ARF D132K03-1
Reactors--General

ARMOUR RESEARCH FOUNDATION
of
ILLINOIS INSTITUTE OF TECHNOLOGY
Technology Center
Chicago 16, Illinois

STUDIES OF REACTOR CONTAINMENT

Task on

STRUCTURAL AND MECHANICAL DESIGN CRITERIA
FOR NUCLEAR REACTOR CONTAINMENT

Report No. 1

An Annotated Bibliography on Reactor Containment

United States Atomic Energy Commission
Contract No. AT(11-1)-528
ARF No. D132K03

August, 1958

Department of Mechanical Engineering Research

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

This is Report No. 1 on a task on "Structural and Mechanical Design Criteria for Nuclear Reactor Containment", ARF No. D132K03, performed by Armour Research Foundation for the United States Atomic Energy Commission under Contract No. AT(11-1)-528, Studies of Reactor Containment. The work reported herein was performed between August 25, 1957, and March 1, 1958.

Foundation personnel who have contributed materially to this report include R.L. Chiapetta and D.B. Singer.

Respectfully submitted,

ARMOUR RESEARCH FOUNDATION OF
ILLINOIS INSTITUTE OF TECHNOLOGY

D.B. Singer, Task Engineer
Structural Analysis Section

R.W. Sauer, Supervisor
Structural Analysis Section

APPROVED:

G.A. Nothmann, Manager
Mechanical Engineering Research Department

STRUCTURAL AND MECHANICAL DESIGN CRITERIA
FOR NUCLEAR REACTOR CONTAINMENT

ABSTRACT

A survey of recent publications concerned with structural and mechanical aspects of nuclear reactor containment was conducted during the first phase period. This report presents an annotated bibliography of 279 such publications. Many of the references listed appeared in the Nuclear Science Abstracts and were published in the period from 1954 through 1957.

TABLE OF CONTENTS

	<u>Page</u>
<u>REACTOR SAFETY AND CONTAINMENT</u>	
I. Reactor Descriptions	1
II. Reactor Safety and Hazards	5
III. Accidents	9
IV. Containment	11
<u>PRACTICAL ASPECTS OF REACTOR DESIGN</u>	
V. Material Properties	13
VI. Stresses	20
VII. Thermal Effects	27
VIII. Joints and Fasteners	39
IX. Vessel Rupture	42
X. Reactor Vessel Design	47
XI. Testing	54
<u>THEORETICAL ASPECTS OF REACTOR DESIGN</u>	
XII. Elasticity and Plasticity	55
XIII. Plates and Shells	59
XIV. Buckling	62
<u>OTHER REFERENCES</u>	
XV. Bibliographies	64
XVI. Miscellaneous	67

REACTOR SAFETY AND CONTAINMENT

I REACTOR DESCRIPTIONS

- 1 Benedict, Manson, (Massachusetts Institute of Technology)
"Nuclear Reactors for Research"
Chemical Engineering Progress 51; February 1955.
A brief discussion of research reactor operating principles is given together with a survey of reactor uses, the types available, their cost, and safety problems.

- 2 Borkowski, C. J.
"Design Review Committee Report on the Homogeneous Reactor Test (HRT)"
Oak Ridge National Laboratory,
CF-55-6-53; June 7, 1955.
The HRT Design Review Committee reported on various aspects of the HRT design including engineering design, equipment layout, pipe drawings, control and instrumentation, reactor kinetics and soup chemistry.

- 3 Ebans, H. T., (Stone and Webster Engineering Corp., Boston, Mass.)
"Structural Features of the Reactor for the Shippingport Atomic Power Station Under Construction at Shippingport, Pennsylvania"
Paper presented at Meeting of American Society of Civil Engineers
October 16, 1956.

- 4 Eklund, Sigvard, (Atomic Energy Company, Stockholm, Sweden).
"The Swedish Reactor"
Nuclear Engineering Part 1, Chemical Engineering Progress Symposium Service No. 11; 1954.
The site and building in which the Sleep Reactor is located are described. Design of the reactor (tank, U fuel rods, D₂O circulation system, heat exchanger, safety rods, power level stabilization, reactivity control, power monitoring, graphite reflector and shielding) which is of the CP-3 type is described.

- 5 Huffman, J. R.
"Materials Testing Reactor Project"
Argonne National Laboratory
ANL-4304; 1949.
A comprehensive survey of the general specifications and requirements for the Materials Testing Reactor located at the Reactor Testing Station was completed. The staff at ANL prepared proposals for a still supporting structure design.

- 6 King, L. D. P.
"A Brief Description of a One Megawatt Convection Cooled Homogeneous Reactor - LAPRE II"
Los Alamos Scientific Laboratory.

LA-1942; April 13, 1955.

General descriptions are given for the LAPRE-II. Complete reactor specifications are listed. The reactor hazards are also briefly discussed.

- 7 Kowarski, L., (Commissariat à l'Énergie Atomique, Paris)
"Development of the Second French Reactor"
Nucleonics 12, No. 8; August 1954.
The history and objectives of the French Atomic Energy program are reviewed briefly, and the design, construction, operation, and contributions of the Saclay reactor in testing, physical research, and as a source of radiation are summarized.
- 8 Lichtenberger, H. V., (Argonne National Laboratory, Lemont, Ill.)
"The Experimental Breeder Reactor"
Nuclear Engineering, Part III, Chemical Engineering Progress Symposium Ser. No. 13, 139-46; 1954.
The basis for the design of the Experimental Breeder Reactor is discussed along with the details of construction and the constants of the reactor. The cooling system and the plant layout are described. The concept of breeding gain and specific power are discussed along with the significance of the development.
- 9 McLain, S.
"Brief of Development Reactor Project"
Oak Ridge National Laboratory
CF-49-1-193; January 10, 1949.
The proposed design of the Materials Testing Reactor is described in detail.
- 10 Wright, W. L.
"The ORR Reactor Building, Reactor Structure and Services"
Oak Ridge National Laboratory
CF-56-5-186; April 19, 1956.
A general description of the design, construction, and facilities of the ORNL Research Reactor is given.
- 11 Author Unlisted
"ORNL Research Reactor Design Data"
Oak Ridge National Laboratory
CF-51-8-216; 1951.
- 12 Author Unlisted
"Calder Hall, Britain's First Commercial-Scale Reactor"
Atomics 5; October 1954
A short description of the Calder Hall nuclear power station emphasizes the auxiliary power system design proposed for use in conjunction with a conventional gas cooled thermal reactor.

- 13 Author Unlisted
 "Catalogue of Nuclear Reactors"
 Atomic Energy of Canada Ltd., Chalk River Project, Chalk River, Ont.
 AECL-200; August 1955.
 Data are summarized from the unclassified literature on 46 reactors known to have reached criticality by July, 1955. The reactors are classified under various categories and construction details are outlined. References covering the more complete sources of information are included for each reactor. The table of contents is cross-referenced from less-commonly used names to the main entry, and a geographical index is included.
- 14 Author Unlisted
 "Nuclear Reactor Data"
 Raytheon Manufacturing Company, 1955.
 An information chart on all known reactor projects throughout the world has been compiled. Types of reactors listed include pressurized water, boiling water, swimming pool, homogeneous, heavy water, graphite, sodium graphite, liquid metal cooled, liquid fuel, and fast breeder. All the basic data available are given for each reactor.
- 15 Author Unlisted
 "Research Reactors, Selected Reference Material, United States Atomic Energy Program"
 Atomic Energy Commission.
 TID-5275; 1955.
 Extensive reference material is given on several types of research reactors, namely: the light-water-moderated, homogeneous fuel reactor; the light-water-moderated, heterogeneous fuel reactor; the light-water and oil-moderated, heterogeneous fuel reactor; the heavy-water-moderated, heterogeneous fuel reactor; and the graphite-moderated reactor. Topics are discussed including descriptions of facilities, assemblies, control system, performance, shielding, experimental facilities, etc.
- 16 Author Unlisted
 "Engineering Test Reactor Engineering Design and Safeguards Report"
 Atomic Power Equipment Dept., General Electric Co., and Kaiser Engineers Division, Henry J. Kaiser Company
 IDO-24020; July 1956.
 The Engineering Test Reactor (ETR) is designed to perform engineering tests on fuel elements and components of nuclear plants. Its main purpose is to provide large experimental facilities with very high neutron fluxes, thus supplementing research reactors already in use. The reactor complex is described in detail and engineering drawings are given for the complete system, including buildings and services. Various operational considerations are evaluated from a reactor safeguards basis.

- 17 Author Unlisted
"Experimental Power and Test Reactors. A Technical and Pictorial Summary"
Division of Reactor Development, AEC
TID-4562; November, 1956.
Diagrams and photographs of various experimental and test reactors are given. Pertinent technical data on each reactor are summarized. The reactors discussed are the PWR, EBWR, SRE, EBR-1, EBR-2, HRE-1, HRE-2, OMRE, LMFR, LAMPRE, MTR, ETR, and SPERT.
- 18 Author Unlisted.
"The Experimental Boiling Water Reactor (EBWR)"
Argonne National Laboratory
ANL-5607; May, 1957.
The final design of the EBWR is described in detail.
- 19 Author Unlisted.
"The Building of Dounreay"
Atomics 8, June, 1957.
The construction of the Dounreay Fast Breeder Reactor is described.

II REACTOR SAFETY AND HAZARDS

- 1 Chastin, J. W.; Redmond, R. F.; Klickman, A.; Anno, J.; and Fawcett, S. L.

"Hazards Summary Report for the Battelle Research Reactor"
Battelle Memorial Institute
BMI-ACRS-601, (Rev); April 4, 1955.

Two reactor accidents and their hazards are considered: (1) a "maximum credible accident" in which an instantaneous increase in reactivity causes a power pulse and a Borax type excursion, and (2) a "maximum hypothetical accident" in which there is complete vaporization of the fuel elements into a cloud. In the latter, which is the most extreme direct radiation, inhalation, and fall-out of the radioactive cloud are considered. It is concluded that the design is safe and that the location provided an adequate exclusion area.

- 2 Cooper, W. E.; and Miller, D. R.
"Safeguards Aspects of Reactor Vessel Design"
KAPL-1524; March 30, 1956.

This paper deals with the problem of design for the high degree of structural reliability required in reactor vessels and with the theoretical evaluation of the mechanical effects of abnormal pressures due to chemical or nuclear incidents. The structural reliability required of reactor vessels exceeds that required of conventional vessels. Furthermore, the existence of new problems or old problems of new significance must be recognized in the design stage of new high-performance reactor systems. Consequently, the extent of engineering effort expended in design and design evaluation must be substantially greater than is normally applied to design of conventional vessels. Similarly, the criteria for limitation of stresses, the quality of design details, and the degree of control of material and fabrication quality must be superior to practices normally used in design and fabrication of pressure vessels.

- 3 Daane, R.; Mela, R.; and DeFelice, J.
"Progress Report on the Study of Low Probability, High Hazard, Fast Reactor Accidents"
Nuclear Development Associates, Inc.
NDA-14-107; November 11, 1955.

The methods being used to analyze highly hazardous, but highly unlikely nuclear accidents are described. The assumptions made require more examination, and, therefore, the results are probably too pessimistic for the model selected by factors of ten or a hundred. Two accidents are described in detail; rapid assembly and slow meltdown.

- 4 Fye, Paul M.; and Noonan, E. C.
"NOL Reactor Safety Program"
Naval Ordnance Laboratory, White Oak, Md.
NavOrd-4286; April 30, 1956.

A proposed program of scaled studies of possible reactor incidents is described. Ordinary explosives will be used to provide a rapid release of the appropriate amount of energy. Scale ratios from 1/8 to full-scale will be used and a study will be made of the properties of various materials which may be used in the construction of pressure vessels. A preliminary study was made of the Borax explosion using a linear scale factor of 1/8. The results were in line with other estimates of the energy release and equivalence to high explosives.

- 5 Graham, Richard H.; and Boyer, D. Glenn
"AEC's Experimental Program on Reactor Safety"
Technical Information Service, AEC
TID-8609.
Heterogeneous and homogeneous reactor experiments designed to establish maximum safety factors for reactor operation are reviewed. Large reactor excursions, safety fuses, molten metal-H₂O reactions, container tests, and fuel ignition are discussed.
- 6 Kasten, P. R.; and Pare', V. K.
"Safety of HRT"
Oak Ridge National Laboratory
CF-54-3-164; March 25, 1954.
- 7 Luckow, Wm. K.; Mesler, Russell B.; and Widdoes, Lawrence C.
"The Nuclear Research Reactor at the University of Michigan: An Evaluation of the Hazards of Operation and Provisions for Limiting These Hazards"
Michigan Memorial-Phoenix Project, University of Michigan
AECD-3669; April 22, 1954.
A hazards evaluation of the proposed Michigan Research Reactor is made for the case of a maximum credible accident, which is believed to be the dissolution of the fuel elements and subsequent release of fission products to the pool water. Factors affecting radiation doses within and outside a specially designed reactor building are discussed.
- 8 Luckow, Wm. K.; and Widdoes, Lawrence C.
"Predicting Reactor Temperature Excursions by Extrapolating Borax Data"
Nucleonics 14, No. 1; January 1956.
Data on the nature and limitations of power excursions in water-cooled, moderated, and -reflected reactors were obtained in the "Borax" experiments. These data were extrapolated to predict the transient behavior of the University of Michigan swimming-pool-type research reactor. Methods of extrapolation are described, and results are summarized.
- 9 McCarthy, W. J., Jr.
"Some Inherent and Design Safety Features of the APDA Fast Breeder Reactor"

Atomic Power Development Associates, Inc.
AECU-3196; June 1956.

- 10 McCarthy, W. J., Jr.; and McMath, F. C.
"Safeguard Features of a Fast-Breeder-Reactor Power Plant"
Transactions of the American Society of Mechanical Engineers 79
January 1957.
In the design of a reactor power plant, special attention to safety is required in order to safeguard the investment in the plant and to insure that danger to the public is reduced to negligible proportions. Some design features relevant to the safety of a fast-breeder-reactor power plant are described. The decision to enclose the radioactive portions of the plant in a gas-tight building to provide a final barrier between the reactor and the public is discussed. Criteria developed for locating the reactor on a safe and economically suitable site are presented.
- 11 Miller, E. C.
"HRT Reactor Hazards"
Oak Ridge National Laboratory
ORNL-2089; August 1956.
- 12 Mills, M. M.
"On the Hazard Due to Nuclear Reactors"
North American Aviation, Inc.
NAA-SR-116; October 1950.
A major hazard associated with any nuclear reactor is the possible contamination of the atmosphere or water supply by the accidental release of the radioactive materials contained within the core. It appears likely that for the reactors which have been built or are presently contemplated, an adequately gas-tight reactor building can remove the public hazard. The absence of mechanical shock, external to the shield, makes this possible. Some development in building construction may be needed to attain this goal.
- 13 Weil, George L.
"Hazards of Nuclear Power Plants"
Science 121; March 4, 1955.
The hazardous conditions that could result from a major accident involving a nuclear power plant are discussed, assuming the most pessimistic conditions that might arise. Precautionary measures that can be taken in plant design, location, and operation are described.
- 14 Widdoes, Lawrence C., (University of Michigan, Ann Arbor)
"Safety Considerations in the Selection of University Research-Reactor Designs and Locations"
Chemical Engineering Progress Symposium, Sec. No. 11, 195-202; 1954.

In the evaluation of the hazard to a given community from the location of a reactor nearby, the behavior of the reactor under all conceivable conditions must first be calculated. From these calculations the maximum credible accident examined carefully, and ways and means of restoring the facility to operation formulated. In addition to this study, the possible results of a total catastrophe should be considered. Factors used in evaluating hazards are discussed, and the results of calculations made for the location of the University of Michigan reactor on the North Campus are given.

- 15 Wieren, H. O.
 "HRT Hazards Report, Chemical Processing System"
 Oak Ridge National Laboratory
 CF-54-12-146; December 10, 1954.
 The mishaps which are most likely to occur during the operation of the HRT fuel processing system are listed. The probable results of each mishap are investigated, and steps are recommended for eliminating the possibility of a hazardous occurrence.

- 16 Author Unlisted
 "AEC Experimental Program on Reactor Safety"
 TID-8009; 1956.

- 17 Author Unlisted
 "The Status of the Reactor Safety Program"
 Technical Operations, Inc., Arlington, Mass.
 NYO-4720; July 17, 1956.

- 18 Author Unlisted
 "Safety of HRT"
 CF-54-3-164; 1955.

- 19 Author Unlisted
 "HRT Safety Calculations"
 CF-55-2-75; 1955.

III ACCIDENTS

- 1 Gilbert, F. W., (Atomic Energy of Canada Ltd., Chalk River, Ontario)
"Decontamination of the Canadian Reactor"
Chemical Engineering Progress 50, 267-71; May 1954.

The restoration program necessitated by an accident which occurred at the Chalk River project when its reactor released into the building and equipment thousands of curies of fission products is described. Some details of the original accident are given along with an explanation of the problems and techniques. Generally, it is hoped to demonstrate that severe pile accidents need not result in hazards to personnel, either associated with the equipment or located in the surrounding district; also that such accidents need not result in complete loss of plant, but that, through the various steps of reconstruction and decontamination, the reactor can be returned to normal operation.

- 2 Hayes, Daniel F.
"A Summary of Accidents and Incidents Involving Radiation In Atomic Energy Activities, June 1945 Through December 1955"
Atomic Energy Commission, Division of Organization and Personnel,
Safety and Fire Protection Branch
TID-5360; August 1956.

