213
2-25-70

1236
THIS DOCUMENT CONFIRMED AS BMI-1901

UNCLASSIFIED

DIVISION OF CLASSIFICATION

BY

DATE 4/27/70

MASTER

A SURVEY OF LMFBR COMPONENT MATERIALS LIMITATIONS AND REQUIREMENTS

P4280

Battelle Memorial Institute
Columbus Laboratories

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

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.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission makes any warranty or representation, expressed or implied with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights, or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

BMI-1901

Category: UC-25 &
Special Distribution

A SURVEY OF LMFBR COMPONENT MATERIALS LIMITATIONS AND REQUIREMENTS

Topical Report, Task 13

Arthur A. Bauer
Jack G. Conner

BATTELLE MEMORIAL INSTITUTE
Columbus Laboratories
505 King Avenue
Columbus, Ohio 43201

Prepared for U.S. Atomic Energy Commission
Under Contract W-7405-eng-92

Report Date: January, 1970

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

Reproduced in the U.S.A. Available from The
Clearinghouse for Federal Scientific and Technical
Information, Springfield, Virginia 22151.
Price: Full Size Copy \$3.00; Microfiche \$0.65

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	1
INTRODUCTION	2
RESULTS OF THE SURVEY.	4
Coolant Transport System.	4
Sodium Pumps.	15
Intermediate Heat Exchangers and Steam Generators	24
Control-Rod-Drive and Fuel-Handling Mechanisms.	36
Core Components	51
CONCLUSIONS AND RECOMMENDATIONS.	64
ACKNOWLEDGMENT	67
REFERENCES	68

APPENDIX A

NOMINAL COMPOSITIONS OF STRUCTURAL STEELS AND NICKEL-BASE ALLOYS .	A-1
--	-----

APPENDIX B

COMPOSITIONS OF WEAR MATERIALS	B-1
--	-----

APPENDIX C

MECHANICAL-PROPERTY REQUIREMENTS	C-1
--	-----

APPENDIX D

STABILITY OF CARBON IN FERRITIC CHROMIUM-MOLYBDENUM STEELS AT ELEVATED TEMPERATURES.	D-1
---	-----

APPENDIX E

MASS TRANSFER AND CORROSION IN SODIUM.	E-1
--	-----

A SURVEY OF LMFBR COMPONENT MATERIALS
LIMITATIONS AND REQUIREMENTS

Arthur A. Bauer and Jack G. Conner

To define the materials needs for selected LMFBR components, operating with sodium coolant at temperatures up to 1200 F, the functions, general design, operating characteristics, history of materials selection, and materials problems and limitations for the coolant transport system, pumps, intermediate heat exchanger and steam generator, control-rod drive and fuel-handling mechanisms, and core components were reviewed. Major materials requirements identified were associated with the need for mechanical-property design data, the problems and materials limitations resulting from corrosion and mass transfer in sodium, the evaluation and selection of wear-resistant materials for guide, bearing, and seal applications, the development of techniques to weld reliable tube-to-tube sheet joints, and the irradiation-induced swelling and ductility loss of structural materials exposed to a fast-neutron-flux environment.

INTRODUCTION

The purpose of this report is to present the results of a survey, conducted during the first half of 1969, on the materials and materials requirements for LMFBR components. Included in this survey were the following specific components:

- (1) Coolant transport system, i.e., piping and valves
- (2) Intermediate heat exchanger and steam generator
- (3) Pumps
- (4) Control-rod-drive and fuel-handling mechanisms
- (5) Core structurals.

Excluded from the survey were the reactor vessel, fuel elements, and core shielding components.

For purposes of this survey, each component was analyzed as to its basic function, general design, operating characteristics, materials selection in past applications, materials problems and limitations, and required materials studies and developments. The LMFBR Program Plan, WASH-1103, served as the basic reference for defining anticipated operating characteristics of the various components. The Plan also served as a source, along with a variety of other publications including those prepared in various component design studies, for defining the functions and design parameters of the components. Additional inputs concerning materials selection experience were acquired from reactor designers and operators, and, when possible, component designers and manufacturers. From these data, the component materials problems and limitations were individually defined and the required materials studies and developments evolved.

Materials problems and limitations arising from environmental factors were defined as those associated with temperature, with the presence of sodium, and with neutron irradiation. In this survey, it was assumed that neutron irradiation exposures would be sufficiently low, if necessary, by design, to be precluded from consideration for all but the core components. A 30-year lifetime was accepted as the system goal, although it was recognized that replacement of certain component elements might be necessary during that period. The ability to make lifetime predictions for such elements is a critical need.

Special note must be made of the fact that component operating conditions as defined in the LMFBR Program Plan were assumed. These conditions represent operating goals for commercial reactors which provide a challenge to both the materials engineer and component and sodium-system designer. Toward this end, temperature is a key and likely decisive factor in establishing materials and, ultimately, component performance, with materials problems and limitations being a direct function of the sodium core-outlet temperature. The 1200 F goal as set forth in WASH-1103 stretches many of the materials involved to the limit of their capabilities. Consequently, the acceptance of less stringent near-term performance goals for components will undoubtedly be necessary until such time as materials and design technology is developed to a state substantially as defined by the requirements and developments outlined in this report.

In the organization of the report, each of the five components has been treated separately. However, the nominal compositions of the materials of interest are assembled as Appendices A and B. Also, because some of the major materials problems are common to many of the components, to avoid unnecessary

repetition, detailed discussions in these areas are presented as Appendices C, D, and E. Appendix C, deals with mechanical-property requirements and Appendix E treats mass transfer and corrosion in sodium. Appendix D is a literature survey in which studies concerned with the stability of carbon in low-alloy (chromium and molybdenum) steels are summarized. The problem of carbon loss and transfer is particularly pertinent to steam-generator materials and this survey provides a basis for some of the recommendations made concerning needed materials studies and developments.

RESULTS OF THE SURVEY

Coolant Transport System

Function

The coolant transport system comprises piping and valves whose function is the containment, transport, and controlled distribution of sodium coolant as a means of transferring heat generated in the core, through an intermediate heat exchanger, to the steam generator. Valve functions in the system are checking (prevention of reverse flow), throttling (flow control), and blocking (shutoff or isolation of a selected portion of system).

General Design

The coolant transport system is expected to consist of large-diameter (up to 48 in.) thin-walled (1 to 1-1/2 in.) piping, including straight sections, curves or elbows, branch connections such as laterals and tees, and reducers. The piping, probably seam welded from formed plate, will be

installed by welding; girth butt welds and a number of fillet welds will be involved. Support for the piping, with its load of sodium, must be provided, and provision must be made to accommodate the expansions which occur both with the introduction of hot sodium into the piping and with changes in sodium temperature during operation. The exterior of the piping will be insulated and will be provided with some device for heating the sodium piping system.

The coolant transport system will consist of a number of loops, up to six in the primary system (reactor core to intermediate heat exchanger and back) and up to three in the secondary (IHX to steam generator and back). The steam generator may include evaporator, reheater, and superheater units requiring diversion of portions of sodium flow between the units. In addition, bypass flow to such devices as cold traps must be provided. Valves will be required in these multiloop systems, depending upon the system design, to provide some or all of the functions of flow control, antiflow reversal, and system shutoff. Antiflow reversal will be provided by check valves, while flow control and shutoff can be secured through the use of butterfly, gate, plug, globe, or ball valves or variations thereof. Common features of these valves include the valve body, valve seat, stem, and valve seal; check valves require no stem and respond to the force of flowing sodium. A dashpot or spring may be required to prevent slamming of the check valve on operation. The other valves may be actuated either by rotary or by translating motion of the stem.

Operating Characteristics

General operating conditions are defined on the basis of the LMFBR Program Plan.⁽¹⁾ For the primary circuit, sodium temperatures up to 1200 F in the hot leg and up to 900 F in the cold leg are envisioned with maximum flow rates in the piping of 149,200 gpm. Maximum internal pressurization is 125 psi. In the secondary circuit, maximum sodium temperatures in the hot and cold legs are 1165 and 850 F, respectively, while maximum flow rates and internal pressurization are 106,000 gpm and 90 psi. Temperature transients associated with reactor scrams or flow malfunctions can lead to thermal-shock rates of 35 F per sec.

Sodium purity is to be controlled at a maximum of 10 ppm oxygen and 20 ppm carbon, with hydrogen limited to its saturation level at 325 F. The sodium cover gas will be either argon (probably) or helium. The exterior of the piping for the primary will either be immersed in sodium, in the case of a pot-type system, or else be under a nitrogen atmosphere. The exterior of the secondary system will probably be exposed to air.

Besides the internal pressure and stresses produced by thermal shock the piping system and valves will be subject to a variety of loads. These include: line expansion forces, arising from changes in temperature of the piping system; weight of the piping contents, and insulation; loads associated with vibration, earthquake, and shock, such as might be associated with fluid sodium hammer or the shock wave resulting from sodium-water reaction at a failed steam-generator tube; bolt loads on flanged joints; and stem loads which may be involved in shearing frozen, oxygen-containing sodium. Design life for all components is 30 years.

⁽¹⁾References are at end of text.

Materials Selection

Current experience with piping in sodium-cooled reactors is limited to 1000 F or less.

The most widely used and considered material for piping and valve-body construction is Type 304 stainless steel, although Types 316, 321, and 347 also are considered and have seen use for this purpose, both in the U.S. and abroad. Ferritic 2-1/4 Cr-1 Mo steel has also been employed or considered for use, principally in the secondary system, while a niobium-stabilized grade of this alloy has been selected for both primary and secondary circuits, as well as the pressure vessel, of the German sodium-cooled KNK reactor.

Hardened surfaces for valves, such as in bearings or valve seats, have generally been of Stellite or Colmonoy. Inconel X has also been used in applications involving sliding surfaces.

In order to prevent stem leakage three types of stem seals, either alone or in combination, have been used: welded metal bellows seals, either alone or in tandem as a means of insuring against bellows failure; freeze seals, accomplished by cooling the area around the stem to freeze the sodium; and stem packing, commonly consisting of graphitized asbestos reinforced with Inconel wire or nickel braid. The packing gland is usually employed as a secondary seal and may include an intermediate gas-purging system.

Materials Problems and Limitations

A number of piping systems have been designed and are in operation at 1200 F and higher, including some loops involving sodium or other liquid

metals. However, experience with sodium in reactor systems is limited to temperatures of 1000 F or below and then to piping of a much smaller diameter than is envisioned for an LMFBR.

Detailed design of piping components is based upon stress-analysis considerations and materials stress limitations defined in various piping codes. The piping designer makes use of available materials property data, in conjunction with various analytical methods, to calculate permissible loads in the piping under the often complex conditions of stress and of loading history during reactor operation. While a code case is available to cover design to 1200 F, it is widely recognized that a more reliable design is required for high-temperature piping. Furthermore, materials property data and behavioral information for temperatures up to 1200 F are generally inadequate, particularly for weld materials, and for all piping and valve materials when a sodium environment is involved. These inadequacies, along with limited design experience at 1200 F and with the lack of proven analytical methods for calculating allowable design stresses, make a design lifetime projection of 30 years extremely uncertain.

While a variety of mechanical-property information has been obtained in sodium, much of this information is inadequate for several reasons. First, the data were obtained for application to fuel-element claddings on specimens of thin dimensions (1/16 in. thick or less) and extrapolation to sodium piping dimensions is uncertain. Application to weld and weld-zone materials is even less certain. In addition, many of the data were obtained in sodium with poorly defined impurity (oxygen and carbon) levels. Finally, sodium effects

are a function of such other variables as sodium flow rate, sodium circuit temperature gradient, and, possibly, sodium mass and hydraulic diameter; the influence of such variables has not been investigated.

Sodium piping is subject to complex and variable stresses and strains. Their nature will be indicated in detailed design analyses of a given system. The application of generalized experimental data (i.e., uniaxial tensile-test data) to the specific conditions encountered at different positions in the piping requires reliable analytical methods. The major analytical methods currently in use by designers (see Appendix C, "Mechanical-Property Requirements") are largely unproven.

Fracture-toughness and crack-growth-rate data for materials operating in the creep range are unavailable. The influence of sodium on crack propagation is unknown.

Mass-transfer effects can be described in general terms and for individual loop systems. However, quantitative data and analytical techniques are not available to permit development of a model which would be independent of a particular loop.

A number of the alloys considered for piping or valve application exhibit specific problems which limit their use and where improvement might be possible. Type 316 stainless steel is slightly stronger than Type 304 stainless, but is more subject to sigma formation on aging with an associated decrease in toughness. By careful control of composition it appears that a "balance" can be achieved to control sigma formation. However, definitive studies to define an optimum "balanced" composition range and the resultant properties have not been pursued. Ferritic 2-1/4 Cr-1 Mo steel is subject to transfer

of carbon to austenitic steels in sodium systems and the rate of carbon loss increases rapidly with increasing temperature. Essentially uniform carbon depletion has been estimated to occur⁽²⁾ across the wall of a 1-in. pipe when exposed in its 30-year lifetime to temperatures in excess of 950 F. Various alloy modifications to alleviate or minimize the mass-transfer problem have been suggested. Efforts concerned with this problem are reviewed in Appendix D, "Stability of Carbon in Ferritic Chromium-Molybdenum Steels at Elevated Temperatures".

A variety of trade-offs are involved in the specification of carbon level (w/o) for a specific grade of Type 304 or Type 316 stainless steel, whether normal ($C \leq 0.08$), L-grade ($C \leq 0.03$), or H-grade ($0.04 \leq C \leq 0.10$). The steels with the higher carbon levels are subject to sensitization, which normally will occur during fabrication of large sections, making them subject to intergranular corrosion if exposed to the atmosphere during shipping or subsequent field assembly. The selection of a low-carbon stainless steel, on the other hand, involves a penalty in the design stresses allowable by code. However, the allowable code stresses do not take into account aging effects which, over the long term, may eliminate the strength advantage offered in specifying a minimum carbon content. The data required to resolve selection advantages are not available.

A realistic specification of maximum carbon content for austenitic stainless steels, based on degradation in materials properties, also is unavailable. This applies both to the base material and to surfaces subject to carburization by mass transfer.

Leakage from piping in sodium loops has at times resulted in rapid and severe attack of the pipe exterior and the production of deep gouges in the surface. This phenomenon could be significant in design since, if undetected, it could reduce the wall thickness to a point where rupture could occur. Caustic attack, resulting from sodium reaction with water in the atmosphere and in the piping insulation, is a suspected cause although recent studies suggest that the attack is basically an oxidation reaction in which sodium oxide serves as a flux for the protective oxide on the steel; insulation has little effect on the reaction.⁽³⁾

Valves present special problems in such areas as the valve seat and stem. Valve-seal problems are generally related to improper design and operation rather than to materials limitations, except in the bellows, where fatigue may cause failure due to vibration.

The valve is particularly sensitive to distortion resulting from thermal stresses, which can distort valve seating surfaces and lead to valve leakage. The valve-seat area is also susceptible to mass-transfer deposits which can prevent complete closure. Experience is generally not available to permit the selection of hard facing materials for seat surfaces for use in 1200 F sodium, and methods of attachment are uncertain. This comment also applies to other bearing surfaces which may be involved in the valve-actuating mechanism.

The valve stem may be subject to extremely high loads in a frozen valve where the shearing of frozen, high-oxygen-containing sodium is required to free it. The strength of stainless steel may be inadequate for this purpose.

Required Materials Studies and Developments

An area of major uncertainty for the structural materials considered for piping application in the LMFBR is the long-term effect on mechanical properties of exposure to sodium at elevated temperatures. Type 304 stainless steel is the major contender for use as piping with Type 316 of secondary interest. Ferritic 2-1/4 Cr-1 Mo alloy steels, or variations thereof, are of potential interest principally in the secondary sodium circuit.

Among the data required by the piping designer are tensile data, including strain-hardening information over a range of strain rates, creep and stress-rupture data for times extending to 50,000 or 100,000 hr, stress-relaxation data, and high- and low-cycle fatigue information. Effects of exposure to sodium and resulting mass transfer, and of thermal aging require investigation. Information is required not only for the base materials but for weld materials and the heat-affected zone. In addition, tests are needed to confirm and refine analytical methods for correlating uniaxial and multiaxial stress behavior, and for estimating creep-fatigue and thermal-fatigue service life as a result of variable cyclic-loading conditions.

Further discussions of the specific mechanical-property requirements noted above are given in Appendix C.

Sodium corrosion and mass-transfer effects also require study. Specific areas which require investigation are effects due to stress, sodium mass, and hydraulic diameter. Conditions under which pitting corrosion and erosion and cavitation occur require definition. The influence of temperature,

temperature gradient, and sodium purity and flow rates requires evaluation along with an evaluation of the effect of transient changes in temperature, temperature gradient, sodium purity and stress level on corrosion and mass-transfer rates.

Methods for the prediction of corrosion and mass transfer by means of models which are independent of a particular loop require development. In support of this goal, studies of the solubility in sodium of such elements as chromium, iron, nickel, and carbon and of their interaction with each other and with other sodium impurities, notably oxygen, should be continued and expanded. Needed studies specifically related to carbon include development of a chemical-potential measurement technique and the subsequent measurement of the chemical potential of carbon in LMFBR structural materials.

A more detailed description of mass-transfer and corrosion study requirements appears in Appendix E.

A number of areas exist where alloy studies should be performed. Stabilized grades of low-alloy ferritic steels (i.e., 2-1/4 Cr-1 Mo) should be evaluated to select alloys which are stable with respect to carbon transfer in a bimetallic, polythermal system involving an austenitic steel. The initial basis for selection must be a decrease in the carbon activity or potential in ferritic as compared with austenitic steels. However, fabricability, weldability, strength, and toughness also must be considered and evaluated.

Studies of property variations with changes in alloy content of Types 304 and 316 stainless steel should be performed to determine whether properties can be optimized by judicious composition control or minor alloy

modifications. Development of "balanced" Type 316 stainless steel compositions for control of sigma formation is recommended. In addition, the effect of carbon content of the austenitic stainless steels, particularly in the aged condition, requires evaluation to determine whether the strength advantages of high-carbon steels in the unaged condition are maintained. Studies to determine the effect of high carbon contents ($C > 0.10$ w/o) on mechanical properties of the austenitic stainless steels also should be performed to develop a specification for maximum carbon content based on degradation in properties both for the base material and for materials subject to surface carburization by carbon mass transfer.

Studies should be performed to define the conditions under which gouging attack at the exterior of sodium piping in the vicinity of a leak can occur. The influence of type of atmosphere and insulating material should be investigated.

Studies to evaluate tensile properties and mass transfer and corrosion behavior in 1200 F sodium of high-strength materials (i.e., Inconel X, Inconel 718) for potential valve-stem or linkage applications should be performed. Considerably higher strengths may be required at 1200 F than can be obtained with the austenitic steels.

Hard facing and bearing materials for use in valve seats and valve linkages require evaluation. The stability of potential materials (i.e., Stellites, carbides) in sodium at 1200 F must be demonstrated. Methods of fabricating and attaching hard facing and bearing materials also should be investigated. The thermal-shock compatibility of these materials with the underlying base materials requires particular consideration and evaluation.

Sodium Pumps

Function

The function of the pump is to move sodium through either the primary or secondary system at a rate determined by heat removal and transfer requirements of the reactor core, intermediate heat exchanger, and steam generator by developing sufficient pressure to overcome system frictional losses and gravity involved in lifting sodium in the system.

General Design

A centrifugal pump consists of a number of elements, including a drive mechanism, a shaft with associated seal and bearings, an impeller, casing, and a pump housing.

In a totally enclosed or hermetic pump design, the drive mechanism and pump internals are all contained within the pump housing such that all elements have access to the sodium. A barrier is placed around the shaft to prevent ingress of sodium and fission products into the drive-motor enclosure. In shaft-sealed units, the sodium and drive mechanism are separated by a rotary face-type mechanical shaft seal. The seal may be either a dry or lubricated type.

