission and hydrogen capture within the reactor tank, and air capture gamma rays

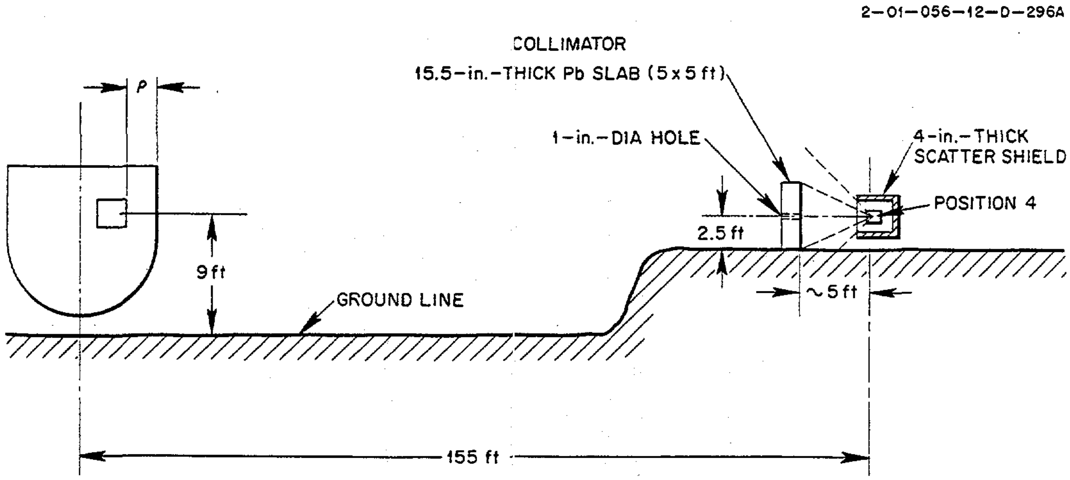

Fig. 9.5 Geometry for TSF Spectral Measurements with Collimator.

using a tightly collimated sodium iodide crystal located at several distances from the reactor. Even though the technique for making these measurements and unscrambling the data was certainly less than refined, the results were very indicative of the need to have a better understanding of the air capture spectra to be successful in calculating the gamma-ray dose measured inside the crew shield.

The preliminary differential-type measurements the spectral measurements that had occupied the early months of 1956 were concluded by late summer and replaced with an experiment using a better defined reactor source. The results of previous measurements, when compared with the calculations that were made concurrent with the experiment, indicated the need for more definition in the experiments and resulting data to understand the magnitude of the errors in the calculational methods for air and ground scattering contributions. To provide this information, the direction of the reactor beam needed to be defined and the location of the radiation entering the crew shield had to be more selective. The reactor source was collimated through the use of two different diameter air-filled cylinders, 8- and 15-in., that were attached to the reactor and could be rotated in both the horizontal and vertical planes as shown in Fig. 9.6. The original GE crew compartment used in the previous measurements was replaced by a compartmentalized one, shown in Fig. 9.7, that was designed and built by Boeing Airplane Company according to ORNL specifications. Measurements were made in the air outside the crew shield and then inside the shield for selected shield voids as a function of source collimator, its orientation with respect to the reactor, and height of both source and detector above ground level.[9]

9.7 VERIFICATION OF CALCULATIONAL MEASUREMENTS

A calculational procedure for the optimization of a divided neutron shield was previously developed at the TSF by M. P. Valerino and F. L. Keller[10] using a reactor shield that was divided into

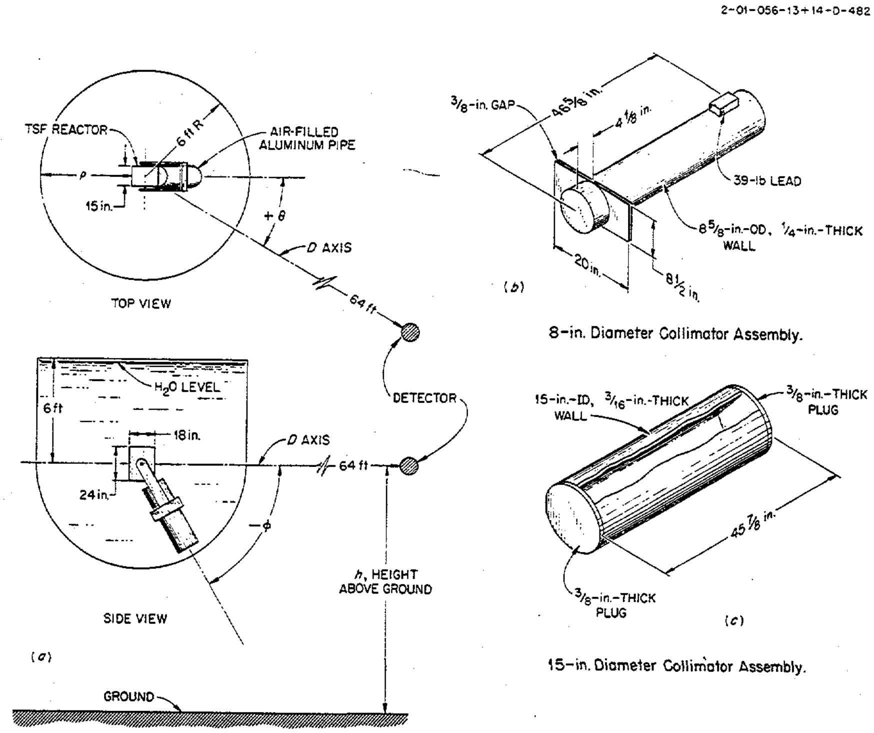

Fig. 9.6 Collimator Geometry in the TSF Reactor Tank.

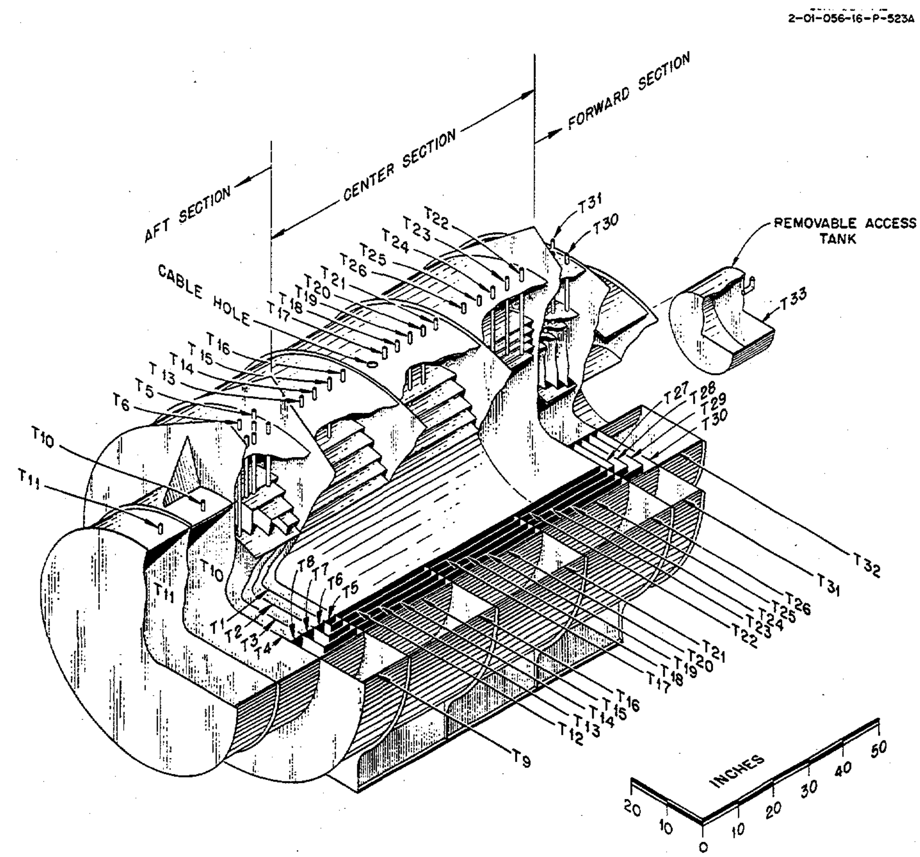

Fig. 9.7 Compartmentalized Crew-Compartment Tank.

conical sections whose vertexes converged at the center of the reactor. In order to confirm this calculational procedure, a compartmentalized reactor shield tank (Fig. 5.3 in Chapter 5), designed by ORNL and fabricated by Convair/San Diego, was combined with the previously mentioned compartmentalized crew shield (Fig. 9.7) for a series of experimental tests. Measurements were made inside the crew compartment as a function of void locations (other sections were filled with water) in both the reactor shield and the compartmentalized crew shield. These measurements were completed by the end of 1956.

9.8 J-57 ENGINE EXPOSURE

The first few months of 1957 were spent preparing for and exposing a jet engine to the TSR source.[11, 12] The exposure provided the means for studying the effects of radiation on control and performance of the engine, determining what radiation damage might occur to the engine components, and measuring what induced activity would result from the exposure. For the test, a Pratt and Whitney Aircraft J-57 model jet engine was operated for 50 hours while exposed to a total tissue dose of 7×10^8 ergs/gm from the TSF reactor (Fig. 9.8). Gamma-ray spectra and dose rates were obtained immediately after exposure and for several days following exposure in an attempt to identify the elements that were activated. The scheme used to make the spectral measurements is shown in Fig. 9.9. Several weeks later the exposure was repeated with only one change in procedure. During the first exposure the oil in the engine was replaced every eight hours, while in the second run there was no oil change. Gamma-ray decay studies were repeated after this second exposure but not in as much detail as after the first run. There were no deleterious effects observed in the performance of the engine during its exposure.

9.9 COLLIMATED NEUTRON MEASUREMENTS

During the next three-month period, measurements of air-scattered fast neutrons that penetrated the sides of the compartmentalized crew shield that had been delayed for the J-57 exposure were resumed (Chapter 9.7). To aid in the calculations of the air-scattered component, it was necessary to generate a profile of the neutron flux leaving the reactor tank as a function of reactor rotation and thickness of water between reactor and side of reactor tank. This was a simple experiment[13] in which the flux leaving the reactor tank and reaching the detector was well collimated so that only the reactor tank was seen by the detector. To measure the air-scattered component, the reactor and crew compartment were raised to 190 ft and measurements made in the crew compartment as a function of water or oil thickness in selected places of the crew compartment.

9.10 AIR CAPTURE GAMMA RAYS

Near the end of 1957, a new approach was made to resolve some of the discrepancies encountered in understanding the results from the many gamma-ray measurements made in the past inside the crew compartment shields. Dose rates had been obtained as a function of reactor orientation and crew compartment mockups but such measurements provided little information as to the origin

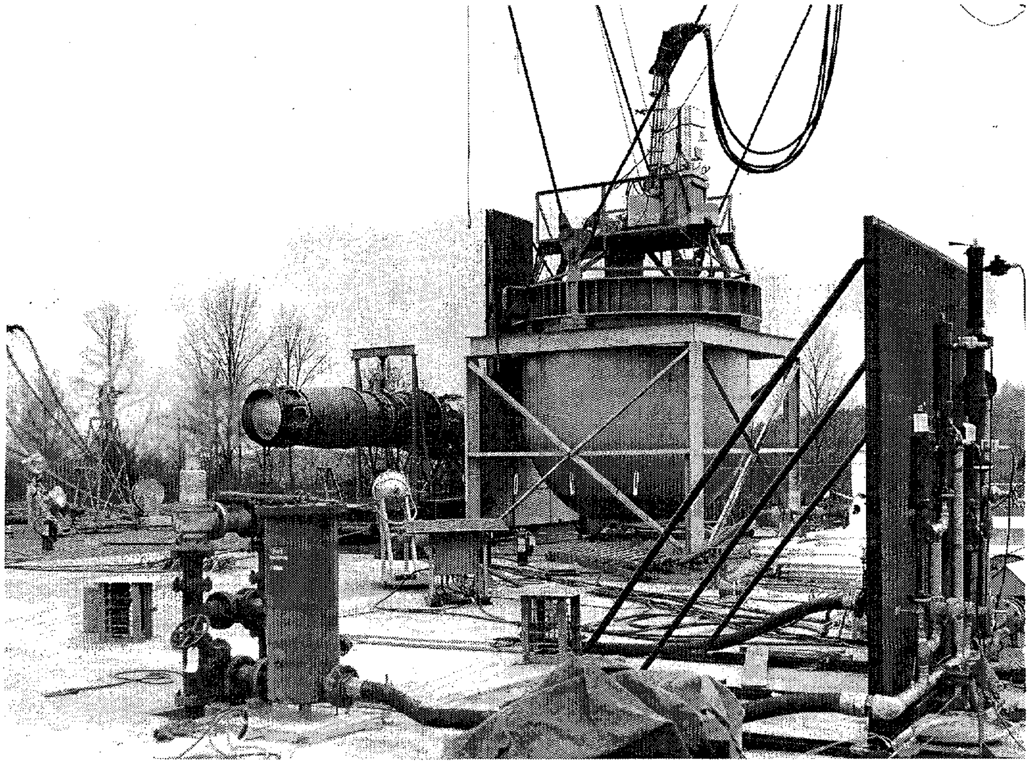

Fig. 9.8 TSF Reactor and J-57 Jet Engine in Operation Position.

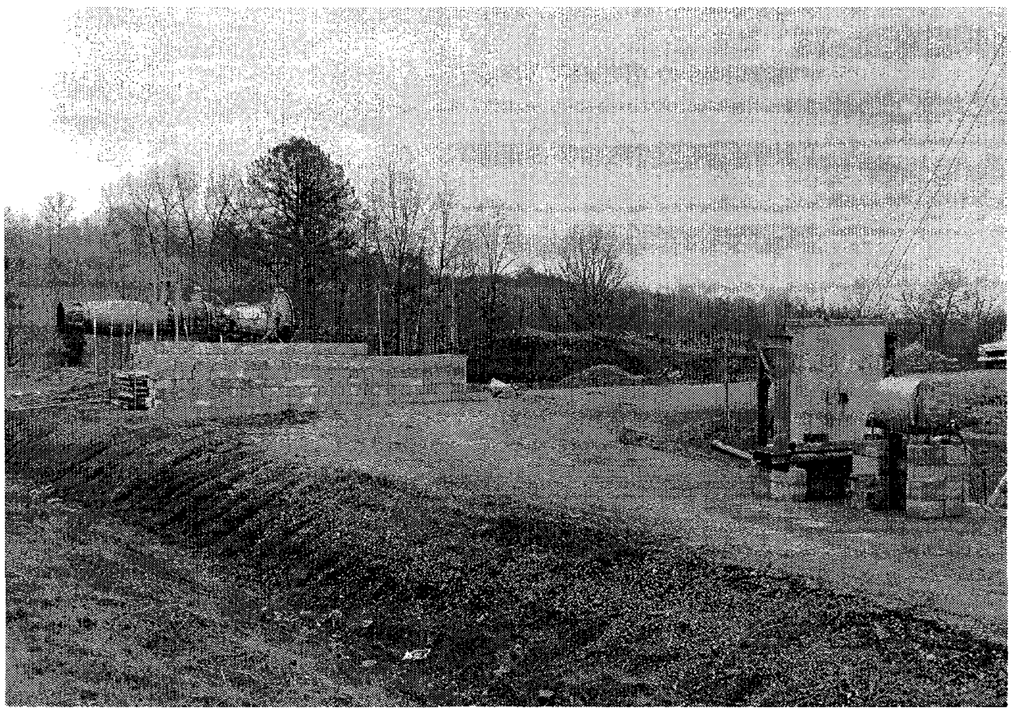

Fig. 9.9 Experimental Arrangement of J-57 Engine and Spectrometer (Extreme Right) for Gamma-Ray Spectrum Measurements.

energy. The measurements did indicate, however, the presence of an intensity greater than that expected from the scattering of gamma rays alone. These findings gave rise to consideration of a second gamma-ray source from neutrons leaving the reactor shield and either being scattered or captured. Inelastic scattering cross sections for neutrons were not well known and to measure them would have been a most difficult assignment for the Tower. Considering all of the possibilities that might help resolve the difficulty, it was decided to try and make a determination of the gamma-ray spectra from the inelastic scattering of neutrons as well as their capture. It was hoped that such a measurement would provide a determination of the gamma-ray dose rate from inelastically-scattered neutrons.

For these measurements[14] the reactor was submerged in the reactor storage pool, while allowing the neutrons to escape skyward through an 8-in.-diam air-filled tube placed against one side of the core, as shown in Fig. 9.10. This procedure minimized other sources of reactor gamma rays from reaching the detector. The detector, shown in the same figure, was well collimated in a concrete block house that viewed the sky through a slit in the blocks. That slit was filled with borated water to eliminate neutron interactions in the crystal. The projection of the area as seen by the detector passed through the cone of radiation projected by the pipe from the reactor as drawn in Fig. 9.11. Control of the thermal neutron capture gamma rays was exercised by placing a cap of borated plexiglas over the end of the pipe extended from the reactor core. Such an experimental procedure provided data from which the intensities for both the neutron capture gamma rays and gamma rays from inelastic scattering of neutrons in air were determined through a series of subtractions.

9.11 ASTR EXPERIMENT

An Aircraft Shield Test Reactor (ASTR) experiment[15] was performed at the TSF in cooperation with Convair, Forth Worth, in early 1958 to obtain data complementary to that obtained previously from their own Nuclear Test Airplane program. Measurements of both neutrons and gamma rays were to be made under conditions similar to those encountered in that program, but this time without the airplane structure present. For this mockup, the ASTR and its shield were attached to a support that allowed rotation of the reactor and shield through 180°. The crew compartment (Fig. 9.12) was placed 65 ft from the reactor and supported by a truss made of aluminum (Fig. 9.13). The ASTR consisted of an approximately cylindrical array of MTR-type fuel elements mounted horizontally between two grid plates (Fig. 4.10 in Chapter 4), where demineralized water served as moderator, reflector, and coolant. The core and moderator section were surrounded by a layer of lead and tanks to contain the water shield. Measurements were made in three phases: (1) repeats of data taken at Fort Worth with the reactor at 12.5 ft above ground level, (2) spectral measurements of the neutron source from ASTR, and (3) measurements at 200 ft in the air for comparison with NTA detector station measurements during flight. The influence of the air, ground, and airplane structure on measurements were derived from this study.

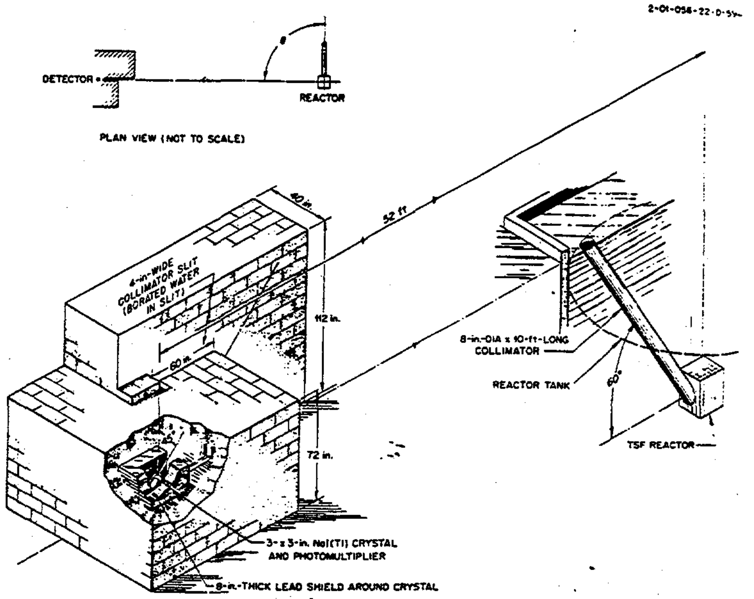

Fig. 9.10 Configuration for the Measurement of the Spectra of Gamma Rays Resulting from Neutron Interactions in Air.

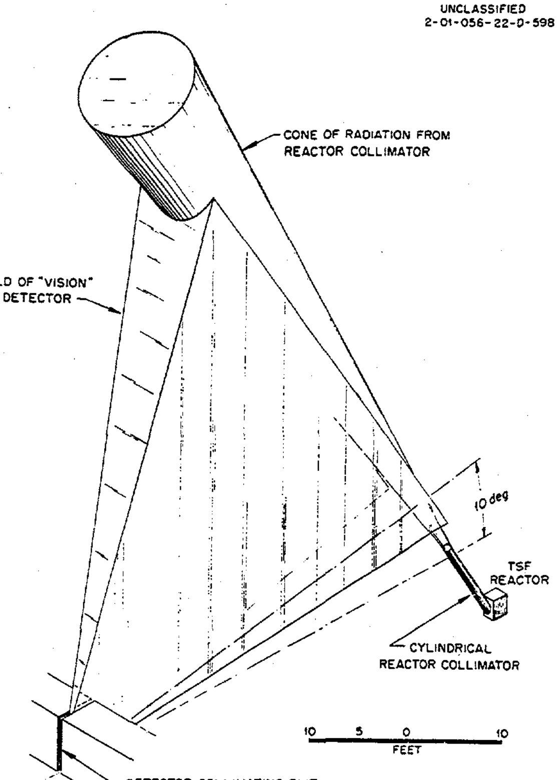

Fig. 9.11 Geometric Relationships of the Detector Lines of Sight and the Core of Radiation from the TSR.

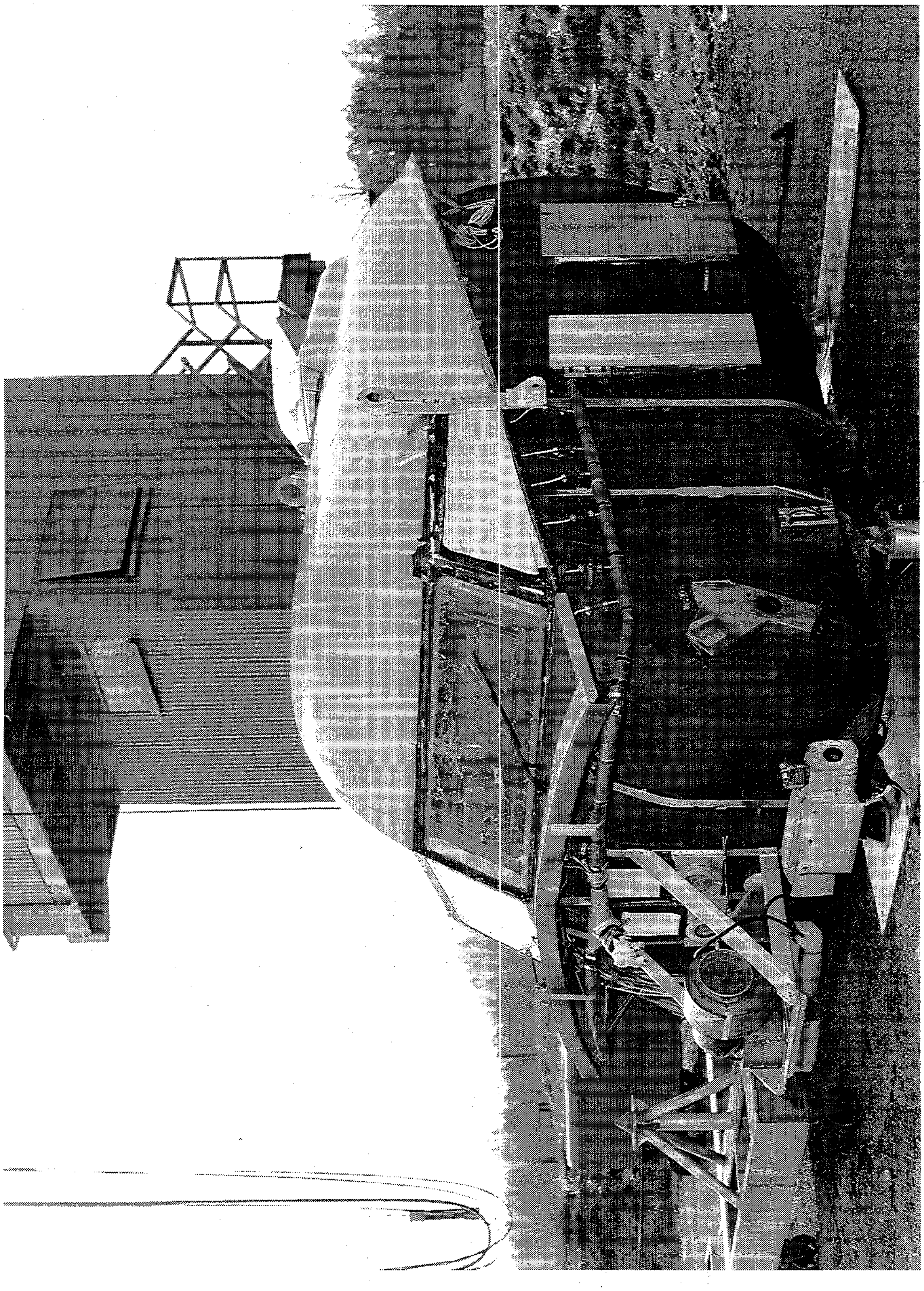

Fig. 9.12 The NTA Crew Compartment.

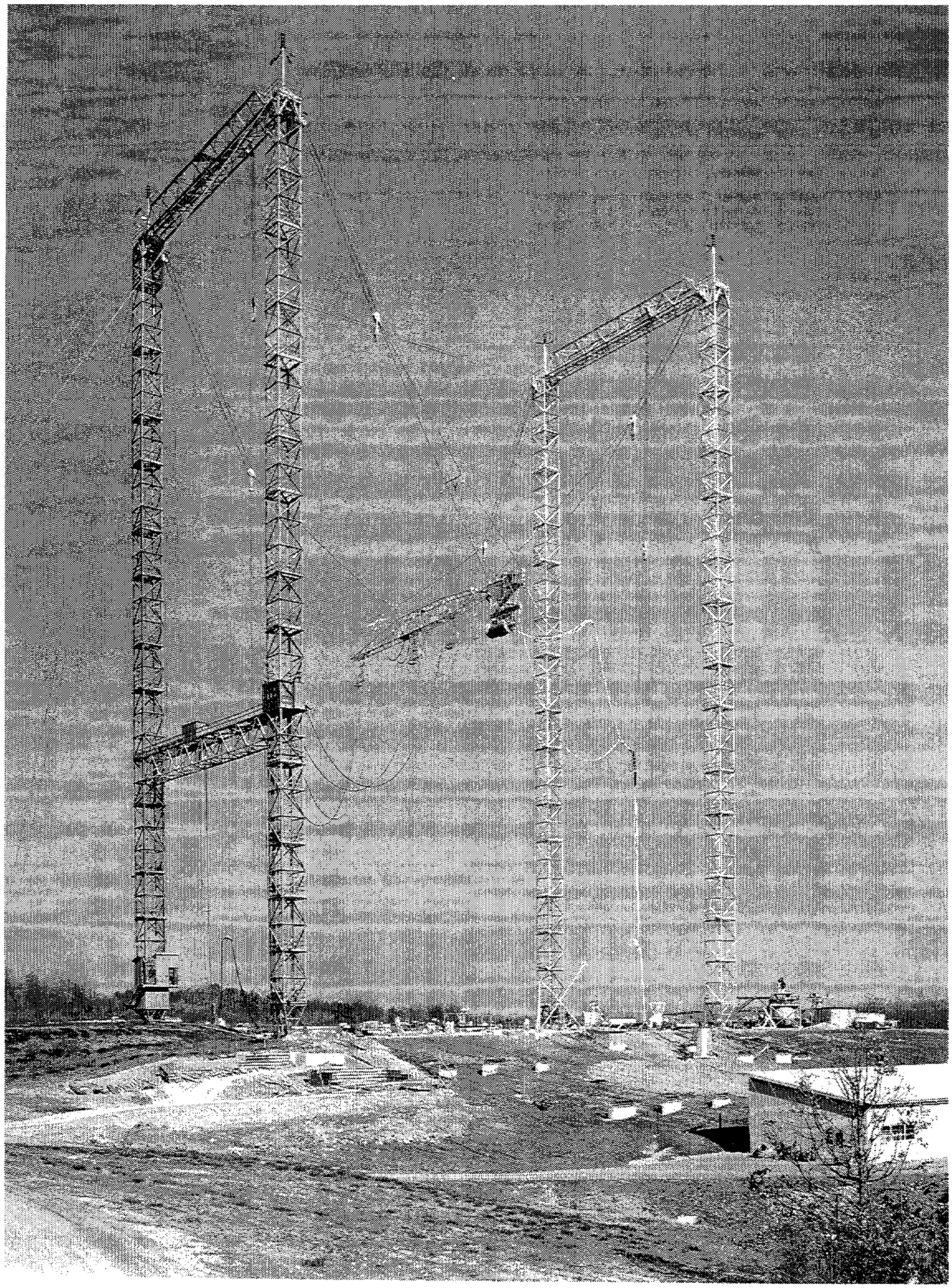

Fig. 9.13 The ASTR Suspended in Air at the TSF.

9.12 RADIATION MEASUREMENTS AT OUTER FENCE

The proposed construction of the 1958 Melton Hill Dam on the Clinch River near the TSF area made it necessary to make a study of the radiation that would reach long distances from the TSR-II and establish assurance that radiation at the dam site from operation of the TSR did not present a problem. Calculations[16] were backed by measurements out to 4000 ft using both the TSR and the ASTR reactors as sources.

9.13 2π EXPERIMENT REPEATS

A series of experiments were carried out in late 1956 for GE at the TSF to aid in the design of a shield for use at their Idaho Shield Test Air Facility (ISTAF). These experiments, designed to measure the ground scattered effect, were made as a function of altitude and shield configuration. Part of the measurements were made with a 2π shield (Fig. 9.14) that fit over the bottom half of GE reactor shield No. 1 to lessen the ground-scattered contribution. Subsequent calculations were not in reasonable agreement with these measurements, and GE requested that some of the measurements be repeated to provide verification (or lack thereof) of the previous measurements and then extend the work to include additional studies. These were completed[17] in December 1958, serving as the last series of measurements to be performed using the TSR-I as a source.

9.14 CAPTURE GAMMA RAYS IN IRON AND LEAD

Gamma rays resulting from an n,γ reaction in material are often the most important source of gamma rays in a reactor shield and an accurate knowledge of their resulting spectra is required for a proper analysis of many shielding experiments. This need was demonstrated by the difficulties that were being experienced in properly calculating the gamma-ray dose rates that were being measured inside crew compartment configurations. The first step in resolving this problem was the measurement of the gamma-ray spectrum from neutron capture in air, described earlier in Sect. 9.10. It was this success that prompted a continuation of this type of measurements[18,19] using a Po-Be source since at this time the TSR was in the beginning stages of dismantlement.

The arrangement that was used is shown in Fig. 9.15, where a bath of oil, 4 × 4 × 3 ft, was used as the means for generating thermal neutrons at the target material as well as providing protection for the NaI scintillator from neutron-induced interactions. Spectra were obtained for several of the common shielding materials that included aluminum, iron, and lead.

9.15 PRATT AND WHITNEY DIVIDED SHIELD MOCKUP EXPERIMENT

The first shield mockup[19] in which the TSR-II was used as a source combined the use of a reactor shield designed by Pratt and Whitney (P&W) Aircraft Company and the compartmentalized cylindrical crew shield used in earlier experiments with TSR-I. (Note: The original reactor is referred to either as TSR or TSR-I.) The P&W shield was an optimized, highly asymmetric, uranium-lithium hydride shield contained in SS that was fabricated to provide data for verification of the calculations in designing an advanced reactor shield (Sect. 5.5). The reactor was positiioned within the shield in the manner shown in Fig. 9.16. The mockup consisted of the reactor shield surrounding TSR-II separated from the TSF compartmentalized crew shield by approximately 64 ft. The compartments within the crew shield were filled with water, oil, or a mixture of oil and an organic boron compound.

This series of measurements were the last ones made under the auspices of the ANP program. The ORNL-ANP program was terminated on June 30, 1961, upon cancellation of the national ANP program.

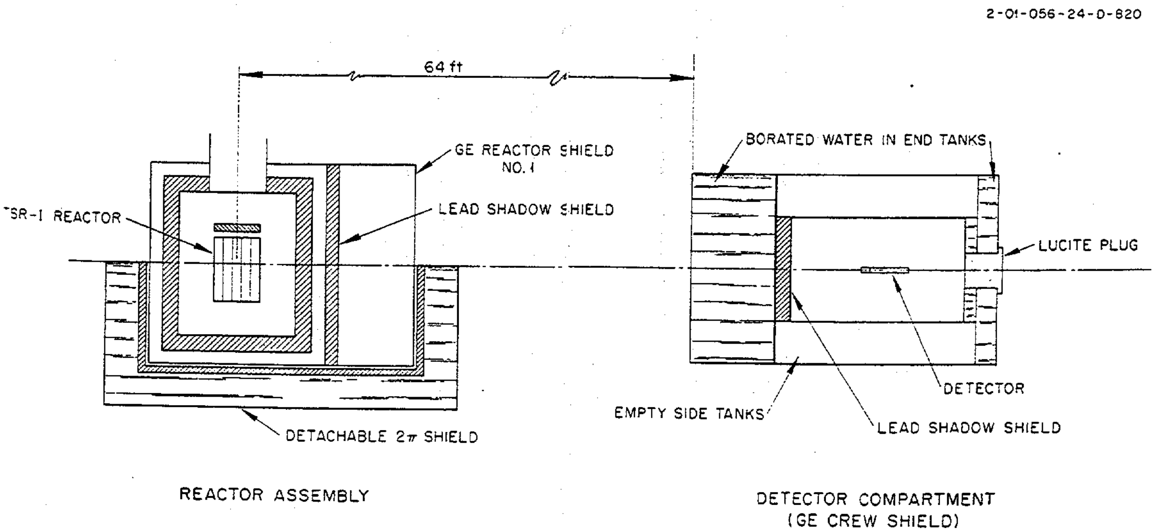

Fig. 9.14 Reactor and Crew Shield Arrangement used in TSF 2π Experiments ($\theta = 0°$).

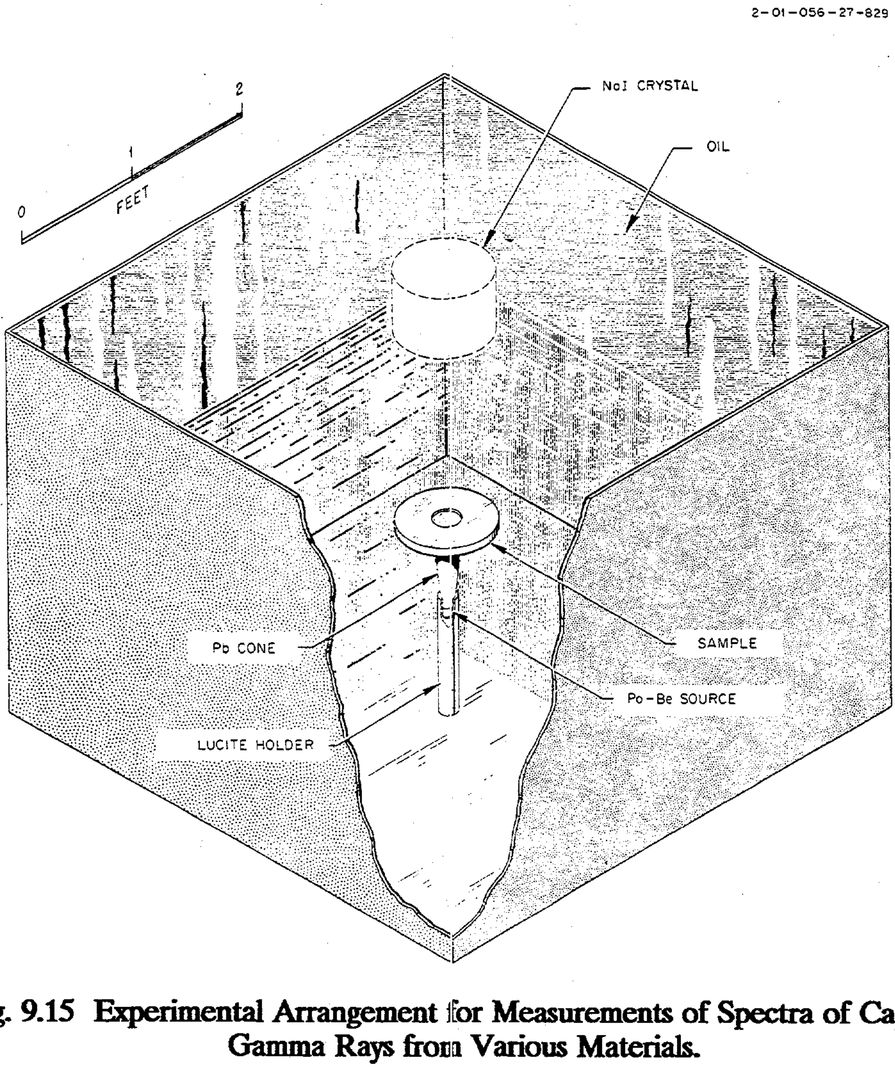

Fig. 9.15 Experimental Arrangement for Measurements of Spectra of Capture Gamma Rays from Various Materials.

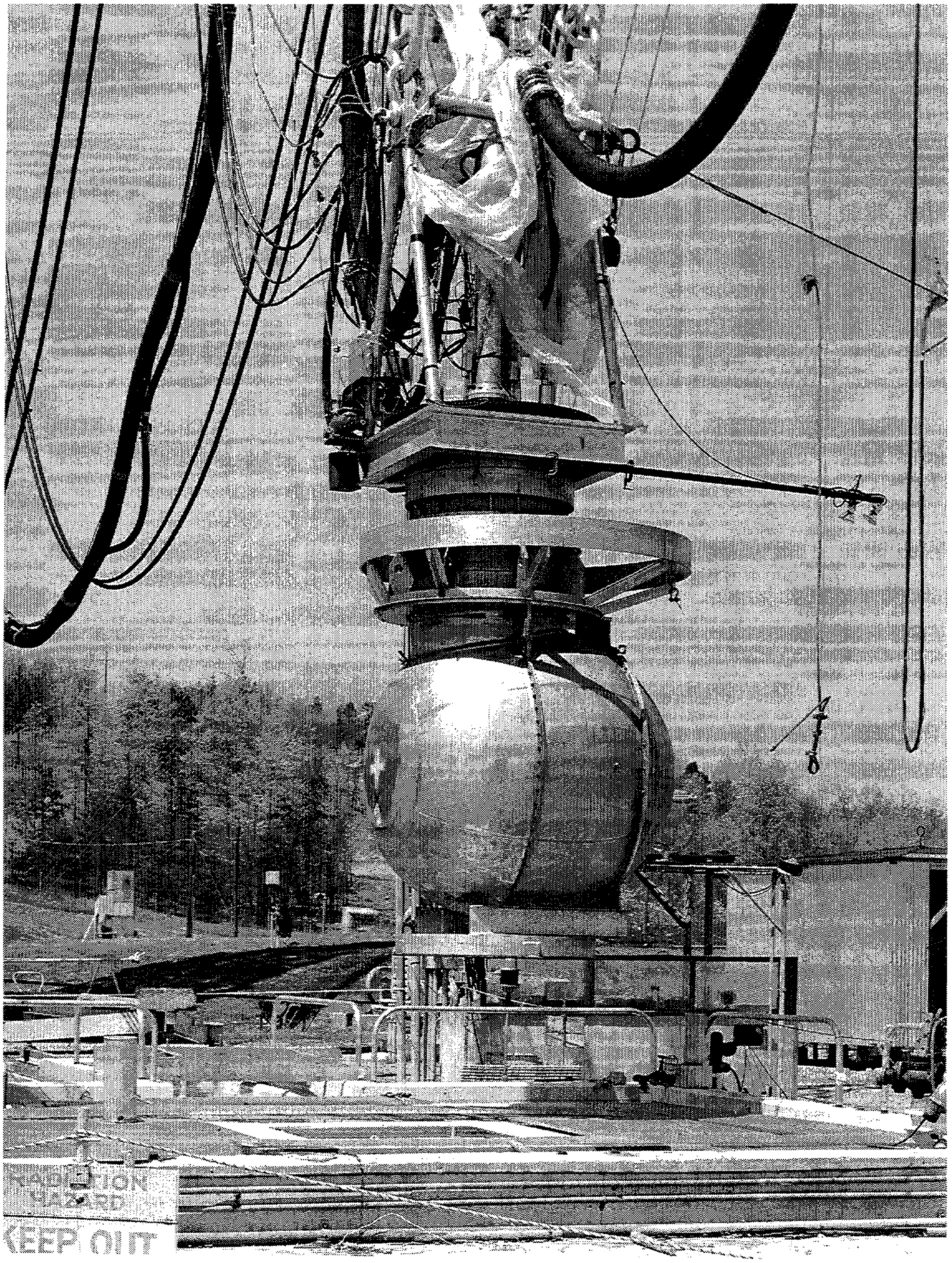

Fig. 9.16 Pratt & Whitney Shield Around TSR-II.

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10.0 DEFENSE ATOMIC SUPPORT AGENCY EXPERIMENTAL PROGRAM

10.1 INTRODUCTION

The Defense Atomic Support Agency (DASA) program was mainly concerned with protection against radiation from nuclear weapon bursts. This concern centered mainly on the development of calculational methods that, once verified by experiments at the TSF, could be used to determine radiation levels in structures other than those measured at the TSF. Through the years from 1960 to 1968, experiments pertaining to cross sections, capture gamma rays, albedos, and shield penetration, were all directed toward providing data that could be used to substantiate the methods used in code development or provide a base against which other data could be compared. Considerable time and effort was spent on design and fabrication so that the experiments truly represented the problems' needs, all of which contributed to understanding the nuclear environment in which the military must operate. The year, 1969, concluded the experimental work at the TSF under the sponsorship of DASA, except for the TORT experiment (Sect. 10.9) performed in 1984.

10.2 ANGULAR DEPENDENCE OF FAST NEUTRON DOSE AND THERMAL FLUX REFLECTED FROM CONCRETE

Considerable interest by DASA in fast neutron and thermal neutron albedo information was generated in connection with solving multibend air duct transmission problems in concrete structures. An experimental program[1] was performed at the TSF, concurrently with Monte Carlo calculations by R. E. Maerker,[2] to obtain the fast neutron dose rate and thermal flux scattered from a concrete slab as a function of the angles of incidence and emergence. For this work (Fig. 10.1) a fixed collimated beam of neutrons from the TSR-II was directed onto a 6-ft-sq, 9-in.-thick concrete slab attached to a turntable capable of rotating the slab through an angle of 180° about the vertical axis and 90° about the horizontal axis. Measurements were made with a well collimated detector system capable of rotating in the horizontal plane through about a 160° arc using the beam centerline as a reference of 0°. Experimental results were obtained for 147 combinations of the angle of incidence of the beam and the polar and azimuthal angles of scattered neutrons.

Fast neutron dose rates were obtained using a Hurst-type dosimeter and a modified long counter (normalized to the dosimeter) as a function of reflected polar and azimuthal angles for four angles of incidence. The TSR-II served as the neutron source with its tightly collimated beam impinging on the concrete slab at either 8 ft or 32 ft away from the end of the reactor collimator.

The procedure for obtaining thermal neutron data was similar but much more elaborate,[3] requiring sixteen different measurements with the BF_3 counter for each of 72 different combinations of the incident angle and the polar and azimuthal reflected angles, to effect a single measurement of the thermal neutron albedo. Cadmium-covered and bare detector readings were taken for the cases of cadmium over the slab, cadmium over the reactor collimator, cadmium over both simultaneously, and both slab and collimator bare for both foreground and background geometries. The thermal contribution from incident epithermal neutrons was determined in the same subtraction process but needing only eight of the sixteen measurements.

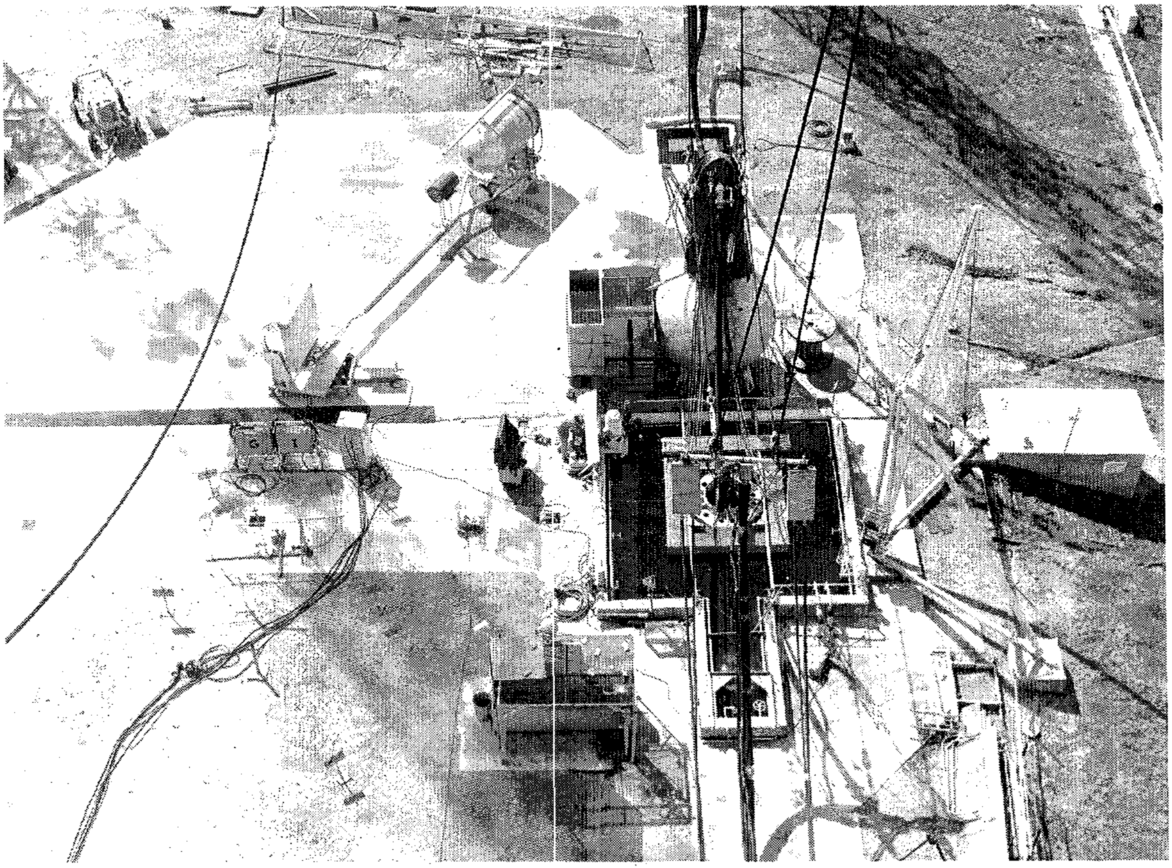

Fig. 10.1 Experimental Arrangement for Measurements of Fast Neutrons Reflected from Concrete.

