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EXPERIENCE WITH CONTROL ROD SEALS USED ON SODIUM-COOLED REACTORS

N. R. Gordon

June 1970

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By

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ABSTRACT

A survey was made to determine the state-of-the-technology on elastomeric reactor seals to provide a better background for design of FFTF seals. The survey revealed only two applications of elastomeric seals within a reactor. Information on silicone rubber and ethylene propylene rubber indicate that they should be considered for some of the penetration seals on the FFTF.

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R. Gordon

INTRODUCTION

The V-A concept for control rods in FFTF requires that the drive mechanism housing be removed during the refueling cycle. This requirement dictates a penetration seal which will provide reliable operation after many removal operations. This requirement can be best satisfied by using an elastomer seal.

A survey has been made to determine the state-of-the-technology on elastomeric reactor seals to provide a better background for selection and design of seals for FFTF.

SUMMARY AND CONCLUSIONS

A request for information on control rod seals to the Liquid Metals Information Center resulted in a combined bibliography of 264 references. Abstracts of 211 of these references were reviewed. Nineteen of the references were read in detail. This literature search revealed only two uses of polymers for seals within a reactor. The control rod penetration on the Fermi reactor was sealed with a silicone rubber gasket and the reciprocating extension rod on the same reactor used Teflon as a backup seal.

The similarity of the elastomeric penetration seal used in Fermi and that proposed for FFTF indicate that successful use of the latter is possible. Although we have no proof that the elastomeric seals used in Fermi were successful, we could find no information which indicated failure. Based on this assumed success with Fermi it is reasonable to expect that the proposed elastomeric seals for the FFTF concept would also be successful.

Information on silicone rubber and ethylene propylene indicates that they are able to retain their elastomeric sealing properties in the temperature and radiation environments of the FFTF. No data have been found which demonstrates the compatibility of either materials with sodium. However, in our opinion, pure hydrocarbon materials such as ethylene propylene should exhibit more stability with sodium than those polymers which have oxygen, halogen, or other reactive functional groups.

RECOMMENDATIONS

We recommend that consideration be given to the use of elastomeric seals for penetrations on the FFTF when conditions are such that the temperature is below 400 °F and the gamma radiation level is below 5×10^3 R/hr.

As part of this recommendation we suggest that a research program be initiated to determine the effects of the FFTF control rod and instrument penetration environments on the stress relaxation of silicone rubber and ethylene propylene. This property is the most valuable measure of a material's long-term sealing capability. The proposed study would involve exposing the test materials to an environment of elevated temperature (250 and 400 °F), exposing the test material to sodium vapor and gamma irradiation for varying periods of time, and measuring the material changes in stress relaxation. The evaluation techniques would be based on the time/temperature superposition principle which has become the accepted method for measuring the viscoelastic responses (stress relaxation, creep, etc.) of elastomers. This principle is based on the fact that viscoelastic properties are a function of both time and temperature. By conducting experiments for short periods of time at various temperatures, a composite curve can be constructed which represents the relaxation or creep of the material for long periods of time at any reference temperature. A more complete description of this experimental technique with accompanying references can be found in Appendix D.

The information derived from this proposed study would supplement the experience gained from the Fermi seals to justify using elastomeric seals for FFTF.

DISCUSSION

A request for information on control rod seals and types of materials which have been used for fast reactor applications was submitted to the Liquid Metals Information Center. In response to this request, the following information was received:

- "Summary of Information on Experience with Control Rod Seals Used on Sodium-Cooled Reactors," compiled by J. K. Balkwill, May 1969. This report appears as Appendix A.
- A bibliography of 53 references on reactivity control devices. This bibliography appears as Appendix B.
- Two hundred and twelve computer printouts with bibliographic data and abstracts of documents dealing with seals on liquid metal cooled reactors. (Information taken from these printouts appear as Appendix C.)

The above information was reviewed and nineteen of the references were read in detail. A bibliography of these references is listed at the end of this report.

POLYMER SEALS

A review of the references cited above revealed only two instances where polymeric materials have been used as seals in a liquid metal cooled reactor.(1) On the Fermi reactor, the safety rod drive extension penetrates the rotating shield plug by means of a seal thimble bolted into the hold-down mounting plate. Two silicone rubber quad ring seals are used to seal the mounting plate penetration near the upper end of the thimble. The temperature at this location is 250 °F which will not affect the properties of the rubber. Also, on these safety rods the bellows seal, attached to the thimble, is backed by a chevron-type Teflon seal which operates on the reciprocating shaft. Teflon was selected for this application because of its low coefficient of friction. The temperature in the area of this seal is also 250 °F.

A Fermi safety rod channel mockup was tested in sodium at reactor temperatures for 3566 cycles with no indicated seal problems. Also, no problems with the polymeric seals have been cited during the operation of the reactor. No references were found which described any development programs which led to the selection of silicone rubber and Teflon for these two seal applications.

APPLICATIONS TO THE V-A CONCEPT

The control rod housing seals for the V-A concept are subjected to an atmosphere of argon gas and sodium vapor at a temperature of 250 °F and a radiation level of 100 n/cm^2-sec and 10^3 R/hr gamma. The instrument cover penetration seals have a somewhat higher temperature requirement (400 °F) and neutron flux (1000 n/cm^2-sec). The sealing characteristics of an elastomer are very desirable for these applications. A better and more reliable seal will be provided, and removal of the control rods will not require replacement of the seal.

Manufacturers' data on ethylene propylene (Du Pont) and silicone rubber (Dow Corning) indicate that they are resistant to temperatures of at least 350 °F in air and probably higher in an inert gas.

In a study conducted for Douglas United Nuclear to evaluate seal materials for the Hanford reactors, these two materials were shown to provide the properties desirable for the application, i.e., radiation resistance and resistance to stress relaxation.[2,3] (Copies of these reports are included in Appendix D.) They appear to retain their relaxation properties for longer than 1 year, and a radiation dose of 5×10^7 R does not significantly affect these properties.

Because of the similarity of the FFTF requirements to those of the Fermi reactor where silicone rubber seals have been used successfully and because of the test results just described, it appears that polymeric seals have potential for FFTF control rods. In light of this potential and the advantages of using an elastomeric material, an evaluation program should be undertaken to determine the reliability of these materials for the desired functions.

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APPENDIX A

SUMMARY OF INFORMATION ON EXPERIENCE WITH CONTROL ROD SEALS USED ON SODIUM-COOLED REACTORS

Compiled By:

J. K. Balkwill
Liquid Metals Information Center
May 1969

SUMMARY OF INFORMATION ON EXPERIENCE WITH CONTROL ROD SEALS USED ON SODIUM-COOLED REACTORS

Compiled By: J. K. Balkwill

INTRODUCTION

A quick overview of the current status of the seal problem for LMFBR control-rod drives is given in the LMFBR Program Plan;[1] pertinent pages are attached. The complete control-rod drive systems for the Dounreay, Fermi, and EBR-II reactors are described by Yevick;[2] his summary table is attached.

A literature survey of mechanical devices with rubbing surfaces operating in liquid metals[3] was published recently by the Liquid Metal Engineering Center. It discusses the characteristics of potentially useful materials that are important in liquid metal applications and describes material compatibility and friction/wear testing programs conducted since 1950. This reference includes information related to seal material problems.

Information on the application of freeze seals for translating rod motion may be obtained from literature concerning valve-stem freeze seals, such as those used on the large throttling and blocking valves in the heat transfer systems at HNPF.[4] These seals exhibited satisfactory operation with the exception that the throttle-valve stem packing had to be replaced occasionally when sodium oxide accumulated on the valve stem in the vicinity of the freeze seal in a sufficient amount to push the packing out of place when the valve was operated. Also there was a slight binding tendency in the blocking valves caused by an accumulation of sodium oxide in the stem freeze seals.[5] As might be expected from this demonstrated tendency to bind, no reference to the use of freeze-type seals for control-rod drive shafts was found.

Detailed information on control-rod seals for specific reactors is discussed below.

EBR-II[6]

In EBR-II, the main shafts of the control-rod grippers operate through labyrinth- or clearance-type seals in the reactor vessel cover. The seals are designed to allow free vertical movement of the shaft with minimum out-leakage of the primary sodium coolant into the upper plenum. Each seal consists of a grooved cylindrical sleeve mounted on the shaft at the proper elevation so that it can slide a distance of 14 in. within a companion sleeve installed in the vessel cover. The cylinder is 12-in. long, with an OD of 2.53 in., and is made of Ampco No. 18-13 alloy (10.6% Al:3.6% Fe; ∿85% Cu). The OD is interrupted by 23 circumferential grooves, each 0.250-in. wide and 0.087-in. deep. The companion sleeve is 29-1/8 in. long, with an ID of 2.56 in., and is made of wrought Stellite No. 6B. The calculated leak rate of each seal is 3.4 gpm of sodium. This rate is based on 100% sodium flow through 67 core subassemblies and a 9.6-psi differential between the upper vessel plenum filled with 900 °F sodium and the 700 °F bulk sodium in the primary tank. The shaft seal has had problems with oxide accumulation and the clearances were increased to assure flushing by sodium flow through the mechanism. Fabrication and operation details of the labyrinth seal and its mating sleeve are given in Table 1.

TABLE 1. EBR-II Control Rod Labyrinth Seal

	Design Information			Operating Conditions			
Component	Material of Construction	Dimensions and Manufacturing Tolerances	Clearances, Loading, and Speed	Operating Medium	Temperature, Velocity, and Pressure	Exposure Environment, hr	Operating Cycles or Hours
Labyrinth Seal -Main Shaft to Reactor Vessel Cover	Aluminum Bronze (Ampco 18-13)	2.535 in./2.531 OD, 12-in. length 23 circumferential grooves 1/4-in. wide 0.076-in. deep	Load: negligible Speed: maximum 8.5 ft/sec Clearance: 0.030 in.	Sodium Liquid	Velocity: 2.27 ft/sec ΔP: 9.6 psi 700 to 845 °F (Neutron flux) 700 °F 300 to 700 °F	≈8,055 ≈27,423 ≈3,942	≈8,470 hr
Reactor Vessel Cover Sleeve	Wrought Stellite 6B	2.559 in./2.567 ID 29-1/8 in. length	Load, negligible; Clearance, 0.030 in.	Same as above	Same as above	Same as above	Same as above

The labyrinth seal was involved in one control-rod-drive operating failure. The Mark-I oscillator-rod drive bearing assembly operated in the region above the sodium pool in argon at about 150 °F. The assembly included a ball bushing that failed after 5800 hr of exposure, allowing the balls to work down the shaft, out the sodium vent holes, and into the adjacent control-rod drive-shaft labyrinth seals. This resulted in the failure of two control rods to drop during a scram. A new drive without ball bushings was subsequently designed.

To prevent primary sodium from entering the main shaft of the control drives, two concentric actuating shafts are welded to stainless-steel bellows which act as internal shaft seals. The actuating shafts are Type 304 SS and the seals are two-ply, semi-welded bellows of Type 347 SS sheet, 0.010-in. thick.

Above the biological shield, the annulus between the main shaft and the guide bearing tube in the rotating plug is terminated by a nesting-type bellows seal. The purpose of the seal is to prevent argon gas leakage from the primary tank without impairing vertical motion of the main shaft. This bellows seal is composed of 198 discs which are welded together at the inner and outer edges. Each disc is made of Type 347 SS and measures 2-9/16-in. ID, 4-1/2-in. OD, and is 0.010 in. thick. During downward travel of the main shaft, the disc portion of the bellows is compressed from 19-7/8 in. to 5-7/8 in. During the control rod "pickup" sequence, the bellows is compressed an additional 7/8 in. to 5 in.

The lower fitting of the bellows is threaded onto the top end of the guide bearing tube, compressing an aluminum gasket to effect a gas-tight seal. Also positioned at the same elevation is a main-shaft guide bearing and Ampco No. 18-13 alloy ring which provides a radial clearance of 0.011 in.

around the main shaft. Sections of the inside diameter of the ring are relieved to permit gas flow in and out of the bellows during control rod strokes and scrams.

The upper fitting of the bellows is threaded to the main shaft, whereupon a gas-tight seal is effected by compressing a graphite-asbestos packing gland. All parts of the seal which fasten to the main shaft can be disengaged to facilitate replacement of the bellows without removing the main shaft and the gripper from the primary tank.

An examination of a control rod which had been in the reactor for about a year showed that the upper and lower guide bushings both made of beryllium copper were severely corroded and/or eroded. This was felt to be significant because the operating environment was roughly the same as for the labyrinth seal. One aluminum-bronze labyrinth seal from a control-rod drive that had been in the reactor about 2 years (see Table 1) showed no indication of wear or dissolution. Plans are in progress to replace these beryllium-copper guide bushings with the same aluminum bronze used for the labyrinth seal.

SRE AND HALLAM

Both the Sodium Reactor Experiment and the Hallam Nuclear Power Facility were designed to have a minimum of operating mechanisms exposed to a sodium environment. Each control-rod assembly operated inside a thimble that excluded reactor sodium and cover gas; a separate helium environment was provided inside the thimble. However, Atomics International has tested a reciprocating-rod seal in 1200 °F sodium for the Sodium Graphite Reactor control-rod development program.[7] The assembly consisted of a 2-in. diameter metal seal used similarly to an O-ring application. In order for the seal to function properly, a film of sodium remained on the control-rod shaft as it passed through the seal. Since the temperature of the seal and shaft was 1200 °F, some vaporization of the sodium occurred followed by condensation in the upper, cooler region of the system. The resulting dress and radioactivity in this upper region negated the otherwise attractive features of this seal. Also the sodium film remaining on the shaft walls could be oxidized even with a low oxygen level in the cover gas and this sodium oxide accumulation eventually could interfere with the operation of the piston. If the seal were continuously immersed in liquid sodium, oxide accumulation would not be a problem if a reasonable sodium purity level were maintained.

FERMI

In the Fermi reactor,[8] the control-rod drive extension penetrates the rotating-shield plug by means of a seal thimble bolted into the holddown mounting plate. Two silicone-rubber seals are used in this low temperature (250 °F) penetration. The extension, which moves through a reciprocating stroke, is sealed by a Type 304 SS self-nesting, welded bellows that does not scram. The bellows is attached to the seal thimble and the rod drive extension and thus isolates the radioactive primary system gas from the atmosphere outside the reactor. A chevron-type Teflon packing seal operating on the reciprocating shaft is used as a back-up seal in case of failure of the bellows seal. Teflon was selected because of the low temperature (250 °F) at the location and because it exhibits the low friction necessary for reliable operation of the extension.

When the control rod was originally designed, the metal bellows appeared to be the only standard sealing method available that was capable of maintaining zero leakage over a long reciprocating stroke (54 in. in this design) while operating for extended time periods in a high-temperature atmosphere containing sodium vapor. However, because the reliability of welded bellows was questionable at that time, the design was to provide for quick replacement of the bellows from the top of the plug without removing the rod drive extension from the reactor. The lower bellows flange connection was therefore bolted to the rod drive extension, but because of difficulties in sealing the bolted flange, it was necessary to seal weld the flange, and the quick-replacement feature was negated. As experience with the bellows has accumulated, their reliability has been found to be very good and the frequency of replacement is expected to be low enough so that removal of the assembly by cutting the seal-weld to remove the bellows can be accepted as standard repair procedure.

Reference 8 also states that the operating control-rod and safety-rod drive mechanisms operated satisfactorily in the reactor for nearly 3 years, and that with the exception of a single lower-latch bellows failure (not to be confused with the reciprocating bellows described above) performance of all bellows seals (approximately 35) was trouble free. The latch bellows that failed was designed to keep the latch-rod cavity leak tight against the entrance of reactor sodium; when the bellows failed, liquid sodium from the pool in the upper plenum entered the latch-rod cavity. When the control-rod extension was raised to the full-up position and held there, for a long enough period, enough sodium froze to interfere with proper rod delatching. As a precaution, after the one failure was discovered, all the latch-rod bellows were replaced with new bellows of the same design. All of the discarded bellows were found to be corrosion-free and leak-tight.

Reference 9 summarizes a literature search on fabrication and operation of bellows in sodium and NaK systems. A tentative specification for procurement of the bellows seal for the FTR control-rod drive is proposed.

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APPENDIX B

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APPENDIX C

BIBLIOGRAPHIC ABSTRACTS OF DOCUMENTS DEALING WITH SEALS ON LIQUID METAL COOLED REACTORS

APPENDIX C

BIBLIOGRAPHIC ABSTRACTS OF DOCUMENTS DEALING WITH SEALS ON LIQUID METAL COOLED REACTORS

Title	Abstract
Electrolytic Removal of Oxygen from Sodium. Technical Status Letter No. 1, December 24, 1964 Through March 31, 1965. **Document No.** MTL-4624 **Author** J. M. McKee	The concept of removing oxygen from sodium by ionic transport through an impermeable wall of solid electrolyte was first explored under contract AF 33(615)-1039 in a 1 yr program ending June 30, 1964. The basic concept feasibility was experimentally confirmed, but the free-standing ceramic cells used were subject to mechanical failure. Under the subject contract, a modified cell design substituting an oxide film on a metallic substrate for the ceramic TJBE will be evaluated. Activities during this first report period were concerned primarily with procuring special alloy elements, modifying the sodium loops to accept metallic cells, and exploring film behavior on a conventional zirconium alloy.

Title

The Design of Totally Enclosed Mechanical Pumps (J. Nuclear Energy)

Document No.

JNUC-1-5/23-54

Author

P. Fortescue

Abstract

This work first analyzes the influence of design factors on performance of sheathed induction motors. The results are presented as a simple general working formulae using the example of a motor for a proposed 50 hp sodium pump. The general conclusion is that motors for this duty should have a large length-to-diameter ratio and should work at lower magnetic loading and higher electrical loading than is normal practice for unsheathed motors. Maintaining an adequate reserve of pull-out torque constitutes a major design limitation in these circumstances. With the design presented, a 6-in. diameter motor, 25-in. long, with a 0.03-in. thick Nichrome sheath should be able to attain 50 hp at 1500 rpm. At an overall efficiency of 80% increasing sheath thickness to 0.045-in. would drop efficiency by about 3%. Using a solid rotor and permitting the liquid metal to penetrate the rotor-sheath gap would involve no serious additional loss. An analysis of the problems of bearing design for use in liquid metals is then made, results being presented in the form of generalized performance charts and a discussion of their implication. The conclusion is that a vertical shaft arrangement with hydrostatic journals and a hydrodynamic thrust bearing offers great advantages. Also shown is a general arrangement drawing which incorporates the previously discussed recommendations for a particular pump design.

Title

Design and Testing of Sodium Pumps for the Hallam Nuclear Power Facility (presented at the Summer Annual Meeting, Los Angeles, California, June 11-15, 1961, of the American Society of Mechanical Engineers)

Document No.

ASME-PAPER-61-SA-39

Authors

R. E. Ball, D. E. Cullman, R. W. Atz

Abstract

The design features of the sodium circulating pumps for the Hallam Nuclear Power Facility are described in detail. Relative costs of various variable-speed drives and gas-sealing devices are discussed. The performance of one pump tested with sodium is presented.

Title

Experience with the Use of Pressure and Level Gages in a Fast-Reactor System (Priborostroenie, No. 1, 23-25, 1965)

Author

A. A. Petrenko

Abstract

Some modifications of previously developed pressure and level gages for Na and NaK coolants to make them suitable for use in a Br-5 fast reactor are described. The temperature range of a rum-type radio-wave level gage was raised from 250 to 350 °C by replacing the gage seal. This modification made it suitable for coolant temperatures up to 500 °C. The reliability of a potentiometer-type level gage was increased by replacing the sectional structure of the sensor and tubes with a simple stainless steel tube, thus simplifying the gage design and preventing the contact of steam with sodium or sodium potassium. The modified pressure gage was found to be suitable for coolant temperatures up to 500 °C. Some future trends in development of instruments for measuring sodium and NaK parameters are discussed.

Title

Bakeable Valve, 12 in. Model II (Engineering Materials)

Document No.

CAPE-1479

Abstract

The valve consists of a large hydraulic cylinder operating through a welded bellows which moves the carrier plate and gate in a simple rectilinear path. Valve closure is accomplished by rollers on ramps near the end of the carrier plate travel. The gate carries a copper disk with the seal surface machined at its outer periphery. Sealing is accomplished against a surface on a mating member external to the valve assembly. A rectangular aluminum foil seal flange permits removal of all valve components for service or replacement without removing the valve from its system.

Title

Alkali Metals Boiling and Condensing Investigations. Quarterly Progress Report II Covering the Period January 1, 1965 Through March 31, 1965.

Document No.

NASA-CR-54405

Authors

F. E. Tippets, G. L. Converse

Abstract

Progress report for quarter ending 3/31/65, in which detail data and analysis are presented. Balance of the boiling test data obtained in the 300 KW Facility during the previous quarter are presented. Boiling heat transfer coefficients and two-phase pressure drop multipliers applicable to boiler design are given. Results show dependence on temperature level, exit quality, mass velocity, and heat flux, co-current or countercurrent flow, and helical or axial flow. Various experiments over a range of parameters were performed in the 100 KW Facility; included were measurement of nucleate heat transfer coefficients and exploration of heated surface oscillations beyond critical heat flux with helical flow. Various condensing tests were performed in the 50 KW Facility.

Title

Alkali Metals Boiling and Condensing Investigations. Quarterly Progress Report 12 Covering the Period April 1, 1965 Through June 30, 1965

Document No.

NASA-CR-54739

Authors

G. L. Converse, F. E. Tippets

Abstract

This is the Progress Report for quarter ending 6/30/56 on program to obtain two-phase heat transfer and fluid flow data for potassium under boiling and condensing conditions. It presents analysis and detailed heat transfer data. Critical heat flux conditions and transition boiling heat transfer coefficients calculated for the 3/4-in. test section and the analytical treatment for same are presented. Boiling data from the 100 KW Facility for various parameters and conditions is presented. The last of the condensing data obtained in the 50 KW Facility have been reduced and are reported. An analysis of the vapor-phase thermal resistance in condensing based on kinetic gas theory to derive the heat transfer coefficient is presented and used for comparison with the data. An analytical treatment of nucleate boiling heat transfer is given, including theoretical prediction of heat transfer coefficients; analysis of relationships between cavity size on heat transfer surface and the wall superheat required to initiate bubble nucleation in potassium; comparison of theoretical predictions with experimental data; and recommendations of design procedure for calculating nucleate boiling heat transfer coefficients.

Title

Pumps for High-Temperature Liquid Systems (Space-Nuclear Conference, May 3-5, 1961, Gatlinburg, Tenn.)(American Rocket Society)

Document No.

ARS-PAPER-1743-61

Authors

H. W. Savage, A. G. Grindell

Abstract

High-temperature pumps have been required for several mobile and stationary nuclear reactor power plant applications using sodium, NaK alloy, or fused salt as coolants. Some of these pumps have operated at temperatures above 1500 °F and some for periods well in excess of 20,000 hr. The incidence of failures has been low. A summary and evaluation of the incentives, courses of development, present status and applicability of the various types of pumps used in the past or now under development indicate design features which may be of interest in developing equipment for space vehicles. The designs and special features of these pumps, primarily centrifugal, will provide a descriptive background of the problems inherent in bearings, seals, lubrication, hydraulic performance, cavitation inception, choices of materials, fabrication procedures, drives, liquid expansion, and size and maintenance considerations.

Title

Testing of Sodium Pipe Joints and Development of Remote Repair Tools for the SDR

Document No.

NDA-84-20

Authors

H. Belofsky, S. Lazarus, B. Minushkin

Abstract

Tools were designed, constructed and tested over working distances approximating full scale reactor dimensions, thus demonstrating the feasibility of remotely repairing and replacing a fuel-coolant tube in an SDR. An expandable mechanical plug was developed to seal the coolant tube to enable repairs. Mechanical joints were tested under operating conditions. Test results are described.

Title

Operating Experience with the Sodium Reactor Experiment (Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy Held in Geneva September 1-13, 1958. Volume 9, Nuclear Power Plants, Part 2)

Document No.

A/Conf. 15/P/452

Authors

F. E. Faris, L. E. Glasgow, D. H. Johnson, R. W. Campbell, J. E. Owens

Abstract

The main conclusions to be drawn from the operation of the SRE indicate that the plant has performed extremely well at levels up to 1/3 of full power. The performance of the reactor and reactor steam plant complex is extremely stable. Excess thermal stresses are not an inherent characteristic of sodium-cooled reactors. With the information gained from the SRE, it should be possible to design a sodium-cooled reactor which is free from these stresses. With the exception of the boiling toluene cold traps, all of the components developed for the SRE have performed well. For large sodium systems, free blow through 1-1/2-in. and smaller lines is determined by the sodium oxide precipitation temperature rather than by the melting point of sodium. Maintaining the oxygen concentration at or below an indicated level of 10 ppm can be easily accomplished under normal operating conditions. Maintenance and modifications of the main primary and secondary sodium systems can be directly and easily accomplished. The shielding for the SRE is more than adequate. In particular, the gap tolerances are more stringent than necessary and economies can be realized in future reactor designs by relaxing them. The gamma heating of concrete structures greatly accelerates the dehydration rate. Reasonably good agreement has been obtained between the calculated and measured critical masses, individual worths of core elements, and

Abstract (contd)

the flux distribution. The slow-acting overall isothermal temperature coefficient for the SRE is positive but small. The fast-acting metal coefficient is negative.

Title

Alkali Metals Boiling and Condensing Investigations Quarterly Progress Report 10 Covering the Period October 1, 1964 Through December 31, 1964.

Document No.

NASA-CR-54308

Authors

F. E. Tippets, G. L. Converse

Abstract

Progress Report No. 10 for quarter ending 12/31/64 in which detail data and analysis are presented. All scheduled testing in the 300 KW Facility was completed - included was operation with two tube sizes, three different insert combinations and with both countercurrent and cocurrent flow. Potassium temperatures exceeding 1700 °F and various exit qualities were obtained. Boiling data over a range of variables, obtained with different size L-605 tubes containing helical inserts, are presented. Primary effort on the 300 KW Project is now directed toward data reduction and analysis with no immediate testing planned. Improvements to the preboiler heater element in the 100 KW Facility were made and boiling operation began. Data with a 3/4-in. test section with no insert are reported. In the 50 KW Facility potassium condensing data obtained with a 5/8-in. ID tube without insert for a range of variable are presented. Comparisons are made of analytical production, and experimental data for condensing pressure agreement is good - the correlation considers both momentum pressure rise and friction pressure drop.

Title

Torus Analysis

Document No.

KAPL-1072

Authors

R. L. Mathews, G. Horvay

Abstract

An analysis is presented of a 60-in.-diameter torus and a continuous, hollow piston ring made of steel tubing with a 3/8-in. tube diameter and a 10-mil wall thickness. It is used to restrict liquid sodium flow in the rotating plugs assembly.

Title

Bore Seal Technology Topical Report. Research and Development Program on Magnetic, Electrical Conductor, Electrical Insulation, and Bore Seal Materials

Document No.

WAED-64.54E

Authors

P. E. Kueser, J. W. Toth, R. C. McRae

Abstract

Thermophysical compatibility, and mechanical property data on ceramic-to-metal seal technology of interest to the design of advanced space electric power systems are presented. It represents a thorough search of the recent world literature on this subject and a bibliographic record on this topic. The application of ceramic-to-metal bore seals to actual designs is described and discussed. The thermophysical and mechanical properties at elevated temperatures of selected ceramic and metallic members are reported. The materials include high-purity aluminum Sub 203, high-purity bed, NB-1ZR alloy, Ta T-111, and NB D-43. Two major methods of joining ceramic to metals were investigated - the metalizing braze method and the direct-bonding, active-metal braze process. The joints produced by the metalizing method were too brittle to ensure an adequate seal. The active-metal braze was used on the representative bore seals. Of the active-metal systems studied for the ceramic-to-metal seal, the 56% Zr, 28% V, 16% Ti system proved to be the most satisfactory. Alkali-metal exposure tests show that for best results, the silica content of the ceramic must be held below 50 parts per million. Static capsule tests were used for the quantitative evaluation at 1000 °F and 1600 °F for compatibility of the bore seal materials and brazed assemblies with very high-purity K, Li, and

Title

Effects of Alkali-Metals on Ceramic-to-Metal Seal Systems (AEC-NASA Liquid Metals Information Meeting, held April 21-23, 1965, Gatlinburg, Tenn.)

Document No.

CONF-650411, PG. 344/358

Author

P. E. Kueser

Abstract (contd)

NaK eutectic alloy. Emphasis was placed upon maintaining very low 0 sub 2 level of the alkali metals throughout the loading procedure.

Abstract

The elevated waste-heat temperature desired for high-powered, space electric power systems has necessitated the selection of potassium, sodium or cesium as the turbine working fluid in rankine cycles. Because the turbine driven alternator contains insulation materials sensitive to corrosive attack by alkali metals, they must be either isolated by a quote bore seal unquote or by shaft seals. A bore seal is used in a radial-gap inductor alternator to protect the stator from alkali-metal vapor. A similar configuration is used in electric motors to protect the stator assembly. The motor also requires a second seal for the rotor assembly. In a lightweight design, high-frequency operation is used which requires a material in the gap with high-electrical resistivity to minimize eddy-current losses. A ceramic material best suits this requirement. To provide a hermetic construction, ceramic-to-metal seals are required. This discussion represents a fundamental investigation on small ceramic-to-metal seals and their constituents to determine construction guidelines.

Title

Some Metallurgical Consideration in the Design and Operation of Liquid-Metal Seals (Transactions American Nuclear Society, 1964 Annual Meeting, Philadelphia, Pa.)

Document No.