Thrust and radial bearings support the vertical shaft. The thrust bearing supports the weight of the shaft along with any hydraulic axial thrust loads developed at the impeller. The radial bearing or bearings support and maintain alignment of the shaft. The thrust bearing is housed with the drive mechanism while at least one of the radial bearings is generally immersed in

sodium. The radial bearings may be either of the hydrostatic or hydrodynamic type with the sodium serving as the lubricant. The shaft, which is immersed at one end in sodium, passes through cover gas and a sodium barrier or seal to the drive mechanism.

The impeller is attached to the lower end of the shaft within the pump casing. Sodium is drawn into the pump casing, through the suction pipe by the impeller, and is expelled by the impeller out through the discharge pipe. The pump casing is supported on or is attached to the pump housing.

The pump, including or excluding casing--depending upon whether a fixed or removable casing forms a part of the design--can be removed from the pump housing for inspection and/or repair.

A thermal shield may be introduced into the pump housing to minimize the possibility of failure of this container by thermal fatigue.

Operating Characteristics

General operating conditions are defined on the basis of the LMFBR Program Plan.⁽¹⁾

The temperatures encountered in the pump are dependent upon whether the pump is located in the hot or cold leg of the sodium system. In the primary system, operation as a hot-leg pump results in temperatures up to 1200 F, whereas cold-leg operation is at temperatures of 900 F or less. For the secondary system maximum sodium temperature is 1165 F.

The pumps are intended to have a sodium pumping capacity of up to 149,200 gpm in the primary and of up to 106,000 gpm in the secondary. The sodium under normal operating conditions is to contain a maximum of 10 ppm

oxygen and 20 ppm carbon, with the hydrogen content limited to less than saturation at 325 F. The cover gas will be either argon or helium.

The pump shaft, which may be hollow, over 30 ft in length, and well over 1 ft in diameter, may be subject to whipping and bending because of its length. This may require the introduction of an additional support bearing if the length cannot be reduced. The impeller (up to 4 ft in diameter) will revolve at speeds near 1000 rpm and higher and will be subject to significant centrifugal forces. Maximum sodium velocities (100 ft per sec and higher) in the reactor system will occur in the vicinity of the impeller tip.

The pump housing and internals are subject to thermal-shock conditions with a maximum temperature-change rate of 35 F per sec. The shock will be particularly severe for the hydrostatic pump bearing because sodium is introduced to the bearing directly from the pump discharge.

Materials Selection

The most common material used in the construction of sodium pumps for reactors has been Type 304 stainless steel. Other stainless steels which have been used for sodium-pump construction include Types 316, 321, and 347. The only exception to the use of an austenitic steel occurs in Germany's KNK reactor, where a niobium-stabilized 2-1/4 Cr-1 Mo ferritic steel has been used in pump construction.⁽⁴⁾

For pump bearings, Colmonoy-faced shafts and sleeves have seen extensive use in sodium at temperatures below 1000 F. Stellite bearings also are in service. Stellites (Stellite 6 on Stellite 12) were substituted in

the Rhapsodie secondary pumps after seizing difficulties were encountered with Colmonoy (Colmonoy 5 on Colmonoy 6) bearings in both primary and secondary pumps.⁽⁵⁾

Most seals that have been employed in liquid-metal pumps have been of the oil-lubricated face type. Freeze seals also have been used. Labyrinth seals have been employed to prevent sodium-vapor flow into the area above the reactor plugs and into the motor housing. Gas-flow barriers have been used to supplement the mechanical seals.

Materials Problems and Limitations

The major materials problems associated with sodium pumps relate to the need for mechanical-property data and analytical methods which can be used in designing for long-time exposure to high-temperature sodium, for additional data defining sodium mass-transfer and corrosion effects, and for mechanical-property, compatibility, wear, and fabrication data which can be used in the evaluation and selection of pump bearing and seal materials.

Despite the availability of a variety of mechanical-property data for materials of interest for use in an LMFBR, these data are generally inadequate. Information concerning the effects of long-time thermal aging and of sodium exposure is lacking, in particular. Also, much of the available data were obtained on specimens of small dimensions, and size effects have not been determined. In addition, the sodium was not always well characterized or typical of sodium expected in an LMFBR so that results, while defining trends, may not be typical.

The pump housing will probably be constructed by seam welding of formed plate. The impeller and pump casing, because of their complicated shapes, will probably be cast but could be welded. The pump shaft may be cast or wrought. Depending upon whether cast, wrought, or welded material is involved and depending upon whether that material has been modified by subsequent treatment (i.e., heat-affected zone in weld or in metal underlying plasma-sprayed bearing overlays), properties will vary. Data are lacking which define these differences.

The pump housing can develop severe thermal stresses under thermal-shock conditions. Some relief for the housing can be provided by introducing a thermal shield which has no major structural requirement other than to bear the thermal gradient developed under transient thermal conditions. Data are needed to evaluate the design requirements imposed by materials limitations stemming from thermal fatigue.

In order to apply general experimental test data to the complex loading conditions encountered in a pump, reliable analytical methods are required for design purposes. While a number of methods are in general use at present, these are largely unproven.

Shaft-distortion problems during operation have been encountered in the EBR-II pump shaft because of inadequate stress-relieving treatments to insure dimensional stability.

Mass-transfer and corrosion effects can be described in general terms for sodium systems. However, data and analytical methods are presently lacking for quantitatively describing these effects independently of a given loop. In addition, data are not available to permit a description of corrosion, erosion,

and cavitation effects in the impeller region, which is subject to the highest relative sodium flow rates in the system.

There is uncertainty regarding the use of Colmonoy's as bearing materials in sodium at temperatures above 1000 F. The uncertainty extends to their wear characteristics and chemical stability in sodium, in particular to loss of carbon. Alternative materials must be considered but sufficient data for evaluation of alternatives are not available. In addition to exhibiting suitable wear characteristics and resistance to degradation of properties as a result of mass transfer, the bearing materials must be resistant to thermal-shock fatigue and be amenable to fabrication in a suitable form. The large differences in thermal expansion between bearing and structural materials normally encountered must be considered and accommodated in the attachment method adopted, as should postfabrication heat-treatment limitations imposed by the structural materials.

Present sealing techniques are generally unreliable. The use of hydrocarbons as seal lubricants provides a possible source of sodium contamination that should be avoided if at all possible.

Required Materials Studies and Developments

No significant materials problems are obvious for sodium pumps operating in the cold leg of the reactor sodium circuit, where sodium temperatures are 900 F or less. The principal materials requirements would be for improved shaft seals and techniques for large-diameter bearing fabrication. Consequently, the following discussion applies principally to a hot-leg pump.

Of major interest for structural materials for use in LMFBR sodium pumps are mechanical-property data which can be applied in designing for a 30-year life in 1200 F sodium. Among the data of specific interest are tensile and compressive-creep data which can be used in treating the loading conditions encountered by the impeller, casing, and housing during normal pump operation, as well as creep-fatigue and thermal-fatigue data, which can be applied to the transient loading conditions encountered by these elements. Data are required for cast and for wrought forms and for weld materials. In addition, data should be obtained for materials in the heat-affected zones adjacent to welds and to plasma- or flame-sprayed bearing surfaces. Effects of thermal aging and of property changes resulting from mass transfer in sodium must be evaluated, with particular attention being paid to representative thick sections.

Mechanical-property tests under carefully controlled conditions also are required to provide data which will verify or serve as a basis for modifying analytical methods presently employed in designing for the complex loading conditions encountered in components. A list of mechanical-property studies required for components, including those specifically applicable to sodium pumps, appears in Appendix C.

Distortion of mechanical and hydraulic elements cannot be tolerated because of the small dimensional clearances involved in pump design. Residual stress and annealing studies should be performed on cast and wrought structures to define adequate stress-relief treatments for specific materials of interest in representative cross sections.

Sodium corrosion and mass-transfer studies at 1200 F are required. The impeller area is critical because of the high sodium velocities in its vicinity. Corrosion studies in sodium at velocities of 100 ft per sec or greater are needed. Cavitation and erosion tests to define the extent of

damage, particularly at the impeller blade surface and in neighboring regions of the diffuser or casing, should be performed. The influence of temperature, temperature gradient, and sodium purity and flow rates on mass-transfer rates requires evaluation. A more detailed analysis of mass-transfer and corrosion study requirements appears in Appendix E.

Fabrication studies will be required for the production of large cast or wrought shapes if materials other than the stainless steels or ferritic alloys are used in constructing the pumps (e.g., Incoloy 800 or Inconel 625). A general review of present fabrication methods should be made and advanced methods, including improved casting techniques, powder metallurgy, and dispersion or composite technology, etc., should be examined as a means of producing component elements (impeller, in particular) with improved mechanical strength and reliability.

Screening tests are required to evaluate potential journal and shaft bearing materials combinations for use in sodium at 1200 F. Such tests should be based on wear* tests involving rotary motion with stop and start operation

* Definitions relating to wear:

Wear - the roughening or removal of material from a surface as a result of relative motion between contacting surfaces.

Self-welding - formation of a metallurgical bond between asperities of two surfaces in contact under pressure.

Galling - wear of a rubbing surface due to intermittent self-welding and fracturing of surface material to produce material transfer and a roughened surface.

Fretting - wear resulting from small relative oscillatory motions between surfaces, often involving a corrosive process.

Spalling - the fracture and loss of coherence of material at a surface as a result of fatigue in rolling contact or impact.

Abrasion - the cutting or gouging of a surface by harder angular particles or asperities.

emphasized. Maximum wear will normally occur at pump startup and shutdown in hydrodynamic bearings and in hydrostatic bearings also unless an independent source of pressurized sodium is provided to lubricate the bearings. Torque measurements can serve as a guide to binding and supplement examination of the bearing surfaces for evidence of wear. Variables which should be studied include temperature, specific load, and relative surface velocities. These should duplicate insofar as possible the conditions expected for operating LMFBR pumps. Sodium purity should also duplicate that expected in the reactor system.

Because changes in surface composition can alter wear resistance, mass-transfer studies related to the bearing-wear studies also are required. Bearing materials should be exposed in sodium loops under conditions where sodium flow rates, sodium purity, temperature, temperature gradient, other loop materials, and surface ratios of these materials to other types of materials are similar to those expected in LMFBR systems. Evidence of material instability resulting in preferential loss of a particular component or of surface degradation should be sought. Because of the importance of bearing stability, the wear tests also should be conducted in a flowing sodium system under similar controlled conditions.

Similar wear studies are required to screen potential seal materials but with tests performed in a dry, inert atmosphere simulating the expected LMFBR cover-gas environment. The exclusion of excess amounts of oxygen will be important in these tests since wear behavior, when sufficient oxygen is present to form a protective oxide film on the rubbing surfaces, is not indicative of behavior in the absence of any oxygen source.

Methods of attaching bearing and seal materials to form shaft and journal members require study and development. A major problem is the large difference in thermal-expansion characteristics of the structural versus bearing materials. Processes which should be investigated include plasma or flame spraying and vapor-deposition techniques. Materials applied in this manner will be limited to their as-applied properties because subsequent heat treatment will be limited by the underlying base material. As longer range solutions, graded bearings, involving a gradual change in composition to accommodate the expansion differences, should be considered; alternatively, methods for producing massive sections of bearing materials which can be spring mounted and which operate against segmented bearing sleeves should be developed. The latter will require the development of methods, and perhaps materials, which will assure reliable attachment and performance.

Improved seal materials development is also considered desirable for the long range. Composite, self-lubricating seal materials, consisting of a solid lubricant in a hard, wear-resistant matrix, should be developed. Techniques of fabrication are required which are applicable to the relatively large dimensions that will be encountered in LMFBR pump seals.

Intermediate Heat Exchangers and Steam Generators

Function

The intermediate heat exchanger (IHX) and steam generator (SG) fulfill similar functions, serving to transfer heat from one coolant circuit to another. Heat, generated in the reactor core, is transferred from the

primary sodium to the secondary sodium coolant circuit in the IHX. The sodium in the secondary circuit, in turn, serves to heat steam in the SG. "Steam generator" is a generic term that encompasses the evaporator (water is heated to form steam), the superheater (wet steam is heated to above its saturation temperature), and the reheater (reheats a portion of steam to superheat temperatures after passing through the high-pressure turbine stage). The reheater is an optional part of the steam-generating system. The evaporator and superheater may consist of individual units or be combined in a single unit.

The IHX is included to prevent possible contamination of the steam with the radioactive sodium of the primary coolant circuit. Contamination could result from a steam-generator tube failure if the steam generator formed a part of the primary circuit.

General Design

The IHX and SG consist essentially of a series of tubes within a shell. Heat passes through the wall of the tubes between the coolant in the tubes and the coolant within the shell. Depending on the design, heat can pass either way. The reactor system will include between two and six primary loops with either one or two IHX's per loop. Each secondary loop will contain a single steam generator.

The tubes are designed to permit thermal expansion without developing excessive stresses. This normally is accomplished in a tube, fixed at both ends, by providing sine-wave or similar bends which can accommodate the strains, or through the use of U-shaped tubes or of bayonet tubes which are fixed only at one end. Depending on design, an expansion joint may be required to

accommodate differential thermal expansion between the tubes and shell. In order to minimize vibration due to coolant flow over the tubes and to direct flow over the tubes for maximum efficiency of heat transfer, tube supports and/or baffles are provided within the shell.

Nozzles welded to the shell provide the junction with the piping which carries coolant to and from the IHX or SG. Within the shell, tube sheets are required to separate and form header or plenum regions for the entering and exiting coolant which flows through the tube. The tube sheets also seal the coolant flowing through the tubes from the coolant on the shell side.

In order to permit replacement of leaking or failed tubes, provisions for removal and replacement of tube bundles can be made. Replaceable bundles can be bolted in place. The modular concept permits the replacement of sections containing failed tubes.

Operating Characteristics

General operating characteristics are defined on the basis of the LMFBF Program Plan.⁽¹⁾ The IHX will operate with maximum primary sodium inlet and outlet temperatures of 1200 and 970 F, respectively. The maximum secondary sodium circuit inlet temperature is 850 F with an outlet temperature of 1165 F. The secondary circuit will be maintained under a positive pressure with respect to the primary. The primary system pressure will be between 4.8 and 115 psig and the secondary circuit pressure between 13.5 and 130 psig.

The steam generator will operate at maximum steam and pressure conditions of 1050 F and 2400 psig. Maximum sodium temperature will be 1100 F.

During operation, with two streams of coolant entering and exiting at different temperatures, the IHX and SG will be subject to severe temperature gradients which must be accommodated in the design. In addition, temperature transients or flow malfunctions can produce thermal-shock rates of 35 F per sec in the IHX and as high as 50 F per sec in the SG. Coolant flow over and around the tubing will cause tube vibrations and in turn cause tube wear at tube supports. Wear and fretting also can occur as a result of thermal expansion of the tubing and motion relative to the tube supports. The possibility of a tube failure in the steam generator leading to a shock wave as a result of the sodium-water reaction, not only in the steam generator but also to the remainder of the system including the IHX, must be considered during operation.

Sodium purity is to be controlled at a maximum of 10 ppm oxygen and 20 ppm carbon, with hydrogen limited to its saturation level at 325 F. Sodium cover gas will be either argon (probably) or helium. Maximum sodium flow rates for each primary loop will be 149,200 gpm and for the secondary, 110,000 gpm.

Materials Selection

Sodium-cooled-reactor experience in the U.S. has involved the choice of Type 304 stainless steel for the IHX with operating temperatures below 1000 F. Common practice is to choose the same material for the IHX as for the other primary components. Other materials used abroad and in test loops have included Types 316 and 321 stainless steel and niobium-stabilized 2-1/4 Cr-1 Mo.⁽⁶⁾

Sodium-reactor steam-generator experience⁽⁶⁾ is quite limited. Much of the U.S. experience involves double-walled tube construction for maximum insurance against possible leakage between the sodium and water sides of the generator. Because of its complexity such construction must be considered as noncommercial. Only Fermi, constructed of ferritic 2-1/4 Cr-1 Mo, has been of single-walled construction. The same material is used for both sides of the EBR-II tubes, which are either metallurgically (with a nickel-base alloy) or mechanically bonded together. For Hallam, 2-1/4 Cr-1 Mo appeared on the sodium side of the evaporator and water side of the superheater while a low-alloy steel and a titanium-stabilized alloy (4 Cr-0.5 Mo-0.5 Ti) were used on the water side of the evaporator and the sodium side of the superheater, respectively. A helium bond was employed between the tubes. Mercury-bonded double-walled tube construction of Type 304 stainless steel was chosen for the SRE. Type 321 stainless steel has been selected for steam-generator application in both France and Britain, while in Germany niobium-stabilized 2-1/4 Cr-1 Mo alloy is being used for sodium-cooled steam-generator construction. All the steam generators referred to involve sodium temperatures in the vicinity of 1000 F or below.

Shells, tube sheets, nozzles, and internal baffling and tube supports have involved these same materials. To optimize tube sheet-coolant compatibility, where dissimilar coolants may be used on either side, weld overlays have been applied, generally on the water side.

The choice of materials in experimental and in planned steam generators for LMFBR application are quite varied with principal consideration being given to Incoloy 800, Types 316, 321, and 304 stainless steel, and

ferritic 2-1/4 Cr-1 Mo steel as well as to stabilized variants of 2-1/4 Cr-1 Mo and to Inconels 600 and 625.

Materials Problems and Limitations

The IHX and SG suffer a number of the same materials problems, many of which result from the lack of experience and data for materials operating in sodium at temperatures in excess of 1000 F. The major areas relate to: (1) the need for adequate mechanical-property design data, including consideration of effects of thermal aging and of sodium exposure; (2) wear and fretting associated with relative movement of tubes in baffles or tube supports; (3) definition of sodium corrosion and mass-transfer rates and effects; and (4) joining of tubes to the tube sheet. Besides these mutual problems of the IHX and SG, the steam generator presents special problems in selection of materials which are compatible with sodium and steam. Possible sodium-water reactions present additional materials problems for consideration.

While a variety of mechanical-property data are available for Types 304 and 316 stainless steel in sodium and to a lesser extent for ferritic 2-1/4 Cr-1 Mo, these data are largely inadequate because they were obtained for specific application to fuel-element claddings, where small dimensions and a relatively short service life are required. In addition, many of the data were obtained in sodium that was inadequately characterized. The availability of suitable design data upon which to base the selection of an alternative material is still poorer as is the availability of data applicable to weld and weld-zone materials. Possible effects of sodium flow rates, of temperature gradients in the sodium circuit, and of sodium mass also have largely been disregarded.

The shell, nozzles, and tubing are subject to complex stresses of varying intensity, duration, and repetitiveness arising both from mechanical and thermal sources. Analytical methods which will permit the design application of general mechanical-test data to these complex conditions are required. The major analytical methods currently in use remain to be proven.

Factors and conditions leading to fretting and wear of tubes at baffles and tube supports are generally undefined. Methods of preventing or minimizing materials loss through appropriate surface treatment must be determined.

Mass-transfer and corrosion effects can be described in general terms for sodium systems and for specific loops. However, quantitative data and analytical techniques are not available to permit engineering design without extensive loop tests.

Critical to the construction of satisfactory IHX's and single-wall-tubed steam generators is the production of reliable tube-to-tube sheet joints and NDT techniques and systems which can be used to test these joints. Welding parameters have not been optimized for maximum weld integrity, and procedures and systems for rigorous examination of the weld joints are lacking. Optimum methods of overlay application to the tube sheets also have not been defined.

The possibility of sodium-water reactions in the SG due to failure of a tube is cause for concern for several reasons. Mechanical-property data required to calculate the response of the shell to sudden high-strain-rate loading, resulting from the pressure wave produced in the reaction, are lacking. The effects of small defects and of thermal aging on materials response, and of the superposition of the pressure wave on existing stress loadings, are

undefined as is the significance of directional property effects and of differences arising from heat to heat and processing variations. The mechanism of wastage and the influence of material composition and alloying on resistance to corrosive, erosive, and vibratory effects anticipated in the sodium-water reaction must be understood.