Criticality incidents, reactor incidents, contamination incidents, fires and explosions involving radioactive material, and miscellaneous accidents involving radiation which occurred from June 1945, through December 1955, are described. Safety measures resulting from review of the accidents are included.

- 3 Lewis, W. B.
"The Accident to the NRX Reactor on December 12, 1952"
Atomic Energy of Canada Ltd., Chalk River Project, Chalk River, Ont.
DR-32; July 13, 1953.

Because of a complex occurrence of mechanical defects in the shut-off-rod system and operating errors which alone would not have caused serious trouble, a power surge occurred in the NRX reactor during preparations for experiments at low power. Some of the cooling arrangements at the time were adequate only for low-power operation. Consequently some of the natural-uranium metal melted and ruptured the aluminum sheathing and tubes which separated the heavy-water, air, and cooling-water systems. As a result some 10,000 curies of fission products from long-irradiated uranium were carried by a flood of 1,000,000 gal of cooling water into the basement. Fused masses of highly irradiated uranium and uranium oxide were left inside the calandria, and the core vessel of the reactor and tubes of the calandria were severely damaged. In such a high-flux reactor where the transient xenon poison may affect the reactivity by 40 milli-k(mk), the shut-off rods have to cover a reactivity range of

about 70 mk. As one lesson from the accident it appears preferable to withdraw the first or safeguard bank of shut-off rods soon after shutting down, instead of making this the first step of the actual start-up.

-4

Robinson, M. S.

"Failure of WAPD-30 High Pressure Loop"

Phillips Petroleum Company, Idaho Falls

AD-134 913; April 19, 1957.

The WAPD-30 high-pressure loop consisted of an AISI Type 316 stainless steel tube having an o.d. of 3.50 in. and an i.d. of 2.90 in. After approximately 3 months and 2060 megawatt days service in the MTR at a temperature of 600°F and a pressure of 2200 psi. the tube developed a leak which necessitated its removal from service. Water and NO₂ gas were known to have collected in the annulus surrounding the tube and observation of the failed tube tip indicated a liquid type corrosive attack from the outside of the tube. No₂ formation during operation and moisture collection in the annulus during shut down periods together with boiling during the start-up of the loop probably concentrated the existing nitric acid into the constant boiling and fuming range of concentration. The corrosion of type 316 stainless steel in constant boiling or fuming nitric acid exceeds 50 mil/yr. and its use, under these circumstances, would be questionable; however, the approximate 1000 mil/yr. corrosion rate of the WAPD-30 tube would not be expected. The additional corrosion rate of the WAPD-30 tube can be probably attributed to the operation temperature and pressure, noncontinuous type of operation and the unknown effects of neutron and gamma irradiation.

IV CONTAINMENT

- 1 Beall, S. E.
"Containment Problems in Aqueous Homogeneous Reactor Systems"
Oak Ridge National Laboratory, Tennessee
ORNL-2091; August 8, 1956.
Containment problems in homogeneous reactors are discussed, with particular emphasis given to the manner in which they have been solved for the HRT. Bases for rigid leakage specifications on reactor components are explained. The design of the reactor enclosure is reviewed with respect to allowable leakage, as well as protection against fragments and internal pressures resulting from a reactor catastrophe.

- 2 Heineman, A. H.; and Fromm, Leonard W. Jr.
"Containment for the EBWR"
Paper presented at the 2nd Nuclear Engineering and Science Conference
1957 .

- 3 Kolflat, Alf; and Chittenden, W. A.
"A New Approach to the Design of Containment Shells for Atomic Power Plants"
Sargent and Lundy, Chicago, Illinois; March 1957.

- 4 Kolflat, Alf; and Chittenden, W. A., (Sargent & Lundy, Chicago, Illinois)
"Containment Vessel Can Be Reduced"
Electrical World, 1957.

- 5 Lindsay, A. L., et al, (Kaiser Engineers, Inc.,)
"Design and Testing of Containment Provisions for the Engineering Test Reactor"
The American Society of Mechanical Engineers, New York, June 1957.

- 6 Raymo, Arthur J.,
"Power Reactor Containment Vessels"
Paper presented at the 2nd Nuclear Engineering and Science Conference
1957.

- 7 Siler, William C.; and Zick, Leonard P.
"Design Considerations for an Atomic Power Reactor Containment Structure"
Paper presented at the 2nd Nuclear Engineering and Science Conference
1957.

- 8 Wise, Walter R. Jr.
"NOL Reactor Vessel Containment Program"
Naval Ordnance Laboratory, White Oak, Md.
NAVORD 4542; June 28, 1957.

Effort is being made to produce basic information which will assist in determining the optimum containment design of nuclear reactor vessels. It is planned that the purpose will be achieved through a fundamental investigation of the structural response of water-filled, steel cylinders to internal static and dynamic loading. Progress of the work consists essentially of designing the mechanical paraphernalia for scaling and constraining the model test cylinders, and designing and developing the means for monitoring the pressure, temperature, and strain phenomena which accompany excursion-simulated loading of the model vessels. Paralleling experimental effort, the energy equations of dynamic equilibrium for right-circular cylinders closed with rigid constraints at the ends have been derived in closed form. These equations constitute the fundamental analytic tool for the work, and in conjunction with experimental data, will determine the maximum energy absorption potential of various materials, configurations, and constraints; and hence, optimum containment design.

PRACTICAL ASPECTS OF REACTOR DESIGN

V

MATERIAL PROPERTIES

- 1 Anderson, Roger A.; and Anderson, Melvin, S.
"Correlation of Crippling Strength of Plate Structures with Material Properties"
Langley Aeronautical Laboratory
NACA-TN-3600; January 1956.
A correlation approach to the crippling-strength analysis of plate structures in new materials and at elevated temperatures is presented. Appropriately defined crippling-strength moduli and correlation procedures are given for predicting the effect of a change in material properties on the strength of a structure. The strength moduli are readily calculated from the effective compressive stress-strain curve for the structural material. The correlation procedures are applicable to multiplate-element components and the accuracy is illustrated with available experimental data obtained in various materials and under different temperature conditions.
- 2 Bohlman, E. G.; and Adamson, G. M.
"Stress-Corrosion Cracking Problem in the HRT"
Oak Ridge National Laboratory
CF-57-1-143; January 31, 1957.
Chloride-induced stress-corrosion cracking has been encountered in the HRT during the preliminary testing. The reactor is constructed of austenitic stainless steels. The cracking has occurred in a secondary system used for detecting leaks in the flanged joints of the primary systems. Tubing used in the leak-detection system was found to be contaminated with chloride introduced during manufacture. Results of the laboratory studies and studies in engineering loop experiments are discussed.
- 3 Class, I.
"Stress Corrosion Phenomena in Mechanically Strained Steel Due to Diffusing Hydrogen"
Translated by R. J. Richardson from Werkstoffe u Korrosion 6, 1955
AERE-Lib/Trans-568.
- 4 Demer, L. J.
"Fatigue Crack Detection Methods. Part 2: Review of Experimental Data on the Initiation and Propagation of Fatigue Cracks in Test Specimens"
University of Minnesota
WADC-TR-55-86, (Pt.2); June 1955.
Current fatigue literature contains numerous references to a few recent studies in which fatigue cracks have been detected very early in the specimen life. This paper attempts to clarify such isolated observations by making a detailed review of the available experimental data in relation to crack initiation and growth in laboratory fatigue test specimens. The data are analyzed considering the nature of the test material, the character of the specimen, the stress level

of the test, and the manner of testing. Conclusions are reached in regard to the effect of these factors on the percentage of fracture cycles at which fatigue cracks are detected. The formula of A.K. Head for the actual growth of fatigue cracks is reviewed, and its agreement with experimental results is discussed.

- 5 Derman, Cyrus
"Some Remarks on the Endurance Limit Problem"
School of Engineering, Columbia University
NP-5764; August 1, 1955.
The determination of the largest possible stress a standard test specimen can be subjected to and still not suffer fatigue failure is considered. The purpose of the experiment is to formulate the problem in the language of probability theory and to suggest a possible approach.
- 6 Dolan, T. J.
"Fatigue as a Factor in Pressure Vessel Design"
University of Illinois, Urbana
Welding Journal, (New York), 33, 265s-75s; June 1954.
The factors influencing the fatigue strength of metals are discussed from the viewpoint of their applicability to design of pressure vessels.
- 7 English, J. L.
"Homogeneous Reactor Corrosion Studies"
Oak Ridge National Laboratory
CF-50-4-98; April 17, 1950.
A resume of the results of corrosion tests with zirconium and stainless steels for the homogeneous reactor in uranyl sulfate solution at 250°C is presented.
- 8 Epstein, Benjamin
"Outline of Research in Life and Fatigue Testing"
Dept. of Mathematics, Wayne University, Detroit
AD-25302; January 31, 1954.
A research project on the statistics of life and fatigue testing was initiated at Wayne University. The principal objective of the research was to develop statistical techniques of estimation and decision making which make maximum use of the information arising from life and fatigue tests.
- 9 Gillig, Frank J.; and Guarnieri, Glen J.
"A Study of the Tensile and Creep-Rupture Properties of Fifteen Heats of C-110M Titanium Alloy Sheet"
WADC-TR-55-164; January 1956.

Fifteen heats of Ti alloy RC-130-A (C110-M) have been sampled and tested at room and elevated temperatures. These data are analyzed for reproducibility and relationships between the room temperature and high-temperature properties at 500°F and 700°F. The results indicate that a correlation exists between the room-temperature strength properties and the yield and ultimate strengths at 500 and 700°F. However, the creep and rupture properties appear to be independent of the short-time tensile strength results even at 700°F, which was the temperature used for creep testing.

- 10 Gross, J. H.; and Stout, R. D., (Lehigh University).
"Plastic Fatigue Properties of High-Strength Pressure-Vessel Steels"
Welding Journal 34; April 1955.
This is the fourth of a series of papers covering a study of the resistance, of steels applicable to pressure vessels, to repeated loading in the plastic range. Fatigue test results are reported for the following steels: A-201, A-225, 48s5, Fortiweld, A-302, 70A, 70B, 90A, and 90B.
- 11 Gross, J. H.; and Stout, R. D.
"The Performance of High-Strength Pressure Vessel Steels"
Welding Journal, 35; March 1956.
Strength and ductility characteristics of A302, 48s5HT, 70B, 70A, 90A, and 90B are tabulated. The effects of stress relieving, strain aging, welding, and post heating on notch toughness of these steels are graphically presented. The 90A steel compares favorably with the standard steels in pressure vessel construction.
- 12 Hausner, Henry H.,
"Materials Problems in Nuclear Power Reactors"
Materials and Methods 40, 84-6; July 1954.
The problems discussed concern: (1) mechanical behavior at elevated temperature including tensile strengths, creep, impact and fatigue strengths and stress to rupture of clad parts at elevated temperatures, (2) corrosion behavior, (3) heat transfer, (4) thermal shock behavior of metal-nonmetal combinations, and (5) radiation damage.
- 13 Hoff, N. J.
"Rapid Creep in Structures"
Brooklyn Polytechnic Institute
PIBAL-254; May 1954.
A survey of the effects of rapid creep upon the stress distribution in structures and of the various modes of creep failure is presented. It is proposed that rapid creep should be tolerated when aerodynamic heating causes high temperatures for only a short time in structural elements of supersonic aircraft.

- 14 Kick, F.
 "Contribution to the Knowledge of the Mechanics of Brittle Materials"
 Translated by F. Hudswell from Dinglers Polytech. J. 247, (1833)
 AERE-Lib/Trans-741.
 The application of the principle that governs the change of shape of ductile bodies to the comminution of brittle substances, such as cast iron, glass, and stones, is discussed.
- 15 Mercks, K. R.
 "Developments Necessary for General Theories of Creep Useful in Stress Analysis"
 Hanford Atomic Products Operation
 HW-31650; March 21, 1954.
 The theory of plastic deformations and the theory of plastic flow are extended to cover the dependence of strain upon stress, temperature, and time. It is hoped that the formulation of these two theories may suggest modifications in creep theory.
- 16 Mott, N. F.
 "A Theory of Fracture and Fatigue"
Journal Physical Society Japan, 10; August 1955.
 The concept of a piled-up group of dislocations is discussed, and its relation to ductile fracture, to brittle fracture, and to fatigue. It is suggested that a large enough stress induces fracture at a piled-up group, and a large number of such small cracks can join together to produce a ductile or fatigue fracture.
- 17 Niederberger, R. B.
 "Evaluation of High Yield Strength, Low Alloy Steels in Heavy Plates for Pressure Vessels"
 Naval Engineering Experiment Station
 AD-135 134; May 10, 1957.
 This report presents the results of tests of material from 3-, 6-, and 8-1/2 in. thick plates of a normalized and drawn, manganese-molybdenum steel. The material, conforming to Grade B of ASTM Specification A302, is a 50,000 psi minimum yield strength steel that is currently being used for pressure vessels of nuclear power plants. It is the first of several steels to be evaluated in the investigation of properties of high strength steels in heavy plates. The test results include tensile and impact data for three locations (surface, one-fourth thickness, and center) and two directions (parallel and normal to the direction of rolling) in each size of plate. A more thorough evaluation of the one-fourth thickness location of the 6-in. plate includes creep, rupture, and fatigue data for 800°F test temperature, and information on the hardenability, thermal expansion characteristics, and susceptibility of the material to strain aging and temper brittleness.

-18

Schijve, J.

"The Fatigue Strength of Riveted Joints and Lugs"

Translated by J. Vanier from Nationaal Luchvaartlaboratorium
NACA-TM-1395; May 1954.

Tests on riveted joints and lugs for the primary purpose of comparing the several types of riveted joints and the effect of various factors on the fatigue strength of lugs are reported. The application of single pulsating tension loads presented no difficulty. At the NLL some series of experiments of riveted joints and lugs have been performed. However, these do not completely cover the important part of the fatigue diagram and more experiments are recommended. A short discussion is given on the stress concentration factor K_t and three fatigue strength reduction factors, K_f , defined in different ways. The relation between K_f and K_t is considered. The various factors which might influence the fatigue of riveted joints are reviewed. For lugs the most interesting experiments and the methods to improve the fatigue strength are discussed.

-19

Sines, George

"Failure of Materials Under Combined Repeated Stresses with Superimposed Static Stresses"

University of California, Los Angeles

NACA-TN-3495; November 1955.

A review of experiments on biaxial alternating stresses and simple combinations of static stress with alternating stress is shown to lead to a general criterion for the effect of static stress on the permissible amplitude of alternating stress. The proposed criterion is then shown to agree with results from tests that have been performed under more complex stress states. Tests were performed to determine the effect of static compression on alternating torsion, the one simple combination that had not been previously investigated. These results also agree with the criterion. It is shown that Orowan's theory of fatigue can be modified to predict the observed effect of mean stress on the permissible amplitude of alternating stress.

-20

Smith, K. F.; and Van Thyne, R. J.

"Selected Properties of Vanadium Alloys for Reactor Application"

Argonne National Laboratory (in cooperation with Armour Research Foundation)

AD-135 136; May 1957.

Selected vanadium-base alloys have been investigated as to tensile and creep strength, corrosion resistance, and thermal conductivity at elevated temperatures for the purpose of making a preliminary evaluation of suitability for use as fuel-element jacketing materials for sodium-cooled reactors. Results indicate the preliminary feasibility of these alloys from all viewpoints under examination and suggest the desirability of studying fabrication problems of alloys and improved reduction methods for vanadium metal.

- 21 Sugihara, Seiji
"Some Tests for Minimum Life of Fatigue Failure Survivorship Functions"
School of Engineering, Columbia University
AD-84867; January 31, 1956.
The statistical analysis of fatigue failure, assuming known characteristic life of machines, was made. The asymptotic theory is used based on the central limit theorem and the δ method.
- 22 Thompson, A. S.
"Empirical Correlation for Fatigue Stresses"
Oak Ridge National Laboratory
CF-55-4-34; April 5, 1955.
Tentative presentation is made of a correlation by which better use of existing experimental fatigue data can be made for predicting the performance of structural materials in regions where no data exist. The correlation would be useful in utilizing data from fatigue failures under cyclic loading in the range of 10^{10} cycles for designs to be used in the range of 0.5 to 10^3 cycles.
- 23 Tucker, J. T., Jr., (Babcock & Wilcox Co., Research Center)
"Studies on the Biaxial Fatigue Properties of Pressure Vessel Steels"
Welding Journal 34; April 1955.
Bowman and Dolan presented data and interpretation of the data on high-strain-range-fatigue of high-strength steels, making comparison with some of the more conventional steels used in the manufacture of pressure vessels. In this paper the author adds a discussion to Bowman and Dolan's paper to include some additional data to supplement those already presented, and to show a comparison of biaxial with uniaxial data.
- 24 Weissmann, Sigmund
"Quantitative Study of Substructure Characteristics and Correlation to Tensile-Property of Nickel and Nickel Alloy"
Journal of Applied Physics 27; November 1956.
The application of a method combining x-ray microscopy and diffraction analysis to the study of the substructure of nickel and nickel alloys disclosed the coexistence of three orders of magnitude of substructural entities. These three orders are structurally interrelated and distinguished by different ranges of size, disorientation angle, and lattice misalignment. The statistical analysis of the data disclosed that for the two largest orders of substructural entities increasing amounts of room temperature deformation cause (1) a decrease in the mean size and a decrease in the spread of the size distribution, (2) an increase in the mean disorientation angle between substructural entities of the same order and an increase in the spread of the distribution of angles, and (3) an

increase in the mean lattice misalignment and spread in the distribution of misalignment. Progressive room temperature deformation causes the low-angle boundaries to become increasingly unsymmetrical and greatly increases the number of the smallest substructural entities. The rise of flow stress as a function of room temperature deformation was correlated to the systematic modification of substructure characteristics.