TANS-7-190-64

Authors

B. Blumenthal, L. R. Kelman, H. V. Rhude

Abstract

Some metallurgical considerations in the design and operation of the EBR-II rotary seal are described. Article with same title and authors issued as ANL-6837

Title

Design and Operation of Freeze-Seal Valves and Pumps

Document No.

CEPR-52-157-56

Authors

R. Cygan, A. M. Stelle

Abstract

Freeze seals have been developed for sealing stationary and rotating shafts, affording a simple means of adapting commercial equipment for use in high-temperature liquid-metal systems. Several designs for sealing sodium at temperatures up to 1200 °F have been satisfactorily tested. Under different operating conditions cooling loads, torque, and temperature distribution were measured. Locating the freeze seal away from the high-temperature region allows the formation of an annulus of frozen metal around the shaft with small cooling loads.

Title

Test of a 24-inch Diameter Model of the HNPF Cerrobend Seal

Document No.

NAA-SR-MEMO-4682

Author

B. W. Admire

Abstract

Top shield of the HNPF core is to be sealed against leakage of helium cover gas by Cerrobend alloy metal in a tongue-and-groove seal at the outer edge of the shield. Cerrobend is a commercial alloy containing 50% Bi, 27% Pb, 13% Sn, and 10% Cd. This alloy melts at 158 °F. A 24-in. diameter seal model was used. The permissible helium leakage rate was 1.0 × 10 to the minus 5 cm^3/sec. Various tinning alloys and fluxes were evaluated. A tinning process, involving 50-50 solder and a coating of Cerrobend resulted in a leak-tight seal. During and after four melt and freeze tests, the seal remained leak tight. Due to difficulties encountered in tinning the test seal, it is very doubtful that complete wetting of the sealing surfaces can be accomplished at the time the seal is installed at the full size reactor. Tubular heaters placed around periphery of seal were used to melt the seal. Installed heat density for the test rig was 544 W/ft of seal. The reactor installation will have 666 W/ft of seal.

Title

Summary Report "Project Freeze Seal"

Document No.

NAA-SR-MEMO-1565

Author

R. Cygan

Abstract

Designs of five shaft freeze seals for the SRE main sodium pumps are described. Data from tests of these seals are presented.

Title

Design and Performance Characteristics of EBR-II Control Rod Drive Mechanisms

Document No.

ANL-6921

Authors

E. Hutter, G. Giorgis

Abstract

The design and performance of the drive mechanisms for the EBR-II control rods are described. Identical drive mechanisms are provided for each of the 12 control rods. A brief description of associated reactor components and systems is given. Performance test results are presented in tabular and graphical form.

Title

Mark A Control Actuator Shaft Freeze-Seal

Document No.

KAPL-M-EDL-75

Authors

P. W. Bissonnette, R. D. Vallee

Abstract

Investigations were conducted to determine the mechanical properties of sodium sheared in a narrow annulus at controlled temperature gradients and shear velocities. The test fixture consisted of a 1-in. diameter stainless-steel shaft centered in a 1.099-in. ID Type-304 SS tube. Sodium shear values were found to be dependent upon the temperature and shear velocity at a given test condition. Sodium shear values obtained were found to be consistent, wherever wetting of components had been assured by contact with hot sodium at temperatures above 500 °F for periods of 1/2 hr or greater. Tests conducted showed sodium shear to have occurred internally, rather than at freeze-seal interfaces within the range of shaft velocities tested. The apparent shear strength of sodium was found to fall off rapidly with successive shearing cycles to steady-state values corresponding to particular shearing velocities. Tests showed that the freeze-seal at rest, a healing action occurred as a function of time, resulting in substantially complete regaining of initial shear strength.

Title

Components

Document No.

PRET-7/3-269/275-64

Abstract

A description of the coolant circulating pumps used for the lithium-cooled reactor experiment is presented. Performance requirements and the shaft-sealing arrangement are discussed. A throttling valve, equipped with a torque-tube stem seal, is described. The valve was tested in 1000 °F sodium. After being subjected to 500 thermal shocks the valve still performed satisfactorily.

Title

High Temperature Metal Bellows Seals for Aircraft and Missile Accessories

Document No.

JEFI-85-281-63

Author

R. J. Matt

Abstract

Several designs of bellows shaft seals are illustrated and discussed. Problems encountered in service, particularly those associated with bellows fatigue failure, are reported.

Title

HNPF-Process Tube-Grid Seal

Document No.

NAA-SR-MEMO-4855

Author

J. R. Charles

Abstract

Experiments were performed to determine the characteristics of two types of 4-in. diameter piston ring seals and to evaluate their suitability for use as the HNPF process tube grid seal. Requirements included a sodium leakage rate of 0.198% of coolant flow rate through the channel, a 10,000 cycle useful life over a 3-in. stroke in 625 °F sodium, and the ability to withstand 25-cycles over a 3-in. stroke in 1000 °F sodium without damage. These requirements were met. However some corrosion problems were encountered at 1000 °F, which led to a recommendation that the ring seals be nitrided previous to use. Testing methods and apparatus are described.

Title	Abstract
Sodium Vapor Transport in a Closed System	The transport of sodium in a closed system is studied and a labyrinth-type vapor seal is investigated for possibilities of limiting sodium vapor transport. Theory predicts that such a seal will be ineffective because it will collect sodium vapor, the large cross-section does not limit diffusion or evaporation, and the seal will not offer resistance to pressure surges. Sodium Transport Rates (area normalized) given in the body of the report, page 8, seem at variance to rates presented in the Summary on page 3.
Document No. KAPL-M-JJK-1	
Author J. J. Kauslarich	

Title	Abstract
Dynamic Shaft Seals in Space. Quarterly Report No. 9.	A program is being pursued to develop techniques for sealing a high speed rotating shaft under conditions of high temperature liquid metal and vapors, high vacuum environment of space and to provide long seal life. Assembly and installation of the seal test rig was completed and satisfactory operation of the supporting facilities was obtained. Seal testing using light turbine oil was accomplished. Reduction of water test data was completed.
Document No. AD-603655	

Title

Dynamic Shaft Seals in Space. Quarterly Report No. 9.

Document No.

AD-603655

Abstract

A program is being pursued to develop techniques for sealing a high speed rotating shaft under conditions of high temperature liquid metal and vapors, high vacuum environment of space and to provide long seal life. Assemb ly and installation of the seal test rig was completed and satisfactory operation of the supporting facilities was obtained. Seal testing using light turbine oil was accomplished. Reduction of water test data was completed.

Title

SIR Freeze Seal Test II

Document No.

KAPL-M-EDL-71

Authors

R. G. Jacoby, P. A. Benson

Abstract

Two arrangements of internal components of a freeze seal model were tested in order to determine the hydraulic characteristics. The internal configuration employing cylindrical knitted mesh packings in the connecting pipes proved the most suitable for the sodium servicing freeze seal, because these packings are more easily fabricated.

Title

Dynamic Shaft Seals in Space. Quarterly Report No. 8.

Document No.

AD-601338

Abstract

A program is being pursued to develop techniques for sealing a high-speed rotating shaft under conditions of high-temperature liquid metal and vapors, high-vacuum environment of space and to provide long seal life. The liquid metal seal test rig was assembled and installed in the test facility. Water seal test results were reduced with the aid of a computerized data program.

Title

Torque Tube Seal - OIC Flow Controller

Document No.

NAA-SR-MEMO-6374

Author

D. J. Hovley

Abstract

The torque tube seal installed in the bonnet of the OIC flow controller was cycled 1000 times each at 1000 °F and 1200 °F. No indication of leakage through the seal was noted during pressure checks performed periodically throughout the test. A sketch of the torque tube assembly is included.

Title

HNPF Process Tube - Grid Seal

Document No.

NAA-SR-MEMO-4334

Author

J. Charles

Abstract

In a test series, piston ring leakages of water and of liquid sodium were measured. Results of the measurements were compared and graphically presented. Leakage of a 4.500 diameter piston ring was less than 0.140% (allowable 0.198%) using 180 °F water which is equivalent to 607 °F sodium. Leakage measurements were taken at start and after 10,034 cycles with sodium at 625 °F. This test was followed by an additional 25 cycles in 1000 °F liquid sodium. Comparative calculations are added. Photographs show corrosion and efficiency of cleaning

Title

The Building of Dounreay

Document No.

ATOM-8-201-57

Abstract

The construction of the Dounreay Fast Breeder Reactor is described.

Title

SRE Engineering Description (Proceedings of the SRE-OMRE Forum. Los Angeles, November, 1956)

Document No.

NAA-SR-1804, pp. 7-40

Author

W. E. Parkins

Abstract

This report describes the SRE and its associated systems. Included are the reactor, sodium heat transfer systems, sodium service system, service cooling system, helium system, nitrogen system, vent system, liquid waste system, and control and safety rods.

Title

Some Metallurgical Considerations in the Design and Operation of the Top Seal of EBR-II

Document No.

ANL-6837

Authors

B. Blumenthal, L. R. Kelman, H. V. Rhude

Abstract

The top seal of EBR-II is a liquid metal seal. Its design and operation raised a number of questions pertaining to the oxidation of the liquid metal. Its compatibility with the container material, and segregation during thermal cycling. An apparatus was built simulating a rotary seal which allowed observation of the behavior of the liquid metal over a period of 70 days, its drossing characteristics, and compatibility with Type 304 SS under the experimental conditions of rotation and temperature. Repeated thermal analyses caused the alloy to segregate substantially. The influence of sodium on the segregation was determined. Specific recommendations are made regarding the design and operation of liquid metal seals.

Title

Gas Ingestion and Sealing Capacity of Helical Groove Fluid Film Seal (Viscoseal) Using Sodium and Water as Sealed Fluids

Document No.

NASA-TN-D-3348

Authors

L. P. Ludwig, T. N. Strom, G. P. Allen

Abstract

An experimental investigation was conducted on gas ingestion and sealing capacity (expressed as a dimensionless sealing parameter, Lambda G) of viscoseals using water and liquid sodium as the sealed fluids. The investigation covered a reynolds number range of 2,100 to 60,500. Results disclosed that a helically grooved rotor viscoseal (smooth bore housing) has increasing gas ingestion rates with increasing Reynolds number when the viscoseal liquid interface becomes unstable. Helically grooved housing viscoseals (smooth rotors) showed no gas ingestion when sealing sodium (Reynolds numbers from 10,400 to 37,200). When sealing water, gas ingestion occurred only within a specific range of Reynolds numbers (2,300 to 8,400). Secondary grooves on viscoseal land areas improved sealing capacity and reduced power absorption. Sodium was sealed at 149 to 329 °C (300 to 625 °F) without measurable liquid loss for operational periods up to 8 hr.

Title

Technology of Liquid Metals

Document No.

ORNL-TR-854

Author

H. W. Savage

Abstract

A review of the development of liquid metal technology at ORNL was followed by summaries of activities at other laboratories in the United States. A general discussion of proposed CNEN liquid metal studies was given with emphasis on facility requirements. Compatibility tests, heat transfer tests and in-pile loops. Considerable discussion was devoted to handling of sodium including storage, charging of systems, purification of sodium, protection of loops, waste disposal and safety requirements. Sodium components were critically reviewed in light of past experience. Components discussed included cold traps, hot traps, valves, pipe joints, heaters, heat exchangers, pumps, insulation and instrumentation.

Title

Design Criteria for Rotary Seals for a Space Environment. Part I. Rotating Fluid Ring Seal Evaluation

Document No.

AFAPL-TR-65-89, Pt. I

Authors

E. Schnetzer, H. Ernst, J. M. McGrew, T. A. Phillips

Abstract

This study covered development of non-contacting fluid ring seals for minimum leakage rates and long life under specified high temperature, low vacuum operating conditions. Seal development, during Phase I of the program, was supported by analysis. Nondimensional parameters were derived to describe basic performance characteristics of rotating fluid ring seals. Three rotating fluid ring seals were designed and evaluated in water. Depending on seal geometry and surface speed, this testing proved that such seals were capable of sustaining pressure differences of appreciable magnitude. Because an interface instability phenomenon (sputtering) was encountered in the easy fluid tests, a detailed development program to eliminate this source of leakage was undertaken which resulted in a dynamic zero leakage (DZL) seal configuration. This configuration was tested in liquid metal during Phase II of the program. A high-speed seal test rig was designed and operated successfully for 172 continuous hours with a total potassium leakage of 3.4 oz. Accounting for start-up and shutdown inventory losses, the seal leakage rate was established to be within the goal of 1 to 10 lb/10,000 hr.

Title

Frozen Sodium Shaft Seal

Document No.

KAPL-1265

Author

W. A. Heywood

Abstract

The feasibility of using a frozen sodium annulus as a shaft seal was investigated. Three phases of the investigation were: (1) the torque required to turn a shaft in a frozen annulus, (2) the leakage of sodium past a frozen annulus, and (3) the leakage of argon past a frozen annulus. Torques up to 60 ft-lb were required to turn a 1-5/8-in. diameter shaft in a frozen sodium annulus 3-in. long and 1/16-in. thick. Sodium leak rates up to 4 g per day were observed with a 15-psi pressure drop across the annulus. Argon leak rates of the order of hundreds of cubic centimeters per minute with pressure drops up to 10 psi were observed.

Title

Components of the Fused-Salt and Sodium Circuits of the Aircraft Reactor Experiment

Document No.

ORNL-2348

Authors

H. W. Savage, G. D. Whitman, W. G. Cobb, W. B. McDonald

Abstract

The development of component and fabrication techniques for the aircraft reactor experiment (ARE) reactor consumed a 4 yr period, during which time the technology for handling high-temperature fluids, including sodium, was extended to equipment operable above 1500 °F. The methods used for determining compatibility of materials under static and dynamic conditions, standards for materials, and techniques for welding, fabrication, and assembly and the design criteria for pumps, seals, valves, heat exchangers, cold traps, expansion tanks, instrumentation, preheating devices, insulation, etc., are described.

Title

Tests of a Mechanical Pipe Joint for Sodium Service

Document No.

NAA-SR-4204

Author

C. Sutherland

Abstract

Mechanical pipe joints of 2-in., 6-in., and 12-in. diameter for use in liquid sodium systems were tested. The joints were subjected to flexural, axial, tensile, and compressive mechanical loads while containing sodium at a maximum temperature and pressure of 1000 °F and 150 psig. Only the 2-in. size performed satisfactorily. The other units leaked sodium during the mechanical load tests after having sealed 1000 °F sodium under no-load conditions. Tolerance limits were not established for flange warpage due to welding. However, it was found that the mechanical pipe joints tested will not seal under the test conditions if they become out-of-round as much as 10 mils.

Title

Improvements in or Relating to Valves for Controlling High Temperature Fluids

Document No.

Patent-UK-777,039

Author

F. L. Speed

Abstract

A valve for high-temperature liquid (sodium) with a self-sealing leak-proof gland of (a plurality of) O-Ring seals arranged as an insert over the spindle on a sleeve-like extension of the valve body. The extension and the spindle are of such length that combined with fins around the extension, a temperature drop is caused between the valve seat and the gland which results in liquid sodium passing through the valve becoming solid at the gland.

Title

Snap-8 Seals-to-Space Development Test Program Volume II - Molecular Pump

Document No.

AGC-2808/TOPICAL (Vol. 2)

Author

J. N Hodgson

Abstract

In order to provide a seal between the Snap-8 working and process fluids in the rotation machinery and the space environment, a molecular pump of the Holweck Type (a smooth rotor running in a helical groove stator) has been investigated. Theory was developed to describe pump performance in the molecular, transition, and continuum flow regimes and a computer program written to perform the analysis. Pump configurations based on the theory were tested and the data obtained agreed well with theory. A prototype design configuration has been selected based on the experimental and theoretical results of this study.

Title

Liquid Metal Seal for Sodium Pump Shafts

Document No.

NAA-SR-MEMO-2184

Author

S. C. Carniglia

Abstract

A survey of liquid metals and alloys was made to select the best material for use as a liquid metal seal for a vertical pump shaft. Fifteen different metals and alloys were considered in terms of their melting points, densities, nuclear cross-section, electrical resistivity, behavior toward materials of containment, tendency toward oxidation by air and solubility in sodium. Lead-Bismuth and mercury were recommended for this application.

Title

Conceptual Design of Large Sodium Valves

Document No.

ACNP-65579

Abstract

This report establishes the valve requirements for large (1000 MWe) sodium-cooled fast reactor plants, reviews the past and present sodium valve technology, and describes some new sodium valve concepts. A search of the best available documents in sodium valves, personal visits to various sodium facilities, and company experience in the design, fabrication, and use of valves are the basis for this report. The new conceptual valve designs were developed for use in a large fast reactor plant. The valve concepts include many features designed to prevent the recurrence of problems which have been encountered with most valves used for sodium service. Cost estimates are presented for the valves selected.

Title

Molten-Salt Reactor Program Quarterly Progress Report for Period Ending Oct. 31, 1958

Document No.

ORNL-2626

Abstract

Progress on the MSRE for the quarter ending October 31, 1958 is reported. A small submerged centrifugal pump with a frozen-lead shaft seal was operated for evaluation tests. Freeze flange and indented seal flange joints that had sealed successfully in high-temperature molten salt lines were tested with sodium. Three commercially available expansion joints were tested and found to be satisfactory for use in both molten salt and sodium lines. A sodium-graphite system was used to prepare Inconel and Hastelloy N tensile specimens for a study of the effect of carburization on the mechanical properties.

Title

Molten-Salt Reactor Program Quarterly Progress Report for Period Ending January 31, 1958

Document No.

ORNL-2474

Author

H. G. MacPherson

Abstract

Progress on the MSRE is reported for the quarter ending January 31, 1958. A hydrostatic bearing was designed for use in liquid metal pumps, which differs from the conventional bearings in that the pockets rotate on the impeller. A bearing of this type has the advantage that the pressure of the pumped fluid would maintain the centering of the impeller in the pump casing. Both oil lubricated shaft seals and bellows seals are being tested in NaK. A combination of NaK and graphite is being used for a carburization test on Hastelloy N (INOR-8). Also NaK testing of brazes using precious metal alloys revealed no evidence of corrosion after 500 hr exposure at 1200 °F.

Title

Molten-Salt Reactor Project Quarterly Progress Report for Period Ending April 30, 1959

Document No.

ORNL-2723

Author

H. G. MacPherson

Abstract

Progress is reported on the MSRE for the quarter ending April 30, 1959. A small frozen-lead pump seal on a 3/16-in. diameter shaft had operated in sodium for 7500 hr with no lead leakage. A similar seal on a 3-1/4 in. diameter shaft had operated in sodium for 3600 hr with an average leakage rate of 9 cm^3/hr. Redesign of the seal was needed to provide better coolant control and packing to decrease the annulus between the seal and the shaft. Endurance testing of a bellows-mounted seal on a centrifugal pump in a NaK loop continued. Negligible leakage had occurred.

Title

Engineering Problems Pertinent to the Use of Sodium Hydroxide in Reactors (Nuclear Engineering - Part 1.)

Document No.

CEPS-50/11-139-54

Authors

E. M. Simons, J. H. Stang

Abstract

Anhydrous, molten sodium hydroxide, has considerable promise as a reactor liquid, because of its low vapor pressure at moderately high temperatures, its rather small cross-section for the capture of thermal neutrons, and its substantial hydrogen content for neutron moderations. It might be used as a moderator alone in an enriched reactor, as a coolant, or, with the addition of nuclear fuel in the form of a solution or slurry, as the combined fuel-moderator-coolant in a homogeneous reactor. Chemical-plant experience with anhydrous sodium hydroxide has been limited to more or less static, open systems with temperatures not much higher than 100 °C above the melting point. A host of new and difficult engineering problems are encountered in the design of closed circulating systems for temperatures up to 850 °C. This paper discusses some of these problems, including suitable container materials, components for high-temperature hydroxide systems (e.g., pumps, seals, bearings, valves, and plumbing) and instrumentation for such systems, including devices for measuring pressure, temperature, flow, and liquid level. Another engineering problem is that of starting up or shutting down a reactor using sodium hydroxide, which melts at 318 °C. Several possible methods of charging the system and keeping the hydroxide molten are discussed.

Title

Wash Cells (Hallam Nuclear Power Facility. Preoperational Test Completion Report.)

Document No.

NAA-SR-9777, Vol 3(AI-P-1133)

Authors

J. Charles, J. Donovan

Abstract

A description of the HNPF fuel and pump wash cells, and the method of testing is presented along with an analysis of results and corrective action taken to render them operational.

Title

Rotary Mechanical Seal

Document No.

Patent-UK-989,794

Abstract

This invention relates to rotary mechanical seal means adapted to transmit rotary motion between differing environments separated by a tight sealed protective barrier.

Title

Improvements in or Relating to Joints Between Tubes of Dissimilar Materials

Document No.

Patent-UK-839,782

Authors

S. Fawcett, W. Rodwell

Abstract

A tube joint with improved sealing characteristics is presented for coupling tubes having different thermal expansion characteristics and of different diameter. Such an arrangement is required in a sodium-cooled, graphite-moderated nuclear reactor where the larger tube forms part of the structure of the reactor, and the smaller tube is a removable structural tube. The invention described comprises a metal sealing ring located between an internal flange on the larger tube and an external flange on the smaller tube. The means for clamping the sealing ring between the external and the internal flange has thermal expansion characteristics that tend to stabilize the variation of load in the sealing ring with temperature changes. Details of construction and application of this type of tube joint in a SGR system are described.

Title

Development of Materials and Fabrication Techniques. SNAP-8 Topical Materials Report for 1963. Vol. 2, Component Materials Development.

Document No.

AGC-2822, Vol. 2, PG. 11-36

Author

R. S. Carey

Abstract

This report investigates physical properties of, and fabrication techniques for, alloys proposed for use in the SNAP-8 power conversion system. An evaluation of the elevated-temperature, long-time strength of AM-350 showed that the equalized heat treatment provided a greater stress-rupture strength at 1300 °F than the solution-annealed heat treatment. Creep and rupture-strength data on 9CR-1MO steel solicited from major manufacturers and users were statistically evaluated to determine the optimum chemical composition of 9CR-1MO steel for maximum strength. A chromium content of 8.84% in combination with a molybdenum content of 1.25% would provide maximum creep resistance. Recommended design values at 1300 °F included a creep strength of 1100 psi (1% in 10,000 hr).

Title

SGR Component Development Technology (Proceedings of the SRE-OMRE Forum, Los Angeles, February 1958)

Document No.

NAA-SR-2600, PG 57

Author

R. W. Dickinson

Abstract

Specific examples of developments in pumps, valve configurations, and cold traps are noted. Experience with SRE leading to the development of these designs is discussed. Studies of alternate materials of construction for SGR systems are covered, with a brief summary of current results at this and other laboratories - prospects for the use of low-alloy steels are particularly noted. The status of sodium-heated steam generators is reviewed, and preliminary results from the operation of the "once-through" steam generator as they affect the heat transfer system are noted. Developments in the method of cleaning sodium-filled cavities are discussed, noting an unusual procedure which has proved satisfactory in preliminary tests. Experiments in progress to determine the necessity for extensive cleaning procedures in sodium system construction are covered. A brief review of experiments under construction or planned for the immediate future is made.

Title

Final Performance Tests of Two-Coolant-Region Sodium Pump Shaft Freeze-Seals

Document No.

NAA-SR-MEMO-4442

Author

F. O. Streck

Abstract

Two-phase, liquid-solid sodium seal was tested for a sodium pump shaft. Two coolant loops with constant flow rates and with almost constant inlet temperatures could keep the solid seal level within an acceptable narrow range for any operating condition of the pump. The solid-phase thickness was always enough to retain sodium in the seal at 10 psig pressure difference across the whole seal. Sodium bulk temperature was 350 to 945 °F. Tetralin was the coolant in both loops with 95 °F input temperature and 5.8 gpm flow in the cooler region and with 240 to 285 °F, 0.8 gpm flow in the warmer range of the seal. A constant 2.56 kW cooling and 1.2 kW heating load could be established at all pump speeds from 0 to 840 rpm during an 1100 hr test run for the given shaft and seal dimensions. The number of test runs and the measured parameter ranges were statistically determined.

Title

Initial Test of Sodium Pump and Instrument Loop

Document No.

NAA-SR-MEMO-1178

Author

R. Cygan

Abstract

An isothermal loop for testing a full scale sodium pump for SRE and Associated Instruments has been operated for approximately 600 hr. A summary of data obtained, together with recommendations, is given.

Title

Reactor Engineering Division Quarterly Report on the Power Reactor Program, January 1, 1955 Through March 31, 1955

Document No.

ANL-5461, Section II

Abstract

Construction and preinstallation performance tests of the EBR-II working model are described. Low frequency induction heating is being studied for the sodium piping system. Convection heat transfer data are presented on eutectic NaK. Results of an EBR-II design study are reported, including a skeleton flow chart of the entire plant and a temperature-enthalpy diagram for the steam generator. A thermal analysis was made of radiator type fuel elements for the EBR-II and results are presented. Results of irradiation tests of EBR-II pin type fuel elements are summarized. A summary of packing gland tests is presented.

Title

Sealing Means for Receptacles Containing Metal in Liquid or Fused State

Document No.

Patent-UK-780,151

Abstract

The conventional solid sealing means, rings of soft wire laid between flanges of the receptacle, are separated from the liquid metal by a cushion of inert gas introduced into a chamber between the sealing means and the interior of the receptacle. This provides a safe seal (e.g. for pipelines and receptacles containing liquefied sodium as coolant) where covers have to be removable, and are sometimes moved by remote control

Title

The Selection, Design Modification, and Analysis of Sodium Valves for Hallam Nuclear Power Facility

Document No.

NAA-SR-5463

Authors

B. Brooks, R. Galantine, F. Bergonzoli

Abstract

Design of 14-in. throttling, blocking, and check valves - and small blocking valves is described. Both liquid- and gas-cooled valve stem freeze seals are discussed. Prototype tests indicated satisfactory performance except for across-the-seat leakage - a design FIX is described. Thermal and stress analyses are reported.

Title

Development of Pump Components for the Pratt and Whitney Aircraft Liquid Metal Turbopump, TP-1

Document No.

PWAC-289

Authors

J. S. Murphy, R. S. Lombard
J. D. Sutherland, J. Farquhar

Abstract

The pump component studies reported herein were conducted in direct support of the development of a 3000 gpm NaK turbopump. This turbopump, which was designed as a directly coupled, vertically mounted unit, will develop a head rise of approximately 400 ft at a shaft speed of 8000 rpm. The design maximum fluid operating temperature is 1300 °F. Included, in the order mentioned, are the descriptions of the four mixed flow impeller tests, full scale impeller and scroll matching test in water, full scale impeller and scroll matching test in subsonic air, hydrostatic stresscoat tests of the prototype pump scroll and dynamic seal and face seal tests. The last sections of this report include the test stand and test unit descriptions of the water and liquid metal center section items, respectively.

Title

Tests of AN-Type Tube Couplings

Document No.

NAA-SR-MEMO-4772

Author

C. Sutherland

Abstract

Tests were conducted to determine the suitability of the AN-type flared tube couplings for use at 1500-psi stress level service in high-temperature sodium systems. Results indicate that such units are not suitable for use in these systems.

C-47

Title

SNAP-8 Seals-to-Space Development Test Program Volume I - Visco Pump

Document No.

AGC-2808, Vol. 1

Author

R. L. Lessley

Abstract

A seal concept was evolved in which mixing of mercury and oil is prevented by venting a section of the shaft to space and permitting a small controlled leakage of each liquid to the space vent cavity. The particular seal-to-space concept advocated for use in SNAP-8 involves use of the visco pump (a device consisting of a shaft with helical grooves rotating within a close-fitting housing). This seal element prevents passage of raw liquid through the seal. It establishes a liquid-vapor interface past which only vapors can leak. Visco pumps function improperly under certain conditions. A condition known as breakdown can prevail, in which case liquid droplets can be lost from the interface and the effectiveness of the seal as a barrier to liquid leakage is destroyed. Tests were conducted with quartz visco pump test sections which permitted viewing and photographing of the liquid-vapor interface during operation of the test rig. Results showed the visco pump interface to be stable for SNAP-8 operating conditions. There were no signs of breakdown. Measurements of drag and pumping coefficients were made for a variety of pump configurations. The visco pump was very well suited for use in the SNAP-8 seals-to-space.

Title

SNAP-8 Seals-to-Space Development Test Program, Volume III - Dynamic Slinger

Document No.

AGC-2808(TOPICAL) Vol. 3

Authors

R. L. Lessley, I. L. Marburger

Abstract

Tests were conducted to demonstrate that dynamic slingers, using oil and mercury as working fluids, are capable of generating stable liquid-vapor interfaces for use in the SNAP-8 seal-to-space. The test rigs contained transparent housing sections which permitted observation of the liquid-vapor interfaces during operation. Results demonstrated that stable interfaces can be obtained for the operating conditions of the rotating assemblies of the SNAP-8 power conversion system.

Title

High-Temperature Centrifugal Pumps (Nuclear Engineering - Part 3)

Document No.

CEPS-50/13-171-54

Authors

H. W. Savage, W. G. Cobb

Abstract

High-temperature centrifugal pumps for liquids above 1000 °F have presented some unusual structural, material, sealing, auxiliary, instrumentation, operational, and reliability problems. These are discussed briefly in connection with applications using liquid metals and fused salts encountered at Oak Ridge National Laboratory. These problems arise owing to the weakening of structural materials and thermal gradients imposed at high temperatures - the possibility of undue mass transfer or chemical attack - the necessity for cooling conventional associated mechanisms - the necessity for leaktightness, especially in radioactive applications - the high dependence upon instrumentation for determination of satisfactory operation and upon reliability for maintaining this owing to the inaccessibility of such pumps when used in radiation fields. Gas-sealed sump type vertical-shaft pumps and frozen-sodium-sealed horizontal-shaft pumps are described, and the design and operating parameters are discussed.