Alloy selection presents the major materials problem for the SG tubes. All materials presently under consideration exhibit limitations and uncertainties. The stainless steels, Types 304, 316, and 321, appear adequate in sodium but are all potentially subject to stress-corrosion cracking. Careful control of water chemistry may alleviate this problem but the solution is uncertain both in terms of long-time exposure and in terms of practicality of operating procedures. The ferritic 2-1/4 Cr-1 Mo alloy steel exhibits satisfactory behavior in steam but is temperature limited because of loss of strength above 1000 F and, perhaps more seriously, because of excessive mass transfer of carbon. Stabilized grades of the ferritic steels have been investigated to overcome the mass-transfer problem, and their use appears to eliminate the problem in isothermal systems, but their effectiveness in a bimetallic flowing sodium system with a large temperature gradient is not proven. Problems with the fabricability and weldability of the stabilized ferritic steels also exist and no experience in welding these materials has been accumulated in the U.S. up to this time. Incoloy 800 is less susceptible to stress corrosion than the stainless steels. However, an exact definition of conditions under which stress corrosion can occur and of methods for preventing it in Incoloy 800 is not available. Mass-transfer problems which may arise on the sodium side of the system as a result of nickel transfer also have not been evaluated.

Similar problems exist for other alloys which may be considered for SG tubing application. All require the development of sufficiently detailed data to evaluate the trade-offs involved in a particular selection and to guide the development of improved alloy materials.

Required Materials Studies and Developments

A major requirement for materials of interest in the IHX and SG is the availability of mechanical-property data which show the effects of long-time exposure to sodium. Among the specific information required are: strain-hardening and high-strain-rate data which can be used to analyze the effects of pressure waves resulting from sodium-water reactions in the steam generator; high-cycle oscillatory fatigue data applicable to vibrating tubing; tensile, creep, and creep-fatigue data which can be used in designing for the stresses associated with various complex mechanical and thermal loadings; tube-burst data for use in evaluating the rupture life of SG tubing; and stress-relaxation data for use in selecting and evaluating bolt materials. Effects of long-time exposure to sodium and resulting mass transfer, and of thermal aging on materials mechanical response, require investigation. Information is required not only for the base materials but for weld materials, the fusion line, and the heat-affected zone. Tube-to-tube sheet joints, in particular, should be subjected to tensile, push-pull fatigue, creep, and creep-rupture tests for evaluation. Fatigue tests of tubing material in steam to determine the effects of water chemistry on fatigue behavior must be performed. In addition, tests are needed to confirm and refine analytical methods for correlating uniaxial and multiaxial stress behavior and for estimating creep-fatigue and thermal-fatigue service life due to cyclic loading conditions.

A general discussion of mechanical-property test requirements for LMFBR components is given in Appendix C.

Experiments are required to evaluate the factors which control fretting corrosion and wear of tubes at baffles and tube supports. Tests involving rubbing motion should be performed at temperatures of 1000 and 1200 F in well-characterized sodium under conditions of controlled purity, rate of flow, and temperature. The tests should include both high-cycle, low-amplitude motion characteristic of tube vibrations and low-cycle, high-amplitude motion such as is associated with thermal expansion. The effect of contact pressure as related to normal force and area of contact also requires investigation. Other variables which require evaluation include materials involved, surface hardness and finish, and the effect of surface treatments and coatings such as nitriding, carburizing, or plasma spraying with Stellite or carbide materials. The object is to define conditions which lead to and which minimize fretting and wear.

The development of methods and models for predicting mass transfer and corrosion, independently of a particular loop, is required. At the present time, design predictions of corrosion and mass transfer in a particular reactor system require extensive loop tests involving mocking up that system and system conditions. Improved analytical methods, advancing the state of the art over the present basically empirical approach, must be developed. Specific studies related to IHX and SG tubing which are required include delineations: of the downstream effect and changes in rate of surface removal with respect to position; of the regions and nature of mass-transfer deposits and their effect on heat-transfer characteristics; and of the surface-carburization occurrence

and rates of penetration, which may be particularly significant in the IHX where carburization of both sides of the tubing can occur.

A more detailed description of general sodium mass-transfer and corrosion study requirements appears in Appendix E.

A critical need for the IHX, and for the SG in particular, is a method of producing sound tube-to-tube sheet joints, with a high reliability, which are capable of NDT inspection and verification. The internal-bore butt welding technique appears promising and should be developed to the point where joints of high reliability can be routinely produced. As an alternative technique, seal welding, in conjunction either with back brazing or explosive bonding to close the resulting crevice between tube and tube sheet, also requires development and evaluation. The back-brazing technique will require development of filler metals which are compatible with sodium. The explosive-bonding approach requires the development of suitable NDT techniques to demonstrate the achievement of a metallurgical bond. Joint evaluation will require extensive mechanical-property tests of representative joints, including push-pull fatigue, creep, and creep-fatigue and thermal-fatigue tests. Effects of exposure to sodium on properties must be defined and a determination of mass-transfer and corrosion characteristics of weld and filler metals is necessary. The steam corrosion behavior of the welded joints also must be investigated.

Since the tube sheet in the SG may well be clad for optimum coolant compatibility at either face, methods of cladding and of preparation for the tube-to-tube sheet joint require evaluation. Possible utilization of explosive techniques requires consideration.

An equally critical need exists for sodium-water reaction studies aimed at determining the mechanism of wastage when a leak develops in the tubing of the steam generator. Profound differences in the rate of wastage have been exhibited by different alloys, but the reasons for the differences have not been defined. Studies of the reaction, to determine the reaction products and mechanisms of their formation, are needed as a means of guiding the selection and development of alloys with maximum resistance to wastage. Methods of tube repair or of plugging also require development.

Stress-corrosion studies to determine the susceptibility of LMFBR candidate tubing materials to chloride and caustic stress-corrosion cracking should be performed. Such studies are needed particularly for those materials for which extensive steam-plant experience is lacking such as Incoloy 800, stabilized variants of the ferritic steels, and possible new developmental alloys. Tests should be performed in steam at 1050 F and at pressures up to 2400 psig. The long-term effects of normal and abnormal feedwater chemical purity, under steady-state, off-normal, and shutdown operating conditions, must be established. Accelerated corrosion tests could be used for screening purposes.

A variety of alloy studies are needed to evaluate and select improved materials for SG application. Stabilized grades or low-carbon varieties of low-alloy ferritic steels which are stable with respect to carbon transfer in bimetallic, polythermal systems involving austenitic steels should be evaluated for SG application. The initial basis for selection must be a decrease in the carbon activity or potential as compared with ferritic steels. However, alloy optimization cannot proceed independently of other alloy requirements

such as strength or, for instance, resistance to wastage in the sodium-water reaction or chloride and caustic stress-corrosion resistance. Consequently, efforts involving alloy evaluation should be aimed at developing an understanding of alloy effects and representative analytical models which can be applied in evaluating the trade-offs involved in alloy selection and modification. Fabricability, weldability, strength, and toughness also must be considered in the evaluation.

Studies also should be performed with present commercial alloys to investigate effects of minor alloy-composition, impurity, and structural variations--such as grain size and impurity form and distribution--on mass-transfer behavior and mechanical strength.

Control-Rod-Drive and Fuel-Handling Mechanisms

Function

Control of a fast reactor is achieved by adjustments in core reactivity through insertion or removal of fuel or nuclear poisons. The function of the control-rod-drive mechanisms is to position control members in order to regulate reactivity at desired levels during reactor operation, and, in the case of a rapid uncontrolled reactivity increase, to provide for a rapid decrease in reactivity by motion of control members in order to shut the reactor down. Complementary to these controls are reactivity oscillators, operated by oscillator-rod drives, which provide information on reactor-core neutronics and stability.

The fuel-handling mechanism is an apparatus whose function is the loading and unloading of fuel-element assemblies and replaceable core components into and out of the reactor and their transfer within the core or to appropriate storage or maintenance areas in the immediate vicinity.

General Design

Current concepts of reactor control involve systems in which drive mechanisms operate from above the reactor and through the reactor shield. Regulation and safety functions may be provided either by the same or by different control members.

The CRDM consists of a drive mechanism, located above the reactor shield plug, which provides the motion for lowering and raising control members, a drive shaft which penetrates the shield plug and which is actuated by the drive mechanism, and a gripper mechanism at the end of the shaft which latches on to the pickup head of the control member. The drive mechanism may be mechanically or magnetically driven; the roller nut and magnetic jack are two preferred designs. Sealing of the reactor atmosphere can be accomplished either between the reactor cover gas and the drive mechanism or above the drive in which case the drive mechanism will be exposed either to the cover gas or to the sodium. Operation of safety members commonly involves free or assisted fall under gravity with assistance provided by a spring or pneumatic device; deceleration may be provided by means of a dashpot or spring. A sensing device may be included which locates the pickup head of a control member for latching by the gripper mechanism. A variety of translating, and, possibly, rotating, bearing, guide, and wear surfaces are encountered in the CRDM.

Oscillator systems consist similarly of a drive mechanism, a drive shaft, and a gripper mechanism which attaches to a nuclear poison member within the core. Rotary or oscillatory motion may be used to provide controlled reactivity oscillations.

Fuel-handling mechanisms can be classified on the basis of whether the design provides either for under-the-plug handling, through-the-plug handling, or for removal of the plug to provide access to the fuel, and whether the design provides for fuel storage either within the reactor pressure vessel or external to the vessel. When the shield plug is removed to provide access to the core, an inert-atmosphere, shielded hot cell is provided above the reactor for transport and transfer operations; storage can be provided either internal or external to the pressure vessel. In the case of through-the-plug handling, double-rotating shield plugs permit positioning of the FHM for loading and unloading of the fuel; used fuel is immediately transported through the plug to external storage. Under-the-plug handling is normally coupled with in-vessel storage systems, although under-sodium transfer to an adjacent storage area is feasible; positioning of the FHM may be accomplished either by means of double-rotating plugs or by means of a single-rotating plug and a rotating or articulated arm.

The FHM consists basically of a drive mechanism which is located above the shield plug; a shaft which provides axial and, possibly, rotary motion; an arm either fixed to the end of the shaft which rotates with the shaft or which moves in a horizontal plane independently of the shaft; and grapples or latches at the end of the shaft or arm which grip the ends of the fuel assemblies or other components for removal. Because of the various motions involved, a

variety of bearing and wear surfaces are incorporated in the FHM. Basic types of bearings used to date include sleeve or journal, ball, roller (both taper and needle), and axial guide. When feasible, attempts are made by design to locate the bearings beyond the sodium environment and to translate the desired motions through seals. Thus, seals around the shield plug, which will contain the reactor environment while permitting rotary motion during fuel handling, are required, as are seals around the shaft, permitting reciprocating and, if necessary, rotary motion. Sodium vapor in the cover gas and beyond the seals must be considered, however.

Operating Characteristics

General operating characteristics are defined on the basis of the LMFBFR Program Plan⁽¹⁾, which envisions reactor operation with a core sodium outlet temperature of 1200 F and a flow of 2×10^5 to 3×10^5 gpm. Thus, the lower drive shaft and gripper mechanism of the CRDM will be immersed in sodium at this temperature with the shaft subject to vibration as a result of sodium flow. These same parts will be subject to neutron irradiation of varying intensity, depending upon distance from the reactor core. The upper portions of the shaft and the drive mechanism may be exposed either to sodium, to the cover-gas atmosphere consisting of inert gas and sodium vapor, or to, simply, an inert-gas atmosphere. Temperatures above the sodium in the cover gas will vary from the sodium temperature at the gas-sodium interface to about 120 F in regions above the shield plug. Parts immersed in the liquid sodium will be subject to thermal cycling associated with normal reactor operation and to thermal shock associated either with coolant-flow failure somewhere in the coolant system or with reactor scrams.

Depending upon the type of FHM system selected and, in particular, for those systems based on under-the-plug handling, portions of the FHM may be exposed to flowing sodium during reactor operation. The mechanism will, in general, see no external load other than that imposed by flowing sodium at these times. During refueling or core-component replacement operations, portions of the FHM shaft, any attached arm, and the grapples or latches will be exposed above the core to sodium at temperatures up to 800 F. Other portions of the shaft and drive mechanism contained in the FHM housing will be exposed to the same cover-gas environment seen by the CRDM. Shield-plug seals will also be exposed to the inert gas-sodium vapor environment.

The CRDM will be required to provide a maximum length of travel for the control members of between 34 and 55 in. with maximum push and pull capabilities, respectively, of 500 and 2000 lb. During steady-state operation, regulating control members will experience up to 120 stop-and-start cycles of 1/4-in. motion per hr (for shim control members the frequency will be much less) and up to a total lifetime number of 8×10^6 cycles; the number of full-length travel cycles is between 500 and 5000. Maximum rate of movement to produce reactivity changes for these members is 15 in. per min. For the safety members as many as 500 scrams may be experienced with maximum initial acceleration of 6 G; deceleration rates may be equivalent.

Refueling in a large commercial LMFBR is estimated to occur a maximum of twice a year with over 300 fuel assemblies being transferred or shuffled during this period. Along with intermittent operation, low sliding speeds are common in FHM's. Maximum load requirement for the FHM is 4000 lb in either push or pull operation.

A 30-year lifetime is the goal desired for all components.

Materials Selection

Reactor operating experience with CRDM and FHM materials has been obtained under conditions where maximum sodium core-exit temperatures have been 1000 F or less during reactor operation and 700 F or below during refueling. For the FHM this experience applies specifically to under-the-plug fuel-handling systems where portions of the FHM remain immersed in sodium during reactor operation.

Type 304 stainless steel is the basic structural material used for CRDM's and FHM's. The trouble-free performance to date with this alloy makes it the principal candidate for future use.

A variety of materials have been used for bearing and wear surfaces in both CRDM's and FHM's, where the materials requirements are quite similar. In a cover-gas environment of argon/sodium vapor, bearing and seal materials have included aluminum-bronze and phosphor-bronze alloys, Colmonoys 4 and 5, and Stellite 6 and 6B. Combinations have included aluminum-bronze alloy (Ampco 22) against a Colmonoy 5 shaft as a sleeve bearing, and aluminum-bronze alloy (Ampco 18-13) against a Stellite 6 sleeve in a drive-shaft seal. Colmonoy 6 versus Colmonoy 6 has also been used in a face-seal combination in argon cover gas. For sodium bearing applications various combinations of the Stellites and Colmonoys generally have been involved, although aluminum-bronze alloys have also been employed. Some combinations include Colmonoy 4 versus Stellite 6 and 6B and Colmonoy 4 versus Colmonoy 70 with Kennametal 95 and Kentanium 138A serving as male and female portions of a conical sleeve bearing in an oscillator device.

Excessive wear and unsatisfactory behavior have been observed for combinations of Stellite 6B with stainless steel and Colmonoy 4 with aluminum-bronze alloy (Ampco 22) in sodium.⁽⁷⁾ Beryllium-copper alloys were found to be incompatible with sodium⁽⁸⁾ as were plain-carbon steel ball-bearing cages which disintegrated in sodium. These were part of stainless steel ball bushings which operated in contact with the 440 C steel shaft of an oscillator drive.

For sliding-wear surfaces in sodium, hard chromium plate on stainless steel has been used as have combinations of nitrided stainless versus Stellite 6B and pads of Stellite 6 against Stellite 6B. Inconel X has seen service as a spring material and also has been used in a variety of the moving parts of safety- and control-rod gripper mechanisms which are subject to wear and shock loading. Other materials used have included Stellites 3 and 6B and hard chromium plate over stainless steel for latch jaws or fingers, and Stellite 6B and Colmonoy 5 cam surfaces; the latter combination was used in the fuel-handling mechanism of Fermi. Nitrided surfaces have been used both in sodium and in the cover-gas environment.

Sealing of the reactor environment has been accomplished principally by welded bellows and packing glands or end-face seals. The latter may be used in conjunction with gas flow to prevent sodium-vapor entry through the seal.

The ability to rotate one or more reactor plugs during fuel handling or similar operations, and to seal off the reactor environment at the same time, has necessitated the use of sliding or dip-ring plug seals. A low-melting metal seal has been most widely used with sealing accomplished by immersing a fin or leg in the sealant when molten. The sealant alloys used have included

Cerrobend (a lead-bismuth-tin-cadmium alloy), a eutectic tin-59 w/o bismuth alloy, and mercury and sodium-mercury amalgam. The latter amalgam is used as an inner seal in conjunction with a mercury outer seal, both of which are continuously molten. The Cerrobend and eutectic tin-bismuth alloys are either completely or partially solid during reactor operation, but molten for plug operation. To prevent reaction with the air atmosphere above the reactor shield plug in the Rhapsodie reactor, silicone oil has been used. The oil provides a protective inert blanket on top of the tin-bismuth alloy.

Materials Problems and Limitations

Problems and limitations of materials in control-rod drive mechanisms occur in assuring the long-term reliability and behavior of bearing, wear, and shaft-seal surfaces in the reactor cover-gas environment and in the selection and behavior of materials for such items as springs, bearing and wear surfaces, and control-rod grippers which will be exposed to a neutron flux and to sodium in the immediate vicinity of the reactor core. For the fuel-handling mechanism the major materials problems also revolve around bearing, wear, and shaft-seal surfaces but, in addition, around the performance of dip-ring plug seals.

One of the most critical areas for bearing and seal surfaces is the cover-gas environment which consists of an inert gas and sodium vapor. Surfaces exposed to this environment are subject to sodium deposition or frosting. The presence of small amounts of oxygen which will oxidize the sodium can lead to seizing of the moving parts. The sodium oxides (and/or carbonates) also may serve as an abrasive, although their qualities in this respect are largely undefined. However, the possibility of accelerated wear of sliding and rotating surfaces in the presence of sodium is of concern.

To prevent the sodium-frosting problem, design efforts involving shaft seals and gas-purge systems are aimed at preventing sodium-vapor ingress to the CRDM and FHM housings. However, equally critical problems exist in an inert atmosphere in the absence of sodium. Wear behavior is generally dependent on the surface oxides which form on contacting surfaces and, in the low-oxygen environment of the cover gas, these may be absent. Similarly, no medium, such as sodium, which can serve as a lubricant at the bearing surface is present. Thus, accelerated wear and associated problems of self-welding and galling may be encountered in a low-oxygen, sodium-free atmosphere with materials combinations which would behave satisfactorily in a less inert environment.

The selection of CRDM bearing materials for long-term operation in 1200 F sodium is uncertain. While proven materials exist for application in 1000 F sodium these same materials may be unsatisfactory at higher temperatures because of either a change in wear or bearing properties or mass-transfer effects. Boron-containing bearing materials should be eliminated from consideration for exposure to sodium in the vicinity of the reactor core during reactor operation because of B^{10} (n, α) reactions with neutrons. At the same time, fast-neutron effects on the properties of other types of bearing materials are largely unknown. The control-rod grippers are subject to particularly severe wear and to shock loads, requiring materials or materials combinations which possess and can maintain strength and toughness, as well as dimensional tolerances in the high-temperature sodium and neutron field encountered above the reactor core. Gripper surfaces must not be subject to galling and self-welding.

Various methods of fabrication and attachment of bearing and wear surfaces are available (e.g., weld overlay, flame or plasma spraying, mechanical attachment, welding, brazing). Depending upon the method used, various problems may be encountered. The method of fabrication affects materials properties and therefore performance; possibilities of improving wear resistance via heat treatment may be limited by the method of attachment selected; effects due to thermal-expansion differences between bearing and base materials, which may arise during thermal cycling, may be influenced by the methods of fabrication and attachment; and uniformity from part to part may be poor if hand fabrication operations are involved.