-25

Yerkovich, L. A.; and Guanieri, G. J.

"Compressive-Creep Properties of High-Temperature Materials"

Cornell Aeronautical Laboratory, Inc.

WADC-TR-54-582; November 1954.

An investigation concerning the behavior of high-temperature alloy and metal-ceramic materials was conducted to provide data regarding the creep characteristics of the test materials when subjected to static compression stresses in the temperature range of 1350 to 1800°F.

-26

Chechulin, B. B.

"On the Statistical Theory of Brittle Strength"

Translated by Lloyd G. Robbins, from Zhur, Tekh. Fiz. 24

AEC-Tr-2651; 1954.

A critical analysis is given of the main points in statistical theories proposed by Weibull and Kontorova and Frenkel. It is established that some inadequacies in these theories result from poor theoretical assumptions concerning a statistical distribution of dangerous defects, which are statistically distributed about the volume of the body and are responsible for the general disintegration of the body when under stress. By assuming a Type III Pearson distribution function as best substantiated for the given case, a new solution of the statistical theory by Kontorova and Frenkel is mathematically presented; moreover, the accuracy of this solution in comparison with the previous solution is demonstrated. It is demonstrated that the Weibull equation constitutes a special case of the proposed formula for brittle strength when the number of defects in the volume of the body is great.

VI STRESSES

- 1 Anderson, L. E.
"Symmetrically Loaded Cylindrical Shell with Fixed Ends"
Oak Ridge National Laboratory
CF-55-9-37; September 7, 1955.
Formulas are developed for the stresses and deflections in a circular cylindrical shell which is submitted to a uniform internal or external pressure. No axial forces are considered to be acting on the shell. The wall thickness of the shell is taken as constant and the ends of the shell are considered to be rigidly held.
- 2 Dubuc, Julien; and Welter, Georges, (Ecole Polytechnique, Montreal, Canada)
"Investigation of Static and Fatigue Resistance of Model Pressure Vessels"
Welding Journal 35, New York; July 1956.
Results of an investigation sponsored by the Pressure Vessel Research Committee indicate remarkable performance for vessels subjected to extreme static and cyclic loading. These tests point out the need for careful design details in order to minimize stress concentration effects which produce fatigue failures.
- 3 Esslinger, Maria
"Static Calculation of Boiler Bottoms" (Statische Berechnung von Kesselboden)
Published by Julius Springer, Berlin
AEC-tr-2676; 1952.
A method of calculating force distribution in boilers of various sizes is discussed, and the equations for numerical solutions are given. The boilers considered are assumed to have cylindrical sides and hemispherical ends.
- 4 Esslinger, Maria
"Static Calculation of Boiler Bottoms" (Statische Berechnung von Kesselboden)
Published by Julius Springer, Berlin
AEC-tr-2850; 1952.
Differential equations are developed for evaluating stresses and strains in cylindrical shell boilers. Boilers having riveted or welded seams and concave or convex ends are considered.
- 5 Hoff, N. J.; Kempner, J.; Nardo, S.V.; and Pohle, F.V.
"Deformations and Stresses in Circular Cylindrical Shells Caused by Pipe Attachment, Part 1, Summary of the Investigation"
Brooklyn Polytechnic Institute, (subcontract to Knolls Atomic Power Laboratory)
KAPL-921; November 1953.
The state of stress and deformation is calculated for a thin-walled circular cylindrical container when a radial load, a circumferential moment, and a longitudinal moment are transmitted to it through the

attachment of a pipe. In a first approximation the load can be distributed along a segment of a generator of a length slightly smaller than the diameter of the pipe. In a first correction to the stress quantities so obtained the pipe can be considered as a rigid body. The solution of the basic problem was obtained by using Donnell's differential equations of the thin-walled circular cylindrical shell. Explicit expressions were derived for all the deflection and stress quantities corresponding to an arbitrary Fourier component of the load function. The correction theory was developed from the differential equations of flat circular plates. The effect upon the stresses and deformations of changes in the physical parameters was explored. The accuracy of the solution was tested by comparing numerical results obtained by it with results calculated from a more complete set of differential equations. In the single case so treated the differences found were well within the requirements of engineering accuracy. Numerical results were also compared with published experimental data, and satisfactory agreement was found.

- 6 Horvay, G.; and Clausen, I. M.
 "Stresses and Deformation of Flanged Shells"
 Knolls Atomic Power Laboratory
 KAPL-M-GH-14; October 21, 1952.

Shell bending formulas, except those for cylindrical shells, do not seem to be available in a suitable form to be used in design without considerable labor. The following formulas for axisymmetric loading of axisymmetric shells are provided: membrane formulas for weight, gas pressure, liquid pressure, and temperature loadings (of conical and spherical shells), zeroth approximation bending formulas for arbitrary thin-walled shells; first approximation bending formulas for spherical and conical shells; flange deformation and stress formulas; and flange and shell interaction formulas.

- 7 Johanson, E. B.
 "Reinforcing Ring for Openings in Pressure Vessel"
 Knolls Atomic Power Laboratory
 KAPL-M-EBJ-3; November 16, 1955.

The feasibility of reducing the stress concentration factor near pressure vessel openings to allowable values through the addition of material near the opening was considered. The addition of bending stresses by the reinforcement material allows only approximate calculations, however, the method of calculation and graphical results are given.

- 8 Kempner, Joseph
 "Deformations and Stresses in Circular Cylindrical Shells Caused by Pipe Attachments. Part 4, Work Charts".

Brooklyn Polytechnic Institute, (subcontract to Knolls Atomic Power Laboratory)
KAPL-924; February 1953.

As a first step in the analysis of shells loaded by pipe attachments, a method was developed (KAPL-922) for the determination of deformations and stresses in circular cylindrical shells under the action of concentrated or approximately concentrated loads. Most of the results are presented in a form suitable for efficient computation of significant displacement and stress quantities. Experience has shown that the use of an extensive set of tables expedites the numerical work involved in the analysis of a given shell. Such a set of tables is constructed on the basis of the displacement and stress equations presented. These tables enable a computer to determine the radial displacements, slopes, bending stresses, and membrane stresses in a circular cylindrical shell simply supported at its ends and loaded midway between the ends by approximately concentrated (or concentrated) radial forces, axial moments, and circumferential moment. It is also shown that the tables can be adapted to a shell analysis for applied radial forces, axial moments, and circumferential moments, away from the midpoint of the loaded generator of the shell.

-9

Kempner, Joseph

"Deformations and **Stresses** in Circular Cylindrical Shells Caused by Pipe Attachment, Part 6, Derivation of Generalized Donnell-Type Equations for Circular Cylindrical Shells with Application to Shells with Line Loads Along Generatrices"

Brooklyn Polytechnic Institute, (subcontract to Knolls Atomic Power Laboratory)

KAPL-926; June 1953.

Mathematical equations for deformations and stresses in a cylindrical shell caused by pipe attachments are derived. The equations are solved for cases where the thin cylindrical shell is subjected to surface and edge loads.

-10

Kerkhof, W. P., (N.V. de Bataafsche Petroleum Naatschappij, The Hague, Netherlands).

"Stresses in Welded Pressure Vessels"

Welding Journal 35, New York 41s-59s; January 1956.

The allowable stresses, design stresses, and rupture theory for welded pressure vessels and boilers are discussed.

-11

Langer, B. F., (Westinghouse Electric Corp.)

"Working-Stress Criteria for Nuclear Power Plants"

Transactions of the American Society of Mechanical Engineers 77

July 1955.

A method is proposed for combining the stresses produced by fluid pressure with those produced by thermal gradients in order to arrive at a practical estimate of the factor of safety in a structure such

as a pressure vessel. Because of the relaxation of thermal stress it is possible to show that certain operating conditions are really safe even though a simple addition of pressure stress and thermal stress gives too high a value. The relative importance in reactor design of such physical properties as ductility, creep, endurance limit, creep-rupture strength, impact strength, and notch sensitivity is discussed.

- 12 Langer, B. F.
"Photoelastic Tests of Pressure Vessel Closures"
Westinghouse Electric Corporation, Atomic Power Div., Pittsburgh
WAPD-CTA (CE)-47; November 16, 1955.
Recommendations for, and discussion of, the investigative program for perforated pressure vessel heads are presented.
- 13 Mathews, R. L.; and Horvay, G.
"Torus Analysis"
Knolls Atomic Power Laboratory
KAPL-1072; February 22, 1954,
A 60-in. diameter torus (a continuous hollow piston ring) is made of steel tubing with a 3/8-in. tube diameter and a 10 mil wall thickness. It is used to restrict liquid-sodium flow in the rotating plugs assembly. A design problem arises. (a) What is the maximum radial compression of the continuous torus, by forces applied on the outside equator, which does not cause permanent deformations? (b) How does the deformation of a slit torque, restrained from shrinkage along the inner equator and squashed by forces applied on the outside equator, compare with the above? The possibility of buckling is to be ignored in these considerations. For simplicity of analysis, the continuous torus may be replaced in the calculations by an equivalent torus of square cross section. The answers so obtained are estimated to be reliable within 15%. For the slit torus, figures are given both for circular and for square cross section. The deflection values corresponding to a maximum stress of 30,000 psi indicate that the outside radius of the continuous torus can be compressed 25 mils without permanent deformation; this compares with the 3.3-mil compression of the slit circular tube. The method of analysis developed can be used equally well for redundant shell structures, such as two concentric cylindrical shells joined at the top and bottom by common flanges. In fact, the first application of the method pertained to such a structure and likewise concerned the rotating plugs assembly.
- 14 Mesnager, M.
"Method of Determining the Stresses in a Circular Cylinder"
Translated by K. Bevis from Compt. Rend. 169, 1919
AEC-TR-2471.

Stresses directed along the radius or perpendicular to the meridian planes are considered along with the stresses parallel to the cylindrical axis in calculating internal stresses developed in a cylinder by annealing.

- 15 Narduzzi, E. D., (University of Montreal & Dominion Bridge Co., Ltd., Lachine, Quebec, Canada)
"High-Pressure Vessels Subjected to Static and Dynamic Loads"
Welding Journal, (N.Y.); May 1954.
Elastic stress-strain relations for external and internal walls for cylindrical shells subjected to internal pressure were derived. Actual stresses were compared to theoretical stresses. The application of various theories of fracture is discussed. Suggestions are made for design formulas.
- 16 Roche, M. E.
"Deformations and Stresses in Circular Cylindrical Shells Caused by Pipe Attachment" Part 7, Work Charts for Off-Center Loading
Knolls Atomic Power Laboratory
KAPL-1025; December 17, 1953.
Stresses and deformations in cylindrical shells due to loads caused by thermal expansion of attached piping systems were calculated. These equations have been used in stress analysis of the SIR Mark A and Mark B reactor containers for loadings due to the expansion of the primary coolant system. A method was developed for evaluating the deformation and stresses in circular cylindrical shells loaded by pipe attachments located away from the mid-point of the generator. By using this method the equations for the off-center case with an applied radial force P^* , axial moment M_x^* , and circumferential moment M_ϕ^* , have been developed. From these equations a set of tables has been developed to facilitate the numerical evaluation of these displacement and stress expressions. These tables lend themselves to the calculation of the displacement and stress quantities without recourse to the general equations. The radial displacements, slopes, bending stresses, membrane stresses, and shear stresses in a circular cylindrical shell simply supported at its ends and loaded at some point away from the center of the generator of the shell can be computed directly from these tables.
- 17 Sampson, R. C; and Leven, M. M.
"Photoelastic Analysis of Stress in Reactor Pressure Vessel STR Core 4 Preferred Head"
Westinghouse Electric Corporation, Bettis Plant
AECU-3268; January 6, 1956.
A model of the STR Core 4 preferred head, having seventeen holes, was tested by the "frozen stress" photoelastic model technique. The highest stresses due to internal pressure occur on the ligaments between holes at the top surface, where the maximum tensile stress was found to be a little more than nine times the value of

the internal pressure exerted on the vessel. In general, lower tensile stresses were found on the hole boundaries at the inner surface of the head.

- 18 Savage, H. C.
"MTR Strain Measurements"
Oak Ridge National Laboratory
ORNL-472; November 22, 1949.
The magnitude of stresses in various parts of the Heterogeneous Pile (MTR) Mock-up under operating conditions were determined. Baldwin-Southwark "SR-4" resistance wire strain gages were used in making all strain measurements from which values of stress were calculated. All stresses were found to be well under the allowable yield strength of the materials involved. A maximum stress of 1875 psi existed in the aluminum tank section D when the maximum water flow rate of 14,500 GPM was used. At this flow rate the pressure in Tank A was 41 psi and 13 psi in Tank E.
- 19 Schlechte, Floyd R.; and Rosecrans, Richard
"Experimental Stress Analysis of Stiffened Cylinders with Cutouts. Shear Load"
Langley Aeronautical Laboratory
NACA-TN-3192; March 12, 1954.
A cylindrical semimonocoque shell of circular cross section was mounted as a cantilever and loaded by a direct shear at the tip. The cylinder was tested with no cutout on the tension side, and with the cutout centered on the neutral axis on one side of the cylinder. The cutout was successively enlarged through six sizes varying from 30° to 130° in circumference and from 1 to 2 bays in length. Strain measurements were made with resistance-type wire strain gages near the cutout on the stringers, the skin, and the rings for each case, and the stresses obtained are presented in tables.
- 20 Voorhees, H. R.; Sliepcevich, C. M.; and Freeman, J. W., (University of Michigan, Ann Arbor)
"Thick-Walled Pressure Vessels"
Industrial Engineers Chemical, 48; May 1956.
Changes in stress distribution in thick-walled pressure vessels under internal pressure at elevated temperatures are analyzed, and a procedure is developed for calculating the time of rupture from creep and stress rupture data.
- 21 Walker, J. W.
"Stress Analysis of the SRP Reactor Tank"
DuPont de Nemours & Co., Savannah River Laboratory, Augusta, Ga.
DP-133 (Del.); October 1955.
Equations are derived for the discontinuity stress in a cylindrical tank that is mounted vertically on a header and is open at the top.

The analysis includes a derivation of equations for the prediction of the deflection and slope of the header by relating the strain energy in bending a circular plate to the strain energy in bending the header. The stresses that result from pressure and temperature are included in the analysis.

- 22 Wallace, J. J.
"Calculation of Wall Stresses in the "A" Tank of Oak Ridge Research Reactor"
Oak Ridge National Laboratory
CF-53-7-205; July 30, 1953.
- 23 Zickel, John
"Stresses Caused by a Step in a Cylindrical Shell"
Knolls Atomic Power Laboratory.
KAPL-M-JZ-2; June 15, 1953.
The effect of a longitudinal and a transverse force on a thin walled cylindrical shell with a built in step for reactor structures is discussed, disregarding stress concentrations from fillets and holes.
- 24 Zickel, J.,
"Stress Analysis of Concentric Plug, Mark A"
Knolls Atomic Power Laboratory
KAPL-M-JZ-3; March 5, 1954.
The upper half of the concentric rotating plug is a pressure vessel of complicated design. It also acts as part of the cover for the container which holds the power unit. Despite its name the plug is eccentric, and therefore, an exact analysis is almost impossible. An approximation can be made by analyzing an axisymmetric structure with dimensions of the narrowest, widest, and average section. The actual stress at these sections will not differ appreciably from the calculated values. Rotations and deflections, however, will not correspond too well because the torsion effect of the flange is neglected.
- 25 Zickel, J.
"A Circular Plate Under the Action of a Centrally Applied Moment"
Knolls Atomic Power Laboratory, Schenectady, N. Y.
KAPL-1454; September 28, 1954.
The analysis of the circular plate under the action of a centrally applied moment is useful in the design of pressure vessels or plates with pipe attachments where the pipes are appreciably stiffer than the plates or shell wall. A detailed analysis is presented with graphs of the existing stress coefficients. The expressions for stresses are developed in a convenient form with coefficients which are always finite. The stresses differ somewhat from those previously published.

VII THERMAL EFFECTS

-1 Aven, R. E. .

"Thermal Stress in HRT Pressure Vessel"

Oak Ridge National Laboratory, Tennessee

CF-55-2-128; February 21, 1955.

By deriving equations and making plots relating temperature in the pressure vessel wall with time, it is possible to compute the rate at which the surface temperature is changing for various temperature differences between the inside and outside surfaces. From this value it is possible to calculate thermal stresses in the pressure vessel wall. The results are presented. It is recommended that when the pressure vessel is heated or cooled, the temperature difference across the wall be limited to a maximum of 50°F. This will limit the rate of surface heating to 260°F. per hour and the thermal stress to 10,000 psi.

-2 Aven, R. E. .

"Heat Removal Systems for the HRT Pressure Vessel"

Oak Ridge National Laboratory, Tennessee

CF-55-3-119; March 14, 1955.

The thermal stress and the rate of heat removal from each surface of the pressure vessel wall as a function of temperature difference were determined by using the temperature distribution equation. The total maximum stress was found by adding the thermal stress, the stresses produced by the reactor design pressure and the expansion of the stainless steel cladding. It is recommended that the pressure vessel be cooled by circulation of cooling water through coils welded to the outside surface.

-3 Aven, R. E.

"Some Heat Removal Considerations of the HRT Pressure Vessel During Operation"

Oak Ridge National Laboratory, Tennessee

CF-55-7-64; July 15, 1955.

