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A MODULAR, LOW-COST,

SODIUM-HEATED

STEAM GENERATOR

NAA-SR-MEMO-6798

MASTER

AEC Research and Development Report

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. Box 309
Canoga Park, California

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CONTENTS

	Page
Abstract	iv
I. Introduction	1
II. Concept Considerations for a Modular Steam Generator	3
III. Technical Plan of Action	7
IV. The Steam Generating System	8
V. Management Plan of Action	9
A. Design and Analysis	9
B. Fabrication Feasibility	9
C. Fabrication	9
D. Testing	10
VI. Development Schedule	12
Technical Staff	14

TABLE

I. Development Schedule	13
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FIGURES

1. Sodium-Heated Modular Steam Generator	5
2. Variation of Heat Transfer Coefficient With Steam Pressure	6

ABSTRACT

This report describes a proposed design and development program for a potentially economical, high-performance, sodium-heated steam generator. The concept employs modular units grouped together to provide for any combined power level desired. Fabrication expense is shown to be further reduced by the reduction in heat transfer area requirements. This reduction is achieved through improved heat transfer coefficients which exist under supercritical steam conditions. The testing of two small units (~ 18 Mw total) is shown to be sufficient to verify performance for any size unit because of the modular concept employed.

15

I. INTRODUCTION

The history of sodium-heated steam generators has been determined by conflicting requirements, with the result that no consistent engineering pattern for high-performance, relatively low cost units has evolved. Early efforts were supported primarily by military programs, with typical, and highly justifiable military requirements. These involved maximum reliability, minimum space, some unique considerations of mechanical shock resistance, very rapid rates of change of power, close attention to health hazards in confined spaces, and occasionally system configurations which were neither conducive to economic operation nor suitable for central station maintenance procedures. Application of liquid-metal-heated steam generators to reactor systems of electric-generating-station designs draws heavily upon previous military experience; a prime example is the Babcock and Wilcox once-through steam generator constructed for the Southern California Edison Company's Experimental Station at the Atomics International Field Test Laboratory. Though different in configuration from previous designs, this steam generator still retains the "third fluid concept," using mercury as a leak detecting fluid between sodium and steam. This design has proved mechanically reliable, but too costly for serious consideration in large plants.

The HNPF steam generators represent a modification of the "third fluid concept" and employ ferritic steel. This design has little possibility for advances in temperature and pressure, because of materials involved and because steam pressure is carried in the steam generator shell, and the sodium-containing tubes are subject to collapse caused by steam pressure on their exterior surfaces. Therefore, further development of this design to improve thermal cycles appears impractical.

Based on experimental findings, increasing confidence in ability to manage sodium-water reactions, as well as a realization that only single-walled steam generators are economically practicable, has led to their adoption in later plants. The Enrico Fermi plant of PRDC is a case in point; a single steam generator serves each loop, although relatively low-pressure superheated steam is generated. The EBR-II steam generators represent an approach to further steam generator subdivision, as 12 units are employed to generate 20 Mw of electricity. Though subdivided, they are highly protected against wall failure, and thus do not represent exact prototypes of economic steam generators.

The current high-pressure steam-generator design being developed by Alco represents a compromise between subdivision, single-wall construction, and low-cost construction materials. It is not expected to be the ultimate in steam generator design, as it has potential shortcomings in capacity-increase capability, working strength of materials involved, susceptibility to crevice corrosion in event of tube failure, and overall protection against corrosion. Though valuable lessons will be learned from its construction and test, it is considered that some of the design deficiencies may be anticipated, and a steam generator configuration may be planned to obviate major difficulties brought out during design and development of the Alco units.

Such a steam generator design has been conceived and is presented herein.