The control-rod shaft passes from the cover gas into the liquid sodium. Within the sodium the shaft is subject to thermal shock as a result of sudden changes in reactor power, to vibration associated with the mode of control operation and with the flow of sodium in the reactor vessel, and to rapid acceleration and deceleration when safety shutdown is required. In addition, the shaft is subject to mass-transfer effects and corrosion in sodium and to neutron irradiation of varying severity, depending upon distance from the reactor core. Mechanical-property values and their changes as a result of mass transfer and corrosion, long-term aging, and neutron irradiation are at present inadequate for design purposes.

The stress-relaxation behavior of spring materials and change in spring constants with long-time exposure to 1200 F sodium in the presence of a neutron flux are unknown. It is uncertain that adequate spring materials exist for application in this environment.

Dip-ring plug seals, involving the use of a low-temperature melting metal or alloy to form the seal, present a number of advantages such as simplicity of operation and lack of potential sources of contamination to the sodium coolant of the reactor. However, the low-melting metals or alloys used in these seals are generally incompatible with the inert gas-sodium vapor environment of the reactor to which they are exposed. Mercury-containing seals alloy with the sodium to form an amalgam which in turn is subject to oxidation at the external surface. This has necessitated the use of a second mercury seal in the Dounreay fast reactor to protect the inner seal. Similarly, sodium is soluble in the tin-bismuth eutectic alloy and is a major constituent of the dross formed at the external surface of the seal in the EBR-II where an oxidizing atmosphere is encountered. Dross formation has prevented normal plug rotation on several occasions.⁽⁸⁾

Required Materials Studies and Developments

Three separate types of screening-test environments must be utilized in evaluating potential bearing and wear-resistant materials: (1) inert gas with presence of sodium oxides, (2) inert gas, and (3) liquid sodium.

Sodium oxides (and/or carbonates) can form on various surfaces exposed to the inert cover-gas environment which will move and contact other surfaces. The abrasive qualities of the oxides are largely unknown and studies to define the wear characteristics of moving surfaces in their presence are desirable in order both to define the potential seriousness of wear problems, where oxide deposition may occur, and to guide selection efforts to materials with maximum resistance to sodium oxide abrasion. Such studies should include oxide

identification, and hardness and toughness characterizations of the oxides and the bearing and wear surfaces as a function of temperature. Wear tests should be designed to insure incorporation of oxides at the contacting surfaces under conditions of variable temperature, specific load, and relative surface speed.

Tests to evaluate and select materials based on wear in a dry, inert atmosphere, in the absence of protective oxide film formation by reaction between the surface and the atmosphere, are required. Tests of materials in sodium or in oxygen-containing atmospheres are unsatisfactory. Such studies should be performed in a low-oxygen, inert-gas environment typical of that expected in the reactor cover gas.

Tests to evaluate and select materials for operation in 1200 F sodium as shaft bearing and guide materials also are required. Evaluation should be based not only on observations of wear, galling, fretting, and self-welding behavior but on compatibility with sodium under conditions where mass transfer can occur. Such an evaluation could probably best be performed in a sodium loop where sodium flow rates, sodium purity, temperature, temperature gradient, materials, and surface ratios of materials equivalent to those expected in a reactor system could be simulated. The resistance of the materials to surface cracking and spalling as a result of thermal transients also requires evaluation by thermal-cycling tests.

Screening tests to select potential bearing and wear-resistant materials for application in any of the three environments noted are regarded as a prelude to actual component tests. It is not practical to duplicate every motion which might occur in application. However, an attempt should be made to duplicate as nearly as possible the expected motions producing wear. Materials

for sliding interface combinations are needed and the tests should duplicate, insofar as possible, the conditions under which the materials are to be applied. Thus, for bearing and shaft combinations, materials wear tests should be based on rotary motion with emphasis on stop-start operation and with torque measurements to detect binding. Pin-on-plate tests are satisfactory for evaluating translating guide-channel combinations as well as translating seals although, for seal-material combinations, relatively large pin surface areas may be desirable. The effect of vibration on wear characteristics should be investigated. Variables which should be studied include temperature, specific load, and relative surface velocities.

Self-welding characteristics, both in cover-gas atmosphere and in sodium, require evaluation. Many of the contacting surfaces operate with discontinuous motion. Tests, involving static loading of pins on a plate, should include an evaluation of specific load and loading time on parting force.

The majority of these screening tests should be performed in sodium and cover-gas atmospheres of a purity typical of that expected in an LMFBR system under normal operating conditions. However, a limited number of tests should also be performed to examine the effect of abnormal operating conditions on wear behavior.

An additional variant requiring study is the method of fabrication of wear-resistant materials (i.e., casting, welding, hard facing, flame or plasma spraying) and its effect on wear characteristics of the material involved. In addition, tests to evaluate the spalling resistance and mechanical stability of various bearing or wear-surface materials under thermal-cycling conditions, both as individual bodies and as attached to the base material, should be

performed. Such tests would be particularly applicable to portions of the system subject to alternate immersion and removal from sodium during fuel-handing operations. Once the fabrication technique is selected for the bearing materials, efforts must be pursued to insure reproducibility of the product and its properties.

High short-term strength and ductility and good resistance to impact at 1200 F are desirable for gripper-mechanism materials, and tests of these properties are required to aid in their selection. Low-cycle fatigue tests to loads above the yield strength and to times to initiate crack formation should be performed. Since the gripper mechanism will be exposed to neutron irradiation, property changes as a result of irradiation must be determined to evaluate potential long-term materials degradation. Wear tests in sodium involving impact and high specific surface loads must be performed to evaluate the characteristics of either these materials or suitable wear-resistant coatings. Sodium-loop tests also are required to evaluate the stability of these materials and/or coatings to sodium under simulated reactor-system conditions. If changes are observed, their significance in terms of wear characteristics and strength will have to be determined.

In order to evaluate potential degradation in spring characteristics by thermal and irradiation-induced aging, the stress-relaxation behavior of potential spring materials requires investigation at temperatures in the vicinity of 1200 F both with and without the presence of a neutron environment.

For the shafts of the control-rod and safety mechanisms, information on fatigue behavior associated with thermal-shock and vibration conditions is required. Both low-cycle, high-strain and high-cycle, low-strain fatigue tests

are desirable. In addition, tensile tests at high strain rates to simulate loading encountered in acceleration and deceleration of safety and control rods should be performed. These studies are required under conditions where mass-transfer and aging effects on materials behavior as well as the effects of irradiation on materials tensile and fatigue properties are evaluated.

Development of low-temperature-melting alloys for use in dip-ring plug seals should be advanced. This could involve improved performance of the eutectic tin-bismuth alloy. Unfortunately, the exact mechanism of dross formation does not appear to be well defined. It is not clear whether sodium dissolved in the eutectic alloy simply oxidizes at the exposed surface to form a complex oxygen-containing compound with tin and bismuth, or whether the solubility of sodium in the alloy first must be exceeded, with sodium-metal compound formation preceding surface oxidation. Studies to determine the sequence and products of dross formation are required along with alloying efforts aimed at increasing resistance of the alloy to oxidation and dross formation. Additions of indium appear to inhibit the rate of dross formation, and definitive studies of the mechanism of improvement are needed. Alternatively, other low-melting-temperature systems should be considered and their oxidation characteristics determined when contaminated with sodium.

Another potential solution to the problem of surface oxidation of the seal alloy is the provision of a protective blanket of silicone oil. However, the long-term effectiveness and compatibility of silicone oil with the rest of the system are undetermined and require evaluation. It is not certain, for instance, that at some sodium concentration in the seal alloy, oxidation and/or dross formation may not still occur. Also, the possibility that the silicone oil may act as a source of contamination for the seal alloy and, ultimately, of the reactor system must be considered.

Core ComponentsFunction

The core components which are considered are the lower grid-plate structure, the ducts, core hold-down and external support structure, and the fuel spacers. Their function is to form a reproducible structure which will support and locate the fuel elements in a stable configuration and to direct sodium coolant flow over the fuel elements for the removal of heat generated in the elements.

General Design

Fuel pins are inserted as bundles in ducts which form a geometric array within the reactor. Wear pads separate the ducts and provide clearance between assemblies for coolant passage. The fuel elements are separated and held in place by fuel spacers which may be in the form of spring clips, a wire grid, or a wire wrapped around the pin.

The ducts rest on a grid plate which contains holes to locate the fuel assemblies. The grid-plate structure normally consists of two plates, which in addition to supporting and locating the fuel assemblies also form a coolant plenum from which sodium is introduced to the fuel assembly through the lower duct nozzle.

Core hold-down and lateral support structures are required to position and hold the subassemblies firmly in place during reactor operations. Such structures must be capable of removal or relaxation in order to permit fuel removal and insertion during refueling operations. To resist any tendency for

upward motion of the core assemblies the ducts may be either locked in the lower grid or a hold-down device above the core may be provided for restraint. This upper hold-down device may also prevent lateral motion by positioning the head of the assemblies. However, lateral support, to prevent bowing or distortion of the core as a result of thermal and irradiation growth, is necessary. The restraining force in the core area is transmitted by the wear pads placed at different heights on the outer walls of the ducts.

The upper hold-down structure may be in the form of grid, fingers, or a plate which contacts and restrains each core assembly; the core cover or some latching device may also provide this function. Lateral restraint, if other than that given by the upper hold-down structure, may be provided by a rigid grid or by a clamp or band around the core.

Operating Characteristics

General conditions in the core region are defined on the basis of the LMFBR Program Plan.^(1,9)

The core will have a sodium outlet temperature of 1200 F. Individual ducts will experience temperatures in excess of this value. The inlet temperature to the core, and the temperature that the lower grid structure and ducts will see, is 900 F or less. However, reactor scrams or flow malfunctions in the primary or secondary circuits can cause either positive or negative thermal transients. The design severity of a negative transient is a maximum of 35 F per sec with a total temperature change of 300 F. A positive transient of similar magnitude, such as might accompany a pump failure in the secondary circuit, can accompany abnormal operation.

The maximum total sodium flow through a typical core is 3.04×10^5 gpm with a maximum sodium pressure at core inlet and outlet of 37 and 94 psig, respectively. Sodium velocity in the core ducts will be between 10 and 30 ft per sec. The sodium will contain maximums of 10 ppm oxygen and 20 ppm carbon, with hydrogen limited to its saturation level at 325 F or less.

The instantaneous flux in the core will be on the order of about 1×10^{16} nv.

Materials Selection

Core structural components in sodium-cooled reactors have been constructed almost exclusively of Types 304, 316, 347, and 321 stainless steel. In the EBR-II, Type 304 stainless steel is the principal structural material, while in Fermi, Types 304 and 347 have been employed for core components. Abroad, the French have chosen Type 316 and the British, Type 321 stainless for structural applications in the core.

Wear surfaces for the prevention of self-welding and galling are required at a number of points in the reactor core. In sodium reactors constructed to date such surfaces have generally involved the Stellites, Colmonoys, and nitrided and chromium-plated surfaces. Some specific applications in Fermi⁽¹⁰⁾ involve Colmonoy 4 wear or spacer pads on ducts, nitrided Type 304 stainless handling heads versus Stellite 6 inlays on hold-down fingers, Stellite 6B bushings for seating the duct nozzle on the grid plate, which serve to prevent erosion due to sodium leakage in the region, and nitrided stainless steel surfaces where the duct nozzle passes through the upper grid plate and rubbing contact occurs. In the EBR-II, chromium plate has been applied to both surfaces where the nozzle and grid plate contact to prevent erosion resulting from sodium leakage through the seat.

Materials Problems and Limitations

Materials problems with core structural components arise from the individual and combined effects of exposure to sodium, high temperature, and neutron irradiation on materials properties, and the requirement that the components must maintain their structural integrity and load-carrying capacity despite the changes that occur.

The grid-plate structure supports the core subassemblies and therefore is quite massive. The upper grid plate will be exposed to a reasonably high neutron flux which will vary, because of self-shielding, across its thickness. The maximum flux and the neutron spectrum at the upper grid face will be dependent upon its distance from the core and the nature of any shielding below the core. A loss in ductility will accompany irradiation. Regardless, the grid plate must be able to sustain cyclic thermal and mechanical stresses arising from transients in sodium flow and heat generation during reactor operation. It must also be capable of withstanding severe thermal shocks associated with flow malfunctions. Recognizing that embrittlement due to irradiation will occur, it may be necessary to design for a removable grid plate. A major uncertainty is the effect of irradiation on the ductility of the grid plate and its related resistance to mechanical and thermal fatigue. Data concerning irradiation-damage variations with neutron spectrum are unavailable for predicting changes in ductility and mechanical strength, and changes in crack-propagation behavior with position and thickness of the grid plate. A further uncertainty arises for materials in future reactors where the neutron spectrum may be quite different from that in current reactors.

The ducts, because of their exposure to high fluxes and fluences in the core region, present problems associated with irradiation-induced swelling due to void formation. While the rate of swelling is most rapid in structures near the core center, the most severe problem is encountered at the core periphery where sharp temperature and flux gradients occur. Since swelling is a function of both temperature and flux, differences in swelling behavior occur across a duct and produce distortions and bowing. It is believed that these can be restrained during reactor operation by core hold-down and lateral support structures. However, the nature of the stresses that develop due to differential growth and the extent to which they can be restrained are unknown. The rate and mechanism of stress relaxation under these conditions of stress generation also are undefined. An understanding of the combined effects of flux, fluence, temperature, stress, and stress relaxation as related to irradiation-induced swelling behavior is lacking. Also, while swelling behavior can be influenced by metallurgical structure and compositional variations, their precise role in control of swelling is undetermined.

There is a general need in core-component design for data that describe the effects of irradiation on the mechanical properties of materials. Properties at end of life must be sufficient to fill the structural requirements in the intended application and the projected lifetime may require adjustment based on the property changes. In addition to structural loads, the components are subject to transient mechanical and thermal stresses due to variations in sodium flow and temperature. Of particular concern are irradiation-induced decreases in ductility and fatigue life. Unfortunately, data generated on cladding materials are often inadequate for component application because of thickness differences.

Mass transfer also will affect the mechanical properties.

Decarburization and preferential nickel and chromium loss with resultant ferrite formation will occur at the surface of austenitic steels in high-temperature portions of the core, while carburization may occur in the low-temperature regions. The combined effects on mechanical properties of surface composition and structural changes due to mass transfer and of irradiation have not been adequately studied and are of particular concern.

Mass-transfer and corrosion effects will be significant in the core because of the high sodium temperatures and reasonably high sodium velocities. While much of the mass-transfer and corrosion data generated for cladding are applicable to core components, particularly the ducts, possible effects of neutron irradiation on rates of corrosion and mass transfer have not been examined. In addition, a need exists for materials which resist erosion and cavitation. Sodium leakage between the duct nozzle and grid-plate seat can lead to excessive loss of sodium flow through the duct as a result of enlargement of the leakage path by erosion and cavitation.

The application of wear surfaces is required at various points in the core structure where galling or self-welding of contacting surfaces may occur. Materials suitable for these applications at 1200 F and higher are needed. In addition to possessing resistance to galling and self-welding, the materials must be stable and not subject to mass transfer in sodium, must retain their wear resistance and mechanical properties in an irradiation environment, and must be resistant to spalling as a result of thermal cycling. Methods of attachment or application of wear surfaces, particularly of wear pads to the core ducts, are a major concern. Methods involving the application

of heat (flame spraying, welding) modify the underlying structure and its irradiation-growth response, thus possibly compounding the problem.

Several materials problems are involved in the selection of a fuel-pin spacer design. If a wire-wrap design is chosen the swelling characteristics of the wire and fuel-pin cladding should match, otherwise the wire will either loosen and thereby lose its spacer function, or tighten, in which case it will tend to bow the fuel pin. Either is undesirable and yet, because of differences in processing procedures and state of stress relief, differences in growth can occur. The extent of the problem is undefined.

A spring-clip or bird-cage spacer design relies on tension in a spring material to hold the fuel pin in place. The availability of a suitable spring material which will not relax as a result of exposure to the core sodium temperatures and high neutron flux is uncertain. Stress-relaxation data under these conditions are unavailable.

If a grid design is involved, the structure will be either welded or brazed. Brazing materials of proven compatibility with 1200 F sodium are lacking, as are data defining the effects of irradiation on the properties of brazed and of welded joints.

Required Materials Studies and Developments

A major need exists to define the individual and combined effects of irradiation and materials variables on the behavior of materials irradiated in a fast-neutron flux ($E > 1$ Mev) to high fast fluences ($>10^{23}$ nvt). Data are needed over the range of 600 to 1300 F. Of concern are irradiation-induced

swelling, loss in ductility, and accelerated crack initiation and propagation, as well as irradiation-enhanced creep, stress relaxation, and corrosion. Irradiation variables include flux, fluence, neutron spectrum, temperature, and stress while material variables relate to alloy composition, to modifications based on minor changes in addition and impurity levels, to methods of processing, and to structure.

Materials for which intensified irradiation studies are required are the basic structural austenitic stainless steels, Types 304 and 316, including both as-fabricated material and material that has been heat affected by welding or by flame spraying for the application of wear surfaces. Weld materials, and braze joints, require irradiation tests as do potential spring materials, including Inconel 718 and Inconel X-750. Irradiation of wear-resistant surfaces, such as are produced by nitriding stainless steels or by overlaying with Stellite or various carbide materials, also is needed.

Data concerning the effects of irradiation on such mechanical behavior as tensile strength, creep fatigue, and stress relaxation are needed. In conjunction with irradiation effects, effects of surface depletion by mass transfer in sodium also should be evaluated. Specimens with surface compositions and structures typical of those that form as a result of mass transfer in sodium circuits should be included for irradiation. Irradiation tests should also be performed on specimens previously exposed in a sodium loop.

It is recognized that a considerable effort has been made and is underway to determine irradiation effects in Types 304 and 316 stainless steel. However, much work is still required to refine our understanding of these effects and to define the influence of irradiation and materials variables,

as mentioned earlier, on materials properties. For this purpose, tensile tests remain the most useful technique for screening the changes in strength and ductility that result from irradiation. Studies of the effects of strain rate on the properties of irradiated material and of irradiation on strain-hardening characteristics should be expanded. Continued studies to define mechanisms of strengthening and of embrittlement and to identify the role of various irradiation-induced defects are needed.

The effect of irradiation on creep rate and creep-rupture life and ductility should be investigated both by postirradiation and by in-pile uniaxial and biaxial tests. Tests of this type are now in progress. A limited number of carefully controlled in-pile uniaxial and biaxial creep tests should be performed and analyzed to describe the effects of flux, fluence, and temperature on creep rate and also to test uniaxial and biaxial creep correlations in order to determine whether these also are applicable to an irradiation environment. Stress-rupture ductility data also should be obtained and analyzed to develop a correlation between the ductility of uniaxially and biaxially stressed irradiated materials. Creep tests should be performed in a manner such that the complete creep curve can be obtained, and analytical models which describe flux, fluence, and temperature effects on creep should be developed.

High-cycle (10^5 to 10^8 cycles) and low-cycle (1 to 10^5 cycles) fatigue tests of irradiated materials are required to determine how fatigue life and the time to crack initiation are affected by irradiation. These data are needed to evaluate effects of in-core vibration and thermal and mechanical transient effects. Methods of integrating variable fatigue damage, as through

a linear summation, should be examined and either modified or developed. Studies to define fatigue-crack growth rates in irradiated materials are required.

An understanding of stress-relaxation behavior for core structural materials must be developed in order to describe the stresses in core components which result from thermal expansion and irradiation-induced swelling or growth. Tests to determine the rate of stress relaxation during irradiation of pre-stressed specimens as a function of flux and temperature should be performed. Stress-relaxation tests also should be performed with spring materials to determine whether their spring characteristics are retained sufficiently for application in an LMFBR core.