It is shown how the temperature at any point in the HRT pressure vessel may be calculated. Temperature profiles vs power level and curves of stress vs heat removal rate from outside surface of pressure vessel at various power levels are presented. The temperature drop across the blanket film and the total tangential stress in the pressure vessel wall are given for a pure D₂O blanket and for a 312.3 g U/liter UO₂SO₄ D₂O blanket solution.

-4 Bareis, D. W.; and Page, W. R.

"Supplementary Engineering Tests for the Brookhaven Reactor"

Brookhaven National Laboratory

BNL-46; November 15, 1950.

An extensive thermal analysis has been made of the structure of the BNL reactor. Use was made of the electric analog technique to obtain de-

tailed temperature patterns for various parts of the reactor structure for a number of different operating conditions. Structural changes recommended on the basis of the temperature information thus obtained were incorporated into the analogs to test the effectiveness of the changes. The analog data for both the original and final designs are presented along with a description of two auxiliary engineering tests which were run to determine air flow patterns over portions of the reactor structure where the amount of forced convection heat transfer would otherwise have been an uncertain factor.

- 5 Baroody, E. M.; Simons, E. M.; and Duckworth, W. H., (Battelle Memorial Institute, Columbus Ohio)
"Effect of Shape on Thermal Fracture"
Journal of the American Ceramics Society 38, 38-43; January 1955.
For radial steady-state heat flow through hollow circular cylinders, the usual application of thermoelastic theory indicates that the heat flow per unit length to cause fracture (W_{max}) is simply the product of two factors: M, characterizing the material, and S, the shape. Attention is given to its validity for cylinders with both circular and non-circular cross sections and to measuring the effect of dimensions on S for circular cylinders. Thermal fracture experiments on circular cylinders of two ceramics indicated the manner in which S varies with the ratios of dimensions which define the shape. Changing all dimensions by factors up to 1.6, with these ratios constant, had no noticeable effect on S. Using theorems derived by Biot, it is shown that the product relationship is expected to hold for a considerably wider variety of shapes than the simple circular tube. Experimental results from tubes having cross sections with a circular inner boundary support the product relation. Information was obtained on the shape factor for these cross sections and for similar tubes with a triangular outer boundary.
- 6 Buessem, W. R., (Pennsylvania State University, State College)
"Thermal Shock Testing"
Journal of American Ceramics Society 38, 15-17; January 1955.
Practical experience has shown that thermal shock tests do not lead to generally useful test data. This is probably due to the fact that thermal shock failure is a complicated function of the external thermal shock conditions and of the temperature functions of five different material properties. These five material functions appear in a different combination in almost every thermal shock case and cannot be extracted from thermal shock test data. It is recommended that these five properties and their temperature dependence be determined by separate standard tests, not employing thermal shock. If the five property functions are known, thermal shock tests proper can be used to determine the maximum thermal shock stresses in any device.
- 7 Carter, S. C..
"Temperature and Stress Distribution in Spheres, Rods, Tubes and Plates in Which the Heat Source is Within the Boundaries of the Solids"

Argonne National Laboratory, Lemont, Ill.
ANL-4690; 1951.

- 8 Chapman, R. H.
"Thermal Stress Analysis of a 6-Ft. Diameter, 50MW Single Region Reactor Vessel"
Oak Ridge National Laboratory, Tennessee
CF-52-12-145; December 12, 1952.
- 9 Clark, Melville, Jr.
"Axial Heat Distribution in Brookhaven Reactor"
Brookhaven National Laboratory, Upton, New York
BNL-123; July, 1951.
- 10 Cooper, W. E.; Roche, M. T.; and Noble, J. L.
"Stresses in a Semi-Infinite Thin-Walled Cylinder Caused by an Exponential Temperature Distribution"
Knolls Atomic Power Laboratory
KAPL-973; March 25, 1955.
The stresses in a long thin-walled circular cylinder subject to given conditions of end fixity and longitudinal temperature distributions have been studied. Two cases of end fixity have been considered, one in which one end is pinned to a rigid wall, and the other in which one end is fixed in a rigid wall. In both cases the other end is free. The temperature is constant for a certain length of the cylinder and then decreases exponentially. The techniques used in the stress analysis are described. Graphs are presented of normalized deflections and stresses as functions of axial distance.
- 11 Crandall, W. B.; and Ging, J., (State University of New York)
"Thermal Shock Analysis of Spherical Shapes"
Journal of the American Ceramics Society 38; January 1955.
A method is described for studying the thermal shock characteristics of a brittle material. An analysis of the thermal stresses developed in a homogeneous isotropic solid sphere has led to the formulation of an equation relating the physical properties of the body to the temperature difference causing failure and time to maximum stress in a single-cycle unsteady-state test. The thermal shock test consisted of plunging a sphere at one uniform temperature into a medium at another temperature. If fracture occurred, the time to **fracture was recorded**. A large number of tests were run to determine the temperature difference which caused 50% of the spheres to fracture. The thermal shock relationships were tested using a high alumina body. The physical properties relating to the thermal **shock** equations were measured, and calculated temperature **differences** causing failure and times to maximum stress were compared with measured values. Sufficient agreement was found to lend support to the theory.

- 12 Davis, D. M.; and Mount, B. H.
"The Calculation of Thermal Stresses in Cylinders with Internal Heat Generation"
Bettis Plant, Westinghouse Electric Corp., Pittsburgh
WAPD-TM-59; 1957.
- 13 Davis, Sam H.
"Thermal Stresses in Lower Pressurizer Walls During a Cold Startup"
Knolls Atomic Power Laboratory
KAPL-M-SMS-58.
A comparison of thermal gradients and stresses in the lower pressurizer walls for cold startup procedures for SAR with and without drainage indicates that a substantial reduction in the membrane stresses can be achieved by choosing the drainage procedure.
- 14 Durham, Franklin P.
"Heat Transfer and Thermal Stresses in Nuclear Reactor Shells"
Los Alamos Scientific Laboratory
LA-1590; September 1953.
A method is presented for determining heat transfer rates and thermal stresses from the gamma-ray energy absorption of nuclear reactor shells for plane, cylindrical, and spherical geometries. Criteria for minimizing thermal stresses are developed, along with the corresponding external cooling rates necessary to minimize the thermal stress. Design charts are presented for rapid determination of approximate thermal stresses and heat transfer rates, along with a numerical example illustrating the use of the charts.
- 15 Field, F. A.
"Temperature Gradient and Thermal Stresses in Bodies With Uniformly Distributed Volume Heat Sources"
Oak Ridge National Laboratory, Tennessee
AECD-3650; February 1955.
Determination of the thermal stresses induced in a solid in which heat is generated may serve to indicate the behavior of a material under operating conditions. Potential application of this to reactor fuel elements is foreseen, and a series of charts has been prepared to facilitate the determination of the maximum temperature differential and maximum thermal stresses induced in flat plates and thick-walled cylinders. Methods for preparing similar charts for other materials and geometries are included.
- 16 Fromm, L. W.
"Design Evaluation of In-Reactor Tube for Argonne Water Loop at MTR (ANL-2)"
Argonne National Laboratory
ANL-5403; March 1955.
A rigorous stress analysis is performed on a pressurized, in-reactor tube test facility (Type 347 stainless steel tube with Type 316 tip) to ensure conformity of the design with ASA and ASME Codes. Detailed

calculations, along with tabulated and graphical data, show that the maximum membrane stress (14,200 psi) and the combined membrane and thermal stresses (22,200 psi) are well below the yield point of the steel. Calculated thermal gradients are lower than those accepted in commercial practice. Experimentally determined stresses and temperature gradients verify or are more favorable than corresponding calculated values. Radiation damage data are presented to show that even at saturation levels, the properties of Type 300 series stainless steels do not approach the limits at which embrittlement failure might be anticipated.

- 17 Goertzel, G.
 "Stress and Strain Due to Thermal Expansion in a Rapidly Heated Sphere"
 Nuclear Development Associates, Inc., White Plains, N.Y.
 NDA-14-46; August 2, 1954.
 The time behavior of a rapidly heated spherical core of a nuclear reactor is studied. Inertial effects are considered. Spherical symmetry is assumed. Equations convenient for numerical evaluation for specified heating are presented.

- 18 Heisler, M. P., (North American Aviation, Inc., Downey, Calif.)
 "Transient Thermal Stresses in Slabs and Circular Pressure Vessels"
 Journal of Applied Mechanics 20,261-9; June 1953.
 The results of computations for determining transient thermal stresses in slabs and circular pressure vessels are presented. The process of solution adopted is to substitute transient temperature formulas into the already available stress expressions. The expressions for thermal shock are transformed by means of a simple integral theorem into a form appropriate for analyzing the thermal processes commonly used to relieve thermal shock. A new dimensionless stress parameter is defined and applied to the determination of optimum heating or cooling times of massive pressure vessels.

- 19 Hilton, Harry H.; Hassan, Hassan A.; and Russell, Hal G.
 "Analytical Studies of Thermal Stresses in Media Possessing Temperature-Dependent Viscoelastic Properties"
 University of Illinois, Urbana
 WADC-TR-53-322; September 1953
 The problem of thermal stresses in thick-walled and thin-walled cylinders and circular plates is analyzed for bodies with temperature-dependent viscoelastic properties of the Kelvin, Maxwell, and Standard Linear Solid type. A numerical example is computed for the Kelvin body thick-walled cylinder which shows that the elastic solution and temperature independent Kelvin body solution are extremely conservative. Alfrey's elastic-viscoelastic analogy is expected to cover viscoelastic thermal stress problems with temperature-dependent material properties.

- 20 Honohan, H. J.
 "Steady State Thermal Stress Distribution in Long Hollow Cylinders Experiencing a Radial Exponential Internal Heat Generation"
Master of Science Thesis, University of Pittsburgh, Pittsburgh, Penna. 1956.
- 21 Horvay, G. .
 "Stresses in a Disk Due to Non-Uniform Heating of Its Periphery"
 Knolls Atomic Power Laboratory
 KAPL-M-GH-6; May 21, 1951.
 An appraisal is made of the temperature and stress transients in disks subjected to edge heating. Treatment is given first to stresses and distortion in a solid disk when steady temperature distribution is maintained. Secondly, the transient in a solid disk due to applied edge temperature is treated. The results are extended to perforated disks of triangular layout.
- 22 Horvay, G.; and Montague, B. J.
 "Thermal Stress in Long, Thin-Walled Cylinders"
 Knolls Atomic Power Laboratory, Schenectady, N.Y.,
 KAPL-M-GH-9; October 17, 1951.
 A family of curves was prepared which give the stresses and deformation in thin-walled cylinders when a sinusoidal temperature drop occurs in the cylinder.
- 23 Horvay, G., (Knolls Atomic Power Laboratory)
 "Thermal Stresses in Rectangular Strips"
Proceedings of the Second U. S. National Congress of Applied Mechanics June 14-18, 1954. American Society of Mechanical Engineers.
 Stresses and deformations ensuing from application of a step function temperature distribution to a rectangular strip are calculated by precise biharmonic eigenfunction expansions, by Fourier integrals, and by self-equilibrating function expansions. Superiority of the latter method, because of its great speed and yet adequate reliability is clearly established. It is also shown that, contrary to expectation (whereby $1/2 E \Delta T = \sigma_0$) the temperature step T is related to the design stress σ_0 by $E \Delta T = 5/6 \sigma_0$.
- 24 Hucks, J. S.; and Gaines, A. L.
 "An Analytical Method of Predicting Temperature Gradients in Thick-Walled Pressure Vessels"
 Paper presented at ASME Conference, 57-A-231; December 1957.
- 25 Hunter, T. A.; Thomas, L. L.; and Bobrowsky, A. R.
 "Thermal-Shock Investigation"
 University of Michigan
 WADC-TR-54-206; September 1954.
 A program of investigation has been undertaken to evaluate the resistance of various material to thermal shocking. A preliminary analysis

of thermal-shock damage has been carried out on a theoretical basis. The results of this theoretical work indicate that the scope of the problem is so wide that purely analytical methods must be supplemented by experimental data. An experimental program has therefore been set up to test fourteen materials for their relative resistance to severe repeated thermal shock from the temperature range of 1600 to 2000°F. A suitable apparatus has been constructed and a standard specimen shape has been devised which give reasonable reproducibility of results. Excursions into the subjects of previous specimen history, mechanical fatigue, and thermal wiggling have also been made.

- 26 Jones, A. R.
"A Study of Temperature Distribution Through a Reactor in the Axial Direction"
Westinghouse Electric Corp., Industrial Atomic Power Section, McKeesport, Penna.
WIAP-M-7; April 23, 1953.
Relationships between the hot-spot temperatures, the coolant inlet temperature and the coolant temperature rise were developed.
- 27 Kapp, R. P.
"Mechanical Loading Versus Thermal Loading"
Westinghouse Electric Corp., Bettis Plant, Pittsburgh,
WAPD-TM-37; January 21, 1957.
While the evaluation of mechanically applied loads on a stress basis is realistic, the evaluation of thermal loads on this basis is shown to be extremely conservative. Evaluation of thermal loading on the basis of strain measurements is proposed.
- 28 Kroeger, H.; and Silks, T. M.
"Thermal Stresses in the Pressure Vessel of the APPR-1"
Alco Products, Inc., Schenectady, New York
APAE-Memo-16; May 1, 1956.
- 29 Langer, B. F.
"Approximate Formulas for Calculation of Thermal Stress in Reactor Vessels, Thermal Shields and Thimbles"
Westinghouse Electric Corp., Atomic Power Division, Pittsburgh
WAPD-CE-39; March 11, 1955.
Approximate formulas for the calculation of temperature and thermal stress in flat or slightly curved plates which are subjected to internal heat generation are given. Formulas for flat plates are exact and are within 5% for cylindrical shells whose thickness-to-diameter ratio does not exceed 1/10.
- 30 Leyse, C. F.
"Temperature and Stress Formulae for Cylinders with Heat Generated in the Material"
Phillips Petroleum Co., Atomic Energy Div., Idaho Falls, Idaho
IDO-16074; March 18, 1953.

General stress formulas for hollow and solid cylinders in which heat is generated in the material, e.g., in irradiations in a reactor, are presented.

- 31 Mack, Rex, C.
"Damage to Metal Plates From High-Intensity Short-Duration Heat Pulses"
Sandia Corporation, Albuquerque, N. M.
SC-3459 (TR); August 25, 1954.

Formulas have been derived whereby it is possible not only to determine whether a flat metal plate is being heated uniformly or nonuniformly, but the extent of any resultant damage to the plate. Under uniform heating the only immediate material damage that can result is melting of the entire mass of the plate. Nonuniform heating may damage the plate as a result of vaporization or thermal stress. When nonuniform heating is a direct result of unsteady heat input, the maximum tensile stresses occur inside the plate and stress-strain measurements at the surface are probably misleading. A formula for optimum rate of heat addition is presented.

- 32 Mahlmeister, J. E.
"Instantaneous Thermal Stresses and Temperatures in Unrestrained Flat Plates and Thin-Walled Cylindrical Tubes"
California Research and Development Co.
AEC-4004 (CRD-T2B-61; LWS-2419C; October 19, 1951.

- 33 Manson, S. S.; and Smith, R. W.
"Theory of Thermal Shock Resistance of Brittle Materials Based on Weibull's Statistical Theory of Strength"
Lewis Flight Propulsion Laboratory
NP-5336; 1953.

Use is made of the concepts of Weibull to predict the behavior of circular disks of brittle materials subjected to peripheral thermal shock. It is found that fracture most probably occurs not at the time when the surface stress is a maximum, but at a later time when the surface stress has fallen somewhat, but a greater volume of material in the interior of the disk has been brought up to moderate stress level. Utilizing the "risk of rupture" concept introduced by Weibull, a general relation is established for relating the conditions of fracture under varying degrees of quench severity. The analysis indicates that for materials having low homogeneity factors, m , - or materials in which the tensile strength differs appreciably from the bending strength - considerable error can be introduced by the use of the conventional maximum stress theory of fracture to relate the fracture conditions under mild and severe quenches. Errors as high as 30% can be expected in some practical cases. It is suggested how thermal shock data can be used to evaluate the material constant m . Analysis of limited data on the thermal shock characteristics of steatite disks indicated that for this material the experimentally determined value of m was sufficiently high to obscure possible small discrepancies arising from the use of the maximum stress theory

of fracture. The value of m as deduced from the thermal shock tests was in very good agreement with the value determined from a statistical study of the bending strengths of twelve small specimens. More data on a variety of materials would, however, be desirable for a full evaluation of the proposed theory particularly for cases involving low values of m for which the largest discrepancies arise.

- 34 McCollom, K. A.; Hanson, G. H.; and Grimoud, A. V.
"Materials Testing Reactor-Structure Temperatures"
Phillips Petroleum Co., Atomic Energy Division, Idaho Falls, Idaho
AECD-3976; February 13, 1953.

At 30 megawatts, the design power level of the MTR, no reactor structure temperatures were observed which exceeded the maximum permissible levels for the various components of the reactor. The maximum design variation within a given section of the thermal shield was 33°F, and the resulting calculated thermal stresses were considered acceptable. From thermal stress considerations the maximum acceptable temperature difference with a concrete structure is approximately 50°F. A qualitative evaluation of the effect of operating conditions on the structure temperatures has also been made.

- 35 McWherter, J. R.
"Thermal Stress in the TBR Pressure Vessel"
Oak Ridge National Laboratory, Tennessee
CF-54-10-146, TBR Memo No. 5; October 28, 1954.