II. CONCEPT CONSIDERATIONS FOR A MODULAR STEAM GENERATOR

Drawing upon past experience, as well as attempting to foresee future requirements for sodium-heated steam generators, the following criteria were established:

- a) Use of single-wall tube construction
- b) Elimination of tube-to-tube sheet crevice on all superheater and reheater units constructed of austenitic stainless steel
- c) Use of ferritic materials, without extensive water-side corrosion protection on all evaporator units
- d) Use of multiple, repetitive, production techniques for production cost reduction
- e) Use of multiple units to minimize single-unit-failure effects on power production
- f) Incorporation of thermal shock resistance capability
- g) Capacity for concept extension to higher pressure and temperature regimes
- h) Suitability for test in as full a scale as possible at reasonable cost.

Subdivision of evaporator units has been determined by effecting an engineering compromise between high temperature strength of 2-1/4 chrome - 1 moly steel, cost of forgings for tube sheets, and decreasing cost with fewer units (requiring less headering, connections, and auxiliary equipment). Tube sheet crevices are proposed to be eliminated in the superheater and reheater units by welding tubes directly to upset connections on the tube sheet, using internal welding techniques developed by Los Alamos Scientific Laboratory, Griscom Russell, and others. The tube sheets themselves are forgings, small enough to be formed in a single operation without the necessity for especially large equipment. This, of course, greatly increases the number of possible suppliers, and offers an attractive potential for cost reduction through competition. Tube length is standard and tubes are straight. Again, this permits use of many potential vendors and allows competition to come into play as no special tube bending equipment is required. Differential expansion between tubes and shell is accommodated by shell

bellows; the shell itself is simply a 6-in. Schedule 40 pipe, readily obtainable from warehouse stock. A preliminary design of one of the proposed units is shown in Figure 1, illustrating the features described above. These units are equally suitable for 2400-psi and 3500-psi (supercritical) operation. Increase in tube wall thickness will be necessary for operation at the higher pressure but standard tubing may still be used. For a 300-Mwe, sodium-cooled nuclear power plant generating supercritical steam, sixty-eight 37-tube modular units are estimated to be required. This estimate assumes that 18% of the heat transferred from the reactor is dissipated into the reheat portion of the cycle; the 82% of reactor heat is transferred to the throttle steam.

Supercritical steam conditions result in high steam-side heat-transfer coefficients, as shown in Figure 2. This phenomenon allows a reduction in steam-generator heat-transfer surface area and a consequent reduction in fabrication and material cost, while giving an increase in overall plant performance. A conventional 2400-psi steam generator has a steam-side heat-transfer coefficient of about $750 \text{ Btu/ft}^2/\text{hr-}^\circ\text{F}$ in the superheater section. The superheater comprises somewhat more than 1/2 the total surface area. By comparison, the steam side heat transfer coefficient at 3500 psi is about $2200 \text{ Btu/ft}^2/\text{hr-}^\circ\text{F}$. Since the steam-side heat-transfer resistance predominates at the lower pressures, increasing the steam-side coefficient by utilization of supercritical conditions, results in a substantial increase in the overall heat transfer coefficient (U) and a decrease in the required heat transfer surface. It is therefore proposed that test units capable of generating 3500-psi steam be constructed; the design philosophy is equally applicable to lower pressure units, although the heat transfer surface must be increased as noted above.

Figure 1. Sodium-Heated Modular Steam Generator

Figure 2. Variation of Heat Transfer Coefficient With Steam Pressure

III. TECHNICAL PLAN OF ACTION

Upon completion of detailed design, two 37-tube modular units will be fabricated. The first unit will be fabricated of austenitic stainless steel, and will be designed to generate 3500-psi, 1050°F steam. This design will utilize forged tube sheets and butt-welded single-wall tubes. The caustic corrosion problem associated with crevices in the stainless tube sheet region will thus be eliminated. In order to up-grade the reliability of the tube-to-tube-sheet joint, a butt weld will be made a significant distance (~ 1 in.) from the tube sheet region. A joint similar to that utilized in the HNPF steam generator will be used except that approximately 5/8-in.-diameter tubes will be employed. It is recognized that this stainless steel unit will be subject to chloride stress corrosion in the event of failure of water treatment, permitting either oxygen or chloride ions to enter the feedwater. However, this condition is endemic with stainless steel units; experience with the SRE stainless steel unit has demonstrated that, with proper feedwater chemistry control, chloride stress corrosion can be avoided.