In addition to concern over irradiation effects on mechanical properties, irradiation-induced swelling or growth represents a major problem areas. Irradiation tests to varying fluences are required over a range of temperatures and fluxes in order to define the interrelated effects of energy deposition and annealing on swelling rate and swelling saturation. These studies should be coupled with microstructural studies to identify defects and voids and their growth, movement, and modification with time. The role of helium, produced in (n,α) reactions, in the nucleation and growth of voids also should be investigated and evaluated. Modeling studies should be performed to analytically describe void formation and growth and resultant swelling behavior.

While sufficient data are available to provide at least a qualitative description of the role of flux, fluence, and temperature on mechanical-property changes and on swelling behavior, such data are almost completely lacking for

defining the role of neutron spectrum, although some theoretical basis exists for estimating its possible influence. Perhaps the greatest need in this connection is for complete characterization and reporting of the neutron spectrum in which irradiation experiments are performed. There is a requirement for the development of improved irradiation monitors to delineate neutron fluences in different energy ranges. Analytical methods for describing neutron-spectrum effects also must be improved. Experiments are required in which variations in neutron spectrum are deliberately sought, both within individual reactors and in different reactors (i.e., EBR-II, Fermi, ETR, etc.), for studies of irradiation effects on mechanical properties and on swelling behavior. Property changes and swelling behavior must be correlated with structural defects and void generation.

The effect of stress on swelling due to irradiation-induced void formation and growth requires definition. Experiments should be performed in which restrained specimens are irradiated and the resultant swelling behavior and accompanying microstructural defect and void production are determined. The stresses applied or developed must be precisely measured in order to determine whether a relationship between stress and swelling rate exists and whether a stress threshold exists above which swelling can be completely restrained. The effects of both compressive and tensile stresses should be examined. Changes in mechanical properties as a result of irradiation under restraint also should be evaluated.

Possible approaches to reduced swelling and loss of ductility during irradiation through modification or control of composition, structure, and/or processing variables require study. Effects of additives, such as titanium,

of cold work, and of carbide agglomeration--which have shown promise in the direction of achieving this goal--should be fully explored. Effects of thermomechanical treatments, of preferred orientation, and of unique structural and processing variations also should be examined. When significant reductions in swelling or ductility loss are realized, studies should be undertaken to define the mechanisms responsible for the improvement.

Studies are needed to determine if fast-neutron irradiation will have any significant effect on the corrosion rates of core-component materials under LMFBR conditions. Sputtering of surface atoms and modification of surface characteristics as a result of fast-neutron damage are possible sources of increased surface-removal rates which should be evaluated. Corrosion and mass-transfer studies are also needed to evaluate and select brazing and wear-resistant materials for use with core components. Tests of these materials should be conducted in sodium loops under conditions described in Appendix E.

Hard surfaces are required at various contacting points in the reactor core where either wear and self-welding or cavitation and erosion must be prevented or minimized. Screening studies, performed in sodium of typical purity expected in LMFBR systems, are needed to evaluate and select materials for these applications. As noted above, mass-transfer studies are required to determine if the materials or surfaces produced are stable in sodium circuits with respect to other structural materials under nonisothermal conditions. Sodium loop tests to investigate this stability and to identify any surface changes which may occur are needed. Subsequent evaluations should include material with similar modified surface characteristics.

Pin-on-plate tests in sodium are recommended to screen the wear and galling characteristics of various materials combinations. Variables which should be evaluated include temperature, specific load, and relative surface velocities. The effect of vibration on wear characteristics should also be examined. Self-welding tests, involving static loading of pins on a plate in sodium, should be performed. The effect of specific load and loading time on parting force should be determined. The spalling characteristics of these materials also should be evaluated by thermal-cycling tests.

The effect of irradiation on hardened surfaces and wear materials requires attention. Tests should be conducted to determine if swelling occurs in these materials and whether hardness and impact resistance are affected by irradiation. Tests should also be conducted to investigate self-welding characteristics during irradiation of surfaces in contact, under load, and in sodium.

Methods of producing, applying, or attaching hardened surfaces to structural components require investigation. Processes that may be considered include preparation of either cast or powder-metallurgical shapes which can be attached mechanically, by welding, or by brazing, applications by plasma, flame-spraying, or vapor-deposition techniques, and use of surface reactions such as nitriding. In applying or attaching a hardened surface, consideration must be given to possible detrimental effects on the underlying material.

CONCLUSIONS AND RECOMMENDATIONS

Significant materials needs for selected LMFBR components, operating with sodium coolant at temperatures up to 1200 F, have been established in this survey. While many of these needs are common to more than one of the components, they can, in general, be identified as being particularly critical for certain individual components. These critical materials needs are identified below.

Coolant Transport System (Piping and Valves). Critical to the design of high-temperature sodium piping for a 30-year lifetime is the need for a variety of mechanical-property data along with a reliable design basis or code. Required particularly are data which are applicable to materials behavior after long-time aging and after long-time exposure to sodium under conditions that result in mass transfer. Data describing the behavior of weldments under similar conditions are also a major need.

Experiments to determine the long-time effect of sodium exposure on the creep and stress-rupture behavior of structural materials (e.g., 50,000- and 100,000-hr tests) should be initiated immediately. Tests to correlate size effects with mechanical-property measurements also should be undertaken. An early start is recommended for experiments to define the effects of long-time sodium exposure and of aging, both in the stressed and unstressed conditions, on the tensile and short-term creep behavior of structural materials. All creep experiments should be performed in a manner such that the complete creep curve is defined.

Experimental data establishing the validity of various analytical methods employed to design for complex stress conditions and loading histories are needed. Specifically, methods are needed which can be applied to estimating service life under creep-fatigue conditions and to calculating creep response under conditions of multiaxial stress, including the strain to failure, from uniaxial-stress data. The applicability of strain-hardening and time-hardening mechanisms in predicting strain response for a given stress history also should be established.

With regard to sodium valves, materials problems seem more related to poor design or improper operation than to inherent materials deficiencies.

Sodium Pumps. A major need, unique to sodium pumps, is the availability of suitable materials for use as large pump bearings and seals. Screening tests to evaluate materials for these applications at the anticipated operating temperature should be performed. Corrosion, erosion, and cavitation behavior of materials in sodium at the high velocities encountered in the vicinity of the impeller tips require study. Methods of fabricating and attaching bearing and seal materials to form shaft and journal members are also critical needs to the pump component.

Intermediate Heat Exchangers and Steam Generators. The ability to predict the extent and location of depletion or deposition of elements subject to mass transfer is important to both the IHX and SG while the ability to minimize carbon transfer and aqueous stress corrosion by the selection of appropriate materials is a critical need for the SG. Consequently, experimental studies which will serve as the basis for development of a mass-transfer model,

independent of a particular loop, and which can be used to predict mass-transfer behavior and to select materials which will exhibit minimum carbon transfer in any sodium loop in an LMFBR sodium circuit, should be initiated. Materials selected must be compatible with the steam side of the generator as well as with the sodium side.

An equally critical requirement for these components is the development of reliable tube-to-tube sheet weld joints and of nondestructive testing techniques and systems which can be used to insure the integrity of these joints.

Control-Rod-Drive and Fuel-Handling Mechanisms. The lifetime of these components will undoubtedly be determined by the performance, in a variety of applications and conditions, of wear and bearing materials. Such materials are needed in guides, axial and rotary shaft and journal combinations, as thrust bearings, for grippers, and in shaft seals. Applications involve exposure to sodium at temperatures up to reactor operating temperature, to a cover gas including sodium and, possibly, its oxides, and to a completely inert environment. Screening tests to evaluate wear, galling, fretting, self-welding, spalling, and sodium compatibility behavior are needed as a prelude to actual component tests. Methods of attachment, and their effect on performance of the wear or bearing surface, require evaluation.

Development of low-melting-temperature alloys for use in dip-ring plug seals should be pursued to assure trouble-free operation during reactor refueling.

Core Components. A comprehensive effort is required to define irradiation-induced swelling behavior and ductility loss of core structural materials in a fast-flux environment. Fuel-element ducts and core grid plates represent especially critical components, but irradiation data acquired will also be useful for such other major structures as core hold-down and lateral support structures. Studies are needed to define not only the effects of temperature, fluence, and flux, but also the effects of stress, stress relaxation, and neutron spectrum as well as the influence of metallurgical structure, thickness, and compositional variations. An understanding of their individual and combined roles must be sought along with a knowledge of the complex effects of mass transfer in a sodium system.

Other core materials for which irradiation and mass-transfer data are needed include weld and braze materials, spring materials, and wear-resistant surfaces.

ACKNOWLEDGMENT

Contributions to the section on "Coolant Transport System" and the "Mechanical-Property Requirements" appendix by E. C. Rodabaugh and to the appendix on "Stability of Carbon in Ferritic Chromium-Molybdenum Steels at Elevated Temperatures" by M. Kangilaski are gratefully acknowledged. Appreciation is also expressed for reviews and comments provided by E. M. Simons on the "Mass Transfer and Corrosion in Sodium" appendix and by W. A. Glaeser and C. M. Allen concerning wear, and bearing and seal materials in the sections on "Sodium Pumps", "Control-Rod-Drive and Fuel-Handling Mechanisms", and

"Core Components". Useful discussions were also conducted with R. W. Bennett on steam generators and intermediate heat exchangers, and with D. R. Ireland on mechanical-property evaluations.

We also acknowledge information provided during preparation of this document by the following organizations: Atomic Energy Commission, Argonne National Laboratory, Battelle-Northwest, Liquid Metal Engineering Center, LMFBR Program Office, United Nuclear Corporation, and Westinghouse Advanced Reactors Division.

This report was prepared with the support of the Fuels and Materials Branch of the Division of Reactor Development and Technology, United States Atomic Energy Commission. Dr. T. Reuther was the project monitor, whose interest and assistance in preparing this report were invaluable.

REFERENCES

- (1) "LMFBR Program Plan, Volume 3, Components", WASH-1103 (August, 1968).
- (2) "Corrosion and Erosion Considerations for Sodium Piping in LMFBR Plants", Memorandum 316, SPLM-953 (June 3, 1969).
- (3) "Fuels and Materials Development Program. Quarterly Progress Report, Period Ending June 30, 1969", ORNL-4440 (1969).
- (4) Herberg, G., Meeting on Reactors, Deutsches Atomforum e.V., Frankfurt a.M., April 14-18, 1969.
- (5) Simmons, W. R., "Components for Main Heat Transport Systems for Sodium-Cooled Fast Breeder Reactors", Proceedings of the National Meeting on Fast Reactor Systems, Materials, and Components, Cincinnati, Ohio, CONF-680419 (April 2-4, 1968).
- (6) Morabito, J. J., and Savage, W. W., "Major Components and Test Facilities for Sodium Systems", Fast Reactor Technology, National Topical Meeting, ANS-100 (April 26-28, 1965).

- (7) "Evaluation of State-of-the-Art of Materials, Bearings, and Seals for Determination of Basic Tests", AI Report No. TI-681-81-040 (September 1, 1968).
- (8) "Ruther, W., Hutter, E. E., Cissel, D., and Crothers, C., ANL, Private Communication (May 13-15, 1969).
- (9) "LMFBR Program Plan, Volume 6, Core Design", WASH-1103 (August, 1968).
- (10) Kovacic, E. C., McHugh, W. E., and Shoudy, A. A., Jr., "Materials Problems and Selections in the Enrico Fermi Fast Breeder Reactor", Nuclear Metallurgy, Volume IX, Symposium on Materials for Sodium-Cooled Reactors, AIME (1963).

APPENDIX A

NOMINAL COMPOSITIONS OF STRUCTURAL STEELS
AND NICKEL-BASE ALLOYS

TABLE A-1. NOMINAL COMPOSITIONS OF STRUCTURAL STEELS
AND NICKEL-BASE ALLOYS

Alloy	Nominal Composition, w/o									
	Fe	Cr	Ni	C	Mn	Si	Nb	Ti	Al	Mo
Type 304 stainless	Bal	19	10	0.08 max L, 0.03 max H, 0.04 ≤ C ≤ 0.10	2.0 max	1.0 max	-	-	-	-
Type 316 stainless	Bal	17	12	0.08 max L, 0.03 max H, 0.04 ≤ C ≤ 0.10	2.0 max	1.0 max	-	-	-	2.5
Type 321 stainless	Bal	18	10	0.08 max	2.0 max	1.0 max	-	≥ 5xC	-	-
Type 346 stainless	Bal	18	11	0.08 max	2.0 max	1.0 max	Nb+Ta, ≥ 10xC	-	-	-
Type 440 C stainless	Bal	17	-	1.0-1.2	1.0 max	1.0 max	-	-	-	0.75
Incoloy 800	45	20	32	0.10 max	1.5	1.0	-	-	-	-
Inconel 600	7	15	Bal	0.04	-	-	-	-	-	-
Inconel 625	1.5	22	Bal	0.08 max	0.2 max	0.2 max	4Nb+Ta	-	-	9
Inconel X	7	15	Bal	0.04	0.70	0.30	1.0	2.5	0.9	-
Inconel 718	18	19	Bal	0.04	-	-	5.2	0.8	0.6	3
2-1/4 Cr-1Mo	Bal	2.25	-	0.15 max	0.45	0.50 max	-	-	-	1.0

A-1

APPENDIX B

COMPOSITIONS OF WEAR MATERIALS

TABLE B-1. COMPOSITION OF WEAR MATERIALS

Material	Nominal Composition, w/o										Comments
	Co	Ni	Cr	Fe	Mn	Mo	W	Si	B	C	
Colmonoy 4	-	Bal	10.0	2.5	-	-	-	2.25	2.00	0.45	-
Colmonoy 5	-	Bal	11.5	4.25	-	-	-	3.75	2.50	0.65	-
Colmonoy 6	-	Bal	13.5	4.75	-	-	-	4.25	3.00	0.75	-
Colmonoy 70	-	Bal	11.5	3.75	-	-	16.0	3.25	2.50	0.55	-
Stellite 3	Bal	3.00	29.0- 33.0	3.0	1.00	-	11.0- 14.0	1.00	-	2.00- 2.70	Others, 1.00 w/o
Stellite 6	Bal	3.00	27.0- 31.0	3.00	1.00	1.50	3.50- 4.50	1.50	-	0.90- 1.40	-
Stellite 6B	Bal	3.00	28.0- 32.0	3.00	2.00	1.50	3.50- 5.50	2.00		0.90- 1.40	-
Stellite 12	Bal	3.00	28.0- 32.0	3.00	1.00	-	7.50- 9.50	1.00		1.10- 1.70	Others, 1.00 w/o
Kennametal K 95	9.0	-	-	-	-	-	86.4	-	5.6	-	Co-91 w/o WC
Kentanium K 138A	20.0	-	-	-	-	-	-	-	14.5	-	54 w/o Ti, 3.9 w/o Nb, 7.6 w/o Ta (Co-65 w/o TiC-15 w/o NbTaTiC ₃)

Other Compositions, w/o: Ampco 22: Cu-14 Al-4.5 Fe-0.5 others
Ampco 18: Cu-11 Al-4.5 Fe-0.5 others.

APPENDIX C

MECHANICAL-PROPERTY REQUIREMENTS

APPENDIX C

MECHANICAL-PROPERTY REQUIREMENTS

The need for mechanical-property data for design purposes is common in varying degrees to all components. Elements of the piping, valves, pumps, heat exchangers and steam generators see much the same loads and environment as, to a lesser extent, do portions of the fuel-handling and control-rod-drive mechanisms. Core components also are subject to similar loads and environments, although these are unique in that they also are exposed to neutron irradiation. In addition to the need for specific mechanical-property data, there is also a need for improved analytical methods to treat creep-fatigue data and creep data under conditions of multiaxial stress and variable stress.

Certain of the data can be acquired in the absence of sodium, particularly where the data are to be used in developing analytical methods. Such tests could be performed both in air or in inert gas. However, properties obtained in nonsodium environments must be confirmed by property determinations in a sodium environment which duplicates as nearly as possible the intended reactor environment.

A variety of mechanical-property data, in particular uniaxial tensile, creep, and stress-rupture data, are available at temperatures in the reactor operating range or slightly higher, for many of the materials of interest. However, those data are not always sufficient and information is particularly lacking which can be used to adequately describe the effects of long-time exposure to sodium, or to define

potential size effects, which may be significant, particularly where changes in surface chemistry and structure occur as is the case for materials exposed to sodium. Thus, a major materials requirement for component design application includes sodium tests with specimens of large cross section to establish and correlate size effects. In addition, much of the existing materials information has been generated in sodium of relatively high oxygen and carbon content with low flow rates. Exposure to sodium under conditions more typical of those expected in reactor systems is necessary.

Three types of test procedures can be employed in studying sodium effects. One procedure is to expose the materials and perform the tests directly in sodium; such a procedure has special applicability to long-time tests, e.g., creep, creep-rupture, and high-cycle fatigue tests. A second procedure is to expose specimens in sodium for various periods of time and then remove the specimen for testing in an inert atmosphere. This procedure can be used most effectively where short-term properties are of interest, such as tensile or low-cycle fatigue properties. Finally, specimens can be prepared to simulate effects of long-term exposure to sodium. Such procedures are particularly applicable to surface carburization and decarburization studies.

If exposure to sodium is involved in test procedures, such exposure should be accomplished in a manner that reasonably duplicates the sodium circuit of a reactor with respect to temperature, temperature gradient, sodium flow rate and purity, and relative surface areas of materials in the different temperature regimes. Some specimens should be exposed at positions characteristic of temperatures in the cold leg,

as well as the hot leg, of the reactor in order to check mass-transfer effects at both positions. This recommended emphasis on controlled-environment loop tests is based on the need to obtain representative surface changes in test specimens for subsequent evaluation.

In selecting materials for testing the probable use of castings in certain components should be recognized. Properties of cast materials can vary significantly from those of wrought materials and tests should be performed to establish these differences.

Tests of weldments are also regarded to be of great importance. In all probability piping will be made from rolled plate with a seam weld. There will also probably be several girth butt welds as well as a number of fillet welds involved in various components. Data on properties of such weldments at elevated temperatures are essentially nonexistent. To insure adequacy in the weld area, it is not sufficient to know that a weld metal has an equal or higher strength to rupture or a higher creep strength. Creep of the base metal may lead to additional loads on the weld metal; if the weld metal has a lower critical strain (as existing data suggest), cracks may initiate in the weld metal.

A weldment commonly involves changes in the base material near the welds. Thus, the high-temperature properties of the weld metal should be determined along with those of the heat-affected zone and that of the unaffected base material. Tests should be performed, both in parallel and transverse directions, on weldments produced using sections of representative dimensions and by welding techniques characteristic of those to be employed in actual component fabrication.

LMFBR components will be subject to mechanical loads from a number of sources, including internal pressure, thermal gradients, thermal expansion, component and sodium weight, shock waves due to seismic disturbances, sodium hammer, and, possibly, sodium-water reactions in the steam generator. The magnitude, number of applications, and duration of the loads must be considered in the design of a component. In order to provide the mechanical-property data needed for analyzing component response under all design conditions, the materials studies described below are required.

Fatigue in the Presence of Creep

At present, the method currently employed to predict service life under conditions of creep fatigue involves the equation:

$$\sum_i \frac{n_i}{N_i} \sum_k \frac{t_k}{T_k} \geq 1, \quad (C-1)$$

where

n_i = applied cycles of strain, ϵ_i

N_i = cycles to failure at strain, ϵ_i

t_k = time duration of stress, σ_k

T_k = time to rupture at stress, σ_k .

Values of n_i and t_k are obtained from the anticipated loading history of a particular component. Values of N_i , for Type 304L and Type 304H stainless steel and the 2-1/4 Cr-1 Mo alloy, are available from ASME Section III, Code Case 1331. Values of T_k , for long-time loadings, are generally

available in published literature; the appropriate value of T_k for relatively short loading time is not as available--although the tensile strength might be used as a guide.