Calculations were made to determine the combined pressure and thermal stresses in the Thorium Breeder Reactor pressure vessel wall as a function of the vessel wall and thermal shield thicknesses. The magnitude of the thermal stress depends on the heat generation in the wall and the method of cooling the wall. Heat is generated in the vessel wall as a result of radiation absorption. The principal types of radiation which contribute to the heat generation include thermal neutrons, fast neutrons, and gamma radiation. A sketch of the proposed vessel is included.

- 36 Mendelson, A.; and Manson, S. S.
"Approximate Solution to Thermal-Shock Problems in Plates, Hollow Spheres and Cylinders with Heat Transfer at Two Surfaces"
Transactions of the American Society of Mechanical Engineers, 78; April 1956.

An approximate method for computing transient thermal stresses in hollow cylinders, plates, and hollow spheres is presented. The method makes use of polynomial approximations to the temperature distribution by means of which the partial differential equation of the problem is reduced to a set of first-order ordinary differential equations. Nonuniform initial temperature distribution and heat transfer from both surfaces can be treated without difficulty making possible the practical solution of problems with relatively little labor for which an exact solution requires a great expenditure of labor and time. Several examples are presented and compare favorably with more accurate solutions.

- 37 Mendelson, Alexander; and Hirschberg, Marvin
 "Analysis of Elastic Thermal Stresses in Thin Plate with Spanwise and Chordwise Variations of Temperature and Thickness"
 Lewis Flight Propulsion Laboratory, Cleveland
 NACA-TN-3778; November 1956.
 An approximate method for calculating thermal stresses in a thin plate with variable thickness and temperature distributions in both the spanwise and chordwise directions is presented. Several examples are worked out and plotted.
- 38 Mercks, Kenneth R.
 "General Concepts of Mechanics and Their Relation to Thermal Stresses"
 Hanford Atomic Products Operation, Richland, Wash.
 HW-36863; May 24, 1955.
 Simple models are analyzed to demonstrate the importance of different material effects of predicted material behavior. The effects of stress, strain, and energy on the interpretation of the effects of thermal gradients are given for an understanding of basic assumptions in stress analysis. By describing material behavior in terms of energy a new approach to the study of material behavior is warranted.
- 39 Ostrander, N. C.
 "Transient Thermal Stresses in Unrestrained Flat Plates and Thin-Walled Cylindrical Tubes"
 California Research and Development Co., Livermore, California
 AECD-3941, (CRD-T2B-47); August 6, 1951.
 The magnitude of transient thermal stresses are calculated for the flat plates and thin-walled cylindrical tubes to be used as materials testing accelerator targets.
- 40 Powell, R. A.; and Poritsky, H.
 "Thermal Stresses in Hollow Cylinders"
 General Engineering Laboratory, General Electric Company
 GEL-90; R54GL269; October 22, 1954.
 The temperature distribution is obtained in a hollow cylinder in which heat is generated at a uniform constant rate and which is cooled internally. The thermal stress distribution under these conditions is then obtained first for an infinitely long cylinder, then for a cylinder of finite axial length. Application of the method to the case of both external and internal cooling is described.
- 41 Richards, Paul B., (Case Institute of Technology)
 "The Temperature Distribution in a Nuclear Reactor Pierced by Circular Cooling Channels"
 Nuclear Engineering Part I, Chemical Engineering Progr. Symposium; 1954.
 The present study has been conducted primarily to estimate the maximum temperature in a reactor pierced by equilaterally spaced circular channels through which the coolant flows. in the course of the mathematical analysis to obtain this maximum temperature, a general method

is devised by means of which the complete temperature pattern within the reactor may be determined in terms of the coolant temperature.

- 42 Robertson, Roy C.
"Study of Effect of Sudden Heating or Cooling on HRT Main Heat Exchangers and Possible Protective Measures"
Oak Ridge National Laboratory
CF-55-11-35.
A study was made on the effect of sudden heating or cooling on the elastic limit of the HRT main heat exchangers. Stresses due to sudden cooling due to incoming feedwater, sudden heating by steam, and sudden loss of steam pressure in the vessel are discussed. Procedures are outlined for protection against these conditions.
- 43 Savage, H. C.
"Thermal Stresses in HRE Reflector Tank"
Oak Ridge National Laboratory, Tennessee
CR-50-4-115; April 19, 1950.
Thermal stresses in the HRE reflector tank are calculated, consideration being given to geometry and temperature gradients. The influence of these stresses on tank cooling considerations is discussed.
- 44 Schulten, R.; and Gans, H., (Max Planck - Institut fur Physik, Göttingen, Germany)
"Temperature Rise in a Thermal Reactor as a Result of Cooling"
Z. Naturforsch; December 1954.
A formula for the maximum possible temperature rise within a thermal reactor as a result of cooling is derived.
- 45 Sonneman, G.; and Davis, D. I.
"Stresses in Long Thick-Walled Cylinders Caused by Pressure and Temperature"
Paper presented at ASME Conference, No. 57-A-256; 1957.
For a long thick-walled cylinder the non-dimensional radial, axial, and circumferential stresses have been evaluated for the condition of axial symmetry for the cases of pressure, logarithmic temperature distribution, and internal heat generation due to nuclear radiation.
- 46 Spiewak, I
"Thermal Stress in the HRE Reflector Tank"
Oak Ridge National Laboratory, Tennessee
CR-51-6-9; June 5, 1951.
The thermal stresses induced in the reflector tank and cover flange of the HRE have been evaluated. They are shown to be safe for normal operating conditions, and to permit overloads up to 6400 kw, in reactor power. The procedure followed in the calculations is presented in detail. It may be applied to similar thermal stress problems.

- 47 Spiewak, I.
 "Thermal Stress in the ISHR Pressure Vessel"
 Oak Ridge National Laboratory, Tennessee
 CF-52-10-81; October 9, 1952.
 Cooling of the outside surface of the pressure vessel is effective in reducing thermal stress and can be accomplished with cooling coils circulating borated water or with air circulation.
- 48 Thompson, A. S.
 "Thermal Stress in Tube-Header Joints"
 Oak Ridge National Laboratory, Tennessee
 CF-55-4-159; April 26, 1955.
 A study was made of the stress and axial variation to temperature in the attachment of a tube to a header using a larger concentric tube between the header sheet and the joint. A low tensile gradient can thus be maintained.
- 49 Vinokurov, L. G.
 "Thermal Stresses in Plates and Shells"
Translated from Izvest. Kazan. Filiala Akad. Nauk. U.S.S.R., Ser. Fiziko-Matemat. i Tekh. Nauk, No. 3, 18-38; 1953.
 (Available, Associated Technical Services (Trans. 96G6R), East Orange, N. J.). AEC-tr-2272.
 By introducing an auxiliary stress, transformation was made of the equations of elastic theory of a three-dimensional body--the equations of the theory of shells in generalized coordinates and in lines of curvature. Examples are given of the application of Galerkin's method of energy to the solution of non-linear problems in elastic theory which involve a nonuniform temperature distribution.
- 50 Wood, P. M.
 "Heating and Thermal Stress in HRT Pressure Vessel During a Shutdown"
 Oak Ridge National Laboratory, Tennessee
 CF-54-7-158; July 27, 1954.
 Calculations are presented on the temperature rise and thermal stress in the HRT pressure vessel when the reactor is dumped.
- 51 Young, G.
 "Thermal Stresses Arising from Defective Strip in Bond"
 University of Chicago, Metallurgical Laboratory
 CP-1707; May 18, 1944.
 Stresses set up by an insulating strip in the bond are estimated and found to be small compared to those normally present without the defect.
- 52 Author Unlisted
 "An Approximate Calculation of the Temperature Distribution Surrounding Coolant Holes in a Heat Generated Solid"
 BMI-T-42; 1950.

VIII JOINTS AND FASTENERS

- 1 Clantice, W. E.; and Stearns, R. W.
"Evaluation of Welded Joints Between AISI 347 Stainless Steel and 70-30 Copper-Nickel Alloy Tubes Under Thermal Shock"
Naval Engineering Experiment Station, Annapolis
EES-060088E; May 23, 1955.

- 2 Fritsch, G. S.
"Thread Investigation"
Allis-Chalmers Mfg. Co., Milwaukee
A-3344; July 25, 1944.
Results of a series of Max-torque tests of threaded bolts, studs and nuts are cited. Tabular and graphic data include bolting strengths and thread damage as determined by these procedures.

- 3 Gall, W. R.
"Selection of Flange Design for High Pressure Service in the HRT"
Oak Ridge National Laboratory
CF-54-6-74; June 9, 1954.
Results are given of an analysis made to establish a basis for selection of the best type of flanges for use in the HRT high pressure systems. On the basis of this analysis the use of a 1500-lb. ring joint weld neck design was recommended for use.

- 4 Haws, C. C., Jr.
"Tentative Recommendations for High Pressure Flange Assemblies"
Oak Ridge National Laboratory
CF-56-7-13; July 3, 1956.

- 5 Horvay, G.
"Analysis of Flanged Shell Joints and the Mechanism of Bolt Failure"
Knolls Atomic Power Laboratory
KAPL-M-GH-15; November 8, 1954.
Formulas are given for the stresses and deformations of a composite plate due to distributed weight loads, edge moments, and edge shear loads; for the stresses and deformations of a flanged shell due to membrane loads, edge moments, and edge shear loads; and for stresses and deformations of a bolt under combined tension, bending, and shear. The results are then combined to analyze the joint formed by a flanged shell-plate-bolt combination.

- 6 McHan, R. L.
"Investigation of Cracking of SIR Mark A Rotating Plug Shield Can Welds"
Knolls Atomic Power Laboratory
KAPL-M-RLM-3; July 2, 1954.
A metallurgical investigation was performed to determine the cause of cracking of rotating plug shield can welds. It was concluded that the cause was due to an undesirable microstructure resulting from a welding

procedure which produced a fully austenitic weld. To correct the situation, filler material should be added to produce a weld containing 5 - 10% ferrite. Future inspection of such welds should utilize the red dye penetrant test rather than visual examination.

- 7 Mehrlinger, F. J.
"Torquing of Bolts for Nuclear Reactors"
Knolls Atomic Power Laboratory
KAPL-1645; August 22, 1956.
As a result of bolting problems of the S1G and S2G reactors, torque tests were made on several different types of bolts. Measurements of the axial stress and effective uniaxial stress in the bolt were determined as a function of torque. The test results are presented graphically.
- 8 Segaser, C. L.
"Zircaloy-to-Stainless Steel Transition Pipe Joint Design Study"
Oak Ridge National Laboratory
CF-55-12-72; December 5, 1955.
A design survey collecting data and procedures to the Zircaloy-to-stainless steel transition pipe joints for 2-region homogeneous reactors was made. An analysis of gasket and bolting stresses consisted of measurements of gasket factors, elastic and creep properties of gasket material, and similar data on bolting stresses.
- 9 Smith, W. R.
"Strength Properties of Joint Designs for Joining Tubes to Tube Sheets"
Hanford Atomic Products Operation
HW-28897; July 31, 1953.
Joints prepared by welding and any combination of welding and rolling exhibited much greater strengths with less variation than those prepared by rolling alone. Rolling prior to welding had no effect on the strength of the joint, and any rolling performed after welding appeared to add little strength to the joint. A groove in the tube sheet did not appear to increase the strength of rolled joints by any significant amount. The joint design used for welding the tubes to the tube sheet is such that the welds can be made with reproducible results. Since the strength of a rolled joint is dependent on the yield point strength of the individual tubes, there is a wide variation in the strength of the rolled joints.
- 10 Taylor, D. K.; and Maloney, J. D.
"Causes of Leakage in HRE Flanges"
Oak Ridge National Laboratory
CF-53-5-125; May 1, 1953.
Tests were made to determine the causes of leakage of the flanged pipe joints used in the HRE, including evaluations of factors such as materials, design, technique of assembly, re-use, yielding

and creep, work hardening, temperature changes, and differential expansion. The test apparatus is described in detail. The cause of the joint failure seemed to be the differential expansion of the parts of the assembly during temperature changes. Design and installation recommendations are listed.

IX VESSEL RUPTURE

- 1 Aven, R. E.; and Zmola, P. C.
"Effects of a Possible Chemical Explosion in the Reflector Tank of the HRE"
Oak Ridge National Laboratory
CF-51-7-105; July 23, 1951.
Possible chemical explosions in the reflector tank of the HRE are discussed. Maximum hydrogen-oxygen explosion pressures and resulting stress in the reactor components are given for a range of reflector temperature. Results indicate that the reflector tank and cover would not fail in this type of explosion. The rate of diluent addition required to keep combustibles at non-explosive concentrations for lower reflector temperatures is also tabulated.
- 2 Aven, R. E.
"Pressure Rise Inside HRT Thermal Shield in Event of Pressure Vessel Failure"
Oak Ridge National Laboratory
CF-55-7-95; July 20, 1955.
- 3 Baker, Wilfred, E.; and Patterson, James D., II.
"Blast Effects of a One-Quarter Scale Model of the Air Force Nuclear Engineering Test Reactor"
Ballistic Research Laboratory, Aberdeen Proving Ground, Md.
BRL-1011; March 1957.
This report presents the results of a series of two simulated nuclear power excursions conducted within a one-quarter scale model of the Air Force Nuclear Engineering Test Reactor. The objective of the tests was to simulate, as closely as possible, large nuclear power excursions of the potential predicted by the "Borax" test data and to experimentally evaluate the effects of the blast on the reactor structure and safety shell. Explosives and propellants were used to simulate the power excursions. It is concluded that the reactor design will contain an explosively-released energy of approximately three times the maximum energy release in the maximum credible power excursion for the AF NETF reactor. The accuracy of simulation of an excursion by explosives and propellants and the applicability and limitations of scaling of blast loading and structural response are discussed.
- 4 Bolger, J. C.; Mack, R. O.; and Gift, E. H.
"Pressure Rise in HRT Shield Due to Reactor and Replacement Heat Exchanger Rupture"
Oak Ridge National Laboratory
CF-57-4-20; April 2, 1957.
Recalculations were made of HRT cell pressures in the event of simultaneous rupture of the reactor core vessel and the fuel and blanket heat exchangers. These calculations will be applicable

after the replacement fuel heat exchanger is installed. The pressure rise is plotted as a function of time. A maximum cell pressure of 34 psig is achieved 1 sec. after rupture.

- 5 Chapman, R. H.
 "Analysis of Spherical Pressure Vessel Having An Energy Source Within the Wall"
 Oak Ridge National Laboratory
 ORNL --1987; October 1954.

- 6 Frost, R. T.
 "Explosive Rupture of Pressurized Water System; Upper and Lower Bounds for Final Pressure, Temperature and Missile Energy"
 Knolls Atomic Power Laboratory
 KAPL-M-RTF-4; April 10, 1956.
 The irreversible expansion of the contents of a pressurized hot water system following an assumed failure of the pressure vessel is discussed. These calculations of final temperatures, pressures, and work are required in the safeguards analysis of the ANARPA pressure cell experiments and may be useful in connection with other similar proposed experiments.

- 7 Hines, Edward; Gemant, Andrew; and Lelley, J. K.; (Detroit Edison Co.)
 "How Strong Must Reactor Housings Be to Contain Na-Air Reactions?"
Nucleonics 14, No. 10; October 1956.
 Experiments and calculations point to 40 psig as the highest pressure to expect for closed-system Na-Air reactions, regardless of the amount of Na or size of the reactor housing. Maximum gas-phase temperature would be about 1000°C.

- 8 Kasten, Paul R.
 "Dynamics of the Homogeneous Reactor Test"
 Oak Ridge National Laboratory
 ORNL-2072; June 7, 1956.
 The safety and stability of the Homogeneous Reactor Test (HRT) have been investigated for a variety of operating conditions and blanket materials. Indications are that the outer pressure vessel will not fail even under the most adverse conditions anticipated and that normal operating procedures and safety restrictions are sufficient to protect the core tank against rupture, stability of the reactor system against reactivity changes, load demand changes, and "walkaway" appears assured in the present design. The HRT is to be fueled with an aqueous highly enriched UO_2SO_4 solution, and may have D_2O , a $\text{ThO}_2\text{-D}_2\text{O}$ slurry, or a natural uranium $\text{UO}_2\text{SO}_4\text{-D}_2\text{O}$ solution as the blanket material at different times. The safety of the reactor is dependent upon the core temperature coefficient of reactivity, which has a value of about $(-2 \times 10^{-3}) \Delta K_E / ^\circ\text{C}$.

-9

Levedahl, W. J.; and Howerton, R. D.

"A Method for Computing Pressure, Fragment Velocities, and Core Reactivity During Explosion of a Reactor Pressure Vessel"

Knolls Atomic Power Laboratory, Schenectady, N. Y.

KAPL-M-WJL-1; August 7, 1956.

A method for analytically determining pressures, radii and velocities of all regions of an exploding reactor is presented. The solution is obtained by numerical integration of differential equations relating pressure, volume, acceleration, and velocity of several concentric shells representing the core, reflector, pressure vessel, shield water, and shield tank. Complex reactors are transformed into equivalent spheres for analysis and results are transformed back to the true configuration. An explosion due to metal-water reaction causes missile velocities only fractionally higher than velocities from an explosion due to fatigue failure at ordinary operating pressures. Shield-fragment velocities may equal or exceed pressure vessel fragment velocities.

-10

McDuffie, H. F.; and Kelly, D. C.

"Homogeneous Reactor Project Quarterly Progress Report for Period Ending January 31, 1955"

Oak Ridge National Laboratory

ORNL-1853; February 16, 1955.