Title

A Potassium-Steam Binary Vapor Cycle for Nuclear Power Plants

Document No.

ORNL-3584

Authors

W. R. Chambers, A. P. Fraas, M. N. Ozisik

Abstract

The basic factors limiting the efficiency of both conventional steam power plants and mercury-vapor and steam binary cycle plants are discussed relative to the potential of a potassium-vapor and steam cycle for nuclear power plant applications. A conceptual design for one embodiment of the latter coupled to a molten-salt reactor is described. Its thermal efficiency is estimated to be 54%, and the associated calculations, including those for the size of the principal components, are appended. The quantities of material required for the heat exchangers and piping for both a coal-fired supercritical-pressure steam plant and a nuclear-powered potassium vapor and supercritical-steam plant are estimated along with the associated costs. The resulting cost and performance data indicate that the nuclear plant with a potassium-vapor and steam binary cycle could give both lower capital charges and a much higher overall efficiency than a coal-fired supercritical-pressure steam plant.

Title

Research Program Related to Vapor Thermionic Converters for Nuclear Applications

Document No.

EOS-3410-FINAL

Author

A. O. Jensen

Abstract

The major objective of the reported research program was to study the effects of long time at temperature on the surface crystal structure of polycrystalline molybdenum substrates and how these changes in surface crystal structure relate to the performance of cesiated molybdenum emitters (and collectors) for use in cesium vapor thermionic converters for nuclear applications. The investigations were primarily limited to molybdenum and/or vapor deposited coatings on molybdenum as a starting point.

Title

Final Technical Summary Report of Research and Development Program of Thermionic Conversion of Heat to Electricity, Volume I

Document No.

GEST-2035, Vol. 1

Authors

R. H. Bristow, L. N. Grossman, Dr. A. J. Kaznoff

Abstract

The material presented in this volume resulted from an experimental program undertaken to solve some of the materials problems on thermionic converter systems. The electrical resistance of high purity aluminas was measured at typical converter operating temperatures. Tungsten and molybdenum based coatings were developed, metallized to alumina and tested in cesium vapor for over 2000 hr. The joining of various retracting metals to metallized alumina using different bonding techniques were investigated. The cesium corrosion resistance of various materials and seals was determined for temperatures up to 1800 °F.

Title

Final Technical Summary Report on Vapor Filled Thermionic Converter Materials and Joining Problems, Plasma Research Pertinent to Thermionic Converter Operation
15 November 1961 - 15 December 1962

Document No.

AD 617348

Authors

M. J. Slivka, R. H. Bristow, M. D. Gibbons

Abstract

An experimental program to determine the compatibility of cesium vapor with ceramics, metals, and metal-ceramic seals is described. Only high-purity alumina ceramic bodies exhibited resistance to attack by cesium vapor at temperatures up to 900 °C. Results from experimental studies on the emission and discharge characteristics of the TA-CS system are also presented. The information from the studies is most directly applicable to the design and construction of cesium-vapor thermionic converters.

Title

SNAP-8 Seals-to-Space Development Test Program Volume IV - Integrated Seal Simulator

Document No.

AGC-2808, Vol. 4

Authors

R. L. Lessley, J. N. Hodgson,
E. A. Haglund

Abstract

The design, fabrication, test and performance evaluation of a shaft seal for mercury was described. The visco seal, molecular pump and dynamic slinger were used to maintain an interface between mercury liquid and mercury vapor. Three sealing arrangements were tested separately followed by a final test of the units combined into one assembly.

Title

Summary of Shaft Face Seal Development Program

Document No.

TIM-909

Author

D. V. Manfredi

Abstract

This report describes a program for the development of a gas sealing system with a minimum operating life of 10,000 hr for shafts of pumps used to circulate lithium, potassium, or NaK at temperatures up to 1600 °F. The vertical shafts, supported by oil-lubricated ball bearings, were to operate up to speeds of 8500 rpm. The purity requirements of the cover gas above the liquid metal were 2 ppm or less of oxygen or moisture. The sealing system consisted of two oil cermet fact seals, one dry gas cermet face seals, a centrifugal dynamic seal, and two sweep gas systems. This report includes seal manufacturing techniques, performance data and experimental techniques evolved during this program. A total of over 150,000 hr of testing was accumulated in accomplishing the objectives of the overall program.

Title

Analysis of the Heat Generation in the Primary Sodium Pipe Tunnels, Intermediate Heat Exchanger Cells, and the Primary Sodium Fill Tank Vault for HNPF

Document No.

NAA-SR-MEMO-7518

Author

P. J. Legendre

Abstract

The following results point out the major changes in magnitude of the hot spot heat generation due to changes in primary sodium piping layout. The peak heat generation in the diaphragm seal at the east end of the reactor cavity is 0.011 W/cm^2. Previous calculations gave 0.0084 W/cm^2. The peak heat generation in the intermediate heat exchanger cells is 0.031 W/cm^2. The previous peak heat generation in the IHX cell, 0.013 W/cm^2, was located on the cell floor beneath the 16-in. primary sodium pipe.

Title

Liquid Metal Research at NASA-Lewis Research Center (Proceedings of 1962 High Temperature Liquid Metal Heat Transfer Technology Meeting, Brookhaven Natl. Lab.)

Document No.

BNL 756(C-35)pg. 262-275

Author

J. P. Lewis

C-55

Abstract

The report summarizes the heat transfer aspects of LRC liquid metal research activities associated with the development of space power generation systems. A stainless steel 150 kW two-phase flow loop operating with sodium to 1700 °F is described, with initial performance data. Nine major component failures are listed. A sodium turbine facility under construction is to operate with a 1500 kW boiler. Two loops, one low pressure, one high pressure (200 psi) will test centrifugal pumps. Boiling and convective heat transfer of sodium up to 2250 °F will be studied in a forced-flow 500 kW loop, constructed of CB-1ZR above 1500 °F and 316 SS below 1500 °F. A large radiator-condenser test facility using potassium heated by a NaK loop is being fabricated. Operation of bearings and seals in liquid metals up to 1600 °F will be studied. A simulated SNAP-8 loop, including NaK and mercury systems, is being constructed. A comprehensive program is in progress for materials evaluation for alkali metal containment in the 1800 to 2400 °F range.

Title

Design and Fabrication of a 2100 °F Forced Convection Lithium Test Loop (Proceedings of 1962 High Temperature Liquid Metal Heat Transfer Technology Meeting, Brookhaven Natl. Lab.)

Document No.

BNL 756(C-35)pg. 326-352

Author

I. L. Gray

Abstract

High-temperature liquid metal technology forms a part of the Martin Marietta Corporation Research and Development Program on direct conversion nuclear reactors. This paper is a progress report on the design, fabrication and installation of a 1-in. diameter, niobium-1% zirconium alloy, forced convection loop that is rated at a temperature of 2100 °F and flow of 20 gpm lithium. The overall system is briefly described and emphasis is placed on the resolution of problems that arose during the design and fabrication stages.

Title

Water Tests of High Speed Centrifugal Dynamic Shaft Seals

Document No.

TIM-914

Author

G. L. Noell

Abstract

Current higher speed liquid metal pumps under development require smaller diameter shaft seals with minimum power consumption. Four models were tested in water at shaft speeds up to 13,000 rpm. A 36-vane dynamic seal, designated DS-11, with a vane height of 0.200 in., and axial vane clearance of 0.010 in. and two circumferential tank slots exhibited the most stable liquid-to-gas interface at high speeds and provided optimum values of vane and torque coefficients.

Title

Mechanical Pumps for Power Reactor Cooling Systems (Progress in Nuclear Energy. Series 4. Technology and Engineering)

Document No.

Book-McGraw & Hill-177-56

Author

G. W. K. Ford

Abstract

The principles on which to base the design of mechanical pumps for reactor coolants are reviewed with special emphasis on the use of shaft seals, special bearing techniques, and gas blankets. A tentative choice is made of the best system to employ with the coolants of major interest in power reactors. Some of these proposals have not yet been tested on large-scale systems.

Title

SRE Core Recovery Program

Document No.

NAA-SR-6359

Author

W. J. Freede

Abstract

During an experimental nuclear power run at the sodium reactor experiment (SRE), a leak occurred which allowed from 2 to 10 gal of organic material, commercially known as tetralin, to enter and decompose in the reactor core. Carbonaceous residue products reduced the sodium coolant flow in fuel channels to such an extent that 13 out of 43 fuel elements were damaged. A substantial quantity of special equipment was developed, tested, and utilized to assess the situation, remove 81 loose fuel slugs plus other debris from the reactor core, and replace 16 moderator cans, 12 of which contained portions of fuel elements. On-site recovery operations were safely and successfully performed. Work progressed 24 hr per day with the crews following a rotating schedule. In no instance did radiation exposure exceed standard AEC tolerances. On the average, exposure was less than a third of this tolerance.

Title

Sodium Pump Development and Pump Test Facility Design

Document No.

WCAP-2347

Authors

H. G. Allen, J. Boyd, B. Cametti, D. A. Maniero, D. R. Nixon

Abstract

The study defines a program for the development of large sodium pumps for use with sodium-cooled reactor systems of 1000 to 1500 MWe capability, and the functional requirements of a related sodium pump test facility for testing large pumps. The future pump requirements of large power systems have been estimated, a type of sodium pump recommended for further development, the development problems identified, and a program research and development prepared to resolve these problems. The functional requirements of a sodium pump test facility for testing pumps for large reactor use have been established.

Title

Sodium Reactor Experiment Incident

Document No.

NUSA-1/3-73-60

Authors

W. B. McDonald, J. H. Devan

Abstract

The SRE was shut down at the end of power Run 14 because tetralin had leaked into the primary sodium system via a thermocouple well in the pump freeze seal. Tetralin decomposition products resulted in blocking the fuel channels and reducing the heat transfer from the fuel elements and intermediate heat exchanger. The result was fuel-element failures and release of fission products into the sodium. Power Runs 8 through 13 were reviewed to show significant chronological events before the shutdown.

Title

Inexpensive Way to Control Oxygen in Sodium Heat-Transfer Systems

Document No.

NUCL-14-34-56

Authors

I. L. Gray, R. L. Neal, B. G. Voorhees

Abstract

Sodium oxides in sodium fluid accelerate metal corrosion and plug systems in cold regions. Rugged, reliable devices have been developed to measure 3 to 300 ppm oxygen in sodium and NaK (200 to 700 °F) and to reduce oxygen to less than 10 ppm.

Title

Mechanical Pumps for Liquid Metals (Colloquium on Liquid Metals. Aix-en-Provence, France, September, 1963)

Document No.

AEC-TR-6354

Authors

J. Baumier, H. J. Gollion

Abstract

The possible solutions to the problems of pumping liquid metals by centrifugal pumps are reviewed. The study is oriented principally toward the problem of shaft seals and hydrostatic bearings.

Title

Components - Pumps (Proceedings of the 1957 Fast Reactor Information Meeting, Chicago.)

Document No.

Book-AEC-199-57

Author

R. A. Jaross

Abstract

Several notable advancements in the field of sodium pump technology have been made in the past year. The successful testing of two 5,000 gpm, 40 psi head pumps (one a centrifugal, sump type and the other an ac linear induction pump) at Argonne National Laboratory has afforded much valuable information and added greatly to the store of experience needed to make an intelligently-designed sodium coolant system. Extensive tests on these two pumps have been made to assist in the selection of pumps for EBR-II. The pumps are described and tests are reviewed.

Title

Plugging Leaks Between Water and Third Fluid System (HG)

Document No.

MSA Memo Report 92

Authors

S. J. Rodgers, J. V. Friel, J. W. Mausteller

Abstract

An investigation has been made of methods of sealing leaks between water (or steam) and mercury systems. Leaks simulating tube-to-tube sheet joint cracks were successfully plugged in less than 1 hr by adding a commercial boiler sealant (Leakure, distributed by Woodward-Wanger Co., Philadelphia, Pa.), to the water side. Addition of 1 wt% magnesium plus 1 vol% Leakure to the mercury system gave plugs in either steam or water leaks. Operating conditions were 500 psig (465 °F) on the water side and 300 psig, 500 °F on the mercury side. Pressures were alternated in some cases. Plugs held 500 to 800 psi differential pressure both at operating water conditions (atmospheric pressure on mercury side) and at room temperature.

Title

Engineering Design of EBR-II (Fuel Process and Fabrication Cycle) (Proceedings of the 1957 Fast Reactor Information Meeting, Chicago.)

Document No.

BOOK-AEC-23-57

Authors

M. Levenson, J. H. Schraidt

Abstract

The EBR-II, in addition to demonstrating advances in Reactor Technology, will demonstrate the use of a closed fuel cycle. The fuel cycle selected for inclusion in the EBR-II plant is designed for remote operation. Fuel processing will be done by melt refining and fuel fabrication by injection casting. The facilities are being designed to achieve maximum flexibility while the equipment is being designed as specific and simple as possible. The 16-shielded shop is described.

Title

Hallam Nuclear Power Facility Reactor Operations Analysis Program Semiannual Progress Report No. 4, February 29, 1964 - September 30, 1964

Document No.

NAA-SR-10743

Authors

D. Teszler, H. Rubinstein, W. Debear, O. Jenkins, Jr., D. Darley

Abstract

From March 1, 1964, through September 30, 1964, the HNPF was in operation 2,748.7 hr, generating 72,547,000 kW/hr of electricity for an accumulated total of 192,460,000 kW/hr. Operations are reviewed. Data are presented in tables and graphs on reactivity history, fuel element exposure, moderator element failures, control rods, in-core temperatures, fuel channel variable orifices, reactor scram analyses, shutdown margin, special studies related to HNPF moderator elements, systems and component analysis, and the component reliability program.

Title

Liquid Metal-Heated Space Radiator-Mounted Thermionic Generator

Document No.

PWA-2369

Author

F. C. Harter

Abstract

The results of an experimental program to evaluate thermionic generators in a configuration appropriate for liquid metal heating at temperatures up to 1500 °C are reported. The thermionic generator evaluated was cylindrical and consisted of a nickel collector surrounding a trilayer emitter structure. The trilayer structure consisted of a tantalum-tungsten alloy tube, which could contain the liquid metal, surrounded by a concentric insulating beryllium-oxide tube and a tungsten-rhenium alloy emitter tube. The generator met or exceeded the performance goals of the program. At an emitter temperature of 1500 °C, the power density measured was 50% greater than the design objective of 4 W/cm^2, and the measured efficiency matched the goal of 10%. The power density of 1.07 W/cm^2, at 1195 °C exceeded the performance goal of 1 W/cm^2 at 1200 °C.

Title

Development of Air Turbine Drive Components for the Pratt and Whitney Aircraft Liquid Metal Turbopump, TP-1

Document No.

PWAC-300

Authors

J. S. Murphy, R. S. Lombard, R. G. Haskell, J. D. Sutherland

Abstract

The components testing program for development of the TP-1 air-turbine drive consisted of: (1) full-scale fiberglass model tests of the TP-1 turbine inlet scroll using subsonic air, (2) subsonic air tests of five tangential inlet type fiberglass turbine inlet-scroll models, (3) cold hydrostatic tests of the prototype TP-1 turbine scroll using a stress-coat technique, (4) evaluation tests of three types of carbon-face seals, and (5) a single endurance test of the most promising carbon-face seal. As a result of these component development tests, satisfactory modifications to the prototype turbine inlet scroll were effected, the operating stress levels in the scroll at design conditions were determined to be noncritical, and the performance of the best carbon-face seal tested was found to be acceptable from a leakage standpoint. A successful endurance test of approximately 1400 hr was conducted on this seal with an average leakage of about 0.5 cm^3/hr.

Title

A Description of SGR Test Installations Available at Atomics International

Document No.

NAA-SR-4411(REV)

Abstract

Illustrations and descriptions of sodium laboratories and test installations, available at the Nuclear Field Test Laboratory of Atomics International, are presented. These installations were constructed to advance sodium-cooled reactor research and development. Considering the amount of information gained from these facilities, concerning liquid metal high-temperature heat transfer systems, the costs are comparatively small. The exceptional ability of liquid metals to transfer heat, and the rapid temperature transients encountered in sodium components under some conditions, have engendered unique designs calling for the test equipment described here-in.

Title

Pumping Liquid Metals

Document No.

ATNE-9-48-58

Abstract

Two centrifugal pumps, usable for circulating either NaK or a sodium, are described.

Title

Thermal Shock Report 15. Thermal Cycling Applied to One in. Socket Weld Tee

Document No.

MSA Memo Report 121

Authors

G. E. Kennedy, E. C. King

Abstract

A tee is used to join two liquid metal streams of different temperatures. This test subjected the tee to thermal and pressure stress. Visual, dye check and mass spectrometer examinations upon completion of the test (1829 cycles and 1048 hr of operation) showed that the test stresses were not excessive.

Title

Corrosion of Beryllium in Flowing Sodium

Document No.

GEAP-3333

Author

W. W. Kendall

Abstract

Samples of seven types of fabricated beryllium were exposed to sodium flowing at 20 ft/sec at 900 °F for 47 hr followed by 520 hr at 1000 °F. The oxide content of the sodium was first reduced by cold-trapping and then gettering with about 1% calcium. On completion of the test, all samples were found to be nitrided and some were joined together at points where the beryllium had been in contact with beryllium. The beryllium nitride film was the thickest on the face exposed to high velocity sodium being about 50-μ thick. The beryllium nitride film was black, hard, and adherent. The beryllium below the surface film did not appear to be affected by exposure to sodium.

Title

A Gas Shaft Seal for the HNPF Sodium Pump

Document No.

NAA-SR-MEMO-2616

Authors

B. W. Admire, F. S. Nayor

Abstract

Development and testing of a leather lip-type gas seal for a 5-in. diameter rotating shaft are described and results and recommendations are summarized. Oil was used as the gas sealing medium. It was concluded that for zero helium leakage to the atmosphere, it will be necessary to use a double seal arrangement and that the lip-type, oil labyrinth gas shaft seals not be recommended for use on HNPF sodium pumps with large rotating shafts.

Title

HNPF Cold Trap Evaluation

Document No.

NAA-SR-4382

Author

R. Cygan

Abstract

Two designs of sodium cold traps for the HNPF have been subjected to full scale tests. Performance features that were investigated included oxide removal efficiency, oxide capacity, pressure drop characteristics, economizer effectiveness, and temperature profiles. Results indicate that both designs should perform satisfactorily in the Hallam plant.

Title

Performance of HNPF Prototype Free-Surface Sodium Pump

Document No.

NAA-SR-4336

Author

R. W. Atz

Abstract

A free-surface centrifugal pump, incorporating a hydraulic bearing running in sodium, was operated at the conditions required for service in the HNPF. After difficulties arising from inadequate shaft clearances were alleviated, the pump performed properly at a flowrate of 7200 gpm of 945 °F sodium at 150-ft head.

Title

High Conductivity Fins for Gas Cooled Liquid-Metal Heat Exchangers

Document No.

IS-587

Authors

W. F. Brown, R. W. Fisher, H. M. Black

Abstract

Copper fins are nickel plated and brazed to the inconel tubes of an air cooled liquid sodium heat exchanger for use up to 600 °C. Methods of plating annular fins for tubes are discussed and test results are presented. Brazing techniques for fin-to-tube joints are explored, and a resistance brazing process is developed and explained. From the results of the tests, it was concluded that resistance brazing is an acceptable process and electroplating is successful on fins only under very limited conditions.

Title

Device for Supporting the Bed of a Nuclear Reactor Which Is Cooled by a Circulating Liquid

Document No.

Patent-US-3,173,846

Authors

M. Gentilly Chauvin, J. C. Margueron

Abstract

This patent claims: (1) a device for supporting the bed of a nuclear reactor cooled by a corrosive liquid metal coolant; a vessel for the reactor; an interior annular shoulder in the vessel; an annular ring mounted on the shoulder; a horizontal annular projection within the ring; a bed mounted on the projection; a flexible metal skirt depending between the ring and the bed; a pair of spaced adjacent edges for the skirt one of the edges being sealed to the ring and the other being sealed to the bed, and a plurality of calibrated outlets in the skirt for the liquid metal coolant. (2) A device as described in claim 1, a skirt comprising two similar parts sealed together along adjacent edges away from the ring and the bed. (3) A device as described in claim 1 including an annular flange connected to the vessel, the shoulder being formed on the flange, and a plurality of yokes bolted to the flange and holding the ring on the shoulder.

Title

Sodium Reactor Experiment Pump Development

Document No.

NAA-SR-1662

Author

R. Cygan

Abstract

Operation of a 6 × 8 × 13 freeze seal type centrifugal pump has been carried out at sodium temperatures up to 1200 °F. Measurements have been made of cooling requirements, pressure drop, torque, temperature distribution, and reliability of the freeze seals employed in the pump. Design and operational techniques are described. This pump design appears promising for large-scale liquid-metal systems.

Title

Pump Tests Conducted for the Development of the Pratt and Whitney Aircraft Liquid Metal Turbopump, TP-1

Document No.

PWAC-297

Authors

H. V. Marman, R. S. Lombard, P. G. Standley, C. W. Grennan, C. Fuchs

Abstract

Initial development tests of the electric drive versions of a turbopump were conducted in both water and liquid metal in the pump turbine laboratory at Canel. The hydraulic characteristics of this 3000 gpm NaK turbopump were established at approximately half design speed during 216 hr of water tests. In addition, calibration of the gas sweep and lubrication circuits was accomplished. It was established that the overall pump efficiency was 74% at a flow rate of 1500 gpm for a speed of 4000 rpm. A total of 60.1 hr were accumulated in hot NaK which included 26.8 hr at a temperature of 1050 °F. The pump speed during these latter tests was restricted to 3000 rpm because of vibration problems caused by the test facility jackshaft. These first liquid metal development tests were prematurely stopped because of test facility problems and pump assembly errors. However, these tests served to indicate that no major design deficiencies exist in the temperature range investigated. To support the above turbopump development effort, a series of tests were conducted on an available 400 gpm pump, designated model S-1, which was modified to incorporate the essential design features of the sealing scheme of the turbopump unit. A total of 1541 hr of hot NaK testing and 195 hr of water testing were completed on this model pump. The liquid metal tests included a successful endurance test for 1148 hr at 1000 °F.

Title

Reactor Engineering Division Quarterly Report October 1, 1954 Through December 31, 1954

Document No.

ANL-5371

Abstract

The apparent thermal conductivity of steel shot in NaK eutectic was measured in a test apparatus. The experiment was designed to simulate the conductivity of uranium in liquid sodium. An alternate fuel element design for EBR-II consisted of a radiator type element in which the coolant flowed through channels in a stack of fuel wafers. A 4-in. sodium loop with a D-C electromagnetic pump driven by a homopolar generator is described. The loop was to be used to test thin-walled stainless or nichrome tubing with fast flowing sodium up to 932 °F. An apparatus is presented for developing sodium-vapor rotary seals and for testing mechanical components, gears, bearings, and couplings, immersed in sodium vapor or liquid sodium. Two newly-developed seal systems were discussed. A dynamic corrosion loop was built to study the effect of circulating liquid sodium on irradiated and non-irradiated uranium. The content of sodium oxide was studied as a function of time. A detailed design study of the EBR-II is presented.

Title

Reactor Engineering Division Quarterly Report, Section II, April 1, 1955 Through June 30, 1955

Document No.

ANL-5471

Abstract

The EBR-II core subassembly is shown. Designs are given for a sodium bond filling device for EBR-II pin type fuel elements. Further attempts were made to produce a protective coating for uranium. Results of experiments of the fission product contamination of NaK bond in contact with uranium during irradiation are reported. Water pressure drop tests of EBR-II core subassemblies were made. Heat capacity characteristics of the EBR-II working model EM pump are given, along with summaries of packing gland tests. A dynamic cold trap liquid metal purification facility was installed in the EBR-II working model, and the initial results are reported. Schematics are given for the sodium heat transfer system. A manually operated electronic sodium level probe is shown.

Title

Reactor Engineering Division Quarterly Report July 1, 1954 Through September 30, 1954

Document No.

ANL-5345 (Del. 2)

Abstract

A preliminary design configuration has been established in EBR-II, employing central inner and outer blankets. Fuel geometries, including a smaller pin type element, are being investigated. Results are reported on tests on a current conductor connection for a high temperature D-C Electromagnetic pump, on sodium valves with conventional packing, and on a rod gripper test facility for the EBR-II. A vacuum cup sampler for high temperature sodium and NaK systems is described. Lifting and loading mechanisms for the EBR-II working model are described.

Title

Design Summary Report of LCRE Reflector Coolant Pumps and Sump

Document No.

PWAC-384

Authors

C. Ferguson, L. Knudsen, R. Lamers, B. Lucas, D. Manfredi, H. Odom

Abstract

The design and development of the LCRE reflector coolant pump unit are summarized. The pump incorporates the design features of a flight-type coolant pump for a nuclear aircraft power plant. The pump is designed to circulate 175 gpm of 700 °F at a head rise of 106 ft. Operating life is anticipated to be 10,000 hr without maintenance.

Title

Test of Third Fluid Valve for Use with NaK

Document No.

MSA Memo Report 84

Authors

W. Milich, E. C. King

Abstract

A 1-1/4-in. bellows "Hoke" valve was tested and proven satisfactory for use with NaK-78. The valve was found to be leak tight on the valve body and across the seat by mass spectrometer tests given after each series of the major tests listed: (1) valve open - 600 psig nitrogen pressure over NaK at 70, 125, 250, 375, and 500 °F for 1 hr each on valve body. (2) 100 cyclic tests of opening and closing with NaK at room temperature at 40 psig, followed by 600 psig across the seat for 1 hr.

Title

Final Report on Development of LCRE Liquid Metal Pumps and Sumps

Document No.

PWAC-386

Authors

C. Ferguson, D. Manfredi, J. Milich, H. Welna

Abstract

The proposed ground test of the Lithium-Cooled Reactor, designated LCRE, required the development of centrifugal pumps mounted in sumps to circulate the lithium primary coolant at 1000 °F, the NaK-78 secondary coolant at 700 °F and the NaK-78 reflector coolant at 700 °F. The design operating conditions for these pumps were: (1) 195 gpm at a head rise of 115 ft for the lithium (LP-1) primary coolant pump, (2) 394 gpm at a head rise of 143 ft for the NaK (NP-1) secondary coolant pump, (3) 175 gpm at a head rise of 106 ft for the NaK (RP-1) reflector coolant pump. All three pumps utilized anti-friction shaft support bearings lubricated and cooled by MIL-L-7808D oil and were powered by gas-cooled electric motors. Primary sealing of the liquid metal was accomplished with centrifugal dynamic shaft seals, which maintained a liquid metal-to-inert gas interface on the rotating vanes of the dynamic seal impeller. Secondary sealing was provided by inert gas sweep systems and cermet-faced mechanical shaft seals. Extensive testing of pump components, as well as water and liquid metal tests of complete pump assemblies, were accomplished to meet the program objectives of high performance and high reliability for the expected total operating lifetime of 15,000 hr. Over 60,000 hr of component face seal testing and over 74,000 hr of pump testing

Abstract (contd)

in liquid metal were conducted. Several successful liquid metal tests of 10,000 hr duration were accomplished with the LP-1 lithium and NP-1 NaK pumps prior to termination.

Title

Design Summary Report LCRE Secondary Coolant Pump and Sump

Document No.

PWAC-385

Authors

C. Ferguson, L. Knudsen, R. Lamers, B. Lucas, H. Odom, H. Marman

Abstract

The design and development of a centrifugal pump for NaK are summarized. This pump, originally intended for 1200 °F operation in the ANP program was subsequently modified for use on the LCRE. The final operating conditions were 700 °F with a flow of 394 gpm against a head rise of 143 ft. Operating life is anticipated to be 10,000 hr without maintenance.

Title

Design Summary Report of LCRE Primary Coolant Pumps and Sump

Document No.

PWAC-383

Authors

C. Ferguson, L. Knudsen, R. Lamers, B. Lucas, D. Manfredi, H. Odom

Abstract

The design and development of a centrifugal pump for lithium are summarized. This pump, originally intended for 1600 °F service in the ANP program was subsequently modified for use on the LCRE. The final operating conditions were 1000 °F with a flow of 195 gpm against a head rise of 115-ft. Operating life is anticipated to be 10,000 hr without maintenance.

Title

Alkali Metal Resistant Wire

Document No.

APL-TDR-64-42

Authors

E. S. Bober, R. E. Stapleton,
W. H. Snavely

Abstract

A ceramic-coated conductor system was developed that was unaffected by 172 hr exposure to K vapor at 850 °C. This system was Metco 201 zirconia plasma sprayed on nickel-aluminide-coated, nickel-clad, silver AWG 8 (Sylvania) wire. This system is not suitable for electrical operation at 850 °C since the resistance of the zirconia coating drops to about 5 ohms at 600 °C. Overcoating the zirconia with aluminum nitride by a vapor plating technique results in a resistance at 850 °C in argon for the multicoat of 0.018 million ohms. The aluminum nitride overcoated zirconia is unaffected by 172 hr exposure. A number of high purity insulators were found to be resistant to potassium vapor at 850 °C. These were alumina, magnesia, and strontium zirconate. A primary problem with plasma spray coatings of these materials is loss of adhesion in the potassium vapor even when a nickel aluminide undercoat is used. Data obtained on lead in insulators of test capsules indicates a decrease in surface resistivity, however, the lowered resistance of these insulators (0.010 million ohms) is still in a useful range. Statorette adherence tests show several insulated conductors will withstand vibration, shock, and acceleration conditions. A statorette coil of nickel-clad silver wire was partially insulated with nickel

Abstract (contd)

aluminide, zirconia, and alumina (98%) coating. The remaining portions of the conductor were coated with nickel aluminide and very high purity alumina (99.9%). The high purity alumina remained on the conductor after 172.