The second test unit, also 37 tubes, will be fabricated of 2-1/4 chrome - 1 moly steel and will be designed to generate 3500 psi, 950°F steam at the turbine throttle. This design will also utilize forged tube sheets and single-wall tubes. Because of the ferritic steel construction, tubes may be either butt-welded or rolled and welded into the tube sheet in a conventional manner. This material is not subject to chloride stress corrosion or caustic attack although it is subject to oxygen attack resulting in pitting. Rolling-in of tubes does not present a problem in this steam generator, as caustic attack on the tube sheet due to sodium-water reaction is not a significant problem.

Upon completion of fabrication, the first steam generator model (austenitic) will be installed in the SCTI. It is anticipated that this installation and test will follow testing of the Alco unit; the full 35-Mw capacity of the SCTI will not be necessary, however, as the individual module will require only 9 Mw of heat for a full scale test. This will, of course, reduce the cost of the test as fuel as an important fraction of the SCTI operation cost (approximately \$240,000 for one year's test at 35 Mw). The SCTI will require modification only in two respects; addition of a feedwater booster pump to supply feedwater to the evaporator and addition of equipment to desuperheat and depressurize the 3500-psi steam for

heat rejection into the existing condenser. Facilities in the SCTI are now being provided for depressurizing 2400-psi steam. This higher pressure steam will require an additional pressure reducing station.

Following test of the first unit, the second unit (ferritic) will be installed in the SCTI for a similar transient and steady-state test. The second unit will be tested only for operation to 950°F and 2400-psi steam conditions. A third test generating 3500 psi, 1050°F steam will incorporate the ferritic unit as the low temperature section connected in series with the austenitic unit as the high temperature section. This combined operation will closely simulate projected operation in a large central station plant incorporating reheat. In this case, reheaters and superheaters are operated in parallel and discharge their sodium into the shell of the evaporator operating in series. Test procedures have not been detailed but may well encompass testing to destruction by deliberately introducing controlled amounts of contaminants into the boiler feedwater, after the "qualification" tests of the unit have been completed. Testing to destruction will also demonstrate ability to manage high-temperature-sodium and high-pressure-steam reactions in an actual failure situation.

IV. STEAM GENERATOR SYSTEM

The complete steam generating system, composed of modules capable of absorbing about 9 Mw of heat apiece, is projected as groups of modules, each group being monitored for leaks at the outlet sodium piping. This monitoring will be accomplished using hydrogen detectors recently developed; the number of units to be grouped will be selected on the basis of an economic study balancing expense of installation and manifolding against potential revenue losses due to outage of varying number of units. Each group of modules would be valved both on the sodium and steam sides in order to isolate them in case of failure of an individual unit. Units will be grouped in order to take maximum advantage of readily available high pressure and low pressure pipe sizes for the steam and sodium circuits respectively. As a part of this program, a cost analysis for a complete steam generating assembly for a 300- to 360-Mwe plant will be prepared. This cost analysis will include piping, valving, structural supports, insulation, heating and associated controls, steam relief valves, instrumentation and sodium-water-reaction product disposal systems.

V. MANAGEMENT PLAN OF ACTION

A. DESIGN AND ANALYSIS

The conceptual design shown in Figure 1 will be detailed and analyzed for steady-state and transient stresses at the temperatures and rate of change of temperature specified for liquid-metal-heated steam generators. Upon completion of this stress analysis and detailed design, the fabricability of the components, which will have been appraised by manufacturing engineers throughout the design phase, will be checked with one or more competent established industrial manufacturers of such parts. These include the tube sheet forgings, attachment of tubes, tubing lengths, finishes, and specifications, shell-side bellows, and shell structure. Nozzle reactions will be assumed similar to those for the steam generators now under construction by Alco.