One justification for using Equation (C-1) is that much of the necessary information is available at present. However, the validity of the equation has not been sufficiently tested and caution must be adopted in its application. Experimental data are required either to establish its validity or to develop an alternative method of analysis. Cyclic-loading tests, on the materials of interest at temperatures of 1000 and 1200 F with various hold times and load ranges, are needed. Data should be obtained in the range of 1 to 10^5 cycles with loads to failure. Both strain-controlled and load-controlled types of tests are needed. Eventual consideration should be given to biaxial tests to establish the relevance of the uniaxial analysis to biaxial conditions.

Thermal Fatigue

In a similar vein, a need for an improved basis for evaluating thermal transient effects is required in order to adequately provide for thermal-fatigue requirements of LMFBR's. At present, determinations are based on isothermal test data, generally those applicable to the highest transient temperature. It is not at all clear that such a procedure is adequate. Improved analytical methods are required, supplemented by test data, to establish a valid thermal-fatigue failure criterion.

Creep Under Variable Stress

This aspect is similar to that of creep fatigue, but with a different objective. It is assumed here that fatigue damage is unimportant. The information desired is the strain history for a given stress history. Considerable work has been done in this important area, but the results are inconclusive. There is some evidence that "strain hardening" is the operating mechanism; other evidence suggests that "time hardening" or perhaps some combination of the two is more appropriate. Whether strain hardening or time hardening is operative appears to be at least somewhat material/temperature dependent. In order to establish the appropriate mechanism, variable-stress tests on the specific materials of interest at temperatures of 1000 and 1200 F are needed.

Stress Relaxation

If the strain response of a material under variable stress can be accurately predicted, it should also be possible to describe stress-relaxation behavior, at least in theory. In piping systems, for example, stress relaxation is particularly important because one of the major loads encountered (restrained displacement of thermal expansion) involves loading followed by relaxation. Strain behavior at local discontinuities also is more typical of a relaxation than a creep process.

In view of the lack of success to date in accurately predicting relaxation behavior from creep tests (constant stress), it is considered desirable to conduct stress-relaxation tests (constant strain) on the materials of interest at temperatures of 1000 and 1200 F for sodium components.

Creep Under Combined Stress

Most available data on creep of materials are based on uniaxial stress tests. In piping and other components the stress state is seldom uniaxial. Accordingly, some hypothesis is necessary to define the multi-axial stress state in actual components. This is an area in which considerable work has been done and reasonably good agreement is usually obtained insofar as creep flow velocity and direction are concerned if either the maximum-shear theory* or the deformation-energy theory** is used. However, additional experimental verification of these theories is required. Also, there is one significant aspect which has not been adequately investigated. This is the matter of defining the critical strain under multiaxial stresses (or strains). For example, if a material (at some elevated temperature in a long-time uniaxial creep test) shows incipient failure (microcracking or perhaps onset of third-stage creep) at 10 percent elongation, the concern is what the critical equivalent strain under equivalent multiaxial stresses would be. If the multi-axial critical strain turned out to be a small fraction of the uniaxial strain, then any component design procedure based on critical strain would have to take this into account.

Carefully controlled and designed tests are required at 1000 and 1200 F under uniaxial stress conditions and under multiaxial stress

* Maximum-shear theory: equivalent stress, $\sigma_e = 1/2(\sigma_{\max} - \sigma_{\min})$.

** Deformation-energy theory: equivalent stress, $\sigma_e = \frac{1}{\sqrt{2}}[(\sigma_1 - \sigma_2)^2 + (\sigma_2 - \sigma_3)^2 + (\sigma_3 - \sigma_1)^2]^{\frac{1}{2}}$, where σ_1 , σ_2 , and σ_3 are principal stresses.

conditions, involving both compressive and tensile stresses and different stress ratios, to establish the theory which most satisfactorily permits the application of uniaxial test data to multiaxial stress conditions. In addition, these tests would serve as a basis for developing a correlation for critical strain under conditions of uniaxial and multiaxial stress.

Short-Time Tensile and Creep Properties

Short-time loadings of components can result from such things as accidental overloads, sodium hammer, earthquakes, and sodium-water reactions. Data are required relating stress and strain, as well as the strain-hardening behavior of materials, at various strain rates at temperatures of 1000 and 1200 F. Stress-strain curves should be determined at strain rates ranging from about 10^{-4} to 10^2 per sec.

Similarly, data on the stress required to produce a small amount of creep as a function of time are required. These tests should be performed using variable loading rates to produce a given strain in a given (relatively short) time. Loading times covering a range from 0.1 sec to 24 hr would probably cover the potential range (sodium hammer could involve higher rates) of application.

Aging Studies

Base-material properties will change with time due to thermal-aging and strain-aging effects and the response of a material to subsequent loads will be affected. This will be of particular significance for materials subject to accidental or intermittent short-time loadings

such as were described above. Tensile and short-time creep tests are therefore required on specimens which have been thermally aged, with either constant or cyclic creep loads and with no load. Aging and test temperatures of 1000 and 1200 F are of interest.

Long-Time Creep or Stress to Rupture

Most data on creep or stress to rupture are obtained from tests lasting 10,000 hr or less. Extrapolation of such data to required lifetimes of 200,000 hr is a well recognized problem, in particular with respect to the strain at initiation of cracking. Obviously, this is an extremely difficult question to answer with experimental data. However, even a few tests to 50,000 or 100,000 hr would give added confidence in present extrapolation methods and such tests at 1000 and 1200 F should be planned and started.

Complete Creep Curves

There is a fairly extensive amount of data published in condensed form on creep or creep-rupture tests of base materials such as Type 304 and Type 316 stainless and the 2-1/4 Cr-1 Mo alloy. However, often the only data reported are the second-stage creep rate or the stress to rupture. These data are adequate for present design methods which use a criterion of some specified maximum secondary-creep rate or some specified stress to rupture. However, to improve design state of the art, it will be necessary to know the complete strain-time history of the test. Such data can, at present, be used to calculate the strain-time history in axisymmetric structures and it is possible, in the next

10 years or so, that analytical methods for nonaxisymmetric structures under creep conditions may become practical. An implied hypothesis is that eventually a more accurate and reliable design method will evolve based on calculating the strain history of critical points (as a function of time and loading) and comparing the accumulated strain and/or strain ranges with "critical" strain or strain-range data from material tests.

It is recommended that data from tests already performed be reviewed in an attempt to document available but unreported details and that all tests in the future be performed with the aim of obtaining and reporting complete strain-time data.

In general, creep data are available for most of the materials of interest, although not always in the temperature range of interest, nor in the desired sodium environment. In addition to long-time data, the greatest needs relate to size-effect correlations and the lack of compressive-creep data. The latter are needed for portions of various components which are subject to compressive-creep loading. The need for uniaxial-multiaxial stress-correlation tests was described earlier and pertains here as well.

Effect of Vibration on Creep

There is some indication that a relatively small-amplitude vibration (0.05 percent) may have a significant effect on creep rate. This effect is probably small and may be fully accounted for by the linear life-fraction rule defined in Equation (C-1). In view of the need for high reliability of sodium components, however, some exploratory tests of this effect, for the materials and temperatures of interest, should be run.

Tube-Burst Tests

Critical to the design of steam generators are the mechanical properties of the tubing after long-time exposure to sodium. Tube-burst tests should be performed to investigate effects of thermal aging, of mass transfer, and of exposure to sodium during testing. Temperatures of interest range from 800 to 1100 F with pressures of 1800 to 2400 psi.

Carbon-Migration Effects

Available information indicates that the carbon content of the material in sodium components may change as a result of transport by the sodium. From the standpoint of the designer, a critical question is whether a material with carbon content reduced by sodium transport is equivalent to a material initially fabricated with that same reduced carbon content. If so, then the designer can use high-temperature properties obtained from materials initially fabricated with various carbon contents. If not, and particularly if the material is relatively weaker than indicated by the reduced-carbon-content property data, the designer will need information from materials with carbon content reduced by sodium transport.

It appears that, in those parts of the system where carbon content is increased, the carbon concentrates in a shallow surface layer. The designer needs to know if this carburized case has any harmful effects--either at high temperature, or at room temperature--such as increased sensitivity to crack initiation. There is the additional question of

the behavior of such materials with composition gradients and how the properties of such materials correlate with the bulk properties of homogeneous material.

Consequently, an effort should be made in the various tests performed where a composition gradient exists, to determine the nature and extent of such gradients and to attempt to correlate property changes with the gradients. Supplementary tests, involving bend or impact tests may be desirable to evaluate fracture behavior and crack initiation in such zones.

Nitrogen-Migration Effects

The extent to which nitrogen migration occurs in sodium systems is not defined. Components or portions of components immersed in sodium would appear to be most susceptible to transfer. If loss of nitrogen occurs from the austenitic stainless steels it may be important to the designer because it can have a detrimental effect on the creep and stress-to-rupture properties. Conversely, the effect of nitrogen pickup is generally unknown. Therefore, tests to evaluate the effect of nitrogen content on stainless steels should be performed. The degree of emphasis may be expanded or contracted, depending upon the effects observed and upon the results of concomitant studies to determine the extent of nitrogen migration.

Metallic Depletion Effects

In addition to carbon and nitrogen migration, mass transfer of metallic elements occurs in sodium systems, with chromium and nickel being lost

preferentially from the surface of austenitic steels and nickel-base alloys. This preferential loss and consequent enhancement in iron content is observed to cause a phase transformation from austenite to ferrite in austenitic stainless steels. The rate of surface depletion and the manner in which the phase interface advances, whether uniformly or by grain-boundary penetration, vary depending upon the material involved. Therefore, tests are required to define these differences and to determine the effect of such surface depletions and phase changes on strength, ductility, and fatigue characteristics. Methods of producing these surface changes other than by exposure to sodium should be investigated for test expediency. Combined effects of metallic depletion and decarburization should be examined.

High-Cycle Fatigue

Information on the behavior of materials exposed to sodium and subject to vibration, such as may be experienced by steam-generator and heat-exchanger tubing, is required. Fatigue studies should be performed generally at temperatures of 1000 and 1200 F on materials of interest. Curves covering a range of 10^5 to 10^8 cycles are of interest.

Specimens should be tested both unexposed and after exposure to sodium as well as in sodium. In addition, materials of interest for use in the steam generator should be tested in high-temperature steam (900 to 1050 F) in order to determine possible degradation effects of the steam.

Fracture Behavior

Crack growth in materials operating in the creep range is a problem which may prove critical to the design of reliable components, particularly where relatively heavy sections or welds may be involved. An understanding of materials behavior and mechanical response in the vicinity of a crack must be developed.

Expanded studies are required in this area, particularly as related to environmental effects associated with long-time exposure to sodium. In this connection, studies to define fatigue-crack growth rates are required. An investigation of delayed-fracture susceptibility, that is, stress-corrosion cracking under a sustained load, also is needed. For example, carburization at crack tips may accelerate the crack-growth process.

A major requirement which must be met in order to fulfill LMFBR component design needs is to provide the various mechanical-property data described in this section. Many of these data can be put to immediate use while some, generally involving strain definition, are required for future high-temperature design analyses based on strain rather than stress limitations. In addition to the need for base-line property data for as-fabricated materials, of critical importance for design purposes are data which define the long-term effects of thermal aging and of sodium mass transfer on mechanical properties. Plans should be made for some minimum number of tests with test periods extending to 50,000 and 100,000 hr. While it is believed that the bulk of the needed information can be obtained by tests, in the absence of

sodium, of specimens prepared with surfaces which simulate those produced by mass transfer in sodium, it is essential that considerable data be obtained in sodium under conditions which duplicate as nearly as possible those which are expected to exist in operating-reactor sodium circuits. A first step must be the development of the necessary sodium test facilities which can provide the environment required.

APPENDIX D

STABILITY OF CARBON IN FERRITIC
CHROMIUM-MOLYBDENUM STEELS AT ELEVATED TEMPERATURES

APPENDIX D

STABILITY OF CARBON IN FERRITIC
CHROMIUM-MOLYBDENUM STEELS AT ELEVATED TEMPERATURES

Introduction

It has been found that if alloys with different carbon activities are placed in contact with each other (or both are in contact with a material that has solubility for carbon, e.g., sodium) then transfer of carbon will take place. The carbon will move from the alloy where carbon has the highest activity to the alloy where carbon has the lowest activity. This carbon transfer has been shown to take place even against a concentration gradient (if relative carbon activities are favorable), with carbon diffusing from the low concentration to the high concentration.⁽¹⁾ Of course, the temperatures must be sufficiently high for significant diffusion to occur.

It has been shown that essentially no decarburization of a ferritic 2.25 Cr-1 Mo-0.1 C alloy occurs in a loop made only from this alloy containing sodium at 650 C.⁽²⁾ However, if there is a temperature differential between the two parts of the sodium loop, then the carbon will be depleted from the hot leg and transferred to the cold leg. If parts of the sodium loop contain austenitic stainless steel, then the 2.25 Cr-1 Mo alloy is decarburized and the austenitic stainless steel is carburized. This carbon transfer occurs because of the lower carbon activity in austenitic stainless steel relative to the carbon activity in ferritic steel. Carbon depletion causes significant weakening of the ferritic steel and carbon pickup results in embrittlement of the austenitic

(1) References at end of this appendix.

stainless steel.

Carbon migration can be minimized by reducing the carbon activity in the ferritic steel, and a number of studies have been performed to investigate the effects of alloying on carbon stability. Short discussions of the techniques used for testing alloy stability along with some actual test results are given below.

Testing Techniques

Two different test procedures have been used to evaluate alloying effects on the decarburization behavior of ferritic chromium-molybdenum steels in the presence of an austenitic stainless steel. In one, diffusion couples consisting of the ferritic and austenitic steels of interest are placed in contact and heated at appropriate temperatures for times sufficient to cause diffusion between the steels. Chemical analyses and metallographic examination are then performed of samples from the areas that were in contact with one another in order to establish the amount of carbon transfer. The technique is relatively simple and provides a definitive description of relative carbon activity for the materials comprising the couple. However, the results are restricted to isothermal conditions and the technique cannot be applied to carbon-stability studies under nonisothermal conditions such as will occur in an LMFBR system.

The second method employed to investigate carbon stability involves placing coupons of the ferritic alloys of interest into a sodium pot or loop constructed of austenitic stainless steel. The coupons then are removed periodically and analyzed for carbon content. While a number of studies have been performed in sodium systems maintained under isothermal conditions, studies also have been performed in sodium loops with temperature differences between

various parts of the loop. While this method permits the additional influence of temperature difference on carbon stability to be investigated, the method also possesses limitations of general applicability since the results obtained are characteristic of the system in which the experiments are performed.

Thus, the important variables which influence test results are:

- (1) Flow rate of sodium
- (2) Temperature difference and temperature gradient
between materials in different portions of a loop
- (3) Purity of sodium (specifically carbon and possibly, oxygen)
- (4) Presence of other materials in the system
- (5) Surface-area ratios of ferritic to austenitic stainless
steels in the system.

It has been found that carbon transfer is accelerated by high flow rates, large temperature differences and gradients, and large surface areas of the austenitic steels relative to those of the ferritic materials. In many studies more than one ferritic alloy has been exposed at a single time so that carbon transfer from one to the other can occur.

Carbon contaminant in the sodium also can provide a source of carburization for austenitic stainless steels and for ferritic alloy steels as well. In a study performed by workers at Atomics International⁽³⁾ the alloys listed in Table D-1 were exposed in sodium loops built of Type 304 stainless steel. One of the loops was contaminated with carbon and all of the ternary iron-chromium-molybdenum alloys containing 5 w/o chromium or less lost carbon while the remainder were carburized. By comparison, in an uncontaminated loop all the alloys were decarburized.

TABLE D-1. CHROMIUM-MOLYBDENUM STEELS WHICH WERE TESTED IN A TYPE 304 STAINLESS STEEL SODIUM LOOP⁽³⁾

Alloy Content, w/o			
Cr	Mo	Ti	Si
1.25	0.5	-	-
2.25	1.0	-	-
5	0.5	-	-
5	0.5	0.4	-
5	0.5	-	1.3
7	0.5	-	-
9	1.0	-	-

Similarly, in a study at General Electric⁽²⁾, specimens of 5 Cr-0.5 Mo-0.5 Ti were held at 650 C in a sodium loop constructed of Type 316 stainless steel. Both the ferritic and austenitic steels were carburized. While the source of the carbon could not definitely be established, it was attributed to contamination in the sodium.

The effect of such variations in loop and test conditions are reflected in the results of the various investigations that have been performed. These variations, along with generally inadequate sodium characterization, make the results qualitative in nature such that comparison evaluations of materials tested in a given sodium system are possible but that quantitative comparisons with data obtained for materials in another system cannot be made.

Alloy Approaches

In ferritic chromium-molybdenum steels, carbon activity has been found to decrease with increasing chromium content.⁽⁴⁾ This effect is illustrated in Figure D-1. In these steels, the carbon is tied up as either Cr_7C_3 or as Cr_{23}C_6 . The higher chromium contents favor formation of the latter, more stable carbide.

A similar effect has been found by Russian investigators⁽⁵⁾, who studied the effect of chromium content on carbon stability in a flowing sodium system at 600 C. As shown in Figure D-2, for chromium contents of less than 5 w/o, carbon depletion occurred, while for chromium contents above 5 w/o, carburization occurred.

Carbon activity can be reduced not only by increasing the chromium, but also by the addition of strong carbide stabilizers to the ferritic steels.

FIGURE D-1. EFFECT OF CHROMIUM CONTENT ON THE ACTIVITY OF CARBON IN THE IRON-CHROMIUM-CARBON SYSTEM AT 1000 C⁽⁴⁾

FIGURE D-2. EFFECT OF CHROMIUM CONTENT ON THE STABILITY OF CARBON IN CHROMIUM STEELS EXPOSED TO FLOWING SODIUM AT 600 C FOR 1000 HR⁽⁵⁾

Figure D-3 illustrates the relative thermodynamic stability of the various carbides. The strongest carbide formers, after chromium, are vanadium, niobium, zirconium, and titanium. Nickel carbide, on the other hand, is less stable than iron carbide, and thus tends to increase carbon activity in the ferritic steels, as does silicon.^(4,5) Consequently, attempts to prevent decarburization of ferritic chromium-molybdenum steels have involved alloying with the strong carbide formers mentioned above and increased alloying with chromium.

Alloy Development Studies

German investigators used diffusion couples to study the behavior of carbon in ferritic chromium-molybdenum steels which were stabilized by various amounts of vanadium, titanium, and niobium.⁽⁶⁾ In their study it was theorized that no carbon or nitrogen transfer would take place if a sufficient amount of stabilizing element was added to tie up all of the carbon and nitrogen in the form of stable compounds. This meant that the following individual alloy additions were required to remove all the free carbon from solution and thereby to prevent decarburization:

$$w/o \text{ Nb} \geq 8 \times w/o \text{ (C + N)}$$

$$w/o \text{ Ti} \geq 4 \times w/o \text{ (C + N)}$$

$$w/o \text{ V} \geq 4 \times w/o \text{ (C + N)}.$$

Excessive additions of the carbide formers are undesirable because of their tendency to embrittle the ferritic alloys. Nickel additions have been found to decrease this tendency.

FIGURE D-3. FREE ENERGY OF FORMATION FOR VARIOUS CARBIDES⁽⁴⁾

FIGURE D-4. CARBON PICKUP IN AUSTENITIC STEEL X8CrNiNb 16 13 VERSUS FREE CONTENT OF NIOBIUM, TITANIUM, AND VANADIUM FERRITIC ALLOYS AFTER TREATMENT FOR 500 HR AT 700 °C⁽⁶⁾

Figure D-4 illustrates the results of this study; namely, that niobium and titanium additions prevent decarburization while the addition of vanadium does not. It was found that even a vanadium addition of 2.63 w/o was insufficient to prevent significant decarburization.