This was a fun experiment because the calculations were being made simultaneously with the measurements, and thus provided an immediate check on the results. The agreement was so good that it created an atmosphere of excitement as to how close the next comparison might be, as several had agreed to the second and third decimal place.

10.3 NEUTRON FLUX IN A RECTANGULAR CONCRETE DUCT

Neutron transport studies in ducts were conducted at the TSF under the auspices of the DASA as part of the reactor and weapons radiation shielding program. The experiment was designed to check on the accuracy of the calculational methods being developed to predict the neutron streaming through multi-directional concrete duct systems. The experiment verified that the calculational methods would be highly successful when the albedo for the material in the wall was determined first before using it in a Monte Carlo transport code that was dependent only on the duct geometry.[4]

To demonstrate the validity of this approach, a series of measurements were made of the single-collision fast-neutron dose rates, and the thermal neutron fluxes along the centerline of a 3-ft-sq duct having one, two, and three-legs as shown in Figs. 10.2, 10.3, and 10.4. The reactor beam was projected at an angle of 45° to the duct centerline, striking only the inside surface of the duct wall that was opposite the reactor collimator. Measurements were made with and without that wall present in the duct. For the fast neutron dose rate measurements, two traverses were sufficient. For determination of the thermal flux, however, six different axial traverses were needed to separate the thermal flux count rates along centerline into those resulting from thermals incident and then reflected from the source wall from those where epi-cd neutrons were incident on the source wall and reappeared as thermals along the centerline.

This was the first use of a spherical BF_3 detector covered with a polyethylene ball as a fast neutron detector, due to insufficient sensitivity of the Hurst-type dosimeter. These BF_3 measurements were intercalibrated with the dosimeter. Close agreement between measurements and calculations through three bends gave credence to the calculational method.

10.4 PRELIMINARY MEASUREMENTS OF THERMAL NEUTRON CAPTURE GAMMA-RAY SPECTRAL INTENSITIES FROM SEVERAL SHIELDING MATERIALS

An attempt was made to measure the gamma-ray spectra resulting from thermal neutron capture for a number of different shielding materials placed in a beam of neutrons from TSR-II. This was the initial part of a study of secondary gamma-ray production in shields.[5] Materials were selected, where possible: (1) to provide single energy gamma rays due to thermal capture in order to accumulate a series of response functions for a 3-in. NaI detector, (2) that had a known cross section for the interaction, (3) that had a known number of gamma rays emitted per thermal capture, and (4) that would provide a range of gamma-ray energies sufficient to provide needed extrapolated response functions. Other materials were then chosen to test the response functions and verify their validity. The geometry for the measurements is shown in Fig. 10.5, where the detector looks at the beam-emergent side of the slab. Relative response functions for the crystal were obtained through the use of single-energy gamma-ray sources and the decay of ^{16}N in the reactor water. Absolute values were obtained using ^{65}Zn and capture gamma rays from hydrogen, lead, and iron. Materials investigated were Al, Fe, Pb, concrete, H_2O, soil, sand, cement, graphite, and polyethylene. The pulse-height spectrum from thermal capture in concrete, shown in Fig. 10.6, is typical of that obtained using a 3-in. crystal.

10.5 SPECTRA OF UNCOLLIDED FISSION NEUTRONS TRANSMITTED THROUGH THICK SAMPLES OF NITROGEN, OXYGEN, CARBON, AND LEAD

An important part of the research program for DASA was the study of neutron transport through the atmosphere, where calculations at this time yielded widely differing results by as much as factors of three to ten when at 1500 meters from the source. In an attempt to resolve some of these

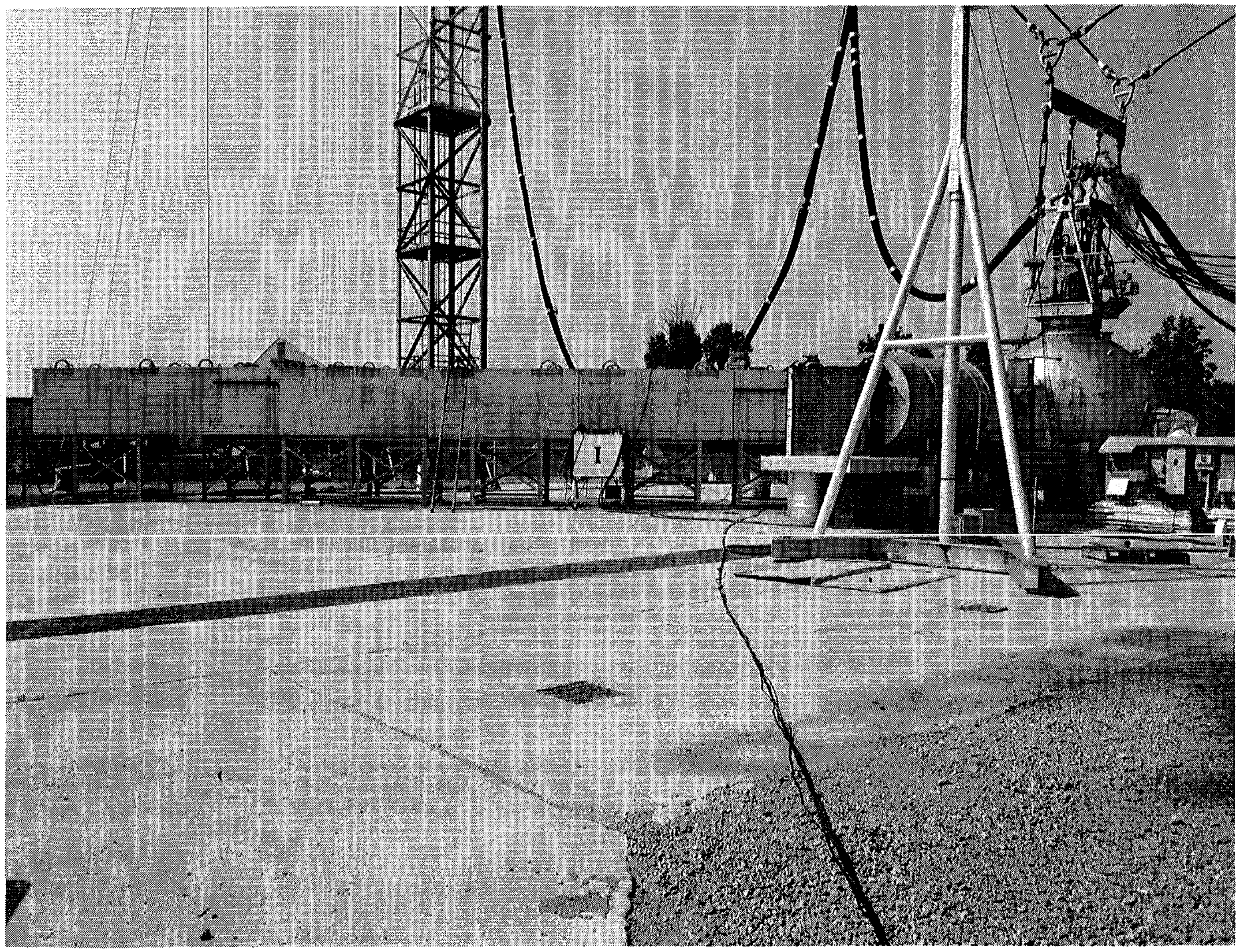

Fig. 10.2 Experimental Arrangement for Measurement of the Fast Neutrons Transmitted down a Concrete Duct.

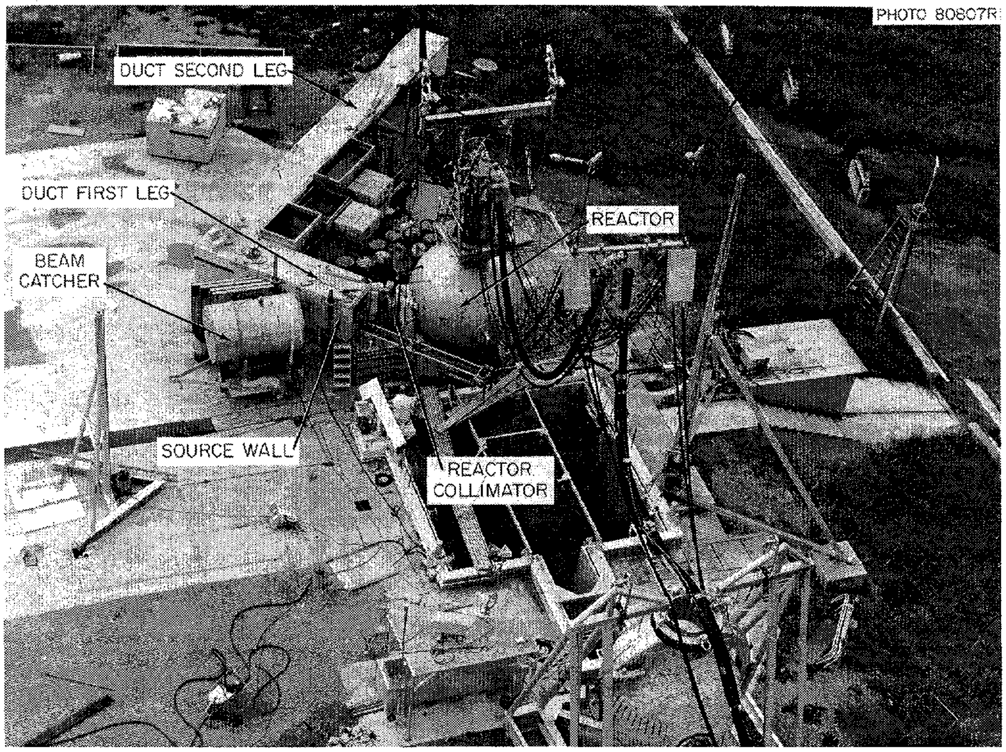

Fig. 10.3 Photograph of Duct with One Bend.

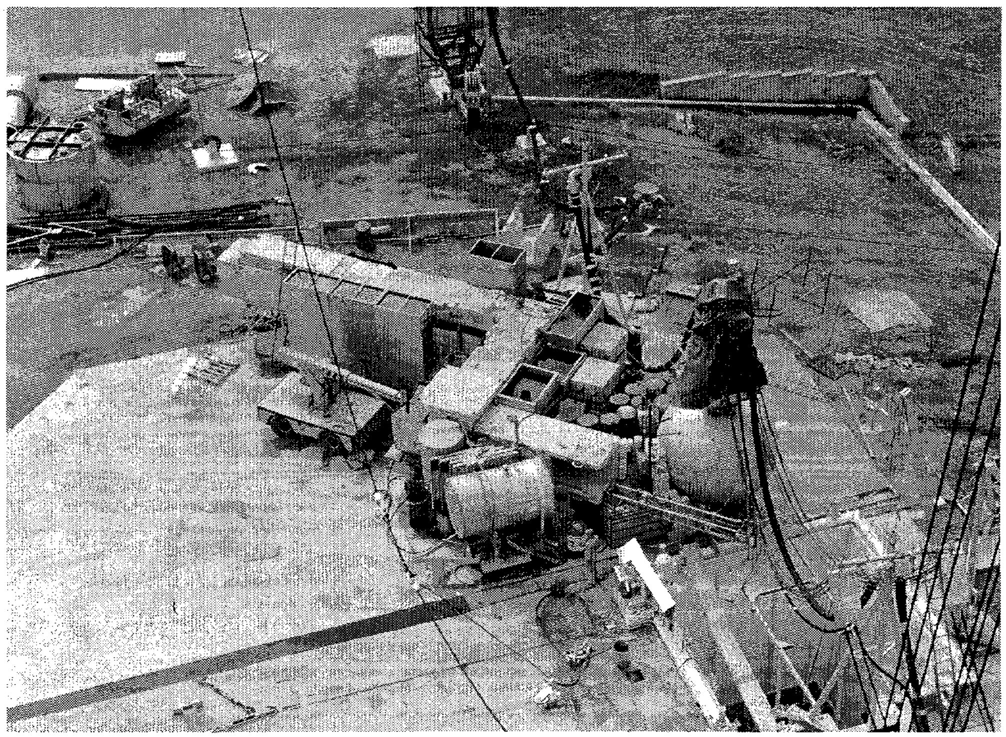

Fig. 10.4 Photograph of Duct with Two Bends.

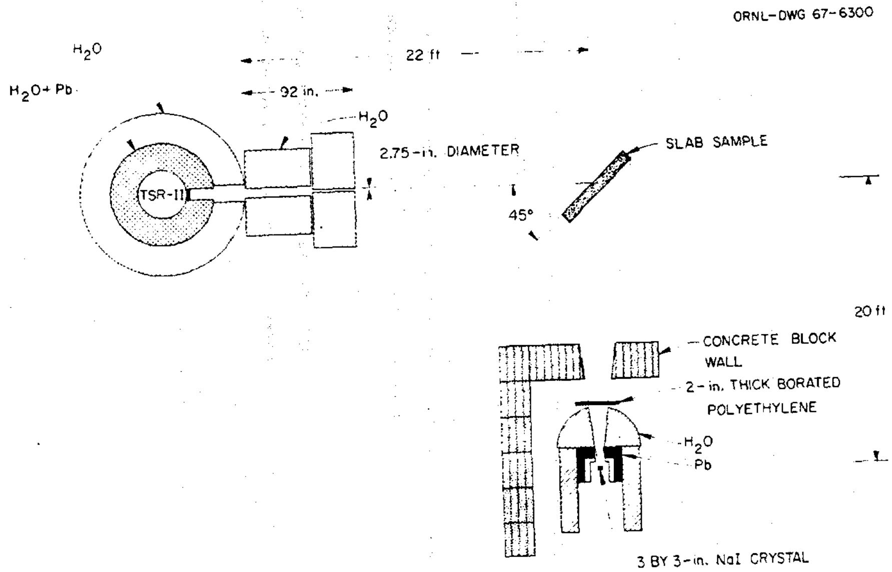

Fig. 10.5 Schematic Diagram of Geometry for Thermal-Neutron Capture Gamma-Ray Experiment.

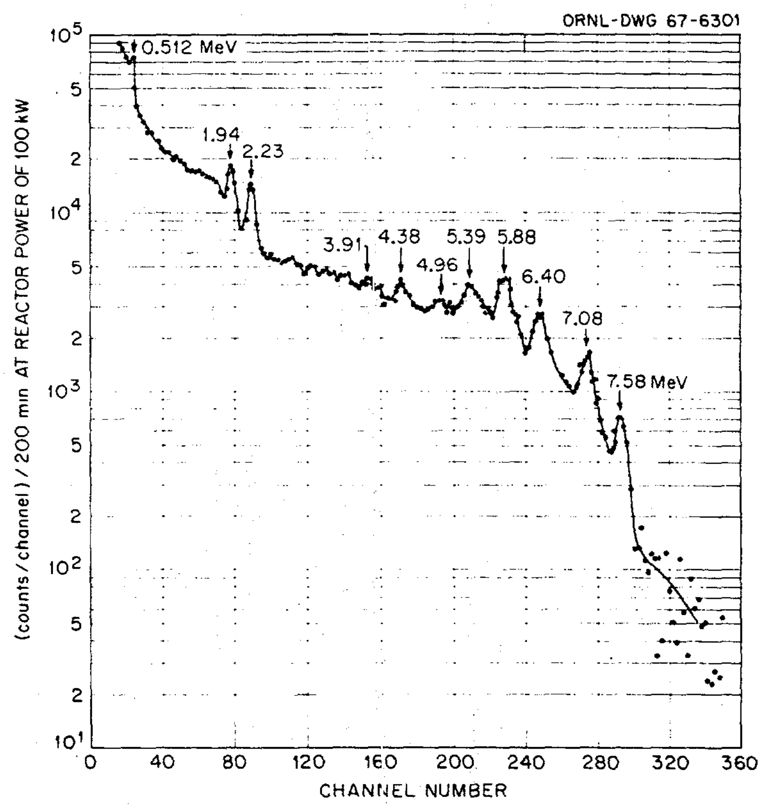

Fig. 10.6 Thermal-Neutron Capture Gamma-Ray Pulse-Height Spectrum for 9-in.-thick Concrete Slab Containing Iron Rod Reinforcements.

differences, a program was initiated at the TSF to make neutron spectral measurements through thick samples where the cross section effects could be emphasized.[6] The initial concern was measurement of the uncollided neutron spectra through samples of liquid nitrogen and oxygen. By using thick samples and good experimental geometry that essentially eliminated any scattered contributions to the detector, it was possible to emphasize the effects in the spectra due to the valleys in the cross sections. The validity of the measurements was checked by using the same technique to obtain neutron spectra through thick samples of lead and carbon since their total cross sections were well known.

The good geometry was obtained by collimating the reactor beam tightly and placing the detector at 100 ft from the reactor with the shield sample halfway in between. The NE-213 spectrometer was well collimated and adequately shielded, as shown in Fig. 10.7, to minimize the background. The result from a measurement through an oxygen sample is shown in Fig. 10.8.

10.6 ANGLE-DEPENDENT NEUTRON ENERGY SPECTRA EMERGENT FROM LARGE LEAD, POLYETHYLENE, DEPLETED URANIUM, AND LAMINATED SLAB SHIELDS

The purpose of this experiment was to provide comprehensive data for comparison with two dimensional discrete ordinates (DOT) transport calculations rather than to simulate any practical shielding application. The experiment[5] was constructed so that it might be represented exactly in cylindrical space dimensions, that is, r-z geometry. The only non-cylindrical components were the square slabs. To simulate an infinite transverse dimension with respect to the source, the reactor beam was collimated to a small area of the slab. Comparisons of the emergent flux data and the calculations were made on an absolute basis. A schematic of the geometry for this experiment is shown in Fig. 10.9. The photograph in Fig. 10.10 shows the reality of the mockup. Angular distributions of the neutron spectra and fast neutron dose rates were made for samples of lead, polyethylene, ^{238}U and for several slab combinations. The detector collimator rotated about a point coincident with the exit face of the last slab in the mockup.

10.7 GAMMA-RAY SPECTRA RISING FROM THERMAL-NEUTRON CAPTURE IN ELEMENTS FOUND IN SOILS, CONCRETES, AND STRUCTURAL MATERIALS

In the mid 1960s, deep concern was expressed regarding the disagreements that existed with respect to the gamma-ray spectra available from neutron capture in elements used in reactors and reactor shields. These discrepancies became even more important in shielding studies as the accuracy of transport calculations improved. By the late 1960s, it was realized that this accuracy was as much determined by the errors in pertinent cross sections as it was by errors in the calculational methods. Thus it was considered vital that additional measurements be made for some of the light elements and nearly all of the heavy elements used in shielding.

For these measurements, both the TSR-II source and the detector were tightly collimated, with a concerted effort to minimize background and other undesired contributions.[7] The NaI crystal was upgraded to a 5-in.-diam right cylinder placed inside the detector shield ball that was designed for use

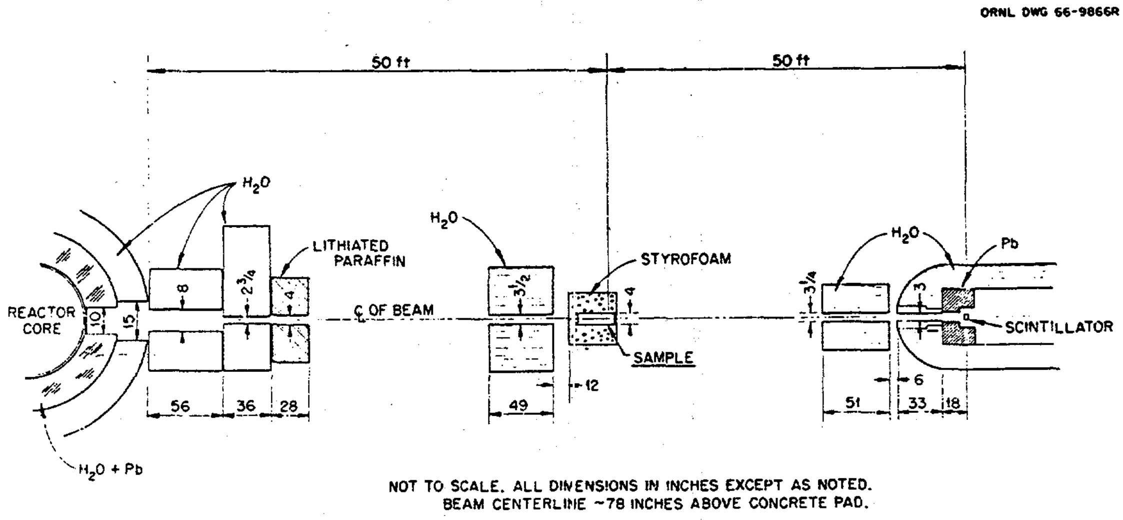

Fig. 10.7 Schematic Diagram of Uncollided Flux Experiment.

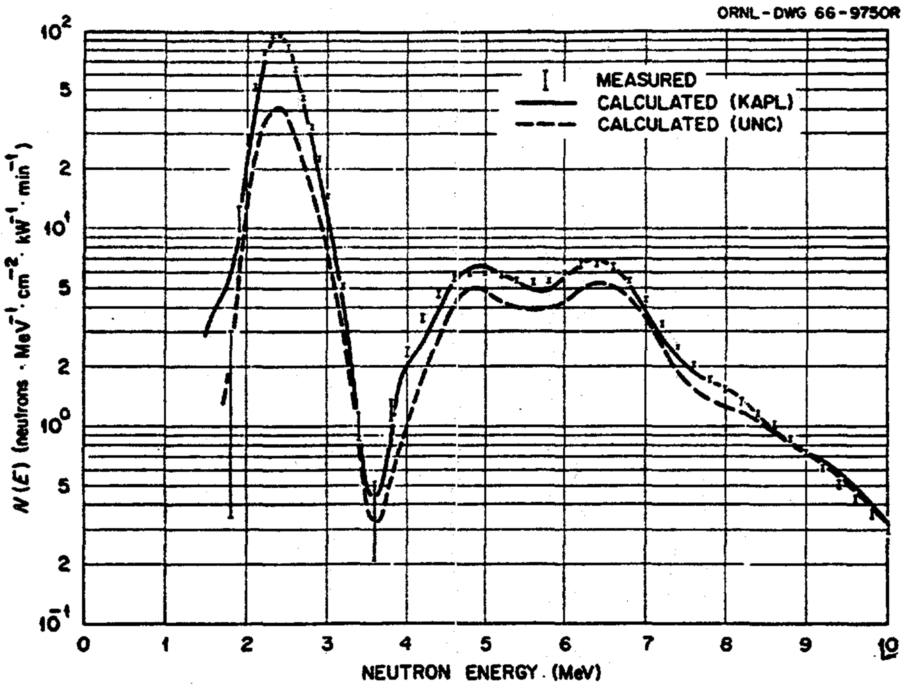

Fig. 10.8 Spectra of Neutrons Transmitted through 24-in. of Oxygen: Comparison of Calculations and Meausurements.

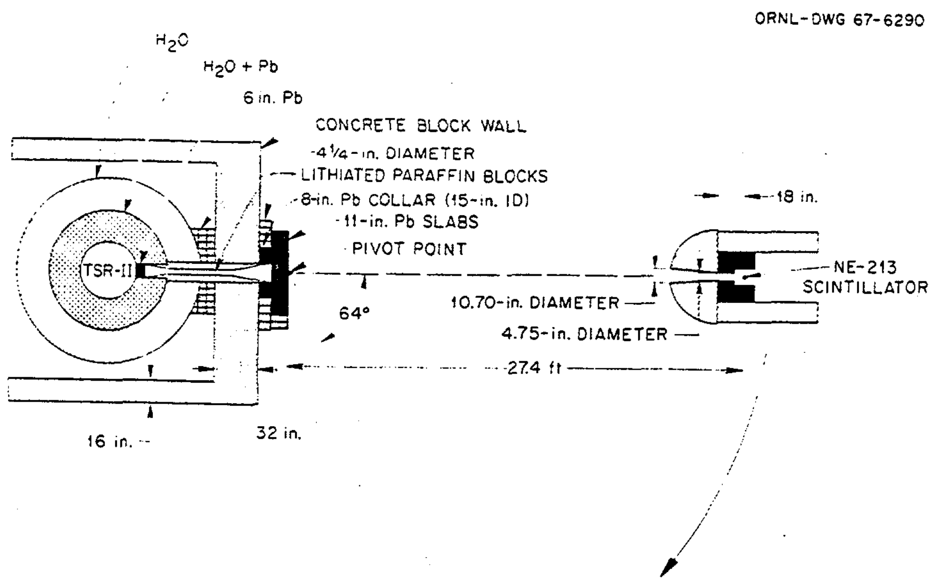

Fig. 10.9 Schematic Diagram of Experimental Arrangement.

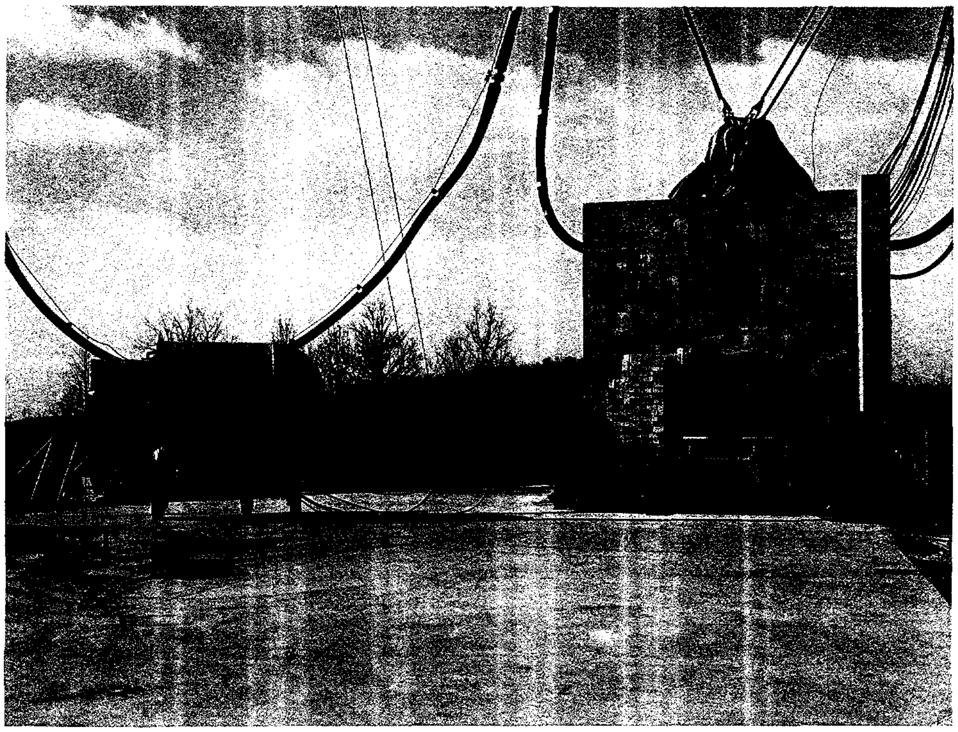

Fig. 10.10 Photograph of Experimental Arrangement.

in the ANP program. Special lead irises were fabricated so that scattering from the collimator was minimal at the detector. Borated polyethylene covered the collimator opening to reduce neutron-induced gamma-ray production within the crystal. The centerline of the detector collimator, shown in Fig. 10.11, intersected at right angles with the axis of the reactor beam. Appropriate mappings were made of the thermal flux incident at the slab surface to generate a source term. Response functions were generated for this detector-housing arrangement using decay and capture gamma-ray sources. Elements exposed were iron, aluminum, copper, titanium, zinc, calcium (CaO), potassium (K_2CO_3), sodium (Na_2CO_3), chlorine (NaCl), silicon (SiO_2), nickel, stainless steel, barium (BaO), and sulfur. Not all elements found in reactor shield studies were measured, but it was certainly a big step in the right direction.

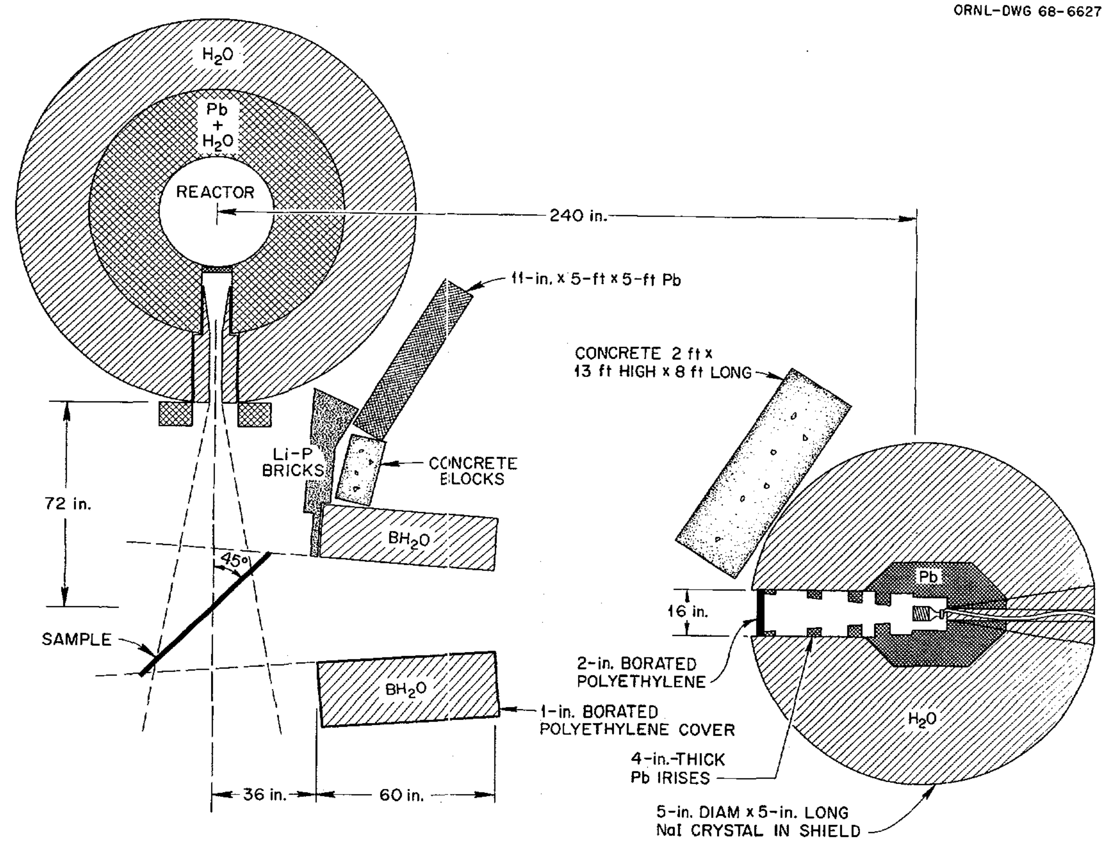

Fig. 10.11 Schematic Diagram of Geometry for Thermal-Neutron Capture Gamma-Ray Experiments.

10.8 EXPERIMENTAL GAMMA-RAY SPECTRA ABOVE ONE MEV FROM THERMAL NEUTRON ABSORPTION IN ^{235}U

The purpose of the experiment was to determine the gamma-ray energy spectrum above one MeV resulting from neutron interactions with fissile materials such as ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu. Three different incident neutron spectra from TSR-II were used: (1) the TSR-II spectrum bare, (2) the TSR-II spectrum filtered by 32-mils of cadmium, and (3) TSR-II spectrum filtered with 2.58 g/cm^2 of ^{10}B. The experimental mockup used the same approach as described previously in Sect. 10.7, with extra collimation placed between the sample and the detector to improve the foreground-to-background ratio for the small foil samples. Twelve more inches of lithium hydride, encased in aluminum, was placed in front of the detector collimator to ensure that neutrons generated in the sample did not reach the detector. Gamma rays generated at the sample were collimated by voids in lithiated paraffin and lead before reaching the lithium hydride. Further background deterrents were also established with borated poly and concrete. Knowledge of the source intensity at the sample was refined by exposing a copper foil the same size as the ^{235}U foil and then normalizing it to foil data for which the source was known. The experimental approach and techniques used in the analysis of the data are described in a report by R. S. Booth.[8]

10.9 VERIFICATION EXPERIMENT OF THE THREE-DIMENSIONAL OAK RIDGE TRANSPORT CODE (TORT)

In the summer of 1984, an experiment was initiated at the TSF to provide data for validation of the radiation transport computer code TORT (Three-Dimensional Oak Ridge Transport) under development at ORNL by W. A. Rhoades.[9] TORT was to be applied to calculations of the radiation that passed through walls and interval voids of large concrete structures that survived the explosive force of the atomic bombs at Hiroshima and Nagasaki. The code was an extension of the two-dimensional discrete ordinates code DOT4 to three-dimensional geometries developed as part of a continuing study supported by the Defense Nuclear Agency (DNA). The experiment was also part of the DNA study, attempting to simulate some of the exposure that might have occurred at the time of the detonations.

In the experiment,[10] a small, simple, single-story concrete structure 183 cm wide, 304.8 cm long, and 244 cm high (Fig. 10.12), was exposed to a collimated beam of radiation from TSR-II with measurements made internally and externally to the building. Variations in the structure were introduced during the experiment by successive changes in the outer wall and within the building. Specifically, blocks were removed from the front wall to form a window; a concrete support pillar within the building was relocated; and a central concrete wall was added to divide the single room into two rooms having approximately the same dimensions. A schematic of one of the mockups is given in Fig. 10.13.

The TSR-II beam was altered by a spectrum modifier to closely resemble the spectra of radiation emitted by "Little Boy," the bomb detonated over Hiroshima. Both neutron flux and the energy deposition from gamma rays were measured using a series of Bonner balls and thermoluminescent detectors.

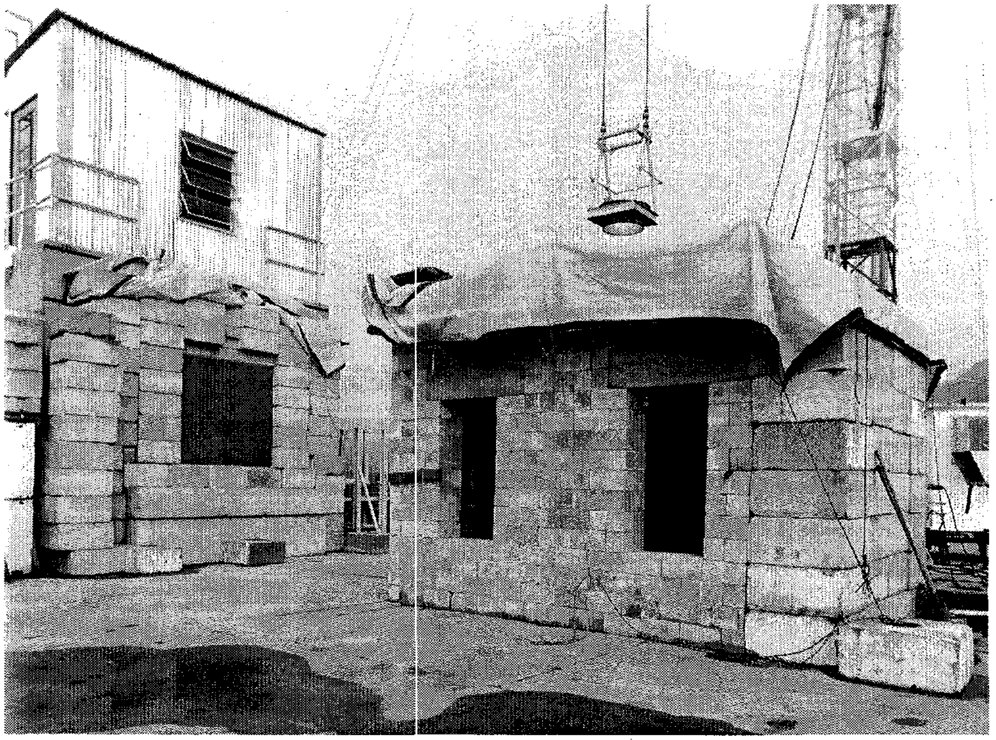

Fig. 10.12 Photograph of TORT Experimental Configuration.

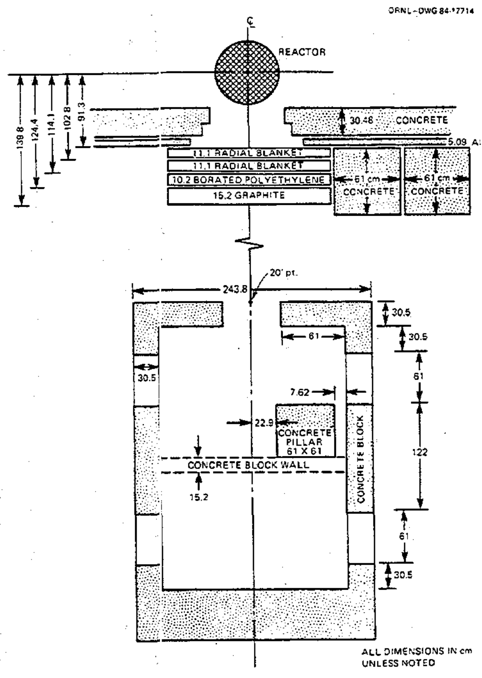

Fig. 10.13 Schematic of SM and Concrete Block House (Item II D).

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11.0 SPACE NUCLEAR AUXILIARY POWER PROGRAM

11.1 INTRODUCTION

The Oak Ridge National Laboratory-Aircraft Nuclear Propulsion (ORNL-ANP) program was discontinued on June 30, 1961, as part of the nationally terminated ANP program. With this loss, the efforts within the TSF shielding group were gradually re-directed toward other programs in the nuclear energy field. Shielding research, in direct support of the Space Nuclear Auxiliary Power (SNAP) program was one of them, wherein reactors were to provide the power source for space vehicles. The work was an extension of a low-energy neutron penetration study initiated in the ANP program, but with emphasis shifting toward studies of the high-energy neutron region where the angular and energy distributions were of great interest to the space program.[1]

Neutron penetration through a shield was always an essential part of a shield design study, but for the space system, where structures such as radiators, reactor controls, and other vehicular parts are a necessity, the scattering of neutrons around the shadow shield was also a vital part of the experimental and calculational investigations. To shadow shield these parts would have meant increases in the overall weight of the system and such weight increases might possibly be excessive for a mission. Thus it was important to develop calculational techniques for determination of these contributions, verification of which would require appropriate experiments.

The calculational program included the development of both a simplified technique for use where there would exist many changes in the early designs of the systems, and a more accurate method that would predict doses in systems with existing flight hardware. The simplified method was a single scattering calculation, in which reactor codes were used to obtain the source energy distribution along with the angular distribution. Monte Carlo techniques were used for the more accurate calculations, determining detailed angular and energy distribution of neutrons penetrating the reactor reflector and parts of the shadow shield. Calculations of the probability of these neutrons reaching the payload included multiple neutron collisions.

A preliminary experiment was performed in early 1962 at the TSF in which neutrons were scattered from a beryllium sample. The measurements were somewhat limited in that the detector remained 90° to the reactor beam while only the sample rotated. However, the experiment was a success, demonstrating that it would be feasible to measure the radiation scattered from such samples in the presence of radiation fields from the reactor source and air scattering. Analysis by single scattering theory indicated the sensitivity of the scattering to the incident neutron spectrum.

That work began what was to become a three-phase program designed jointly by ORNL and Atomics International (AI) in support of the SNAP systems. Phase I included measurements and calculations of neutron scattering by elements that had reasonably well known cross sections, some of which were to be utilized in various components of the SNAP systems.

Phase II was aimed at developing shielding technology pertinent to the design of lithium hydride (LiH) shadow shields for use in the SNAP program. Shields of LiH and polyethylene were studied, with the results from both experiments serving to verify the Monte Carlo calculations to give confidence in the methods.

For Phase III, an experiment was performed using a full-size SNAP shield and reactor system to determine the neutron intensity at the payload position. These results provided verification for the further development of calculation methods, including the modification of the 05R Monte Carlo neutron transport code for increased efficiency in deep penetration studies.

By 1971, the SNAP program had been drastically reduced at the TSF. Some effort was extended, however, to repeat measurements of both neutron transport and gamma-ray production in uranium. This was done due to the large discrepancies between calculations and experimental results when studying the laminated shields. Studies of an iron sample were also included to help demonstrate the reliability of the measurements since the cross sections were well known and the calculational ability using iron had been well demonstrated. By year's end, the use of the SNAP reactor was almost nil, with only two short runs made during 1972. The last test run on the reactor was made February 18, 1973. All operations of the reactor ceased on May 9, 1973, when orders were received to dismantle it for storage.

11.2 NEUTRON-SCATTERING EXPERIMENTS WITH COLLIMATED BEAM AND DETECTOR

The first SNAP experiment, completed the first half of 1962, was preliminary in nature and performed to determine the feasibility of measuring neutron scattering from support tabs and beryllium reflector segments that were part of the various SNAP reactor control systems.[2] The experiment did not use the SNAP reactor, however; instead, the geometry, simplified as given in Fig. 11.1, consisted of the collimated TSR-II reactor and detector shields placed so that their collimators intersected at right angles. TSR-II was used since its collimated shield provided a beam of neutrons in the horizontal plane and could be easily matched by the collimator in the detector shield. A slab of beryllium, as the neutron target, was placed at the point of intersection of these two collimator centerlines and rotated about a vertical axis to provide selected angles of incidence with complementary angles of scattering when referenced to the sample. The observed ratios of foreground-to-background readings indicated such measurements could be made with a reasonable degree of confidence if some improvements were made in both the collimation of the reactor beam and the detector.

A second series of measurements[3] were made following improvements to the reactor and detector collimator systems. Several reactor collimators were fabricated as selection of the proper collimator was to be part of this experiment. The detector collimator was considerably reduced, along with the void area around the detector, in an attempt to minimize the amount of air-scattered neutrons reaching the detector, as shown in Fig. 11.2. Measurements were again made at 90° to the reactor beam centerline for cylindrical samples of aluminum, beryllium, carbon, and iron. Results indicated that for measurements using the appropriate reactor collimator with a 6-in.-diam aluminum cylinder in the beam, the foreground-to-background (no sample) ratio was 100 to 1.

As part of this experiment, V. V. Verbinski[4] and associates made a measurement of the fast neutron spectrum for this reactor shield collimation using a shielded-diode spectrometer. Knowledge of the spectrum incident on a sample was important as input data for Monte Carlo calculations to be used in analyzing the experimental data.

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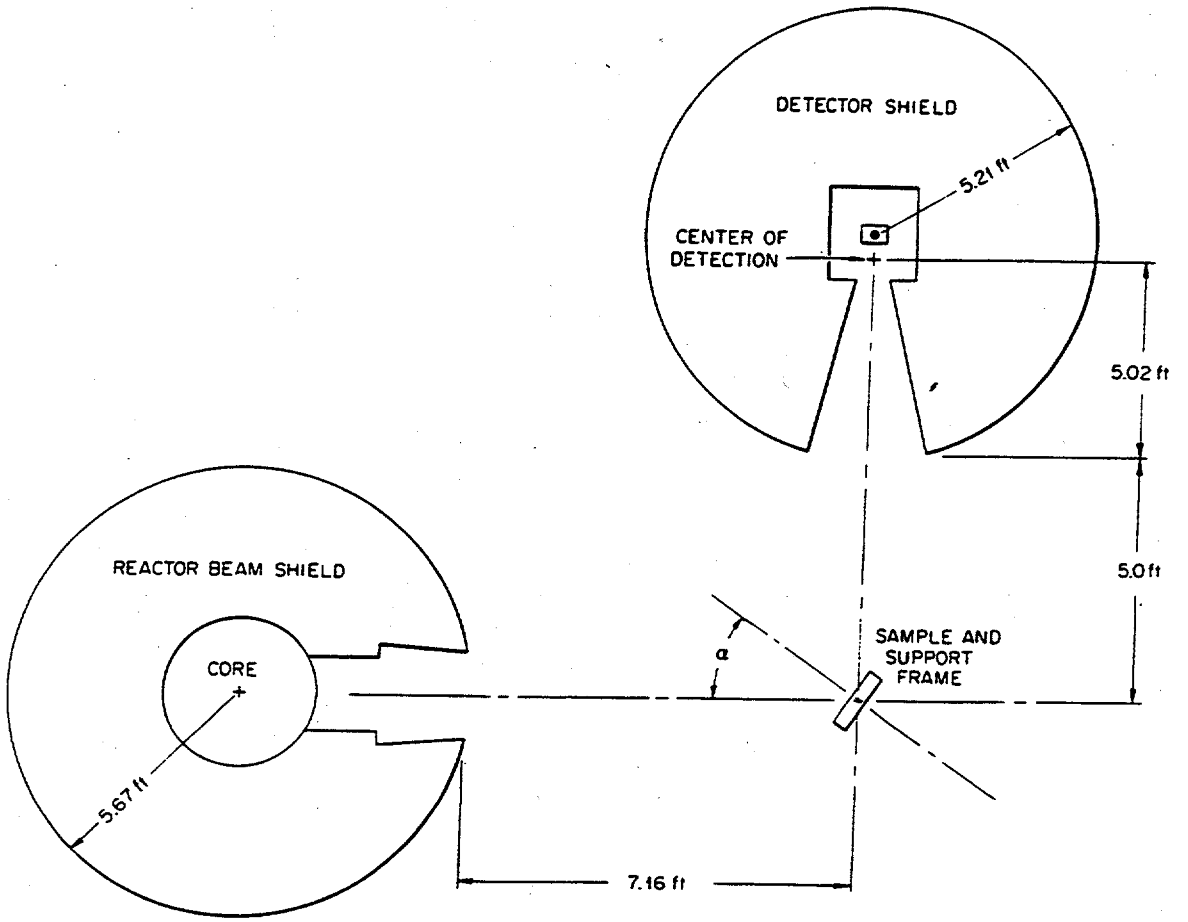

Fig. 11.1 TSF Neutron-Scattering Experimental Assembly.