Title

Design and Operation of a Sodium-to-Lithium-to-Air Heat Transfer System (Period from September 1949 to June 1953)

Document No.

APEX-327

Authors

A. Crocker, R. Potter, R. Spera, T. McLay, S. Esleeck

Abstract

This report describes the design and operation of a 50-gpm liquid-metal circulating system transferring heat from sodium to lithium to air. It also includes a detailed listing of equipment with its cost. Much detail has been included so that the report is a complete history of the work accomplished and a modus operandi of the system. The report is divided into four sections: (1) the summary presents the overall conclusions of the report, problems revealed and as yet unresolved, conclusions incidental to the liquid-metal circulating system but significant to the experimental methods; (2) overall system design deals with such design aspects as circuitry, size, and materials of the overall system; (3) in component design, operation, and results primary components of the liquid metal system are treated individually; (4) design descriptions with revisions thereto and methods of operation are included, and difficulties and their corrective measures are enumerated. The section on

Abstract (contd)

overall operation shows the interrelation of the components in the three separate circuits tested. Specific components involved, operation time, and general results are given for each circuit with the purpose of keeping a chronological record of the entire test.

Title

Sodium Reactor Experiment Power Expansion Program-Heat Transfer Systems Modifications

Document No.

NAA-SR-10379(REV.)

Authors

W. J. Freede, J. K. Roberts

Abstract

Under the power expansion program (PEP), modifications have been made to the sodium reactor experiment (SRE) facility to improve plant reliability and permit an increase in power to 30 MW_t, with a reactor coolant outlet temperature up to 1200 °F. Pertinent changes in the heat transfer systems include: (1) installation of a new main intermediate heat exchanger; (2) replacement of sodium coolant pumps; (3) increase in diameter of certain main primary sodium coolant lines from 6 to 8 in.; (4) reduction in number of valve and freeze traps, (5) utilization of flowing vent lines; and (6) installation of a circulating nitrogen atmosphere cooling system for removing excess incidental heat from galleries.

Title

Quarterly Status Report on LAMPRE Program for Period Ending November 20, 1961

Document No.

LAMS-2647

Abstract

The operations and activities carried out in conjunction with LAMPRE I are summarized. Corrosion testing of containers made out of various tantalum alloys is described, and data are presented on the tensile properties of high-purity tantalum sheet. A ternary phase diagram is presented for molten Pu-Ce-Co alloys. Melting points are given for a number of ternary-U alloys and systems. A preliminary test of 5 ceramic crucibles with Pu-U-Mn fuels was made at 900 °C. A processing method was devised for recovery of uranium and plutonium from sodium paste blankets or paste cores. Results of an analysis of a one-region, direct contact reactor using U-Pu fuel is given. A small-scale critical test of a circulating fuel, direct contact core called pint bottle experiment is described. Sodium test facility operation is described briefly.

Title

Performance of HNPF Primary Pump

Document No.

NAA-SR-MEMO-5988

Author

H. P. Schroeder

Abstract

A proof test of a primary sodium pump, prior to its installation in HNPF, is described. During the test performance curves were developed, temperature information recorded. Performance of the preheater was also observed. At the conclusion of the test, the pump was dismantled and inspected. All data indicates the pump should perform satisfactorily in the HNPF.

Title

Performance Test of a Two-Coolant-Region Sodium Pump Shaft Freeze-Seal

Document No.

NAA-SR-MEMO-4119

Author

F. O. Streck

Abstract

The operation of the freeze-seal type sodium pump requires a shaft freeze-seal capable of retaining sodium. A prototype two-coolant-region freeze seal for application on HNPF sodium pumps had been designed and constructed. It was tested under environmental conditions to determine its operating characteristics and sodium retaining capabilities.

Title

Centrifugal Dynamic Shaft Seals

Document No.

MEEN-86-48-64

Authors

G. M. Wood, D. V. Manfredi, J. E. Cygnor

Abstract

The essential features of centrifugal dynamic seals are illustrated. An impeller with vanes on one face, and the other face smooth is mounted on a shaft and rotated inside a housing with close-fitting radial and axial clearances. Besides serving as the primary seal against the pressurized liquid, the centrifugal types of dynamic seals offer other advantages: (1) the capability of degassing the pumped fluid; (2) positive liquid-liquid control during operation for arbitrary orientation of the shaft, and (3) the capacity to accommodate transient pressure fluctuations of moderate degree without losing liquid-level control.

Title

The Development of an Insulated Thermionic Converter-Heat Pipe Assembly, First Quarterly Technical Report June 1965

Document No.

NP-15433

Author

W. E. Harbaugh

Abstract

Under a contract toward the development of a long-life, insulated thermionic converter, integrated with an efficient, constant temperature heat transfer device, the cylindrical converter, RCA Type A-1198 B will be adapted for heating by means of a heat pipe to operating temperatures of 1500 °C. The module available from prior work was life tested for more than 1000 hr. Irreparable damage was caused by equipment failure and the test was halted. Progress was made in the development of high-temperature, ceramic-metal seals. A program has been detailed for the production of ultra-pure cesium and the accurate analysis of the result. New approaches toward lowering the collector work function are being investigated. The promising results of adding xenon to the operating converter is being further investigated to optimize performance. The heat-pipe test vehicle has been designed for the evaluation of processing techniques. The process of insulating the emitter from the heat pipe at high temperatures is proceeding on schedule. The converter design has been modified to facilitate the use of a heat pipe as the prime source of heat.

Title

Developing Screw Type Shaft Seals for Potassium Environment

Document No.

REDE-17-48-66

Author

A. E. King

Abstract

A shaft seal, called the viscoseal, is described. It is a screw-thread pump achieving a pressure differential by pumping a viscous fluid axially along the clearance between the threads and the shaft. The fluid within the pump acts as the seal between the high and low pressures. At laminar conditions, the maximum pressure head developed by the pump varies linearly with speed, viscosity, and length, and exponentially with radial clearance. At high-speed, turbulent conditions, the head developed can become quite large since it is no longer a linear function of speed but instead is an exponential function. Tests in 375 °F potassium at shaft speeds to 20,000 rpm demonstrated the feasibility of using noncontact liquid and vapor screw-type shaft seals in a potassium-cooled and lubricated electro power generator for space environment.

Title

The Fast Neutron Power Reactor. (Solid Fuel Reactors.)

Document No.

BOOK-AW-13-58

Authors

J. R. Dietrich, W. H. Zinn

Abstract

The reactor physics and reactor safety of the fast neutron power reactor is discussed. Sodium plant technology is reviewed by presenting the properties of sodium and NaK. Also current technology is reviewed for components such as heat exchangers, steam generators, pumps, and sodium handling systems.

Title

Status of SNAP-8 Electrical Generating System. (Space Power System Engineering.)

Document No.

BOOK-AP-483/502-66

Author

R. W. Powell

Abstract

The SNAP-8 is a 35-kW turbo-electric nuclear space power system using a mercury rankine cycle. It will operate continuously for 10,000 hr after remote automatic startup. Possible applications for SNAP-8 include manned space stations, manned lunar bases, and deep space probes. The unique features of the four-loop system are described. A brief report is given on the status of the program including the nuclear reactor system. The facilities for non-nuclear and nuclear testing of the power conversion system are described. The developmental design features and the confirming test data are presented on the following significant components of the power conversion system - the boiler, the condenser, the auxiliary start loop heat exchanger, the turbine-alternator assembly, the composite seal to space, the HG motor pump assembly, the NaK motor pump assemblies, and the lubricant-coolant pump assembly. A 1/16 scale four-loop corrosion loop system which is being operated concurrently with the SNAP-8 program will be described. Extrapolation of this data is expected to yield performance data on the mercury containment materials ability to survive the 10,000 hr space requirement. The results of the nuclear radiation tests on materials and components are discussed.

Title

Sodium Reactor Experiment. (Annual Technical Report. AEC Unclassified Programs, January - December 1957.)

Document No.

NAA-SR-2400, Pt 1, Sec 2

Author

L. E. Glasgow

Abstract

A program conducted to obtain experimental engineering information from the SRE and to improve the performance of SRE is described. Pertinent areas covered include installation of an eddy current brake to control post-scram flow decay, installation of an EM pump in the moderator system to provide compensation for grid plate leakage, temperature difference limitations in the intermediate heat exchanger, modifications of the centrifugal sodium pumps to prevent leakage and improve their performance, problems encountered with bellows seal valves, problems encountered with conduction Type EM pumps and consequent replacement with linear induction pumps, fuel element damage due to frozen sodium and consequent change in operating and handling techniques, and problems encountered in removing moisture from the gallery inert gas system.

Title

SGR Advanced Studies. (Annual Technical Report. AEC Unclassified Programs, January - December 1957.)

Document No.

NAA-SR-2400, Pt 1, Sec 3

Author

R. W. Dickinson

Abstract

Evaluation of concepts for large sodium graphite reactors and progress in the development of system components and materials are described. Evolution of core vessel designs and selection of the calandria as the optimum design are discussed. The use of a commercial terphenyl as a cleaning agent for sodium vessels is described. Design of a large component test loop is discussed along with the design of a NaK-filled capsule for fuel element irradiation testing. A description of a re-entrant tube construction type steam generator and plans for its testing are presented. Freeze seal and gas seal developments are discussed. Compatibility studies between various chrome steels, sodium, and austenitic steels are described.

Title

Sodium Graphite Reactors. (Annual Technical Report. AEC Unclassified Programs, January - December 1957.)

Document No.

NAA-SR-2400, Pt, Sec 1

Author

R. W. Dickinson

Abstract

The use of the SRE as an experimental tool to demonstrate the feasibility of the sodium graphite reactor concept is discussed. Problems and solutions are related. The installation of an eddy current brake and throttle valves to control postscram flow decay to reduce thermal shock is reported. A new design of moderator elements, replacing the zirconium cladding with Zircaloy 2 cladding, plus other improvements is discussed. Other areas of interest include sodium component handling equipment, control of sodium fires, moderator handling facilities and equipment, compatibility of zirconium in oxygen bearing sodium at 1000 °F, the zirconium nitriding problem, and compatibility of low alloy steels with 1000 °F sodium. A replacement calandria core is also discussed.

Title

Piping Flexibility Analysis of Cavity Piping

Document No.

NAA-SR-MEMO-3436

Authors

C. P. Hughes, C. P. Craig

Abstract

An analysis of HNPF piping configurations, for the purpose of establishing forces and moments on the diaphragm seals and determining maximum resultant stresses due to thermal expansion, is presented. Sketches of piping runs and calculational methods are included.

Title

Pressure and Temperature Transducers for High Temperature and Nuclear Radiation Environments (Proceedings of the 1964 National Telemetering Conference, June 2, 3, 4 Biltmore Hotel, Los Angeles, Calif.)

Document No.

CONF-658-1, pg 1-1

Author

S. A. Hluchan

Abstract

Volumetric type transducers for high temperature (400 to 1500 °F). Nuclear radiation and difficult environments are described. A high-temperature fill fluid, NaK is used to provide a hydraulic transmission link between a seal element contacting the process medium and an electrical output pressure sensor located in a more favorable environment. The design and performance of gage and differential pressure measurement systems utilizing a complete liquid fill are discussed. Also, experimental data relating vapor pressure and temperature for NaK in the temperature range of 1500 to 2000 °F. is presented in a discussion of a NaK vapor type temperature transducer.

Title

Sodium Graphite Reactor Quarterly Progress Report, September-November, 1953.

Document No.

NAA-SR-956

Author

G. M. Inman

Abstract

For a central station reactor power plant of the sodium-graphite type, two designs have been investigated. The first operates as a converter using slightly enriched uranium fuel and produces 150 electrical MW. The second operates as a thermal breeder using U233-Th alloy fuel and produces 300 MW_e. Consideration has also been given to the problems associated with the design and operation of the sodium reactor experiment. Engineering studies specifically related to the above reactor designs were carried out. The greatest emphasis was on experimental and analytical work not related to any particular design, but of general importance to sodium-graphite type reactors. The experimental work comprised various tests related to the properties of materials, as well as the operation of reactor components.

Title

Reaction Rate of Solid Sodium with Air

Document No.

IECH-49-1931-57

Authors

W. H. Howland, L. F. Epstein

Abstract

This study was undertaken to determine the influence of temperature and composition of solid sodium on its rate of reaction with air. A quantitative re-examination was made of the observation that the reaction speed of alkali metals and air is markedly dependent on purity. The rate of reaction is greater for filtered sodium than for the relatively pure distilled material, particularly at lower temperatures, and the difference between the rates tends to vanish at the melting point of the metal. These experimental results suggest that the reaction is catalyzed by the presence of small amounts of impurity and that the catalysis reaction has a high energy of activation.

Title

Research on Liquid Metals as Power Transmission Fluids

Document No.

WADC-TR-57-294

Author

R. H. Blackmer

Abstract

The eutectic alloy of sodium (23 wt%) and potassium (77 wt%), known as NaK-77, has been determined from a technical survey of liquid metals and salts as the most feasible liquid known for 10 to 1000 °F hydraulic system applications. A single-cylinder test pump in an inert atmosphere glove-box has pumped NaK-77 up to 3600 psi at 100 °F and up to 2000 psi at 1000 °F. A total of about 100,000 cycles at 1 cps and an average pressure of 1500 psi have been accumulated. Results of literature survey, consultation, and laboratory tests are included in this report.

Title

Sodium Reactor Operating Experience

Document No.

CEPR-57-54-3/61

Author

R. E. Durand

Abstract

The SRE is described, and details of its heat transfer system are given. Pumps in all heat transfer systems are centrifugal with frozen sodium shaft seals. Reactor operations and design modifications are discussed. An eddy current brake was constructed to reduce post-scram convection flow. An improved sodium-to-sodium heat exchanger was made to eliminate temperature stratification problems. A drying system was installed to control the gallery moisture content. Auxiliary coolant tetralin leaked into the sodium system, and its decomposition products fouled 13 of the 43 fuel process channels, resulting in fuel jacket destruction. The plant was shutdown and modified to prevent recurrence of hydrocarbon in-leakage. A steam/fuel washing facility was installed to remove the sodium residue from core elements. Equipment was developed for removing moderator elements and fuel element pieces from the reactor. Maintenance and modification procedures are briefly discussed.

Title

Engineering and Constructing the Hallam Nuclear Power Facility Reactor Structure

Document No.

NAA-SR-7366

Authors

J. Mahlmesiter, W. Haberer, D. Casey, J. Susnir, T. Ricci, W. Peck

Abstract

The Hallam Nuclear Power Facility reactor structure, including the cavity liner, is described, and the design philosophy and special design requirements which were developed during the preliminary and final engineering phases of the project are explained. The structure was designed for 600 °F inlet and 1000 °F outlet operating sodium temperatures and fabricated of austenitic and ferritic stainless steels. Support for the reactor core components and adequate containment for biological safeguards were readily provided even though quite conservative design philosophy was used. The calculated operating characteristics, including heat generation, are summarized. Shop fabrication and field installation experiences are also briefly related. Results of this project have established that the sodium graphite reactor permits practical and economical fabrication and field erection procedures - considerably higher operating design temperatures are believed possible without radical design changes. Also, larger reactor structures can be similarly constructed for higher capacity (300 to 1000 MW_e) nuclear power plants.

Title

Sodium Graphite Reactors. (Annual Technical Progress Report, AEC Unclassified Programs, Fiscal Year 1959. Sections 1-A, 1-B, 1-D.)

Document No.

NAA-SR-3850, Sec. 1A,B,D.

Authors

L. E. Glasgow, R. W. Dickinson

Abstract

General research and development progress on the SGR concept is reported. Operation of and experiments conducted at the SRE are discussed. Subjects covered include material-sodium compatibility, improved reactor system and component design, testing of reactor components, new operation, maintenance or fabrication techniques, and development of testing loops and rigs. More detailed coverage is given in topical reports listed at the front of the section.

Title

Valves for the Hallam Reactor

Document No.

PRET-4(4-54)56-61

Abstract

A survey of the sodium valves to be used in the HNPF is presented. Discussion centers on the seating and stem sealing provisions required for 950 °F sodium service.

Title

Sodium Pump and Auxiliary Components for Hallam

Document No.

PRET-4(2-59)61-61

Abstract

A brief description of the prototype centrifugal pumps tested for the HNPF was presented. The shaft freeze seal and the hydrostatic bearing were cited as the principal development items.

Title

Survey of Sodium Pump Technology

Document No.

WCAP-2255

Author

D. R. Nixon

Abstract

This report reviews the current status of sodium pump development as related to nuclear power applications. The report includes a description of the design features and performance characteristics of the more important types of sodium and sodium-potassium alloy (NaK) pumps. Some requirements for sodium pumps for future large liquid metal reactor systems are presented with some preliminary consideration of the potential of various pump types to meet these requirements.

Title

Reactor Coolant Pumps - Sodium Freeze Seal Cooling Loads

Document No.

NAA-SR-MEMO-1843

Author

R. W. Atz

Abstract

A summary of the cooling load data for the SRE sodium coolant pump freeze seals is presented.

Title

Hydraulic Studies of the LSGR Process Channel

Document No.

NAA-SR-11123

Author

J. H. Brindley

Abstract

Hydraulic tests were performed with water on a full-scale mockup of the large sodium graphite reactor (LSGR) process channel consisting of a process tube containing an 18-rod cluster of fuel rods, cookie cutter spacers at 1-ft intervals, and a variable orifice assembly. Reported are pressure-loss flow-rate characteristics of entrance and exit components - fuel rod spacers - the fuel bundle - and a unique variable flow-control device. Principles of dimensional similitude were used to convert water test data to those of LSGR coolant conditions. The sodium flow rate through the process channel at the design core pressure drop of 21.0 psi can be controlled from 45.5 to 13.4 lb/sec with the variable orifice. Radial velocity-profile studies in the fuel bundle demonstrated that the maximum-to-average velocity ratio is 1.33. Fuel rod spacer drag coefficients, based on the De Stordeur correlation, were found to be constant at 1.5 over a Reynolds number range of 1,000 to 20,000.

Title

Working Fluids Program (NASA-AEC Liquid Metals Corrosion Meeting, December, 1961, Brookhaven Natl. Lab.)

Document No.

TID-7626(Pt.1), pg. 111-114

Authors

D. A. Kirk, J. A. Roth

Abstract

Discussions pertaining to the thermophysical properties of ruthenium at 2000 °F and below, and pertaining to determination of impurities in this metal are presented. Research on high-temperature liquid metal condensation is also reported.

Title

Final Report, LCRE Valve Development

Document No.

PWAC-401, Pt. II

Author

D. E. Smith

Abstract

The lithium-cooled reactor experiment required liquid metal valves capable of 10,000 hr operation in NaK and lithium at maximum temperatures of 1200 and 1000 °F, respectively. To arrive at valve designs compatible with this liquid metal service, a development program of component and valve assembly tests was initiated. Approximately 38,000 hr of liquid metal testing was accumulated on a total of 11 valves prior to the termination of the program.

Title

Selected Operating Experience of Commission Power Reactors (Presented at American Institute of Electrical Engineers, June 1961, Ithaca, N.Y.)

Document No.

TID-13305

Author

J. O. Roberts

Abstract

Operating history and statistics pertaining to the SRE, EBR-I, and SIR are presented. From this history it is concluded that the feasibility of using sodium as a heat transfer medium in large nuclear power plants is established. A brief account of bellows seal valve failures, IHX stratification problems, fuel element and moderator element cladding damage, and fission product retention by sodium and decomposed tetralin is given. The EBR-I core meltdown incident, and the lack of reactor stability which caused the incident, are described.

Title

The Design and Development of a Liquid Metal Mechanical Pump

Document No.

A.E.R.E. R/M 95

Authors

F. L. Speed, K. A. Tomblin

Abstract

A description of a single-stage centrifugal pump for laboratory sodium service is given. Sealing of a vertical 2-5/16-in. shaft while rotating at 3000 rpm is described. Results of a hydraulic test using water are also reported.

Title

Pumps for Liquid-Metal Flight Controls (ASME-EIC Fluids Engineering Conf., Denver, Colo., April 1966)

Document No.

ASME-66-FE-20

Author

J. R. Granan, R. C. Kumpitsch

C-101

Abstract

Efforts to develop a hydraulic system for 1000 °F operation for aircraft service using NaK-77 as the working fluid are reported. Principle effort has been concentrated on development of a pump capable of generating a differential pressure of 3000 psi at 1000 °F. Both positive-displacement and multi-stage centrifugal pumps were tested for this service. Test results are not disclosed but centrifugal pumps are indicated as having the most promise for future development. The principal areas requiring further development are the shaft seals and bearings. Results of several seal and bearing material compatibility tests are given.

Title

Thermionic Radiatior System (Power Systems for Space Flight)

Document No.

BOOK-AP-535/549-63

Author

K. E. Buck

Abstract

Current plans for high power level space power systems are based on the use of high-temperature, liquid-metal cooled reactors coupled to a rankine cycle power conversion system. An alternate approach is the thermionic radiator concept in which similar reactors are used to supply thermal energy to the cathodes of radiator-mounted thermionic converters. The thermionic radiator system could be a backup to dynamic conversions systems if the liquid metal cooled reactors under development are designed to satisfy the requirements of both systems. The thermionic radiator system should be considered also as the prime conversion system. Aerojet-General Nucleonics is engaged in a program to determine the feasibility of a thermionic radiator system. The primary objective of the first phase was to demonstrate the operation of a thermionic converter, using a liquid metal to supply thermal energy to the cathode. In addition, some limited systems analysis was performed. The first phase just recently has been completed, and the objectives fully were attained.

Title

Cesium Thermionic Converters and Generators for Solar Space Power Systems (Power Systems for Space Flight.)

Document No.

BOOK-AP-733/761-63

Authors

P. J. Brosens, S. S. Kitrilakis

Abstract

The research work on thermionic converters accomplished at thermo electron is reviewed. Data obtained for converters, using niobium, tantalum, tungsten, iridium, and rhenium emitters, as a function of interelectrode spacing and converter operating temperatures, is given in the form of maximum power density maps. General design requirements for solar thermionic generators for space applications are discussed. Work done at thermo electron on cylindrical monodiode and cubical multidiode generators is described. Although no inherent limitation has been observed on the life and reliability of thermionic devices, life of the order of several thousands of hours and higher reliability must be and remains to be demonstrated. Effects of cesium interelectrode pressure and heat conduction across the cesium vapor were investigated. No corrosion by the cesium vapor was observed.

Title

High Temperature Alkali Metal Resistant Insulation

Document No.

AD-602440

Authors

W. H. Snavely, R. E. McVay, R. E. Stapleton

Abstract

Materials were selected for 850 °C potassium vapor exposure. A transformer was designed and selected as a test device to determine the combined effect of 600 °C potassium vapor, fluctuating magnetic fields and high ac voltage on insulators, potting compounds, magnetic materials, and electrical conductors. Potassium vapor corrosion tests are in progress on the materials that will be used in the final test transformer. Ceramic to metal seal development is being studied extensively as terminal seals for electrical feed throughs will be needed for electrical testing in potassium vapor at elevated temperatures.

Title

Dynamic Corrosion Test of Type-347 Stainless Steel in Sodium

Document No.

KAPL-M-NGM-1

Authors

N. G. Mills, R. F. Koenig

Abstract

As part of the program to compare various sodium purification treatments, a 6-month dynamic corrosion test employing thermal circulation was made with aged and filtered sodium to determine the corrosion and erosion resistance of Type-347 SS. The test was scheduled for a year's run but a leak developed, ending the test after 6 months. The temperature relationships were observed for indications of plugging by metal or oxide deposition. No plugging was encountered. The results showed that Type-347 SS was very resistant to corrosion and erosion in aged and filtered sodium at 500 °C (932 °F). It was also shown that such a system can be operated with moderate flow rates (0.3 ft/sec) and relatively small openings (3/16 in.) without difficulty. The attack by aged and filtered sodium was to be compared with that by calcium-gettered sodium. (Due to other considerations, the test with calcium-gettered sodium was not made).

Title

EBR-I and EBR-II Operating Experience (Fast Reactor Technology. Detroit, April 1965)

Document No.

ANS-100, pg 25-40

Authors

M. Novick, F. D. McGinnis, G. K. Whitham

Abstract

EBR-I was shut down after over 12 yr operation in a program which proved the feasibility of the fuels breeding concept, demonstrated the practicality of operating a reactor power plant with a liquid metal heat transfer system, provided useful information relative to the safety of fast reactors, and demonstrated that a fast reactor fueled principally with plutonium can be safely controlled by normal operators without any unusual special measures being taken. Operation of EBR-II has proved to be routine during periods when it was operable. Normal startup procedures can be completed and the reactor made critical in about 2 hr. Continuation to power proceeds smoothly, and the synchronization of the generator with the outside lines offers no difficulty. Prior to the shutdown in October, 1964, the reactor had accumulated a total of 16,928 $MWhr_t$ (705 MWd/tonne) of operation and 3,459 $MWhr_e$ of electrical production. Filling of EBR-II with tank-can sodium is described. Use of cold traps to purify the sodium and to collect pipe scale is also discussed. The use of a plugging meter to monitor cold trap performance is reported. Resolution of early difficulties experienced with the 5000 gpm primary sodium pumps and the 6000 gpm secondary sodium pumps is also described.

Title

Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending June 30, 1957

Document No.

ORNL-2340, Parts 1-5

Authors

W. H. Jordan, S. J. Cromer, A. J. Miller

Abstract

Progress is reported on reactor and facility construction, component development and testing, instrument and controls development, engineering design studies, design physics, materials and components inspection, heat transfer studies, phase equilibrium studies, physical properties of molten materials, production of purified mixtures, analytical chemistry, development studies of Ni-Mo alloys and Inconel, welding and brazing, corrosion and mass transfer, materials fabrication, metallographic examinations of engineering test components after service, nondestructive testing and radiation damage.

C-107

Title

Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending December 10, 1955

Document No.

ORNL-2012 (Pt. 1, 2, 3) (Del.)

Authors

W. H. Jordon, S. J. Cromer, A. J. Miller, A. W. Savolainen

Abstract

System flowsheets and instrumentation lists were prepared in an initial attempt to define the entire art. The fuel-to-NaK heat exchanger design was modified to overcome interfaces at the headers and to provide additional space in the region of the headers for the support struts. The problem of cooling the fuel fill-and-drain tank was studied. Full-size models of the impeller and the volute for the art sodium pump are being tested with water. Tests of an art fuel pump are under way with high-temperature (1400 °F) Nak as the pumped fluid. Operation of intermediate heat exchanger Test A was continued, and Test Stand B was placed in operation. A small heat exchanger test stand is operating with a 20-tube fuel-to-NaK heat exchanger, a 500-kW NaK-to-air radiator, and a circulating cold trap. The test data obtained with the intermediate and small heat exchanger test stands have been correlated. A test assembly for use in the development and testing of cold traps and plugging indicators is being fabricated. Several Inconel forced-circulation loops were operated in a study of the effect of the oxide content of the sodium being circulated. Thermocouple assemblies were exposed to sodium in seesaw apparatus so that the effect of various amounts of Chromel-Alumel in the weld nugget could be studied. A third 500-kW NaK-to-air high- conductivity-fin radiator, two 20-tube fuel-to-NaK heat exchangers,

Abstract (contd)

and two 100-tube bundles for intermediate heat exchangers No. 3 were fabricated. The two NaK-to-air radiators that failed in service were examined.

Title

Materials for Pump Bearings, Valves, and Seals for Fluorides and Sodium (ANP Materials Meeting November 16-18, 1954)

Document No.

ORNL-2685, pg 4-16

Authors

W. H. Cook, E. E. Hoffman

Abstract

For bearings, valves, and seals for operation in 1500 °F sodium systems, metals and metal alloys were found to be unsatisfactory because of their high wear rates, galling and self-welding. Some carbides, nitrides and oxides showed good resistance in static tests. Tungsten carbide and titanium carbide cermets showed the most promise as bearing materials in static, dynamic, and wear tests.

Title

Sodium Graphite Reactor Quarterly Progress Report June-August, 1953

Document No.

NAA-SR-878

Author

G. M. Inman

Abstract

Engineering was continued on the development of sodium cooled, graphite moderated type reactors. General studies were carried out as well as studies specifically devoted to the following: (1) Full scale power-only plant. (2) Thirty-megawatt pilot plant, the SGR. (3) Sodium reactor experiment, the SRE. This work consisted of theoretical analysis of various aspects of nuclear performance - economic investigations of different fuel elements, cooling system and plant arrangements - and experimental investigations related to the properties of certain materials and to the development of components. In addition to a summary of the general design features of the SRE, a program was prepared outlining the proposed use of this installation.

Title

Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending June 10, 1956.

Document No.

ORNL-2106, Parts 1-5

Authors

W. H. Jordan, S. J. Cromer, A. J. Miller, A. W. Savolainen

Abstract

Progress is reported on the aircraft reactor test (art) design - art physics - art instruments and controls - component development and testing - art, ETU, and in-pile loop operations - phase equilibrium studies - chemical reactions in molten salts - physical properties of molten materials - production of fuels - compatibility of materials at high temperatures - analytical chemistry - dynamic-corrosion studies - general-corrosion studies - fabrication research - welding and brazing investigations - mechanical-properties studies - ceramic research - nondestructive testing studies - heat transfer and physical properties - radiation damage - fuel recovery and processing - and critical experiments.

C-111

Title

Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending March 10, 1956.