B. FABRICATION FEASIBILITY

Upon completion of this design and fabrication review, contracts will be let for procurement of key items. At present, these key items appear to be the tube sheet forgings and shell bellows. Three units will be obtained from each of several manufacturers, together with their estimate for quantity production. In addition, photostress models will be made of the tube sheets in order to verify calculations. Photostress studies will be performed by a laboratory specially skilled in this technique.

The tube sheet samples will be set up in a special fitting duplicating temperature and pressure conditions expected in service. Provision will be made for administering thermal shocks to tube and tube-sheet joints, with stresses mocked up simulating those experienced under full steam pressure conditions. Satisfactory completion of these tests should be considered confirmation of the design of this area of the steam generator, incorporating such changes as may have been shown necessary during the testing process.

C. FABRICATION

The detailed designs developed will be turned over to a heat exchanger manufacturer, for confirmation of the heat exchanger calculations, and construction of test units as outlined in Section III above. Liaison will be maintained with the selected vendor throughout the manufacturing process, to insure that specifications

are being adhered to, and that quality control is being maintained. Also, fabrication problems, and simplifications which may be made as a result of fabrication experience, can be factored into the design during this period. Delivery of the test units will be effected by the manufacturer, and warranties will be requested only to the extent of good workmanship and materials inasmuch as these are experimental units.

D. TESTING

The SCTI steam system will be modified to accommodate 3500-psi steam. It will not, however, be necessary to make these modifications extensive, as it is anticipated that these units will have a thermal capacity of about 9 Mw, rather than the 35-Mw capacity available for 2400-psi operation. A smaller steam line will be installed taking the 3500-psi steam from the steam generator to a pressure reducing station sized for 9 Mwt. Steam pressure will be reduced to 2400 psi, whereup it may be passed through the regularly installed pressure reducing station to the condenser. Because of the low steam flow, compared with the installed capacity, transient operations will be performed by manual control, as automatic operation will be difficult under these conditions. The additional expense involved in manual operations will be offset by the lower fuel cost involved in testing this steam generator.

Cyclic operations will be performed in accordance with the test program established for the Alco unit; particular attention will be paid to measuring water-side height fluctuations during violent transient conditions in order to establish (or confirm the lack of) need for separating apparatus. It is clear that, under normal operating conditions, such an apparatus is not necessary; however, prediction of water-side behavior is sufficiently difficult under transient conditions that experimental justification for its elimination is required. This separator would serve to prevent "slugging" the inlet superheater tube sheet with saturated water if significant carrier water existed during the transient.

The shell-side bellows will be closely observed through high-temperature strain-gage measurements and thermocouple readings. These measurements will be correlated with separately conducted bellows flexure tests, in order to insure that the lifetime results obtained from previous bellows testing predict long and satisfactory bellows performance in service.

Artificial nozzle reactions will be imposed simulating the maximum specified; these loads will be added to those calculated in order to bring moments and thrusts up to the required standard. Thrust and moment will be monitored by strain gages throughout the test to observe changes due to permanent sets in the material, if such should occur. Upon the completion of the series of tests, estimated to last approximately nine months, a complete test report will be written, containing an evaluation of both the ferritic and stainless steel module type supercritical steam generators. This series of tests, because of the multiplicity of modular units, is expected to demonstrate, in full scale, the engineering and economic feasibility of this type of steam generator for sodium-cooled reactor systems.

VI. DEVELOPMENT SCHEDULE

The work is divided into two phases:

Phase 1, requiring approximately 18 months, will involve the design, analysis, and construction of two modular units. It includes modification of the SCTI for testing the 3500-psi unit and installation of the first modular unit into the SCTI for test.

Phase 2, requiring approximately 9 months, consists of the test program as follows:

- a) Test of the austenitic unit to 3500-psi and 1050°F steam conditions
- b) Test of the ferritic unit to 2400-psi and 950°F steam conditions
- c) Test of combined units to 3500-psi and 1050°F steam conditions
- d) Tests to destruction or failure of combined units.