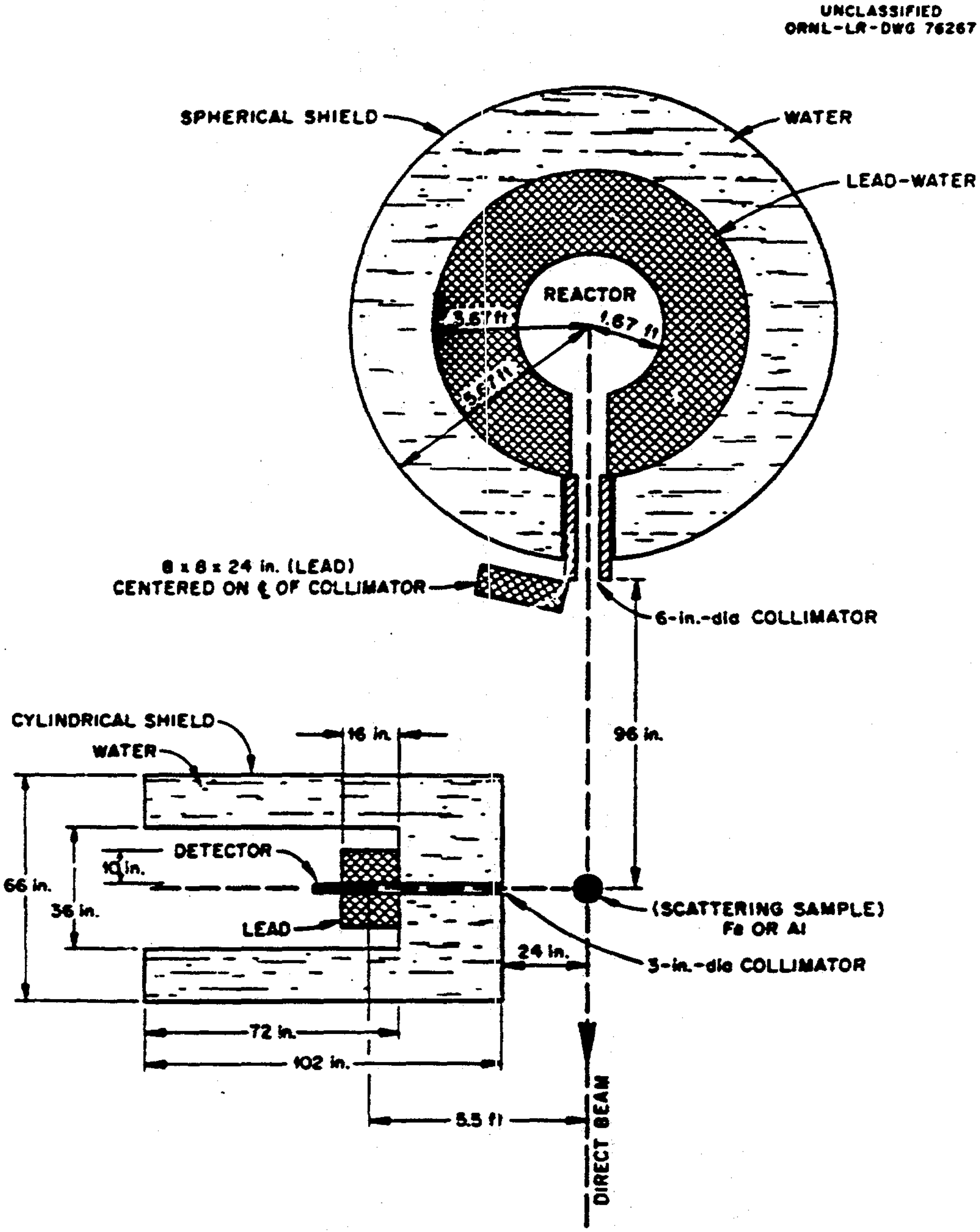

Fig. 11.2 Configuration of Preliminary SNAP Neutron-Scattering Experiments.

11.3 NEUTRON SCATTERING EXPERIMENT WITH SNAP CONTROL DRUM MOCKUP

With confidence from the results of the previous experiment discussed in Sect. 11.2, a more sophisticated experiment was jointly planned by the technical staff at AI and ORNL that would measure the intensities of the neutrons scattered from beryllium and nickel in geometries simulating control drums and reflector pieces of the hydride-shielded SNAP reactor system.[3] The experiments were performed to: (1) evaluate calculational methods developed at AI for predicting the contribution to the dose at the payload from neutrons scattered by the reflector control drums and associated parts, and (2) demonstrate that this type of measurement was feasible in the presence of the atmosphere which also contributes to the scattered dose. The TSR-II was placed inside an aluminum vessel beneath which was a 3-ft-thick volume of water, with the walls of the water container truncated to simulate a SNAP lithium hydride shadow shield. A traversing mechanism was mounted beneath the water vessel that supported a long cylinder which could be moved along the bottom. The collimator tube contained a neutron detector and the axis of this collimator coincided with the slope of the side of the water vessel as seen in Fig. 11.3. Attached to the side of the aluminum vessel surrounding the reactor were hinged samples of a beryllium drum or cylinder of nickel whose free end could be spaced at various locations away from the reactor. To perform the experiment, the detector collimator was submerged in the water of the reactor maintenance pool. Measurements were made as a function of the detector location beneath and outside the solid angle projected by the shadow shield for positions of the beryllium and nickel.

11.4 DOSE RATES FROM FAST NEUTRON SCATTERING IN CARBON, ALUMINUM, IRON, AND BERYLLIUM CYLINDERS

The work described here[5] was a continuation of the program initiated in Sect. 11.2, but was interrupted for measurements described in Sect. 11.3. Some changes were made, however, to both the reactor collimator and the detector shield for this series based on the previous findings. The gamma-ray flux from the reactor was attenuated by a 2-in.-thick insert placed in the reactor collimator next to the reactor. Extra lead was placed outside the reactor shield to reduce gamma-ray background. The detector shield was again the cylindrical tank, this time with a hemispherical head that was penetrated by a cone-shaped, air-filled collimator. The detector shield assembly was mounted on a three-wheeled carriage (Fig. 11.4) attached to a pivot point on the vertical centerline of the sample so that the assembly could be rotated about the sample while maintaining a constant distance. This arrangement provided the means for measuring the neutron flux as a function of scattering angle in the horizontal plane. In all, some 60 individual measurements were made for cylindrical samples of different diameters of aluminum, beryllium, carbon, and iron.

11.5 FAST-NEUTRON TRANSMISSION THROUGH AN "INFINITE" SLAB OF LITHIUM HYDRIDE

Phase II of the three-phase program in support of the SNAP program began with this experiment in late 1963.[6] It was aimed at developing shielding technology pertinent to the design of the LiH

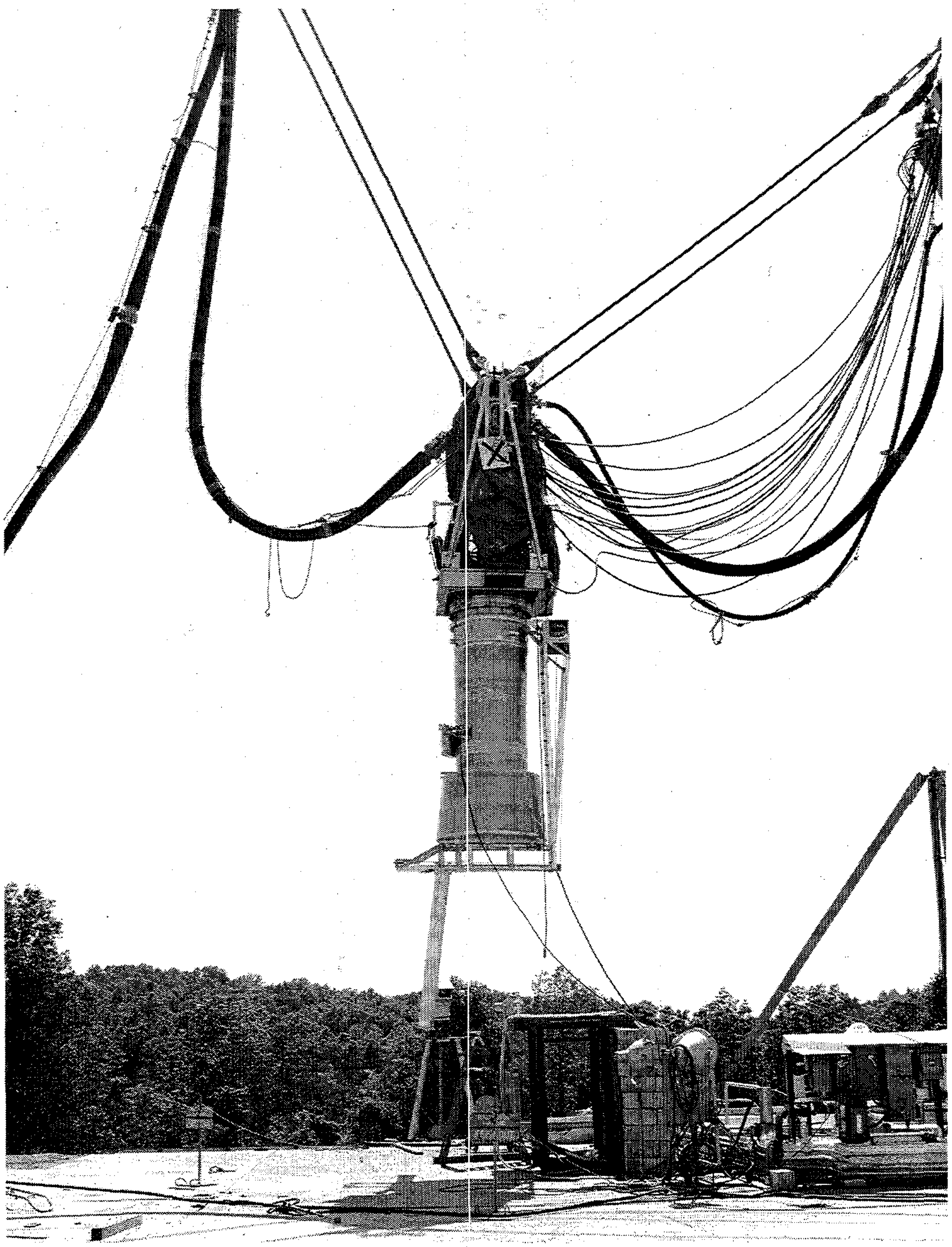

Fig. 11.3 Mockup for Measurement of Neutron Intensities Scattered from Beryllium in Simulated Control Drum Position.

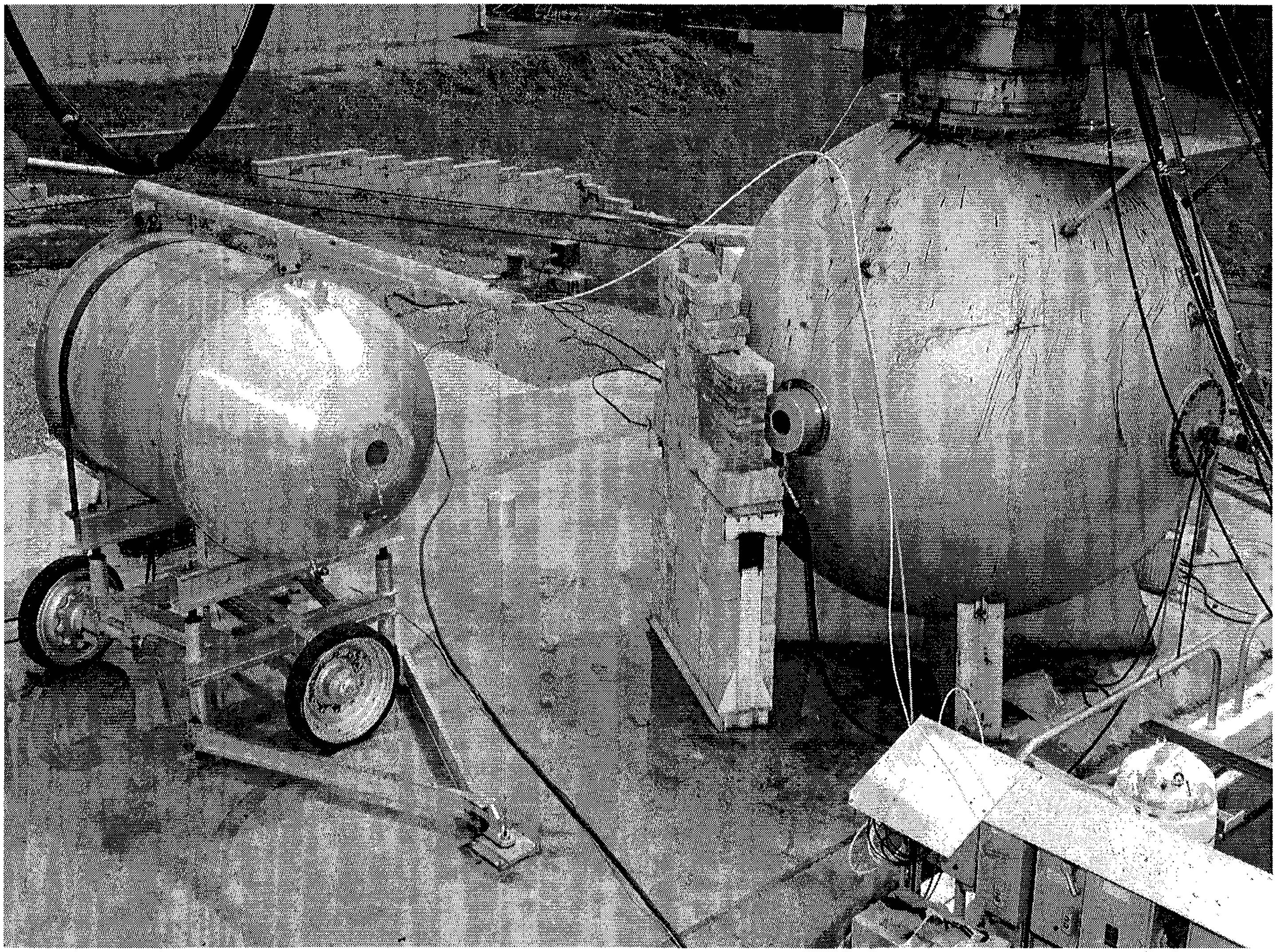

Fig. 11.4 Mockup for Measurement of neutron Scattering from Cylindrical Samples.

shadow shields including both measurements and calculations. Extensive measurements were made of the fast-neutron dose transmitted through thick lithium hydride (LiH) shields utilizing the TSR-II source and infinite-slab geometries. The thickness of the 5-ft by 5-ft LiH slabs varied from 1 to 30 inches. Concurrent calculations were made using a 05R Monte Carlo code to explore the advantages, if any, of using a conical-shell shield rather than a disk-shaped shield for space satellite power plants. The experimental values provided a basis for verification of the code.

11.6 FAST-NEUTRON DOSE RATE TRANSMITTED THROUGH A LITHIUM HYDRIDE SHIELD SURROUNDED BY AN IRON-OIL COLLAR SHIELD

Since measurements of the radiation transmitted through a LiH shadow shield would be done in air, it was considered vital that some form of auxiliary shielding, in the shape of a collar, be placed around it to minimize the contribution from air-scattered neutrons to the measured dose.[7] The shield design, based on Monte Carlo calculations, consisted of an oil-filled tank that contained movable iron slabs surrounding the LiH shield as shown in Fig. 11.5. The measurements to be performed were in support of those calculations. However, before these measurements could be made, it was necessary that measurements be made to check on possible cracks within the LiH or between it and its container. Thus the study was done in two phases.

The first phase consisted of measurements to check for possible voids between the LiH and the walls of its container.[8] For this, the LiH shield was centered in an oil-filled cylindrical container through which only the larger end of the LiH protruded. A tightly collimated beam of neutrons (approximately 1.25 inches in diameter) impinged on the smaller end of the shield as shown in Fig. 11.6, and, while rotating the LiH and oil container simultaneously, measurements were made with a tightly collimated detector beyond the larger end of the LiH shield. A picture of the mockup is shown in Fig. 11.7. Throughout the lateral and rotational movement of the shield, the detector axis and the reactor collimator axis remained coincident. Though the LiH was found to be solid, it was necessary to fill separations observed between the LiH and its container with mineral oil.

In Phase II, measurements were made of the transmitted neutron dose rates for different iron slab thicknesses and their locations within the oil around the LiH to optimize the shield design.

11.7 NEUTRON SPECTRA FROM TSF-SNAP REACTOR

A special collimator designed by E. A. Straker[9] in 1966 for the core-mapping experiments using the SNAP reactor as a source is shown in Fig. 11.8. The design met several criteria, one of which was that the direct intensity transmitted by the collimator be relatively uniform over the detector surface area while keeping the intensity scattered from the collimator walls to less than 5% of the radiation seen by the detector. The collimator was used in conjunction with the NE-213 organic scintillator to measure the neutron energy distribution above 0.8 MeV emerging from the bottom of the reactor. A 1/E detector (Fig. 11.9) developed by T. V. Blosser[10] for use in another program was used to measure the low-energy neutron spectrum up to about 10 keV.

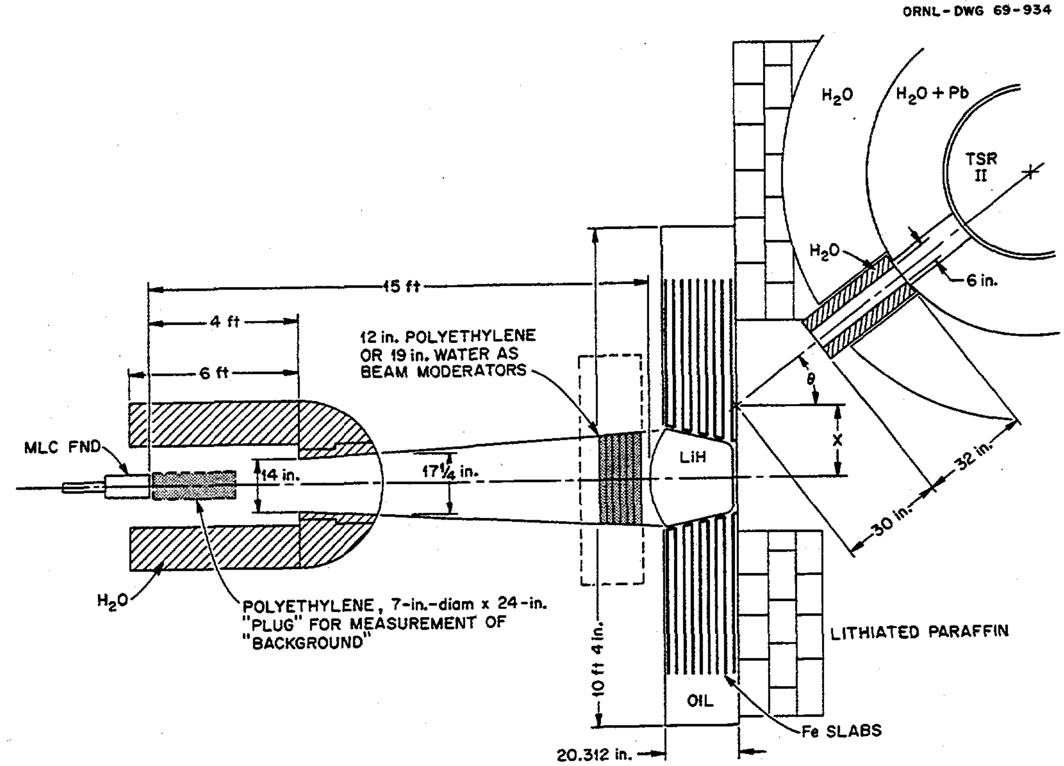

Fig. 11.5 Schematic for Measurement of the Neutron Leakage through the LiH Collar Shield Configuration.

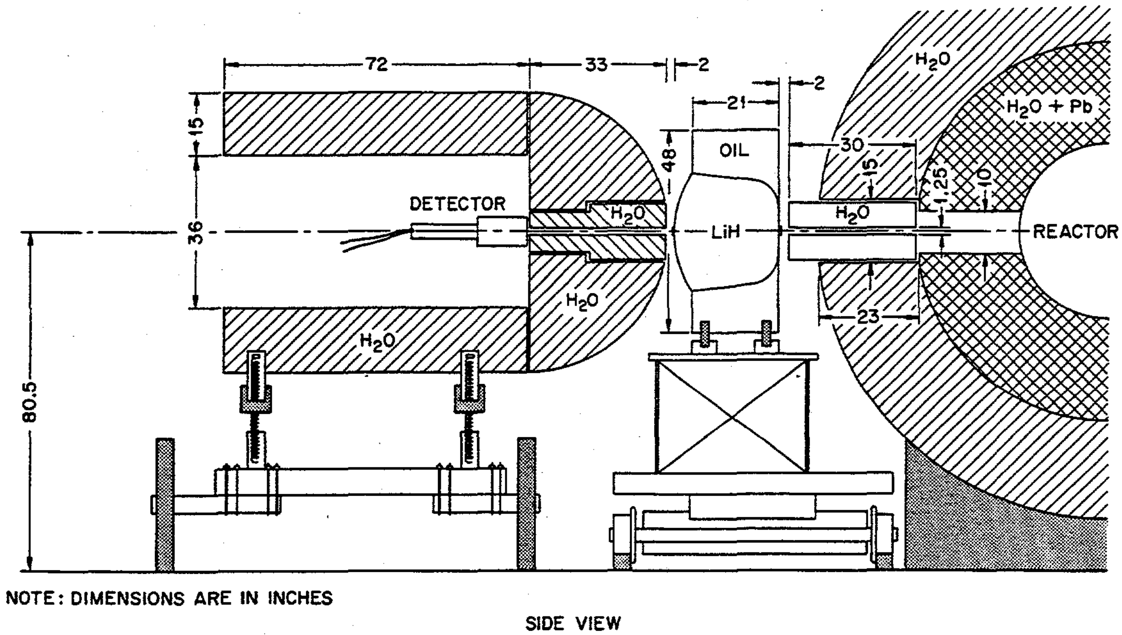

Fig. 11.6 Configuration for Mapping of the Neutron Leakage through the LiH Shield.

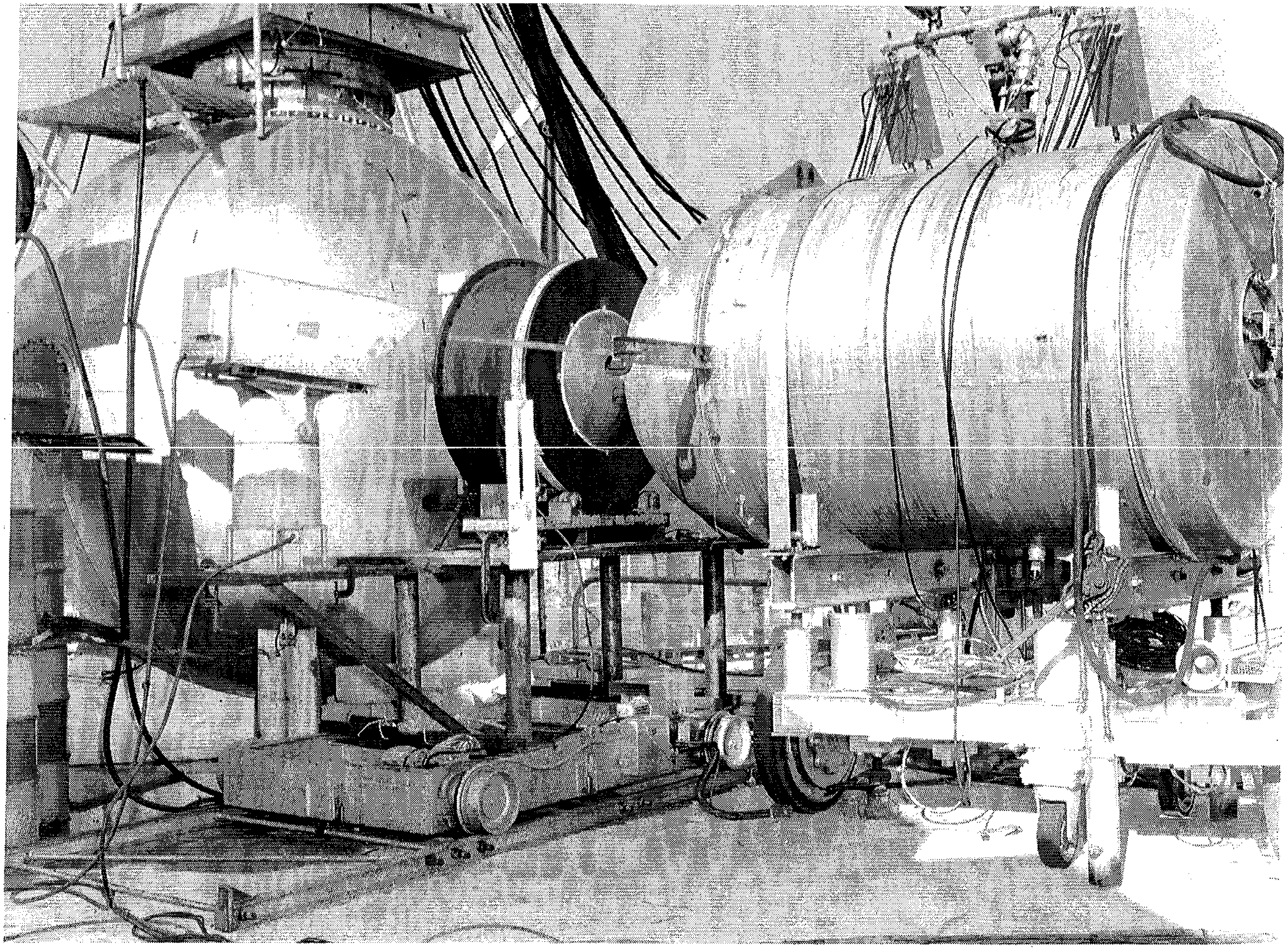

Fig. 11.7 Experimental Arrangement of the LiH Shield with respect to the TSR-II Reactor and the Detector Shield.

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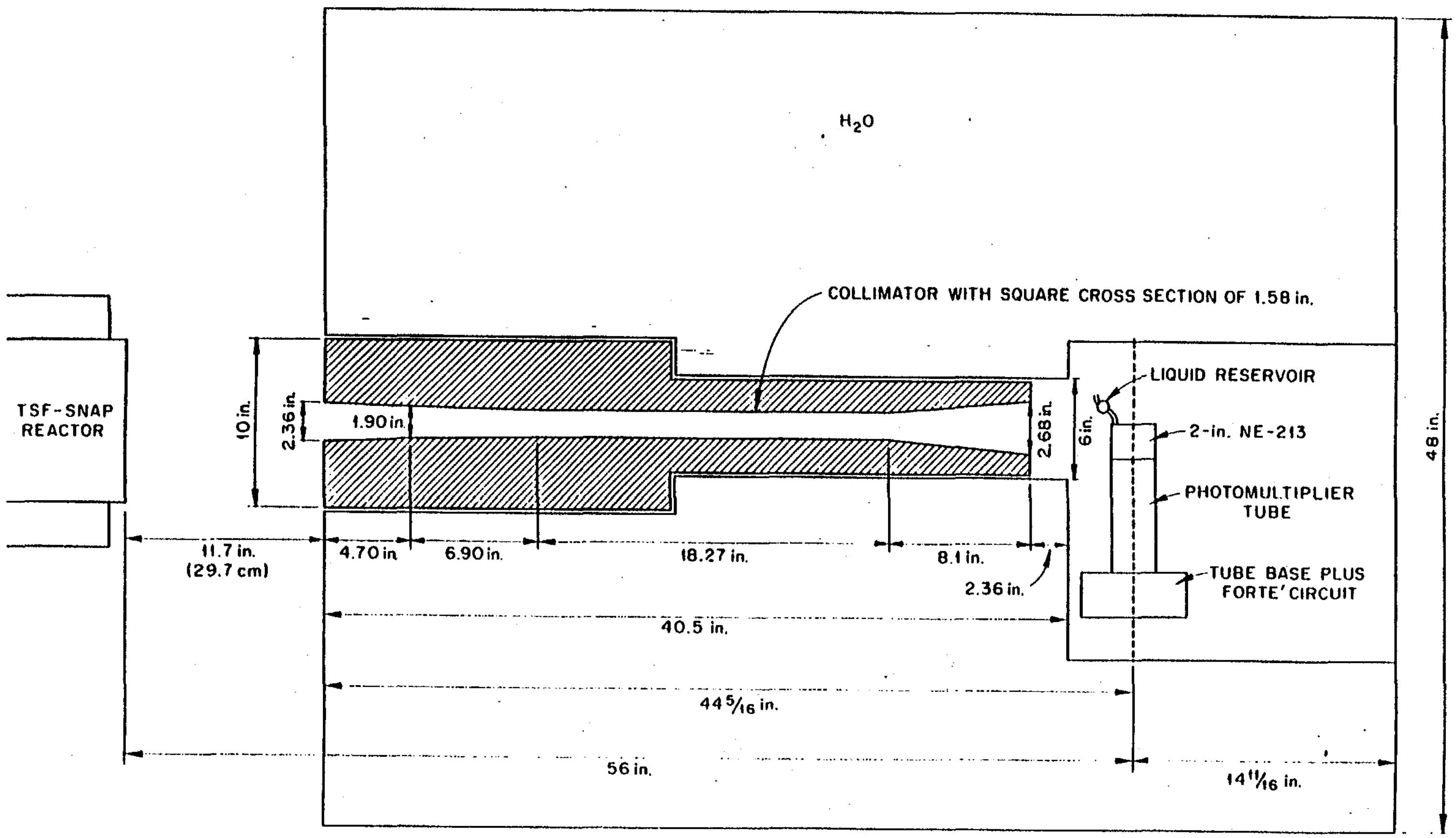

Fig. 11.8 Schematic Diagram of Geometry for Measurement of the Neutron Leakage Spectrum from the TSF-SNAP Reactor.

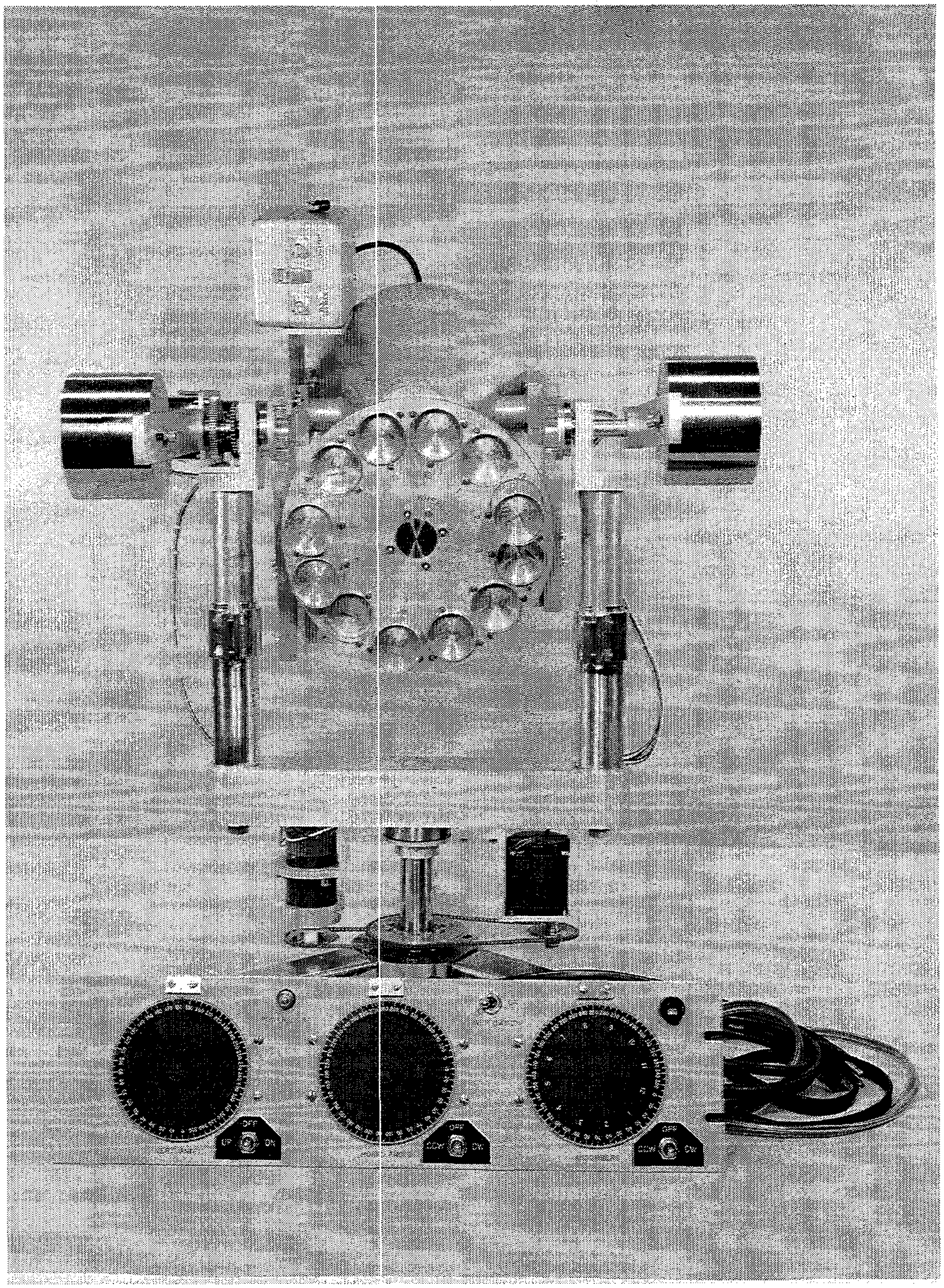

Fig. 11.9 The Epithermal-Neutron Spectrometer.

The NE-213 scintillator was placed inside a lead-water shield containing the above collimator, the shield being mounted on a support mechanism that both rotated the detector in a vertical plane about the horizontal axis of the detector and moved it in a line perpendicular to the reactor's vertical axis (Fig. 11.10). These measurements were repeated during the late-1968 to early-1969 time span, this time providing much more detail by including dose profiles beneath the reactor to check the accuracy of the predicted spatial distribution of the neutron environment.

11.8 NEUTRON SPECTRAL MEASUREMENTS THROUGH TYPICAL SNAP SHIELDING MATERIALS

Measurements were made of the neutron spectra above 1 MeV transmitted through typical SNAP shielding materials placed directly beneath the SNAP reactor when placed over the reactor pool.[11] The scintillator was placed in the same lead-water shield described above (Sect. 11.7) and placed 28 ft below the reactor near the bottom of the drained reactor handling pool. The pool was covered with 2 ft of concrete through which a 5-ft-diam iris allowed the detector to look at the bottom of the SNAP reactor as shown in Fig. 11.11. Additional shielding was provided as necessary by placement of water tanks, lithiated paraffin bricks, and poly or lead in and around the iris to improve the background. Shielding samples for which neutron spectral measurements were obtained included lead, ^{238}U, tungsten powder, hevimet, lithium hydride, and laminated slabs of lead, lithium hydride, and ^{238}U.

11.9 GAMMA-RAY SPECTRAL MEASUREMENTS THROUGH SHIELDS OF SINGLE MATERIALS AND LAMINATES

For the SNAP program during 1970, the effort was centered on measurement of the gamma-ray spectra transmitted by shields of single materials and shields with laminates of heavy metals and lithium hydride. For this work, slabs of material were placed beneath the SNAP reactor and measurements made with a 5-in. NaI crystal about 28 ft beneath the slabs using the same mockup arrangement as previously shown in Fig. 11.11. Materials studied were slabs of lithium hydride, lead in thicknesses from 1.50 to 6 in., ^{238}U in thicknesses from 1.50 to 4.50 in., and tungsten powder in thicknesses up to 6 in. of natural tungsten. For the laminated slabs, various combinations involved different thicknesses of lithium hydride, lead, and uranium. To aid in interpolating the data, measurements were also made for thin metal samples that included tantalum and Hastelloy. Results indicated there were still large discrepancies between measurements and calculations for thick lead and uranium shields. These measurements were the last series scheduled for the SNAP program, but some repeats were made later in 1971 to clarify differences between measurements and calculations.

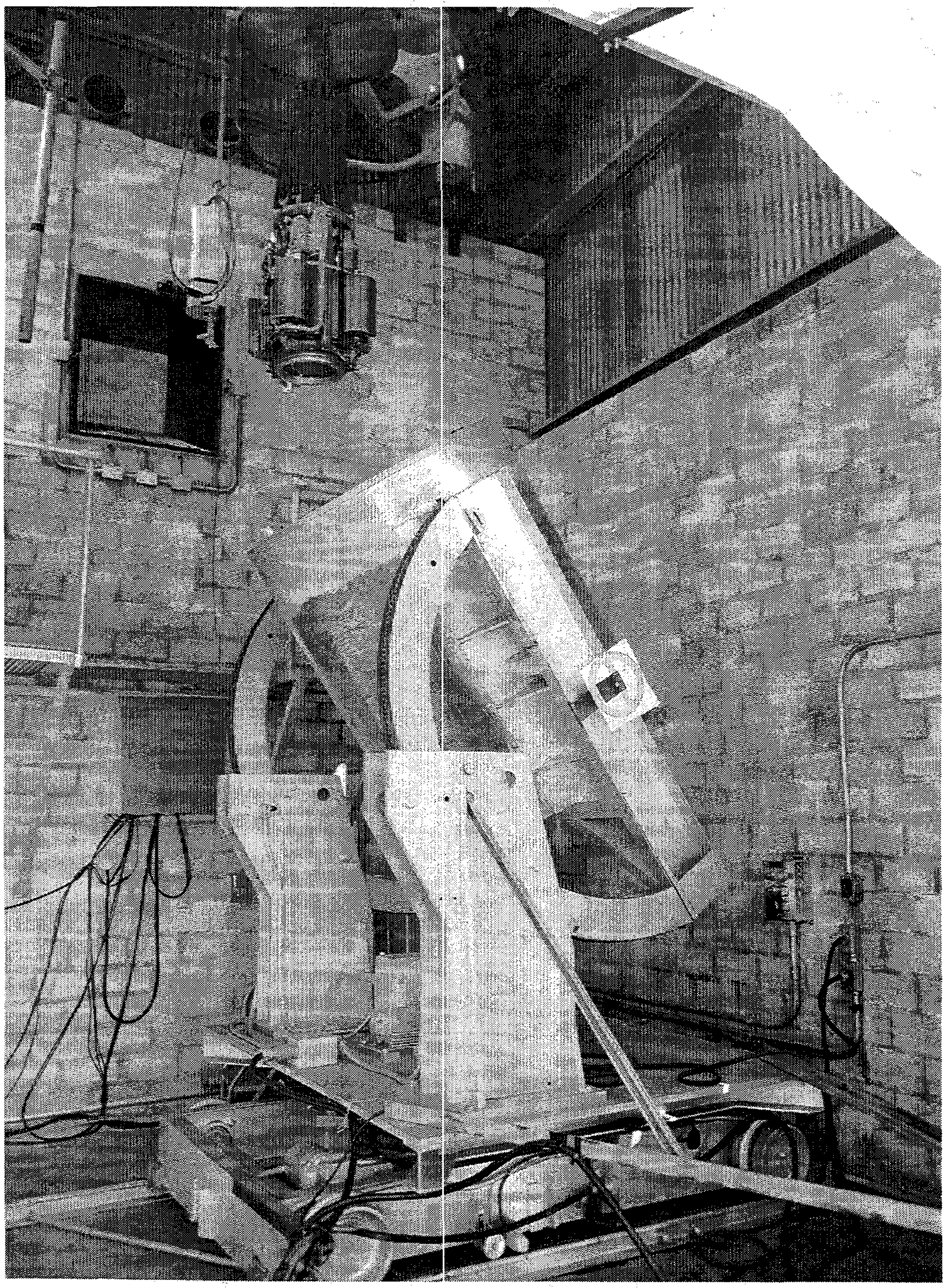

Fig. 11.10 Photograph of Experimental Arrangement.

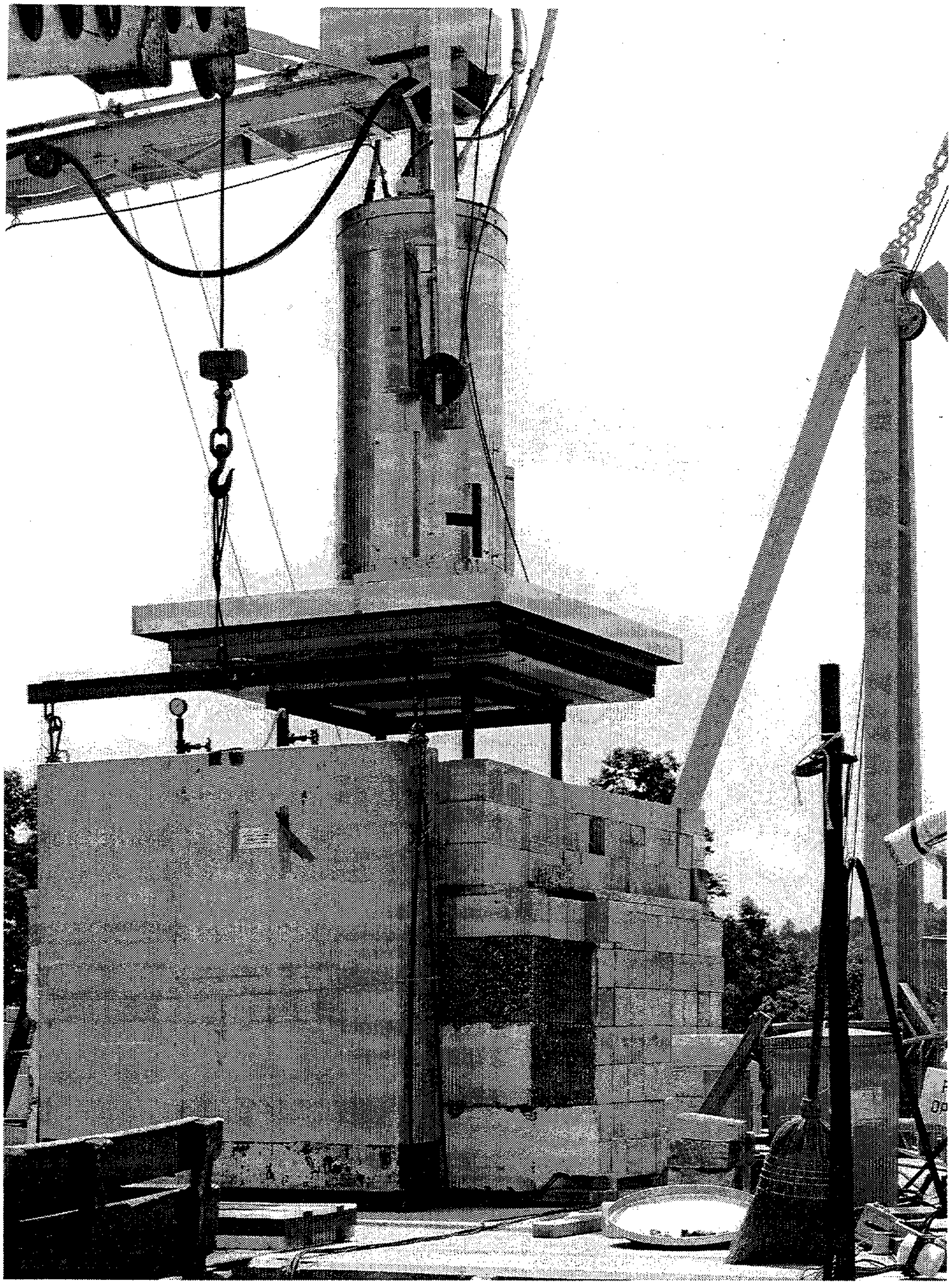

Fig. 11.11 Typical Mockup for Measurement of Radiation Transmission through Shield Samples using SNAP Reactor as Source.

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12.0 LIQUID METAL FAST BREEDER REACTOR PROGRAM

12.1 INTRODUCTION

Once the U.S. decided that the way to meet the nation's future energy needs was to develop the Liquid Metal Fast Breeder Reactor (LMFBR), it was proposed that a reactor plant should be built to test the concepts using liquid sodium as a coolant.[1] The resultant facility, the Fast Flux Test Facility (FFTF) was built at Richland, Washington.

The lead designer responsible for its construction was Westinghouse Electric Corporation's Advanced Reactor Division (WARD). Many problems presented themselves, and as the design progressed, WARD recognized the inadequacies in existing shielding analysis methods. Realizing the tremendous needs in that field, WARD requested through the Atomic Energy Commission's Reactor Development and Technology Division, in 1969, that the shielding analysis group at ORNL assist them in evaluating their shield designs for the FFTF reactor. At that time, the analysis group had only three major codes for computing radiation transport, all of which had been developed under the auspices of other shielding programs. A close look at the problems presented to them by WARD resulted in a realization that further development of the codes as well as cross section data would be necessary. Thus a methods development effort was initiated, which included the need for integral experiments at the TSF, for verification and validation of these codes.