Document No.

ORNL-2061, Pt. 1, 2, 3

C-112

Abstract

Synthetic lubricants in the UCON LB series were investigated to determine their stability as pump lubricating fluids in the art. Additional tests of the performance of the art sodium pump with water were made. Difficulties were encountered in high-temperature tests of the fuel pump with NaK. Two Inconel forced-circulation loops in which NaK was circulated completed 1000 hr of operation with a temperature gradient of 300 °F and a maximum fluid temperature of 1500 °F. Inconel tube-to-header joints brazed with Ni-Cr-Ge-Si low-cross-section alloys were tested in sodium and NaK in see-saw apparatus. Two Type 316 SS thermal-convection loops were operated with sodium to study the effect of a diffusion cold trap on the amount of corrosion and mass transfer observed in such a system. A series of differential-thermal-analysis was performed to determine the solubility of lithium in NaK. Screening tests of elastomers for possible use as valve seat materials in NaK circuits were initiated. Metallographic investigations are underway on NaK-to-air radiators that failed after various periods of service at high temepratures.

C-113

Title

Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending March 31, 1957

Document No.

ORNL-2274, Parts 1-5

Authors

W. H. Jordan, S. J. Cromer, A. J. Miller

Abstract

A summary of the progress on the ANP project for the quarter ending March 10, 1957 is presented. The work covers aircraft reactor test design, reactor physics, instrumentation and controls, component development and testing, reactor and facility construction, in-pile, art, and ETU operations, chemical analysis, compatibility of materials at high temperatures, metallurgy, general corrosion studies, fabrication research, welding and brazing investigations, nondestructive testing and inspection of materials and components, radiation damage and fuel recovery and reprocessing, and critical experiments and reactor shielding.

Title

Ventilation Systems at Atomics International (Sixth AEC Air Cleaning Conference, July 1959)

Document No.

TID-7593, pg 228-231

Author

A. R. Piccot

Abstract

A brief summary of ventilation systems employed on reactors, hot cells, and critical facilities designed and/or operated by Atomics International is presented. The two power reactors discussed are contrasted by use of a rather loose building containment system in one (sodium reactor experiment) and a very tight vapor container in the other (PIQUA organic moderated reactor). Similarly, of the two hot cells described, one operates with a comparatively large volume air flow, the other with a very low ventilation rate.

BNWL-1387

Title

Test of a 1 Inch Combined Valve Connect and Disconnect

Document No.

NP-7413

Authors

W. Milich, E. A. Schultz, E. C. King

Abstract

A 1 in. combined valve connect and disconnect intended for use in the primary coolant vent lines of the SIR Mark B System was subjected to a series of performance tests. These tests included valve cover welding tests, a series of static pressure and seat leakage tests, and a series of sodium cycling tests. The static pressure and seat leakage tests were run at room temperature, using nitrogen at 100 psig and 200 psig. Additional static tests were run at 500 °F using sodium at 200 psig. The valve was subjected to 10 cycling tests, wherein sodium at 500 °F, 100 psig was pumped through the valve for 10 min periods. After each cycle the valve disconnect joint was dismantled and rebuilt with a new O-Ring. Performance in all cases was satisfactory.

Title

Hazard Analysis of a Sodium-Water Reaction in an SRE Pipe Gallery

Document No.

NAA-SR-MEMO-7743

Author

H. B. Dietz, Jr.

Abstract

This hazard analysis will consider the effects of a possible sodium-water reaction in the SRE pipe galleries. The presence of a small amount of oxygen in the gallery nitrogen cover gas, which could enter into the reaction, will also be evaluated. The possibility of water leakage from the earth into the underground galleries is remote and the chance of a sodium-water contact is even more remote - however, this hypothetical study was made to evaluate the results of such an occurrence and to recommend protective measures. In approaching the problem, all available information which concerned the history of the galleries was studied including any water leakage and its method of entry. To establish a maximum sodium leak rate, the log books and reports were searched and the applicable data were analyzed. A literature study of the sodium-water-oxygen reaction was made and calculations were performed to determine the effects of such a hypothetical incident on the personnel and system. The beneficial recommendations which accrued as the analysis progressed are included in Section XII of this report.

Title

Performance of a Frozen Sodium Seal on a 120 gpm Duriron Centrifugal Pump

Document No.

MSA-MR-61

Author

V. K. Heckel, E. C. King

Abstract

Operation of a 120 gpm centrifugal pump installed in a 3-in. test loop was described. Performance of the pump shaft seal and instrumentation were also described. Comments regarding the liquid level indicator, pressure gage and flow control valve were given.

Title

Sodium Pump Development Study (Proceedings of the Sodium Components Information Meeting, Palo Alto, August, 1963)

Document No.

SAN-8002, pg 258

Author

J. H. Wright

Abstract

A brief discussion is given of current technology relating to design and operation of pumps for sodium service. Conceptual drawings are included of a proposed mechanical shaft sealed pump, and of a sodium pump test facility with two independent parallel loops of 60,000 gpm each.

Title

Development of a Continuous Meter for Oxygen in Sodium (Proceedings of the Sodium Components Information Meeting, Palo Alto, August, 1963)

Document No.

SAN-8002, pg 276

Authors

H. Steinmetz, G. Stern

Abstract

An electrochemical cell is described for measurement of oxygen in sodium. Upper temperature limitations are 600 °F for a zirconia cell, and 700 °F for a thoria cell. Temperatures below 500 °F result in resistances too high to permit convenient measurements. A representative curve, showing the response of an electrochemical cell to changes in oxygen content as fixed by cold trap temperature, is shown.

Title

EBR-II Materials Experience (Proceedings of Sodium Components Development Program Information Meeting, Chicago, June 1965)

Document No.

CONF-650620, pg 127

Author

C. H. Scheibelhut

Abstract

The 2-1/4 Cr-1 Mo steel section of the EBR-II reactor sodium piping system was installed without pickling to remove an estimated 400 lb of mill scale. About 4 months of draining, temperature cycling, and cold trapping removed about 130 lb of sodium oxide. Large amounts of particulate material were present in the sodium, which was thought to be temperature expanded pieces of mill scale. A description of the EBR-II secondary sodium system is included. A description is also included of a gripper mechanism which is used to grasp subassemblies that are removed and inserted in the reactor.

Title

Solutions of Alkali Metals in Polyethers and Vapor Pressures of Alkali Metals

Document No.

NYO-2182

Authors

B. R. Sundheim, F. Cafasso

Abstract

The absorption spectra, the electron paramagnetic resonance spectra, the electrical conductivity and some solubility relationships have been studied in two alkali metal ether systems. The results are compared to the chemical, optical and magnetic properties of similar solutions in liquid ammonia or amine solvents, in order to determine the general nature of the ether solutions. A plausible interpretation has been put forward. This report also includes a description of a spectral cell to be used in the spectrophotometric measurement of the absorption of sodium or potassium resonance lines above NaK vapor. The cell was designed and built to make vapor pressure measurements.

Title

Effect of High Temperature Sodium on Austenitic and Ferritic Steels. Test Facility Design and Operation Procedures. (Topical Report No. 1)

Document No.

MSAR-63-161

Authors

R. C. Andrews, K. R. Barker

Abstract

MSA Research Corporation is engaged in a continuing test program studying the effects of air, helium and sodium on Type 316 SS and 2-1/4 Cr-1 Mo steel under creep, creep-to-rupture and fatigue conditions. The design and fabrication phase of this program has been completed. Two identical but separate and independent loops have been designed and fabricated at MSAR to accomplish the above program. One loop is engaged in testing 316 SS specimens and the other in testing 2-1/4 Cr-1 Mo steel specimens. Both loops are fabricated from 316 SS. The test facilities are complete for obtaining creep, creep-to-rupture and fatigue data in dynamic sodium systems at elevated temperatures. The systems have been constructed and are in operation. The test results will be reported in separate topical reports as the various phases of the test program are completed. This report sets forth the design criteria, a description of the test facilities and the operational procedures of these test facilities.

Title

Flow Controller Evaluation for Sodium Service

Document No.

NAA-SR-7534

Author

R. A. Winborne

Abstract

Tests were conducted to evaluate the performance of a 10-in. flow controller in sodium at temperatures up to 1200 °F. Torque tube reliability was determined by extensive mechanical cycling tests and further checking with a helium mass-spectrometer leak detector. Flow control performance and valve vibration characteristics were tested and evaluated and flow rate limits were established. The valve assembly was subjected to a thermal shock test program simulated to represent the most severe conditions likely to be encountered during a minimum 10-yr lifetime.

Title

Nonequilibrium Electric Conductivity of Wet and Dry Potassium Vapor

Document No.

APL-TDR-64-106

Authors

A. W. Rowe, J. L. Kerrebrock

Abstract

The nonequilibrium electric conductivity of two-phase potassium vapor has been studied both theoretically and experimentally. This study is important in connection with the design of rankine-cycle alkali-metal magnetohydrodynamic (MHO) generators. The theoretical work indicates that small condensation droplets (mean radius less than 50 Å units) will detrimentally affect the achievement of elevated electron temperatures by Joule Heating, and will also reduce the vapor ionization associated with this temperature. These effects, which arise from large electron-drop collision energy losses, are negligible when the mean droplet radius is greater than 200 Å units. A high-temperature (maximum temperature equals 2000 °K) closed-loop facility has been constructed for the study of nonequilibrium ionization in wet and dry potassium vapor. Preliminary experimental results on the electric conductivity of potassium in a subsonic test section are given. The results suggest that the condensation existed as a myriad of nucleation-sized (mean radius approximately 10 Å units) droplets which reduced the nonequilibrium conductivity appreciably below that of the dry vapor.

Title

Trends in Sodium Equipment

Document No.

NUCL-19-65-61

Authors

R. W. Dickinson, J. B. Williams

Abstract

Experience with mechanical components of sodium-cooled reactors is summarized. New design approaches, overcoming thermal-stress and other problems, and exploiting sodium's superior heat transfer ability are outlined. Heat exchangers, steam generators, piping, nozzles, valves, pumps, and instrumentation are discussed. A chronology of sodium-cooled reactors is given.

Title

The Pressure at the Bottom of the Shaft Seal in the Main SRE Sodium Pumps

Document No.

NAA-SR-MEMO-2172

Authors

R. W. Atz, C. W. Griffin

Abstract

A series of water tests were performed on the SRE main secondary sodium pump to measure pressure on the pump shaft seal during operation with sodium. As a result the recirculatory flow holes in the impeller were enlarged sufficiently to reduce the seal pressure from 45% to 15% of pump discharge pressure.

Title

The Separation of Potassium Isotopes by Molecular Distillation

Document No.

K-1650

Authors

R. McGill, R. Browell, J. Grisard, E. Blumkin, E. VonHalle

Abstract

The technical feasibility of accomplishing the separation of the principal potassium isotopes K-39 and K-41 on a large scale by the molecular distillation of potassium metal has been established. Following preliminary tests, a 10-stage experimental molecular still was designed, constructed, and operated to locate the optimum operating temperature and to investigate the effect on the separation factor of several process variables. The experiment was successful and led to the design and construction of a 15-stage prototype cascade unit for operation at the optimum evaporating temperature of 270 °C. This equipment is now in operation. Conceptual design of a molecular distillation cascade to produce one ton per year of potassium enriched to 99.93% K-39 has been completed - analysis of the cascade construction and annual operating costs leads to an estimated unit cost of product of about $420/lb economic effects of operating this cascade at other K-41 depletion factors and other production rates are discussed.

Title

SNAP-8 Mercury Pump Motor Assembly Thermal Analysis

Document No.

AGC-TM-395-64-1-208

Author

E. B. Bridges

Abstract

A steady-state thermal analysis of the heat generated by a carbon to steel shaft seal is presented. The seal serves to exclude mercury at 270 °F from the windings of a 7800 rpm electric motor. Heat developed by the seal and conducted from the motor is rejected to space through a liquid to liquid heat exchanger. Several different heat exchanger designs were reviewed to obtain the maximum heat transfer in the space available. The selected design was shown to cool the seal faces to 285 °F using 2600 lb/hr of 210 °F having an inside diameter of 0.08 in.

Title

Transport and Chemical Control in the Dissolution of Metals in Mercury

Document No.

AERE CE/R 1998

Authors

J. A. R. Bennett, J. B. Lewis

Abstract

Dissolution rates of tin, lead, and zinc in mercury were determined. Cylindrical metal specimens remained stationary - the cylindrical, metal container with the solvent was rotated. Transport (diffusion) is the controlling process in the solution of tin and lead. Both transport and chemical solution rate are significant in the solution of zinc.

Title

Thermal Cycling and Leakage Tests of 2-in. Valves for Sodium Service

Document No.

NAA-SR-5275

Author

C. J. Baroczy

Abstract

Tests were performed to determine the effect of thermal cycling on the across-the-seat leakage characteristics of valves considered for use in the auxiliary coolant system of the Hallam Nuclear Power Facility. Twelve 2-in. valves were thermally cycled in a sodium loop between 300 and 650 °F and periodically tested for across-the-seat sodium leakage. Five different valve types, representing eight manufacturers, were sodium-leak tested at pressure differentials of 10 to 30 psi, and temperatures between 395 and 685 °F. Five bellows-seal globe valves had a leakage pattern which indicated a maximum leakage rate at a pressure differential of 20 psi. The torque-tube valve had considerable leakage in some tests owing to inconsistent valve closing. A freeze-seat valve, a solenoid valve, and four flexible-seat valves had no leakage. The flexible-seat valves were particularly easy to open and close. No valve-stem sodium leakage was observed for any of the valves tested.

Title

NaK Free Convection Cooled Shaft Freeze Seal for SRE Pumps

Document No.

NAA-SR-MEMO-3984

Author

F. Perez

Abstract

The feasibility of using NaK for cooling of shaft freeze seals in SRE sodium pumps by natural convection was demonstrated analytically. The analysis showed NaK could remove 2.6 kW of heat from the shaft seal region of the pump while it was operating at 1000 °F and 1250 rpm.

Title

Sodium Level Detectors in the Enrico Fermi Atomic Power Plant

Document No.

APDA-300

Author

G. H. Reicks

C-126

Abstract

Two types of continuous level detectors are installed in the Enrico Fermi Atomic Power Plant to measure sodium levels. One type is the Moore Special Model 15 bellows-type detector, and the other is the Taylor NaK-Filled Diaphragm Detector. During the period of the conceptual design of the Fermi Plant, all known level detectors for high-temperature operation in liquid metals were evaluated. Moore Special Model 15 Detectors were selected from those available as most nearly fulfilling the rigid specifications that had been previously established. Several years later, the Taylor NaK-Filled Detectors were evaluated and also were installed in the Fermi Plant. The Moore Special Model 15 Detectors will measure levels with accuracies of approximately ±3 in. of reading. An advantage is that the detectors are capable of measuring levels with a cover gas pressure below atmospheric and, with the properly observed precautions, down to a full vacuum. The Taylor NaK-Filled detector has operated satisfactorily with a high degree of accuracy, small temperature shift, good linearity and reproducibility, and with a minimum of maintenance required. Both of these detectors are difficult to install and shield properly as well as difficult to remove and replace once they are installed. They have moving parts which could create maintenance problems, and accessory instrumentation is required for remote calibration.

BNWL-1387

Title

Solubility of Sodium Monoxide in Liquid Sodium

Document No.

KAPL-1653

Authors

O. N. Salmon, T. J. Cashman, Jr.

Abstract

The solubility of sodium monoxide in liquid sodium in the temperature range 100 to 500 °C has been obtained, using the methods of Pepkowitz and Judd, by analysis of the oxygen content of sodium samples taken from liquid sodium in equilibrium with excess sodium monoxide. The method of least squares has been applied to the data to derive the following solubility equation:

$$\mathrm{Log}_{10}\ Y = -1816/T + 1.266$$

in which Y is the solubility of Na_2O in sodium, expressed as wt% oxygen in sodium, and T is the absolute temperature of the sodium in K. A brief description of corrosion of Type 347 SS specimens by sodium saturated with sodium monoxide at 500 °C is also presented.

Title

Condensation of Metal Vapors. Mercury and the Kinetic Theory of Condensation.

Document No.

ANL-6948

Author

D. J. Wilhelm

Abstract

Nusselts and improved theoretical film-condensation heat transfer rates for metal vapors were roughly 5 to 30 times as large as observed experimental rates. The literature survey and subsequent analyses revealed serious shortcomings in the kinetic theory of condensation. Supporting reasons were not found for the common assumption that the condensation and evaporation coefficients are identical. Design details and operating difficulties of mercury vapor rotating disk condenser test (which comprised the original and a major portion of this program), aimed at evaluating the vapor-liquid interfacial resistance, have been included. A bibliography of 295 references is given.

Title

A Study of Various Types of Freeze Type Liquid Metal Valves

Document No.

NAA-SR-MEMO-2920

Author

E. O. Dryer

Abstract

A discussion of various valve sealing and flow controlling arrangements is given. The use of bellows, freeze seals, seal welds, and packing are considered in an attempt to develop more reliable valves for liquid metal service.

Title

Metallurgical Investigations of Sodium Heat Transfer Rig

Document No.

AERE R/R 2190, pg 15

Author

A. G. Ward, J. W. Taylor

Abstract

A sodium heat-transfer loop operated approximately 1400 hr at 600 to 1100 °F. Results of metallurgical investigations of failures which occurred during operation and terminal examination of components are given. For comparison, results of 1000 hr static tests with low and high oxygen concentrations in sodium at 300, 450, and 600 °C are reported. Stabilized austenitic stainless steel (18-8-1) used in the loop contained free carbide. During loop operation decarburization occurred at grain boundaries. No corrosion of nickel was observed. This information appears in an appendix to accession LM-00077, which reports heat transfer results and operating experience of the test.

Title

Experience Obtained on a Liquid Sodium Heat Transfer Rig 1954/1956

Document No.

AERE R/R 2190, pg 1

Authors

E. C. Firman, R. I. Hawes, D. W. G. Sturge

Abstract

Nusselt Numbers are correlated with Peclet Numbers for sodium flowing at moderately low Reynolds Numbers in a double annulus heat exchanger. Results qualitatively confirm findings of others who reported coefficients below theoretical minimum. Effect of tube eccentricity, gas entrainment, and oxide deposits on depressing the heat transfer coefficient are discussed. Loop operated approximately 1400 hr. General operating experience is described, e.g., heater failure at copper-steel interface, flow blockage caused by oxide deposits. Results of metallurgical investigation of this loop are reported in accession LM-00078.

Title

Mercury Corrosion Loop. (SNAP-8 Topical Materials Report for 1963. Vol. 2, Component Materials Development.)

Document No.

AGC-2822, Vol. 2, pg 53-89

Author

R. S. Carey

Abstract

An adjustable choked nozzle located in the superheated mercury-vapor section of component test loop 2 was tested for 400 hr. This run resulted in corrosion-product buildup at the venturi throat and on the pintle tip. The operating characteristics of a mercury chempump evaluated during this test were found to be satisfactory. Leaks developed at the O-Ring seal in the impeller case and an auxiliary weld-seal assembly was fabricated to replace the O-Ring. The designing of corrosion loops 3, 4, and 5 was completed and is fully described herein. Four thermal-convection loops of 9 Cr-1 Mo steel were fabricated, operated, and evaluated to establish a basis for the design of a mercury-corrosion-product separator. The corrosion pattern was determined for mercury boiling and condensing temperatures of 700 and 1075 °F. Separator concepts using iron and columbium wool and magnets were evaluated in two of the thermal-convection loops. The results indicate that an effective liquid separator is attained through the use of a large surface area and long mercury dwell time. A cyclone-type vapor-phase corrosion-product separator was designed and fabricated for testing in CTL-2. Design parameters are presented.

Title

SNAP-8 Topical Materials Report for 1963. Vol. 2, Component Materials Development

Document No.

AGC-2822, Vol. 2

Author

R. S. Carey

Abstract

Materials investigations conducted in support of the design, development, fabrication, and testing of components for the SNAP-8 power conversion system are reported. Details of the work are noted in accessions LM-00326 through LM-00330.

Title

Study of Liquid Metal, NaK-77, for Application in Flight Control Systems. Vol. 3. Investigation of System Applications and Components

Document No.

ASD-TDR-62-597(Vol. 3)

Authors

R. C. Kumpitsch, J. R. Granan, P. J. Kroon

Abstract

The use of liquid metals as the working fluid in hydraulic systems for high temperature service is discussed. Applications for these systems are cited for space vehicles, missiles, aircraft and nuclear propulsion devices. Preliminary development of pumps and valves for this service is described. A ten-stage centrifugal pump operated in NaK for 20 hr before a bearing seizure halted the test. A vane-type positive displacement pump was tested in 100 °F NaK for 5 hr and in NaK at 500 °F for 2 hr. Tests were performed on the shaft seal and bearings prior to their use in the pump. Nak at 200 °F was supplied to the bearing and seal tester at pressures from 5 to 45 psi. Flow through the bearing was 0.3 gpm. The seal held pressures from 5 to 10 psi and leaked 15 cm^3 NaK during tests at shaft speeds from 3000 to 36,000 rpm. Seal and bearing surfaces were in satisfactory condition at end of test. A poppet-type valve and a two-stage valve were fitted with electromagnetic pump to use NaK as the actuating fluid. No performance data was given. Attempts to use alumina for pump throat material were unsuccessful. Several materials compatibility tests were performed to determine wear rate, friction coefficient and bearing properties for use in 1400 °F NaK. Flame-plated carbides may be suitable provided the application, finishing and substrate preparation is properly done.

Title

Characteristics and Performance of 5000-gpm ac Linear Induction and Mechanical Centrifugal Sodium Pumps

Document No.

A(CONF. 15)P/2158

Authors

O. S. Seim, R. A. Jaross

Abstract

A centrifugal pump and a linear induction electromagnetic pump were tested in sodium at 900 °F and 850 °F respectively. Both pumps developed a head of 40 psi at a flow of 5000 gpm. The centrifugal pump was equipped with a pressurized bearing and was driven by a two-speed electric motor. The electromagnetic pump had a thermal barrier and water cooling to permit 850 °F operation of its class "H" electrical insulation. Test facilities for the pumps are described.

Title

An Investigation of Nickel-Base Brazing Alloys and Brazed Joints for Service in Liquid Sodium

Document No.

NAA-SR-11326

Author

Suk K. Lee

Abstract

A series of nickel-base brazing alloys were tested in a liquid sodium thermal convection loop for a period of 10,000 hr at temperatures of 1200 and 700 °F. The tests were conducted to establish the feasibility of a brazed calandria joint concept for the sodium graphite reactor application. The results indicated that the Ni-Cr-Si-B (CM53) and Ni-Si-B (CM52) were the most resistant to sodium attack and that the Ni-Cr-Si (CM60) would only be suitable for short term use (less than 1500 hr) or lower temperature (below 700 °F) applications. Although CM52 and CM53 experienced a loss of both silicon and boron, the depletion was not accompanied by the formation of voids. Therefore the integrity would not be impaired. The copper-manganese brazing alloys were attacked by sodium even with the exposure temperatures limited to 700 °F. Cobalt additions to this alloy appeared to have lowered its sodium resistance. The apparent reaction found at the surface of specimens exposed at 1200 °F included loss of the minor alloying constituents, silicon and boron. This loss of alloying elements was confirmed by X-ray fluorescent analyses. The leaching of these two elements from the alloy by the high temperature sodium was apparently accomplished without the development of visible vacancies in the depleted zones. A comparison of the 5000-hr-1200 °F

Abstract (contd)

exposure test results indicated that Nicrobraz 135 is far more stable than CM52 in sodium. This is reasonable because the silicon and boron content of Nicrobraz 135 is substantially lower.

C-136

Title

Plugging Leaks Between Water and Third Fluid System (NaK)

Document No.

MSA-MR-117

Authors

S. J. Rodgers, J. W. Malsteller

Abstract

An investigation was made to find a NaK additive which could be used to seal NaK-water leaks in systems such as the S2C evaporator and superheater. A mixture of 1 wt% iron oxide and 0.2 wt% asbestos plugged a leak of 0.3 liter NaK/hr, but only partially plugged a 1.0 liter NaK/hr leak. The leak was completely plugged when operated in the vapor phase, but the seal failed when the leak area was exposed to water. The seal formed in the 0.3 liter NaK/hr leak held at a Δp of 560 psi for 1 hr and at a Δp of 250 psi overnight. An iron oxide-asbestos mixture can be used as a sealant but will have limited applications.

Title

Calandria Core Weld Joint Development

Document No.

NAA-SR-9821

Author

J. G. Roberts

Abstract

With the advent of the large SGR suspended reactor with a calandria core containing a lattice work of process channels, it became necessary to study and design methods for the installation, maintenance or replacement of the process channels. With regard to welded joints and machine parting characteristics, a development program was initiated to determine joint configuration and replacement processes. The design and initial test of cutting and welding equipment developed to remotely cut and reweld the bottom process tube joint are discussed in this report.

Title

The Sampling of Alkali Metal Systems with the Modified MSA Sampler

Document No.

ORNL-2147

Authors

G. Goldberg, A. S. Meyer, Jr., J. C. White

Abstract

Two significant modifications have been made to the MSA Alkali Metal Sampler: (1) the substitution of vacuum-tight, modified Wilson Seal for the packing gland, and (2) the replacement of the gate valve with a vacuum-tight, teflon-packed, Jamesbury Valve. A complete description of the modified sampler and a stepwise procedure for its application to the sampling of dynamic or static systems of alkali metals are given.

Title

Improved Design for Fuel Handling on Sodium Graphite Power Reactors

Document No.

NAA-SR-MEMO-5503

Authors

T. F. Edziak, R. O. Crosgrove, J. F. Stolz

Abstract

Operational experience at the SRE and evaluation of alternate methods have resulted in the development of an optimized fuel-handling system for the Hallam Nuclear Power Facility Reactor. Increased reliability, easier maintenance, improved safeguards, and broader flexibility are some of the characteristics of this system, which features atmosphere seals, for use in a conventional unshielded building with an air atmosphere.

Title

Alkali Metals Boiling and Condensing Investigations. Volume II. Materials Support

Document No.

GE63FPD66

Authors

J. W. Semmel, Jr., W. R. Young, W. H. Kearns

Abstract

Materials support was provided for alkali metals boiling and condensing investigations that required the construction of a 300 kW heat transfer loop from L-605 and a 100 kW loop from the CB-1 zirconium alloy. In addition to the preparation of specifications for these materials and assistance in constructing the heat transfer facilities, experimental investigations were conducted to document several aspects of the behavior that were pertinent to the loop construction and operation. Experimental work was performed in the following areas: oxidation of L-605 in air and combustion gas compatibility of L-605 with ceramic insulation materials, grain growth in bent L-605 tubing, aging and embrittlement of L-605, corrosion of L-605 by potassium, welding L-605, joining L-605 tubing to molybdenum tubing, corrosion and diffusion bonding of stellites No. 6 and No. 12 hard facing materials in potassium, welding Cb-1 Zr, joining Cb-1 Zr tubing to Type 316 SS tubing, emittance of Cb-1 Zr and Lb-2 and TiO_2-Al_2O_3 coatings applied to Cb-1 Zr, spraying alumina and molybdenum on Cb-1 Zr tubing to form a resistance heating element, and joining tantalum to alumina by brazing and diffusion bonding.

Title

Corrosive Environment Ceramic to Metal Seals for Space Power Applications. (Sixth National Symposium, Materials for Space Vehicle Use, Seattle, Wash., Nov. 1963, Vol. 3)

Document No.

CONF-375-Vol. 3-p/16-63

Authors

L. Reed, R. C. McRae

Abstract

Work on the problems of fabricating seals for high temperature operation (300 to 1000 °C) in atmospheres of lithium, sodium, potassium, cesium or mercury for alternator bore seals has to a large degree defined the problem areas. Alumina ceramics brazed to a metal member using the conventional techniques practiced in the electron tube industry are unsatisfactory in all cases mentioned. Alkali metal vapors leach out silica which is present in the secondary phases of most ceramic bodies. With a mercury vapor environment the main avenue of attack is through the braze when using alloys containing copper, silver, and gold. In all cases the final working assembly presents problems which are both metallurgical and mechanical in nature. The rigid dimensional tolerances on the thin ceramic membrane in the electromagnetic flux gap of alternators requires special considerations in design, metalizing and brazing. Special silica free ceramic bodies have been fabricated and are being investigated for the alkali metal vapor applications. Metalizing and brazing materials inherently resistant to the appropriate operating atmosphere are being evaluated for mutual compatibility, vacuum tightness, and mechanical properties with various ceramic-to-metal seal test assemblies. While potentially satisfactory ceramic-metal bore seal

Abstract (contd)

structures for mercury and potassium environments at 500 °C are now available, developmental work is not complete. The mechanical feasibility of large, thin walled (0.050 in. or less), cylindrical assemblies has not been established.

Title

Characteristics of Fast Settled Bed Reactors. (From Conf. on Breeding Economics, and Safety in Large Fast Power Reactors, Argonne, Ill., Oct. 1963)

Document No.

BNL-7592

Author

J. Chernick

Abstract

The fast settled bed reactor is an example of a high temperature, particulate fuelled, breeder concept on which preliminary studies are being carried out at Brookhaven. The reactor may be regarded basically as a pot of small diameter, densely packed fuel lumps through which a liquid metal is pumped downward for heat removal. During reactor shutdown, the flow is reversed and the bed is fluidized to facilitate removal, reshuffling or reprocessing of the fuel and fertile material. Preliminary design studies have been based on UC-PuC spheres of 1/4- to 1/8-in. diameter cooled by liquid sodium at velocities of about 2 ft/sec. Because of the substantial pumping power required to obtain such velocities in large power reactors, some compromises in core geometry have been studied.