HRT - A summary of a hazards report is presented. Possible blast effects from a rupture of the pressure vessel were considered sufficient to justify the inclusion of a 1.5 to 2-in. thick blast shield around the pressure vessel. A maximum pressure of 29 psig is expected in the reactor equipment cell in the case of the instantaneous release of core and pressure vessel liquids, and release of heat exchanger liquids through 6-in. steam lines. Designs for the reactor pressure vessel and core tank were completed, and stresses in the carbon steel wall of the pressure vessel were calculated.

-11

Miller, E. C.

"Possibility of Brittle Pressure Vessel Failure"

Oak Ridge National Laboratory

CF-54-4-112; April 16, 1954.

In connection with HRT design, the history of brittle failures of welded engineering structures is reviewed. Available data on the impact strength of ASTM-212 steel are included.

-12

Porzel, F. B.

"Design Evaluation of BER (Boiling Experimental Reactor) in Regard to Internal Explosions"

Armour Research Foundation, Illinois Institute of Technology

ANL-5651, Project D090, for Argonne National Laboratory; January 1957.

An analysis of the safety aspects of the Argonne National Laboratory BER with regard to internal explosion has been made. The various methods by which protection from shock waves can be provided are

examined, and it is shown that a new type blast shield would be the most practical solution. The general background of the blast shield design is developed and the mechanism of absorption is discussed. The design rules are then applied to the BER and a blast shield of the proper materials and dimensions is developed.

- 13 Rosenthal, M. W.
"Hazard to HRT Containment Cell from Zirconium-Water Reaction or D₂ Explosion"
Oak Ridge National Laboratory
CF-56-8-156; August 24, 1956.
The energy release, explosion hazard, and cell pressures developed with and without an explosion have been examined for a number of situations involving rupture of the high-pressure system of HRT.
- 14 Svenson, N. L.
"The Bursting Pressure of Cylindrical and Spherical Vessels"
Paper presented at 1957 Conference of the American Society of Mechanical Engineers, No. 57-A-15; 1957.
- 15 Wood, Phillip M.
"A Study of Possible Blast Effects from HRT Pressure Vessel Rupture"
Oak Ridge National Laboratory
CF-54-12-100; December 14, 1954.
- 16 Wood, Phillip M.
"A Further Study of Pressure Rise in HRT Shield"
Oak Ridge National Laboratory
CF-55-1-3; January 3, 1956.
Assuming simultaneous failure of the HRT pressure vessel and one heat exchanger, an upper limit of the pressure rise in the equipment cell is calculated.
- 17 Zmola, P. C.; and Welton, T. A.
"Effects of Possible Chemical Explosion in the Core Tank of the HRE"
Oak Ridge National Laboratory
CF-51-7-3; July 3, 1951.
The effects of chemical explosions in the HRE core tank are estimated for an instantaneous detonation and for a radially expanding cylindrical shock front. The methods used are outlined briefly, and results are tabulated. It was concluded that an explosion in the gas core would not be of sufficient violence to rupture or deform the core tank.
- 18 Author Unlisted
"Army Package Power Reactor"
Description and Supplement, Alco Products, Inc., Schenectady, N.Y.,
AECD-3731; October 14, 1955.
The APPR-1 is described and the various hazards are reviewed.

Because of the reactor's location near the nation's Capitol, containment is of the utmost importance. The maximum energy release in any possible accident is 7.4 BTU's, which is completely contained within a 7/8-inch thick steel cylindrical shell with hemispherical ends. The vapor container is 60-feet high and 32 feet in diameter and is lined on the inside with 2 ft. of reinforced concrete which provides missile protection and is part of the secondary shield. All possible nuclear excursions are reviewed and the energy from any of these is insignificant compared to the stored energy in the water. The maximum credible accident is caused by the reactor running constantly at its maximum power of 10Mw and through an extremely unlikely sequence of failures, causing the temperature of the water in the primary and secondary systems to rise to saturation; whereupon a rupture occurs releasing the stored energy of 7.4 million BTU's into the vapor container. If the reactor core melts during the incident, a maximum of 10^8 curies of activity is released. While it appears impossible for a rupture of the vapor container to occur except by sabotage or bombing, the hazards to the surrounding area are discussed in the event of such a rupture occurring simultaneously with the maximum credible accident.

X REACTOR VESSEL DESIGN

- 1 Bowes, W.H.
"Design of a Head and Keyed Closure for a Nuclear Reactor Pressure Vessel"
Atomic Energy of Canada Ltd., Chalk River Project, Chalk River, Ontario
NPG-9(AECL), AECL-349 ; May 30, 1956.
A design study was carried out to determine the feasibility of engineering a welded, penetrated vessel head, and of securing the head by means of helical keys. The fuel rod pitch, core diameter, and internal pressure used in the design were proposed for NPD at the time this study was started. These values have since been altered: diameter and spacing have been increased, pressure reduced. The welded head and keyed closure are feasible for the given design conditions. It is probable that a similar vessel could be designed to meet the present requirements of NPD.
- 2 Brown, Wayne S.; and Kistler, S.S.
"The Design of Pressure Vessels for Extreme Pressure"
Utah University, Salt Lake City, College of Engineering
NP-6000; June 1, 1956.
Elastic theory was found to be useful in gaining insight into stresses in vessels designed for extreme pressure. High pressures must be retained by tensile members, the design of which can be accomplished through elastic theory. Weak materials are useful in high pressure design since there is no indication that they distribute stresses hydrostatically even when fragmented. Powders are found to support loads in a columnal fashion.
- 3 Bush, P.D.
"ETR⁴ Design and Construction"
Nucleonics; March 1957.
The story of testing and selecting metals for vessel and piping and cement for shielding and of solving some unusual construction problems is told.
- 4 Cooper, W.E.
"Design Stress Analysis, SIR (Mark B) Container Assembly"
Knolls Atomic Power Laboratory, Schenectady, New York
KAPL-M-WEC-2; December 31, 1953.
The work which has been done on the design stress analysis of the major components of the SIR (Mark B) container assembly is reviewed.
- 5 Cooper, W.E.
"Structural Design Basis, SAR Reactor Components"
Knolls Atomic Power Laboratory, Schenectady, New York
KAPL-M-WEC-7; June 20, 1955; Revised November 15, 1955.
A structural design basis to provide limitations upon stresses and deformations and to outline the general design requirements for the SAR is given.

- 6 Davis, Frank W.
"Feasibility Study of Pressure Vessels for Nuclear Power
Generating Reactors"
Division of Reactor Development, AEC
AECU-3062; December 1955.
- 7 Gall, W.R.; and Mallon, D.J.
"Mock-Up Design Report"
Oak Ridge National Laboratory, Tennessee
ORNL 701; October 27, 1949.
Design work done on the mock-up of the Materials Testing Reactor (MTR) is discussed in detail. Each detail is explained as to the basis for design and reasons for particular solutions that were selected. The outline starts with the tanks and then takes each item from the top plug down through the grid supports to the bottom of the tank. Calculations are not given in detail, but results are given and general methods used are explained. Alternate designs are briefly discussed in some cases so that it will be known what others were considered and what the reasons were for their elimination. The reflector and the control system have been covered only briefly without going into the same detail as for the mechanical design of the tank, grids, etc. Conceptual design of the reactor for the most part was complete before design of the mock-up started, so that this was primarily a detailing job. Where conception was required it related to mechanical details and not to the over-all picture of the arrangement of the reactor.
- 8 Gasche, Fred
"Design of Reactors and Closures"
Ind. Eng. Chem., 48, 838-40; May 1956.
A study was made of design and fabrication of equipment for service at elevated pressures and temperatures.
- 9 Jasper, T. McLean
"Multilayer Vessels for High Pressure and Temperature"
Chem. Eng. Progr., 52, 521-27; December 1956.
Problems in designing, fabricating, testing, and inspecting pressure vessels are reviewed. The simplification of such problems by the use of multilayer vessels is discussed.
- 10 Kerkhof, W.P., (The Hague, Netherlands)
"Stresses for Pressure Vessels and Boilers up to 650°"
Welding Journal, 33, 239s-51s; May 1954.
The problem of the stresses occurring in boilers and pressure vessels has been studied with the aid of both the theory of plasticity and theory of elasticity, taking into account the residual stresses in the weld and in the non-welded plate, and the stress-relieving effect of the hydrostatic test. The calculated maximum total stress occurring is nearly always equal to the yield point of the material, and independent of the allowable membrane stress and weld efficiency used for the

-11 calculation. As such a high stress always occurs, it cannot have any influence on the rupture of the vessel. Rupture can only be due to cracks or other faults in manufacture and/or design. Such faults result in stress concentrations far above the yield point. In this case the state of stress is triaxial. It is known that the material only fails as a result of those high stress concentrations. As the total stress occurring is not dependent on the membrane stress used for calculation, the latter should be as high as possible. An allowable membrane stress of 66% of the yield point is recommended. The permissible superimposed bending stress can also be 66% of the yield point. In case pure bending stresses exist, the allowable stress can be equal to the yield point. The test pressure should load the vessel to a stress 1.3 times the proposed allowable stress and test pressure limit the deformation of the material in such a way that no rupture can occur due to excessive strain. The weld efficiency should always be 100%. The use of the factor of safety is not based on sound reasoning. Stress relieving is not necessary with a view to strength and should only be carried out for improvement of the mechanical properties of the material and perhaps in the case of very thick shells. Radiographing should be carried out when a possible rupture would cause an explosion and in all other cases where a rupture is dangerous. In some cases, if the weldability of the material is low, radiographic examination is also required.

-12 Kooistra, L. F., (The Babcock & Wilcox Co., Alliance, Ohio)
"Effect of Plastic Fatigue on Pressure-Vessel Materials and Design"
Welding Journal 36, 120s-30s; March 1957.

Compiled data have been correlated to establish the parameters governing pressure-vessel design with respect to the plastic fatigue characteristics of the material. General conclusions for the practical application of the findings toward setting up safe design limits, as well as recommendations for further research, are presented.

-13 Langer, B. F.
"Design Basis for Reactor Vessels"
Westinghouse Electric Corp., Atomic Power Div., Pittsburgh
WAPD-CE-43; October 18, 1955.

The design basis to be used for reactor pressure vessels for naval service is specified. Type of loading and operating conditions are given. Methods of formulating stress and minimum stress values to be met are listed.

-14 Langer, B. F.
"Revisions to Design Basis for Reactor Vessels"
Westinghouse Electric Corp., Bettis Plant, Pittsburgh
WAPD-CTA(CE)-52; Revisions to WAPD-CE-43; March 13, 1956.

- 15 Maker, F. L.
"Design of a Large Vacuum Vessel"
Livermore Research Laboratory, California Research & Development Co.
LRL-157; June 1954.
A horizontal vacuum vessel 60-ft. in diameter by 87-ft. long was designed and built. Many novel design features are described. These included use of both longitudinal and circumferential stiffeners and spherical segment heads turned inward, with compression ring reinforcement at the edges. The design is described, and the method of determining the distribution of the load between shell, longitudinal stiffeners, and circular ribs is developed, as well as the computation of the discontinuity stresses at the edge of the head.
- 16 Major, C. J.
"Charts for Pressure Vessel Design"
Chemical Engineering 62, No. 4, 172-5; April 1955.
- 17 Mong, B. A.; and Douglass, R. M.; (Babcock and Wilcox Co., Akron, Ohio)
"Engineering and Construction of Nuclear Power Plants. Reactor Vessels"
Nucleonics 13, No. 6, 66-9; June 1955.
The problems involved in the design and fabrication of vessels to contain reactor cores are discussed. In addition to the usual consideration of pressure, temperature, and corrosion resistance, the following problems must be studied: radiation damage to the vessel materials; thermal stresses due to radiation heating; exceptional reliability since the vessel cannot be examined periodically; leaktightness; and closer dimensional tolerances. Vessels for pressurized-water reactors, aqueous homogeneous reactors, and liquid-metal reactors are described.
- 18 Schneider, R. W., (Travelers Insurance Co., Hartford, Conn.)
"Help in Designing Pressure Vessels"
Chemical Engineering 63, No. 2, 171-3; February 1956.
A rapid graphical method of estimating maximum stresses accurate enough to justify the use of minimum safety factors of pressure vessels with flat heads is discussed.
- 19 Segaser, C. L.
"Pressure Shell and Closure Design for the Homogeneous Reactor"
Oak Ridge National Laboratory, Tennessee
CF-49-11-152; November 15, 1949.
The proposed design for a pressure shell based on a spherical internal shell of 19-1/2 in. radius is analyzed for stresses, closures, and flange stress and leak detection.
- 20 Segaser, C. L.
"Thermal Shield and Pressure Shell Design for 15-Foot Homogeneous Reactor Tank at 50KW per Liter Specific Power"
Oak Ridge National Laboratory, Tennessee
CF-51-6-42; June 12, 1951.

The principal loading producing stress in the pressure shell and thermal shielding of large homogeneous spherical reactor tanks are due to the internal pressure of the fluid contained and the temperature differentials induced in the shell walls by absorption of leakage radiation from the reactor core. The tangential thermal stress induced by these temperature differentials becomes so large at high reactor power levels that operation may be limited to some lower power unless means are provided to reduce these stresses to tolerable values. A 15-ft. diameter single shell homogeneous spherical reactor is apparently limited to a power level not exceeding 2.5KW per liter if the total stress in the pressure shell is not allowed to exceed 15,000 psi, as recommended by the 1950 ASME Boiler Code for Unfired Pressure Vessels. The optimum thickness of shell required for such a single shell reactor to contain fluid at 1,000 psi pressure is 6" at 2.5KW per liter. Two methods are known for reducing the magnitude of thermal stress. The thickness of the shell may be decreased and/or the shell may be cooled simultaneously on both sides to the same temperature differential in the wall, and, hence, a lesser thermal stress. It is shown that a reduction of shell thickness decreases the thermal stress, but the total stress from the combined loading still greatly exceeds the assumed maximum allowable stress of 15,000 psi. To be economically feasible, Lane and Briggs have shown that a homogeneous reactor must operate in the order of magnitude of 1 million KW. At this total power, a 15-ft. diameter reactor must operate at a power density of 20KW per liter. At 20KW per liter, the optimum pressure shell thickness is approximately 2", but the total minimum tangential shell stress exceeds 40,000 psi. In order to operate at a power density of 50KW per liter, the amount of leakage radiation from the reactor core absorbed by the pressure shell must be reduced to where the total combined stress from pressure and temperature will not exceed the upper allowable design stress for the pressure shell material. It is shown in this report that a total of 6" of iron thermal shielding, is required. With 6" of shielding material and cooling space, the inner radius of the necessary pressure shell is 100", which to contain the fluid at 1000 psi pressure, requires a 4-1/2" thick steel wall. The maximum stress in the welded joints will, therefore, be not greater than 13,750 psi, which will meet the ASME Boiler Code requirement for a welded vessel at joint efficiencies of 90%.

-21

Segaser, C. L.

"HRT Evaporator Design Study"

Oak Ridge National Laboratory, Tennessee

CF-54-5-2; May 3, 1954.

The design of a standard calandria-type of shell and tube evaporator is discussed, and the specifications are included for use in the HRT.

-22

Stewart, D. D.

"A Description of Chalk River Experience with Heavy Water Reactor Vessels"

Atomic Energy of Canada Ltd., Chalk River Project, Chalk River, Ontario
CRIO-684, AECL-402; January 1957.

Chalk River experience with NRX "calandria" reactor vessels is summarized. The original calandria is described and compared with the second one as to materials, fabrication, testing and modifications. Proposed changes for the spare calandria are discussed.

-23

Van Winkle, R.

"Design Criteria for HRT Sample Line Coolers"

Oak Ridge National Laboratory, Tennessee

CF-55-1-29; January 5, 1955.

-24

Vasta, John; and Dunham, Frank W.; (U. S. Navy Department, Bureau of Ships, Washington, D. C.)

"30-Foot Diameter All Welded Test Tank"

Welding Journal 33, 422s-32s; August 1954.

Some unique problems encountered in the design and construction of a 30-ft. diameter all-welded steel test tank are described. The theoretical solution to some of these problems are given and compared with the experimental data taken during the proof testing of the tank.

-25

Zick, Leonard P., (Chicago Bridge and Iron Company)

"Design of Welded Pressure Vessels Using Quenched and Tempered Steel"

Welding Journal 34, 442s-8s; September 1955.

Nine welded pressure vessels made of T-1 steel, a quenched and tempered material, were tested to destruction. These tests dramatically indicated the strength and toughness of this material under both static and impact loads at low temperature and at stresses considerably above the ultimate strength of most materials now listed in the Codes. These tests raise new and shed light on many old controversial issues pertaining to the design of pressure vessels. Although one particular steel was used in the tests, many of the issues discussed apply more broadly to notch tough steels having high yield and ultimate strengths and high yield to ultimate strength ratios. Economic considerations suggest the utilization of the greater elastic strength by permitting allowable design stresses greater than one-fourth of the ultimate strength. The use of material having a minimum yield strength of 90,000 psi suggests design stresses greater than the yield of some carbon steels. Refinements of design details such as connections, junctions and attachments and increased inspection rule requirements may well be justified.

-26

Author Unlisted

"Classified Progress Report of the Reactor Science and Engineering
Department for 1949"

Brookhaven National Laboratory, Upton, New York

BNL-48; April 1950.

Efforts during the report period have been divided between two fields: research, and evaluation and redesign of the reactor. Greater effort has been directed toward re-evaluation of the structural design of the reactor, tests on various components, and engineering design aimed at correcting such structural defects as have been uncovered. Along with these have been included studies for the start-up procedure and actual testing and readying of the reactor auxiliary equipment and instrumentation. Progress has been made on certain aspects of the research program as well.