It is anticipated the program would follow the schedule shown in Table I. An accelerated schedule is possible, and depends upon earlier availability of the SCTI for testing. The test program could be shortened by elimination of the separate unit tests if only combined unit data are desired.

TABLE I
DEVELOPMENT SCHEDULE

	FY 1962				FY 1963				FY 1964			
<u>PHASE 1</u>												
Heat Transfer and Structural Analysis												
Steam Generator Design and Analysis												
Process, Piping, and Instrument Design												
Weld and Tubesheet Development												
Tube Sheet Testing												
Procurement of Long Lead Materials												
Test Procedure												
Steam Generator Fabrication												
Steam Generator Installation												
SCTI Modification												
Report												
<u>PHASE 2</u>												
Unit No. 1 (austenitic) Test												
Unit No. 2 (ferritic) Test												
Combined No. 1 and No. 2 Tests												
Combined No. 1 and No. 2 Tests to Failure												
Report												

This section
removed from master
not essential to report

11/9/62

TECHNICAL STAFF

R. W. Dickinson Director, Sodium Reactors Department

Education: University of California, A. B. , 1941,
 Physics/Mathematics
 U. S. Naval Post Graduate School, M. S. , 1958
 Mechanical Engineering
 Massachusetts Institute of Technology, S. M. , 1953,
 Nuclear Engineering

Experience: Mr. Dickinson joined Atomics International in 1956, and is presently the Director of the Sodium Reactors Department where he is responsible for directing the engineering, component and systems development for sodium cooled reactor programs and the operation of Atomics International's Sodium Reactor Experiment. In this position he also has cognizance of the Advanced Epithermal Thorium Reactor (AETR) project. Mr. Dickinson is a member of a Sodium Components Technical Working group which has been organized to advise the AEC on the current AEC sponsored components development program.

Prior to joining Atomics International, Mr. Dickinson had been a Naval Officer on active duty since 1941, resigning in 1956 with the rank of Commander, USN. From 1953 to 1956, his assignment was with the Bureau of Ships and Atomic Energy Commission. For two years he was the Head of the Nuclear Components Section of the Naval Reactors Branch, responsible for the design review and approval of development and testing of machinery and components exclusive of the reactor and its pressure vessel and controls, for all naval nuclear power plants and the Shippingport power reactor. For the naval plants such as the Sea Wolf and its land-based prototype this included all propulsion machinery; turbines, gears, condensers, T. G. sets, etc. ; as well as boilers, heat exchangers, pumps, piping, pressurizers in primary coolant systems utilizing pressurized water and liquid metal.

Prior to his nuclear naval experience he had been Assistant Planning and Estimating Superintendent for New Construction and Conversion at Mare Island Naval Shipyard for two years, Squadron Engineer for a submarine squadron for two years, and served five years aboard submarines, principally in an engineering capacity. He is qualified to command submarines, and now holds the rank of Captain, USNR.

Societies: American Nuclear Society
 American Society for Metals

Publications: Sodium Graphite Reactors, C. Starr and R. W. Dickinson
 Atoms for Peace Publication for Geneva Conference, 1958

W. J. Hallett

Group Leader, Sodium Components, Sodium Reactors Department

Education:

University of Colorado, B. S. , 1944, Chemical Engineering
University of Tennessee, graduate work in heat transfer, 1946

Experience:

Mr. Hallett joined Atomics International in 1950 and is the Group Leader of the Sodium Components Group. In this capacity he is responsible for the development of components for sodium cooled reactors and for the environmental test programs for such components. He has directed component development work for the Hallam Nuclear Power Facility, the Sodium Reactor Experiment, the Advanced Epithermal Breeder Reactor, and the Advanced Sodium Graphite Reactor programs. This work has involved the development of items for high temperature liquid metal service such as electromagnetic and mechanical pumps; instrumentation for flow, pressure, and liquid level measurement; valves and related process system components; nuclear reactor control mechanisms; and heat transfer and fluid dynamics studies related to reactor components.