Title

Section IV. Power Reactor Technology. Components

Document No.

PRET-8-133-65

Abstract

A description of the design and the results of testing of the EBR-II control-drive mechanisms are given. Problems in the selection, fabrication, installation, and testing of the NPR primary-loop process tubes and fittings are described.

Title

Liquid Metal Fuel Reactor Experiment. Testing of Small Croloy 2-1/4 Bellows Designed for Liquid Metal Service

Document No.

BAW-1083

Author

F. J. Soltys

Abstract

Tests were conducted to determine the feasibility of using a bellows-type sealing mechanism in various LMFRE components. Products of four companies were tested under conditions designed to simulate reactor operations. Only one model, a single-ply welded-disc bellows supplied by company "A", met minimum requirements. Plans to investigate the nature of mass transfer under test conditions were dropped because there was no evidence of this phenomenon.

Title

Development and Testing of a Fusible Metal Joint for a Fast Reactor

Document No.

Sou. Tech. Con.-19-27-65

Authors

J. Hainzelin, C. P. Zaleski

Abstract

The eutectic alloy of bismuth and tin (58% Bi, 42% Sn) was chosen as joint material for two rotary closure plugs for the tank of the fast reactor rapsodie. Results of mechanical tests ultimate tensile strength, elongation, compressive strength, cracking, Brinell Hardness, and impact strength on this alloy are given. The application procedure adopted to obtain maximum adherence of the Bi-Sn alloy to adjacent 18-8 stainless steel surfaces is described. Description and results of pressure tests on a mockup of the joint assembly to determine gas tightness of joint in both liquid and solid states are presented.

Title

Design Development Tests of Some Components for the 1C MW_e SDR

Document No.

NDA-84-21

Authors

C. Bolta, B. Minushkin, H. Steinmetz, O. Sullivan

Abstract

A sodium leak from a fuel-coolant tube and a water leak from a calandria tube were deliberately induced to study the effect of these leaks on a barrier. A program directed at improving leak detection devices and developing better methods of leak detection is reported. Prototype fuel-coolant tube closure devices were experimentally evaluated at simulated reactor operating conditions to obtain useful design data. Devices to join the calandria tubes to the tube sheet while maintaining a gas-tight seal were experimentally evaluated.

Title

Molten-Salt Reactor Program Quarterly Progress Reports

Document No.

ORNL-2799

Author

H. G. MacPherson

Abstract

Operational testing of the modified Fulton Sylphon Bellows Mounted Seal in a PKP type of centrifugal pump circulating NaK was suspended in the quarter ending April 30, 1960. The reason for the suspension was that oil leakage from the lower seal of the shaft had increased. The accumulated operating time at abort was 18,834 hr at 1200 to 1225 °F. This abstract covers ORNL-2799, 2890, and 2973.

Title

Molten-Salt Reactor Project Quarterly Progress Report for Period Ending January 31, 1959

Document No.

ORNL-2684

Author

H. G. MacPherson

Abstract

Progress on the MSRE for the quarter ending January 31, 1959 is reported. A small frozen lead pump seal on a 3/16-in. diameter shaft operated for 5000 hr with no lead leakage. Test operation of a similar seal on a 3-1/4-in. diameter shaft was initiated. Mechanical property tests were made on Inconel and Hastelloy specimens that were carburized by exposure for 4000 hr at 1200 °F in a system containing sodium and graphite. Carburization decreased the tensile strength and elongation of Hastelloy N, increased the tensile and yield strength of Inconel but decreased the elongation. Studies of composites of Hastelloy N and Type 316 SS have shown the alloys to be compatible at temperatures to 1800 °F, indicating usefulness in fused salt to sodium heat exchangers.

Title

Design of Impurity Traps for FFTF Flow Circuits (Progress Report Conceptual Design Fast Flux Test Facility, Vol 3, Design Analysis)

Document No.

BNWL-CC-175, pg III-1

Author

J. M. Davidson

C-145

Abstract

Reliable operation of sodium heat-transfer systems requires that control of fluid purity be maintained. Removal or control of oxygen and carbon are primary considerations. Sampling of the fluid, both discrete and continuous, to determine the levels of various impurities is mandatory. Analysis of the available data led to the selection of the cold-trap design shown. The design includes: (1) an integral economizer unit, (2) a bypass valve for control of heat rejection and resultant control of the trapping temperature (3) stainless steel mesh trapping region, and (4) NaK circuit as the trap coolant. Basically, the hot carbon trap consists of a large surface area of stainless steel in a container through which the sodium passes. A convenient means of obtaining a large surface area in a reasonable volume is to employ corrugated stainless steel foil and to separate the layers of corrugated material by flat strips of foil. The hot oxygen trap is similar in function and design to the hot carbon trap. It consists of a large surface area of zirconium in a container located in a bypass sodium flow system. In the FFTF flow circuits with a 1000 °F maximum operating temperature, the hot oxygen traps can be operated at 1200 °F. For the closed-loop systems, it may be advantageous in terms of hot-trap

BNWL-1387

Abstract (contd)

container and heater design to operate at 1500 °F when the fluid temperature at the loop exit is 1400 °F. Zr-Ti and Zr-Al alloys are better oxygen getters than pure zirconium.

Title

Development of a Large Metal Ultrahigh Vacuum Simulation Chamber

Document No.

NAA-SR-6490

Author

F. J. Kamensky

Abstract

A large ultra-high vacuum chamber was built for environmental testing of components for the SNAP Program at temperatures as high as 1000 °F. The chamber employs diffusion, electronic, and cryogenic pumping to handle high gas loads at high temperature and ultra-high vacuum. A unique internal heating system, connections, assemblies, flanges, and test set-up jigs are described in detail.

C-147

Title

Development of High-Temperature Alkali-Metal-Resistant Insulated Wire. Quarterly Progress Report No. 1.

Document No.

AD-404273

Authors

E. S. Bober, R. E. Stapleton, W. H. Snavely

Abstract

The program effort is directed toward the development of an insulated electrical conductor resistant to saturated potassium (850 °C) and mercury (538 °C) vapors. Surveys of candidate insulators, conductors, and coating methods are substantially completed. Exposure of a number of candidate insulators, uninsulated conductors, and metal-ceramic seals in potassium and mercury vapor was initiated to provide preliminary corrosion data. The mercury vapor exposure tests are completed. Evaluation of a number of coating methods was initiated and preliminary results from the work are promising.

Title

Monthly Technical Report for July, 1962

Document No.

PRDC-TR-61

Abstract

Research and development progress on the Fermi Reactor Project for July, 1962 is reported. Pertinent topics include tryouts and modifications to the offset handling mechanism, vessel and shielding engineering, studies on the thermal conductivity of graphite in sodium, reactor kinetics studies and experiments to determine the reason for sodium hammer in the system piping. In the field of system safety, experiments involving the reactions of sodium with environmental moisture, oxygen, and carbon dioxide were performed.

Title

A Review of the Operation of the Dounreay Fast Reactor

Document No.

CONF-BNES-66-2/2

Authors

K. J. Henry, A. G. Edwards

Abstract

A review of DFR Operation is presented. Various statistics on availability, production and operating hours are presented. Performance of the fuel and reactor components are discussed. Recent reactor modifications are presented. Coolant purity and fission product contamination are reviewed.

Title

Liquid Metals Technology Abstract Bulletin for the Period September, 1962 to February, 1963

Document No.

LM/TAB-21

Abstract

An abstract bibliography on current (9/62 to 2/63) liquid metals technology literature is presented. Forty-nine abstracts of published literature in this field are included.

Title

Liquid Metal Cooled Reactors. (Reactor Handbook, Second Edition, Vol. 4, Engineering.)

Document No.

BOOK-IP-612/681-64

Authors

S. McLain, J. H. Martens, L. E. Crean, W. E. Parkins

Abstract

Problems associated with the use of liquid metals as heat transfer media in nuclear systems are discussed. The advantages and disadvantages of the liquid metals as coolants are discussed, with emphasis on Na and NaK systems. The problems include mechanical, metallurgical, and operational aspects of structural materials, fuels, moderators, control and safety elements, pumps, piping, valves, heat exchangers, steam generators, traps, and instrumentation. Brief descriptions of the principles of liquid metal-cooled reactors are given.

Title

Pumps, Piping, and Fittings - Liquid Metals. (Reactor Handbook, Second Edition, Vol. 4, Engineering)

Document No.

BOOK-IP-116/120-64

Authors

S. McLain, J. H. Martens, C. F. Bonilla, R. C. Werner

Abstract

Pumps, piping, and fittings for light water, heavy water, and liquid metals are discussed. Design and fabrication of tanks and pressure vessels are discussed.

Title

Monthly Technical Report, Report Period - July, 1958

Document No.

PRDC-TR-13

Abstract

Research and development progress on the FERMI Reactor Project for July, 1958 is reported. The design and fabrication status of major reactor components is shown in chart form. Other pertinent activities reported include the corrosion test loop operation, and vessel fabrication status. Testing of temperature and pressure instrumentation, a sodium-water reaction test, a test of a cerrobend rotating shaft seal, and ultrasonic cleaning tests, are also described.

Title

Design of an Experimental Bowable Fuel Element for the SRE

Document No.

NAA-SR-10152

Author

C. L. Peckinpaugh

Abstract

An experimental, bowable fuel element was developed to study temperature oscillations in the second core loading of the SRE. The bowable element featured two sets of wire spring mechanisms which were remotely operated during at-power reactor operation. One set bowed the five fuel rods outward to the coolant channel wall while the second set extended between the fuel rods to the channel wall. Miniature thermocouples were mounted on the springs and throughout the cluster. The spring mechanisms operated successfully during full power reactor operation in a high-temperature sodium environment. Epoxy seals for thermocouple penetrations through shield plugs were adequate. However, difficulty was experienced with core gas leakage past dynamic O-Ring seals for spring actuators. Radiation streaming from straight-through gaps in shield plug penetrations from actuators and thermocouples was below permissible exposure rates. The miniature thermocouples had a 92% reliability over the 1 month of experimental operation.

Title

Quarterly Progress Report on Reactor Development, 4000 Program

Document No.

ANL-5398

Abstract

Progress is reported on the operation of EBR-I and the design of EBR-II. A pin type fuel element, designated for use in EBR-II is described and results of hydraulic testing and irradiation effects on the fuel element are discussed. The correction of sealing problems on an EM pump is reported. The corrosion test loop was operated for purification purposes, then used to study the mass transfer effects of sodium on irradiated uranium.

Title

Sodium Technology and Equipment of the BN-350 Installation (British Nuclear Energy Society. London Conference on Fast Breeder Reactors, May 17-19, 1966)

Document No.

CONF-BNES-66-58/1

Authors

A. I. Leipunskii, M. S. Pinkhasik, Y. E. Bogdasarov, V. M. Poplavskii

Abstract

This paper is devoted to a detailed description of the main engineering equipment in the BN-350 installation and the experimental work carried out in support of the chosen designs, and to an explanation of certain problems of sodium technology. The design and experimental work on the main circulating pumps, the intermediate heat exchanger and the steam generator are examined.

Title

Development Tests of the TP-1 Liquid Metal Turbopump and Components

Document No.

PWAC-318

Authors

H. V. Marman, R. S. Lombard, P. G. Standley, C. W. Grennan

Abstract

The development status of a 3000 gpm liquid metal turbopump, is described in this report. This includes discussions of the various pump and turbopump tests conducted in water and liquid metal.

Title

Power Losses and the Initial Torque of a Shaft in a Frozen Sodium Seal

Document No.

SJAE-10-376-61

Authors

A. V. Drobyshev, N. M. Turchin

Abstract

Measurements of power consumption and temperature distribution in a frozen sodium seal as used on a pump shaft are reported.

Title

Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending June 10, 1952

Document No.

ORNL-1294

Author

W. B. Cottrell

Abstract

Research and development progress on the aircraft nuclear propulsion project is reported. NaK-to-NaK and sodium-to-air heat exchangers were operated for long periods. Efforts devoted to the development of instruments for measuring flow, liquid level, and pressure are described. Successful operation of a frozen sodium pump shaft seal is reported. A sodium circulation loop, fabricated from Inconel, had to be shut down due to a plugged line. The plugging was later found to be due to the mass transfer of nickel from the Inconel.

Title

Aircraft Nuclear Propulsion Project Quarterly Progress Report for Period Ending March 10, 1953

Document No.

ORNL-1515

Author

W. B. Cottrell

Abstract

Progress on the ANP project is reported. Pertinent information categories include: ac induction pump testing in sodium, tests of various types of seals in lead and sodium, heat exchanger testing, distillation cleaning of NaK from discharged lines, vortexing studies in loop surge tanks, detection of NaK vapor in helium by flame testing, materials compatibility studies involving cermets, beryllium, Inconel, Incoloy, quartz, and nickel in lead, sodium, and NaK, and thermal conductivity studies for sodium hydroxide and lithium hydroxide.

Title

Test of PK Pump Split Purge Gas Labyrinth

Document No.

CF-59-1-5

Author

D. L. Gray

Abstract

A split gas purge arrangement was applied to the shaft of a centrifugal pump of the vertical shaft overhung impeller sump type. The pump was operated with NaK at 1200 °F and 3000 rpm for 3378 hr. The test was terminated when purge supply ports leading directly to the shaft annulus plugged. The split purge did keep gross NaK vapors from the lower seal.

Title

The Enrico Fermi Fast Neutron Breeder Reactor. Components - Experience to Date (Proc. of Fast Neutron Breeder Reactor Conf. Dec 63, Detroit, Mich.)

Document No.

CONF-631201, pg. 16/22

Author

A. Amorosi

Abstract

A review is given of the operation experience with the reactor, and specific problems encountered and the way in which they were handled are outlined. Mechanical design, fuel handling, seals, pump and check valve, steam generator, remote maintenance, and corrosion are discussed.

Title

SNAP-8 Electrical Generating System Development Program. Quarterly Report No. 0390-04-17

Document No.

NASA-CR-54219

Abstract

The objective of the SNAP-8 program is to design and develop a 35-kW electric generating system for use in various space emissions. The progress during the seventeenth quarter of contract performance is reported. The status of the various necessary systems analyses is presented. Design and fabrication of components-turbine alternator assembly, NaK and mercury pump-motor assemblies, bearings, seals-to-space, mercury boiler and condenser, mercury injection system, electric insulation, and electrical components - are reported. Testing the several component test loops is discussed. Materials evaluation of components exposed to mercury corrosion and erosion is presented.

Title

Quarterly Technical Progress Report (On) SNAP-50/SPUR, for Period Ending March 31, 1965.

Document No.

APS-5152-R2

Abstract

Mechanical testing results on TZM, K-94, and K-601 alloys are reported along with those for arc-cast W and Mo-Ti-C systems. Development and testing bearings and seals for use with liquid-K are reported. Results of turbomachinery tests and analyses are included along with development and testing of potassium boilers. Work on condenser design and development, control system analysis, and auxiliary power supply analysis is summarized.

Title

SNAP-8 Seals-to-Space Development Test Program. Volume 1. Visco Pump. Report No. 2808.

Document No.

NASA-CR-54234(Vol. 1)

Author

R. L. Lessley

Abstract

In order to permit use of oil-lubricated anti-friction bearings in the SNAP-8 power conversion system rotating assemblies, a seal is required to prevent mixing of mercury and oil. A seal concept was evolved in which mixing of mercury and oil is prevented by venting a section of the shaft to space and permitting a small controlled leakage of each liquid to the space vent cavity. The particular seal-to-space concept advocated for use in SNAP-8 involves use of the visco pump (a device consisting of a shaft with helical grooves rotating within a close-fitting housing). This seal element, when functioning properly, prevents passage of raw liquid through the seal. It establishes a liquid-vapor interface past which only vapors can leak. Visco pumps function improperly under certain conditions. A condition known as breakdown can prevail, in which case liquid droplets can be lost from the interface and the effectiveness of the seal as a barrier to liquid leakage is destroyed. Since the visco pump technology is not sufficiently advanced to permit prediction of conditions under which breakdown will occur, a test program was required to demonstrate the adequacy of the visco pump operating with SNAP-8 fluids (mercury and ET-378 oil) and operating conditions. Tests were conducted with quartz

Abstract (contd)

visco pump test sections which permitted viewing and photographing of the liquid-vapor interface during operation of the test rig. Results showed the visco pump interface to be stable for SNAP-8 operating conditions.

Title

SNAP-8 Seals-to-Space Development Test Program. Volume II. Molecular Pump. Report No. 2808.

Document No.

NASA-CR-54234 (Vol. II)

Author

J. N. Hodgson

Abstract

The seals-to-space of SNAP-8 require a means of restricting the flow of mercury and lubricant vapor out of the vent to space. This flow restriction is accomplished by use of a molecular pump - a highly effective form of flow restrictor based on the principle of the Holweck Vacuum Pump. Rather than using the static action of the conventional annulus or labyrinth, the molecular pump employs a dynamic action which moves molecules counter to the pressure gradient to space. The theoretical performance of the molecular pump was derived in detail as part of the flow restrictor development program. Tests were conducted on both optimum and non-optimum sets of configurations at temperatures of 300 and 375 °F. The correlation between the theory and test data was good. Three basic problems were encountered during the test program - shaft seizure occurred several times during the initial phase of testing, equipment malfunction caused interface temperatures to be well in excess of 400 °F, and entrained air in the

Abstract (contd)

bearing lubricant leaked into the molecular pump area causing pressure ratios to be low enough to lead to the premature conclusion that the molecular pump was not functioning. Once these problems were brought under control, a successful test program was carried out. Both pressure ratio and leakage data are in good agreement with theoretical predictions. The test program served to answer a number of questions regarding the effectiveness of the molecular pump as a flow restrictor.

Title

SNAP-8 Seals-to-Space Development Test Program. Volume III. Dynamic Slinger. Report No. 2808.

Document No.

NASA-CR-54234 (Vol. III)

Authors

R. L. Lessley, I. L. Marburger

Abstract

The seal-to-space concept advocated for use in the SNAP-8 power conversion system rotating assemblies utilizes two basic elements. The first element creates a liquid-vapor interface (i.e., a line of demarcation between liquid and vapor). The second element restricts leakage of the vapors which emanate from the liquid-vapor interface. A dynamic slinger has been selected as the liquid-vapor interface element for the current SNAP-8 seal-to-space configuration. In order for the slinger to fulfill its intended role, however, the liquid-vapor interface must be stable. Otherwise droplets may be generated by the interface which can possibly pass on through the seal. A series of tests was conducted in which oil and mercury slingers were operated at the specific conditions encountered in the SNAP-8 rotating assemblies. The purpose of these tests was to demonstrate that the slinger interfaces actually do provide a positive demarcation between liquid and vapor and thus create a barrier to prevent leakage of raw liquid. The slinger housings contained transparent sections which permitted observation of the interface during operation. Pumping and drag data were also collected. The oil slinger tests revealed the presence on the slinger static wall of a film that could serve as a vehicle for the transport of oil past the interface. It was found that this problem could be avoided by coating the slinger walls with a non-wettable material such as teflon.

Title

SNAP-8 Seals-to-Space Development Test Program. Volume IV. Integrated Seal Simulator. Report No. 2808.

Document No.

NASA-CR-54234 (Vol. IV)

Authors

R. L. Lessley, J. N. Hodgson, E. A. Haglund

Abstract

A seal-to-space was developed to permit use of oil-lubricated bearings in the SNAP-8 mercury-vapor turbine assembly. Small leakages of oil and mercury to a space vent cavity are permitted (1 to 10 lb/yr) in order to avoid intermixing of the two fluids. Since improper functioning of the seal would result in a complete loss of mercury and lubricating oil inventory, the development of the seal-to-space has been conducted in a very deliberate manner. The program included individual tests of each seal element and culminated with the test of a complete seal operating in the thermal environment of the SNAP-8 turbine seal. The results of this test are described. The basic seal concept utilizes a dynamic slinger to establish a liquid-vapor interface as a barrier to the flow of liquid. A molecular pump is used to restrict leakage of vapors emanating from the liquid-vapor interface. Since the leakage rate is very sensitive to the temperature of the liquid-vapor interface, provisions for cooling of the interface are included in the seal design. Two methods of cooling the interface were evaluated in separate test rigs - the Model A and Model B Seal Simulators. These test rigs duplicated very closely the configuration of a SNAP-8 turbine seal. They included a dummy turbine wheel and turbine housing into which 700 °F mercury vapor was introduced. This provided faithful simulation of the actual thermal environment of a SNAP-8 turbine seal-to-space.

Title

Materials for Components in Fluoride and NaK Flow Systems

Document No.

BMI-1178

Authors

S. J. Basham, R. W. Endebrock,
J. H. Stang, E. M. Simons

Abstract

Screening studies were made of a wide variety of materials that might be useful for certain special components in high-temperature (1200 °F to 1500 °F) No. 30 fluoride salt or NaK reactor flow systems. The program covered separate research investigations of static and dynamic materials compatibility and sleeve bearing-journal combinations. By statically loading various specimens immersed in fluoride salt at 1200 °F and in NaK at 1500 °F, the self-welding tendencies of a large number of combinations were evaluated to see if they might be suitable as valve seats and plugs. To find materials compatible for nonhydrodynamic sliding applications such as face-type seals, one material was rubbed against another, under load, while immersed in salt or NaK at 1500 °F.

APPENDIX D

RADIATION EFFECTS ON SELECTED ELASTOMERS

INTERIM REPORT

and

VISCOELASTIC EVALUATION OF SILICONE

RUBBER AND ETHYLENE PROPYLENE

(Formerly published as BNWL-736, January 1968)

N. R. Gordon

RADIATION EFFECTS ON SELECTED ELASTOMERS
INTERIM REPORT

By: N. R. Gordon

INTRODUCTION

Dow Corning 2096 silicone rubber has been used successfully in reactor applications for the past few years. During this time, several elastomers have been introduced which may be superior to the silicone rubber in a radiation environment. The purpose of this research program is to provide a comparison of several elastomers for use in reactor applications.

SUMMARY

The effect of radiation on tensile strength, elongation, and hardness was determined for eight elastomers. The superiority of any of these materials is difficult to determine using only ultimate properties. The fabrication variables appear to induce greater variation than does the radiation. These studies will continue, using viscoelastic analysis to determine time-temperature effects on stress and strain.

EXPERIMENTAL

The initial efforts of this program have been to study the effects of radiation on ultimate tensile strength, elongation, and hardness of elastomers. Eight materials were chosen for this study including five silicones, one nitrile and two ethylene-propylene copolymers. They were molded into flat sheets and two sizes of standard O-rings.

Tensile and elongation measurements were made with a Scott tensile tester at an elongation rate of 20 in./min. All tests were made at 21 ± 3 °C (70 ± 5 °F). Stress-strain relationships were determined on some of the samples with the use of a spark recorder. This instrument is operated manually to indicate tensile strength on a calibrated graph paper at predetermined increments of elongation. Photographs of the equipment testing flat tensile specimens and O-rings are shown in Figures 1 and 2, respectively. Hardness of the specimens was determined with a Shore A-2 hardness tester.

Specimens were irradiated in air to total doses of 5×10^6 R and 5×10^7 R. Three sets of specimens (ten flat tensile specimens and four specimens of each size O-ring in each set) were tested for each batch of material. One set was used as a control and one set was tested at each of the two radiation intensities. Table 1 shows a summary of all of the specimens tested.

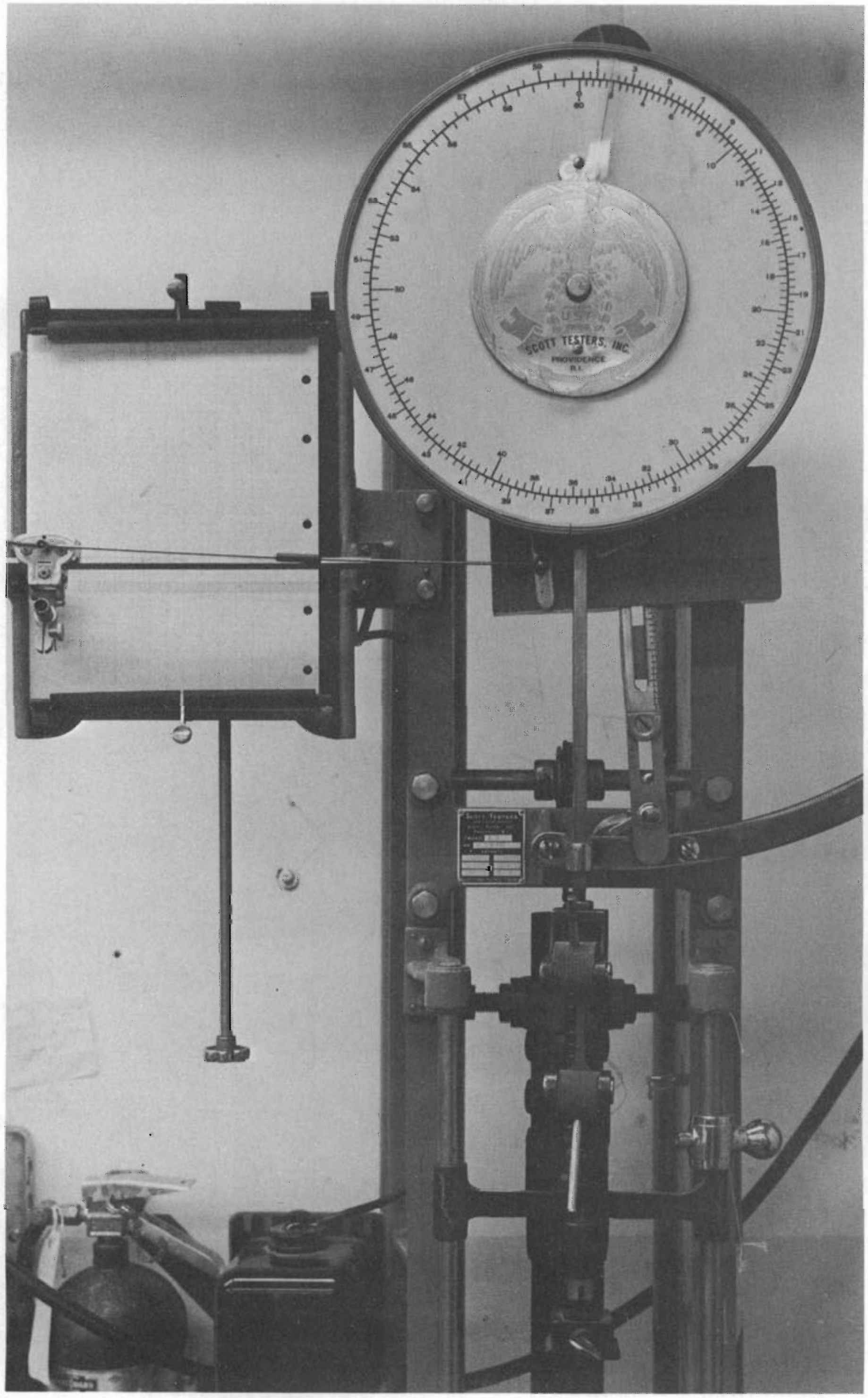
The majority of the silicone samples were molded by The Plastics and Rubber Company. Three of the samples were molded from two batches of material. Parker Seal Company molded one batch of Dow Corning 2096 silicone and their proprietary formulations of nitrile and ethylene-propylene.

RESULTS AND CONCLUSIONS

Table 2 provides a summary of the results obtained on the eight elastomers. It contains comparative values for:

- Ultimate tensile strength and elongations for unirradiated specimens and those irradiated to 5×10^7 R.
- Variation in tensile strength, elongation and hardness induced by irradiating to 5×10^7 R (restricted to changes within individual batches).
- Tensile strength variation of unirradiated specimens induced by molding.