-27

Author Unlisted

"Charts to Pressure Vessel Design"

Product Engineering; October 14, 1957.

XI TESTING

- 1 Bunk, A. P., (Chicago Bridge and Iron Co.)
"Welding Procedure Qualification Tests for Six High-Yield-Strength Steels"
Welding Journal, (N.Y.) 34, 197s-206s; April 1955.
Welding procedure qualification tests, made in accordance with Section IX of the ASME Code for Unfired Pressure Vessels, were conducted on three pairs of steels with minimum yield strength of 50,000, 70,000 and 90,000 psi, respectively. The purpose was to establish suitable welding procedures which would meet the requirements of the Code. The more important results of this study are summarized.
- 2 Gross, J. H.; and Stout, R. D., (Lehigh University, Bethlehem, Pa.)
"Properties and Weldability of High-Strength Pressure Vessel Steels in Heavy Sections"
Welding Journal, (N.Y.) 36, 157s-67s; March 1957.
The effects of plate thickness and of position in the plate on the mechanical properties and weldability of high-strength pressure-vessel steels in heavy sections are studied.
- 3 Halpine, P. A.
"Outline of PWR Reactor Vessel Testing Program"
Westinghouse Electric Corp., Atomic Power Div., Pittsburgh, Pa.
WAPD-RD-22; April 2, 1955.
- 4 Palladino, N. J.
"PWR Pressure Vessel Test Program"
Westinghouse Electric Corp., Atomic Power Div., Pittsburgh, Pa.
WAPD-RD-40; May 16, 1955.
- 5 Robinson, H. N.
"Test of HRT Cell"
Oak Ridge National Laboratory, Tennessee
CF-55-11-62; November 9, 1955.
The HRT cell was pressure tested and proven to be capable of containing a static gas pressure of 30 psig. The test is described and typical data are presented.
- 6 Speeds, J. A.
"Temperature and Pressure Cycling of a Full Diameter Test Vessel"
Babcock and Wilcox Co., Research Center, Alliance, Ohio
BW-5421; June 11, 1953.
A full diameter test vessel was used to carry out a series of temperature and pressure cycles. Only minor surface defects were discovered. No propagation of a known defect in the weld occurred. The closure studs did not show any trend of relaxing due to fatigue test. Results from these tests justify full confidence in the vessel design.

THEORETICAL ASPECTS OF REACTOR DESIGN

XII ELASTICITY AND PLASTICITY

- 1 Bridgman, P. W.
"Plastic Properties of Steel. The Collapse of Hollow Cylinders Under External Pressure"
Harvard University, Cambridge, Massachusetts
WAL-111/7-3; October 8, 1943.
Problems of plastic behavior of metals at large deformations are investigated by a special method of collapsing hollow cylinders. Mathematical analysis and complete tabulations are presented.
- 2 Cooper, R. M.; and Naghdi, P. M.
"Propagation of Non-Axially Symmetric Waves in Elastic Cylindrical Shells"
Engineering Research Institute, University of Michigan, Ann Arbor
AD-135757; April 1957.
This paper is concerned with the predictions of phase velocities and amplitude ratios, for non-axially symmetric wave propagation in an elastic cylindrical shell, by employing two systems of equations of motion; these two systems of equations were previously employed in a similar study but limited to torsionless axisymmetric wave propagation. Comparison is again made between the predictions of the two systems of equations including the case of pure torsional motion.
- 3 Curtis, C. W.
"Propagation of Elastic and Plastic Deformations in Solids"
Lehigh University, Institute of Research, Bethlehem, Penna.
NP-5776.
The propagation of strain pulses along solid bars was investigated. The pulse was produced by applying a time-dependent compressive force to one end of a freely suspended bar or system of bars. With the condition of loading specified, the resulting strain as a function of position and time was determined.
- 4 Friedrich, C. M.
"Energy Capacity of Elastic-Plastic Beams"
Westinghouse Electric Corp., Atomic Power Division, Pittsburgh
WAPD-TN-511; June 1955.
A procedure is presented for calculating strains and deflections in elastic-plastic steel beams in terms of energy capacity or starting velocity. Graphs based on the calculations for quick determination of peak strain, maximum permanent strain, peak deflection, maximum permanent deflection in steel beams of various cross sections, and loading conditions are given.
- 5 Gubkin, S. I.; and Bogdanov, Ye. S.
"The Path of the Deformation Process as a Basic Characteristic of the Deformed State of a Plastic Body"
Translated from Doklady Akad. Nauk S.S.S.R., 1953
NSF-tr-125.

The physical and geometrical deformations which are necessary to give a complete description of the deformed state are discussed. It is concluded that the only deformation variable proportional to the expended work and to the change in the physical state of the body is the physical intensity of deformation. Therefore, the dependence of the intensity of the state of stress on the process path is the sole characteristic of the plastic behavior of a metal. In the general case this dependence is represented by a set of curves, one coordinate being the intensity of the state of stress and the other the deformation path. For stable and very plastic metals, when the state of stress does not affect the resistance to deformation, the family of curves merges into one general curve, which is a universal characteristic of the material.

- 6 Gubkin, S. I.; and Downar, S. A.
 "Determination of the Coefficient of Diffusion in Plastic Deformation"
 Translated from Doklady Akad. Nauk S.S.S.R. 93, 1953
 AEC-tr-2249.
 A description of a new method of determining the coefficients of diffusion and self-diffusion in the plastic deformation of a metallic body is given. The method is distinct in that it determines the coefficients of diffusion and self-diffusion directly in the process of plastic deformation. A table giving the sequence of operations in the experiment showing the nature of the method is attached.

- 7 Horvay, G.; and Born, J. S.
 "Precise Closed-Form Solutions of Some Mixed Boundary Value Problems of Plane Elasticity"
 Knolls Atomic Power Laboratory, Schenectady, New York
 AECU-3107; 1954.
 Precise solutions are given for the problem of the semi-infinite strip extending from $x=0$ to $x=\infty$, stress-free along the edges $y=\pm 1$ when $x=0$ is subject (a) to a quadratic shear displacement and zero normal stress, (b) to a cubic normal displacement and zero shear stress. These are the two simplest (non-trivial) end problems of the semi-infinite strip, antisymmetric in y , that may be formulated. The corresponding symmetric problems, (a) linear shear displacement and zero normal stress, (b) quadratic normal displacement and zero shear stress, were solved in earlier papers.

- 8 Riparbelli, Carlo
 "On the Growth of Plastic Deformation in a Bar Submitted to Longitudinal Impact"
 Cornell University, Graduate School of Aeronautical Engineering
 NP-5369; January 1953.
 A scheme to interpret and compute the growth of plastic deformation in a semi-infinite bar undergoing longitudinal impact is outlined.

The deformation of a bar element is interpreted as, the resultant of two deformation components: (a) elastic deformation, propagating with the speed of elastic waves of longitudinal deformation, and (b) a deformation subsequent to the establishment of the stress, called "creep", the result which is the plastic deformation. This last is understood to be a local phenomenon, not propagating. The deformation of a semi-infinite bar submitted to end impact is computed on the basis of such a scheme. The computation is done step-by-step, each step corresponding to one element of a bar and to one time interval. At every step, two phases are considered: (a) propagation of the deformation wave as if it were elastic and (b) increment of deformation due to creep. A discussion is given of the static stress-strain relation in light of the observed phenomena.

-9

Steele, M. C.; and Eichberger, L. C. .

"Investigation of the Behavior of Overstrained Thick-Walled Cylinders. Technical Report No. 2. The Overstraining of Thick-Walled Mild Steel Cylinders"

University of Illinois, Urbana

TAM-79; February 1955.

Experimental results are presented on the behavior of three mild steel cylinders (6 in. O.D. x 3 in. I.D. x 8 in. long) overstrained by internal pressure. Particular attention is made to the mechanism of yielding and the variation and character of strains measured at the bore and outside surfaces. Yield initiates and propagates into the cylinder walls by the formation of a limited number of Lueders' lines. Initial yield is caused by a maximum shear stress higher than that necessary to propagate yield. Lueders' line patterns (recorded on end face) are defined by two groupings situated approximately at opposite ends of a diameter. They follow closely the maximum shear stress trajectories. The non-homogeneous mechanism of yielding is clearly shown up by variations in strain circumferentially. Fully yielded pressures are some 10 to 15 per cent lower than those predicted by theories when yield stresses from tension tests conducted at normal rates of loading are used. Axial strains, to a first approximation, are elastic, which is in agreement with a theory due to Koiter but in marked disagreement with Thomas' theory. Small differences between experimental and theoretical shapes of pressure-circumferential strain curves are attributed to the non-homogeneous mechanism of yielding. Creep or slow yield is a feature of the tests. Lueders' lines require time, 15 to 30 minutes, to reach an equilibrium state at a specific pressure stage. Corresponding creep is shown up in the circumferential strains.

-10

Steele, L. C.; and Eichberger, L. C.

"Investigation of the Behavior of Overstrained Thick-Walled Cylinders. Technical Report No. 3. The Overstraining of Medium-Carbon Steel

Thick-Walled Cylinders"

University of Illinois, Urbana

TAM-83; April 1955.

Experimental results are presented on the behavior of four medium-carbon steel cylinders (6 in. O.D. x 3 in. I.D. x 8 in. long) overstrained by internal pressure. Particular attention is given to the mechanism of yielding and the associated strains measured at various locations on the bore and outside surfaces. The yield mechanism consists of the propagation through the cylinder walls of a finite number of Lueders' lines. Initiation of yield requires a maximum shear stress higher than that found from the lower yield in a tension test. Lueders' line patterns are symmetrical but follow closely the maximum shear stress trajectories. Creep of slow yielding is a feature of the mechanism. Full yield in the cylinders is accompanied by an increase in the number and width of Lueders' lines and corresponding increases in circumferential strains. The nonhomogeneous mechanism of yielding is reflected by variations in strain circumferentially. Average bore and outside diameter circumferential strains compare favorably with theories due to Thomas and Koiter. Axial strains are essentially elastic in keeping with Koiter but in marked disagreement with Thomas. Circumferential strains exhibit significant creep. A comparison is given between the Lueders' line patterns in mild steel and medium-carbon steel cylinders.

(For preceding report in series see TAM-79).

-11

Zhukov, A. M.

"Poisson's Coefficient in the Plastic Range"

Translated from Izvest. Akad. Nauk S.S.S.R. Otdel Tekh. Nauk, No. 12
AEC-tr-2277; 1954.

The change which occurs in Poisson's coefficient (the ratio of the fractional transverse contraction to the fractional longitudinal extension of a body under tensile stress) with the increase of plastic deformation is discussed. Experimental results are reviewed and found to be very inconsistent. Using mechanical tensometers, an attempt was made to determine the change in the Poisson coefficient in relation to a longitudinal deformation directly from the experiment for a simple elongation. Tests were conducted on solid Al and steel cylindrical samples -- 30mm in diameter. Results show that the Poisson coefficient increases sharply when plastic deformations appear up to an elongation of --1%. After this a smooth and very small increase occurs in the coefficient.

-12

Author Unlisted

"Resistance of Metals to Plastic Deformation"

NSF-tr-176; 1953.

- 1 Eagle, H. A.
"Design Charts for Analysis of Circular Plates With Central Holes"
General Electric Co., Aircraft Nuclear Propulsion Dept., Cincinnati
APEX-315; April 15, 1957.
The results of the stress analysis of circular plates, with central hole, supported at both inner and outer edges and loaded by a uniform pressure are described. The four cases of ideal edge fixity investigated are: both edges simply supported, clamped inner edge and simply supported outer edge, simply supported inner edge and clamped outer edge, and both edges clamped. The solution for various ratios of hole to plate diameter have been obtained by the method of superposition using IBM digital computers and the results are presented in the form of non-dimensional design charts. The results of the analysis are limited by the usual restrictions of the thin plate, small deflection theory.
- 2 Hetenyi, M.; and Timms, R. J.
"Analysis of Annular Shells With Applications to Welded Bellows"
Knolls Atomic Power Laboratory
KAPL-1089; 1954.
A method is presented for the calculation of stresses and deflections in ring-shaped shells of circular cross-section, subjected to axially symmetric loadings. The solution is derived without the restriction imposed for toroidal shells by previous investigators, that the radius of curvature of the cross section is to be small compared with the mean radius of the torus. The range of applicability of the method is extended to include the slightly arched convolutions used in the construction of welded bellows. By a rational reduction of the general solution approximate design formulas are obtained for the maximum stresses and deflections in bellows under axial forces and pressure loadings, and the calculated values are compared with experimental data.
- 3 Hoff, N. J., (Polytechnic Institute of Brooklyn, N. Y.)
"Boundary-Value Problems of Thin-Walled Circular Cylinder"
Journal of Applied Mechanics 21; December 1954.
The homogeneous differential equations of Donnell's theory of thin cylindrical shells are integrated. Expressions are obtained in closed form for the displacements, membrane stresses, moments, and shear forces.
- 4 Kelber, Charles N.; and Spinrad, Bernard I., (Argonne National Laboratory, Lemont, Illinois)
"Two-Group Iteration Method for Annular Cylinders"
Nuclear Science and Engineering 2, 219-20; April 1957.

A brief description is given of an iterative technique which was developed to study the critical properties of a homogeneous annular region reflected externally and internally.

-5 La Coas, W. D.

"An Investigation of Metal Shells as Cushioning Devices for the Component Recovery Program"

Sandia Corporation, Albuquerque, New Mexico

AECU-3441; March 15, 1957.

A preliminary investigation of the energy absorption and failing characteristics of metal shells that were subjected to a compressive load has been completed. Results of the tests indicate some advantages of metal shells as cushioning devices as compared to aluminum honeycomb materials.

-6 Stanek, Floyd J.

"Spherical Segment with Circular Hole at Vertex Loaded Axisymmetrically Along the Edges"

Oak Ridge National Laboratory, Tennessee

ORNL-2207; January 3, 1957.

The results of a stress analysis of a spherical segment with circular hole at the vertex which is loaded symmetrically with respect to its axis along the edges of the shell are given. Results were obtained and are given for two cases. General loading, i.e., normal force, radial shearing force, and bending moment applied to each of the two edges of the shell and General Loading at inner edge with outer edge rigidly fixed. The classical, small-deflection theory of thin shells was used. This method involves the evaluation of four hypergeometric series and their derivatives which give the solution to a fourth order differential equation for the shearing force, Q_ϕ . The above problem is then extended by attaching an axially loaded circular cylinder to the inner edge of the shell. The manner in which this was done is shown. Several cases were analyzed, and the results are summarized. The cases differ primarily in their relative geometrical dimensions and outer edge conditions. These differences are summarized. The relative dimensions for Case A1 were selected, so as to coincide, as nearly as it was possible to do so, with a physical model from which the principal stresses were obtained experimentally. A comparison is then made between the theoretical and the experimental results. In general, the two results are in good agreement.

-7 Author Unlisted

"Uniform Notation System for Pressure-Vessel Shell Theory"

Welding Journal, (N.Y.) 34, August 1955.

A final revision incorporating the comments and suggestions of leading authorities in the fields of shell design and structural mechanics is presented. The intent of this paper is to simplify the reading of shell theory literature and to facilitate communication among investigators in these fields. Definitions of terms and standard symbols

are given for shells of revolution in general and also for the right circular cylinder. It is felt that necessary modifications of this notation for special shapes and types of loading can best be given in a specific context.

XIV BUCKLING

- 1 Benthem, J. P.
"On the Buckling of Bars and Plates in the Plastic Range. Part II"
Translated from Nationaal Luchtvaartlaboratorium, Amsterdam
NACA-TM-1392; January 1954.
A review is made of existing literature concerning comparison with experiment of various theoretical formulas for buckling of plates in the plastic region. The significance and relative merits of various theories are discussed.
- 2 Bijlaard, P. P.
"Investigation on Plastic Buckling for Cases of Simply Supported Plates, Subjected to Non-Homogeneous Stress Distribution"
Cornell University
NP-5609.
The case of plastic buckling of long clamped plates under eccentric compression in their plane was solved earlier by writing the differential equation in terms of finite difference equations with second order differences, using nine spacings. In this note a brief discussion given of the plastic reduction factor η , the ratio between the plastic and buckling stresses for a given stress distribution.
- 3 Gast, Paul F.; and Bournia, Anthony
"Buckling of an Elliptic Cylinder; First Roots of the Zero Order, Modified Mathieu Function"
Hanford Atomic Products Operation, Richland, Washington
HW-38698; July 21, 1955.
- 4 Gast, Paul F.; and Bournia, Anthony, (Hanford Atomic Products Operation, Richland, Washington)
"Finding the Buckling of an Elliptic Cylinder"
Nucleonics 14, No. 4; April 1956.
Equations are given for reactors in the form of a right elliptic cylinder. These are particularly applicable to a propulsion reactor which might be mounted in a space where one horizontal dimension must be limited.
- 5 Kaplan, A.; and Fung, Y. C.
"A Nonlinear Theory of Bending and Buckling of Thin Elastic Shallow Spherical Shells"
California Institute of Technology
NACA-TN-3212; August 1954.
The problem of the finite displacement and buckling of a shallow spherical dome is investigated both theoretically and experimentally. In the theoretical approach the nonlinear equations are converted into a sequence of linear by expanding all of the variables in powers of the center deflection and then equating the coefficients of equal powers. Experimental results seem to indicate that the classical criterion of buckling is applicable to very shallow

spherical domes for which the theoretical calculation was made. A transition to energy criterion for higher domes is also indicated.

-6

Reiss, Edward L.