Mr. Hallett was previously Supervisor of a unit responsible for the development of mechanical components for various reactor programs including the Organic Moderated Reactor Experiment, the Piqua Organic Moderated Reactor, and the Sodium Reactor Experiment. When he first joined Atomics International he was engaged in research in liquid metal technology including corrosion studies involving sodium, NaK, bismuth, and tin, and their utilization as heat transfer fluids for high temperature systems. He was later responsible for the development of fuel fabrication methods and for the fuel handling techniques for the Sodium Reactor Experiment. His close association with the development of liquid metal technology has led to his assignment to various review committees at Atomics International, the purpose of which is to review and advise on various reactor programs.

From 1949 to 1950, Mr. Hallett was with the Fairchild Engine and Airplane Corporation, NEPA Division, where he was Group Leader with technical staff duties. His work involved the study of materials for nuclear propulsion systems for aircraft and included investigations of liquid metals as heat transfer fluids. From 1944 to 1947 he was with the Tennessee Eastman Corporation as an engineer, in the Engineering Development Group.

Societies:

American Institute of Chemical Engineers
Alpha Chi Sigma
Sigma Tau

W. J. Hallett (Continued)

Publications: Dynamic Corrosion of Graphite by Liquid Bismuth
 Report NAA-SR-188, 1952 (co-author)

 Na PuP Mock-up Experiment, Memo NAA-SR-424
 1952 (co-author)

 SRE Fuel Element Manufacturing Procedure
 Memo NAA-SRE-1165, 1955 (co-author)

 Fuel Rod Loading Using Reprocess Slugs
 Memo NAA-SR-1221, 1955 (co-author)

 Experimental Development of a Fuel Handling System
 for the Sodium Reactor Experiment
 Report NAA-SR-2004 (co-author)

 SRE Fuel Element Damage
 Report NAA-SR-4488, 1959 (co-author)

A number of other project papers have been authored by
Mr. Hallett on materials studies and on the development
of reactor components.

J. B. Williams Project Engineer, Advanced Sodium Graphite Reactor Program, Sodium Reactors Department

Education: University of Washington, B.S., cum laude, 1946,
 Mechanical Engineering
 Willamette University, Graduate Study, 1944
 Mechanical Engineering

Experience: Mr. Williams joined Atomics International in 1954, and is presently the Project Engineer for the Advanced Sodium Graphite Reactor Program. In this capacity, he is responsible for the direction and guidance of the technical effort in the development of the Sodium Graphite Reactor Concept. Included in this assignment is the management and guidance of component development, advanced engineering and planning of future SGR systems, economic studies, and design of prototype reactor parts. In addition, he is the Project Engineer responsible for the design and installation of a replacement calandria core for SRE. He has previously served as Project Manager for the Danish Research Reactor installed by Atomics International for the Danish AEC near Copenhagen, Denmark. Prior to this, he was the Project Engineer on the design of a medical research reactor for the University of California at Los Angeles. He was the responsible design engineer for the KEWB reactor, which was constructed for study of transient behavior of aqueous homogeneous reactors of Atomics International. Mr. Williams has considerable experience in the design, analysis and construction of nuclear reactors of all types.

 Prior to joining Atomics International, Mr. Williams was a design engineer at the Radiation Laboratory of the University of California at Berkeley and Livermore, California (1950-1954). In this capacity he did considerable work on linear accelerator design, as well as on high-vacuum systems.

 In addition, he was previously a design engineer in the Project Engineering Department of the General Electric Company, at Hanford Works, Richland, Washington (1947-1950), where he did mechanical engineering design on nuclear components on the Hanford reactors and separation plant, plus overall design work on plutonium critical mass equipment.