In a companion study by German investigators, carbon transfer from a variety of ferritic steels was investigated in static sodium in the presence of austenitic stainless steel at 620 C.⁽⁴⁾ Table D-2 lists the compositions of the ferritic and austenitic stainless steels that were utilized in the study. Figure D-5 shows the change in carbon content that occurred in the austenitic stainless steels as a result of exposure in sodium to the various ferritic steels. The results confirm previous observations that niobium-stabilized ferritic alloys and ferritic alloy steels with high chromium (11.8 w/o) contents are the most resistant to decarburization by austenitic stainless steels.

A similar study has been conducted by MSAR to evaluate various modified ferritic alloys for possible LMFBR application.⁽⁷⁾ Table D-3 lists the compositions of the alloys which have been investigated. Specimens of these alloys were exposed to static sodium at 650 C in an austenitic stainless steel system. Specimens were removed periodically and tested, after which the carbon content was measured. The change in carbon content as a function of time in the sodium is illustrated in Figure D-6. Significant carbon transfer takes place among the test specimens, thus masking the transfer behavior between the austenitic stainless steel and the various alloys. Two of the alloys, stabilized by titanium-zirconium-vanadium (Alloy 3) and by niobium-tantalum (Alloy 4) additions, appear to be relatively stable with respect to carbon transfer. Alloy 4 is a commercial alloy resulting from the previously described German studies.

TABLE D-2. COMPOSITION OF STEELS USED

Steel	Designation	Alloy Content, w/o							
		C**)	C	Cr	Ni	Mo	Mn	Si	Special
1	10CrMo 9 10	0.076	0.10	2.25	-	1.0	0.5	0.3	
2	10CrMoV 9 10	0.12	0.10	2.25	-	1.0	0.5	0.3	0.3 V
3	10CrMoNb 9 10	0.12	0.10	2.25	-	1.0	0.5	0.3	Nb 10 X w/o C
4	20CrMo (W) V 121	0.22	0.20	11.8	0.5	1.0	0.5	0.3	0.3 V
5	X8CrNiNb 16 13	0.057	0.08	16	13	-	1.2	0.5	Nb 10 X w/o C
6	X10CrNiTi 18 9	0.076	0.10	18	9	-	1.2	0.5	Ti 5 X w/o C

FIGURE D-5. CARBURIZATION OF AUSTENITIC STAINLESS STEELS BY VARIOUS FERRITIC STEELS AT 620 C IN CONTACT WITH SODIUM(4)

FIGURE D-6. CARBON VARIANCE AFTER 1000, 3000, AND 4000 HR AT TEMPERATURE IN 650 C (1200 F) RETORT SODIUM (STATIC CONDITIONS)(7)

Number in parentheses refers to compositions as given in Table D-3.

TABLE D-3. COMPOSITION OF CHROMIUM-MOLYBDENUM STEELS
WHICH WERE TESTED IN A STAINLESS STEEL
SODIUM LOOP AT 650 C⁽⁷⁾

Alloy	Alloy Content, w/o						
	Cr	Mo	V	Ti	Zr	Ni	Nb/Ta
1	2.25	1.0	--	--	--	--	--
2	2.25	1.0	0.6	--	--	--	--
3	2.25	1.0	0.2	0.2	0.14	--	--
4	2.25	1.0	--	--	--	0.6	0.56
5	5	0.5	--	--	--	--	--
6	5	0.5	--	0.5	--	--	--

TABLE D-4. ALLOYS USED TO MAKE DIFFUSION COUPLES
WITH TYPE 316 STAINLESS STEEL⁽⁸⁾

Cr	Alloy Content, w/o				
	Mo	Nb	Ti	V	W
2.25	0.9	--	--	--	--
2.25	1.0	0.58	--	--	--
2.25	1.0	0.99	--	--	--
2.25	1.0	1.42	--	--	--
2.25	1.0	--	0.36	--	--
2.25	1.0	0.48	0.10	--	--
2.25	1.0	0.84	--	0.07	--
2.25	1.0	--	--	--	1.50
2.25	1.0	0.47	--	--	0.62

In a recent Japanese study, involving the use of diffusion couples, specimens of 2.25 Cr-1 Mo, containing various amounts of niobium, titanium, and tungsten were explosively joined to Type 316 stainless steel and then heated at 600 and 700 C.⁽⁸⁾ Compositions of the alloys investigated are given in Table D-4. It was found that the specimens containing 1.0 and 1.4 w/o niobium did not undergo any decarburization and that specimens of these same alloys exhibited large improvements in stress-rupture life at 550 C as compared with unstabilized 2.25 Cr-1 Mo alloy.

In a study at Atomics International, a number of modified chromium-molybdenum steels were exposed in a dynamic sodium loop which was made of austenitic stainless steel.⁽⁹⁾ The surface-area ratio of the austenitic stainless steel to the ferritic alloys was about 1000 to 1. The loop conditions included a temperature of 650 C and sodium with an oxygen impurity of 60 ppm and a flow rate of 0.13 ft per sec. The alloys investigated are described in Table D-5. After 1500 to 3000 hr in the loop it was found that those alloys containing the greatest amounts of titanium and niobium, in combination, were the most stable with respect to carbon depletion. The carbon content of the 2.25 Cr-1 Mo-0.47 Ti-0.4 Nb and 2.25 Cr-1 Mo-0.40 Ti-0.42 Nb-1.3 Ni alloys did not change while 2.25Cr-2 Mo-0.4 Ti-0.44 Nb and 5 Cr-1 Mo-0.38 Ti-0.46 Nb alloys increased in carbon content. All the other alloys underwent carbon losses. The relative magnitudes of the carbon transfer also are given in Table D-5.

Russian investigators have studied the stability of carbon in a large number of ferritic steels of the compositions shown in Table D-6.⁽¹⁰⁾ These alloys incorporate carbide stabilizers such as vanadium, niobium, and tungsten

TABLE D-5. CARBON TRANSFER IN MODIFIED CHROMIUM-MOLYBDENUM STEELS WHICH WERE TESTED IN A STAINLESS STEEL SODIUM LOOP⁽⁹⁾

Alloy Content, w/o						Time, hr	Total Carbon ^(a) Change, percent	Carbon Change in 1000 Hr, percent
Cr	Mo	Ti	Nb	V	Ni			
2.25	1.0	--	--	--	--	2000	-0.065	-0.032
2.25	1.0	0.3	--	--	--	3000	-0.055	-0.018
2.25	1.0	--	0.37	--	--	3000	-0.075	-0.025
2.25	1.0	--	--	0.6	--	3000	-0.045	-0.015
2.25	1.0	--	0.1	0.6	--	2000	-0.035	-0.017
2.25	1.0	0.35	0.12	--	--	3000	-0.02	-0.007
2.25	1.0	--	0.73	--	--	2000	-0.015	-0.007
2.25	1.0	0.47	0.4	--	--	3000	--	--
2.25	1.0	0.4	0.42	--	1.3	2000	--	--
2.25	2.0	0.4	0.44	--	--	2000	+0.01	+0.005
5	1.0	0.38	0.46	--	--	2000	+0.025	+0.012

(a) Original carbon contents ranged from 0.10 to 0.12 w/o.

TABLE D-6. CHEMICAL COMPOSITION OF PEARLITIC
AND CHROMIUM STEELS⁽¹⁰⁾

Steel	Element Content, w/o						
	C	Si	Cr	Mo	V	Nb	W
12KhMF	0.11	0.20	1.03	0.25	0.20	--	--
EP-79	0.16	0.83	1.10	0.36	0.22	0.33	--
EP-78	0.13	0.52	1.16	0.50	0.20	--	--
1Kh1M1FS	0.15	0.52	1.20	1.03	0.26	--	--
12Kh2MFSR	0.14	0.53	1.76	0.56	0.24	--	--
1Kh2M2F	0.14	0.95	2.19	1.54	0.32	--	--
1Kh2M	0.15	0.08	2.25	0.99	--	--	--
EI-531	0.17	0.57	2.51	0.65	0.31	0.64	--
1Kh5MS	0.14	1.54	6.32	0.70	--	--	--
EI-802	0.14	0.10	11.6	0.67	0.42	0.30	0.30
EI-993	0.20	0.38	12.4	0.52	0.31	0.52	0.59
EP-38	0.17	1.33	13.6	0.78	0.41	0.37	--

as well as high chromium additions. Tests of these alloys were performed at 500, 600, and 700 C in sodium and with an oxygen content of 20 to 50 ppm and a flow rate of 15 to 40 ft per sec. Table D-7 compares the change in carbon content of the various steels for exposure periods ranging from 2000 to 6500 hr.

The steels that showed no or only slight carbon losses were EI-531 and EI-802, both of which had significant carbide stabilizer additions. Comparison of the test results for EI-993 and EI-802 (which have about the same composition except for carbon content) suggests that excess carbon of EI-993 (from 0.20 to 0.14 w/o) is lost until an equilibrium value is reached, after which no further carbon loss occurs. The loss of carbon from Alloy EP-38 is unexplained although carbide destabilization due to the large silicon addition may be responsible.

In another Russian study the carbon loss from various ferritic steels in flowing sodium at 500 C was determined.⁽⁵⁾ The sodium velocity was 3 ft per sec and its oxygen content was less than 30 ppm. Results of this study, summarized in Figure D-7, indicate that the steels containing niobium and titanium getter carbon from the unstabilized ferritic steels.

In this same study⁽⁵⁾ it was observed that the rate of decarburization increased with stress. This effect is shown in Figure D-8 for specimens exposed to 550 C sodium for 500 hr at stress levels of 0, 5, and 7 kg per mm². The depth of depletion is seen to increase with stress.

In a Czechoslovakian study⁽¹¹⁾, specimens of chromium-molybdenum steels were placed in flowing sodium loops at 500 and 600 C for 1000 hr; the oxygen content of the sodium used in these tests varied from 3 to 280 ppm. Specimens of both 20 and 40 mils thicknesses were exposed in the loops. It

TABLE D-7. DECARBURIZATION OF PEARLITIC AND CHROMIUM
STEELS IN A FLOW^(a) OF SODIUM⁽¹⁰⁾

Steel	Test Time, hr	Starting	Carbon Content, w/o After Holding in Sodium at the Temperature Shown		
			500 C	600 C	700 C
12KhMF	2000	0.11	0.06	0.03	--
EP-79	2000	0.16	0.1	0.09	--
EP-78	2000	0.13	0.06	0.05	--
1Kh1M1FS	2000	0.15	0.06	0.09	--
12Kh2MFSR	2000	0.14	0.08	0.02	--
1Kh2M2F	2000	0.14	0.1	0.1	--
1Kh2M	1800	0.15	0.1	0.06	--
EI-531	1800	0.17	0.17	0.14	--
1Kh5MS	2000	0.14	0.14	--	--
EI-802	6500	0.14	--	0.13	0.13
EI-993	6500	0.2	--	--	0.14
EP-38	6500	0.17	--	--	0.05

(a) Flow velocity: 5 m per sec.

FIGURE D-7. VARIATION OF CARBON CONTENT IN THE STEELS AS A FUNCTION OF TIME IN FLOWING SODIUM AT 500 C⁽⁵⁾

(1) 2 Kh 2 1.9 Cr-0.15 C	(4) 2 Kh 2F 2.14 Cr-0.92 V-0.16 Si-0.14 C
(2) 2 Kh 2M 1.8 Cr - 0.89 Mo-0.13 C	(5) 2 Kh 2T 2.00 Cr-0.99 Ti-0.11 Si-0.18 C
(3) 2 Kh 2V 2.20 Cr-0.72 W-0.12 C	(6) 2 Kh 2B 2.15 Cr-0.60 Nb-0.50 Si-0.12 C

FIGURE D-8. EFFECT OF STRESS ON DECARBURIZATION OF 2.25 Cr-1.0 Mo STEEL IN 550 C SODIUM AFTER 500 HR⁽⁵⁾

Stresses: Curve 1 - 0 kg/mm²
 Curve 2 - 5 kg/mm²
 Curve 3 - 7 kg/mm²

was found that 1 Cr-3 Mo, 1 Cr-3 Mo-0.1 Nb, and 1 Cr-0.2 V alloys were decarburized both at 500 and 600 C, whereas 1 Cr-2 Mo alloys with additions of either 0.6 w/o niobium or 0.5 w/o titanium did not lose carbon at either temperature. Decarburization, when it occurred, was not accelerated with an increase in oxygen content above 3 ppm.

This latter observation confirms a British reference⁽¹²⁾ to an independence between rate of decarburization and sodium oxygen content. However, it conflicts with the conclusions of Russian workers⁽¹³⁾ who, based on tests involving sodium with 50, 500, and 1000 ppm oxygen, reported a decarburization dependence on oxygen content for low-alloy, ferritic steels.

Thermodynamic Analysis

A study was performed at Argonne National Laboratory in which a thermodynamic approach was applied to the prediction of carbon transfer in systems involving sodium and austenitic and ferritic steels.⁽¹⁴⁾ In this study the thermodynamic activity and solubility of carbon in iron-chromium-nickel alloys were calculated for the temperature range of 500 to 800 C by the extrapolation of available higher-temperature data for the iron-chromium-carbon and iron-nickel-carbon systems. The results were used in conjunction with phase-equilibria information and selected data for the solubility of carbon in sodium to calculate the total carbon content in iron-chromium-nickel alloys (carbon in solution plus carbon in the form of carbide) in equilibrium with carbon concentrations in sodium. The results of such a calculation are shown in Figure D-9 for an iron-18 w/o chromium-8 w/o nickel alloy. The calculations show that iron-chromium-nickel alloys with varying nominal initial carbon contents may undergo carburization or decarburization depending upon the carbon concentration in sodium and upon the temperature.

FIGURE D-9. EFFECT OF CARBON LEVEL IN SODIUM ON CARBON CONTENT OF Fe-18Cr-8Ni ALLOY

FIGURE D-10. EQUILIBRIA BETWEEN CARBON-SATURATED Fe-Cr-8Ni ALLOYS AND SODIUM CONTAINING CARBON

FIGURE D-11. TEMPERATURE DEPENDENCE OF EQUILIBRIUM IN SYSTEM INVOLVING CARBON IN SODIUM AND IN FERRITIC AND AUSTENITIC STEELS

The results of calculations of the equilibrium between carbon in sodium and in carbon-saturated iron-chromium-8 w/o nickel alloys are shown in Figure D-10. Concentrations of carbon in sodium higher than those defined by the curves shown for the various contents will result in the formation of carbide ($M_{23}C_6$ type) in the steels. Significantly, the carbon concentrations in sodium at which carbide formation in the austenitic alloys is predicted to proceed are much lower (i.e., about 1 ppm or less at 650 C [1200 F]) than the maximum carbon content specified for LMFBR sodium (20 ppm max). The effect of surface chromium depletion due to mass transfer on the potential for carburization of the austenitic alloys also is illustrated by this figure as is the effect of carbide formation which will deplete chromium from solution in the alloy matrix; both decrease the carburization potential.

The driving force for carbon transfer from a ferritic steel to sodium and to an austenitic steel under isothermal conditions also was calculated with the results illustrated in Figure D-11. The equilibrium relationship between carbon concentrations in sodium and in an iron-2.25 w/o chromium-1 w/o molybdenum alloy is shown along with the carbon concentrations in sodium in equilibrium with an iron-18 w/o chromium-8 w/o nickel alloy. The curves predict carbon loss from a ferritic 0.1 w/o carbon alloy to the austenitic alloy at all temperatures above about 525 C, with the driving force increasing as the temperature is increased. The driving force for carbon transfer in a nonisothermal system, with the austenitic steel at a higher temperature than the ferritic steel, is predicted to be less than in an isothermal system from these curves.

The results of this study must be considered qualitative in nature because of the limitations of available experimental data and the assumptions necessitated by these limitations. However, they are valuable in defining

experimental needs for the application of thermodynamic and kinetic data to the problem of predicting carbon transport in steel-sodium systems.

Fabrication and Welding of Chromium-Molybdenum Steels

The ferritic chromium-molybdenum steels can be formed by both cold and hot conventional working processes. The hot-working range is 1500 to 1800 F. Tube fabrication is normally a hot-working operation except for final finishing and sizing procedures.

The chromium-molybdenum alloy steels are readily welded by most welding processes, including shielded metal arc, submerged arc, gas-shielded arc, atomic hydrogen, and electroslag.⁽¹⁵⁾ In welding, it is necessary to preheat the material before welding, and to prevent the material from cooling below that temperature until the final postwelding heat treatment has been completed in order to prevent the cracking that results from martensite formation. As the alloy content is increased, higher preheating temperatures are required. Table D-8 lists the recommended preheating temperatures for the various chromium-molybdenum steels. If a low-carbon base or weld metal is utilized, then lower preheating temperatures are permissible. For example, if a weld metal contains less than 0.05 w/o carbon, then the preheating temperature may be lowered 100 F from those shown.

In general, the weld metal that is selected has the same chemical composition as the parent material. With all the weld metals, attempts are generally made to minimize the carbon content. Minor repair work has been successfully performed on chromium-molybdenum steels by using austenitic stainless steel as the weld metal. However, the use of austenitic stainless steel is not recommended for welds which are subjected to cyclic temperature

TABLE D-8. RECOMMENDED MINIMUM PREHEAT TEMPERATURES FOR
WELDING CHROMIUM-MOLYBDENUM STEELS⁽¹⁵⁾

Cr-Mo Steel	Minimum Preheat Temperature for Thickness Shown, F		
	To 1/2 In.	1/2 to	Over 2-1/4 In.
		2-1/4 In.	
1/2Cr-1/2Mo	70	200	300
1Cr-1/2Mo	250	300	300
1-1/4Cr-1/2Mo	250	300	300
2Cr-1/2Mo	300	300	300
2-1/4Cr-1Mo	300	300	300
3Cr-1Mo	300	300	300
5Cr-1/2Mo	300	300	300
5Cr-1/2MoSi	300	300	300
5Cr-1/2MoTi	300	300	300
7Cr-1/2Mo	400	400	400
9Cr-1Mo	400	400	400

service because of the large difference in thermal expansion between the ferritic and austenitic steels. Recommended postwelding heat-treatment temperatures for the various chromium-molybdenum steels are given in Table D-9. As with the preheating temperature, the heat-treatment temperature increases with increasing alloying additions.

Methods of welding stabilized chromium-molybdenum steels are not well defined in this country but appear to be successfully applied in Europe. Recommendations for the welding of both the unstabilized and niobium-stabilized forms of 2.25 Cr-1 Mo are similar.⁽¹⁶⁾ Compositions appear in Table D-10. Preheating uniformly to between 200 and 300 C (390 and 570 F) is necessary. Welding is accomplished with weld rod of a composition similar to that of the base material. However, nickel is not included in the niobium-stabilized weld rod because its addition causes embrittlement. Manual-electrode, TIG, and submerged-arc processes can be applied. The best ductility in the weld is achieved in manual welding if very thin electrodes (maximum, 4.0 mm) are used and a great number of passes involving thin layers are made. Where the seam roots cannot be welded together, TIG welding is recommended while blanketing the reverse side with argon or forming gas. Submerged-arc welding can be accomplished using both stabilized and unstabilized weld rod with the requirement that an anneal and temper treatment and a simple temper treatment are required, respectively, for the two types of rods. Heat treatment is a basic requirement for all welds. Annealing temperatures are 930 to 960 C (1705 to 1760 F) and 970 to 1030 C (1780 to 1885 F), respectively, for the unstabilized and stabilized materials. Corresponding tempering temperatures are 730 to 780 C (1345 to 1435 F) and 680 to 750 C (1255 to 1380 F). Air cooling follows both

TABLE D-9. RECOMMENDED POSTWELD HEAT-TREATMENT
TEMPERATURE RANGES FOR CHROMIUM-
MOLYBDENUM STEELS⁽¹⁵⁾

Alloy Type	Temperature Range, F
1/2Cr-1/2Mo	1100-1300
1Cr-1/2Mo	1200-1350
1-1/4Cr-1/2Mo	1250-1375
2-1/4Cr-1Mo	1250-1400
3Cr-1Mo	1250-1400
5Cr-1/2Mo	1300-1400
5Cr-1/2MoSi	1300-1400
5Cr-1/2MoTi	1300-1400
7Cr-1/2Mo	1325-1400
9Cr-1Mo	1325-1400

TABLE D-10. CHEMICAL COMPOSITION OF UNSTABILIZED
AND STABILIZED 2.25 Cr-1 Mo STEELS⁽¹⁶⁾

Designation	Alloy Content, w/o						Nb
	C	Si	Mn	Cr	Mo	Ni	
10 Cr Mo 9 10	0.15	0.15-0.50	0.40-0.60	2.0-2.5	0.4-1.1	--	--
10 Cr Mo Ni Nb 9 10	0.10	0.15-0.50	0.40-0.80	2.0-2.5	0.9-1.1	0.3-0.8	10 x %C

the anneal and temper treatments. If only tempering is required, a minimum anneal of 1/2 hr between 650 and 750 C is recommended with the time increased if temperatures at the lower end of this range are used.