The FFTF effort was mainly concentrated in three areas: (1) the evaluation and development of calculational techniques for predicting the fluence of neutrons penetrating a sodium and stainless steel shield protecting the grid plate, (2) the penetration of neutrons through a solid iron shield covering the top of the reactor tank, and (3) the prediction of the neutron spectra arriving at the bottom surface of the top-head iron shield. The experiments were essentially of two types: prototypic mockups that represented specific areas of the FFTF, and basic experiments that provided data for analysis of neutron penetration through important materials used in the FFTF design. Experiments of the first type, prototypic mockups, included a series of measurements to 1) determine the importance of neutron streaming through coolant passages found in reactor grid plate shields, 2) study the neutron streaming through inlet and outlet, cooling pipe areas that penetrated a wall of the heat exchanger vault, and 3) evaluate the neutron transmission through planar and cylindrical annular slits to determine the effectiveness of the rotating top-head shield design. The basic type experiments included a study of the effectiveness of inconel as a neutron shield, as compared to stainless steel, and measurements of the absolute fission rates for ^{239}Pu and ^{235}U during the Fast Test Reactor (FTR) and Demonstrating Plant Reactor first fission experiment. Thus, from 1970-1975, when the FFTF design was frozen, a large portion of the TSF time was devoted to the FFTF program.

Throughout this period, however, other integral type experiments were performed that were within the LMFBR program, but had no direct connection with the FFTF work.[2] Measurements were made for GE to demonstrate that the neutron transport through shields of large diameter stainless steel on boron carbide-filled rods could be adequately described using a homogeneous model. Benchmark studies were made that served as standards for calculation of the neutron penetration through such materials as sodium, iron, and stainless steel. Gamma-ray studies were initiated to measure both the energy deposition within the mockups using thermoluminescent detectors (TLDs) and the energy spectrum as a function of mockup thickness. Total cross section measurements were made in the 5 keV to 1 MeV region for samples of iron, nickel, oxygen, chromium, sodium, and uranium using the hydrogen-filled proton recoil spectrometer.

During the FY74 period, work for the FFTF program was phasing out and support was initiated for the Clinch River Breeder Reactor (CRBR) program, beginning with experiments studying the adequacy of calculational techniques for predicting the counting rate of the Low Level Flux Monitor (LLFM). Measurements were made throughout the mockup of a lower axial shield region designed by Atomics International (AI) to provide data for evaluation of the calculational methods being used to predict streaming through the control rod penetrations. Earlier TLD work for the FFTF program was extended to measure energy deposition rates within shield configurations typical of the LMFBR. Measurements were made for comparison with calculations of the ORNL designed CRBR Upper Axial Shield, a shield that consisted of varying amounts of stainless steel followed by sodium and iron. Neutron penetration studies were made behind a radial shield design by Westinghouse of stainless steel and inconel. Neutron transmission studies were made for a prototypic CRBR coolant pipe chaseway.

ORNL's Shielding Analysis Section, as part of their support effort in the FFTF program, performed an analysis of a shield designed by Westinghouse to prevent neutron streaming in the area of the top-head cavity. Never before in the history of TSF's shielding work had it been possible for TSF personnel to make measurements for an operating reactor system that would demonstrate the validity of such calculations. Such an opportunity was provided during the startups of the FFTF reactor, and measurements were made to test the effectiveness of that design and compare the measurements with the calculations.

With the advent of the smaller Liquid-Metal-Cooled Reactor (LMR) concept, a principal concern of its shield designers was the availability of a shielding material that would reduce the size and mass of the reactor's shield while maintaining the neutron flux within design limits. One of the materials considered was boron carbide (B_4C). Studies were made at the TSF, comparing its shielding properties to that of stainless steel in an Alternate Shielding Material Experiment.

12.2 NEUTRON TRANSPORT IN IRON AND SODIUM MOCKUPS OF REPRESENTATIVE GRID PLATE SHIELDS

The solution to a number of design problems pertinent to the LMFBR program and the FFTF design in particular required the prediction of neutron energy fluence at a number of points external to the core.[3] One such problem was to minimize radiation damage to the grid plate which supported the fuel elements. A proposed shield design called for small diameter SS rods located at the lower end of the fuel rods, a configuration that would allow the sodium coolant to pass through the system. Prediction of radiation transport in shields containing large arrays of cylindrical rods was a severe challenge to the most sophisticated discrete ordinates or Monte Carlo methods at that time. To provide guidelines, a program of experimental measurements were made of the neutron spectra transmitted through shields of iron, sodium, and iron pins in sodium. The neutron source was provided by the SNAP reactor in a typical configuration, shown in Fig. 12.1, and measurements were made directly below and at 26 ft beneath the shield sample.

The shield samples were all cylindrical, having a nominal diameter of 18 in. Shield thicknesses of 8, 16, and 24 in. were fabricated for iron and iron pins in sodium. The 0.375-in.-diam pins were held in a square array, the pitch being 0.519 in. A 16-in.-high can of sodium completed

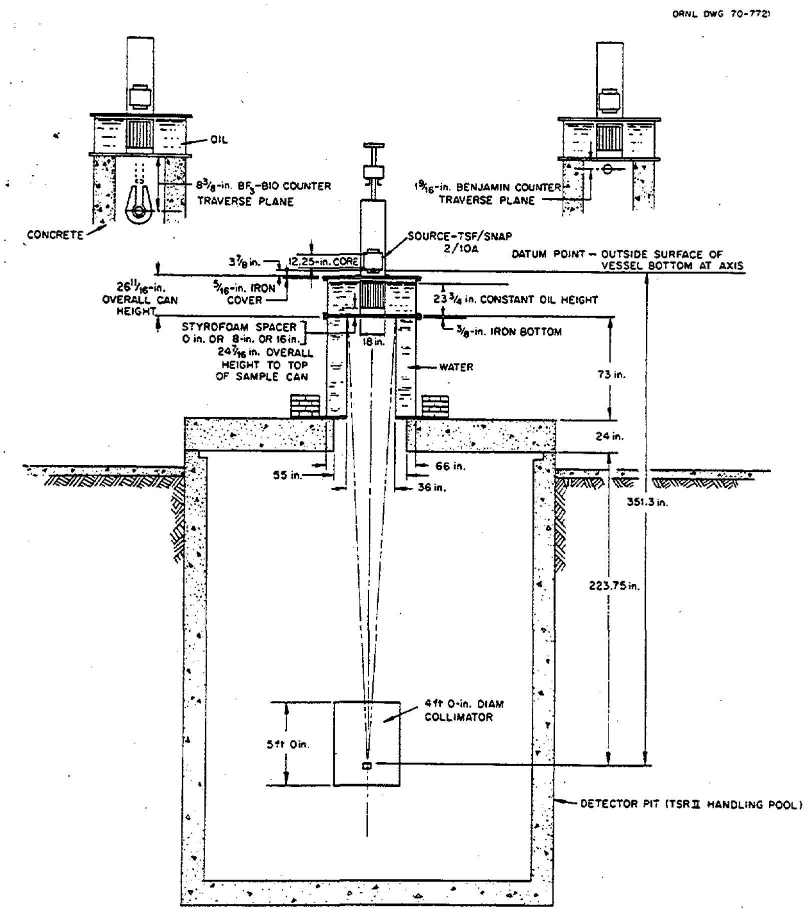

Fig. 12.1 Experimental Configuration for FFTF Grid Plate Shield Measurements.

the group of samples tested. During the measurements, each sample was immersed in a cylindrical tank filled with oil that was 24 in. high. Lead bricks were placed around the samples to reduce the gamma-ray background. The height difference between sample and tank was filled with styrofoam.

12.3 NEUTRON TRANSMISSION THROUGH PLANAR AND CYLINDRICAL ANNULAR SLITS IN IRON SHIELDS

The newly fabricated Benjamin-type hydrogen-filled detectors were used for the first time in the series of experiments to measure the low energy part of the neutron spectrum from about 50 keV to 1.5 MeV. During 1970, an experiment was performed at the TSF to measure the streaming of neutrons through planar and cylindrical annular slits in thick iron shields.[4,5] The experiment was part of a study to determine the shielding effectiveness of the rotating top plug design in the reactor pressure vessel head of the FFTF. The top plugs consisted of a nominal 25-in.-thick thermal shield of alternate layers of steel and inconel topped by about a 22-in.-thick reactor vessel head constructed of SA-508 steel. Though it was expected that this thickness would be adequate, it was recognized that with the numerous penetrations through the shield to accommodate components of the reactor system, the introduced gaps could serve as significant radiation streaming paths. Specifically, the annular gaps around the rotating plugs comprising parts of the three in-vessel handling machines were considered to be of greatest concern.

For the TSF measurements to be pertinent to these problems, the energy spectrum of the neutrons incident on the experimental configuration had to be similar to the energy spectrum of the neutrons incident on the underside of the FTR thermal and top-head shields. To accomplish this, a spectrum modifier of 8-in. of Fe followed by 60-in. of Na was placed between the reactor source and the iron containing the slits to be studied. That spectrum modifier arrangement is shown in Fig. 12.2. The study using planer slit mockups was mainly concerned with the neutron transmission through the voids as a function of slit depth, width, and number of slit offsets, which in this experiment, were limited to two as seen in Fig. 12.3. The primary interest in the study of mockups having annular slits was the change in neutron penetration when a single offset was introduced midway through the shield. The offset was chosen to occur at the separation point between the thermal shield mockup and the vessel head mockup as shown in Fig. 12.4. Due to the limited breadth of the reactor beam source, it was necessary to limit the diameter of the annular slit to 22.5 in. The nickel placed between the iron slabs was included in the thermal shield to more closely model stainless steel in the real assembly. A diagram of the mockup is given in Fig. 12.5.

12.4 EXPERIMENTAL INVESTIGATION OF NEUTRON STREAMING THROUGH A PROTOTYPIC LOWER AXIAL SHIELD FOR FFTF

A series of measurements were made at the TSF to determine the relative importance of neutron streaming through various geometric arrangements of coolant passages found in a nearly prototypic reactor grid plate shield for the FFTF.[6,7] The passages were necessary to allow flow of the sodium coolant through the fuel elements. The design placed the FTR core support grid plate close to the core requiring that some type of shielding would be required to prevent excessive radiation damage to it. For this work, the TSF-SNAP leakage spectrum was modified to represent that incident on the FTR grid plate by filtering it through 12 inches of SS, 1-in. polyethylene, and 0.25-in. boral before impinging on the grid plate mockup. A typical experimental mockup is shown in Fig. 12.6.

Design of the lower axial shield insert called for at least one offset, as seen in Fig. 12.7, with the holes for coolant flow in the second section rotated 90° following a plenum space separating the two sections. The voids within the iron insert plug sections were filled with sodium, while an aluminum mesh was used to represent sodium in the collar. Three removable plugs (lower axial shields), supplied by WADCO Corporation, a subsidiary of Westinghouse Electric Corporation, were used in the experiment.

12.5 BENCHMARK EXPERIMENT FOR NEUTRON TRANSPORT IN THICK SODIUM

During the summer of 1971, a task was undertaken to supply the TSF with sufficient sodium thickness to perform an in-depth measurement of the neutron transport through sodium.[8] Results from earlier experiments conducted for this purpose had somewhat limited usefulness in that the diameter of the sodium samples was much smaller than the total thickness studied. The design of the FFTF

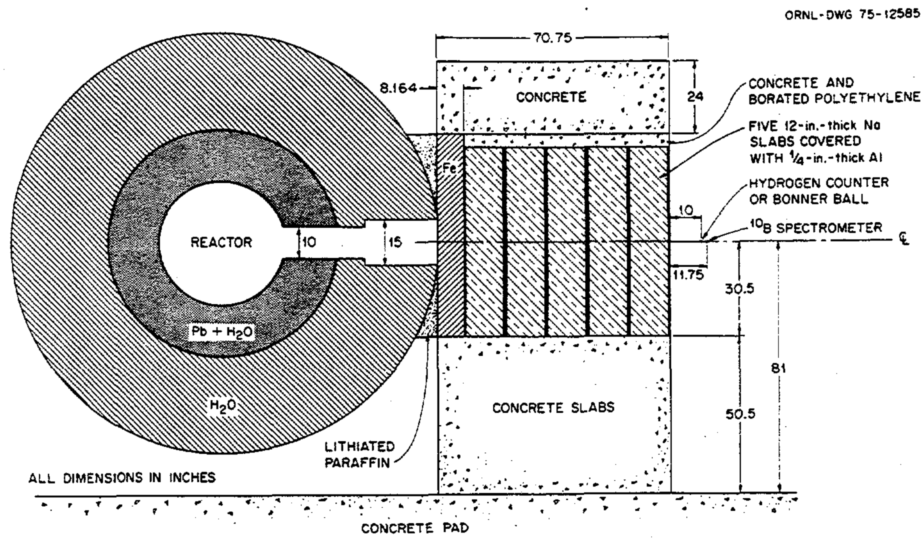

Fig. 12.2 TSR-II Fe-Na Spectrum Modifier Arrangement (Vertical Cross Section).

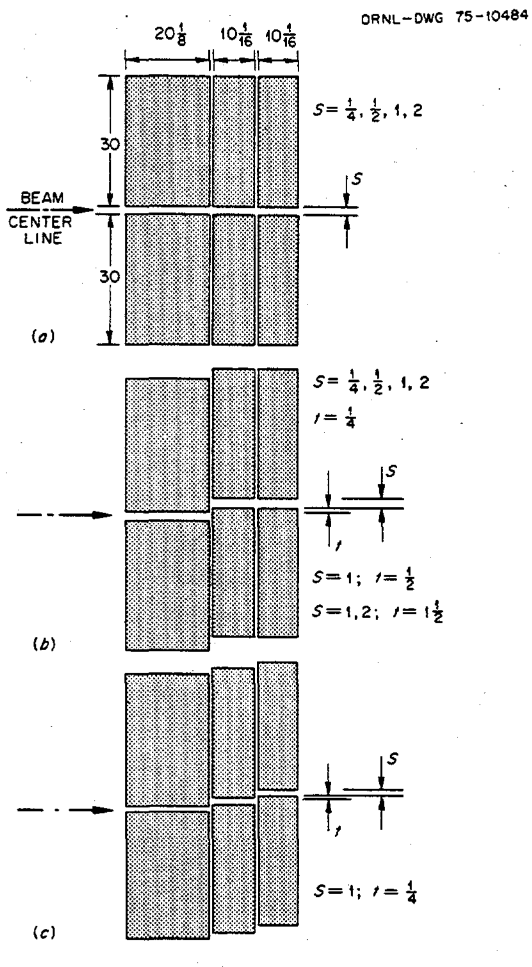

Fig. 12.3 Horizontal Cross Sections of Plane-Slit Configurations:
(a) Straight Slit; (b) Slit with Single Offset; and (c) Slit with Double Offset.
Dimensions in inches.

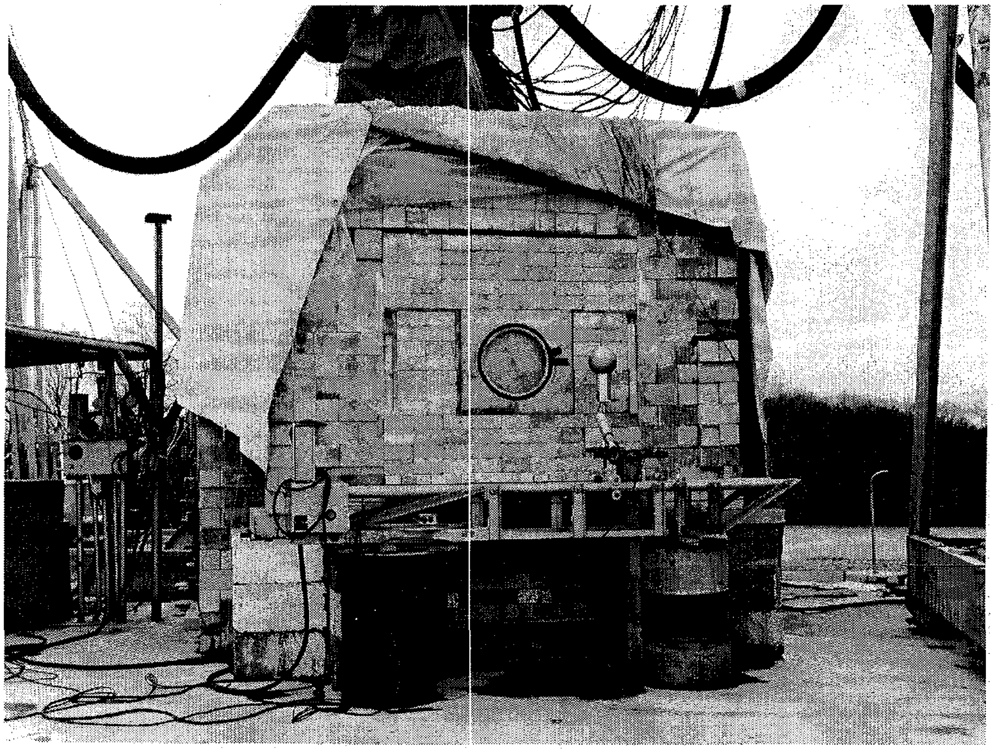

Fig. 12.4 Annualar Slit with Single Offset.
(Note: Left side of assembly was adjacent to Spectrum Modifier.)

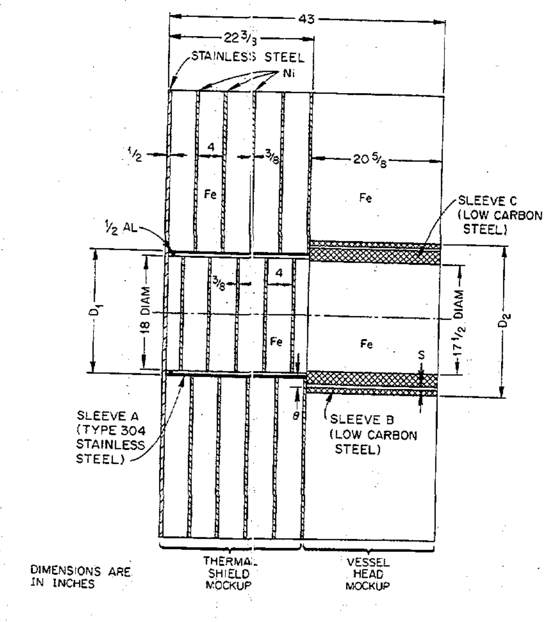

Fig. 12.5 Front Face of Annular-Slit Assembly. Instrument Located at Lower Right of Assembly is a Bonner Ball Detector. Brinks Around Slit are Lithiated Paraffin.

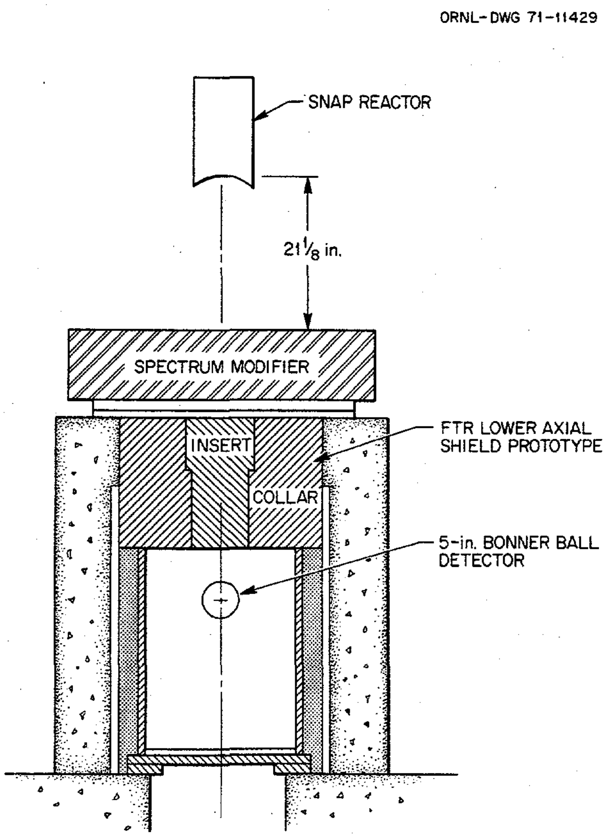

Fig. 12.6 ORNL TSF Prototype Grid-Plate Shield Experiment Configuration.

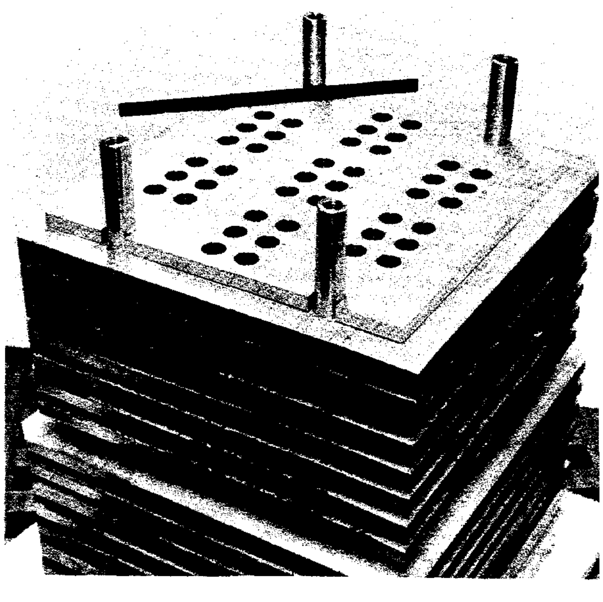

Fig. 12.7 Array of Filler Plates for Grid-Plate
Shield Mockup.

called for a pool of sodium above the core to be 15 ft thick and 20 ft in diameter. To do an experiment for verifying calculations, the mockup should contain a sodium volume of similar dimensions to that of the pool. The thickness of 15 ft was obtained by filling a combination of four large, 11-ft-diam, cylindrical tanks that were 2.5 and 5 ft thick.

The selected diameter was adequate since the diameter of the reactor source was only 15 in. These tanks were made of aluminum, and supported by a concrete saddle that surrounded the cylindrical surface of the tank. The tanks were filled at ORNL by TSF personnel during a hot summer period. Fig. 12.8 shows a photograph of all four sodium tanks after their arrival at the TSF, being closely checked by J. J. Manning of the TSF staff. Detailed measurements were made as a function of sodium thickness using a collimated beam of TSR-II neutrons as a source.

12.6 A BENCHMARK EXPERIMENT FOR NEUTRON TRANSPORT IN IRON AND STAINLESS STEEL

An experiment was performed at the TSF in early 1972 to provide verification of the accuracy of available neutron cross sections used in transport calculations of deep neutron penetrations through a maximum of 3-ft iron and 18-in. SS.[9] Previous measurements were for relatively small thicknesses, but since the top shield for the FTR was relatively thick, accurate experimental results were necessary to give validity to these calculations. For the iron measurements, the TSR-II beam was highly collimated, as shown in Fig. 12.9, thus limiting the effects of side leakage even for the thick sample. The measurements with SS in the beam were done later that same year after the concrete facing was added to the reactor's spherical shield (mentioned earlier in Sect. 5.7.).

12.7 ORNL TSF PIPE CHASEWAY NEUTRON STREAMING EXPERIMENT - PHASE I

The reactor design of the FFTF included inlet and outlet pipes that penetrated the 4-ft-thick reactor cavity walls. These pipes were located above the height of the core such that neutrons entering the reactor cavity at an angle would have to undergo at least one scattering event to stream out of the penetration. Neutron paths to the heat exchanger were possible through the sodium, the insulation around the pipes, and the voids around the insulation. This experiment[10] tested the neutron streaming along these paths in a typical mockup, as shown in Fig. 12.10. Three configurations were studied: (1) an empty pipe chase, (2) a 16-in.-diam sodium-filled pipe, and (3) the sodium-filled pipe surrounded by insulation. Again the results were obtained to validate the analytical methods to be used in predicting the flux levels in the FFTF heat exchanger vault and piping.

12.8 TSF PIPE CHASEWAY STREAMING EXPERIMENT - PHASE II

Experiments were performed at the TSF in support of the analysis of the neutron transport through the sodium coolant pipes and in a heat transport system (HTS) pipeway of the FFTF. Sodium inlet and outlet pipes penetrate a 4-ft-thick concrete wall surrounding the reactor cavity, pass through

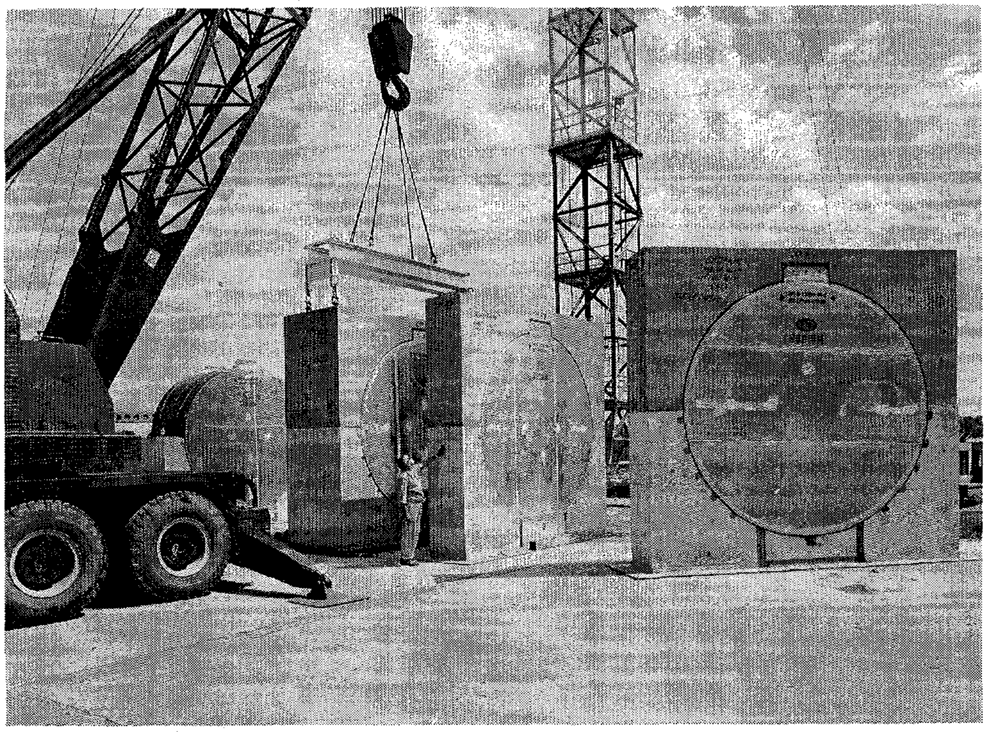

Fig. 12.8 Photograph of Sodium Tanks and Concrete Collars.

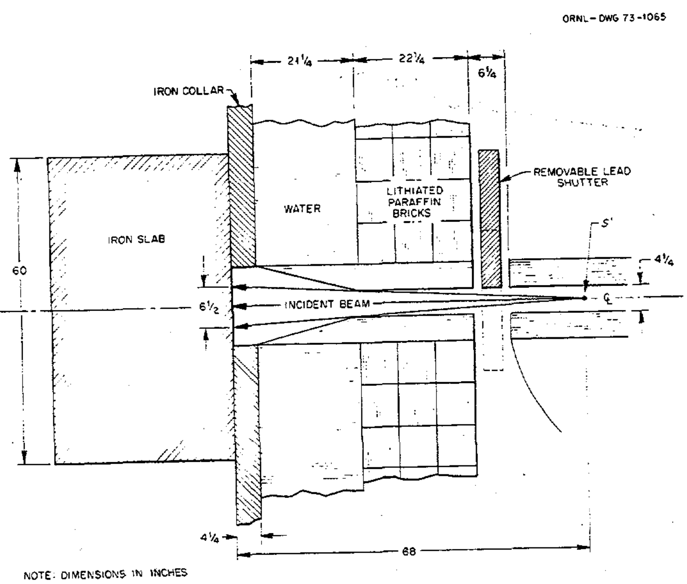

Fig. 12.9 Experimental Configuration for the 4.25-in.-diam Collimator with the 3-ft-thick Iron Slab in place. This Collimator was used for all the measurements except those made behind 18 in. of Stainless Steel.

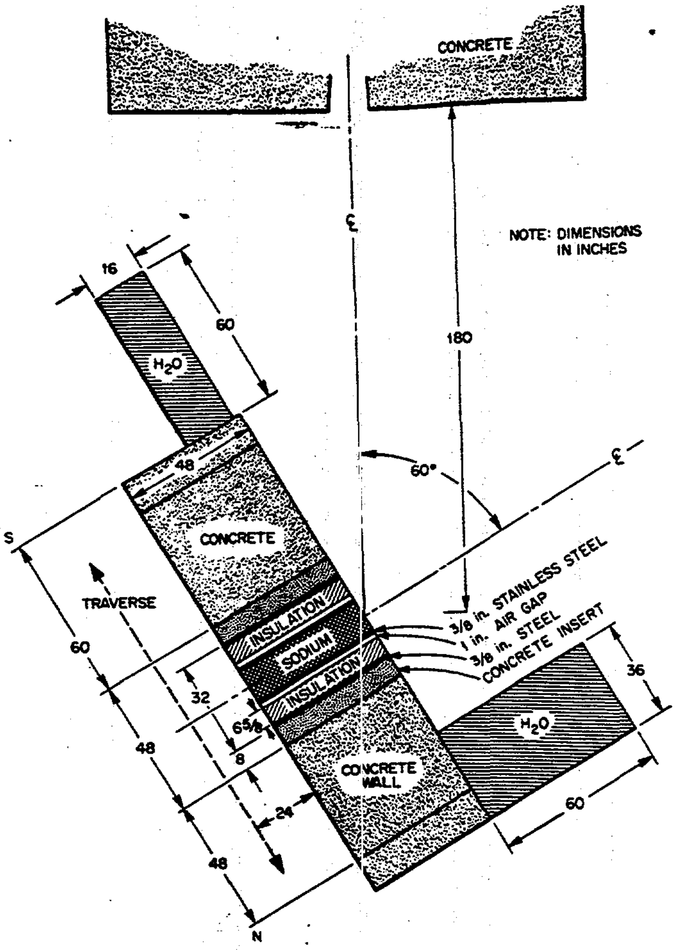

Fig. 12.10 Geometry for the First Phase of the TSF Pipe Chase Streaming Experiment.

a pipeway, and then enter heat exchanger vaults through a concrete shield. Due to the location of the pipe penetration above the level of the reactor core, the neutrons entered the pipeway at an angle. In the Phase I experiment described in Sect. 12.7, a steel-lined pipe-chase mockup was placed in the beam at an angle to measure the resultant radiation scattered beyond the chaseway. Phase II,[11,12] of the HTS of the FFTF was modeled by a concrete-walled plenum lined with iron. Radiation from the TSR-II entered through a hole in one wall, was scattered with the plenum, with some fraction of the scattered neutrons exiting through a pipe-chase mockup in the opposite wall. Measurements were made of the flux distribution in the cavity and beyond the pipe chase with and without the insulation around the pipe. A schematic of the configuration is shown in Fig. 12.11, and an outside view of the mockup shown in Fig. 12.12.

12.9 GENERAL ELECTRIC AND WESTINGHOUSE RADIAL AND AXIAL SHIELD MEASUREMENTS

A fast reactor experimental shielding program for the General Electric (GE) demonstration plant[13] was initiated at TSF in late 1972 to compare the relative neutron shielding effectiveness of slabs containing B_4C filled SS tubes with slabs containing the same amount of B_4C in a homogeneous mixture. Several slabs of each tube diameter, 1, 2, or 5.25 in., were provided, with the space between

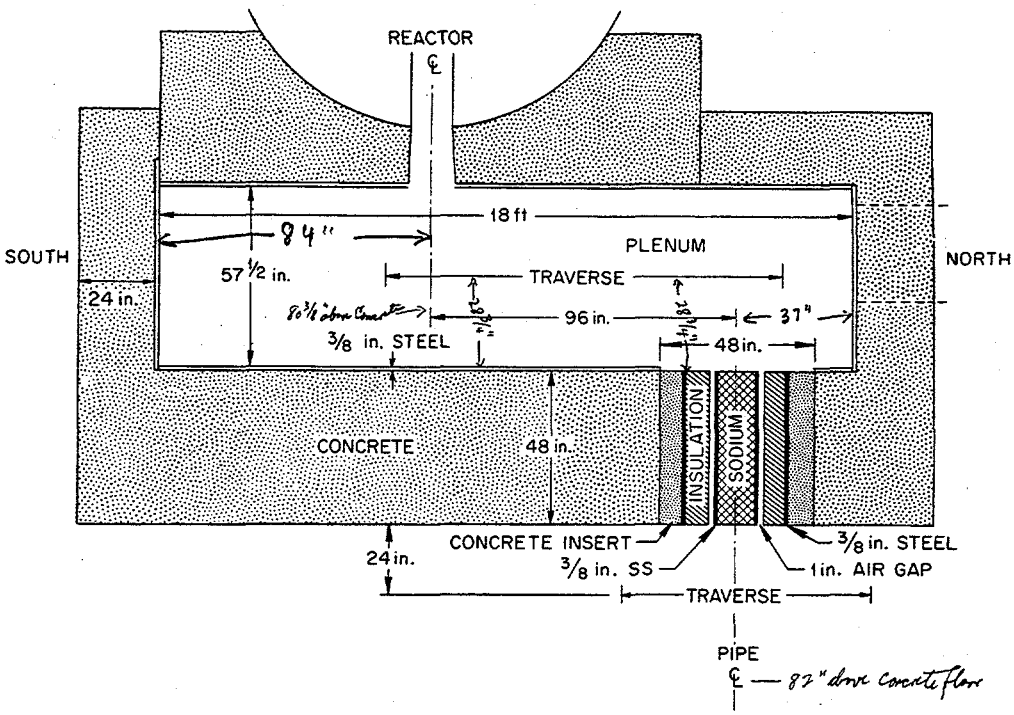

Fig. 12.11 Schematic of TSF Pipe Chaseway Streaming Experiment - Phase II
Plenum simulates HTS Pipeway without Pipes.

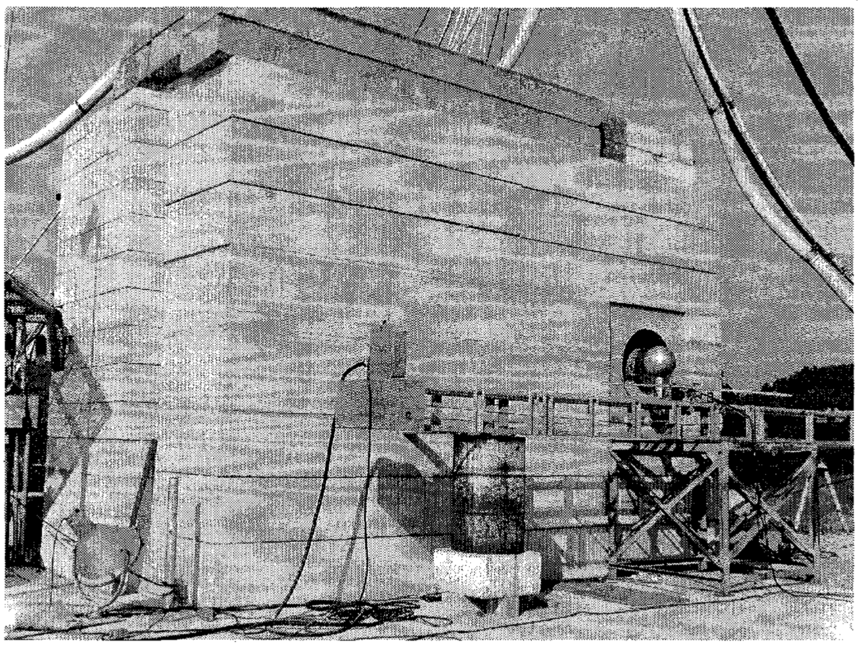

Fig. 12.12 Outside view of Experimental Setup.
(Note: The access opening in the left was closed during the experiment.)

adjacent tubes filled with aluminum, giving the slabs a smooth surface. Radial traverses were made behind mockups of these slabs to observe neutron streaming through the aluminum areas. Benchmark studies of neutron transport through B_4C were included. The measurements were preceded by either the GE spectrum modifier (SM) of 18-in. SS, or the Westinghouse SM of 4-in. SS + 6-in. Na + 1-in. Boral. A typical mockup is shown in Fig. 12.13.

12.10 FTR IRON/POLYETHYLENE SECONDARY GAMMA-RAY EXPERIMENT

Gamma-ray spectral measurements in late-1972 to early-1973 were made as part of the FTR program[14,15,16] to determine the uncertainty in the production and transport calculations of the secondary gamma rays generated in the FTR head compartment shield. The results of the measurements were also included as part of the FTR gamma-ray benchmark experiment. Two different neutron source spectrums were used—the TSR-II beam and a modified TSR-II spectrum which was similar to that incident on the FTR cover. Three different thicknesses of iron were studied, 10, 22, and 34 in., along with the thicker iron samples backed by laminations of borated polyethylene and iron.

During this study, a program was also initiated testing thermoluminescent dosimeters as a means of determining the gamma-ray energy deposited within the iron. Attempts were made using two different TLD materials, CaF_2:Dy and $CaSO_4$:Dy, both in powder form.

12.11 WESTINGHOUSE AND GENERAL ELECTRIC DEMONSTRATION PLANT EXPERIMENTS

Three shielding experiments were devised by GE, Westinghouse, and Atomics International to aid in the design of an LMFBR Demonstration Plant. The initial part of the experimental programs, described earlier in Sect. 12.9, was completed by August 1972 before it was temporarily suspended for the more pertinent FTR gamma-ray measurements described previously in Sect. 12.10. During this delay, changes[17,18] were made to the Demo program plan to also include some gamma-ray spectral measurements. The program ended by July, but not before a successful attempt was made to use a pulse-rise-time discrimination technique applied to a gas-filled proportional counter to discriminate between neutrons and gamma rays in the 10-140 keV energy region. The detector, described earlier in Sect. 7.10, was filled to 3 atmospheres with 85% hydrogen, 10% methane, and 5% nitrogen.

A typical mockup included a spectrum modifier, a radial blanket region, a SS reflector region, and an inconel restraint region followed by additional shielding of sodium and SS. The radial blanket slabs, fabricated just for this experiment, contained a closely packed array of aluminum-clad natural UO_2 fuel pins immersed in sodium as shown in Fig. 12.14. Part of the GE mockups included an axial shield fabricated of SS rods immersed in sodium, also fabricated just for these measurements, that was being evaluated to determine the feasibility of using large (5.25-in.-diam) SS rods in the coolant above the reactor core.

12.12 FTR INCONEL EXPERIMENT

An experiment was performed during the month of July 1973 to investigate the neutron-attenuating and gamma-ray production properties of inconel, since inconel had been chosen for the

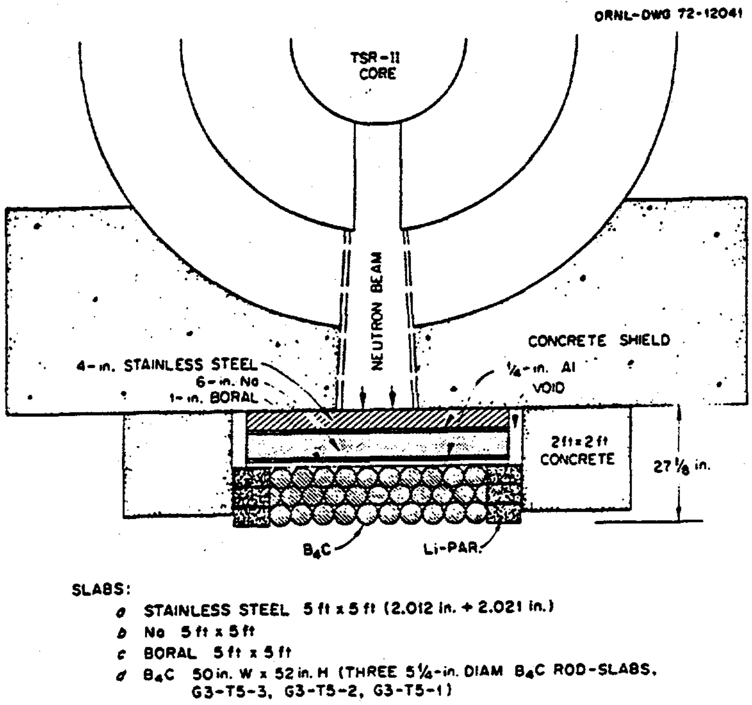

Fig. 12.13 G. E. Radial Shield Configuration, Three Rows 5.25-in.-diam B_4C rods, Item VIII.A.

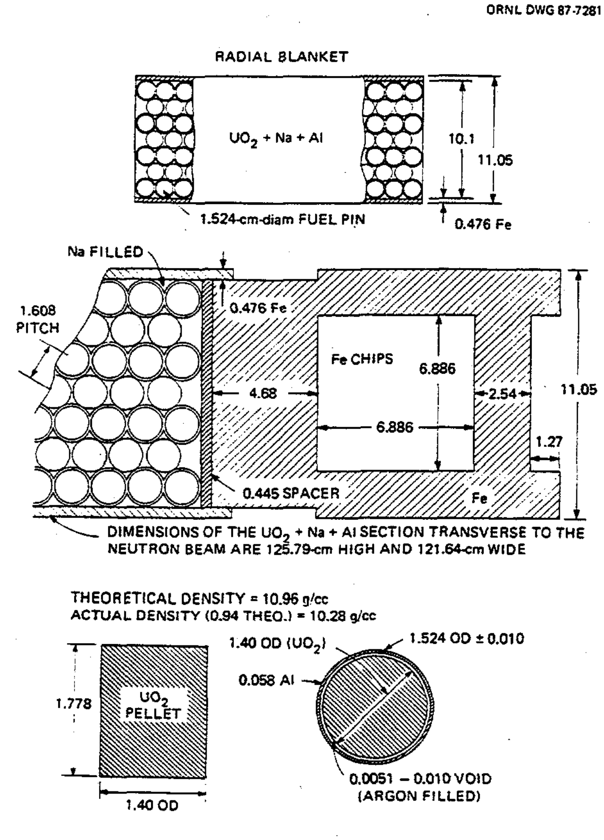

Fig. 12.14 Simulation of the Radial Blanket Assembly and Details of UO_2 Fuel Pins.

reactor's reflector region in the FFTF.[19,20,21,22] The primary constituent of inconel is nickel, and because the cross sections for nickel were not well known, there was some doubt as to the validity of the calculations that had been performed. Thus, the experiment was needed to provide experimental results against which the cross sections could be tested. Two thicknesses of inconel were studied, obtaining neutron and gamma-ray spectra beyond the mockups along with integral neutron flux measurements.

12.13 FTR AND DEMONSTRATION PLANT REACTOR FIRST FISSION EXPERIMENT

The Fast Test Reactor (FTR) and Demonstration (Demo) Plant reactor first fission experiments were conducted to provide data on the absolute fission rates of ^{239}Pu and ^{235}U as a function of shield thickness.[23,24] The experimental configuration included a spectrum modifier followed by slabs of SS, inconel, sodium, and blanket materials which simulated the shield design for each reactor. Thicknesses studied represented typical materials found between the reactor and stored fuel for both reactor designs. Measurements of the fission rates were obtained for detectors utilizing parallel plates coated with thin layers of either ^{239}Pu or ^{235}U so that the efficiency of the counters could be precisely determined.

12.14 FTR AND DEMONSTRATION EX-VESSEL EXPERIMENT

An experiment was conducted at the TSF to determine the effectiveness of adding a graphite moderator around the ex-vessel detectors in order to increase their counting efficiency for both the FTR and Demo shield configurations.[23] The effectiveness of the monitors to accurately indicate reactor power depended on the degree to which they could distinguish between core neutrons and neutrons from stored fuel. This experiment demonstrated to what extent the presence of graphite enhanced the count rate. The reactor cavity shield was simulated by a steel-lined shield of concrete blocks surrounding the graphite for both the FTR and Demo shield designs. For the Demo shield, the Na thickness was reduced by 30 cm and the axial shield in the FTR mockup was replaced by slabs of iron and SS. Measurements were made with detectors bare and surrounded by 12-in. of graphite, except on the side adjacent to the shield. On that side, the graphite was varied to obtain the maximum count rate. The mockup for the FTR experiment is shown in Fig. 12.15.

12.15 CALCULATIONS OF GAMMA-RAY HEATING IN IRON: ANALYSIS OF TSF MEASUREMENTS WITH THERMOLUMINESCENT DOSIMETERS

The reliability of thermoluminescent dosimeters ($CaSo_4$:Dy and CaF_2:Dy powders) for measuring gamma-ray heating continued to be a concern. Results from a previous experiment, where both the Co^{60} source and TLDs were embedded in iron, were lower than calculations by as much as 30%. A new experiment[23,24] was proposed in which the ^{60}Co source was again embedded in iron, but the TLDs were placed at intervals on the surface of the iron as shown in Fig. 12.16. To add credence to the measurements, the experiment also included measurements with the NaI spectrometer.

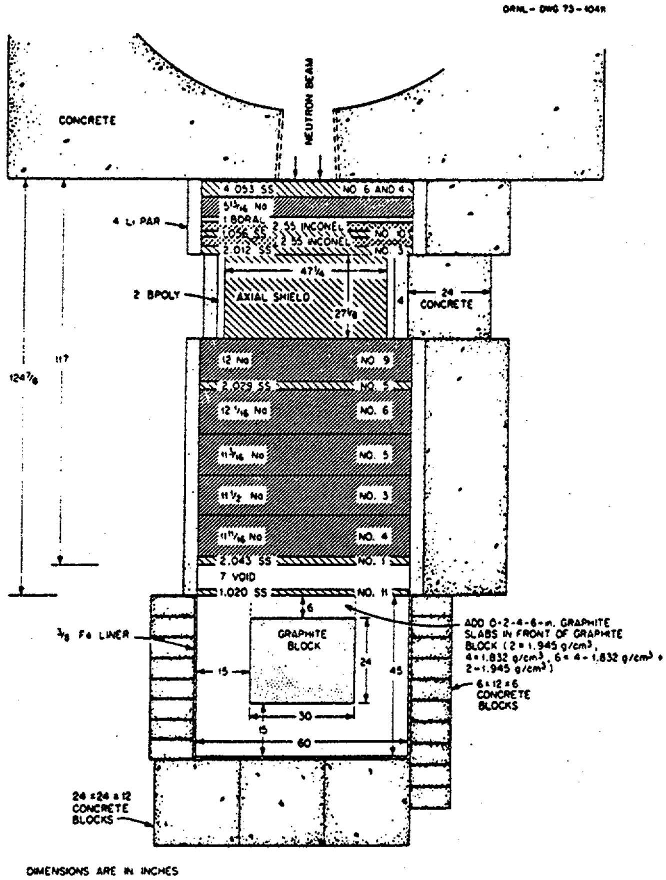

Fig. 12.15 Shield Configuration for FTR External-Vessel Experiment.