Societies: Sigma Xi
 Tau Beta Pi
 Zeta Mu Tau
 American Society of Mechanical Engineers

Licenses: Registered Professional Mechanical Engineer, State of California, No. M-1160

J. B. Williams (Continued)

Publications: "Design of a Calandria Core for the Sodium Reactor Experiment," NAA-SR-3936, 1959

 "Sodium Facilities Volume," NAA-SR-Memo 4411, 1960

 "Technical Justification for ESCR Power Rating," NAA-SR-Memo-4082, 1959 (co-author)

 "Uranium Oxide 19-Rod Fuel Cost Study," NAA-SR-Memo-2696, 1958

Additional papers in the field of nuclear power and research reactors have been co-authored by Mr. Williams.

W. F. Anderson Research Specialist, Sodium Reactors Department

Education: Lafayette College, B. A., 1948, Electrical Engineering
 B. S., 1948, Civil Engineering
 California Institute of Technology, M. S., 1954, Civil Engineering; graduate work 1954-1955

Experience: Mr. Anderson joined Atomics International in 1955 and presently is in the Sodium Reactor Engineering Group of the Sodium Reactors Department as a Responsible Engineer on the Thermal Stress Fatigue Studies Program. In this position he directs research on the basic properties of materials as well as on the analysis and testing of simulated typical reactor geometries, primarily investigating their ability to withstand damage induced by cyclic thermal stresses. In addition, he performs heat transfer, stress and vibration analysis on reactor components and serves as a consultant to other engineers on problems in this field.

Previous duties included stress analysis on components for the SRE and OMRE and structural evaluation of the Graphite Moderated, Organic Cooled Reactor with Superheat. He is a co-holder of the patent on the Tabbed Plate Fuel Element for the OMRE.

Prior to joining Atomics International, Mr. Anderson worked for Southern California Edison Company in power plant design, construction inspection, startup and operation. He also carried out special studies on the analysis of stresses imposed by short circuits on electrical bus conductors and on the structural reliability of high voltage disconnect switches under earthquake loading. Starting in 1948, Mr. Anderson taught at the Arizona State College at Tempe for two years, followed by various positions in field engineering work including one year in South America.

Societies: Phi Beta Kappa
 Tau Beta Pi
 Sigma Xi
 ASCE
 Associate Member ASME Committee on Behavior of Pressure Vessel Materials
 Member Main Committee on Pressure Vessel Research, Welding Research Council

Publications: Paper No. 57 NESC-113 published by ASME, "Heat Transfer Considerations in the Use of Organic Reactor Coolants," by T. T. Shimazaki and W. F. Anderson

Development of Zirconium Moderator Cans (co-author)

High-Temperature Strain Fatigue Testing With a Modified Direct-Stress Fatigue Machine, NAA-SR-4051

L. T. Bogaert Senior Mechanical Engineer, Systems Unit, Sodium Reactor Engineering Group, Sodium Reactors Department

Education: University of Wisconsin, 1937-1939 and 1942-1943, Graduate work, Mechanical Engineering
Air Technical Service School, Air Force 1944, Graduate work, Mechanical Engineering and Metallurgy

Experience: Mr. Bogaert joined Atomics International in 1956 and is presently working as a Senior Mechanical Engineer responsible for design coordination of precision heavy equipment. He also acts in a consulting capacity making recommendations regarding the feasibility and fabrication costs of both reactor components and developmental projects. Previously, he was responsible for determining the methods, tooling and fabrication of the HNPF moderator cans assemblies. This included the design and fabrication of AI's 40-ft long automatic seam welder, capable of welding helium leak tight seams to very close tolerances at speeds up to 40 in. per minute.

Mr. Bogaert also served as AI Division Equipment Engineer to the NAA Corporate Office. In this capacity, he coordinated AI's equipment specifications and requirements with those of other NAA divisions. He was also responsible for the procurement of all standard and special capital equipment for Atomics International. Upon joining AI, Mr. Bogaert was assigned to the staff of the Chief of Manufacturing and responsible for the fabrication of the SRE vessels and shields and the OMRE core tank.