Conclusions

Decarburization of ferritic chromium-molybdenum steels in sodium systems constructed of or containing austenitic stainless steels is of sufficient concern to have prompted studies in a number of countries aimed at preventing or minimizing this occurrence. The principal approach to carbon stabilization has involved alloying to decrease carbon activity in the ferritic steel. From the experimental results that have been reported it appears that optimum carbon stabilization is obtained by additions of chromium, titanium, and/or niobium. Only minor improvement in decarburization resistance is obtained by vanadium and tungsten additions.

Depending upon the total carbon content of a system a particular alloy can either gain or lose carbon.^(2,3,7,9,10) However, it appears that the carbon content of an alloy will tend to approach some quasi-equilibrium level with time⁽¹⁰⁾, the level being determined by its alloy content and the total carbon in the system.

The effect of oxygen content on decarburization rates is uncertain, being subject to conflicting reports.⁽¹¹⁻¹³⁾ However, accelerated decarburization with increasing levels of stress has been observed.⁽⁵⁾

Predictions of carbon transfer in sodium systems involving austenitic and ferritic steels have been made using available thermodynamic and carbon-solubility data. An increase in carburization potential for austenitic by ferritic steels with increasing temperature and a decrease in

potential when the austenitic steel is at a higher temperature than the ferritic steel is calculated.

Tubing fabrication and welding procedures for the unstabilized ferritic chromium-molybdenum steels are well developed. However, the stabilized grades present special problems in welding which appear to have been overcome in Europe by careful control of weld preheating and postheating procedures.

References

- (1) Wever, H., and Flender, H., "Die Bergauf-Diffusion des Kohlenstoffs in ein-und Mehrphasigen Eisen-Kohlenstoff-Legierungen", Arch. Eisenhuettenw., 35, 65 (1964).
- (2) Rowland, M. C., Plumlee, D. E., and Young, R. S., "Sodium Mass Transfer: XV, Behavior of Selected Steels Exposed in Flowing Sodium Test Loops", GEAP-4831 (March, 1965).
- (3) Hayes, W. C., and Shepard, O. C., "Corrosion and Decarburization of the Ferritic Chromium-Molybdenum Steels in Sodium Coolant Systems", NAA-SR-2973 (1958).
- (4) Motz, J., "Untersuchung Einiger Warmfester Ferritischer Stahle auf Ihre Verwendbarkeit in Natriumanlagen bis zur Temperaturen von Etwa 600 C", Z. Metallk., 55, 489 (1964).
- (5) Bykov, V. N., et al., "The Problems of Corrosion Resistance of Structural Materials in Sodium", CEMA Symposium on Atomic Power Stations With Fast Reactors, Vol I, JPSR-48330 (July 1, 1969), pp 426-461.
- (6) Baerlecken, E., Lorenz, D., Krauz, E., and Schlegel, D., "Development of a Low-Alloy Ferritic Steel for High Temperature Sodium Service With Special Regard to Carbon Transport", Alkali Metal Coolants, Proceedings of a Symposium, Vienna, 25 November - 2 December, 1966, IAEA, Vienna (1967), p 145.
- (7) Andrews, R. C., and Barker, K. R., "Topical Report No. 14, MSAR Evaluation of Candidate Ferritic Alloys for LMFBR Service", MSAR-69-139 (July, 1969).
- (8) Uamamoto, S., and Onishi, T., "Study on Steel Tubes for Steam Generator of Fast Breeder Reactor", Tetsu To Hagane, 54, 284 (1968).
- (9) Anderson, W. J., Sheffield, G. S., and Berkle, A. J., "Development of Ferritic Steels for High Temperature Sodium Service", Part II, NAA-SR-7544 (November 30, 1963).
- (10) Zotov, V. V., Nevzorov, B. A., and Umnyashkin, Ye. V., "Corrosion Resistance of Construction Materials to Sodium", Liquid Metals NASA-TT-F-522 (1969), pp 300-317.
- (11) Ilinchev, G., "The Effect of Oxygen Content in Sodium on the Corrosion and Mass Transfer of the Ferritic and Some Stainless Steels", CEMA Symposium on Atomic Power Stations With Fast Reactors, Vol I, JPSR-48330 (July 1, 1969), pp 462-477.

- (12) Brown, A. F., and Hayes, D. J., "Observations on the Decarburization and Graphitization of Steels by Sodium", J. Nucl. Mater., 27, 187-193 (1968).
- (13) Lyashenko, V. S., and Nerzorov, B. A., "The Mechanism of Carbon Transfer in Liquid Sodium", Conference On Corrosion of Reactor Materials, Salzburg, June 4-9, 1962, IAEA, AEC-TR-5408 (1962).
- (14) Natesan, K., and Kassner, T. F., "Calculations of the Thermodynamic Driving Force for Carbon Transport in Sodium-Steel Systems", ANL-7646 (December, 1969).
- (15) "Welding of Chromium-Molybdenum Steels", Welding Handbook, Section 4, American Welding Society (1966), pp 63.40-63.47.
- (16) Herberg, G., "Welding of the High Temperature Ferritic Steels 10 Cr Mo 910 and 10 Cr Mo Ni Nb 910", Meeting on Reactors, Deutschen Atomforum e. V., Frankfurt/Main, April 14-18, 1969.

APPENDIX E

MASS TRANSFER AND CORROSION IN SODIUM

APPENDIX E

MASS TRANSFER AND CORROSION IN SODIUM

Mass transfer in flowing sodium systems occurs as material dissolved in one region is deposited in another. Two mechanisms exist for this transfer. In a nonisothermal system most elements are more soluble in sodium at elevated than at low temperature. Material dissolved in high-temperature regions will tend to precipitate in low-temperature regions when the solubility is exceeded. Transfer also can occur as a result of the chemical-potential differences that develop as a result of the temperature differences. Either may be sufficient to produce mass transfer in a monometallic system and also may be active in a bimetallic system. However, in a bimetallic system differences in chemical potential or activity of a given element in the different metallic or alloy species provide an additional and, quite often, predominant mechanism for mass transfer.

The importance of each mechanism in the transfer of specific elements is not completely understood. However, it appears that in the transfer of metallic elements, such as chromium and nickel, solubility differences in sodium at different temperatures, perhaps related to complexes formed with other impurities in the sodium, play the major role in transfer. For carbon, on the other hand, differences in chemical potential appear to be of greatest significance, although complexes may be involved.

Mass transfer can have serious effects, both in the region of removal as well as in the region of deposition, which must be taken into

account in component design. In the removal region, the simple material loss appears as a part of the corrosion allowance. The much more pronounced preferential depletion of individual components, proceeding from the surface and extending into the interior of a component material, can produce significant changes in behavior of the materials not only because of specific chemical composition changes, but also because of associated phase changes. Carbon depletion will occur to much greater depths than will depletion depths for the metallic elements because of its much greater diffusivity. It is for this reason, and because of the large effects of carbon content on mechanical properties, that carbon transfer is of greatest concern to the performance of component materials.

Carbon deposition as opposed to depletion, leading to surface carburization is also of concern because of the resulting changes in materials characteristics. The embrittlement, which accompanies carburization, can be severe. In addition to property changes, mass-transfer deposits are undesirable for other reasons. They tend to be of low density and when deposited on tubing, as in the IHX or steam generator, impair heat transfer. There is also a tendency for deposition to occur in areas with low sodium flow rates, as at valve seats, which can result in unsatisfactory performance.

For these reasons it is desirable either to minimize mass transfer or else to be in a position to predict where changes will occur, the extent of depletion or deposition, and the effect on system performance and reliability.

A knowledge of corrosion behavior is necessary to permit specification of corrosion allowances for surface-material loss and degradation

that may occur over the life of a component. Since this concern overlaps that due to mass transfer and because experiments or procedures often provide information on both mass transfer and corrosion, the two problems are treated together. In fact, it is difficult at times to completely separate the two processes.

Much of the required information can be developed by loop studies with careful control of system conditions. Conditions of importance which will influence mass transfer and corrosion rates include temperature, temperature differences and gradients, sodium purity and flow rates, and relative surface areas of the different materials involved in the system as well as their relative physical positions.

It is recognized that extensive loop studies have been or are being performed and that much useful information relating to corrosion and mass transfer has been developed. However, each loop system is unique, and it is not always possible to directly relate behavior in one loop to another, or to a given reactor system. For this reason, loop tests are required under conditions which will duplicate, insofar as possible, those expected in an LMFBR sodium circuit. In addition to loop studies, supplementary supporting studies (e.g., studies to develop thermodynamic and kinetic data) are needed to provide information for the development of mass-transfer and corrosion models which are independent of a particular loop. The specific studies recommended are detailed below.

Loop Studies

Materials of interest for use in LMFBR systems should be exposed in loop systems under conditions representative of those expected in

operating LMFBR systems. Maximum temperature in the loop should be 1200 F with a temperature difference in the system of 300 F or greater. Sodium purity for most studies should be maintained at ≤ 10 ppm oxygen and 20 ppm carbon although provision for excursion to, and tests at, a level of 30 ppm oxygen should be included.

Materials to be studied should include not only the structural materials of primary interest such as stainless steels, ferritic alloys, and Incolloys but also weld-zone, brazing, and potential bearing materials. During reactor operation a number of bearing surfaces, as in the control-rod drive mechanism, valve seats, and the sleeve-and-journal bearing of a hot-leg pump, will be immersed in sodium at 1200 F. The stability of these bearing surfaces, which may include various carbides as well as nitrided surfaces, is critical to their utilization.

Extensive metallographic, structural, and chemical analyses of specimens exposed in sodium loops are needed in order to characterize surface changes. Regions of material loss or depletion and of deposition must be identified. Surface layers, including any oxide or barrier layers, should be investigated and their rate of formation and change with time should be determined. Elements exhibiting preferential depletion and migration must be identified and the depth and extent to which either depletion or deposition and penetration has occurred must be determined.

Changes in mass-transfer behavior with loop temperature and temperature differences in the loop should be investigated. The influence of sodium flow rates and impurity levels on these surface effects also requires study.

In addition to mass-transfer effects, a number of sodium corrosion effects are either unknown or incompletely defined. A better understanding should be possible through the following:

- (1) Flow-rate studies should be extended to 100 ft per sec or more. This is typical of flow rates encountered in the vicinity of the pump impeller. Present data are limited to a maximum of 40 ft per sec and there is uncertainty in extrapolating to higher flow rates.
- (2) Applied stress is not known to influence corrosion behavior in short times but may in long-time exposure to sodium. Studies should be undertaken to determine whether such an effect occurs.
- (3) Studies should be undertaken to determine the importance of sodium mass and of related downstream effects on metal loss. In addition, a quantitative description of the effect of hydraulic diameter is desired. Such information is required for design purposes in scaling up from loop to reactor dimensions (up to about 48 in.).
- (4) Studies should be performed to evaluate the effect of transient changes in conditions of loop operation on corrosion and mass-transfer rates. Evaluations should include changes in sodium purity, stress level, temperature, and temperature gradient.
- (5) Pitting corrosion has been observed in a General Electric loop. It may have been a unique occurrence. However,

attempts to duplicate this effect and to determine the conditions of occurrence should be made in order to establish whether an allowance for such corrosion must be made in reactor system design.

- (6) The conditions under which erosion and cavitation occur in sodium and the nature and extent of damage produced have been examined but have not been defined completely. Erosion may occur in regions of either jetted or vortex flow. However, in certain cases of apparent erosion it is likely that particulate matter carried by the sodium has been responsible for much of the attack. Cavitation attack is expected to be most severe on the pump impeller and on check valves during a sudden closure. Studies to define the occurrence and severity of both are required.

Supporting Studies

These studies should be aimed at defining the processes involved in mass transfer and in developing new materials or methods for the minimization of mass transfer. The following specific investigations are recommended:

- (1) Further studies of iron, chromium, and nickel solubility in sodium as a function of temperature should be performed, in particular to provide information on the effect of other impurities on solubility. These studies could be extended to some of the minor alloy additions such as

molybdenum, niobium, and titanium which have important strengthening and stability effects in stainless steels.

- (2) Studies to define the solubility of carbon in sodium as a function of temperature should be continued as should studies of the form in which the carbon is present, either as a solute or as a complex of some sort. Included should be determinations of impurity effects; in particular, those of oxygen and chromium which may affect the complexing behavior.
- (3) The influence of sodium oxygen content on decarburization rates and processes should be investigated. Past studies have produced conflicting conclusions. Confirmation of a reported dependence on stress of decarburization rates also should be sought and, if confirmed, a quantitative description of the dependence established. For a description of previous work in these areas see Appendix D.
- (4) Studies to define the chemical potential of carbon in sodium and in alloys of interest for use in LMFBR components should be undertaken with specific consideration of steam-generator tubing and piping materials. Measurements are required as a function of temperature over the range from 800 to 1200 F. These studies should be correlated with analytical attempts to predict alloying effects on carbon potential. Since stainless steels are subject to aging, with associated carbide precipitation and modification,

which can be responsible for changes in carbon activity or chemical potential with time, aging effects and changes in potential with aging time should be included as a variable in these studies.

For these studies a reliable method for the accurate measurement of carbon solubility and/or chemical potential in sodium is required. Studies should be undertaken to develop and evaluate an optimum technique.

- (5) An alloy selection and development effort aimed at improvement of alloys currently being considered for LMFBFR components should be undertaken. The studies described in Item (4) above should serve as the basis for such development.
- (6) Techniques for analyzing and interpreting the variety of corrosion and mass-transfer data should be developed. Models are required which will predict corrosion and mass-transfer behavior independent of a particular loop and, eventually, of a given reactor system. This should be a major eventual objective of all studies performed.
- (7) Diffusivity data are lacking for carbon in the stainless steels of interest for use in components. Such data are required in order to calculate the carbon gradients which develop in these materials with time as well as to estimate accurately rates of carbon loss and absorption. In addition to diffusivity rates, the processes involved in carburization and decarburization are not

precisely known. For instance, precipitated carbides in a solid metal will act as a source of carbon, but only the carbon in solution in the matrix phase is free to diffuse under action of the potential gradient. The distribution of carbon between carbide and matrix phases is largely unknown. Furthermore, aging effects tend to change this distribution as well as the form of the carbides and thereby the potential gradient. Thus detailed studies of the carburization and decarburization processes, both with and without the presence of sodium, are required in order to provide the required information.

- (8) Diffusivity data for nitrogen in the stainless steels of interest are also required in order to calculate possible rates of migration through piping walls and rates of nitrogen depletion from, or possible absorption in, component parts immersed in sodium.

The corrosion and mass-transfer studies described in this section will fill a major design need for LMFBR components. Loop studies can provide many of the needed data with the requirement that tests be performed under conditions that duplicate, insofar as possible, those anticipated in operating LMFBR's. Conditions which must be controlled include temperature, temperature differences and gradients, sodium purity and flow rates, relative surface areas of the different materials involved, and their relative physical position in the system. While there will be a continuing need for loop studies, there are limitations to their use

in examining design variables since each sodium loop circuit is unique and any change in system conditions leads to changes in corrosion and mass-transfer behavior. Consequently, the long-term emphasis should be placed on studies to provide information which can be used to develop sodium corrosion and mass-transfer models which are independent of a particular loop.

SPECIAL DISTRIBUTION

AEC — Division of Reactor Development and Technology — HQ

Assistant Director, Engineering Standards
Assistant Director, Nuclear Safety
Chief, Gas Cooled Projects Branch
Chief, Components Branch (3)
Chief, Fuel Engineering Branch
Chief, Fuels and Materials Branch (3)
Chief, Reactor Vessels Branch
Program Manager, LMFBR Office

AEC — RDT Site Offices

Senior Site Representative
Argonne National Laboratory — CH and ID
Atoms International
General Electric Company — Sunnyvale
Gulf General Atomic
Oak Ridge National Laboratory
Westinghouse — Madison
Assistant Director for Pacific Northwest Programs, RDT

AEC — Division of Naval Reactors, HQ

Steele, R H

AEC — Division of Research, HQ

Stevens, D K

AEC — Division of Space Nuclear Systems, HQ

Powers, J A

Aerojet General

Murray

Argonne National Laboratory

Reuther, W
Li, C Y
Nevitt, M V
Kassner, T
Shewmon, P

LMFBR Program Office — ANL

Kelman, L R
Davis, S

APDA

Jens, W

Atoms International

McDonald, J
Pearlman, H
Shively, J H
Ash, E

Babcock and Wilcox

Kovach, P (Alliance, Ohio)
Baroch, C (Lynchburg, Virginia)
Rowe, J P (Alliance, Ohio)
Probert, Paul B (Barberton, Ohio)

Battelle Memorial Institute — Pacific Northwest Laboratory

Bement, A L
Claudson, T T
Knecht, R L
Marshall, R P
Tobin, J C
Finch, L

Brookhaven National Laboratory

Gurinsky, D H

Byron Jackson (Vernon, L A)

Hoppel, R

Cimax Molybdenum Corporation

Semchyshen, M

Combustion Engineering, Inc

Chernock, W P
Muenchow, H O (Windsor, Conn)

Crane Company

Murphy

Foster Wheeler Corporation

Wolowodink, W

Gulf General Atomic, Inc.

Howe J P
Hull, L J

General Electric Company — Breeder Reactor Development Operation

Zebroski, E L

General Electric Company — Nuclear Fuels Department

Richards, R B

General Electric, Knolls Atomic Power Laboratory

Wojcieszak, R F

IIT Research Institute

Parkh, N

Iowa State University

Carlson, O N

Los Alamos Scientific Laboratory

Baker, R D
Hall, D B

Massachusetts Institute of Technology

Pelloux, R M N
Backofen, W A

MSA Research Corporation

Andrews R C

NASA — Lewis Research Center

Manson, S S

Naval Air Systems Command

Machlin, I

Naval Research Laboratory

Lange E A
Steele, L E

Naval Ship Research and Development Center

Gross, M R

Oak Ridge National Laboratory

Slaughter, G
Martin, W R
McCoy, H E
McHargue, C J
Weir, J R
Greenstreet, W L
DeVan, J

The Ohio State University

Spretnak, J W

Rockwell Manufacturing

Mellville, B
Douty, R A

Sandia Laboratory (Albuquerque)

Jacob J

Stanford University

Sherby, O

Teledyne (Boston)

Cooper, W

United Nuclear Corporation

Goldman, K

University of California, Lawrence Radiation Laboratory

Brewer, L (Berkeley)
Roberts, L W (Livermore)

University of California Los Angeles

Tetelman A S

Westinghouse Electric Corporation, Advanced Reactor Division

Hobson, R G
Ray, W E
Gunson, W
Beckett, E F

Westinghouse Electric Corporation, Astronuclear Laboratory

Goldberg, D C

Westinghouse Electric Corporation, Bettis Atomic Power Laboratory

Fillnow, R H

C. F. Braun & Company

Detman, Roger F (Alhambra, California)

Liquid Metal Engineering Center (AI)

Dickinson, R W, Director
Droher, Joseph LM Information Center