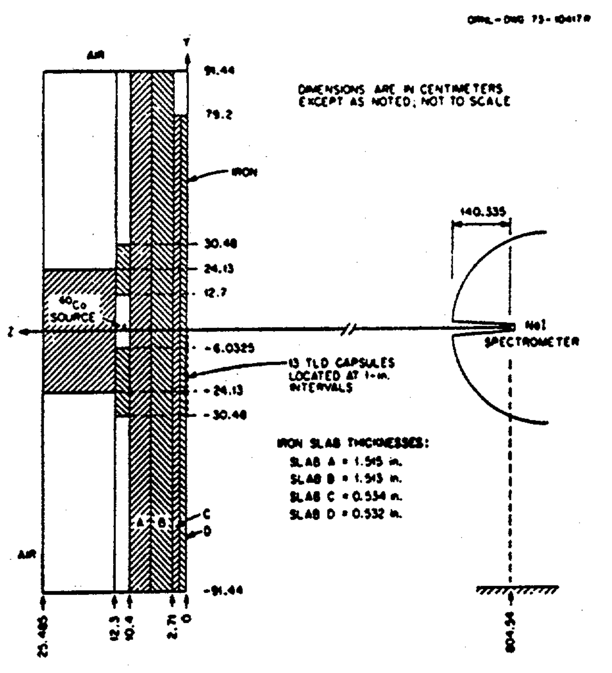

Fig. 12.16 Y-Z Plane of Configuration for ^{60}Co Gamma-Ray Experiment.

Agreement between dosimeter measurements and calculations was about ±20%. Agreement between calculated and measured spectra was very good for angles of 0, 45°, and 60°. Such agreement gave validity to the calculational method, but left some doubt as to the reliability of the procedures used in the TLD work.

12.16 TOTAL CROSS SECTION EXPERIMENT

An experimental program[25] was undertaken in late-1973 to verify the accuracy of the total cross sections for several materials in the neutron energy range from about 5 keV to 1 MeV using the H_2 proton recoil spectrometer. Measurements for Fe, Ni, O, Cr, Na, and U were obtained since these elements were of interest to the LMFBR program. Carbon was included since its cross sections were well known and it could be used to provide an evaluation of the method. SS was also added to ensure that the energy correlation in the Fe, Ni, and Cr total cross section peaks and valleys were sufficiently accurate.

Analysis of the experimental data, however, showed that the background measurements with the NE-213 were not properly handled. It was necessary to repeat the measurements[26] in the spring of 1974, this time limiting the number of samples to iron, nickel, carbon, SS, and chromium, but increasing the number of sample thicknesses for each element. To reduce the gamma-ray intensity in the reactor beam, the collimator system was revamped to that shown in Fig. 12.17. This arrangement reduced the background considerably and the resolution of the detector was greatly improved.

12.17 ROD DROP EXPERIMENT FOR INSTRUMENTATION AND CONTROLS DIVISION

As was often done, experiments in progress would be interrupted for another experiment whose priority had suddenly been upgraded. Such was the case during the previous experiment (Sect. 12.16), when it was interrupted for a brief rod-drop measurement for the ORNL Instrumentation and Controls (I&C) Division using the Demo-shield configuration. The measurements were originally made to see if reactivity changes due to rod insertions in the core could be measured by detectors external to the reactor shield. Earlier results produced anomalous readings from a detector in a position that would be external to the pressure vessel in a commercial reactor system, when compared to various readings internal to the shield. In the repeat runs, the anomalies were resolved, indicating that rod-drop effects can be readily predicted by detectors either internal or external to the shield.

12.18 NEUTRON TRANSMISSION THROUGH AN ATOMICS INTERNATIONAL-LIQUID METAL FAST BREEDER REACTOR LOWER AXIAL SHIELD MOCKUP

This experiment was performed to measure the intensities and energy distributions of neutrons transmitted through a mockup designed by Atomics International (AI) to test their early design of the lower axial region of the LMFBR demonstration plant.[27,28] The main objective of the experiment was to provide data for evaluation of the calculational methods for predicting the streaming through control rod penetrations and to determine the neutron penetration in the lower axial shield. Of specific interest was the extent to which neutrons streaming through control rod channels, extending from the reactor

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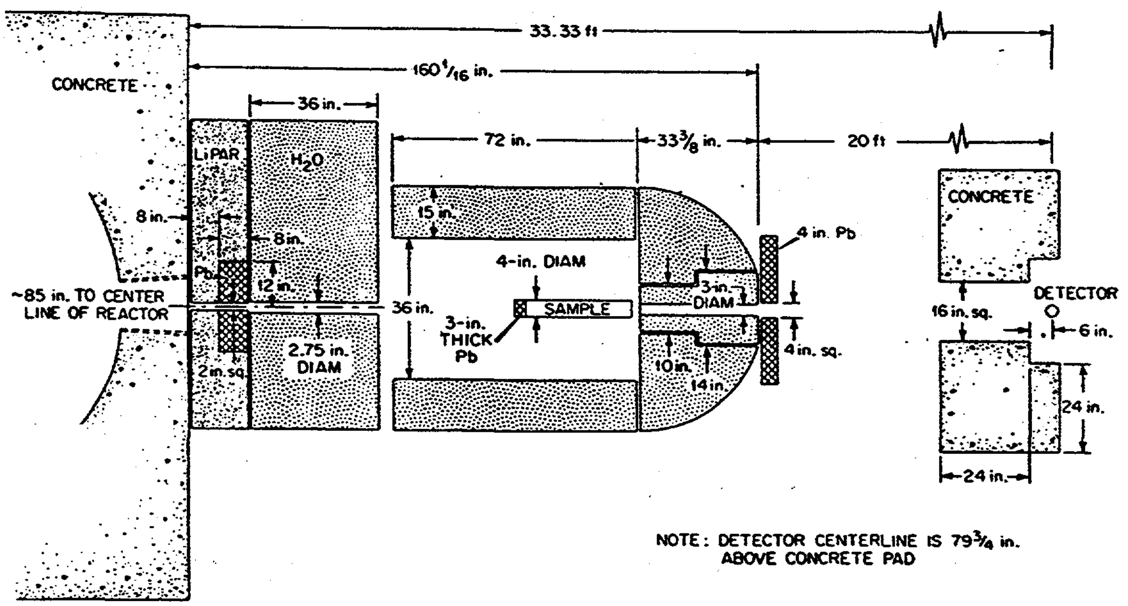

Fig. 12.17 Experimental Setup for the Total Cross-Section Experimental Check (Top View), May 1974.

core down to the upper grid plate, would increase the radiation damage to the grid plate during reactor operation.

The mockup consisted of: (1) a lower axial blanket of UO_2, sodium, and SS, (2) a flow region of sodium and SS, (3) a SS upper grid plate perforated with sodium plugs, (4) a sodium plenum region, (5) a SS lower grid plate, and (6) a sodium region simulating a pool of sodium. The mockup was preceded by a spectrum modifier whose filtered spectrum simulated that expected from a fast reactor. These components were contributed by several sources. The blanket, transition regions, and upper grid plate were fabricated by AI, while the spectrum modifier came from Westinghouse's Advanced Reactor Division. The plenum and pool sodium regions were simulated with Na-filled tanks that had been built at AI for earlier experiments, and the lower grid plate was modeled by SS slabs from the TSF.

The blanket and flow transition regions contained three holes that could be filled with plugs that represented different locations of control rods, but only the plugs in the center hole were varied to simulate (1) a sodium-filled control rod channel, (2) an empty channel, and (3) no channel (homogeneous plug). A schematic of the experiment is given in Fig. 12.18, showing the location of the plugs within the mockup. Measurements were made beyond various sections as they were added to the mockup and in the holes as provided for *in situ* measurements.

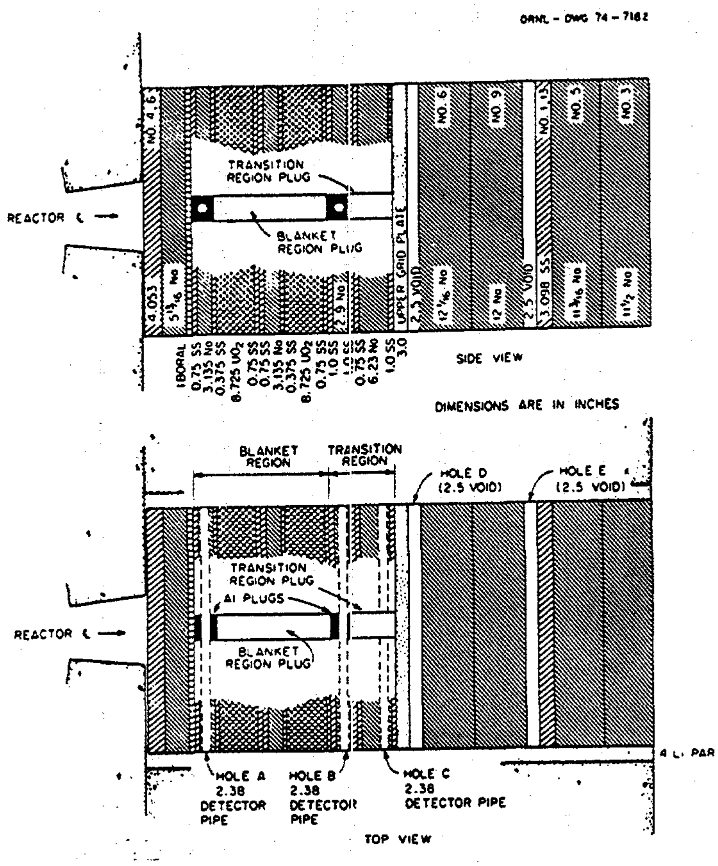

Fig. 12.18 Center Plug Arrangement for Dectector Measurement in Holes A, B, D, and E.

12.19 FISSION RATE DETERMINATION OF SIMULATED FAST TEST REACTOR STORED FUEL FOR THE FAST REACTOR EXPERIMENTAL SHIELDING PROGRAM

An experiment was performed which simulated the three-dimensional problem of predicting the fission radiation emitted by the stored fuel within the FTR.[29] The results provided experimental verification of the theoretical models normally used to describe stored-fuel activity. A spectrum modifier was needed to provide the proper energy source, followed by the radial blanket (slabs of UO_2 in sodium fabricated for earlier experiments), and the radial shield, which was simulated using slabs of SS, inconel, and iron, as seen in Fig. 12.19. To minimize costs, the stored fuel was simulated by three cylindrical annuli of ^{235}U placed within a solid aluminum block rather than a sodium-filled tank (Fig. 12.20). The effect of the stored fuel was clearly demonstrated by comparison with measurements made using a dummy aluminum block. The experiment provided a good test of the theoretical models used to calculate stored fuel activity.

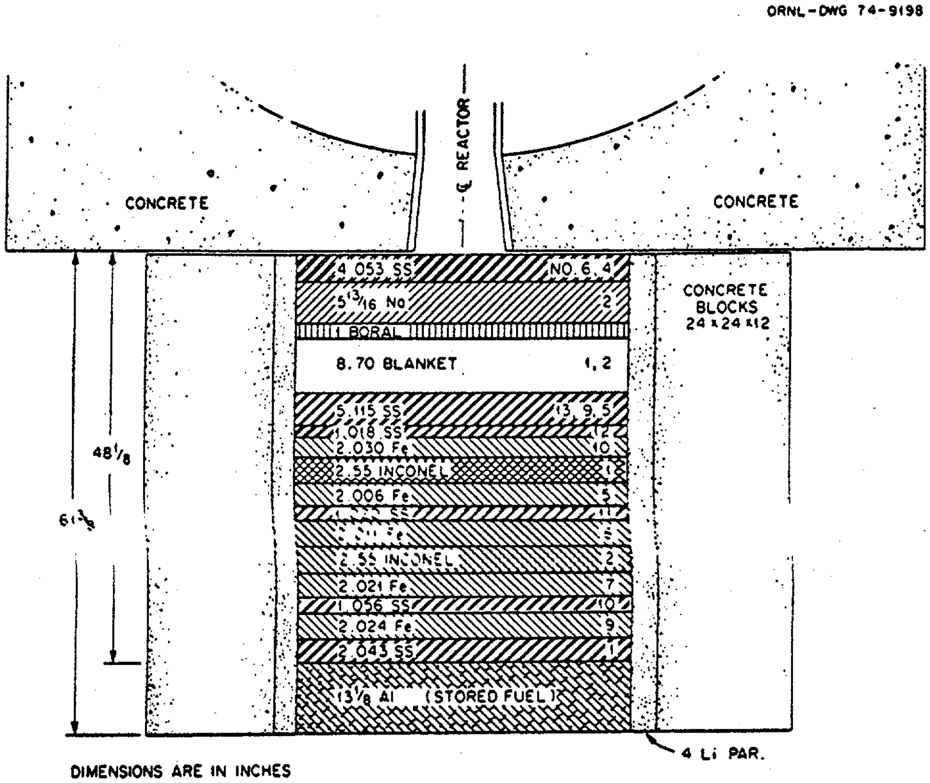

Fig. 12.19 Configuration III: Configuration II Plus 12-in. Stored Fuel, Item III

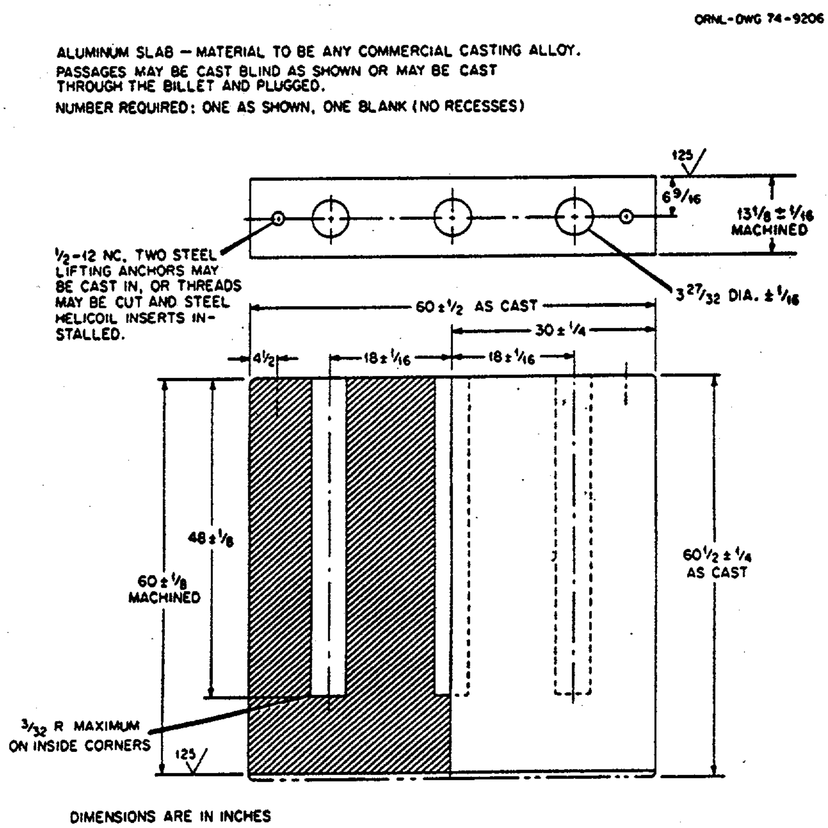

Fig. 12.20 13-in. Aluminum Slab Representing the Stored Fuel Region.

12.20 SECONDARY GAMMA-RAY PRODUCTION IN REINFORCED CONCRETE SHIELDS

An experiment was performed at the TSF during the last few months of 1974 to measure the gamma-ray fluxes and energy deposition rates in various concrete shield configurations. There was concern as to what effect the secondary gamma rays generated within the large concrete shields proposed for the LMFBR would have on the fluxes beyond the concrete, since the concrete would contain large amounts of reinforcement iron. For this experiment[30] (referred to as the "rebar" experiment), measurements were made of the energy deposition rates within the concrete configuration and of the gamma-ray intensities behind the slabs.

The concrete slabs used were of two types: one had only enough reinforcement along the edge of the slabs for handling support; the other had rod reinforcement throughout the slab as seen in Fig. 12.21. Data were obtained as a function of total concrete thickness and location of the rebar slabs within the mockup. *In situ*-type measurements were made with TLDs and an ion chamber, while penetration measurements were made with Bonner balls and the sodium iodide spectrometer.

12.21 RADIATION HEATING IN IRON AND STAINLESS STEEL SHIELDS FOR THE CRBR PROJECT

An experiment was started at the TSF in late 1974 to measure heating rates within shield configurations typical of the LMFBR using calcium fluoride (CaF_2) and lithium fluoride (LiF) TLDs and a high pressure argon-krypton-filled ion chamber (IC).[31,32] Measurements of the gamma-ray spectrum leaving the mockups, along with Bonner ball neutron data, were made for several of the configurations to provide reference data for comparison with calculations. Overall, 42 different experimental arrangements were investigated.

A secondary objective was to compare the TLD procedures used by ORNL, Argonne National Laboratory (ANL) (both at Chicago and Idaho Falls), and the Massachusetts Institute of Technology (MIT) to measure heating rates in fast reactor systems. Personnel from these facilities participated directly in two phases of the experiment involving mockups of iron and SS. Intercomparisons of the independently determined results, along with comparisons with the ion chamber and calculational results, provided a measure of the reproducibility and credibility of using TLDs to measure heating rates.

A typical mockup for the experiments is shown in Fig. 12.22. The slabs were spaced so that small voids existed between them into which the TLDs could be placed. The TLDs were exposed in iron capsules that are displayed in Fig. 7.11 (Sect. 7.12). After exposure, the TLD chips were heated individually in a TLD reader and the light output converted to current, which when integrated and multiplied by the calibration factor, was converted to energy deposited in the iron or SS.

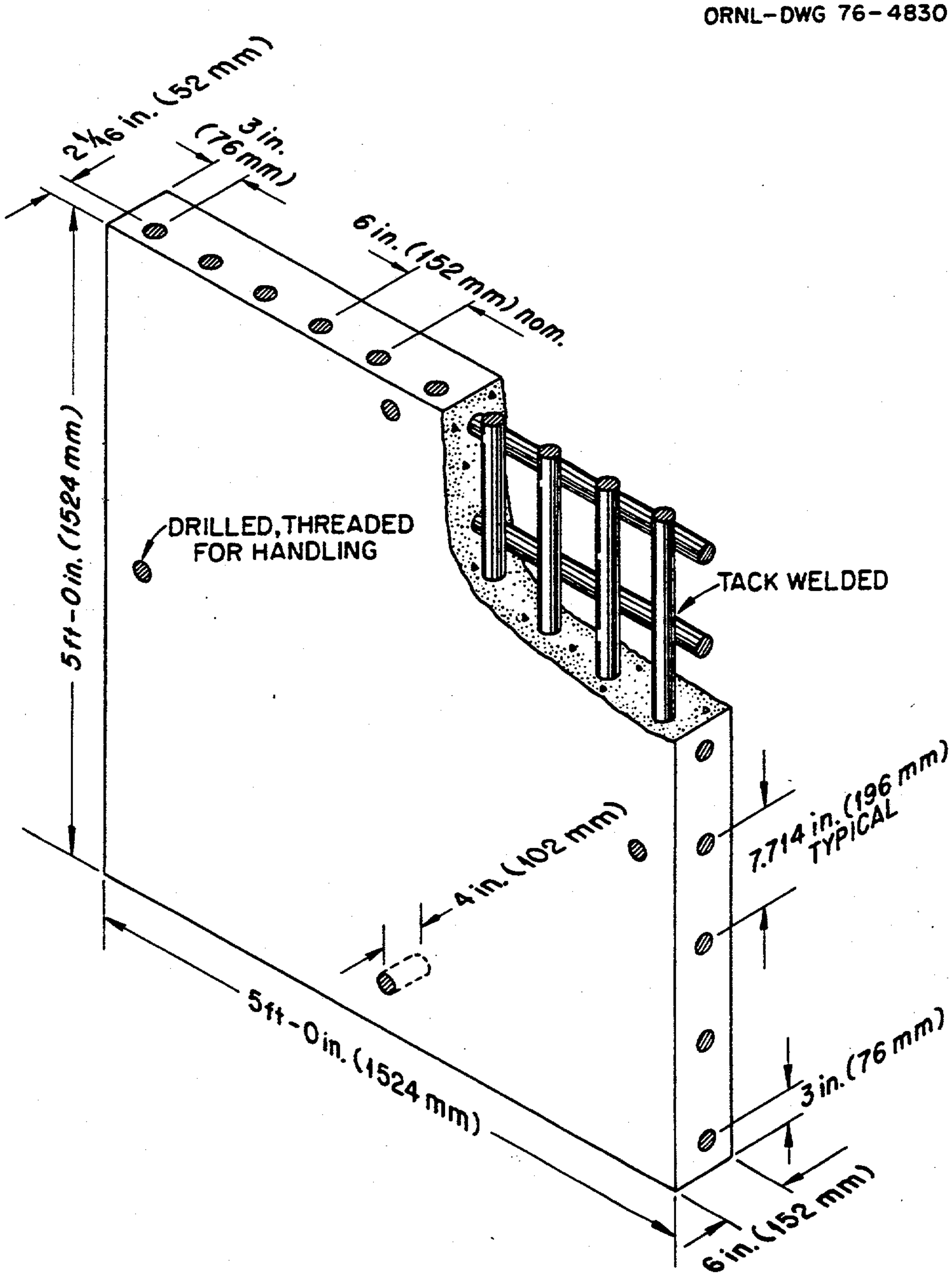

Fig. 12.21 Sketch Showing Construction of Rebar Slab. All Reinforcing Bars are No. 11 Steel, comprising approximately 7.6 vol% of the Slab.

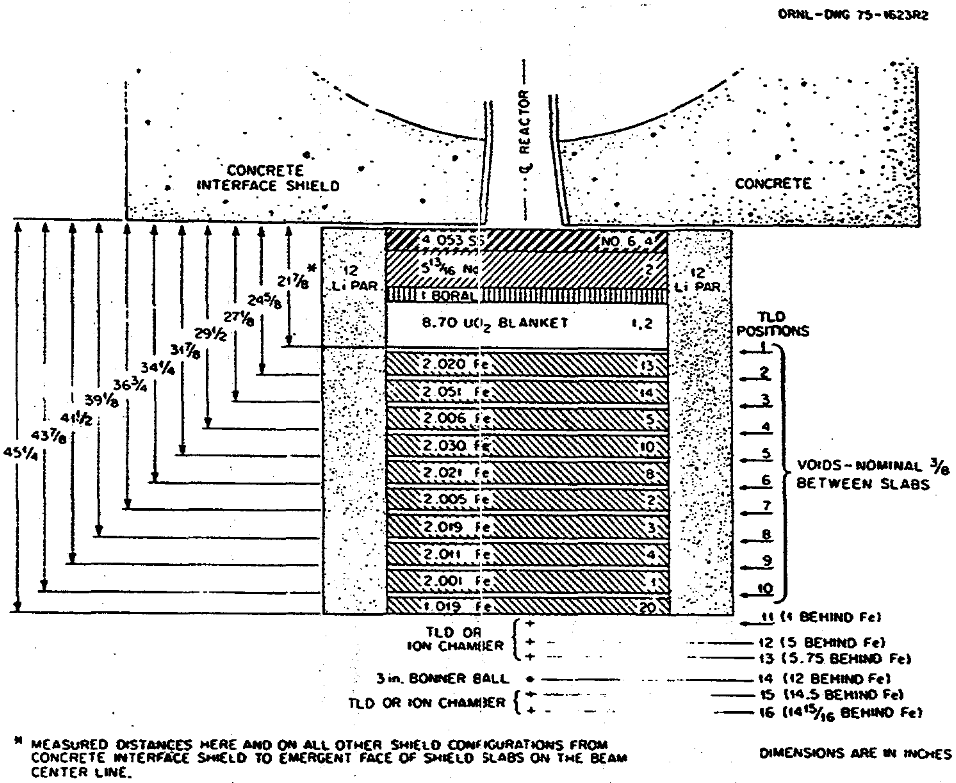

Fig. 12.22 The "Iron Shield" Configuration, Spectrum Modifier + 8-in. UO_2 Blanket + 19-in. Fe (including voids), Item II.

12.22 MEASUREMENTS AND CALCULATIONS OF THE CRBR UPPER AXIAL SHIELD EXPERIMENT

The intensity of the neutron penetration through the top deck shield proposed for the Clinch River Breeder Reactor was considered a controlling factor in the biological dose existent on the top deck. Calculations were made using standard transport methods and cross sections, and an experiment was performed to test both of them.[33]

For this work, the reactor was placed within the new concrete reactor shield shown in Figs. 5.11 and 5.12 (Sect. 5.9), at which location the shield sample was only 91.3 cm from the center of the core. The shield was modeled by adding one or more components at a time, with neutron measurements made behind each component (or group of components) as they were added to the configuration. This procedure provided a large number of comparisons with calculations so that trends and/or discrepancies could be readily observed. The mockup consisted of varying amounts of SS, followed by up to approximately 450 cm of sodium, and ending with approximately 90 cm of iron, as shown in Fig. 12.23. The estimated attenuation factor for the full mockup was about 1 x 10^{12}.

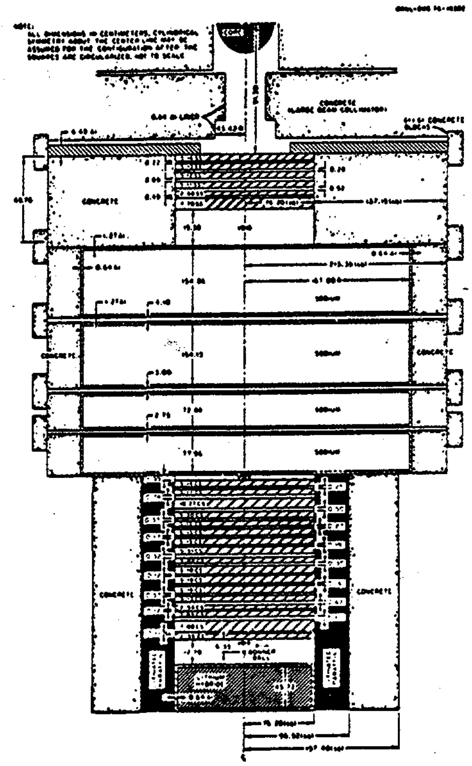

Fig. 12.23 Top View of the Geometry of the Measurements Behind 30.90 cm SS + 459.83 cm Na + 90.06 cm CS.

12.23 GAMMA-RAY HEATING MEASUREMENTS IN LMFBR SHIELDS WITH THERMOLUMINESCENT DOSIMETERS

Attempts to measure the gamma-ray heating deposited in iron and SS using TLDs was described earlier in Sect. 12.21. The technique was relatively new at this time and the proper operating procedures using TLDs were not fully established. Exposures were also made by personnel from ANL at Idaho Falls and Chicago, and from the MIT, with the intent of intercomparing results, to establish confidence in the independent methods developed at the different facilities. The comparisons suggested a need for improvement, which led to the development of a lengthy handling procedure by W. Y. Yoon,[34] a graduate student working at the TSF, that provided consistency of results with good reliability. His effort resulted in his doctoral thesis from the University of Tennessee, and provided a procedure for the TSF that was capable of determining the gamma-ray energy deposition in a mixed neutron and gamma-ray field to within a ±3% on an absolute basis. His work included a detailed evaluation of the experimental technique employed, a verification of the gamma-ray spectral correction factor used to relate the energy deposited in the TLD to that deposited in the capsule material, and the development of the neutron dose correction factors for both CaF_2:M_n and ^7LiF thermoluminescent materials.

Verification of this procedure was established through a series of measurements using single-energy gamma rays, making the measurements in air and for different iron thicknesses. These results, when compared to both data obtained with a high-pressure iron-equivalent ionization chamber (Sect. 7.13) and to discrete ordinates calculations, were within the established criteria.

The procedure was then given the ultimate test by comparing the TLD data with that obtained with the ion chamber and calculations for a typical shield mockup. Careful measurements and readouts, along with precise spectral correction procedures, produced results within the desired limits.[35]

12.24 RADIATION HEATING STUDIES IN A STAINLESS STEEL AND SODIUM SHIELD

The shield configuration for these measurements consisted of a spectrum modifier followed by stainless steel, sodium, and more stainless steel. *In-situ* measurements were made using both LiF and CaF_2 thermoluminescent dosimeters (chips) at several locations beyond the spectrum modifier.[36] The purpose of the work was two-fold; first, to establish a series of benchmark-type measurements of radiation heating in a stainless steel and sodium system and secondly, to determine the accuracy of calculated heating rates using a standard design cross-section set based on ENDF/B-IV, together with derived response functions for each detector.

12.25 NEUTRON STREAMING IN A CRBR PROTOTYPIC COOLANT PIPE CHASEWAY

An experiment was conducted at the TSF to determine the neutron distribution throughout a mockup of the primary exit coolant pipe chaseway for the CRBR.[37] The measurements provided benchmark data to evaluate the discrete ordinates and Monte Carlo techniques for predicting neutron transport in chaseways containing sodium coolant pipes for the CRBR, and to provide data on important neutron scattering regions of the sodium pipe that were not included in earlier pipe chaseway experiments.

The mockup included two 90° bends (Fig. 12.24) plus additional shielding walls in the shape of chokes. Even though it was intended that this configuration simulate in full scale the current CRBR design, it was necessary to scale down some of the cavity dimensions due to lack of adequate space on the pad area and available concrete shielding material. The sodium coolant was modeled using Na_2CO_3 powder enclosed in iron pipe and the cylindrical surfaces of the pipes were enclosed in Kaylo insulation. The length of this coolant pipe, that passed through the three legs of the CRBR duct, was divided into four sections that allowed measurements to be made at pre-selected locations within the chaseway. Neutron entry into the first leg of the chaseway was through a concrete choke 152.4 cm in diam following the sodium in the spectrum modifier, as shown in Fig. 12.25. A photograph of the final arrangement of the concrete structure shown schematically in Fig. 12.24 is pictured in Fig. 12.26. Neutron flux measurements were made through the doubld bend passageway as a function of the sodium coolant pipe length. Exclusive of the spectrum modifier, the three legs of sodium coolant attenuated the neutron flux by about a factor of 10^9.

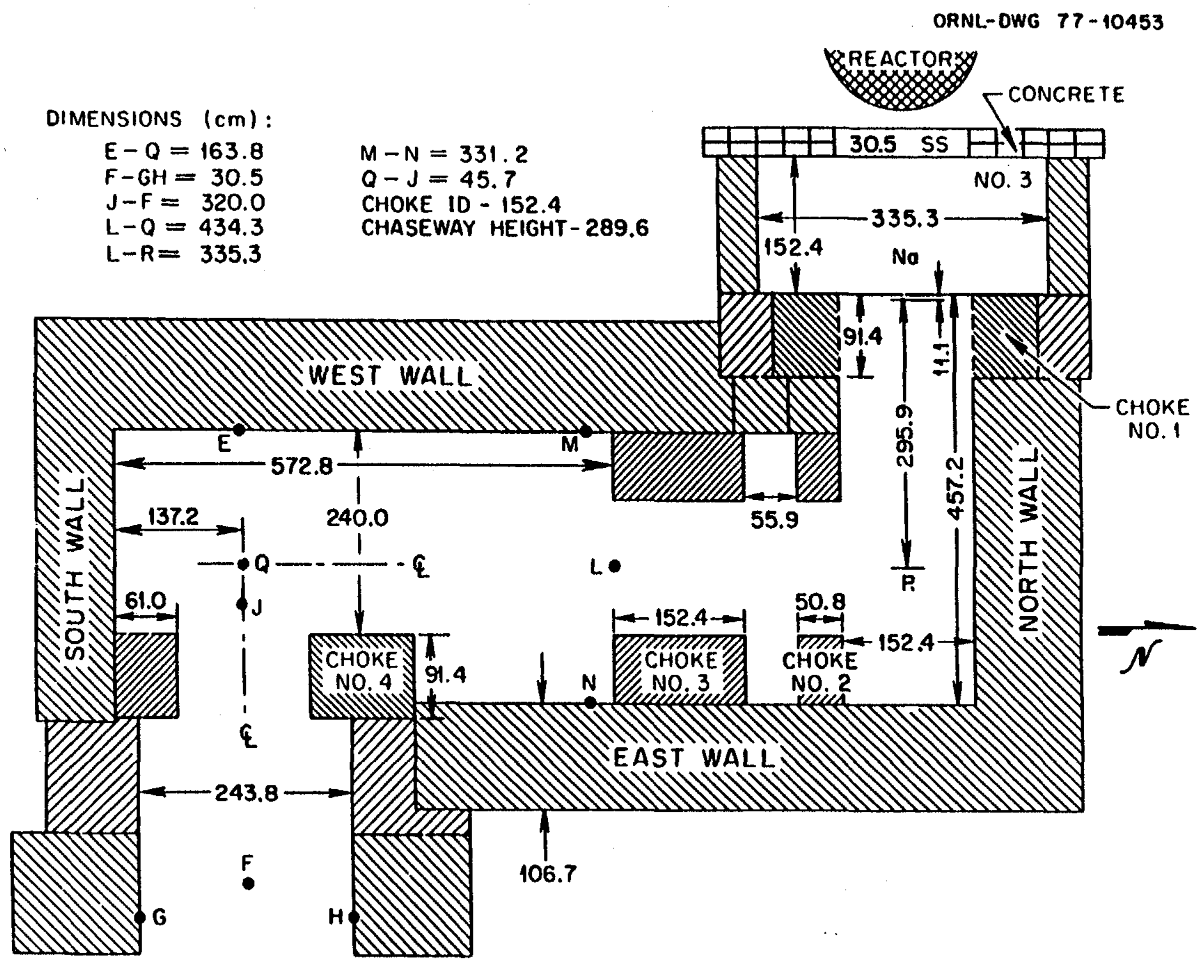

Fig. 12.24 Schematic of Pipe Chaseway Configuration through Horizontal Midplane.

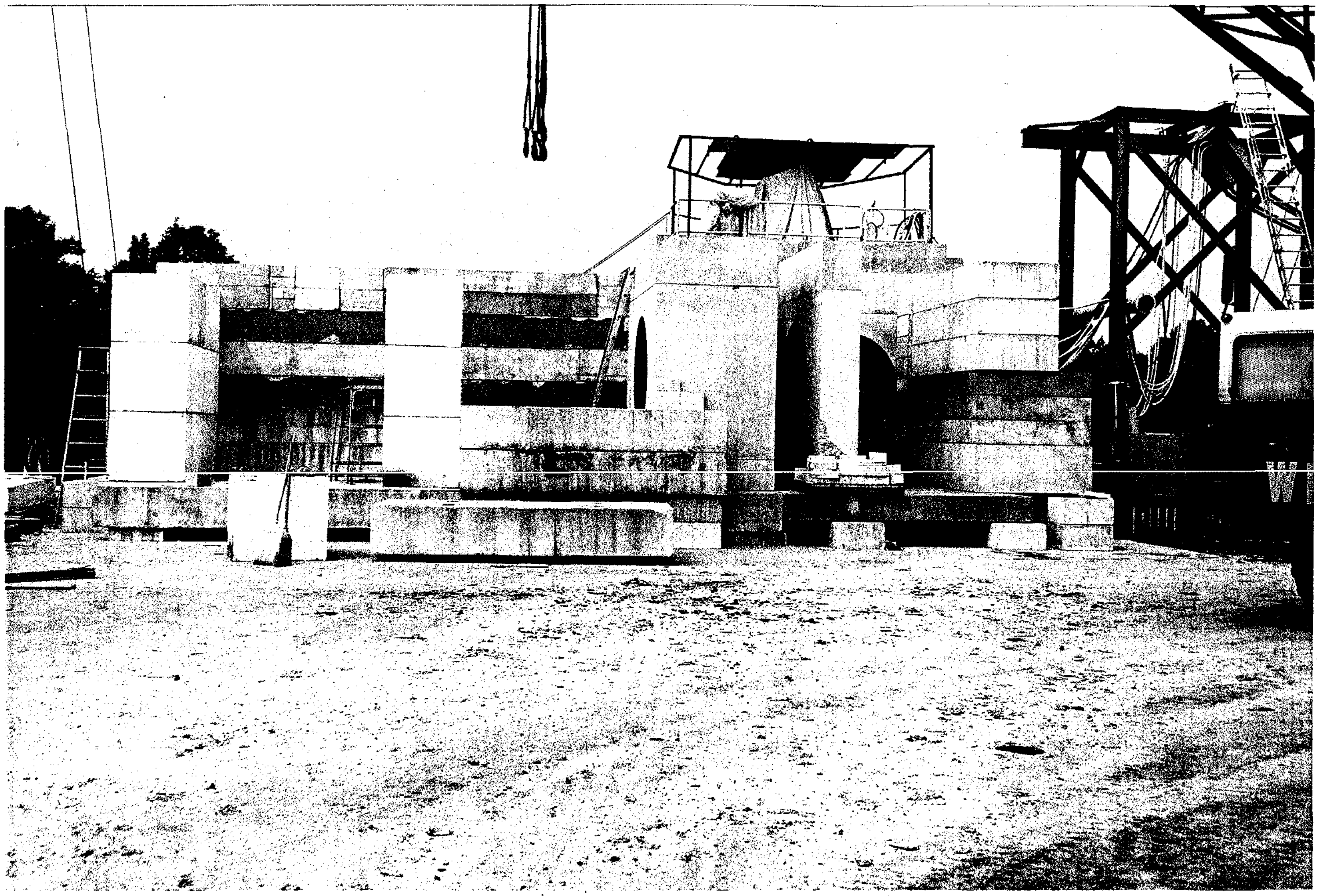

Fig. 12.25. Arrangement of First Three Chokes with respect to the Axis of the first Leg of the Chaseway on the right.

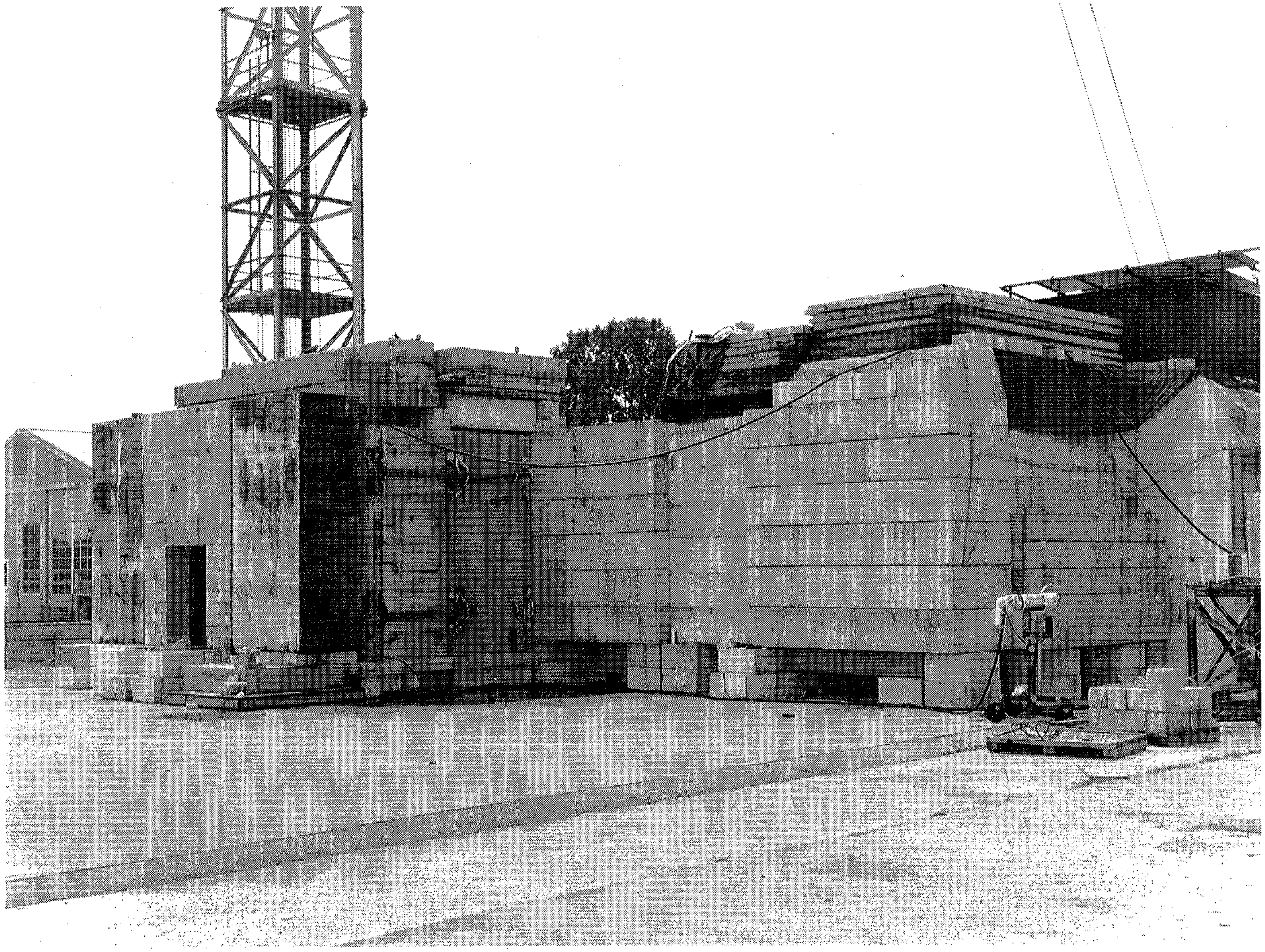

Fig. 12.26 Final Arrangement of Concrete Slabs forming outside Wall of the Pipe Chaseway.

12.26 MEASUREMENTS AND ANALYSES OF THE CRBR INCONEL-STAINLESS STEEL RADIAL SHIELD EXPERIMENT

A request was made by WARD that the TSF do a series of experiments to determine quantitatively the advantage of replacing stainless steel with inconel in the radial shield.[38] Independent studies were made for different thicknesses of each material with measurements obtained behind each thickness. The reactor source was modified to provide a source spectrum similar to the one incident on the CRBR radial shield.

12.27 MAPPING OF NEUTRON FLUXES ABOVE TOPHEAD OF FFTF

The first FFTF region to be examined by the analytical section of ORNL's shielding group was the lower axial shield, which was followed almost simultaneously with calculations of the upper axial shield. This led to a study of streaming through clearances around the rotating plugs through the thermal shield and top head, for which experiments had been performed at the TSF using straight and annular slits in iron. Improvements in the calculational methods indicated considerable leakage radially outward from the vessel with streaming upward to the top of the reactor cavity. This led to concern regarding the cavity fluxes resulting from storage of fuel assemblies in this region. Once aware of the problem, WARD proposed placing a shield in this region to reduce the streaming, which again required analysis by ORNL for several different designs.

It is to these latter calculations that the ORNL measurements[39] made in the head compartment at the FFTF were addressed. It was the first time that experimental measurements could be made in a completed reactor system to study a shield designed by the ORNL analytical group and demonstrate the reliability of the calculations. The purpose of the measurements were: (1) to test the effectiveness of the shield protecting the top head compartment in which instruments and equipment were located and maintenance activities were performed; and (2) to provide data that would indicate how accurately the neutron fluxes had been predicted by the ORNL calculations. As an additional benefit, the measurements also provided data that could be interpreted to give approximate operational dose rates.

The locations of the reactor vessel head shield and the stored fuel assemblies are shown in Fig. 12.27. It should be noted that the top head shield was also penetrated by control rod drives, in vessel handling machines, instrument trees, etc., which were centrally located around the reactor's vertical axis. Calculations had indicated, however, that due to neutron streaming in the cavity, the highest neutron intensities should be expected near the ring of bolts that secure the shield to the reactor support structure. That region was thoroughly investigated.

Because of the conglomeration of ducts, pipes, and instruments, it was not possible to make measurements on the shield itself. Instead, all the data, except one measurement, were obtained in the head compartment on the maintenance deck. Measurements were made with and without different quantities of stored fuel in the reactor cavity.

The initial measurements were started in August 1981 while the reactor was going through its initial startup phase. There were, however, unforeseen problems that caused delays and prevented the reactor from reaching its many programmed power plateaus as scheduled. This limited the amount of data that could be obtained, necessitating a return visit for a second series of measurements in November of that year, followed by a third series in April of the following year.