From these results, we can conclude that Material VIII (ethylene-propylene) is least affected by irradiating to 5×10^7 R. This material also showed the smallest sample-to-sample variation. Material I (silicone) had the greatest



Neg 0660753-1

FIGURE 1. Testing of Flat Tensile Specimen

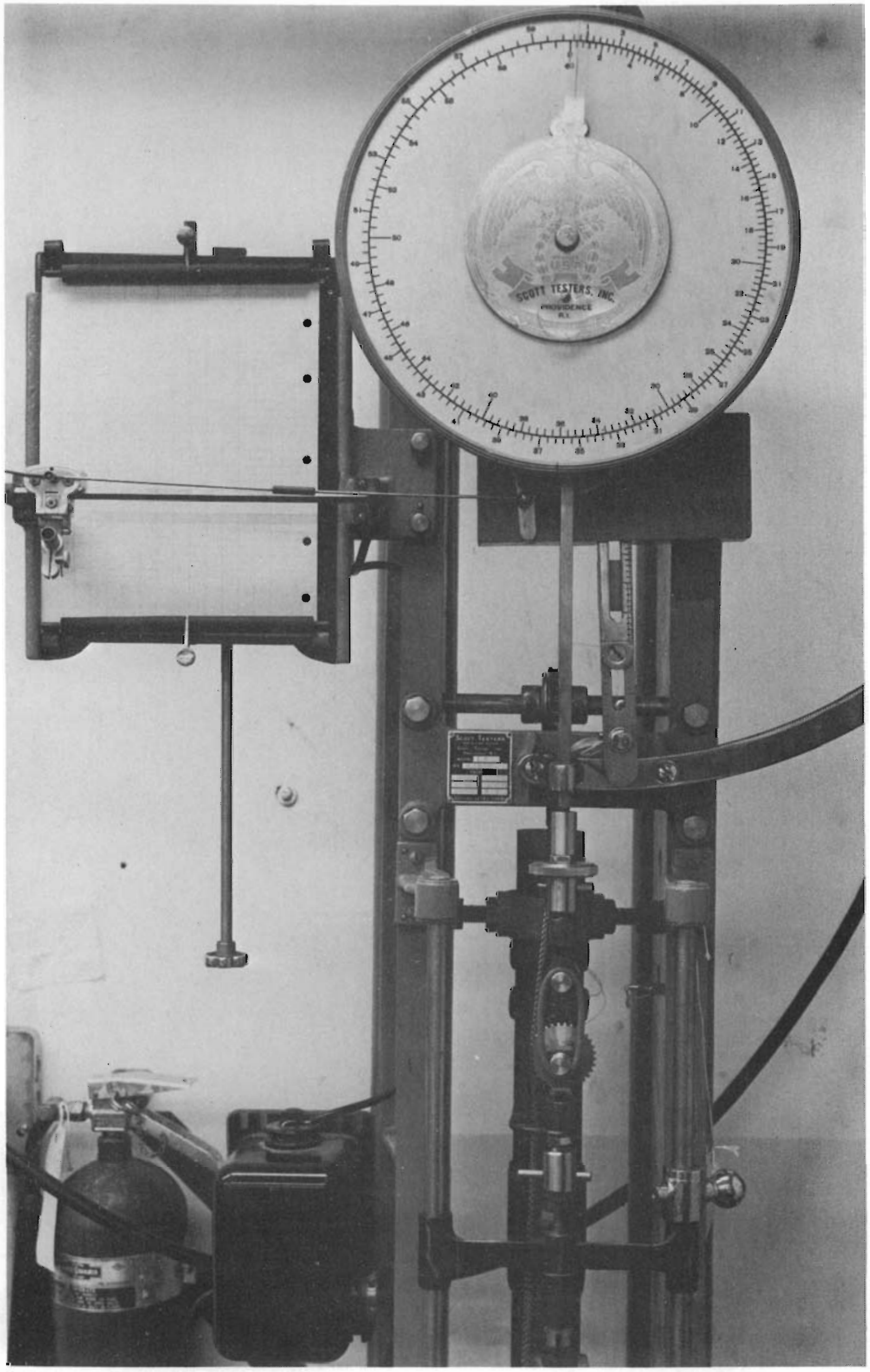


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FIGURE 2. Testing of O-Rings

TABLE 1. A Summary of All Specimens Tested

Material Code and Type	Average Properties*	Flat Tensile			Size 330 O-Ring			Size 325 O-Ring		
		Control	5×10^6r	5×10^7r	Control	5×10^6r	5×10^7r	Control	5×10^6r	5×10^7r
I (silicone)	T	1141	1043	935	459	493	363	420	467	393
	E	205%	190%	46%	---	242%	55%	---	---	---
	H	69	72	87	50	60	83	50	62	80
II (silicone)	T				577	503	395	529	465	389
	E				>300%	>300%	75%	>420%	295%	90%
	H				54	52	78	38	49	72
III (silicone)	T	674	508	348	382	329	235	372	314	148
	E	457%	310%	64%	>300%	>300%	82%	>500%	460%	24%
	H	35	38	68	38	48	74	38	49	72
IV (silicone)	T	970	948	754	447	524	443	524	514	622
	E	312%	191%	66%	235%	195%	45%	310%	280%	65%
	H	60	65	82	62	72	82	60	65	84
V (silicone)	T	1007	974	902	635	622	510	634	644	439
	E	283%	188%	60%	255%	190%	45%	225%	200%	48%
	H	65	65	82	68	74	84	68	73	85
VI (silicone)	T	656	676	382	418	394	281	369	366	336
	E	679%	389%	87%	>300%	208%	50%	350%	355%	100%
	H	28	36	60	54	66	82	55	59	80
VII (nitrile)	T				1371	1353	1278	1323	1397	1417
	E				---	---	515%	---	---	600%
	H				59	62	63	58	60	64
VIII (ethylene-Propylene)	T	1532	1404	1528	1109	1039	787	1065	1112	798
	E	174%	146%	108%	300%	265%	135%	300%	335%	170%
	H	80	82	83	80	82	85	81	82	87

TABLE 1. (Contd)

Material Code and Type	Average Properties*	Flat Tensile: Control	Flat Tensile: 5×10^6 r	Flat Tensile: 5×10^7 r	Size 330 O-Ring: Control	Size 330 O-Ring: 5×10^6 r	Size 330 O-Ring: 5×10^7 r	Size 325 O-Ring: Control	Size 325 O-Ring: 5×10^6 r	Size 325 O-Ring: 5×10^7 r
IX (ethylene-propylene)	T	2246	2112	1469	1079	1078	766	1109	1024	717
	E	341%	323%	192%	425%	405%	235%	530%	500%	310%
	H	68	68	69	68	69	69	70	69	69
X (silicone)	T				585	393		576	464	
	E				310%	165%		375%	230%	
	H				57	64		57	66	
XI (silicone)	T	759	729	619	545	590	450	571	582	484
	E	298	251	75	385	330	93	415	177	118
	H	45	51	67	38	47	67	42	48	68
XII (silicone)	T	921	864	1001	822	844	521	739	799	640
	E	207%	167%	54%	385%	300%	78	358%	328%	101
	H	56	60	79	52	57	77	53	58	77
XIII (silicone)	T	510	498	369	362	415	227	347	378	250
	E	350%	326%	120	470%	420%	95	500%	433%	133
	H	38	46	55	38	39	56	38	39	57

* *T = Tensile Strength (psi)*
E = % Elongation
H = Hardness (Shore A)

TABLE 2. A Summary of Results from Eight Elastomers

Material Code	*	Ultimate Tensile (unirradiated)	Ultimate Elongation (unirradiated)	Ultimate Tensile (5×10^7)	Ultimate Elongation (5×10^7)	Tensile Variation (radiation)	Elongation Variation (radiation)	Hardness Variation (radiation)	Fabrication Tensile Variation (unirradiated)
I, X, XII	T	1141-921	207-205	1001-935	54-46	+9 to -18%	-74 to-78%	+26 to +41%	1141-420
	O	822-430	385-358	640-363	101-55	-7 to-37%	-72 to -80%	+45 to +66%	63%
V	T	1007	283	902	60	-10%	-79%	+11%	1007-635
	O	635	225-255	439-510	45-48	-20% to -31%	-79 to -82%	+25%	37.2%
II, III, & XIII	T	510	350	369	120	-28%	-66%	+45%	577-347
	O	347-577	420-500	227-395	75-133	-26 to -32%	-74 to -80%	+45 to +50%	40%
VI, XI	T	656-759	298-679	382-619	75-87	-18 to -42%	-75 to-87%	+49 to +115%	359-369
	O	369-545	350-415	281-484	50-118	- 9 to -33%	-72 to -76%	+46 to + 76¢	45%
IV	T	970	312	754	66	-22%	-79%	+37%	970-447
	O	447-524	235-310	443-602	45-79	- 1 to +15%	-79 to -81%	+32 to 40%	54%
VII	T	1323-1371		1278-1417	515-600	- 7 to +7%		+ 7 to +10%	
	O								
IX	T	2246	341	1469	192	-35%	-44%	<1%	2246-1079
	O	1079-1109	425-530	717-766	235-310	-29 to -35%	-42 to -45%	<1%	52%
VIII	T	532	174	1528	108	<- 1%	-38%	+4%	1532-1065
	O	1065-1109	300	787-798	135-170	-25 to -29%	-43 to -55%	+6 to 7%	30%

* *T represents flat tensile specimens.*
O represents O-rings

sample-to-sample variation. These changes indicate that the molding processes can have a considerable effect on the ultimate strength of a molded part. This fact is further verified in that some of the specimens irradiated to 5×10^6 R had higher tensile strength than the control samples in the same batch.

Figures 3 through 10 are stress-strain curves for some of the samples studied. These curves are probably of greater significance than the ultimate values of tensile strength and elongation since they present a relationship between the two. These curves show that the modulus of elasticity of ethylene-propylene (Figure 6) is affected the least by radiation.

We may generally conclude that the results obtained to date are insufficient for making engineering recommendations. However, future studies will be made using viscoelastic analysis to determine the effects of time and temperature on stress and strain of elastomers. In this way, properties directly related to engineering applications can be measured. In addition, the effects of radiation in environments other than air will be determined. In most applications, the radiation environment will be comprised of elevated temperatures, water, and steam.

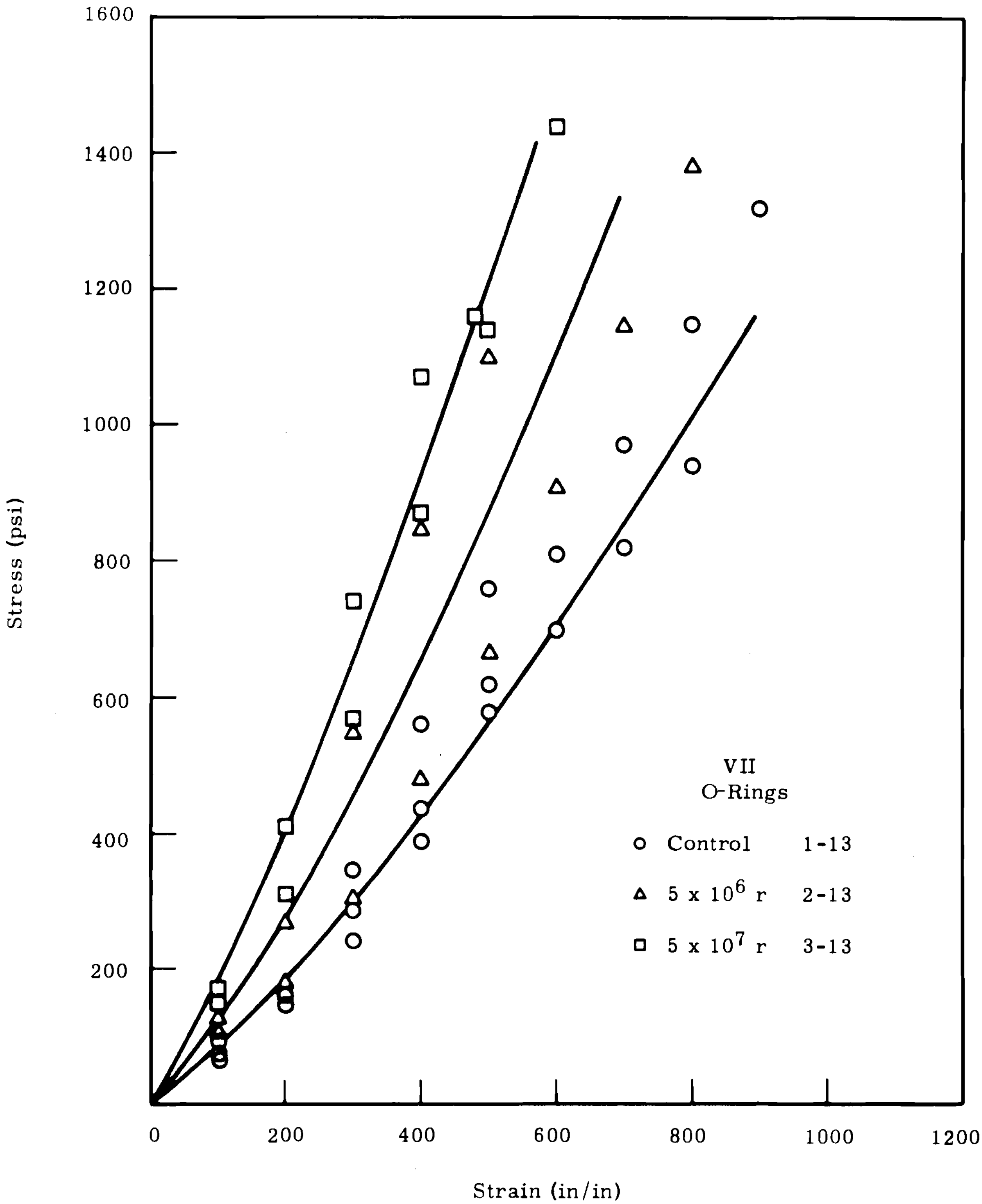

FIGURE 3. Stress-Strain Curves for Nitrile O-Rings No. VII

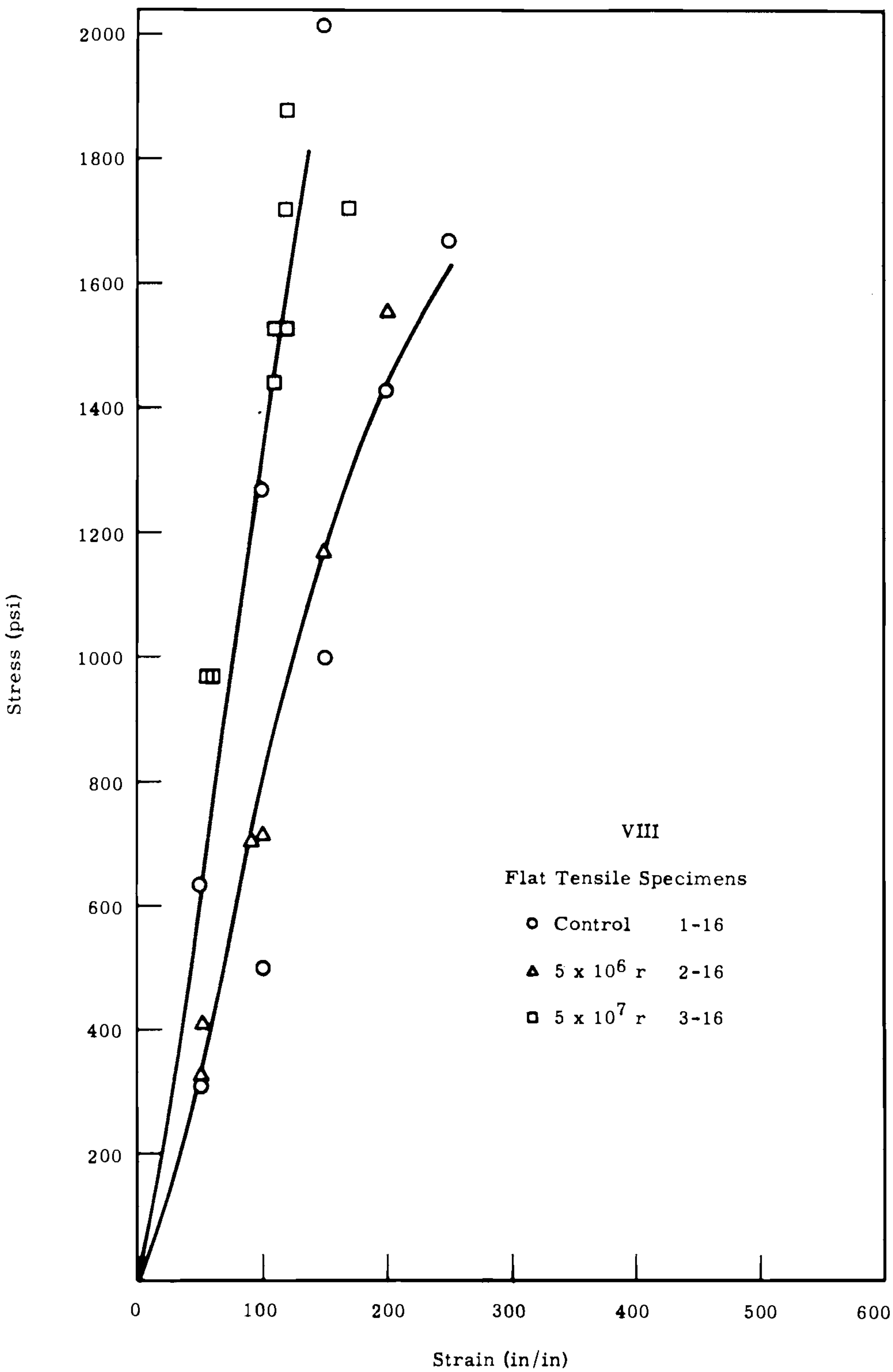

FIGURE 4. Stress-Strain Curves for Ethylene-Propylene Flat Tensile Specimens No. VIII

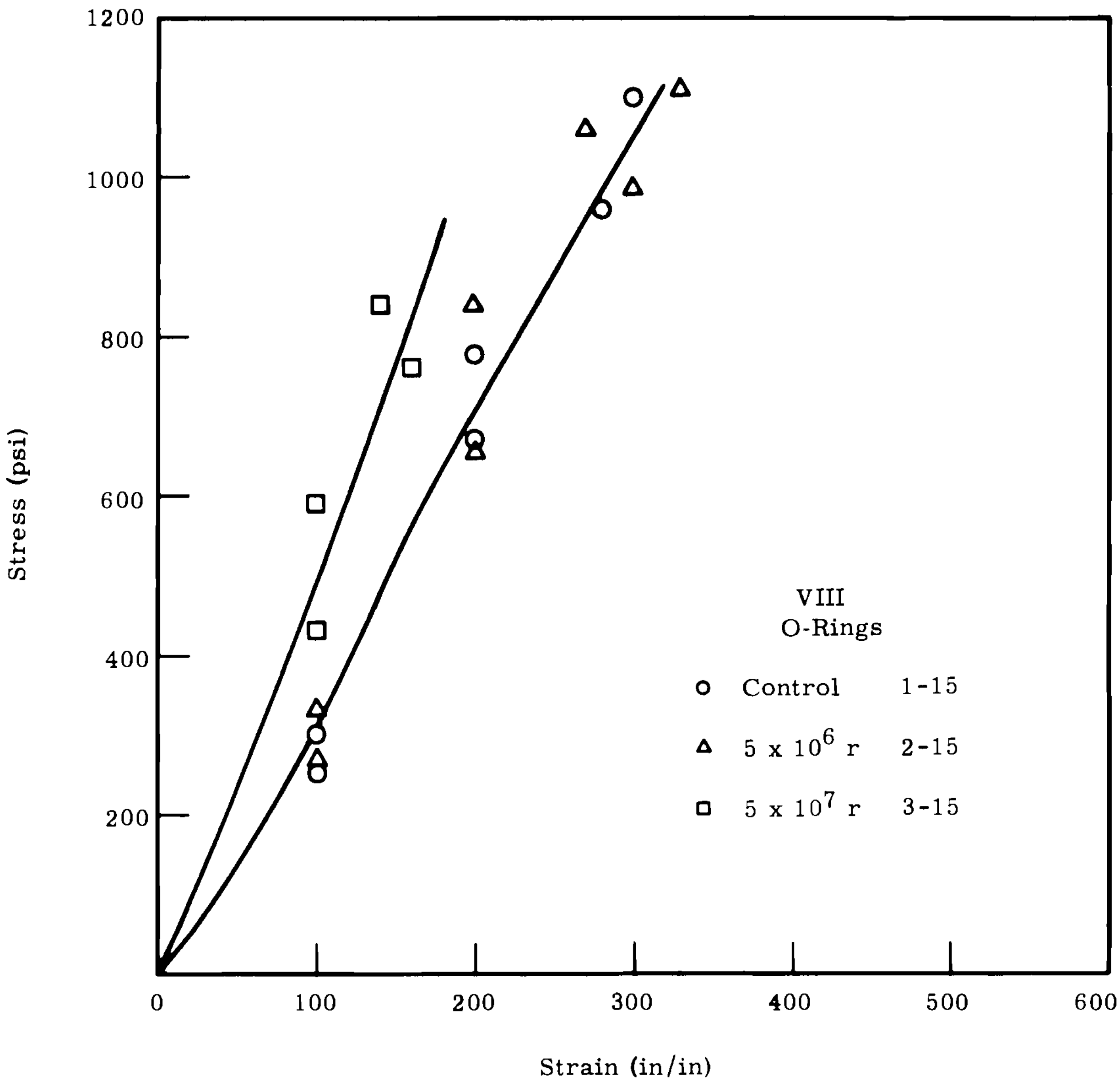

FIGURE 5. Stress-Strain Curves for Ethylene-Propylene O-Rings No. VIII

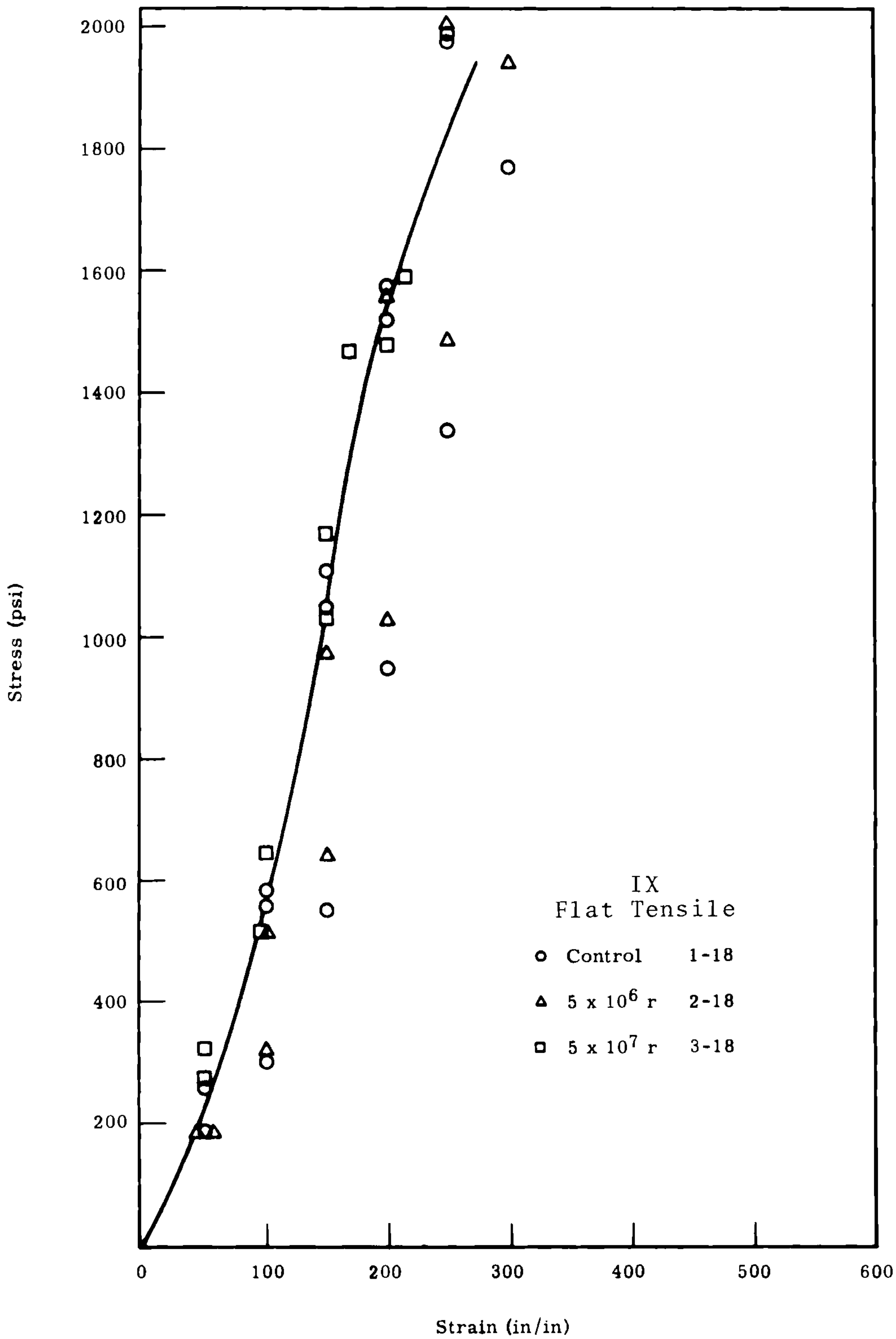

FIGURE 6. Stress-Strain Curves for Ethylene-Propylene Flat Tensile Specimens No. IX

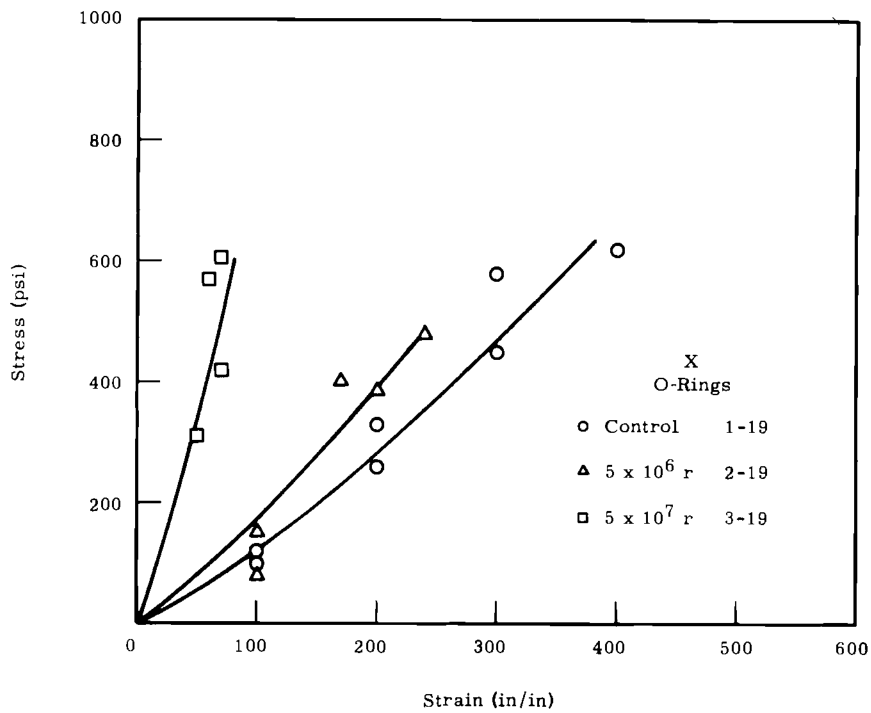

FIGURE 7. Stress-Strain Curves for Silicone O-Rings No. X.

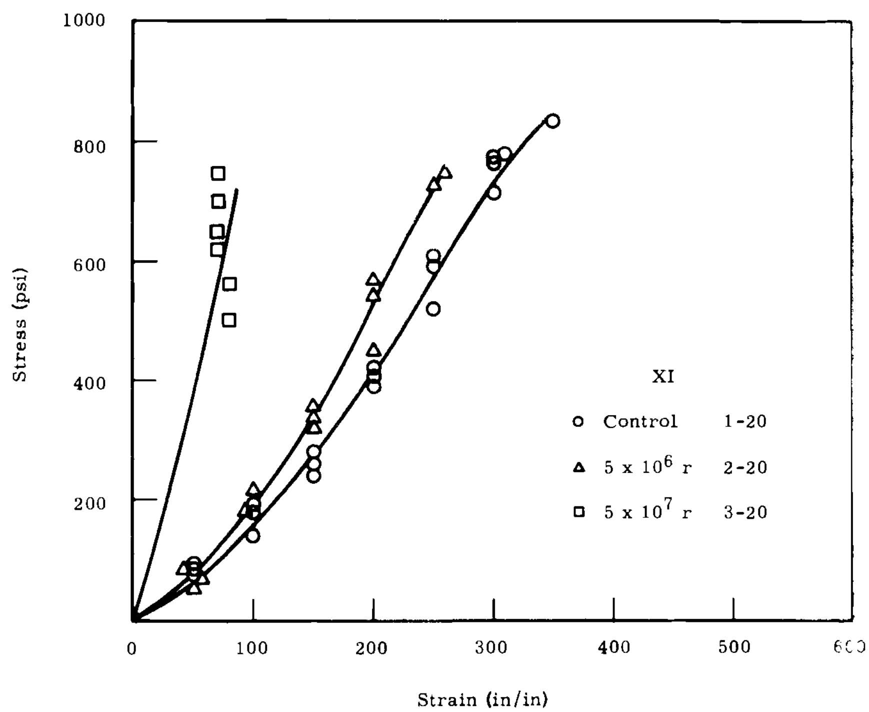

FIGURE 8. Stress-Strain Curves for Silicone O-Rings No. XI

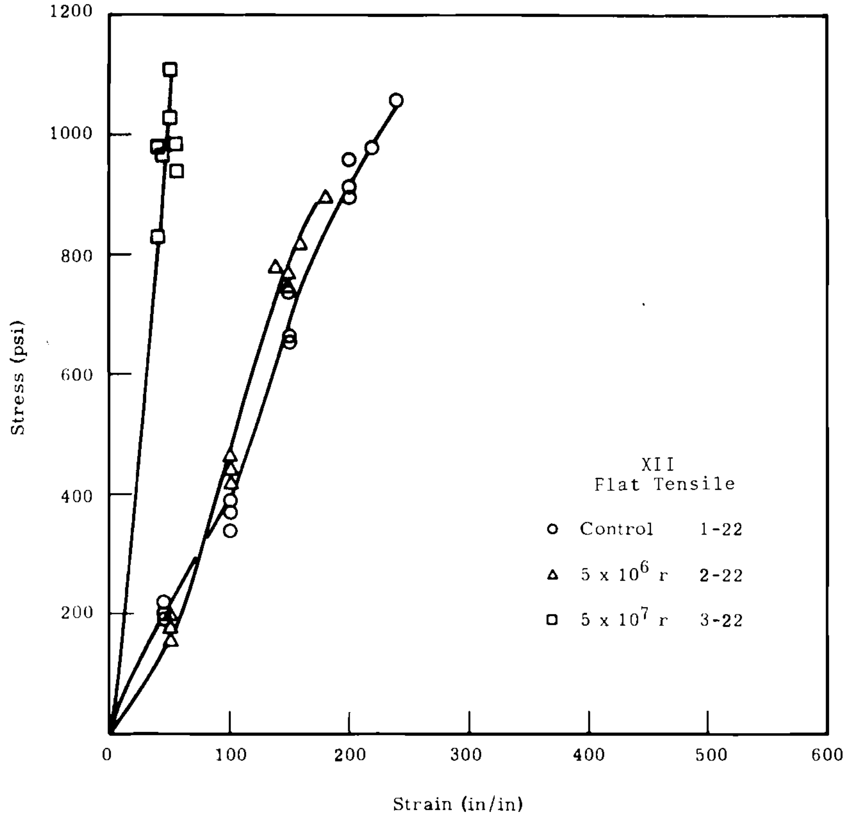

FIGURE 9. Stress-Strain Curves for Silicone Flat Tensile Specimens No. XII

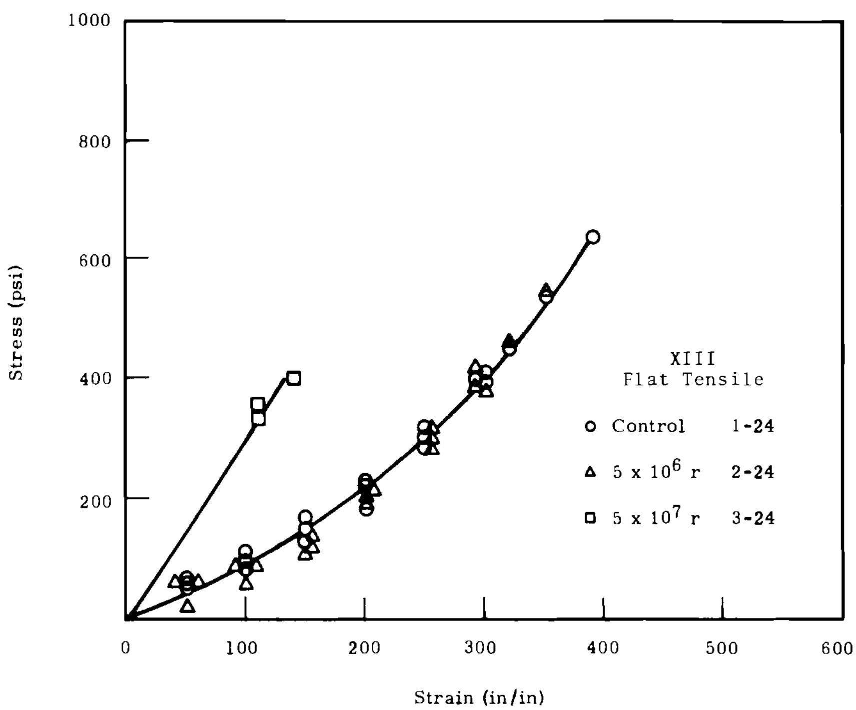

FIGURE 10. Stress-Strain Curves for Silicone Flat Tensile Specimens No. XIII

VISCOELASTIC EVALUATION OF SILICONE RUBBER AND ETHYLENE PROPYLENE

By: N. R. Gordon

ABSTRACT

Initial evaluation studies were made on two materials to determine their suitability as alternates to a silicone rubber for reactor applications. Of the two materials studied, ethylene propylene and chlorosulfonated polyethylene, the ethylene propylene has the lowest compression set and stress relaxation; and consequently, appears to be the best candidate. It has a higher stress relaxation rate than the silicone but its radiation resistance is superior.