"Buckling of Shallow Spherical Shells Under External Pressure"

New York University (AEC Computing Facility)

NYO-7967; April 1, 1957.

A formula for the initial buckling loads for clamped, shallow spherical shells under uniform external pressure is obtained by combining the solutions of two linearized versions is a linear eigenvalue problem, while the other is the bending problem for a shallow cap in the linear theory of elasticity. The formula which is obtained in a simple manner, yields buckling loads that are in better agreement with experiments than previous approximate solutions to the non-linear problem.

-7

Yoshimura, Y.

"On the Mechanism of Buckling of a Circular Cylindrical Shell Under Axial Compression"

Translated by Y. Yoshimura and edited by Y. C. Fung and E. E. Sechler.

NACA-TM-1390; 1951.

The present paper deals with the buckling of a circular cylindrical shell under axial compression from the viewpoint of energy and the characteristics of deformation. It is shown first, both theoretically and experimentally, that the reason why the buckling of a cylindrical shell is quite different from that of a flat plate is attributable to the existence of a nearby developable surface far apart from the original cylindrical surface, which is equivalent to the existence of an approximately inextensional finite deformation. Based upon this result, the experimental fact that the buckling is really not general but local, that is, that the buckled region is limited axially to a range of 1.5 times the wave length of the lobe, is explained by the theoretical result that the minimum buckling load is smaller in the local buckling than in the general buckling case. The occurrence of local buckling is affirmed also from the viewpoint of the energy barrier to be jumped over during buckling, and from a comparison of the theoretical post-buckling state with the experimental results. Finally, the local buckling with the load applied by a spring is analyzed, and it is proved that the minimum buckling load increases with an increase of rigidity of the spring.

OTHER REFERENCES

XV BIBLIOGRAPHIES

- 1 Andersen, J. R.; and Nestler, D. E.
"Shock Wave Propagation in Solids: A Survey of the Literature"
University of Pennsylvania
AD-39616; 1952.
- 2 Perks, William, I.; and Chwick, Alexander
"Bibliography on Thermal Elasticity"
Brooklyn Polytechnic Institute
AD-36788; January 1953.
- 3 Delano, Warren E.
"Investigation of Strength and Buckling Characteristics of Transverse Bulkheads in Cylindrical Shells, A Bibliography"
Massachusetts Institute of Technology
AD-40826; March 15, 1954.
- 4 Householder, Alston.
"Bibliography on Numerical Analysis"
Oak Ridge National Laboratory
OCL-4540; July 5, 1955.
- 5 Majors, Harry Jr.
"Thermal Shock and Fatigue: A Literature Survey"
Alabama University, Tuscaloosa, Bureau of Engineering Research
NP-6168; June 1956.
- 6 Nash, William A.
"Bibliography on Shells and Shell-Like Structures"
David W. Taylor Model Basin
DTIC-863; November 1954.
Approximately 1450 papers and books pertinent to experimental and theoretical work on shells and shell-like structures are listed in this compilation. The entries are listed in chronological order, and both author and subject indexes of the papers are included. The listing is believed to be current through December, 1953.
- 7 Nash, William A.
"Bibliography on Shells and Shell-Like Structures (1954-1956)"
Florida University, Gainesville, Engineering and Industrial Experiment Station
NP-6362; June 1957.
Books and papers published from 1953 through 1956 pertinent to experimental and theoretical work on shells and shell-like structures are listed. The entries are listed alphabetically by authors, and a subject index is included.

- 8 Postell, Paul E.; and Voress, Hugh E.
"Unclassified Bibliographies of Interest to the Atomic Energy Program"
Technical Information Service, AEC
TID-3043; October 2, 1953.
This compilation contains annotated references to bibliographies prepared by the AEC project and to some from commercial, scientific, and other government organizations. The bibliographies are arranged by subject, and multiple listings are made for those which cover more than one subject. A table of contents lists the subjects covered. Author indexes are included.
- 9 Scott, Thomas W.
"Unclassified U. S. Atomic Energy Commission Reports on Civilian Power Reactors, and Test Reactors"
Technical Information Service Extension, AEC
TID-3501; November 1956.
- 10 Voress, Hugh E.
"Unclassified Bibliographies of Interest to the Atomic Energy Commission" Supplement 2
Technical Information Service, AEC
TID-3043; September 1955
This compilation contains annotated references to bibliographies considered pertinent to the AEC program. Reports available at the Technical Information Service between July 20, 1954 and August 25, 1955, were reviewed. Selected published literature was also included. The bibliographies are arranged by subject and multiple listings are made for those which cover more than one subject. A table of contents lists the subjects covered and author and report number indexes are included.
- 11 Voress, Hugh E.
"Unclassified U. S. Atomic Energy Commission Reports on Reactor Safety: A Literature Search"
Technical Information Service Extension, AEC
TID-3503; November 1956.
- 12 Author Unlisted
"A Bibliography of Selected AEC Reports of Interest to Industry. Part 5, Mechanics and Mechanical Engineering"
Technical Information Service, AEC
TID-3050(Pt.5); January 1955.
A bibliography of 139 selected references to non-classified reports on developments in the field of mechanics and mechanical engineering compiled by the AEC Industrial Information Branch is presented as an aid to industry in its interest in technological developments in the atomic energy program. A subject index provides a detailed guide to the specific material covered by each document listed in the bibliography.

- 13 Author Unlisted
"Nuclear Engineering, A Selected List of References"
Technical Information Service, AEC
TID-3055; April, 1954
This bibliography contains 480 annotated references to AEC reports and to the open literature. A list of pertinent bibliographies, an author index, and a report number index with availability information is also included.
- 14 Author Unlisted
"Nuclear Science: A Bibliography of Selected Unclassified Literature"
Technical Information Service, AEC
TID-3070; April, 1955...
A total of 1517 references are listed in this compilation. These include selected non-published USAEC reports and published articles in technical books and journals. An author and a report number index with availability information are also included.
- 15 Author Unlisted
"Corrosion: A Bibliography of Unclassified Report Literature"
Technical Information Service Extension, AEC
TID-3302; July 1956.
This bibliography contains 111 annotated references to unclassified reports on the corrosion of alloys, commercial alloys, ceramics, and plastics. References are included to reports written prior to December 15, 1953. Author, subject, and report number indexes are provided.

- 1 Aven, R. E.
"Mechanical Stability Considerations in the Design of Pressurizers for Large Homogeneous Reactors"
Oak Ridge National Laboratory, Tennessee
CF-52-1-77; January 11, 1952.
The build-up of mechanical oscillations in intermediate and large homogeneous reactors can place limits on pressurizer dimensions. Equations are developed for evaluating these limits.
- 2 Bledsoe, L. P.; and Daly, F., (Newport News Shipbuilding & Drydock Co., Va.); Elder, G. E., (Electro-Mechanical Labs., White Sands Proving Ground, N. Mex.); and Gall, W. R. and Miller, E. C., (Oak Ridge National Laboratory, Tennessee)
"Fabrication of the Homogeneous Reactor Test Vessel Assembly"
Welding Journal, (N. Y.) 35, 997-1006; October 1956.
The problems of the fabrication of the HRT vessel include those which are of a special nature occasioned by the unusual requirements of design and application. The selection of materials of construction of the pressure vessel, the fabrication of the heads and nozzles and flanged end connections to the pressure vessel, the development of the closing weld for joining the hemispherical halves, fabrication of blast shield, and the Zircaloy-2 core tank are described.
- 3 Bush, Philip D.
"ETR: Design and Construction"
Nucleonics 15, No. 3, 48-56; March 1957.
The methods are presented of the testing and selecting of the materials for the pressure vessel, piping and cement for shielding, and of solving some unusual construction problems in the ETR.
- 4 Cooper, W. E.
"Structural Significance of Applied Loadings. A Summary of Design Methods"
Knolls Atomic Power Laboratory, Schenectady, N. Y.
KAPL-M-WEC-10; April 2, 1957.
The various methods used for the evaluation of the structural significance of applied loadings are reviewed. Included are discussions of the various industrial codes as well as the structural design basis for reactor pressure vessels for naval service, and a discussion of the various activities which are directed toward modification or improvement of these methods. In addition one subsection deals specifically with the subject of thermal stress in nuclear reactors as something which is "different" and not necessarily considered as completely in the design of conventional apparatus.

- 5 Davids, Norman; and Kumar, Sudhir
 "Theoretical Analysis of Scabbing in Materials. The Basic Theory of Scabbing in Materials with Two Solids in Contact"
 Pennsylvania State University, University Park
 NP-5694; May 31, 1955.
 Basic relationships for scab formation in a solid are developed from the point of view of elastic materials. Relationships giving the thickness of scabs are obtained for semi-infinite plates and thin rods on the basis of normally-incident pressure pulses of arbitrary form. The effect of a backing medium has been expressed in terms of impedance matching relations between the two media, and these used to determine quantitatively the reduction in stress. Criteria for required thickness are developed on the basis of momentum considerations. A preliminary treatment is included for spherically diverging waves arising from a point explosion in a semi-infinite medium. Some available data are made use of in a discussion for the purpose of evaluating time constants of typical pressure pulses used in the report.
- 6 Erb, Carl W.
 "Hydrogen Embrittlement of Carbon Steel"
 Knolls Atomic Power Laboratory, Schenectady, New York
 KAPL-M-C E-1; July 13, 1955.
 An investigation on H embrittlement of carbon steel revealed that a steel containing a minimum alloy of 0.5% Mo should not suffer H embrittlement under SAR conditions. H effects due to irradiation in the pressure vessel are not included.
- 7 Gall, W. R.
 "MTR Report on Reactor Structural Steel"
 Oak Ridge National Laboratory, Tennessee
 CF-49-12-50; November 29, 1949.
 This is one of a series of design reports on the Materials Testing Reactor written to transmit necessary information to the design contractor. This report outlines the requirements of a structural steel framework within the concrete shield for the purpose of supporting and maintaining alignment of the reactor tank. The design of this structure as shown on ORNL drawings is also given. This proposed design consists of four columns, a horizontal frame of beams at the sub-pile room ceiling, a set of brackets attached to tank section A and four angle brackets to support the thermal shield. The following components are supported or resisted by the steel structure: the reactor tank and its contents; the thermal shield and graphite; concrete above the sub-pile room and two feet of concrete above the thermal shield; vertical forces due to internal water pressure in the tank.

- 8 Heisig, C. C.
 "Pressure Vessel Windows for Use at High Pressure"
 Oak Ridge National Laboratory, Tennessee
 CR-52-8-140; August 6, 1952.
 A brief description is given of most of the types of windows that have been used at high pressure, and comments by the designers on their utility are included.
- 9 Hogerton, J. F.; and Gross, R. C.
 "The Reactor Handbook. Vol. 2, Engineering"
 Atomic Energy Commission, Washington, D. C.
 AEC-3646; May 1955.
 Various phases of reactor engineering are discussed, specifically light-and heavy-water-cooled systems, liquid-metal-cooled-systems, gas-cooled systems, aqueous fuel systems, liquid-metal fuel systems, fuse-salt systems, handling and control, and reactor designs.
- 10 Horvay, C.
 "Problems of Mechanical Analysis in Reactor Technology"
 Knolls Atomic Power Laboratory, Schenectady, N. Y.
 American Institute of Chemical Engineers, New York; 1956.
 The role of the structural specialist in the atomic power field is outlined, and some of his tasks at KAPL are described. Thermal stress and shock phenomena receive particular attention. Shell and perforated plate problems are also considered. A brief discussion of design principles is included.
- 11 Kasten, Paul R.
 "Holding Force Required for HRT Pressure Vessel"
 Oak Ridge National Laboratory, Tennessee
 CP-55-5-166; May 25, 1955.
 A study has been made of the reaction force arising from breaking the inlet fluid nozzle of the pressure vessel, the initial pressure and temperature being 2,000 psi and 300°C. Based upon reasonable assumptions, the maximum upward distance the pressure vessel could travel would be ~4' if no holding force were present. With a 10,000-lb additional weight, the maximum distance would be ~2-1/2' and with a 20,000-lb additional weight the maximum distance would be ~1.6'. To ensure zero upward movement a holding force of 50,000 lbs would be required.
- 12 Kinnier, H. L.
 "Suggested Design Practices for Steel Reinforcing and for Expansion and Contraction Joints in Concrete Shielding for Nuclear Reactors"
 Oak Ridge National Laboratory, Tennessee
 CP-56-9-34; September 10, 1956.

- 13 Leeser, D. O.
 "How Nuclear Radiation Affects Engineering materials"
Materials and Methods, 40; August 1954.

- 14 Lloyd, R. G.; and Brooks, E. A.
 "Analysis and Recommended Modifications for Shield Structure --
 Brookhaven National Laboratory for the H. K. Ferguson Co., Inc."
 Babcock and Wilcox Company, New York
 M-4512; July 24, 1950.
 This report cancels and supersedes all previous information re-
 leased by the Babcock and Wilcox Co. relative to the shield
 structure, either verbal or written. An analysis was made of
 the shield structure at the Brookhaven Reactor and recommenda-
 tions are presented for reinforcing and altering the design to
 raise the strength level and increase the service life of the
 structure. Drawings and supporting calculations for the recom-
 mendations are included.

- 15 McLain, Stuart
 "Materials for Nuclear Reactors"
Chemical Engineering Progress 50, 240-4; May 1954.
 Materials choices for typical reactors are summarized, including
 source and fissionable materials, cladding materials, moderators,
 coolants, structural materials, shielding materials, and control
 elements.

- 16 McLain, Stuart, (Argonne National Laboratory, Lemont, Illinois)
 "Nuclear Power Reactors. I"
Electrical Engineering 74; January 1955.
 Fundamental technological problems connected with the design of
 power reactors are illustrated with a discussion of the Argonne
 Research Reactor, Boiling Experimental Reactor, and the NAA
 Sodium Reactor Experiment Programs.

- 17 McLain, Stuart, (Argonne National Laboratory, Lemont, Illinois)
 "Nuclear Power Reactors. II"
Electrical Engineering 74, February 1955.
 Discussions developed in part I are extended to the ORNL Homo-
 geneous Reactor Test and the Argonne Experimental Breeder
 Reactor - II.

- 18 McWherter, J. R.
 "Boiling Reactor Shield Design Study"
 Oak Ridge National Laboratory, Tennessee
 CF-53-1-230; January 9, 1953.

Preliminary drawings of a building and shield for boiling reactor experiments were completed. The underground shield has a removable cover, is 35' square and 30' deep with 6 cells separated by 3' concrete walls. A 10' wide operating gallery permits unshielded equipment during operation.

- 19 Mikhalopov, G. S.; Kesselring, K. A.; and Trocki, T.
"Engineering Development Problems in Atomic Power Plants"
Knolls Atomic Power Laboratory
AECU-2755; 1953.
The various phases in the development of a commercially successful atomic power plant are discussed. The problems that require the most development are experimental work are outlined. Reactor cooling considerations and mechanical design, heat transfer studies, heat exchanger fabrication and pump development are reported.
- 20 Murphy, J. J.; Soderberg, C. R. Jr.; and Rossheim, D. B.; (I. W. Kellogg Co., New York)
"Considerations Affecting Future Pressure-Vessel Codes"
Welding Journal, New York, 35 582s-96s; December 1956
Code requirements for pressure vessels are presently associated with extremely ductile materials and safety factors sufficiently ample to limit consideration to principal stresses evaluated by simple formulas. Advancements in physics have extended present knowledge of the fundamental behavior of metals, and metallurgical progress has provided new, improved, and special-purpose materials, many with physical properties enhanced by chemistry, heat treatment, or cold work with concurrent reduced ductility. The horizons of both analytical and experimental stress analysis have been vastly extended. Also new and improved inspection techniques and equipment are available. Since World War II, pressure vessel demands of the process, power, and other industries have rapidly increased in number, size, and complexity. In many cases this requires more precise design and fabrication, with justification of lower safety factors and improved economics. This trend toward balanced or adequately proportioned pressure vessels is further accentuated by new critical materials whose greatly increased cost justifies the objective of higher working stresses as obtained from improved design, fabrication, materials, and inspection. Tentative basic principles for revision of the ASME pressure-vessel codes to a broader base to suit the wide variations in service and economic demands are indicated.
- 21 Robertson, A. F.
"Effects of Thermal Radiation on Materials"
U. S. Department of Commerce, National Bureau of Standards
10A-113; November 1, 1953.

- 22 Segaser, C. L.
"A Study of Design Criteria for a Proposed HRT Heat Exchanger
Blast Shield"
Oak Ridge National Laboratory, Tennessee
CF-55-7-84; July 18, 1955.
In order to prevent the release of radioactive products during
a rupture, it has been recommended that a blast shield be
installed around the HRT heat exchangers. The shield should
be of sufficient strength to contain both the pressure and
impact forces which result from a heat exchanger failure.
Complete calculations concerning such a shield are included.
- 23 Warde, John M.
"Materials for Nuclear Power Reactors"
Oak Ridge National Laboratory, Tennessee
Materials and Methods 44, No. 2; August 1956.
Materials are currently the limiting factor in utilizing the
vast quantities of thermal energy released by nuclear fission.
Selection criteria for engineering materials used in the design
and construction of nuclear power reactors include such unprece-
dented demands as low neutron absorption and resistance to radia-
tion damage. This manual surveys those materials now available
that can be or have been used in harnessing nuclear energy for
generation of power.
- 24 Wilson, J. C.; and Perggren, R. G.
"Effects of Neutron Irradiation in Steel"
Paper presented to the American Society for Testing Materials,
Atlantic City, New Jersey; June 1955.