FFTF REACTOR

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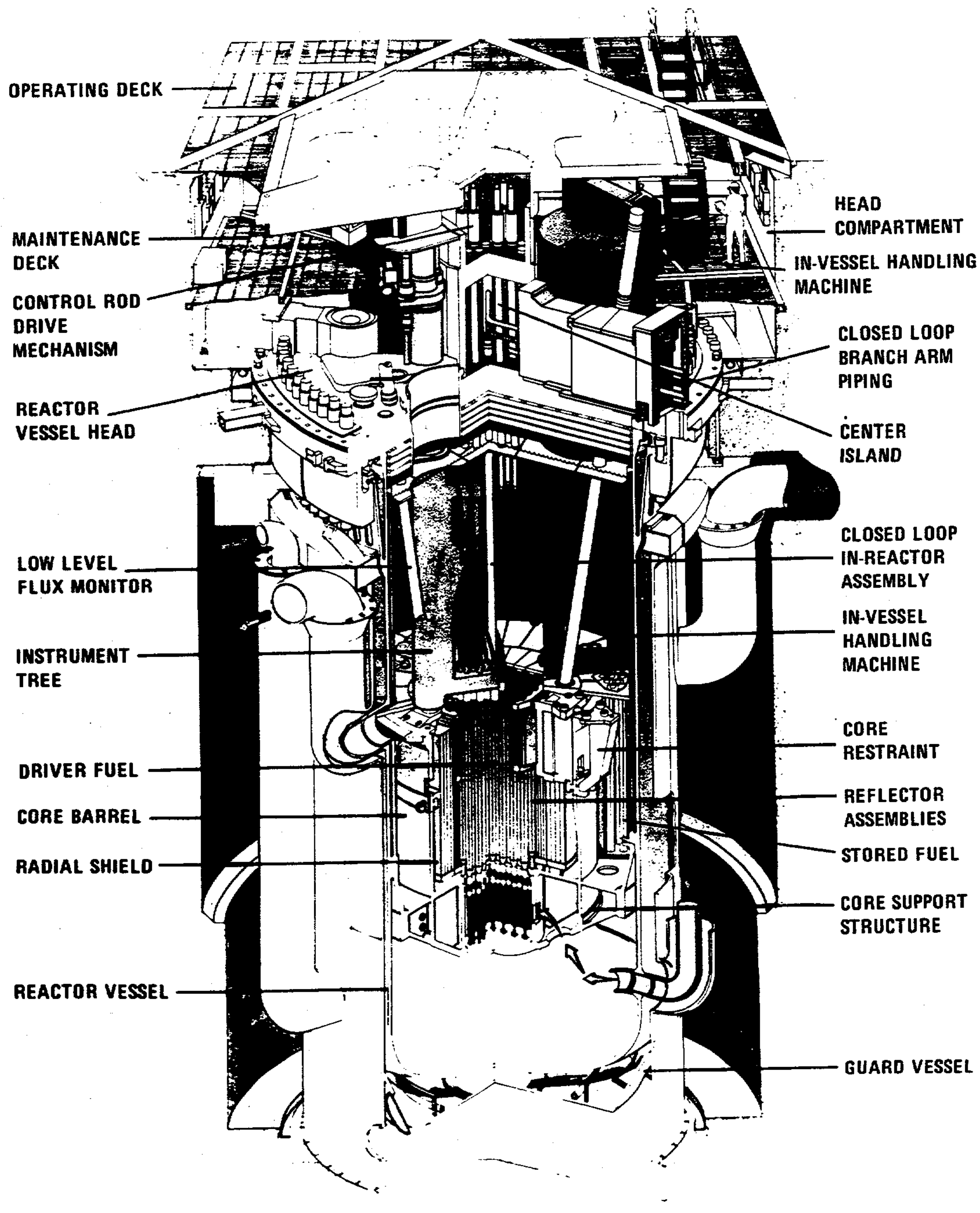

Fig. 12.27 Cutaway Drawing of a Model of the Fast Flux Test Facility.

Because data measurements had no priority with respect to the reactor operating schedule, the amount of time available for taking data was somewhat limited, reducing the program of measurements to just two traverses at different locations across the gap of the headball area, which took a considerable amount of time; and a series of individual measurements at selected locations. The two traverses are indicated in Fig. 12.28. Calculations were accurate to within a factor of two, which was considered to be excellent given the factor of 10^{14} total attenuation from the core to the head compartment. The picture in Fig. 12.29 shows the location of the Bonner ball detector on the maintenance deck along the 270° traverse line (indicated in the previous figure).

12.28 LIQUID METAL REACTOR ALTERNATE SHIELDING MATERIALS EXPERIMENT

A principal concern of shield designers of liquid-metal-cooled reactor systems was the amount of shielding necessary to limit the neutron flux intensity at selected locations within the system. Beginning with the Large Scale Prototypic Breeder Reactor (LSPB) design and later with liquid-metal-cooled reactor (LMR) concepts, shield designers in the U.S. began to actively investigate the advantages of newer, lighter shield materials. The large size of the LSPB was requiring a voluminous amount of shielding within the vessel to protect components and limit activation. The cost of stainless steel as a shield was enormous, and its weight posed a significant problem with respect to the seismic response of the system. Also, the move to smaller LMR concepts required better shielding material due to the reduction in space.

The most attractive material appeared to be boron carbide. Since previous shield studies used mainly steel for shield material, very little information was available about the attenuation properties of B_4C, and B_4C plus other materials. It was also recognized that B_4C in combination with steel could possibly be advantageous for near core shielding. Thus an experiment was designed to study its properties as a shielding material.

The Alternate Shielding Materials Experiment[40] was divided into two phases, one for study of B_4C shield designs near the core, and the other for study of materials near the heat exchanger and stored fuel locations. Each study required a different modification of the reactor source. Measurements were made behind successive layers of shielding as they were added to the mockup.

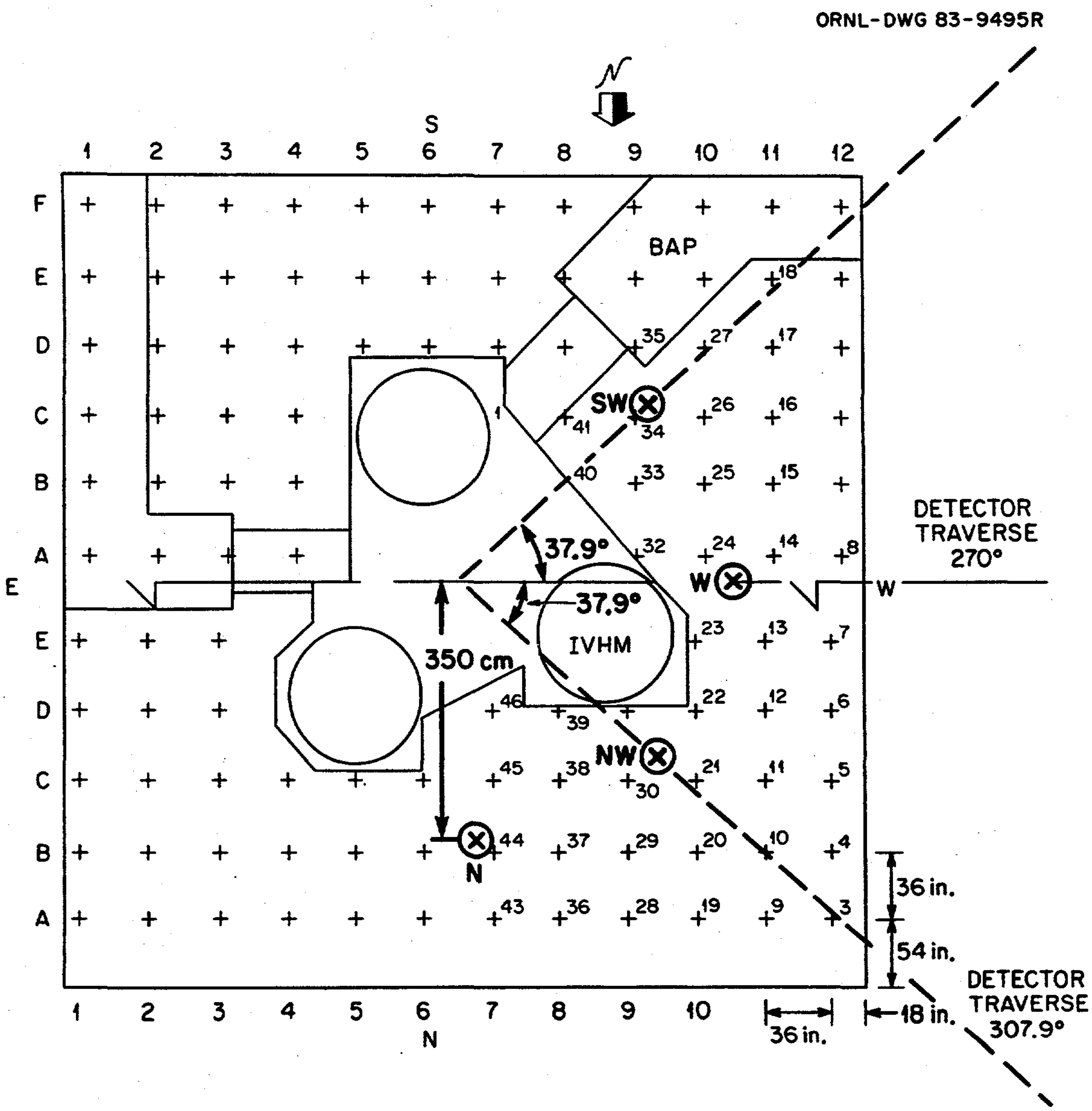

Fig. 12.28 Matrix of Detector Locations for Measurements in Sectors 2 and 3.

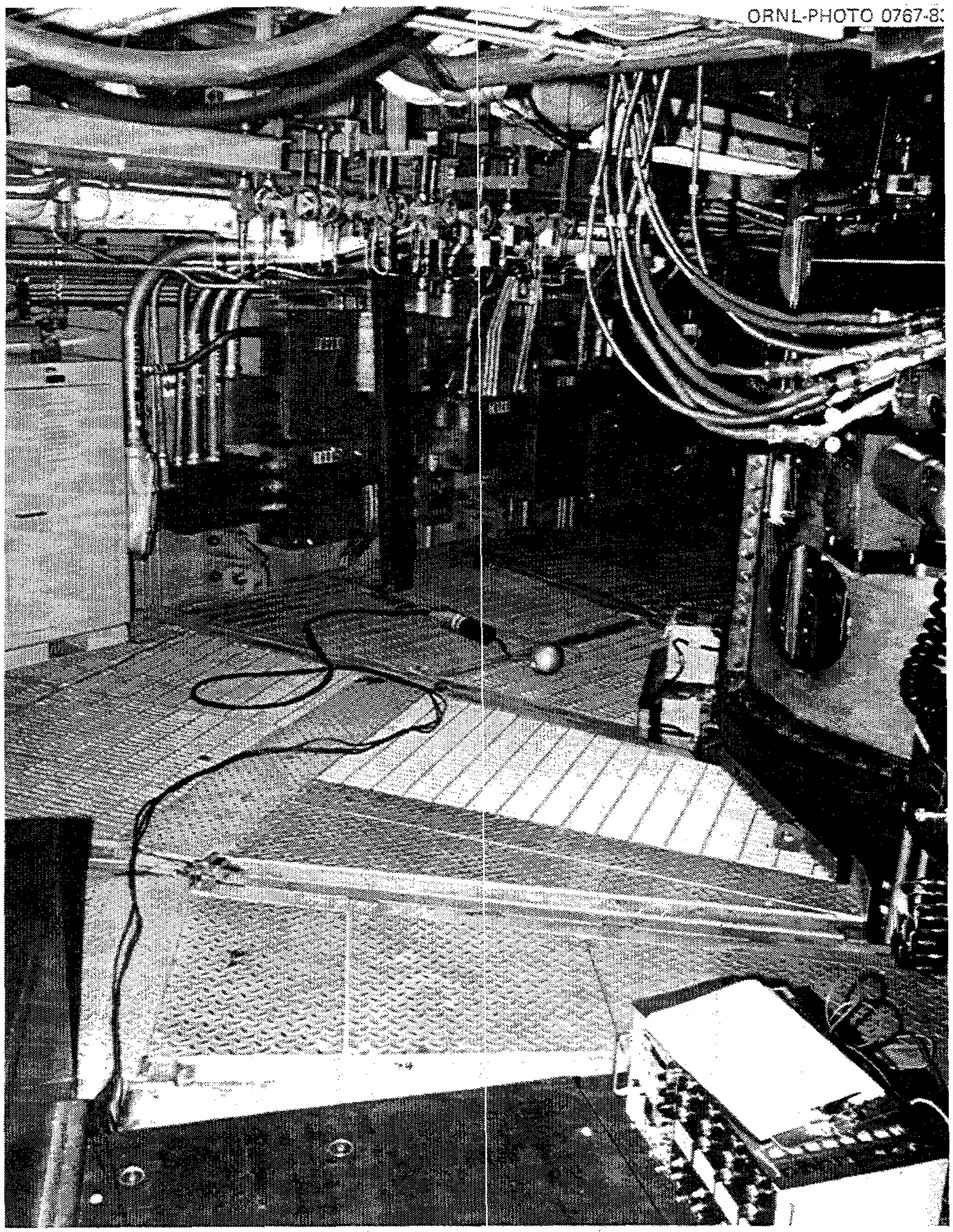

Fig. 12.29 View of the West Side of the Maintenance Deck looking South to North.

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13.0 GAS-COOLED FAST REACTOR PROGRAM

13.1 INTRODUCTION

Studies of the concept of a Gas-Cooled Fast Reactor (GCFR) using helium as a coolant were initiated in the United States in 1962 by the General Atomic Company (GAC).[1] The following year, the Atomic Energy Commission (AEC) provided funding for the company's fuel element development work, and within several years a few of the U.S. utilities were supporting GAC in a broad GCFR program. The first conceptual design of a demonstration plant was completed in 1970, a design that used information based on a minimum extrapolation of the technology for the High-Temperature Gas-Cooled Reactor and the Liquid-Metal Fast Breeder Reactor. By 1974, advances in technology led to a new design for the demonstration plant. By 1975, the total number of participating utilities had increased to 57, with several of the European utility companies joining the effort. The utility-supported program emphasized work on the overall plant design, licensing, and citing, while the AEC supported work on reactor physics and component development. The AEC also supported a number of other GCFR programs at other facilities, such as the reactor shielding analysis work in the Neutron Physics Division at ORNL. The ORNL analysis group was asked by GAC to review the shielding system for the new GCFR plant design to identify areas of concern. ORNL utilized an approach that was similar to that taken for the previous FFTF effort, that is, a thorough analysis effort supported by selected experiments. This approach was needed, since the transport methods then in use had never been applied to geometries that contained so many voids in which neutrons born within the core could so easily stream to regions outside the core. The state-of-the-art at that time was severely tested. These caused a need for development of new codes, these codes would then require testing through experimental data to validate them. Thus in 1975, a program was initiated by both ORNL and GAC, and by the time the U.S. GCFR program was canceled in 1980, six of the eight planned experiments had been completed.

The original GCFR concept was for a 300-MW(e) demonstration plant containing a fueled region 39.2 in. high containing 265 fuel and blanket elements. The fuel and blanket elements were to be suspended from a 24-in.-thick steel grid plate above the core. The fuel rods (270/element) would contain mixed plutonium-uranium oxide pellets, with pellets of depleted uranium on each end of the rods to form the axial blanket. The radial blanket around the fuel could be pellets of uranium or thorium oxide. The system would be cooled by helium flowing downward through the core and then upwards around the outside of the core's thermal shield and out to the steam generator cavity. The reactor and its auxiliary systems were to be contained in a prestressed concrete reactor vessel (PCRV) that had an outside diameter of 84 ft and was 80.5 ft high.

13.2 SINGLE/MULTIPLE-CELL NEUTRON STREAMING EXPERIMENT

In a reactor core having long, small-diameter fuel rods surrounded by low-density coolant as for the GCFR, it was considered important to know what effect the coolant passageways would have on the neutron flow through the core and the resulting damage that could occur in nearby structural components. Since calculational methods at that time could not accurately predict such streaming, it was deemed necessary that an adequate two-dimensional model should be developed in which the neutron streaming paths could be approximated. Before such a model could be applied with confidence to the multiannulus GCFR core, however, it was considered necessary to demonstrate the adequacy of the model itself.

A simple geometry experiment,[2] was proposed in 1975 for this purpose—a mockup that consisted of a single, air-filled pipe, centered in a water-filled tank as seen in Fig. 13.1. An aluminum clad, slightly enriched UO_2 rod or an iron rod was placed inside the pipe. Such an arrangement provided a definite streaming path for the neutrons, thus requiring a proper calculational model for testing different angular quadrature sets in the calculations. Measurements were made behind this mockup to provide the necessary data against which the calculated results could be compared. As usual, the TSR-II source was modified for these measurements.

13.3 NEUTRON STREAMING THROUGH A LATTICE OF GCFR-TYPE FUEL PINS

The results from the single-cell experiment above were not conclusive, in that the comparisons between experiment and calculation failed to indicate a particular calculational model satisfactory for predicting the neutron streaming. However, ORNL, in consultation with General Atomic Corporation expanded the mockup (shown in Fig. 13.2) to include 894 UO_2 pins arranged in a triangular pitch inside an 18-in.-diam steel pipe surrounded by water.[3] Each pin was 54 in. long (48 inches active fuel length) with a nominal diam of 0.0452 in. The pins were preceded by a spectrum modifier to generate a typical fast reactor spectrum. Measurements were made beyond the pin lattice on centerline and at small angles to the centerline. Results showed that the homogeneous and heterogeneous adaptations of the discrete ordinates code bracketed the results from the measurements.

13.4 THE GCFR GRID-PLATE SHIELD DESIGN CONFIRMATION EXPERIMENT

Three years later, after extensive pre-analysis and planning, the next GCFR[4] experiment was conducted at the TSF to provide data against which the validity of the calculational methods used in the design of the grid-plate shield could be tested. The grid-plate shield was positioned directly above the GCFR core in the original reactor design to prevent the radiation damage to the grid plate from reaching established tolerances due to neutrons streaming up the coolant passages in the core. An experiment was needed because analytical methods at that time were not developed sufficiently to calculate streaming in geometries where a high void fraction existed. An experiment using a prototypic mockup of the GCFR fuel pin subassemblies and the proposed grid-plate shield was developed, components were fabricated, and performed.

The experimental configuration, shown in Fig. 13.3, consisted of four basic segments: 1) a concrete shadow shield placed directly in the collimated beam of the TSR-II to limit the

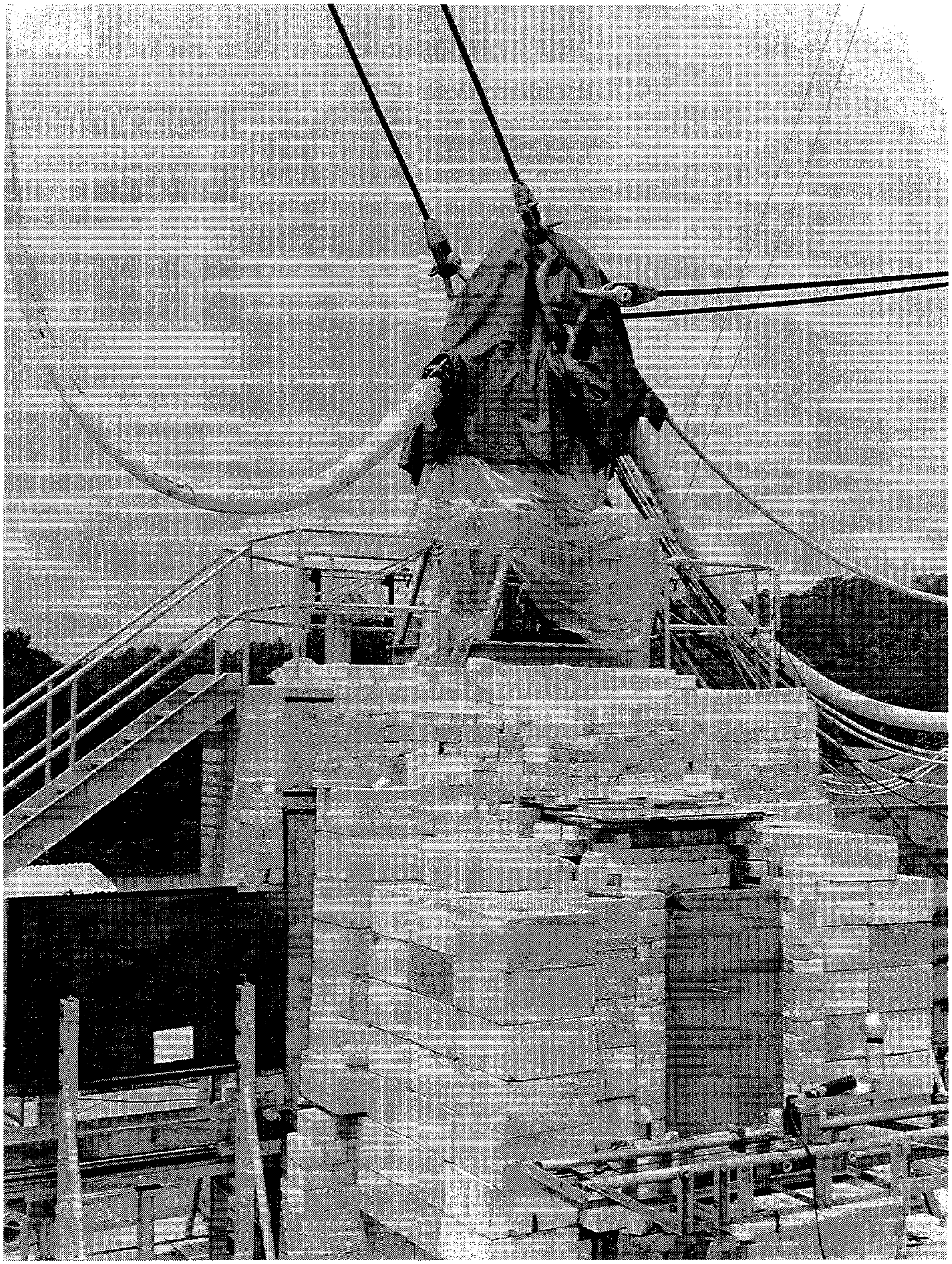

Fig. 13.1 Mockup of GCFR Single-Cell Neutron Streaming Experiment.

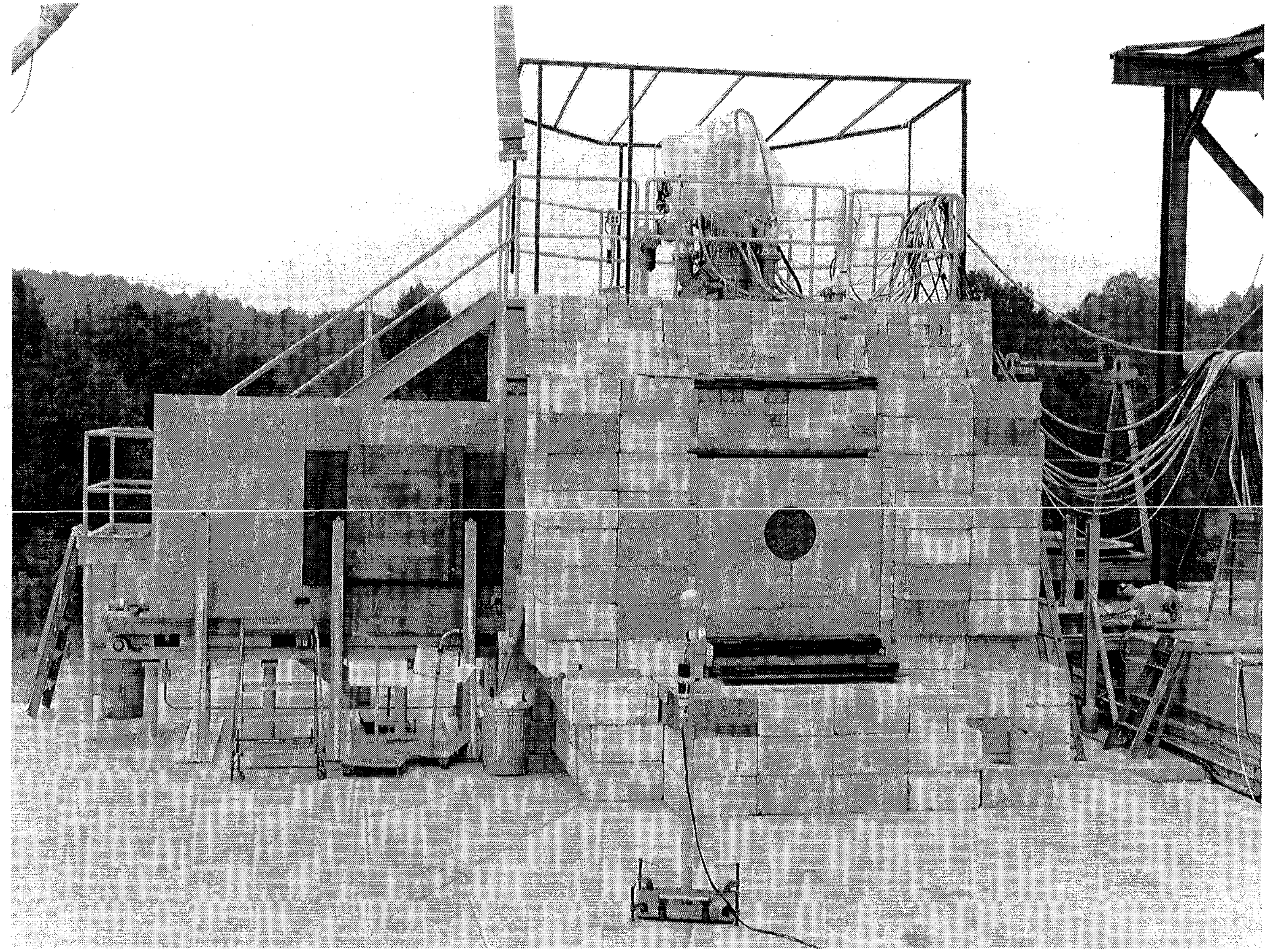

Fig. 13.2 Mockup for Measurement of Neutron Streaming through a Lattice of GCFR-Type Fuel Pins.

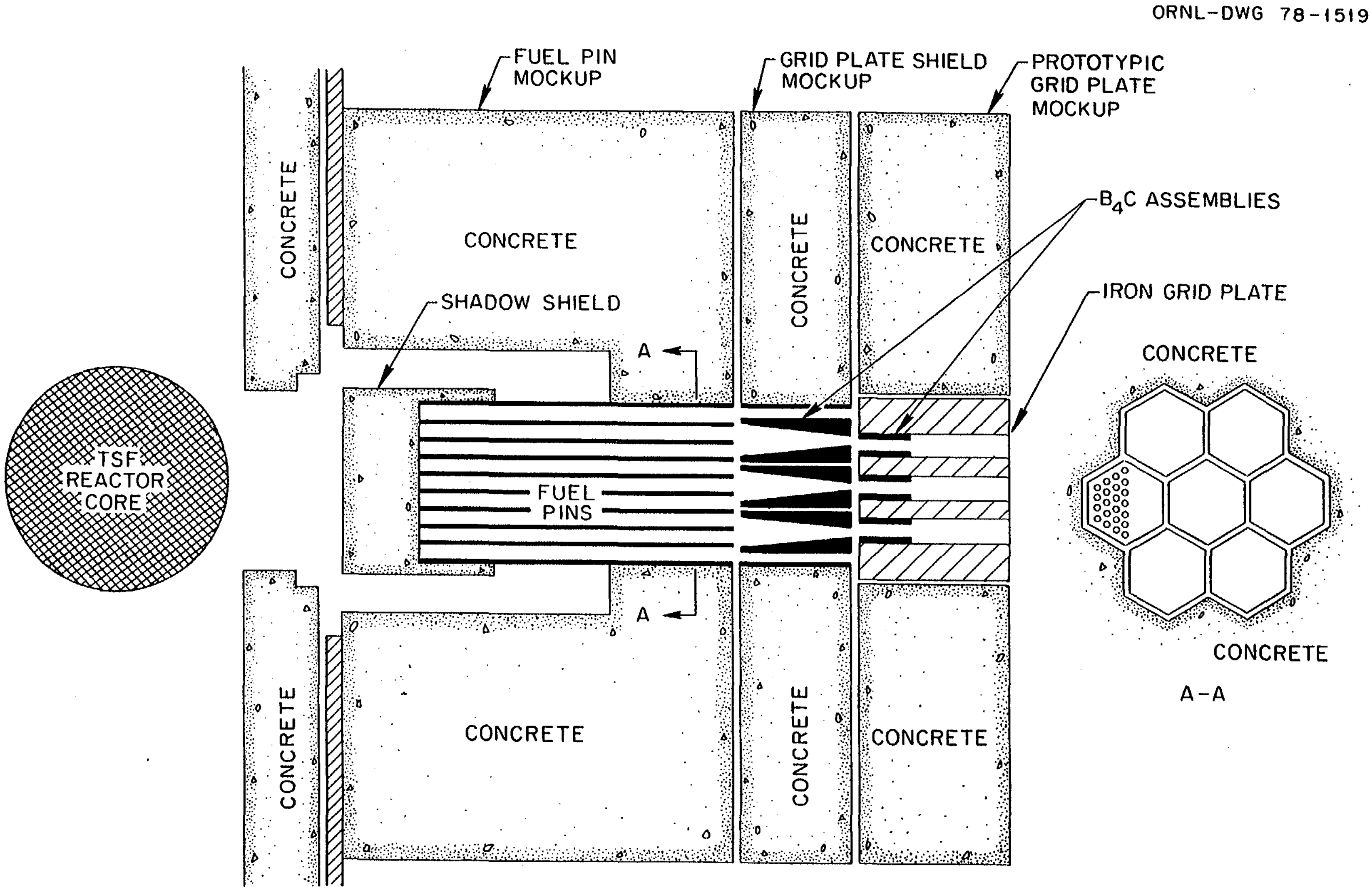

Fig. 13.3 Final Configuration which includes addition of Prototypic Grid Plate.

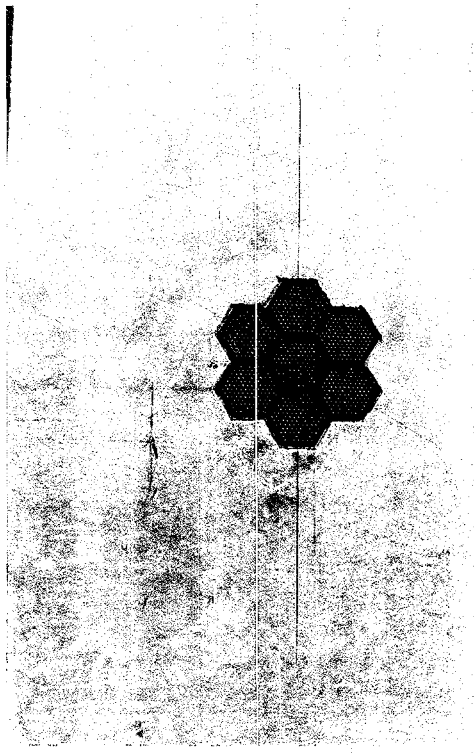

Fig. 13.4 Mockup of the Fuel Pin Subassemblies.

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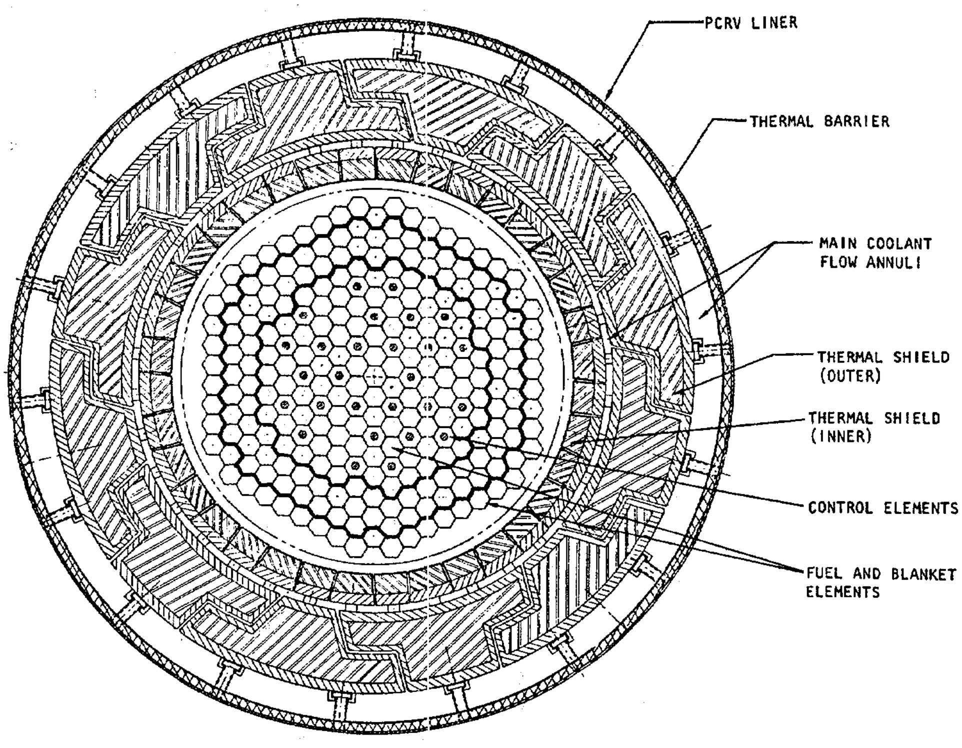

Fig. 13.5 Top View of a Design for the Out Radial Shield.

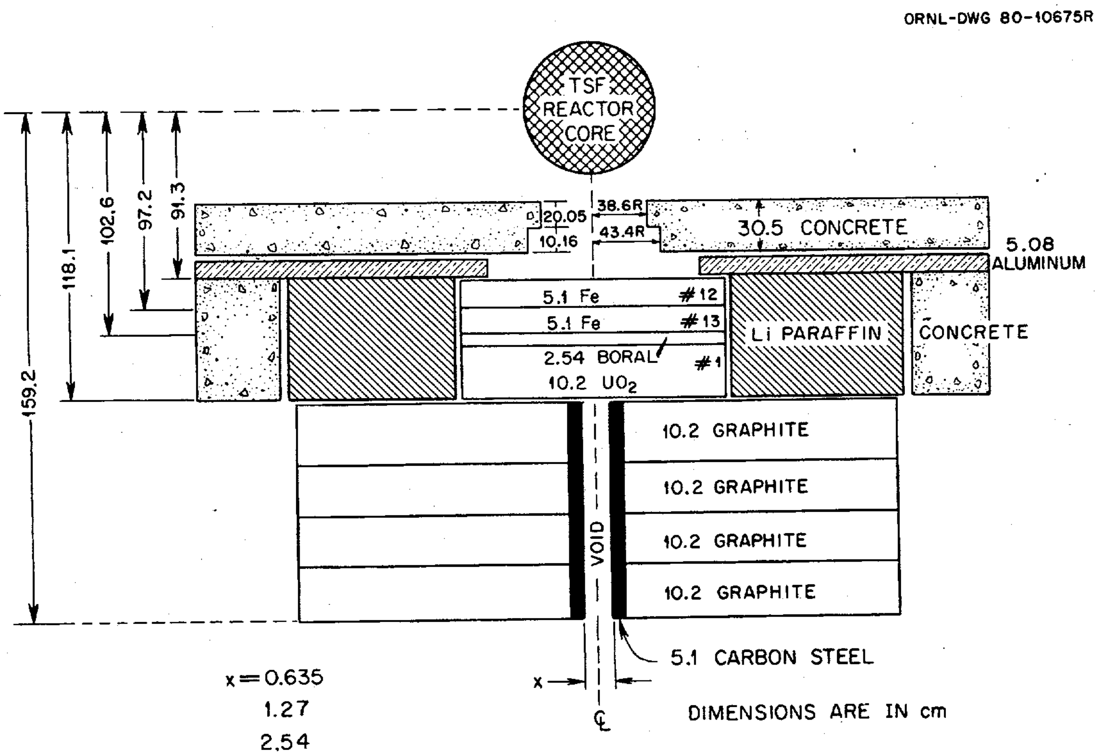

Fig. 13.6 Mockup of Spectrum Modifier and Graphite Shield with Fe Liners - Single Slit (*Items II-A, II-B, II-C, II-D*).

TSR-II neutrons so that the source term would come mainly from the pins beyond the concrete; 2) a simulated GCFR core of fuel pins (pictured in Fig. 13.4) in hexagon subassemblies positioned beyond the shadow shield followed; by 3) a prototypic grid-plate shield, and 4) a prototypic grid plate. The material in the gridplate shield subassemblies was boron carbide (B_4C) and the grid plate was made of iron with partial (B_4C) inserts. The individual segments were introduced into the configuration sequentially, with measurements after each segment. The variables in the experiment were the number of fuel pins per subassembly and the spacings between subassemblies.

13.5 GCFR RADIAL BLANKET AND SHIELD EXPERIMENT

The purpose of this experiment was to study the effects of neutrons leaving the core in a radial direction, passing through thick and dense laminated regions, and reaching the prestressed concrete reactor vessel.[5] Specifically, the experiment was intended to verify the calculations used in determining the effectiveness of the radial shield components in protecting the prestressed concrete reactor vessel from radiation damage. The shielding consisted of an inner and outer shield, with the design calling for the inner shield to be removable. The inner shield was to be boronated graphite and SS, the outer row to be graphite bounded by boronated graphite and SS. The shields were designed to limit the total lifetime fluence at any point in the PCRV to less than 1 x 10^{19} neutrons per square centimeter.

The scope of the experiment included neutron penetration studies through both the ThO_2 and UO_2 as radial blankets, with emphasis primarily on the ThO_2, since no deep penetrating data existed for THO_2. The measurements helped establish the uncertainty in relevant cross section data and provided the bias factors for the calculated source term for the radial shield. The ThO_2 slabs were fabricated at National Lead by filling SS containers with ThO_2 powder; the slabs providing up to four blanket rows for testing. The UO_2 slabs, fabricated for earlier LMFBR experiments, were iron vessels filled with UO_2 pellets surrounded by aluminum and sodium, (Sect. 12.11).

The spectrum modifier was followed by three segments of a typical mockup that included the radial blanket, inner radial shield, and the outer radial shield. Both neutron and gamma-ray fluxes were measured as segments were added.

13.6 GAS-COOLED FAST BREEDER REACTOR PLENUM SHIELD DESIGN - SHIELD HETEROGENEITY EXPERIMENT

This experiment focused on neutron streaming issues related to the outer radial shield, which also extended into the upper and lower plena.[6] The measurements for this experiment could be thought of as a refinement on the studies made in the previous experiment (Sect. 13.5), in which the radial shield was modeled with slabs of graphite and steel.

In the actual design of the reactor, (Fig. 13.5), the outer radial shield consisted of several interwoven pieces of graphite and borated graphite enclosed in stainless steel envelopes. In this experiment, the concern was centered on possible neutron streaming between abutting pieces of SS and SS with graphite, a concern deep enough to warrant measurements behind mockups typical of those that would be present in the reactor shield. However, to simplify the geometry for the experiment, that portion of the reactor shield was modeled with solid layers of graphite, or graphite plus a steel-lined slit (Fig. 13.6).

The experimental program was divided into several phases: (1) mockups using a single slit with varying widths in graphite only, (2) a mockup where the slit in the graphite was lined with several inches of iron to model the stainless steel envelope slits, (3) mockups having three slits, one between the iron pieces as in (2) and one between each iron piece and adjacent graphite to model the SS-graphite abutments, and (4) mockups with a single offset through graphite only.

13.7 GAS-COOLED FAST BREEDER REACTOR PLENUM SHIELD DESIGN - EXIT SHIELD EXPERIMENT

This experiment was the sixth and final experiment performed at the TSF as a part of the five-year integral shield testing program for the GCFR.[7] The exit shield was designed to be an integral part of the fuel element, with the shield concentrated along the fuel element axis, leaving a surrounding coolant path (mockup in Fig. 13.7). The design was intended to minimize the neutron streaming into the upper plenum while still providing sufficient space for coolant flow. There was also a major concern about the validity of the calculations for predicting neutron streaming through the exit shield in the vicinity of a control rod. The control rod mockup was divided into equal segments so that a study could be made of the neutron streaming within the exit shield for three different axial positions of the control rod: (1) positioned within the core, (2) positioned within the core and axial blanket, and

(3) positioned within the blanket and exit shield. Measurements were also made with and without B_4C in the subassembly liner.

The experimental configuration consisted of four basic segments: (1) the concrete shadow shield used for earlier GCFR experiments; (2) a simulated GCFR core directly behind the shadow shield; (3) a cross section of a prototypic exit shield with the control rod; and (4) a cross section of the shield without the control rod in the center subassembly (Fig. 13.8).

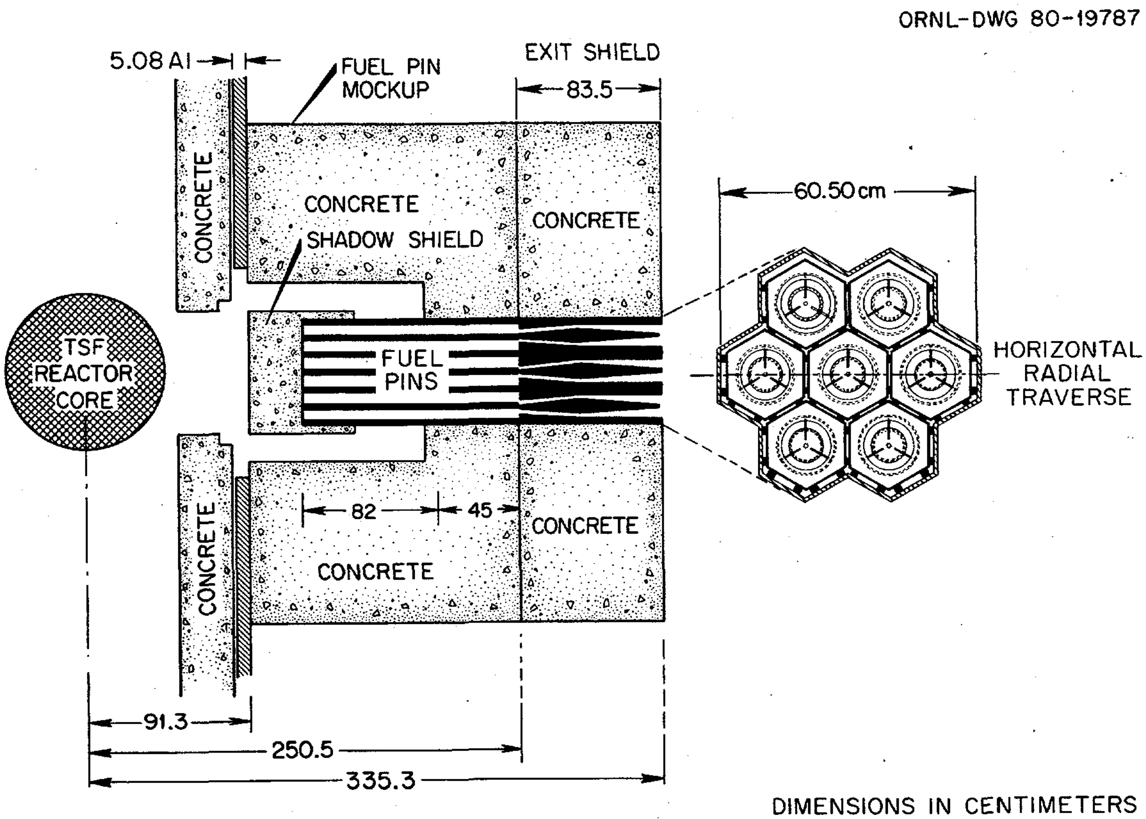

Fig. 13.7 Experimental Configuration consisting of Seven Fuel Pin Subassemblies and Seven Exit Shield Subassemblies (Configuration II.A).

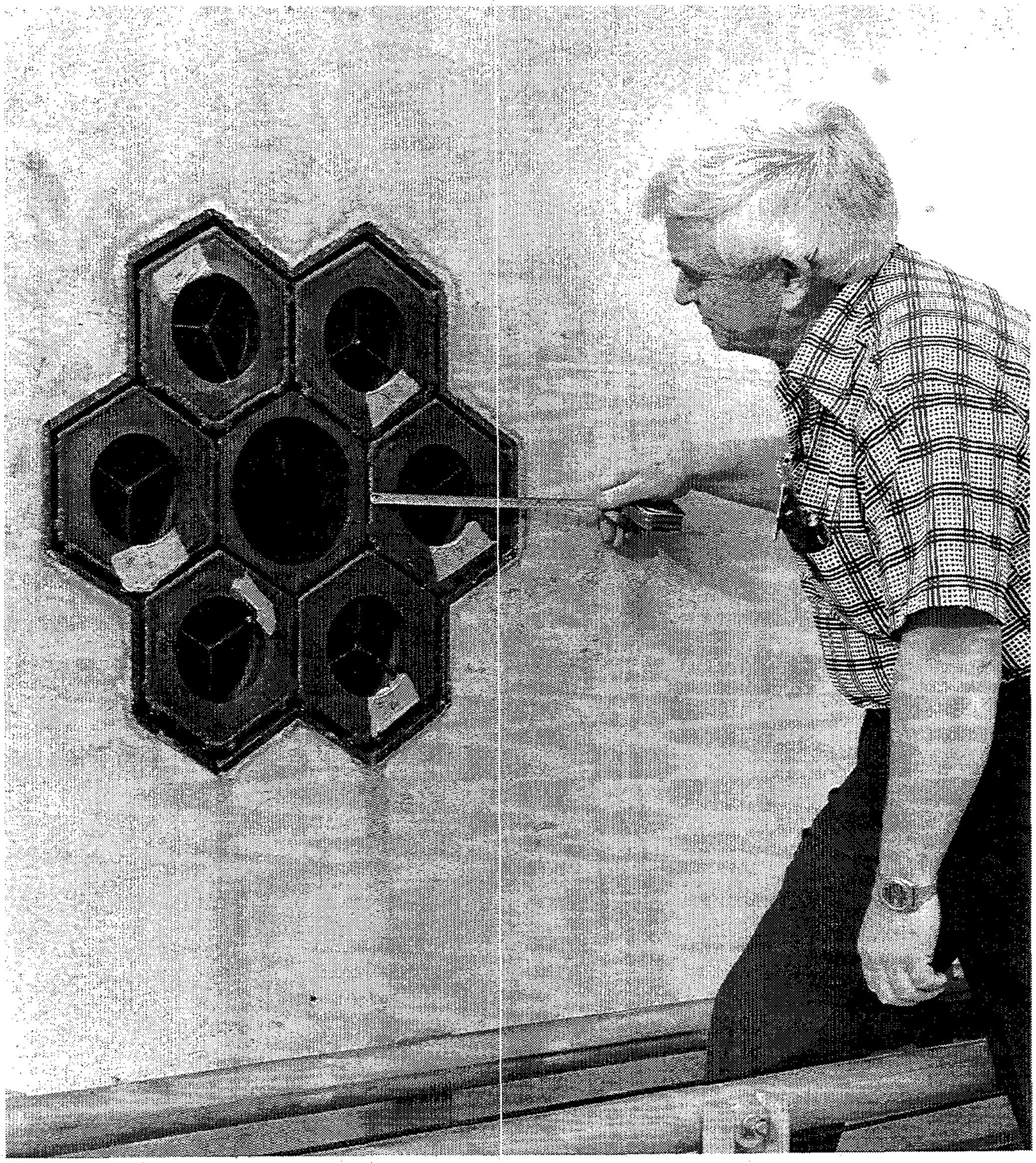

Fig. 13.8 Mockup of Exit Shield (Six Subassemblies) with Central Control Rod Subassembly Housing.