Prior to joining Atomics International, Mr. Bogaert was project manager at Century Engineers, Inc., on the Sidewinder Missile and Launcher development and pilot production program. While there, he also managed the Tullahoma Wind Tunnel Balance System "A".

Mr. Bogaert worked for ten years for the Kimberly Clark Corp., Machinery and Erection Division as a field erection engineer, production engineer and superintendent of Manufacturing and Development.

Societies: ASTE

D. F. Casey Heat Transfer Specialist, Sodium Reactor Engineering Department

Education: University of California, B. S. , 1949, Mechanical Engineering
 University of California, M. S. , 1953, Mechanical Engineering
 UCLA Graduate Work, Mechanical Engineering

Experience: Mr. Casey joined Atomics International in September 1955, and is presently a heat transfer specialist in the Sodium Reactor Engineering Department. Since joining Atomics International, Mr. Casey has been involved in heat transfer and fluid flow analysis and design of high temperature nuclear power plants for stationary and propulsion purposes. This has encompassed proposals, preliminary and final analysis and design of gas cooled, organic cooled, and sodium cooled reactors. This work required a thorough knowledge of reactor engineering, liquid metal technology, and high temperature heat transfer.

 Experience in the requirements of compact reactor concepts has been acquired through work on the Nuclear Ramjet Project summarized in the report, NAA-SR-2318 Summary Report, Evaluation of Nuclear Reactor for Ramjet Application.

 Prior to coming to Atomics International, Mr. Casey was employed by Northrop Aircraft as a Senior Research Engineer engaged in heat transfer analysis.

Societies: American Society of Mechanical Engineers
 American Nuclear Society
 National Society of Professional Engineers
 Sigma Xi

Licenses: Registered Professional Mechanical Engineer, State of California

Publications: "Theory of the Manometer Accelerometer," Review of Scientific Instruments Vol. 24, No. 11, Rev. 1953 (co-author)

 "Heat Transfer to Lead Bismuth in Turbulent Flow in an Annulus," Transactions ASME, Vol. 79, No. 7, October 1957 (co-author)

 "After-Scram Thermal and Hydraulic Performance of the Sodium Reactor Experiment," presented at Annual Meeting of the ANS, 1958 (co-author)

R. Cygan Research Specialist, Sodium Components Group, Sodium Reactors Department

Education: Rensselaer Polytechnic Institute, B. S. , 1944, Mechanical Engineering

Experience: Mr. Cygan joined Atomics International in 1950 and is presently Research Specialist in the Sodium Components Group where he is in charge of work being done at the Large Component Test Loop at Atomics International's Santa Susana testing facilities. He is also the Responsible Engineer for the development of components for sodium systems for the Hallam Nuclear Power Facility. He has done considerable high temperature liquid metal corrosion and heat transfer work and development work for Sodium Reactor Experiment and HNPF components including pumps, valves, piping, bellows, heat exchangers, and cold traps.

From 1944 until 1950 Mr. Cygan was successively; a Development Engineer at General Electric, where he was engaged in a wide variety of laboratory and production testing of electrical and mechanical equipment, hypersonic wind tunnel evaluation and construction and operation of liquid metal test equipment; an Associate Mechanical Engineer at the New Mexico School of Mines in charge of jet engine damage trials for Navy BuOrd project; and Consulting Mechanical Engineer with the Allied Engineering Service in charge of heating and ventilating design for commercial and residential construction. Mr. Cygan has been actively engaged in liquid metals work for over twelve years covering almost all phases of this technology.

Societies: Sigma Xi

Licenses: Registered Professional Engineer (Mechanical) in California

Publications: "Sodium and NaK Supplement to U. S. Navy Liquid Metals Handbook," (contributing author)

 "Circulation of Lead-Bismuth Eutectic at Intermediate Temperatures," NAA-SR-253

 "Compatibility of Sodium, Graphite and Stainless Steel," NAA-SR-258 (co-author)

 "Sodium Reactor Experiment Pump Development," NAA-SR-1662

 "HNPF Cold Trap Evaluation," NAA-SR-4382