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14.0 HIGH TEMPERATURE GAS-COOLED REACTOR PROGRAM

14.1 INTRODUCTION

General Atomic Technologies' 1984 design for their 900-MW(e) High Temperature Gas-Cooled Reactor[1] (HTGR) contained an array of thousands of small diameter coolant passageways immediately below the core which merged to become several large diameter holes penetrating the core support blocks and emptying into the lower plenum region. The helium coolant gas passing through these holes offered very little resistance to neutron flow, and this flow was seen as possibly a means whereby the neutron flux reaching the lower plenum could exceed the design criteria. Calculations were made, but the geometry of the region was quite complex, and there were concerns that calculations might under predict the neutron flux arriving in the plenum region.

Thus, it was recommended that the design be tested at the TSF to verify the calculational methods and possibly the cross-section data. The experiment simulated the regions modeled in the calculations, except for the differences in the small coolant-hole and boron-pin patterns. The experiment was divided into two phases. Phase I studied the neutron streaming through the upper boron pin layer for two different loadings of boron and portions of the large coolant holes that preceded the lower boron pin layer. Phase II concentrated on the neutron passage through the lower boron pin layer followed by the support blocks.

14.2 PHASE I HIGH TEMPERATURE GAS-COOLED BOTTOM REFLECTOR AND CORE SUPPORT BLOCK NEUTRON-STREAMING EXPERIMENT

Phase I of the experiment was performed during FY 1983. In this phase, the first four of eight segments that comprise the full experimental mockup were utilized. Fig. 14.1 shows a schematic of the experimental configuration containing the four separate graphite pieces; identified as the upper boron pin layer, a reflector layer, crossover layer and a follow-on layer. The same figure shows the spectrum modifier of iron and graphite preceding the mockup which was used to alter the TSR-II spectrum to more nearly resemble the neutron spectrum emerging from the bottom of the HTGR core. A picture of the seven hexagons of pins and/or coolant holes emerging from the face of the upper pin layer is shown in Fig. 14.2. A picture from behind the last section used in Phase I is shown in Fig. 14.3.

The measurements were designed to determine the extent of neutron streaming through the upper portion of the bottom reflector, which then provided a measure of the flux level incident on the lower plenum region, and to investigate the effectiveness of borated graphite pins in reducing the neutron streaming. In part one of this phase, the number of graphite and borated graphite pins in the upper pin layer were nearly equal. In part two, the graphite pins were replaced with borated graphite. Data were obtained behind each successive layer of graphite as they were added to the mockup for each of the two pin arrangements.

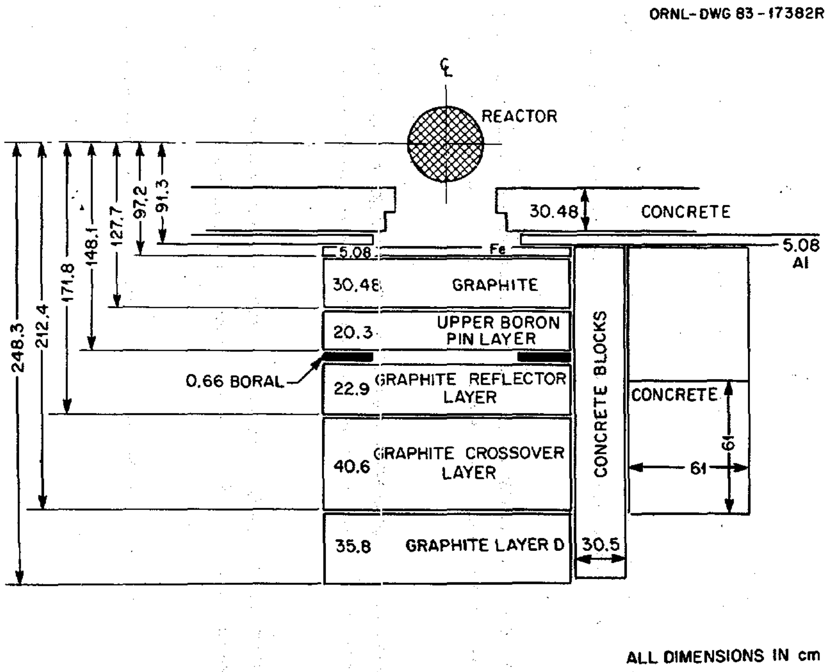

Fig.14.1 Schematic of Full Mockup: SM + Upper Boron Pin Lay + Boral Shroud + Reflector Lay + Crossover Layer + Follow-on Layer.
Note: Concrete Covers Lateral Sides of Configuration.

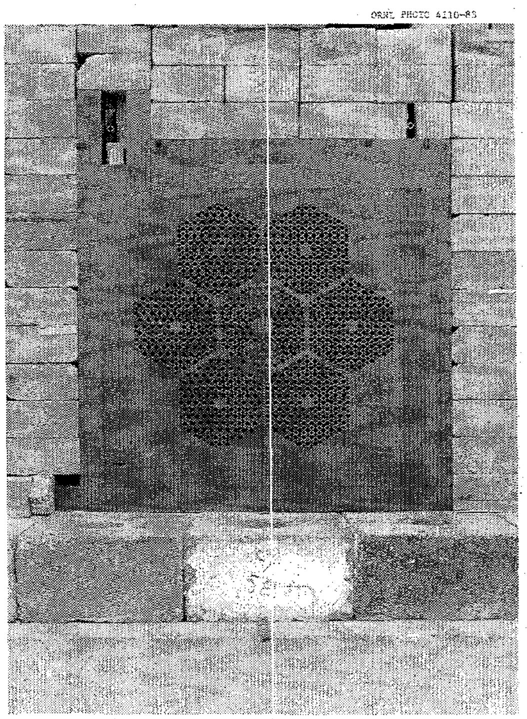

Fig. 14.2 Photograph of Upper Boron Pin Layers.

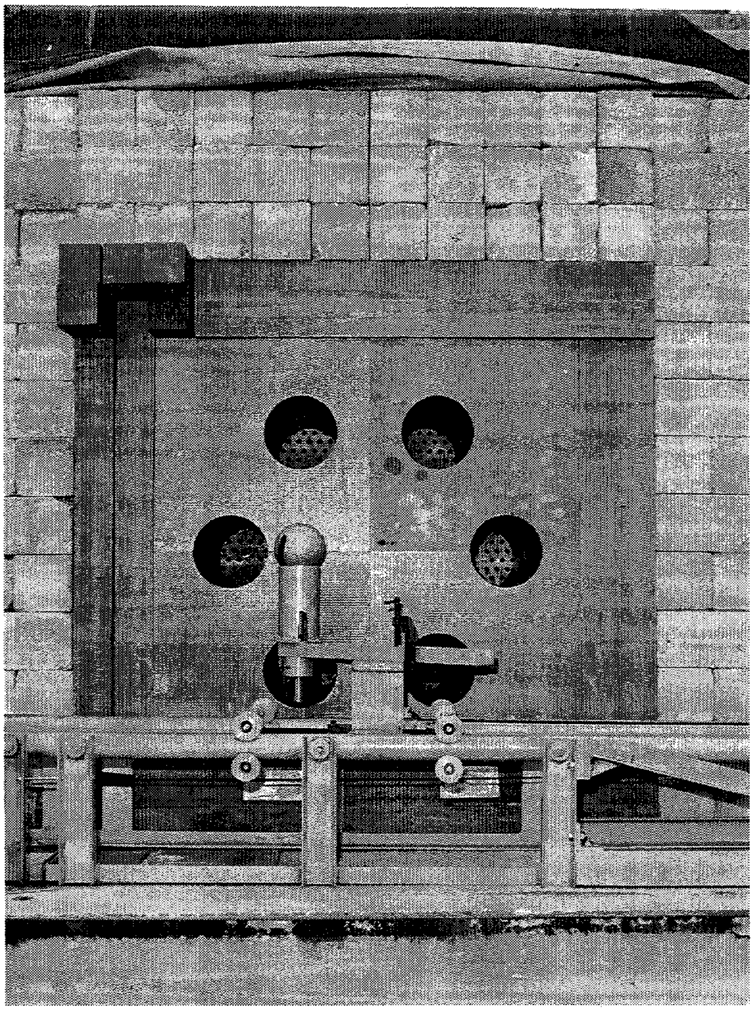

Fig. 14.3 Photograph of Follow-on Lay.

14.3 PHASE II HTGR BOTTOM AND CORE SUPPORT BLOCK NEUTRON-STREAMING EXPERIMENT

In 1984, the second phase of the measurements[3] was performed, involving the last four of the eight segments that comprised the full experimental mockup. These were: (1) the lower boron pin layer, (2) a transition layer in which six coolant holes merged into one large hole (shown in Fig. 14.4), (3) a layer for continuation of the large coolant hole, and (4) a mockup for the core support floor and prestressed concrete reactor vessel (PCRV). These segments were preceded by the first four layers and the spectrum modifier used in Phase I, as the arrangement shows in Fig. 14.5. Data was obtained for each segment as it was added as in Phase I. Although this series of measurements was relatively extensive in terms of planning, fabrication, and measurements, it was the only experiment in support of the calculational methods used in the design of General Atomic's HTGR.

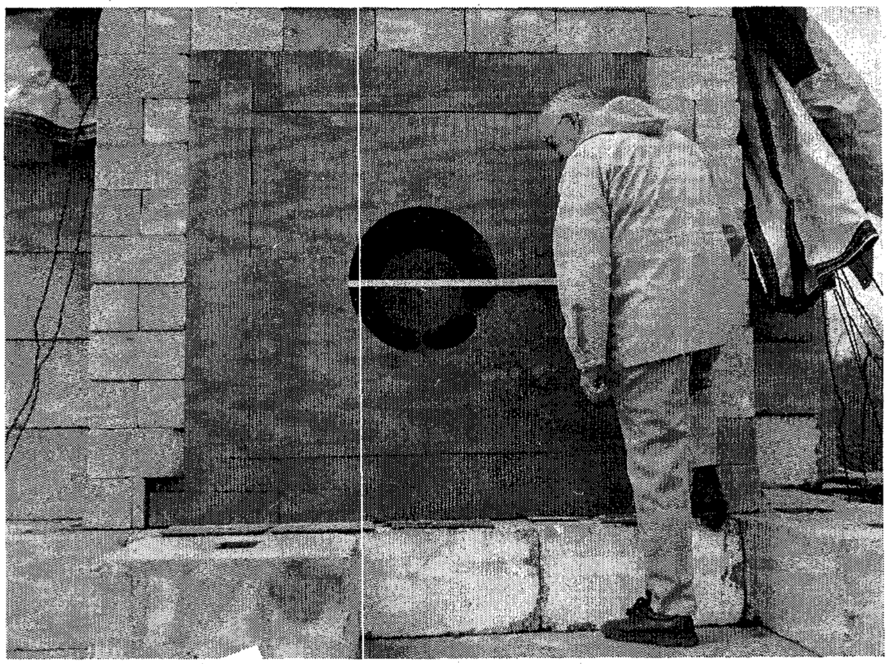

Fig. 14.4 Measurement of Transition Layer Void Diameter by staff member J. J. Manning.

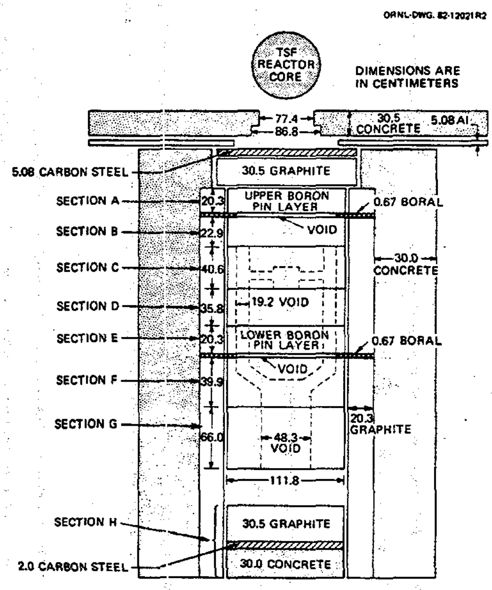

Fig. 14.5 Plan View of the Full Configuration for the HTGR Bottom Reflector and Core Support Block Neutron - Streaming Experiment. Phase I Measurements involved sections A - D, and Phase II Measurements involved sections E - H.

14.4 REFERENCES

1. C. O. Slater, "Analysis of Phase I of the HTGR Bottom Reflector and Core Support Block Neutron-Streaming Experiment," ORNL/TM-9252 (July 1984).

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15.0 JAPANESE-AMERICAN SHIELDING PROGRAM OF EXPERIMENTAL RESEARCH: JASPER

15.1 INTRODUCTION

In September 1985, an agreement was signed between the United States Department of Energy (DOE) and the Japanese Power Reactor and Nuclear Fuel Development Corporation (PNC) to perform a series of experiments at the TSF.[1] The Japanese-American Shielding Program of Experimental Research (JASPER) was a cooperative effort designed to meet the needs of both participants in support of the Advanced Liquid-Metal-Cooled Reactor Program. All of the experiments performed were planned jointly and derived through an iterative process of submitted proposals and consensus.

The experiments, listed in the order of their performance, included studies of: (1) the attenuation in several radial shield mockups, (2) neutron streaming in a fission gas plenum, (3) shield effectiveness of different axial shield designs, (4) source multiplication effects within in-vessel stored fuel assemblies, (5) sodium activation in an intermediate heat exchanger mockup, (6) neutron streaming in narrow gaps, (7) the feasibility of an in-vessel flux monitor, and (8) the attenuation properties of special materials.

The first experiment was started in March 1986 and completed the same year. The second one, the fission gas plenum experiment, was relatively short, starting in January of the next year (1987), and finishing by the end of February. The third experiment, a study of the effectiveness of different axial shield designs, was planned to begin the first part of April, but was delayed since all of the reactors at ORNL were shut down the last week of March by orders of the DOE-Oak Ridge Operations office. For nearly a year, the fate of the facility and the JASPER program were in limbo until March 26, 1988, when orders were received to permanently dismantle the facility. Such was not to be the case, however, due in part to pressure asserted by PNC to complete JASPER. After an interruption of nearly three years, permission was given to restart the reactor on November 30, 1989.

The JASPER program was actually restarted on December 11, 1989, with a single run, but measurements over the next several months were sporadic as the reactor experienced some operating problems. It was shut down on the first of March, 1990, for repairs, and would not start again until nearly the end of July. Following a series of preliminary runs, the axial shield mockup was once again placed in the reactor beam on September 10. Measurements were completed for that phase of the program in early January 1991. It was followed immediately by measurements for the In-Vessel Fuel Storage (IVFS) and Intermediate Heat Exchanger (IHX) experiments. A part of the IHX measurements were included in the IVFS experiment since both programs required the use of the same vessel of pins. These two programs were completed by the middle of February 1992.

Measurements for the next three experiments, namely, the Gap-Streaming, Flux Monitor, and Special Materials studies, were completed by the end of July 1992. The final two months of the program, August and September, were spent making additional measurements for the Axial Shield and IVFS phases of the program. These re-runs were completed before the end of September, 1992, bringing the JASPER program to a close.

Subsequently, word was again received to prepare the TSF for standby, with eventual dismantling of the reactor core projected for 1995.

15.2 JASPER PROGRAM RADIAL SHIELD ATTENUATION EXPERIMENT

This experiment was the first in a series of six experiments planned for the TSF as part of a cooperative effort between the United States Department of Energy and the Japan Power Reactor and Nuclear Fuel Development Corporation.[2] The experiment was conducted to: (1) provide validation data for calculating the shield effectiveness of combinations of SS, graphite, and boron carbide shield designs, (2) verify the accuracy of related radiation transport methods and nuclear data, and (3) substantiate the effectiveness of shield design currently proposed by advanced LMR designers in Japan and the United States.

Nine separate investigations were proposed, but only seven were completed in the allotted time of eight months. Two different spectrum modifiers were used to provide a near-core source spectrum and a source spectrum near the internal heat exchanger. The mockups tested contained slabs of SS, graphite, boron carbide, boral, sodium, and aluminum, a typical mockup of which is shown in Fig. 15.1. Measurements were made behind successive layers of one or more slabs as they were added to the configurations.

15.3 JASPER PROGRAM FISSION GAS PLENUM EXPERIMENT

The second JASPER experiment was designed to investigate neutron transmission through heterogeneous and homogeneous mockups[3] of the upper gas plenum region designed for use with the advanced reactor concepts. The data provided verification of the assumptions and calculational methods used to determine the leakage from the plenum and provided associated uncertainty evaluations.

The program was comparatively short, the mockups to be investigated consisted of only two thicknesses for each of the heterogeneous or homogeneous mockups. The heterogeneous plenum consisted of a 22-cm-diam arrangement of 512 SS tubes passing through sheets of aluminum as shown in Fig. 15.2. The plenum was then surrounded by concrete as shown in Fig. 15.3. The homogeneous plenums were made of solid plates with void spacings.

15.4 JASPER PROGRAM AXIAL SHIELD EXPERIMENT

The Axial Shield Experiment[4] was programmed to extend the studies of the different axial shield designs beyond the fission gas plenum while providing a comparison of the neutron attenuation characteristics of boron carbide and stainless steel (SS) as they were integrated into the designs. The configurations studied contained hexagons having three different internal geometrical designs that consisted of: (1) a central blockage type in which the coolant flowed around a central shield plug; (2) a rod bundle type in which the shield material was in the form of small rods spaced for coolant flow; and (3) an annular type in which the coolant flowed centrally through the fuel assembly. The two shield materials were boron carbide and stainless steel, with the sodium coolant modeled by aluminum. The photograph in Fig. 15.4 displays the array and their overall width, as seen by N. Ohtani of Japan's PNC. In Fig. 15.5, B. D. Rooney provides a display of the various individual assemblies.

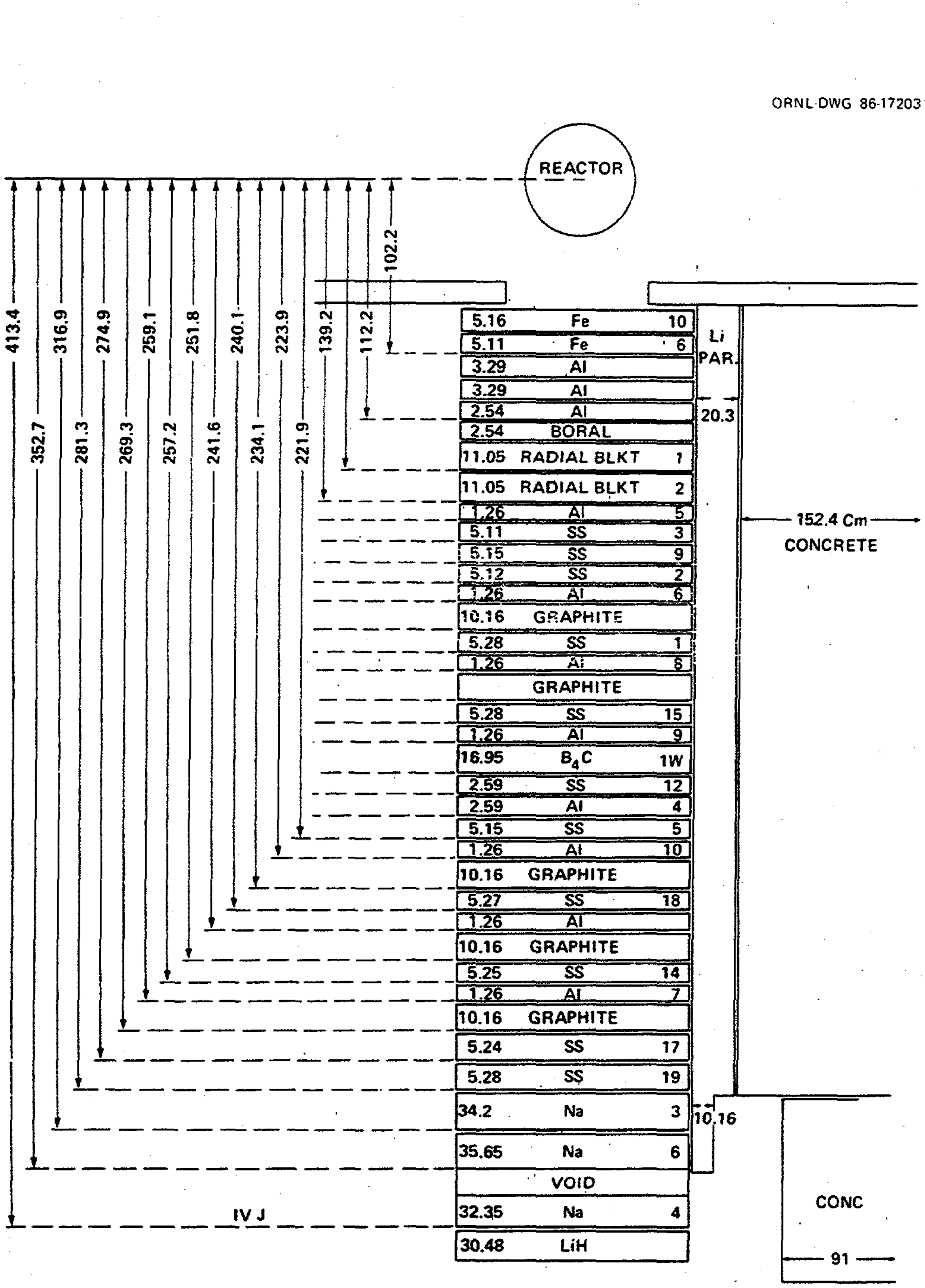


Fig. 15.1 Schematic of SM1 plus Shield Configuration for item IVJ. Note: Lithiated Paraffin covers lateral sides of configuration.

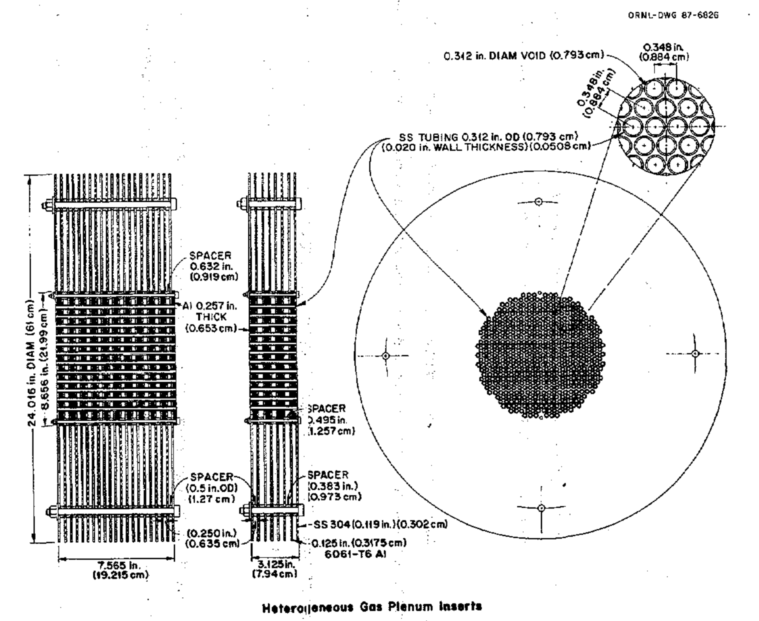


Fig. 15.2 Schematic of Heterogeneous Fission Gas Plenums (Items IV, V).

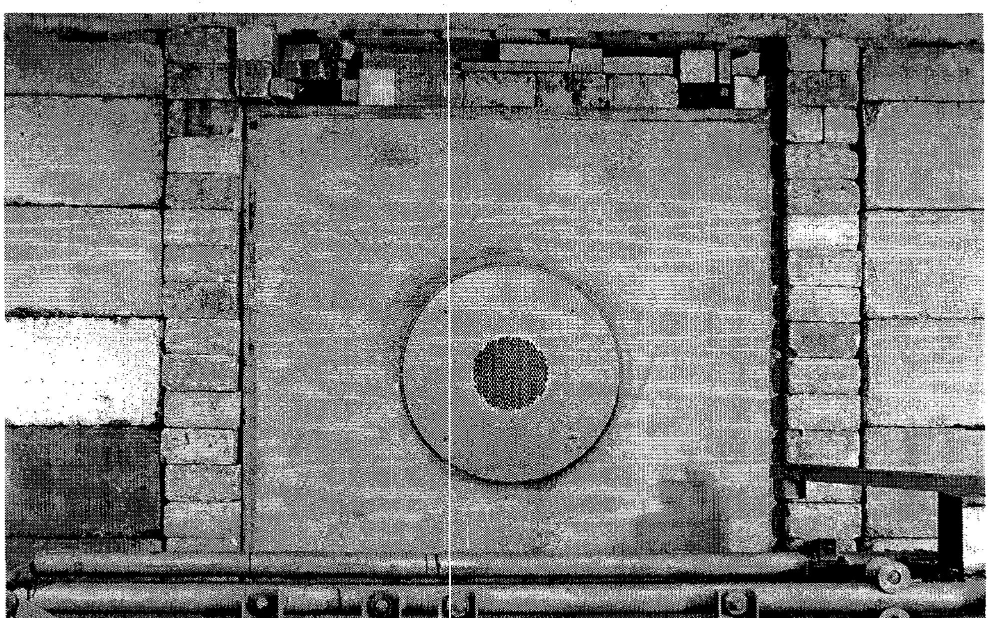


Fig. 15.3 Photograph of 8-cm Heterogeneous Fission Gas Plenum Mockup (Item IV).

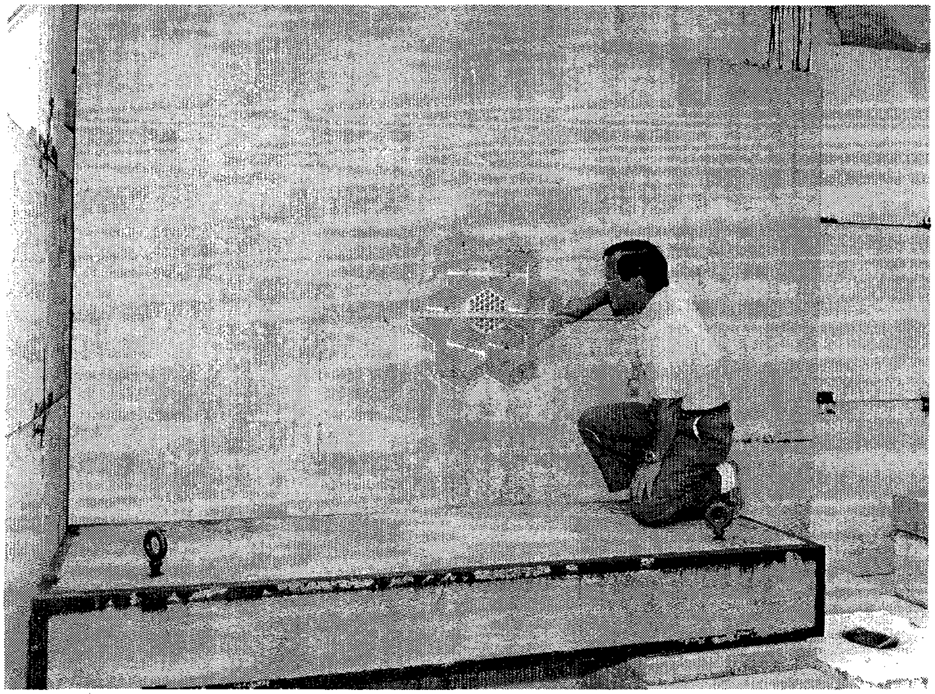


Fig. 15.4 Photograph of a typical Axial Shield Mockup.

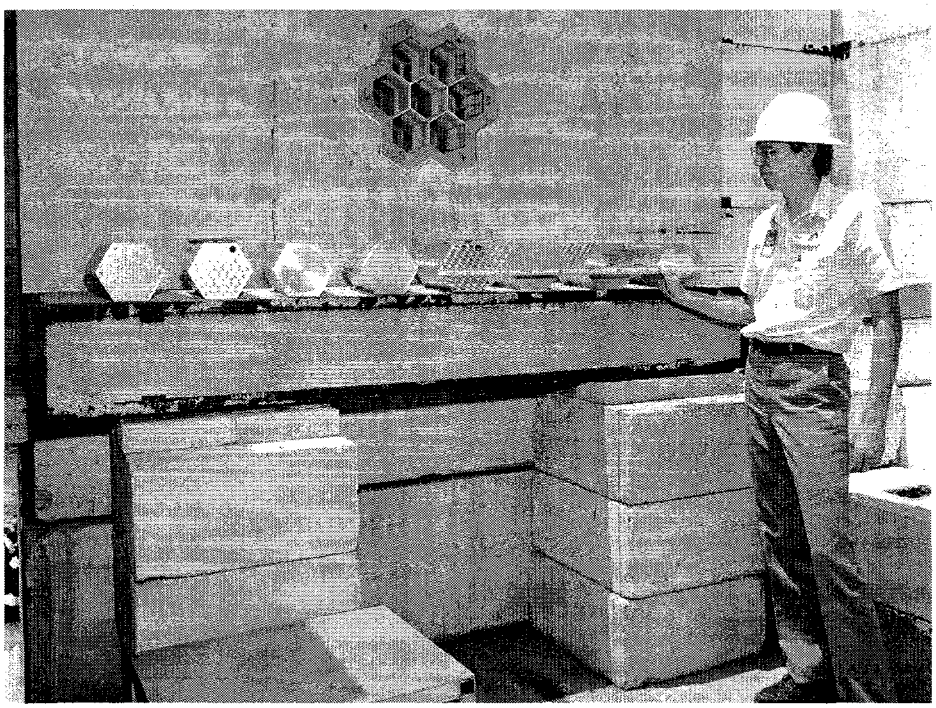


Fig. 15.5 A Display of the various Hexagonal Segments forming the Axial Shield.

15.5 JASPER PROGRAM IN-VESSEL FUEL STORAGE EXPERIMENT

Presence of stored fuel within the reactor vessel creates significant shielding and flux monitoring problems for designers of Liquid-Metal-Cooled Reactor (LMR) systems. Performance of previous experiments of this type at the TSF (not for this program) demonstrated the difficulty in determining the multiplication of neutrons within the fuel array. This experiment[5] was designed to once again attempt such a study with the hope of resolving the difficulty.

Fuel pins containing 4.81 wt% of enriched ^{235}U were used to model the fissile material planned for storage within a LMR vessel. Three different pin arrays were studied: (1) a thick slab designed to provide an array of the maximum number of pins available (1148 pins); (2) a heterogeneous arrangement of three bundles separated by aluminum (sodium substitute) to represent individual fuel assemblies; and (3) a so-called homogeneous slab containing nearly the same number of pins as in the heterogeneous slab, but providing a more uniform distribution of pins and aluminum in slab forms. Schematics of the three fuel pin assemblies are given in Fig. 15.6.

15.6 JASPER PROGRAM INTERMEDIATE HEAT EXCHANGER EXPERIMENT

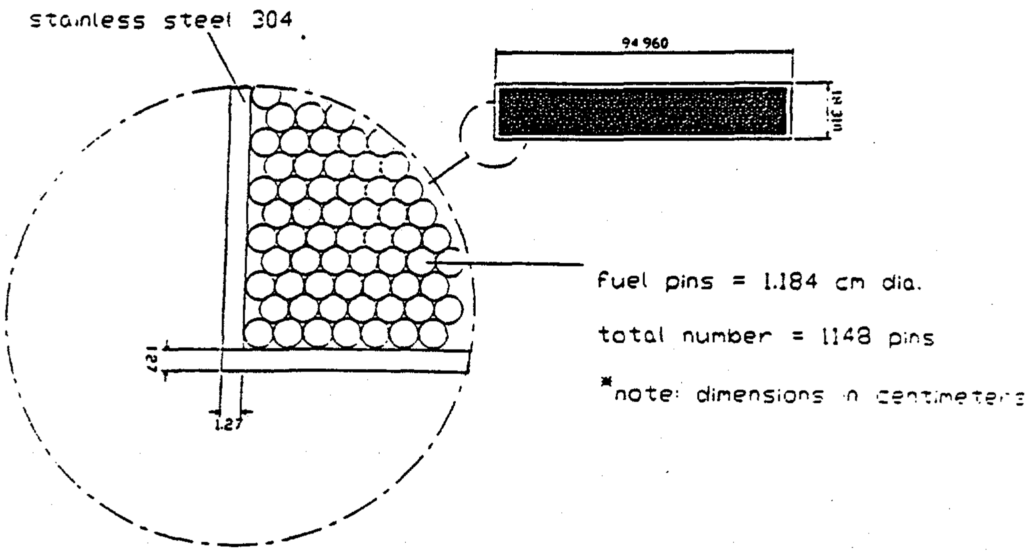
Calculation and prediction of the secondary coolant activation in a LMR system is a difficult problem because of the associated structure complexities and the many neutron transport paths that may be available in that area. Thus this experiment was designed to investigate and establish activation rate distributions within different intermediate heat exchanger (IHX) mockups. Configurations for three different locations of the IHX were studied for the U.S. part of the program: below the stored fuel, in the area of the stored fuel, and in the Gas Plenum region of the stored fuel. The Japanese considered only one area for their IHX vessel; surrounded by B_4C located radially away from the reactor core.

A typical mockup for the U.S. series of measurements is shown in Fig. 15.7. Fig. 15.8 shows A. Shono, a representative from PNC, attached to the TSF staff, setting up a Bonner ball detector behind one of the mockups for the Japanese part of the program.

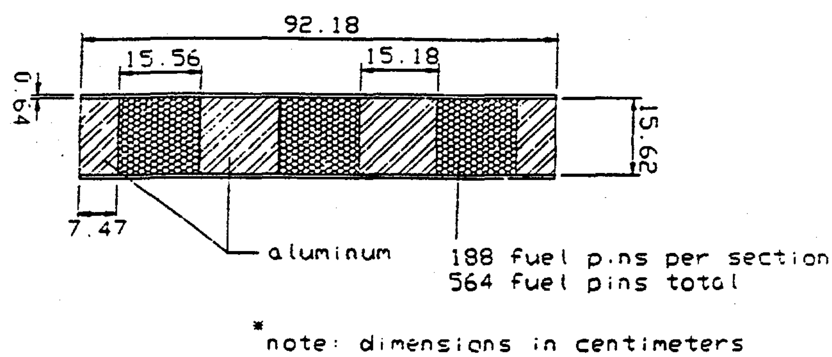
15.7 JASPER PROGRAM GAP STREAMING EXPERIMENT

This experiment[7] was designed to investigate neutron streaming in annular gaps as a function of gap width and/or gap offset that were representative of spacings between various components in the reactor enclosure system. Such measurements are important because the neutron dose rate above the reactor vessels usually results from neutron streaming in these voids. Again, the data was intended to provide a base against which the calculational methods may be tested, hopefully with reduced uncertainties.

Two iron-lined, concrete filled vessels were fabricated, one of solid concrete and the other containing a central cylinder of concrete surrounded by an annular void followed by additional concrete. Cylindrical sleeves of iron and/or concrete were then inserted into the gaps to vary the width of the void and/or generate single offsets. A schematic of the basic components is shown in Fig. 15.9, and a picture of the exit face of a mockup can be seen in Fig. 15.10.



Schematic of thick IVFS mockup (slab #1).



Schematic of heterogeneous IVFS mockup (slab #2).

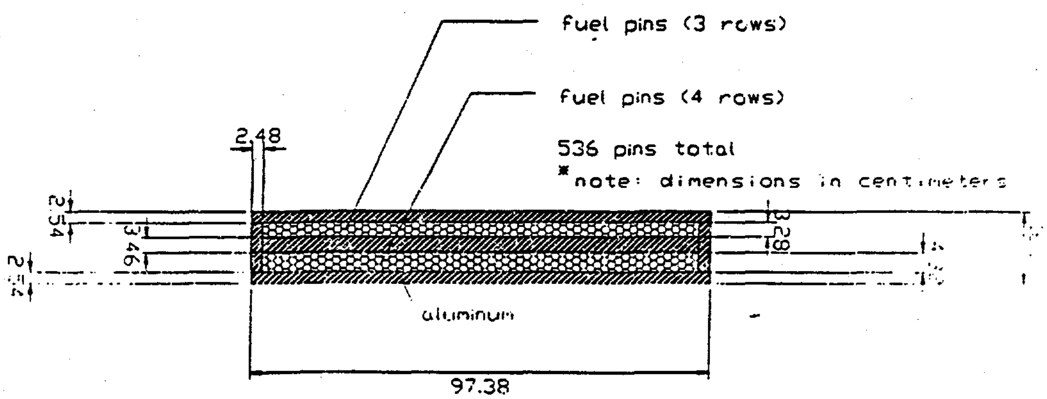


Fig. 15.6 Schematics of the Thick, Heterogeneous, and Homogeneous IVFS Mockups, reading top to bottom.

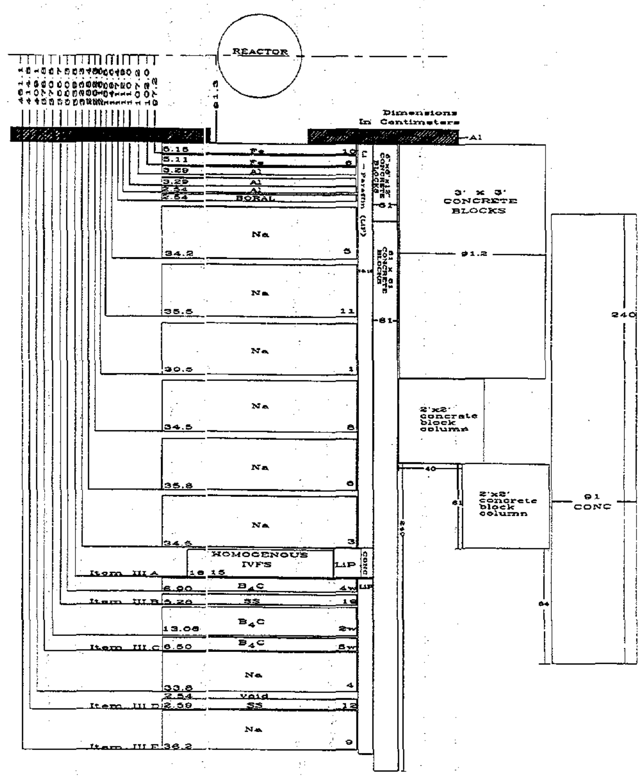


Fig. 15.7 Schematic of SM-2 plus Shield Configuration for Item III.

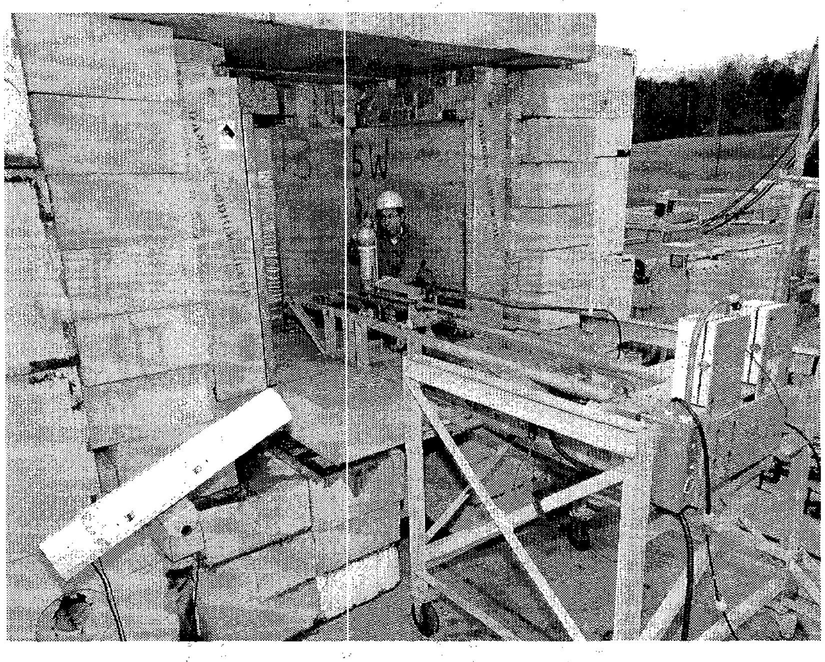


Fig. 15.8 A photograph of a typical IHX Shield Mockup.

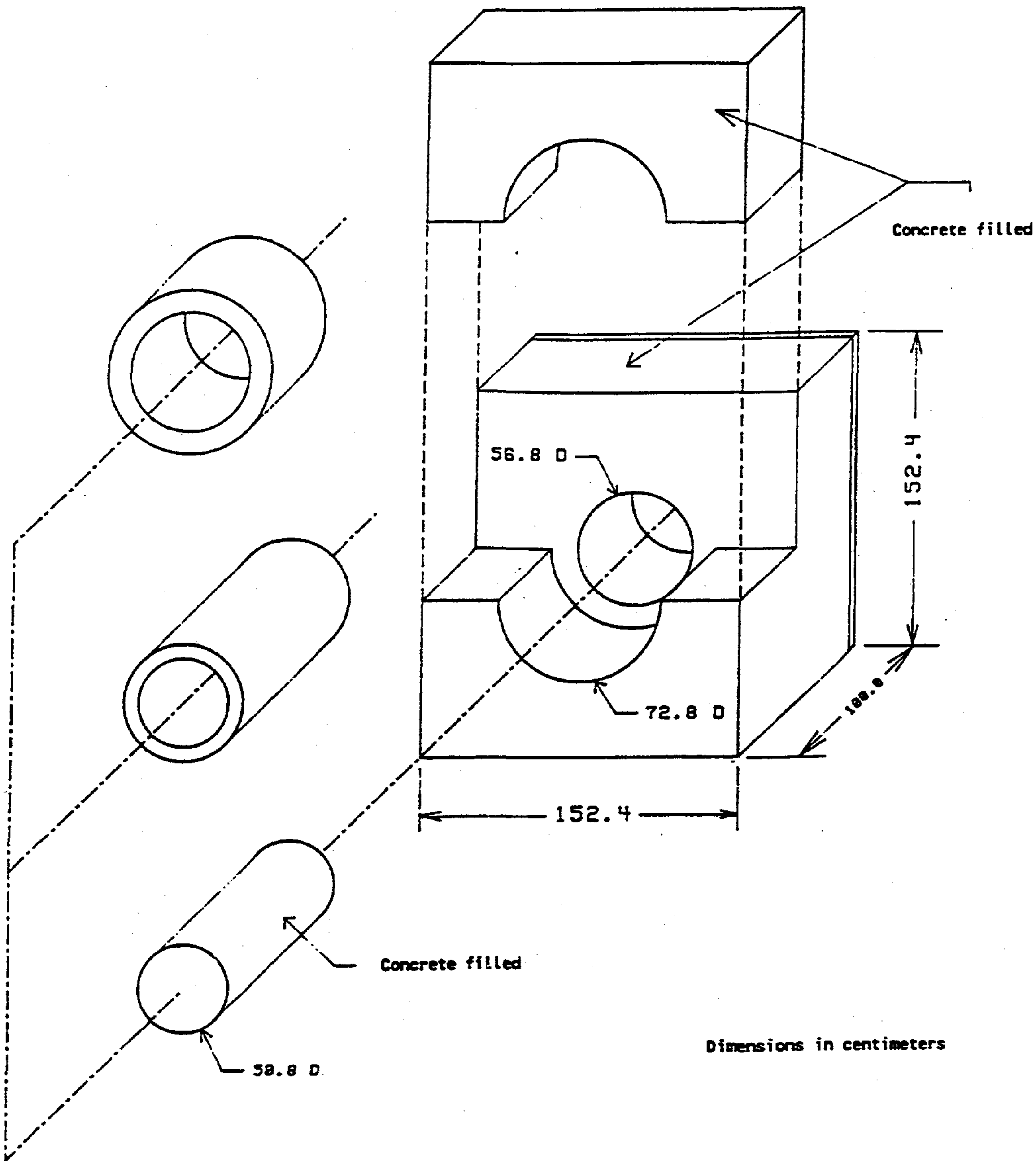


Fig.15.9 Basic Components used to construct the configurations for the Gap Streaming experiment.

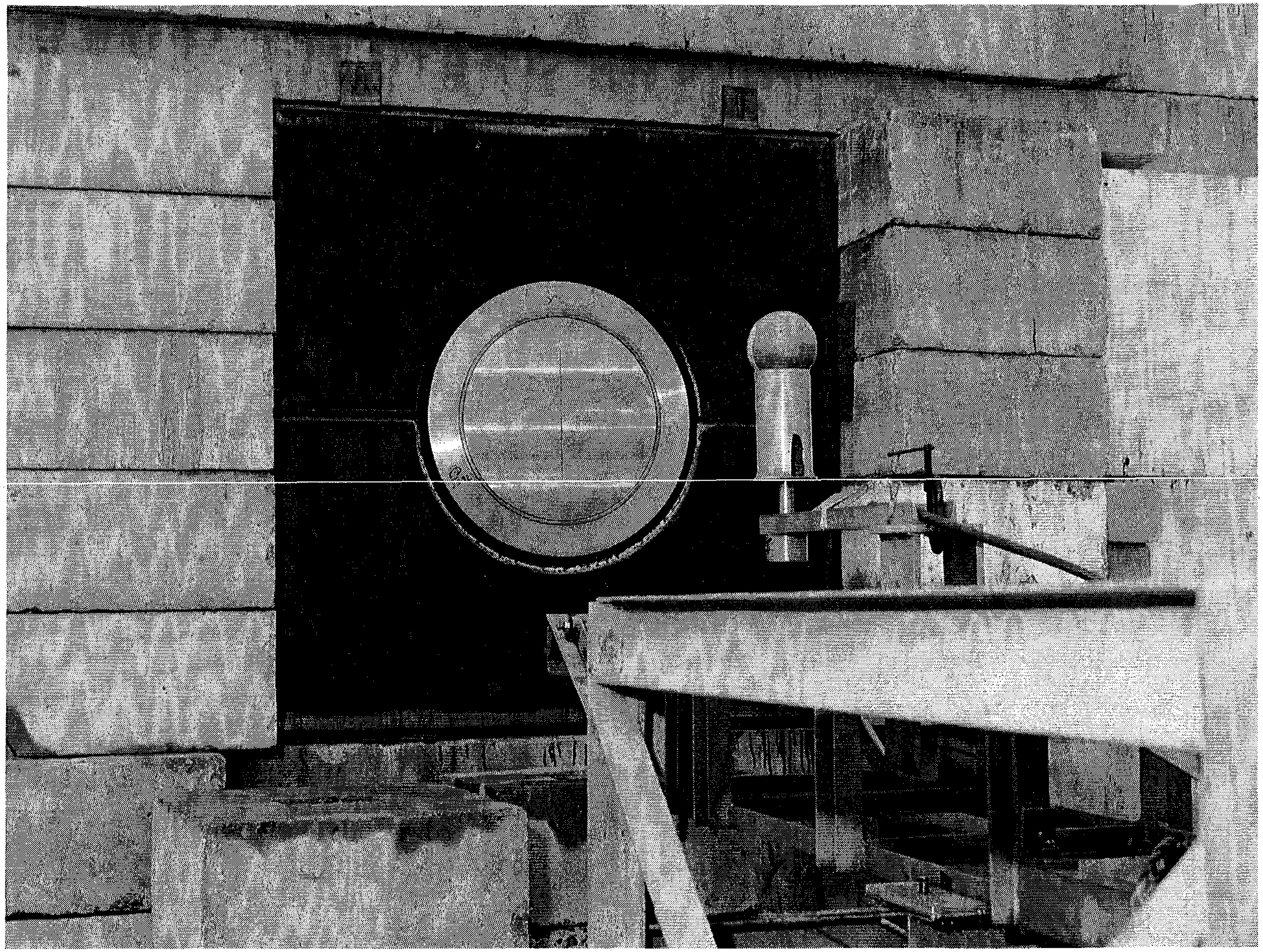


Fig. 15.10 Photograph displaying the Exit Face for the Annular Slit Mockup and Bonner Ball setup.

15.8 JASPER PROGRAM FLUX MONITOR EXPERIMENT

This experiment was designed to study the shielding concerns in the area where nuclear instrumentation systems (NIS) might be placed within the reactor vessel to monitor the neutron flux. The feasibility of such a system can be affected by the core shielding design, and where the design includes a "window" that allows neutron streaming, there was a concern about the ability to provide additional shielding for components beyond the detectors. Two configurations were studied[8]: one that used an aluminum window in the B_4C slab that provided data for the Japanese tank-type design, and the second that represented a possible arrangement for the U.S. LMR design using the Axial Shield Mockup, where the central hexagon was made of aluminum. The latter simulated the flux monitor directly above the core, a position comparable to the Japanese loop-type NIS arrangement. A typical mockup for the U.S. version is shown in Fig. 15.11.

15.9 JASPER PROGRAM SPECIAL MATERIALS EXPERIMENT

This was the last in the series of programs conducted for JASPER. Materials considered for this experiment[8] originally included Be, BeO, ZrH_2, YH_2, Hf, and B_4C fully enriched with $_{10}B$ and B_4C depleted of $_{10}B$. Availability and cost eliminated most of the materials; however, there was a slab of Zirconium at the TSF from previous experiments. To simulate the hydride part of ZrH_2, slabs of polyethylene were purchased and added to the mockups. Studies were made for a combination of Zr, B_4C, and polyethylene and then just B_4C and polyethylene.

15.10 JASPER PROGRAM AXIAL SHIELD RE-MEASUREMENT EXPERIMENT

Analysis by PNC of the Axial Shield experiment, described in Sect. 15.4, indicated an exceedingly high background source of neutrons from penetration through the concrete surrounding the hexagonal array. To mitigate this effect, repeat measurements were made with 10.16 cm of lithiated-paraffin between the spectrum modifier and the axial shield, while leaving a void in the area preceding the seven hexagons and the B_4C collar surrounding them. To further improve the foreground-to-background ratio, the same shaped slab of lithiated-paraffin bricks was also placed behind the axial shield. A schematic of the mockup is given in Fig. 15.12. (Measurements were made only for the mockups where the six outside hexagons contained B_4C and the contents of the central cylinder varied as in the original measurements.) Several measurements were included where the lithiated paraffin slab following the axial shield was removed to explore the effectiveness of that slab. These measurements were the last for the series of programs prescribed for JASPER, and with its completion, brought to an end the operation of the TSF on September 30, 1992.

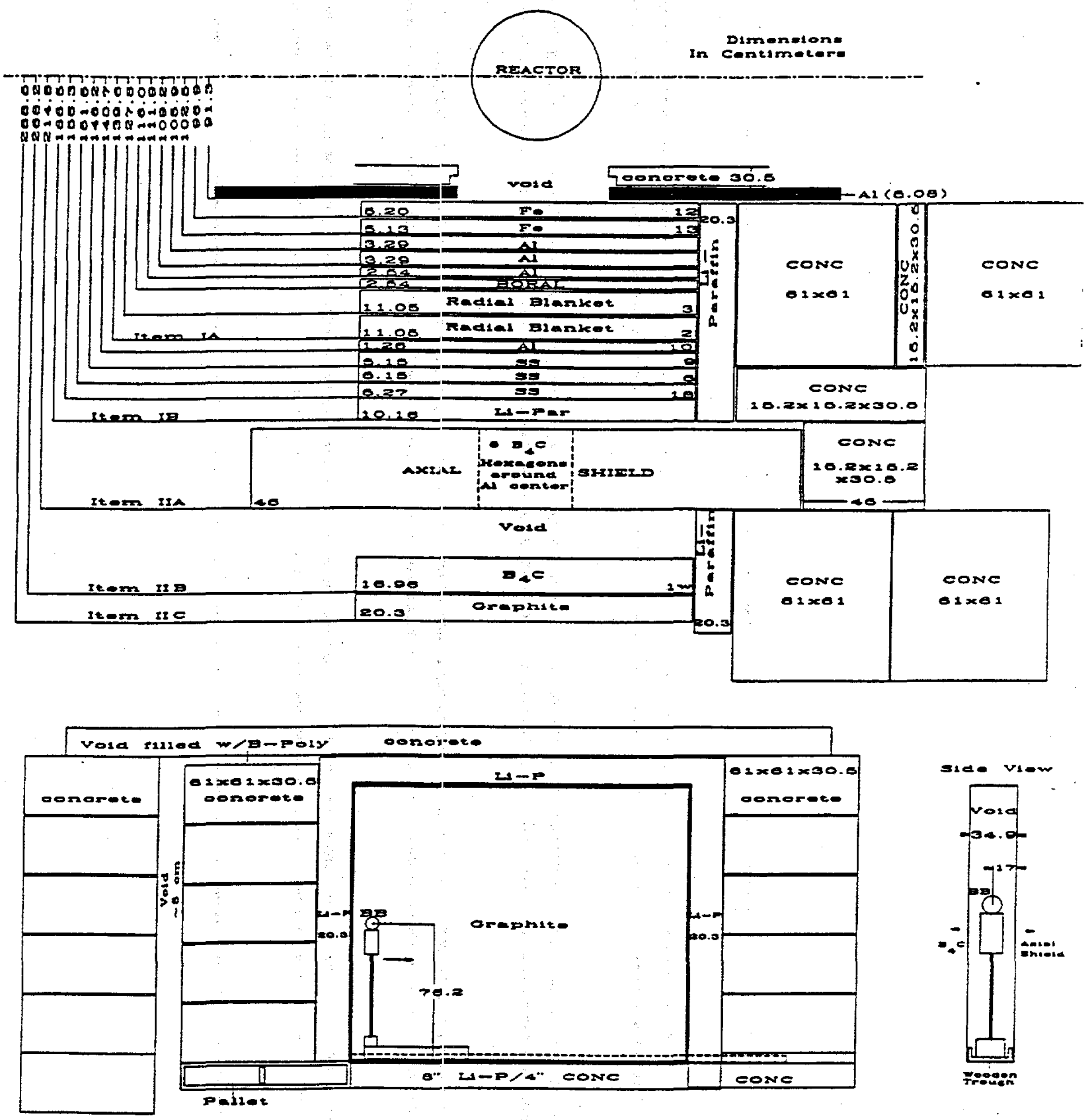


Fig. 15.11 Schematic of SM plus Shield Configuration.

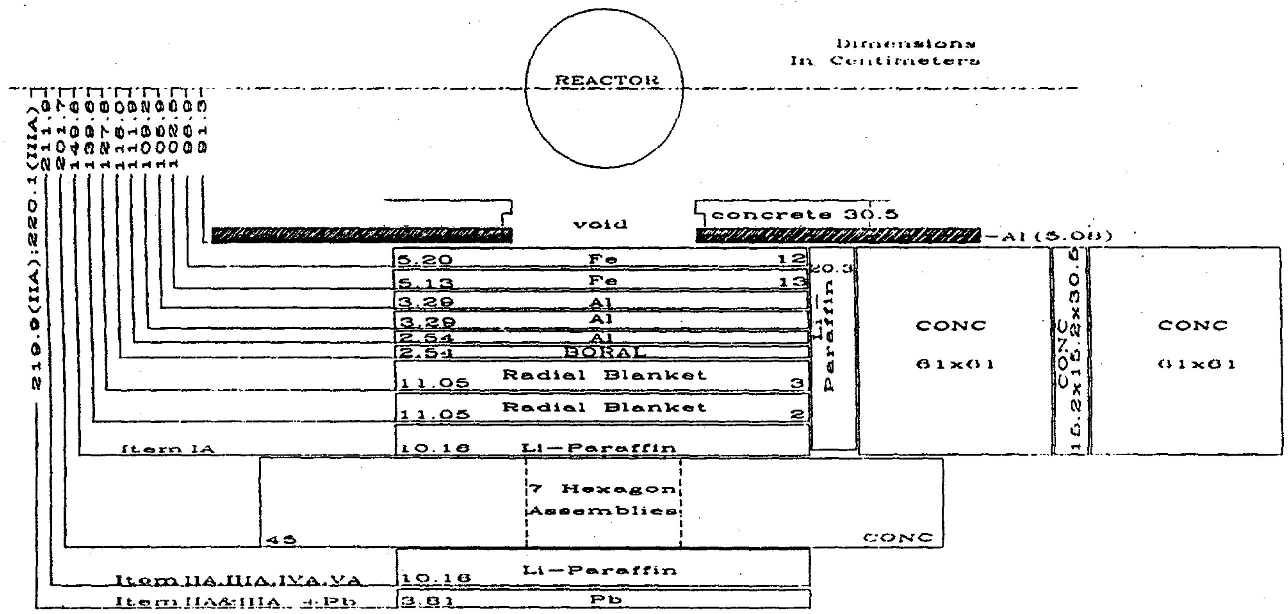


Fig. 15.12 Schematic of the Axial Shield mockup plus Lead Slab.
Note: Lithiated paraffin covers four sides of the SM-1.

15.11 REFERENCES

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3. F. J. Muckenthaler et al., "Measurements for the JASPER Program Fission Gas Plenum Experiment," ORNL/TM-10422 (June 1987).

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16.0 NON-PROGRAMMATIC EXPERIMENTS

16.1 INTRODUCTION

Throughout its lifetime, the TSF has shown its versatility on many occasions as it was confronted with requests to perform a variety of experiments which were certainly not envisioned when the facility was planned. In this chapter, experiments are described that have no direct connection with any one particular program, but are an important part of the TSF history.

16.2 PROJECT ORANGE PRIMATE EXPOSURE

The adaptability of the TSF was displayed early in its existence (1955) when it interrupted a differential-type mockup experiment in progress for the ANP program to expose a group of 25 Rhesus monkeys to massive neutron radiation doses.[1] The project, designated ORANGE by the Air Force, was preliminary in nature, and was performed to provide information for a more extensive experiment being planned for a later date. Prior experiments with primates had been limited to gamma-ray sources only and the need for studying the neutron effects was deemed necessary to obtain a more realistic approximation of the doses received from an atomic explosion. Chemical dosimeters, ion chambers, and the Hurst-type fast neutron dosimeters were used to measure the exposure dose. An experimental arrangement for the exposure is shown in Fig. 16.1.

16.3 SOIL AND CONCRETE EXPOSURE FOR LOCKHEED RADIATION DAMAGE FACILITY

Samples of concrete and soil from Marietta, Georgia, were irradiated at the TSF in 1956[2] to provide data for estimating the activation to be expected from exposure of these materials to 100 hours of reactor operation at the Lockheed Radiation Damage Facility. Each sample was a $3 \times 3 \times 1$ block, one containing soil and four others containing concrete. The four concrete samples consisted of plain concrete, barytes concrete, and one each of those types containing an admixture of 1% boron. The exposure lasted 20 hours at full power, followed by a study of the radiation decay using a 3×3-inch NaI crystal.

16.4 RADIATION SHIELDING TESTS OF FIGHTING VEHICLES

In early 1961, the shielding properties of armored vehicles were studied experimentally at the TSF.[3] The program resulted from an agreement between the United Kingdom and the U.S. to measure the attenuation characteristics of specific military equipment and research models of mutual interest. The equipment tested consisted of two British Avis-43 shields, a U.S. armored tank, a radiologically protected pod, and six geometric shapes that were designed by the Ballistics Research Laboratory. Participating in the work were groups from the U.S. Army Tank Automotive Center (ATAC) from Detroit Arsenal, General Dynamics/Fort Worth, and ORNL. The objectives were to compare measurements using ordinary armored vehicles, simulated designs for the future, and to aid in verification of theoretical studies toward optimization in design.

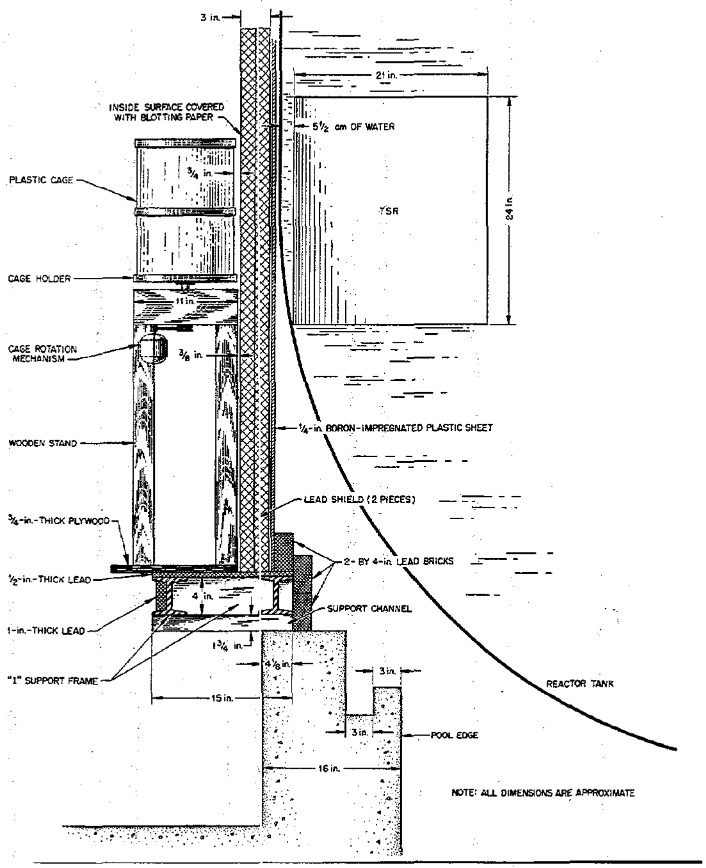

Fig. 16.1 TSF Experimental Arrangement for Monkey Irradiation (Side View).

16.5 IN-AIR RADIATION MAPPING OF TSR-II SOURCE

In 1961, an extensive program of "free field" mapping[4,5] of radiation escaping from the TSR-II reactor was made at several distances from the reactor. The term "free field" refers to minimal interference or bias to radiation reaching the detector caused by ground scattering, structures, equipment, etc. Measurements at long distances from the reactor were made atop three towers that were constructed at selected distances from the reactor as shown in the plan view in Fig. 16.2, and in the aerial photograph in Fig. 16.3. Detectors could be positioned at selected heights on these towers up to a maximum of 150 ft.

Two different reactor source spectra were used, the difference being the magnitude of the fast neutron leakages. The first spectrum was obtained by replacing the lead and boral surrounding the core inside the pressure vessel with water and aluminum and then surrounding the vessel with an

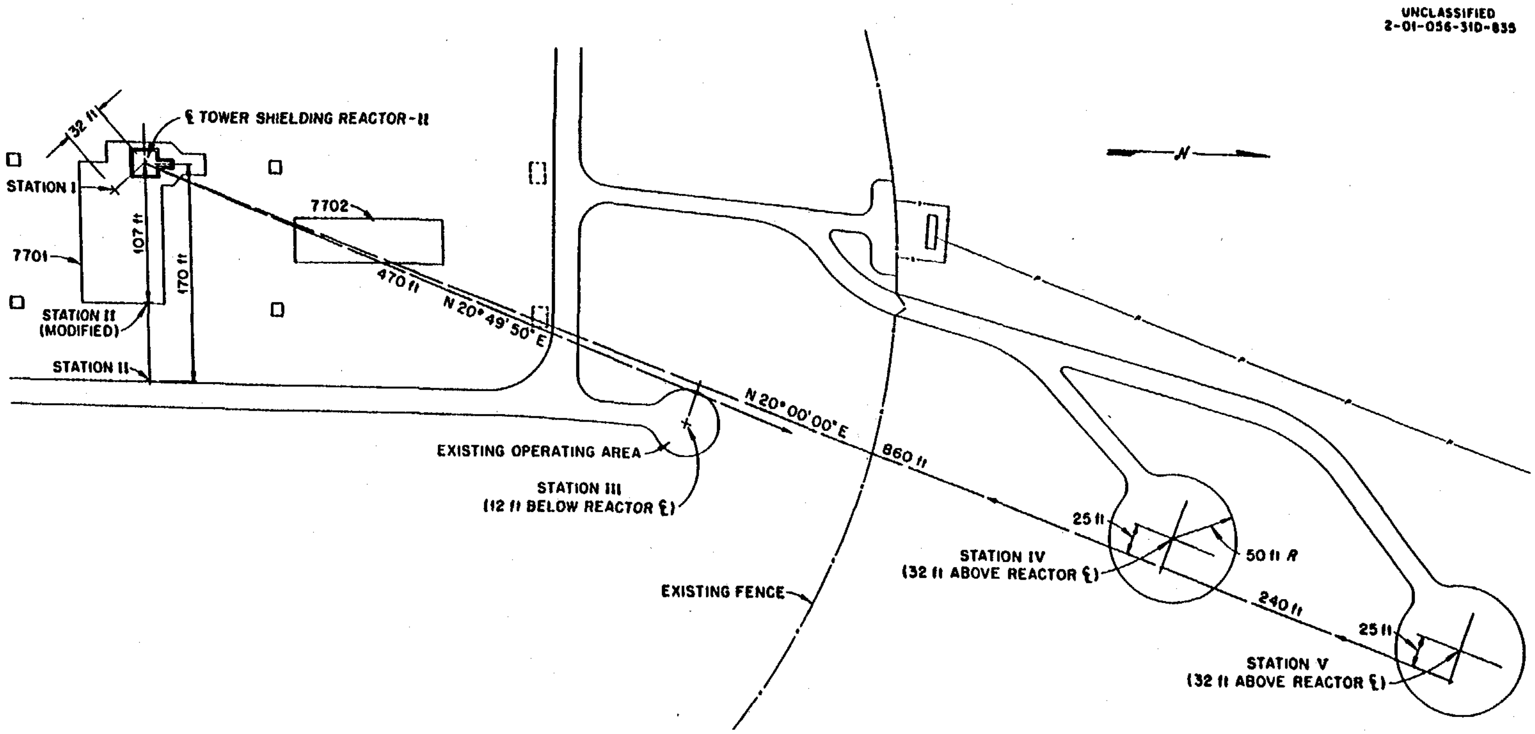

Fig. 16.2 Location of Detector Stations for In-Air Measurements at TSF.

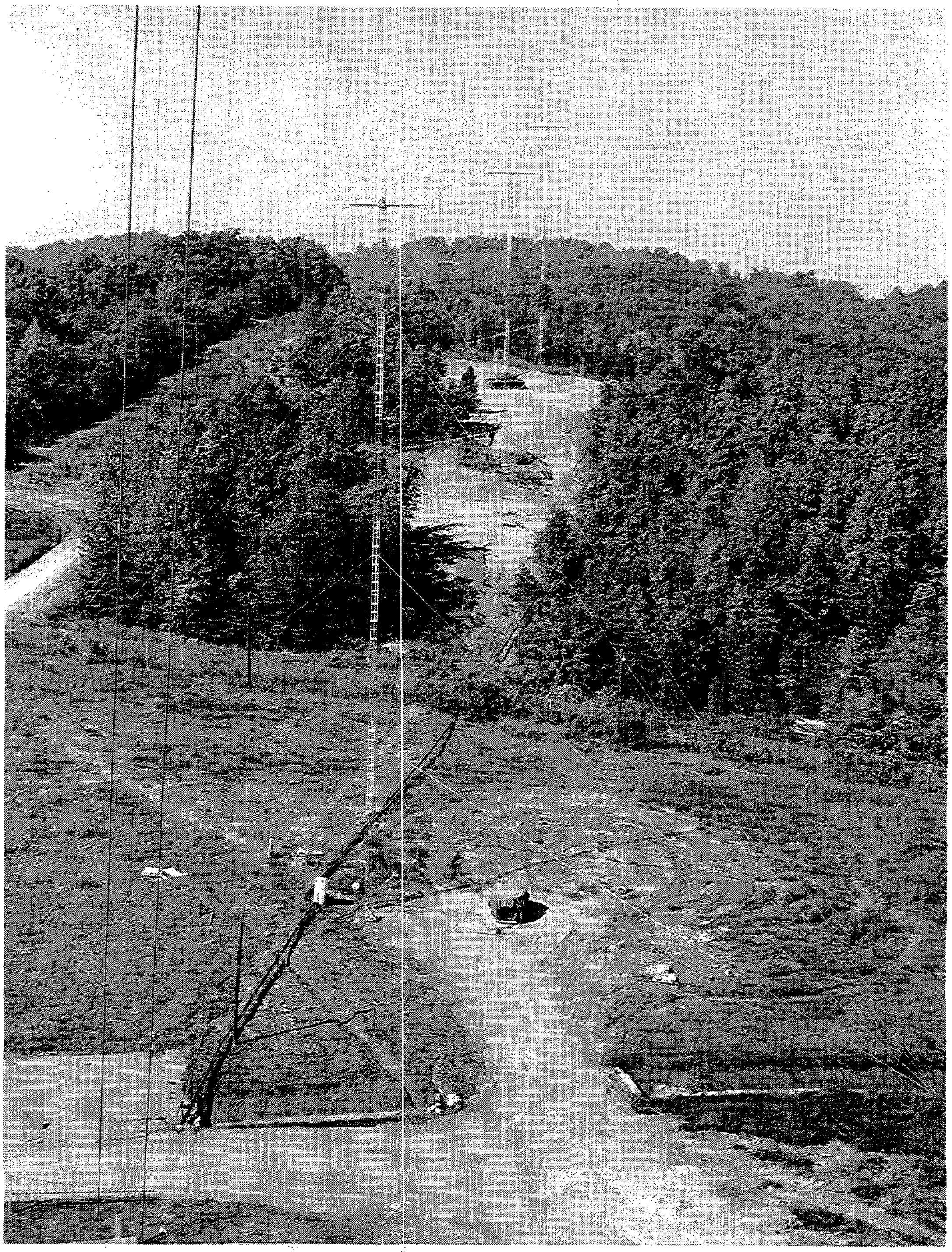

Fig. 16.3 Special Tower Stations III, IV, and V as viewed from the main TSF tower.

additional shield of lead, the combination called COOL-I. The COOL-II shield, which provided a harder fast neutron leakage spectrum, was obtained by slipping a shield of lead, water, and borated water around the COOL-I shield. These two shields were shown previously in Fig. 5.6 (Sect. 5.6). These shields give a fast neutron to gamma-ray dose rate ratio of 1.5 at large distances from the reactor, while providing sufficient thermal flux to contribute 30% of the gamma-ray dose rate from neutron capture in nitrogen, as measured in the field.

16.6 SHIELDING STUDIES AGAINST PROMPT WEAPONS RADIATION

One of the problems facing the military (Department of Defense) in the late 1950s was the penetration of prompt weapons radiation into shielded and open underground facilities. To aid in the understanding of the transport of neutrons through the air and subsequently through shields and silos, a series of measurements were made starting in 1961, sponsored by the Air Force[6] and Office of Civil Defense[7].

For the Air Force-sponsored part of the program, the TSR-II was modified to give a leakage flux similar to that from a weapon and measurements were made inside concrete-lined wells (silos) as representative of typical weapons installations. The holes were 4 ft in diam, 20 ft deep, and placed at different distances from the reactor to facilitate studies of different angles of radiation penetration by varying the reactor height. The three holes were identical, except for one which contained an opening perpendicular to the hole, as seen in Fig. 16.4. This horizontal opening was well protected from the direct radiation by the earth above it, thus activity measured in the hole had to be from scattering in the well. Neutrons were measured by means of a Hurst recoil-proton proportional counter dosimeter, and the gamma-ray dose rates were measured with an anthracene scintillation detector.

Upon completion of the measurements by the middle of 1962, the experimental program was then shifted to investigate the radiation protection afforded by various personnel shelters against prompt weapons radiation (sponsored by the Department of Defense's Office of Civil Defense). The ultimate goal was to generate calculational methods for estimating protection factors against weapons radiation. Studies were to be made of the attenuation for ordinary concrete thicknesses, the buildup of radiation levels within the cavities by scattering from the walls, and the transmission of radiation down a tunnel (passageway) with two right angle bends as shown in Fig. 16.5. Again the TSR-II isotropic source was modified using the COOL-I shield to provide radiation similar to that from a nuclear weapon. The bunkers at each end of the passageway were 12-ft cubes, constructed so that the concrete thickness on the top of one bunker and on the face of the other could be varied. Typical measurements of dose rates for neutrons and gamma rays were obtained along with values of the thermal flux.

16.7 U.S. NAVAL RADIOLOGICAL DEFENSE LABORATORY SHIELDING EXPERIMENTS

A very brief experiment was performed to measure the dose distribution inside a steel structure placed in a radiation field simulating the initial radiation field from a nuclear weapon. Measurements were made as a function of wall thickness, location of detector within the compartment, and as a function of distance from the source. A simple display of the mockup is shown in Fig.16.6,

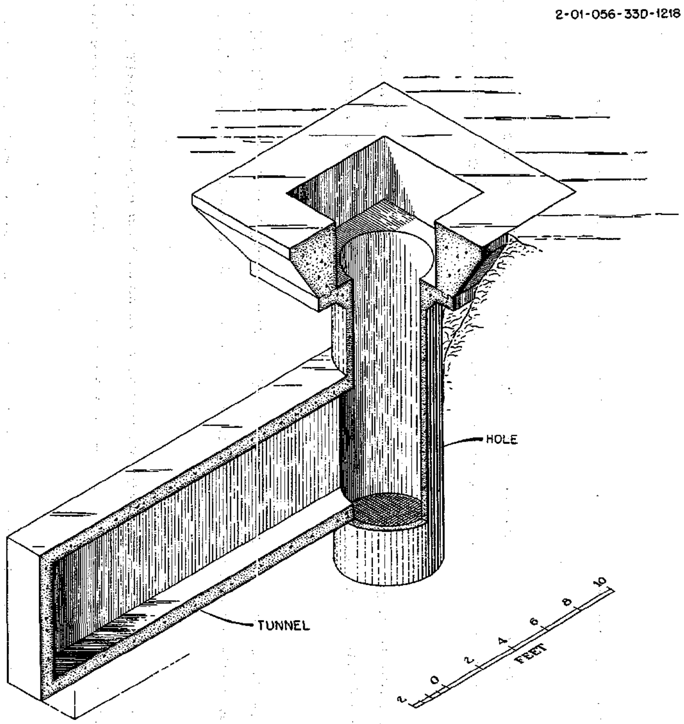

Fig. 16.4 Cutaway View of Hole-Tunnel Configuration.

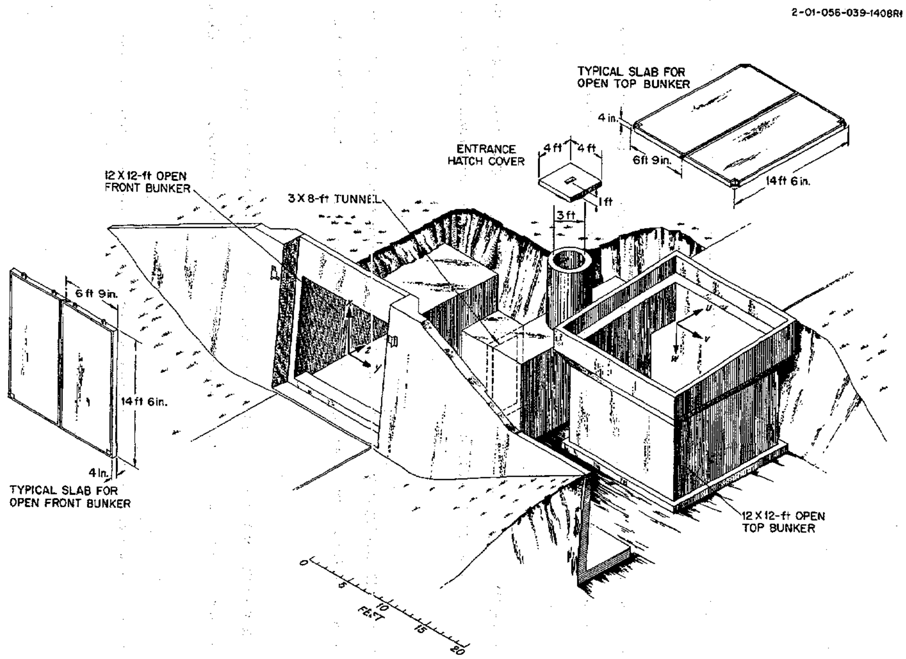

Fig. 16.5 Schematic of Concrete Bunkers and Interconnecting Tunnel.

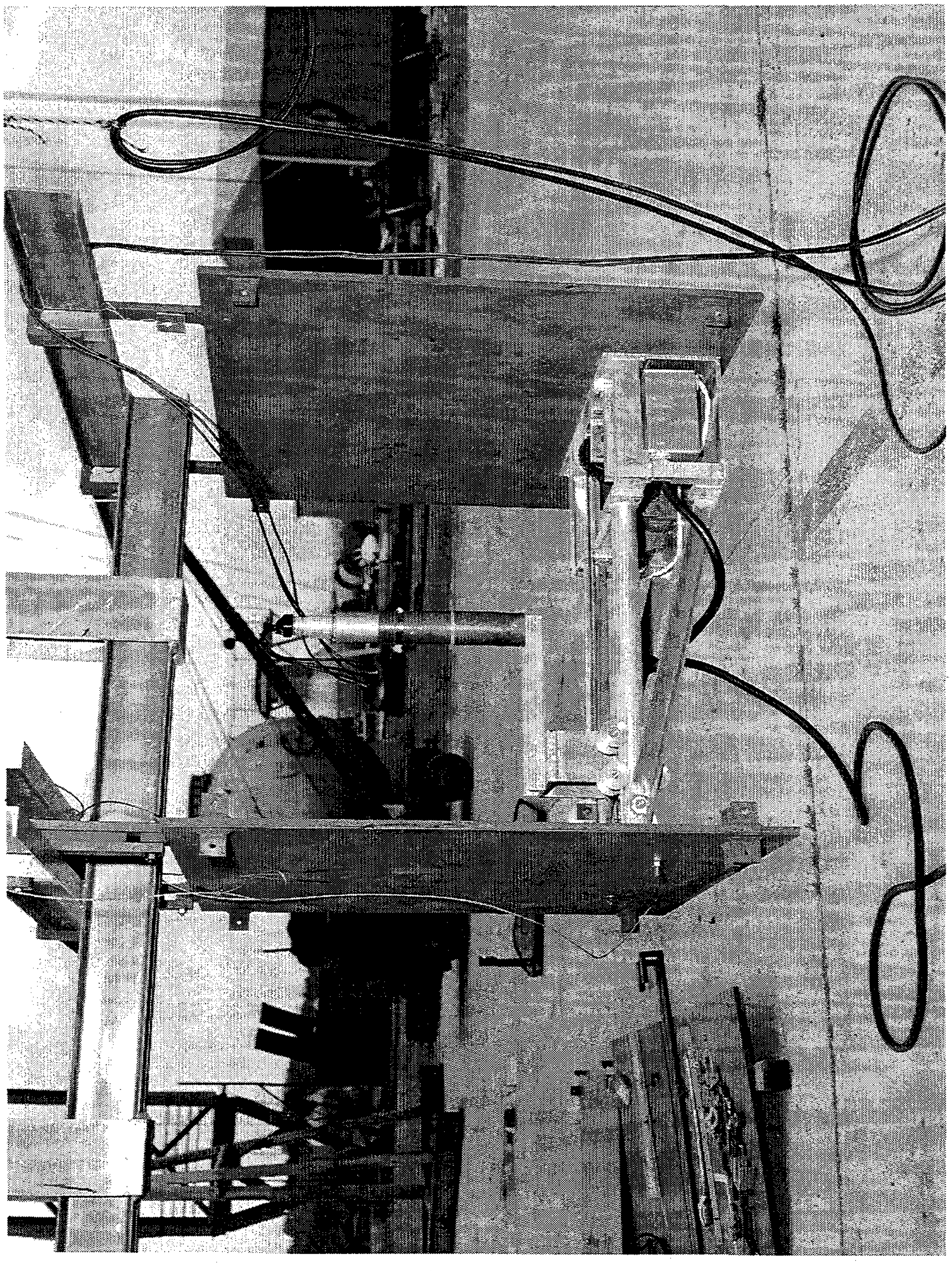

Fig. 16.6 Typical Workup for Measurements inside Simulated Wall Structures for Naval Vessels.

with two walls removed to display the detector and traverses. These results were used to develop attenuation and geometry factors for compartmentalized steel structures inside a naval vessel in a radiation field.

16.8 NASA DUCT EXPERIMENT

In June 1971, the TSF exposed a single 8-in.-diam, half-term helical duct installed in a large tank of water to a collimated beam of neutrons and gamma rays from the TSR-II. Fast neutron flux and gamma-ray dose rate profiles were obtained, beyond the exit of the ducts, for mockups of just plain water and water plus lead irises around the duct.

16.9 DETERMINATION OF THE NEUTRON ENERGY AND SPATIAL DISTRIBUTION OF THE NEUTRON BEAM FROM TSR-II IN LARGE BEAM SHIELD

Fabrication of the large beam shield for the TSR-II, shown previously in Fig. 5.12 (Sect. 5.9), was completed in late 1974. The shield consisted of a layer of SS and water, which acted as a thermal shield, surrounded on all sides by thick concrete, except for the beam port. With the reactor in this shield, an air void was formed beneath the reactor between the hemispherical shape at the bottom of the pressure vessel and the flat concrete flooring beneath it. A similar space on top of the fuel was filled with lead. Thus, to present a nearly isotropic source to the configurations below the core, it was necessary to replace an equivalent volume (and shape) of the lead and replace it with pressurized air tanks.

The new shield design permitted the mockups to be placed approximately 91 cm from the center of the reactor and this, along with an enlarged collimator diameter that exposed the mockup to the whole core, the intensity of the beam incident on the configuration was 200 times that obtained previously with the reactor in its spherical shield. This increase was necessary to provide greater accuracy in the measurements, an improvement that was needed to upgrade the reliability required for verification of the increasingly more rigid calculational methods.

Measurements[8] were made to determine the neutron spectrum and to define its spatial distribution using the NE-213 liquid scintillator, hydrogen recoil spectrometers, Bonner balls, and Hornyak button. Relatively small corrections were made where necessary, to account for the different geometries of the various detectors.

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17.0 CASK DROP PROGRAM

17.1 INTRODUCTION

The safe transport of radioactive materials from one facility to another is of prime importance.[1] Strict United States and international regulations govern both the design and use of the packages that contain radioactive material during shipment. Before such a package can be used, the structural integrity of it has to be demonstrated to show that it will indeed survive severe and improbable accident criteria, as set down by the regulations, without release of any of its radioactive contents.

To perform these tests, it was seen necessary to have a facility capable of lifting heavy loads, at least 50 tons, to variable heights and provide a surface upon which the load could land without yielding to the impact. The TSF was capable of providing the ability to lift heavy loads as well as providing an extensive work area. Thus in 1978, a test pad area was constructed at the TSF that was virtually unyielding. It contained 600 metric tons of 5-cm-rebar-reinforced concrete shaped in the form of a stepped pyramid, having a larger base than impact area, topped by a 70-ton armor-plated surface that was 6.1 m long, 2.5 m wide, and 51 cm thick. The pad was located nearly midway between the two eastern towers of the four-tower rectangle, originally designed for use in reactor shield studies.

The casks to be dropped were attached to the hoist cables running from the two tower legs, the same cables that were used for lifting reactor shield samples. The maximum height possible for a cask drop was 200 ft. A view of the pad under construction is shown in Fig. 17.1, while a photograph of the finished pad surface is shown in Fig. 17.2. Because of its unique capability, the TSF was identified as a facility that could be made available to both government and nongovernment users. It was even possible to test packages not designed for carrying radioactive materials. A typical example of a test drop is shown in Fig. 17.3.

The drop tests[2] were recorded by high- and normal-speed motion picture cameras along with detailed measurements of the test pieces following each impact. Data acquisition capabilities also included signal conditioning and high-speed recording equipment for a variety of sensors such as accelerometers and strain gauges, and on-line spectral analysis.

The tests were done under the auspices of the Transportation Technology Group of the Chemical Technology Division at ORNL. This group was involved in the testing and developing of safe shipping packages over a period of 20 years.

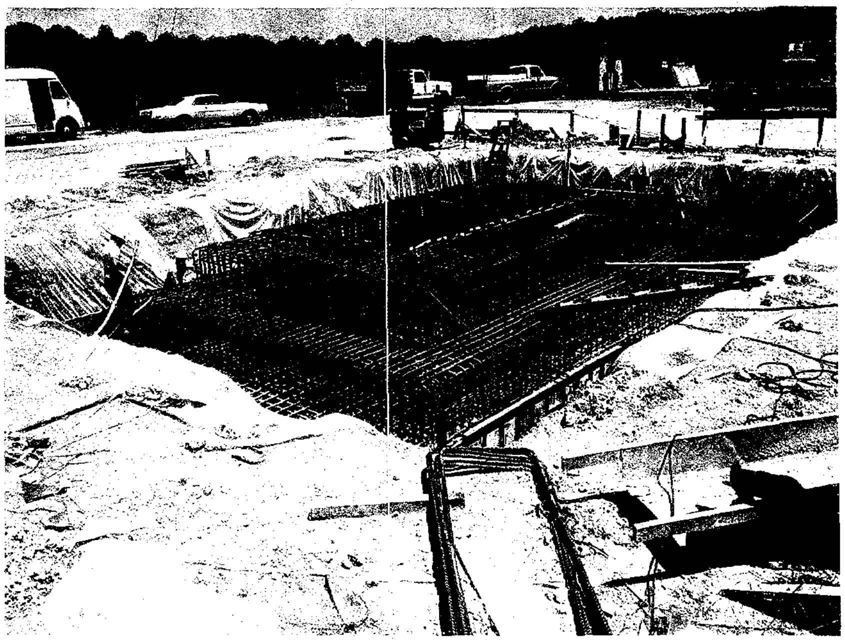

Fig. 17.1 Re-bar Framing for the Concrete Base Supporting the Impact Pads.

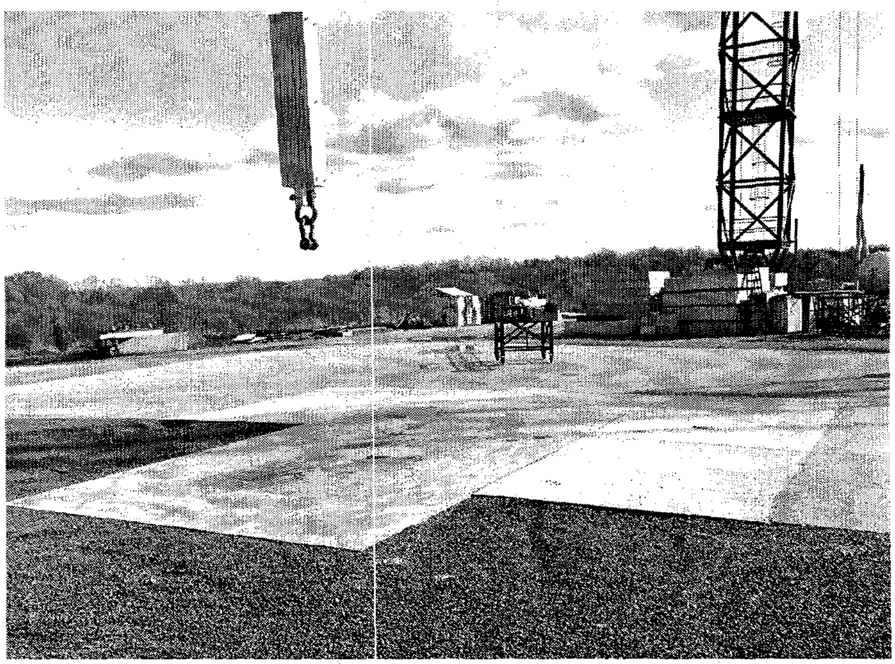

Fig. 17.2 The Armor Plate Surface of the Impact Pad Surrounded by a Concrete Edging.

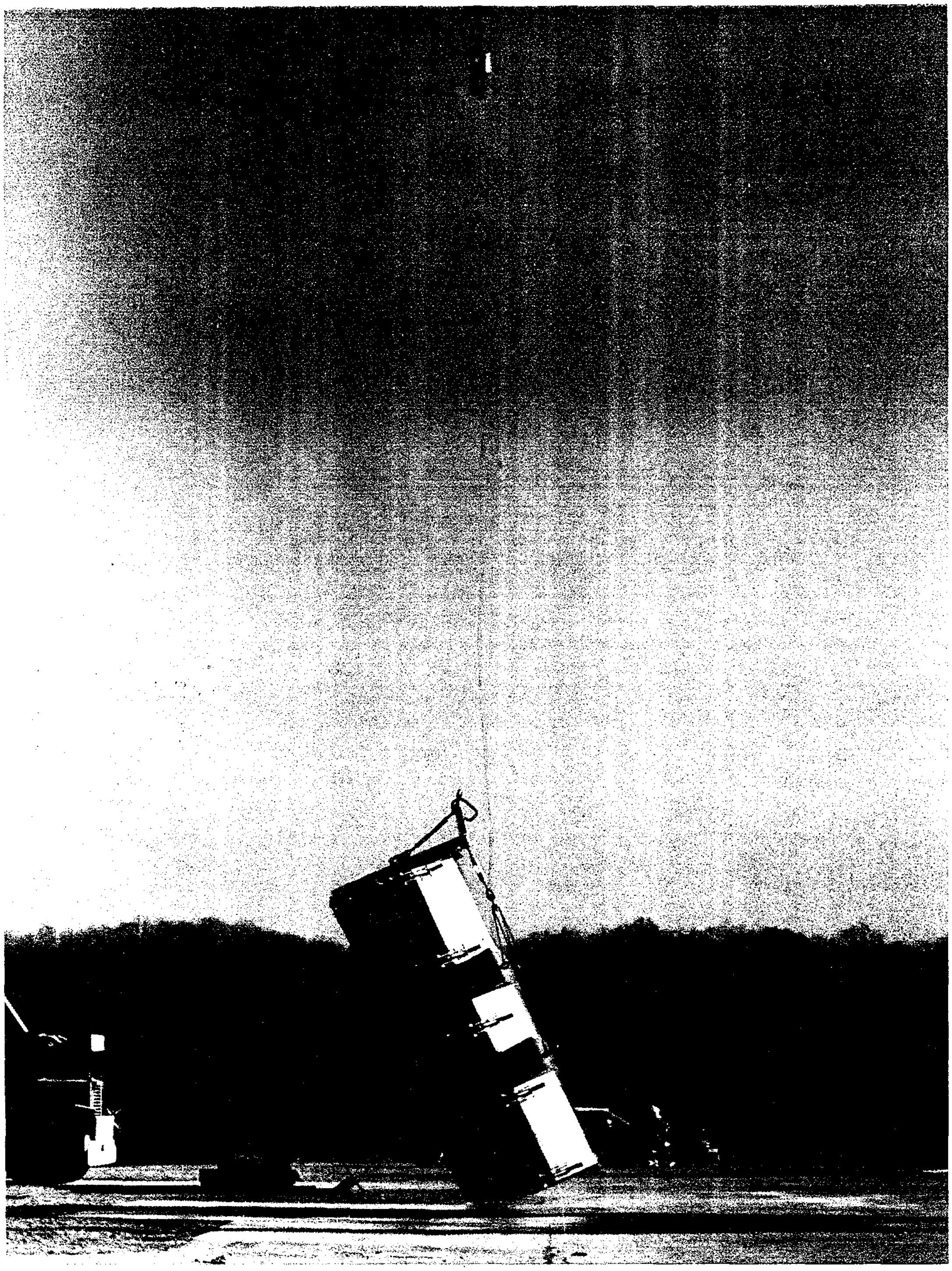

Fig. 17.3 A typical example of a Test Drop.

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