VISCOELASTIC EVALUATION OF SILICONE RUBBER AND ETHYLENE PROPYLENE

By: N. R. Gordon

INTRODUCTION

The research described in this report is a preliminary effort to determine which elastomeric materials can be used as O-rings in Reactor applications. At the present time only one material, a silicone rubber, is specified for use in the Hanford reactors. The specification of additional materials would create a competitive supply with the possibility of cost savings. The purpose of this research was to evaluate the usefulness of three materials (ethylene propylene, chlorosulfonated polyethylene, and chlorosulfonated polyethylene blends) for reactor applications.

SUMMARY AND CONCLUSIONS

This report describes a preliminary investigation of three materials proposed as alternates for silicone rubber in O-ring applications. The materials included in this study were silicone (for comparison), ethylene propylene, chlorosulfonated polyethylene, and blends of chlorosulfonated polyethylene with 50% neoprene or 50% nitrile rubber. The properties initially measured on these materials were stress-strain relationships, tear strength, and compression set. A knowledge of these properties would provide a means of selecting which materials warranted further investigation. On this basis further studies on the chlorosulfonated polyethylene materials were discontinued because of their comparatively high compression set values.

The second phase of the program was the determination of time-dependent properties by stress relaxation measurements.

These measurements show the changes in a material under constant strain as a function of time. The results show that the silicone rubber has a much lower rate of relaxation than does ethylene propylene, although the rate for the latter appears within acceptable limits.

As a final phase of the work, some stress-relaxation experiments were conducted on irradiated silicone and ethylene propylene. These tests show that even though the ethylene propylene relaxes faster than the silicone, it is much more radiation resistant. Consequently, the modulus of ethylene propylene changes less than silicone over an extended period of time in a radiation environment.

Based on the limited amount of work on radiation effects the ethylene propylene appears superior to the silicone. However, it is recommended that the studies of the effects of radiation on stress relaxation be continued to provide a more complete characterization of both materials.

DISCUSSION

BACKGROUND

One of the most demanding uses for elastomers in the Hanford reactors has been that of a cooling tube O-ring seal. This seal is subjected to both hot water and radiation. To date, a silicone elastomer has demonstrated the longest useful life for this particular application. A nitrile elastomer with superior radiation resistance was tried at one time, but its creep and stress relaxation properties were so poor that it had an unsatisfactory life span.

At the present time there is no backup material qualified as a replacement for the silicone rubber. A two-phase testing program is being instituted to find other useful materials. The first phase of this program is a preliminary study on materials that appear suitable for reactor use. Phase two of the program tests these materials under operating conditions, using a testing jig installed on the rear face of a reactor. This report describes results from phase one of the program.

Past Evaluation Techniques

In the past, testing techniques for evaluating elastomeric materials for reactor applications have been quite restrictive. The major test parameters have been ultimate tensile strength and ultimate elongation at an extension rate of 20 in./min.[1] In addition to these, the hardness of the materials has been determined. The effect of radiation on the materials has been determined by exposing test specimens to a ^{60}Co gamma flux. Although the data obtained by these techniques have been accurate and reproducible, they are not adequate for making reliable recommendations for reactor use. The time dependency of the material's properties has been ignored, leaving a lack of knowledge of creep or stress relaxation. Such a premise can lead to untimely and unexpected failures.

Proposed Evaluation Techniques

Rough screening of materials was obtained by measuring compression set, tear strength, and stress-strain curves. Compression set normally provides an indication of the time dependence of a material under stress. Tear strength provides a measure of a material's strength during handling. Stress-strain curves provide information on the interrelations of strength and elongation as well as the ultimate properties.

A more accurate evaluation of the materials can best be accomplished by measuring one of the time-dependent properties, i.e., creep or stress relaxation. Since there is a mathematical relationship between the two, stress relaxation is usually determined because the experiment is easier to perform.

Stress Relaxation and Creep

Stress relaxation of a material is defined as: "the time dependent decrease in force applied by a material which is stretched to a predetermined length and held at that point," or "the decrease in stress at constant strain." Creep is defined as: "the time-dependent change in length of a material under a constant load," or "the change in strain at constant stress."

Stress relaxation in tension is defined mathematically by the equation:

$$E(t) = \frac{F(t)}{A_o \ \Delta \ell / \ell_o} = \frac{\sigma(t)}{\varepsilon} \tag{1}$$

where E(t) is the modulus of elasticity as a function of time, F(t) is the applied force as a function of time, A_o is the unstressed cross sectional area, ℓ_o is the unstressed length, $\Delta\ell$ is the change in length, $\sigma(t)$ is the stress as a function of time, and ε is the applied strain. A typical stress relaxation curve is shown in Figure 1.

Creep is generally defined mathematically as:

$$D(t) = \frac{\Delta \ell (t) / \ell_o}{F/A_o} = \frac{\varepsilon(t)}{\sigma} \tag{2}$$

where D(t) is the tensile compliance as a function of time, $\varepsilon(t)$ is the strain as a function of time and σ is the applied stress. A typical creep curve is shown in Figure 2. It would appear that Equation (2) is merely the reciprocal of Equation (1). However, this is not exactly true. In Equation (1) force is measured as a function of time and in Equation (2), the change in length is measured as a function of time. If the material is isotropic and if the total strain is kept low the difference in modulus between the two is negligible and either can be used for design purposes. Stress relaxation was measured in this study since an O-ring normally fails by that mechanism.

Boltzmann Superposition Principle

In a creep experiment, if stress σ_o is applied to a material at time zero, the strain after time t_o is given as:

$$\varepsilon(t_o) = \frac{1}{E(t_o)} \sigma_o \tag{3}$$

The addition of another stress at time t_o will then give a strain at time t of

$$\varepsilon(t) = \frac{\sigma_o}{E(t)} + \frac{\sigma(t)}{E(t - t_o)} \tag{4}$$

indicating the additive nature of the property. This is the basis of the Boltzmann superposition principle and serves as a definition of a linear viscoelastic material. A linear material whose structure (crystallinity, degree of cross linking, etc.) does not change during the course of the experiment will obey this principle.[2] Because of this restriction, an elastomer in a reactor environment does not function as a linear viscoelastic material since its structure is continually affected by the radiation environment. However, if testing is performed outside of a radiation field, the material will obey the principle. By irradiating the material to various radiation levels then measuring the resulting changes, an indication of the influences of the radiation environment can be obtained.

Time-Temperature Superposition

The viscoelastic functions of materials are not only time dependent but also have a temperature dependency.

Theoretical studies by Rouse[3] were conducted on polymer solutions to describe this dependency in terms of relaxation time, τ_p. From these studies he developed the relationship:

$$\tau_p = f(a, \zeta_o, 1/T) \tag{5}$$

where a is the monomer length, ζ_o is the frictional coefficient and T is the absolute temperature. The temperature dependence of a in this function is expected to be very small. Using the Rouse function and the relationship between viscosity, η, and relaxation time in a solvent-free system, Ferry[4] has developed a constant for the ratio of temperature dependent properties. He refers to this constant as the "shift factor" defined as:

$$a_T = \frac{\eta \; T_o \; \rho_o}{\eta_o \; T \; \rho} \tag{6}$$

The subscript o denotes properties at an arbitrary reference temperature T_o. By applying the "shift factor" to experimental results, Ferry then formulated the equation:

$$\text{Log } A_T = -C_1^o(T - T_o)/(C_2^o + T - T_o) \tag{7}$$

where C_1^o and C_2^o are empirical constants. Williams, Landel and Ferry[4] further simplified this equation by using the glass transition temperature of the material, T_g, as the reference temperature and obtained the relationship

$$\text{Log } A_{T_g} = -C_1^g(T - T_g)/(C_2^g + T - T_g) \tag{8}$$

The form of this equation has been found to fit many different systems over the temperature range from T_g to t_g + 100.

By using these relationships, it is possible to measure a time-dependent property of a material for short periods of time at several elevated temperatures and combine the data into a master curve extending to longer times at a lower reference temperature. The applicability of the equation is limited to amorphous polymers up to temperatures about 100 °C above their glass transition temperatures. This relationship can also be applied to semicrystalline polymers if the mechanical properties are controlled by the amorphous regions of the polymer. Also, if the experiments are performed at the same strain, the contributions of the crystalline regions are held constant and superposition can be made on the basis of the amorphous components.

Using time-temperature superposition, master curves for polyethylene at yield strain have been developed which coincide quite accurately with actual burst tests for polyethylene pipe.[5] This provides fairly conclusive proof that stress relaxation experiments utilizing time-temperature superposition can provide data adequate for design purposes.

EXPERIMENTAL

Materials

The elastomers studies in this program were silicone rubber, chlorosulfonated polyethylene, chlorosulfonated polyethylene blended with 50% nitrile or 50% neoprene elastomer, and ethylene propylene.* The silicone was in the form of 2-in. diameter O-rings; the chlorosulfonated polyethylene and its blends were 1/8-in. thick sheets; and the ethylene propylene was in the form of 2-1/4-in. diameter O-rings and

* *The silicone used was Silastic 2096 from Dow Corning; the Ethylene Propylene was Nordel from Du Pont; the Chlorosulfonated Polyethylene was Hypalon 20 from Du Pont.*

1/8-in. thick sheet. Initial test data on the silicone and ethylene propylene were so widely scattered that it became necessary to postcure them to obtain valid results.

Initial Screening Tests

The stress-strain relationships, compression set and tear strength of the materials evaluated were determined initially to screen out those materials which would not be comparable to the silicone for reactor use. Stress-strain curves were measured per ASTM D412-62T for the flat sheets and ASTM D1414-56T for the O-rings at an extension rate of 2 in./min. Stress-strain curves are shown in Figure 3. Compression set was measured per ASTM D395-61 method B after 22 hr at 100 °C. Notch tear strength was measured per ASTM D624 die C. Results of compression set and tear strength measurements are shown in Tables 1 and 2, respectively. Compression set measurements were not made on the silicone rubber because no flat specimens were available.

The compression set of the chlorosulfonated polyethylene was so much higher than the other two materials that it was rejected from further consideration. No further tests were made on this material.

TABLE 1. Compression Set[a]

	Measured	Manufacturer's Literature
Chlorosulfonated Polyethylene	68%	75%
Ethylene Propylene	12.2%	12 to 47%[b]
Silicone	--	7%[c]

(a) Measured per ASTM D395-61 Method B - 22 hr at 100 °C.

(b) 47% is for ordinary compound - 12% is for a special compound.

(c) After 70 hr at 300 °C.

TABLE 2. Notch Tear Strength

Chlorosulfonated Polyethylene	101 lb/in.
50% Chlorosulfonated Polyethylene 50% Nitrile	120 lb/in.
50% Chlorosulfonated Polyethylene 50% Neoprene	142 lb/in.
Ethylene Propylene	233 lb/in.
Silicone	30 lb/in.

(a) ASTM D624 die C

(b) Dow Corning Literature

Silicone rubber had the lowest ultimate tensile strength and elongation of all the materials tested; however, this is of little relative importance for O-ring applications where the part is stretched to only a small percent of its ultimate strain.

Stress Relaxation Tests

Stress relaxation tests were conducted on O-ring test specimens. These specimens were extended at a rate of 2 in./min between 1/2-in diameter rollers to a constant strain (ε = 0.5 for silicone, ε = 2.0 for ethylene propylene) and allowed to relax. The stress change was recorded continuously on a strip chart. Tests were conducted for periods of 5 min. Temperature of the sample was maintained at ±0.5 °C in an Instron environmental test chamber. Five specimens were tested at each test temperature and averaged to give one curve at each temperature. The lowest test temperature used (1.5 °C) was taken as the reference temperature. The log of the modulus of elasticity was then plotted as a function of time using T_0/T as a vertical shift factor. Figures 4 and 5 show these curves for silicone and ethylene propylene, respectively, at the various temperatures to which they were tested. The

ρ/ρ_0 part of the vertical shift factor was ignored since the density change is a function of thermal volume expansion and this quantity is very low for the two materials tested. Figure 6 shows the master curves obtained for the silicone and ethylene propylene. All data were reduced to log $E_{(t)}$ at a reference temperature of 1.5 °C. (See Appendix for a description of the method for generating these curves.) From this curve it is apparent that the time-dependent modulus of the silicone rubber changes less than that of the ethylene propylene.

A limited amount of work was performed on the effects of radiation on the stress relaxation of silicone and ethylene propylene as represented in Figures 7 and 8, respectively. These figures show that the change in modulus caused by the radiation environment is much less for the ethylene propylene than it is for the silicone. Using a family of curves such as these, it is possible to plot the modulus of the material as a function of time and radiation dose rate. Such a curve is shown in Figures 7 and 8 at a rate of 10^6 R/hr. Although it is unlikely that these curves depict exactly what ahppens in a reactor, they do provide a valuable method of comparing the two materials. This process would also be useful for designing a part with a maximum and minimum modulus of elasticity. If these limits were designated on the stress relaxation curves, the designer could predict with reasonable accuracy the useful life of the part for any radiation level, or the maximum allowable radiation for a specified life span.

RECOMMENDED ADDITIONAL STUDIES

The foregoing studies have demonstrated that simple tests such as stress-strain relationships and compression set do not provide adequate data for comparative evaluation of materials. This can only be accomplished by using time-dependent properties such as stress relaxation. Additional work should first complete the study of effects of radiation on the stress relaxation of ethylene propylene and the silicone. The curves on silicone are necessary for a basis of comparison, and ethylene propylene appears to be a promising material.

A review of the literature-supplied information on chlorosulfonated polyethylene appears to conflict with the conclusions on this study.[6] Although no values were given for compression set, the stress relaxation quoted is very similar to ethylene propylene as shown in Figure 8. This would indicate that either the compression set measurements are not valid criteria for evaluating time dependent properties of a material, or that the stress relaxation measurements in the referenced paper are for a significantly different chlorosulfonated polyethylene than those studied in this work. However, both materials were made by the same manufacturer and bear the same designation.

Because of the uncertainties involved, further studies should be conducted on the chlorosulfonated polyethylene to determine its stress relaxation and how this property is affected by radiation.

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5. *See p. 189 in Reference 2 above.*

6. *S. F. Kurath, E. Passaglia and R. Pariser, "The Dynamic Mechanical Properties of Hypalon-20 Synthethic Rubber at Small Strains," Journal of Applied Polymer Science, vol. 1, p. 150, 1959.*

APPENDIX

METHOD FOR DEVELOPING A MASTER CURVE

The master curves for the silicone and ethylene propylene shown in Figure 6 were developed from the relaxation curves at varying temperatures (shown in Figures 4 and 5). Tables 3 and 4 contain the accumulated information to form the master curve. The values in the Δ log A_T column were obtained by measuring the horizontal separation between the individual relaxation curves in Figures 4 and 5 with a linear scale. Log A_T then becomes a sum of the Δ log A_T values with the reference temperature (1.5 °C) having a log A_T value of zero. The calculated value of log A_T is obtained by using the equation:

$$\log A_T = -C_1^o(T - T_o)/(C_2^o + T - T_o) \qquad (9)$$

The values of C_1^o and C_2^o are obtained by plotting $T - T_o/\log A_T$ as a function of $T - T_o$ (Figures 9 and 10). From the curve, C_1^o is -1/slope and C_2^o is intercept/slope. The variation from a straight line for ethylene propylene at the higher temperatures results because these temperatures exceed the glass transition temperature by more than 100 °C and Equation 9 is no longer valid.

With the values of log A_T the master curves can be formed. First the individual relaxation curves are shifted vertically by multiplying E by T_o/T. They are then shifted horizontally by the factor log A_T. The master curves are shown in Figures 11 and 12.

TABLE 3. Determination of Log A_T Values for Silicone

°C	Δ log A_T	log A_T	T - T_o	T - T_o/log A_T	log A_T (calc)
1.7					
	6.6				
13.0		- 6.6	11.3	-1.72	- 6.19
	3.35				
19.0		- 9.95	17.3	-1.74	- 8.88
	0				
25.5		- 9.95	23.8	-2.39	-11.5
	-1.5				
29.5		- 8.45	27.8	-3.29	-12.9
	3.35				
35.0		-11.80	33.3	-2.82	-12.5
	0.35				
37.8		-12.15	36.1	-2.97	-15.5
	3.45				
40.0		-15.60	38.3	-2.46	-16.2
	4.60				
43.3		-20.20	41.6	-2.06	-17.1
	0				
49.0		-20.20	47.3	-2.35	-18.6
	6.4				
54.5		-26.60	52.8	-1.98	-19.9

TABLE 4. Determination of Log A_T Values for Nordel Ethylene

°C	Δ log A_T	log A_T	T - T_o	T - T_o/log A_T	log A_T (calc)
1.5					
	1.05				
7		-1.05	5.5	-5.23	-1.22
	0.93				
12.8		-1.98	11.3	-5.72	-2.27
	1.31				
25		-3.29	23.5	-7.13	-3.98
	0.69				
32		-3.98	30.5	-7.66	-4.23
	0.94				
43.3		-4.92	41.8	-8.52	-5.73
	1.30				
54		-6.22	52.5	-8.44	-6.47
	3.03				
64		-9.25	62.5	-6.77	-7.02

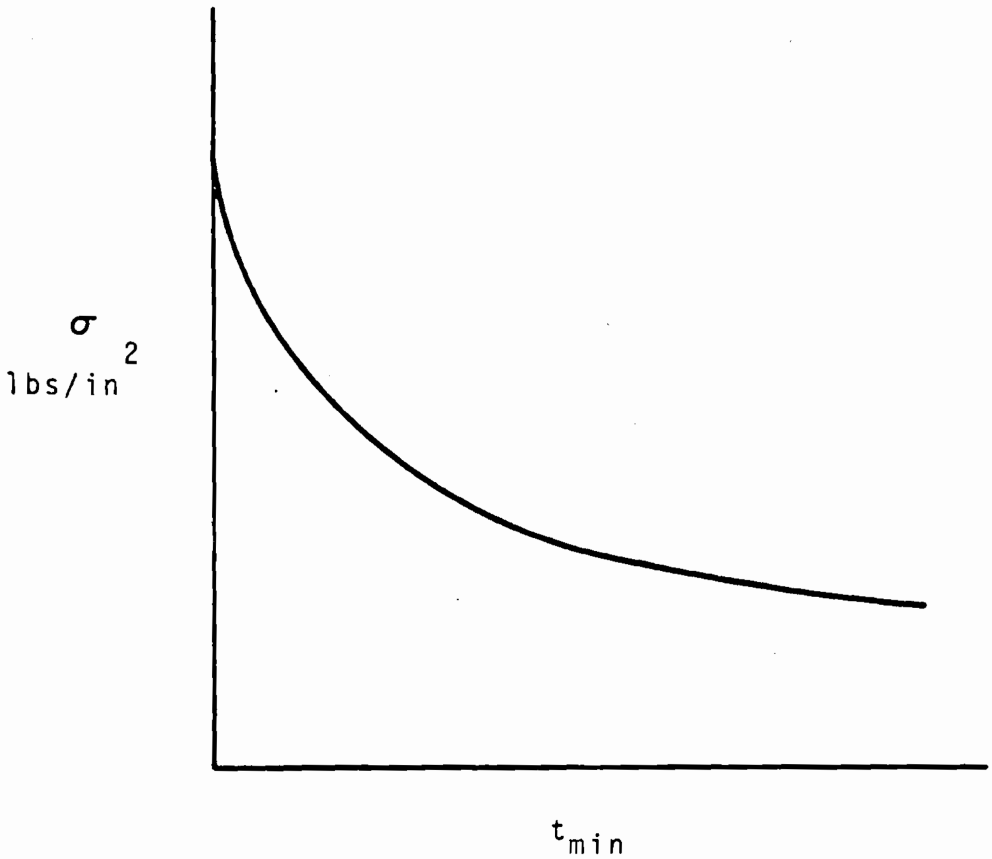

FIGURE 1. Typical Stress Relaxation Curve

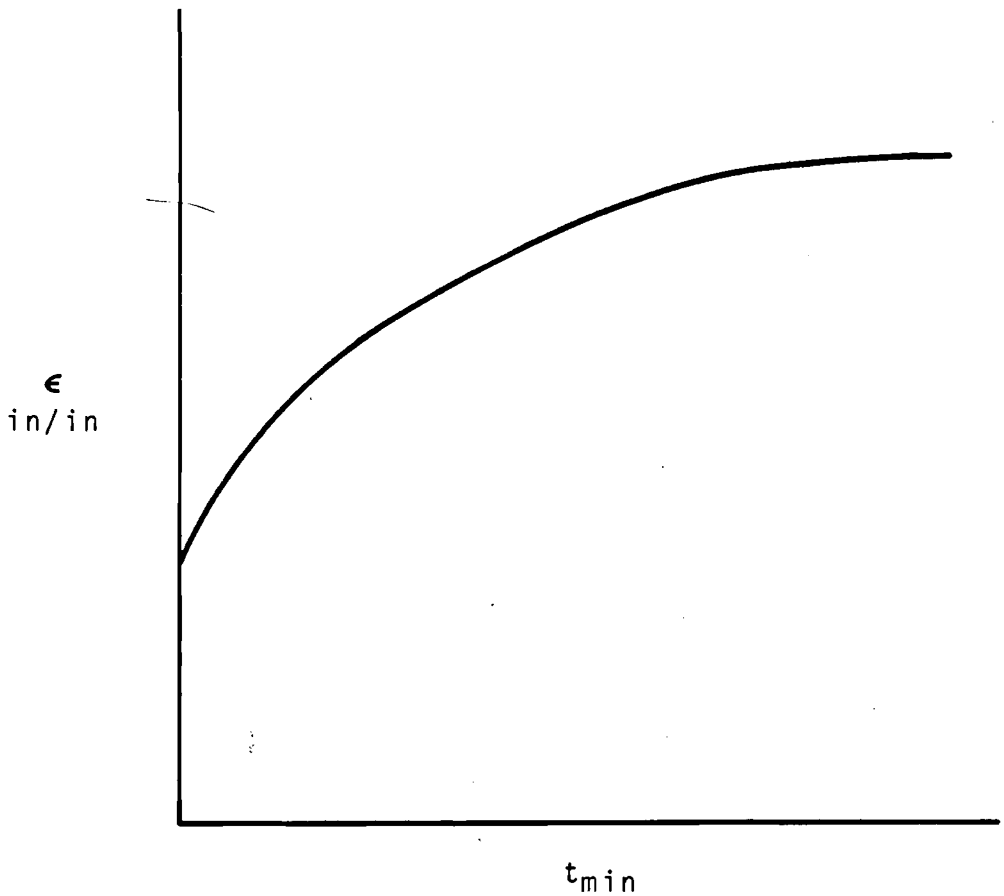

FIGURE 2. Typical Creep Curve

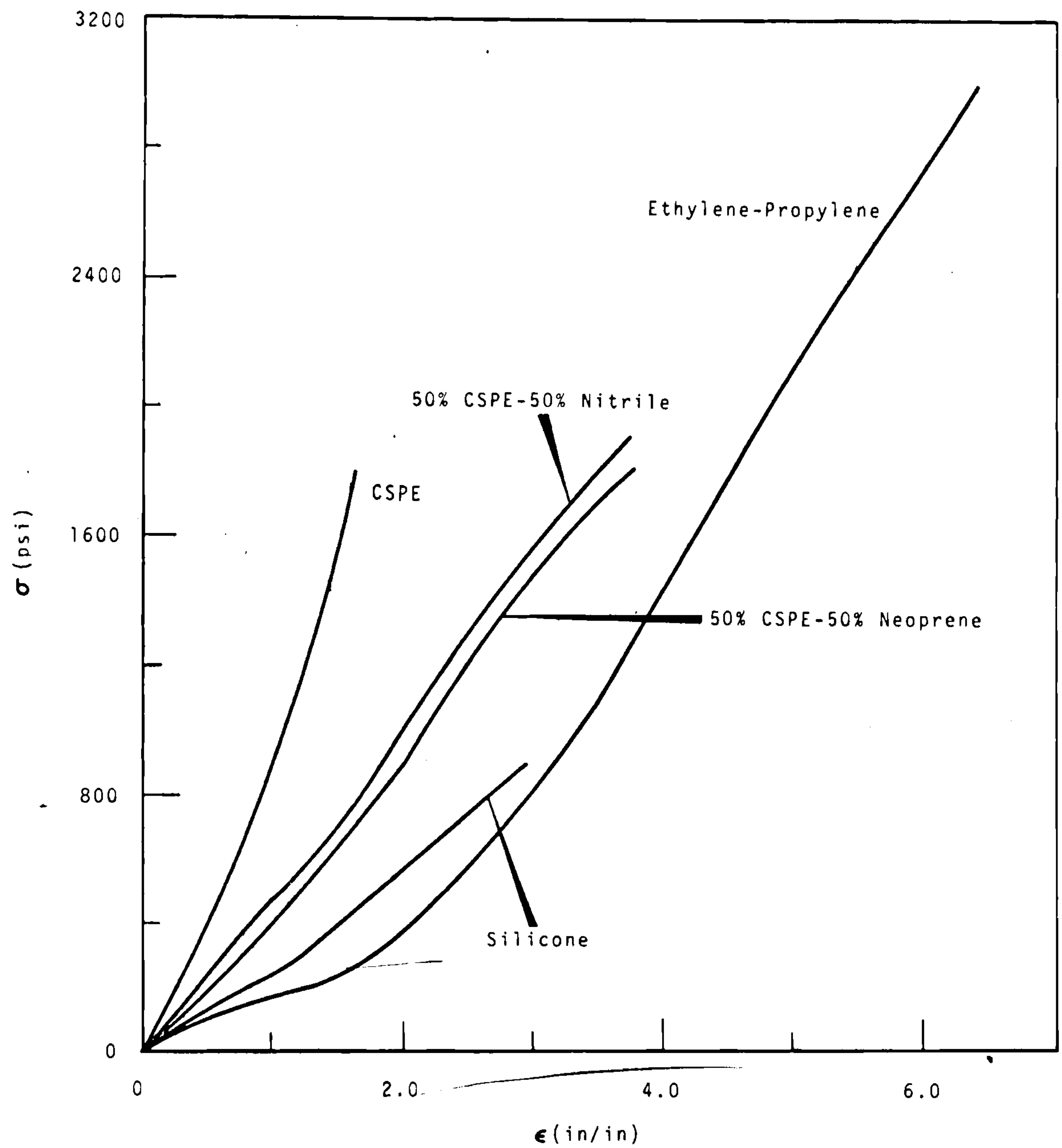

FIGURE 3. Stress Strain Curves

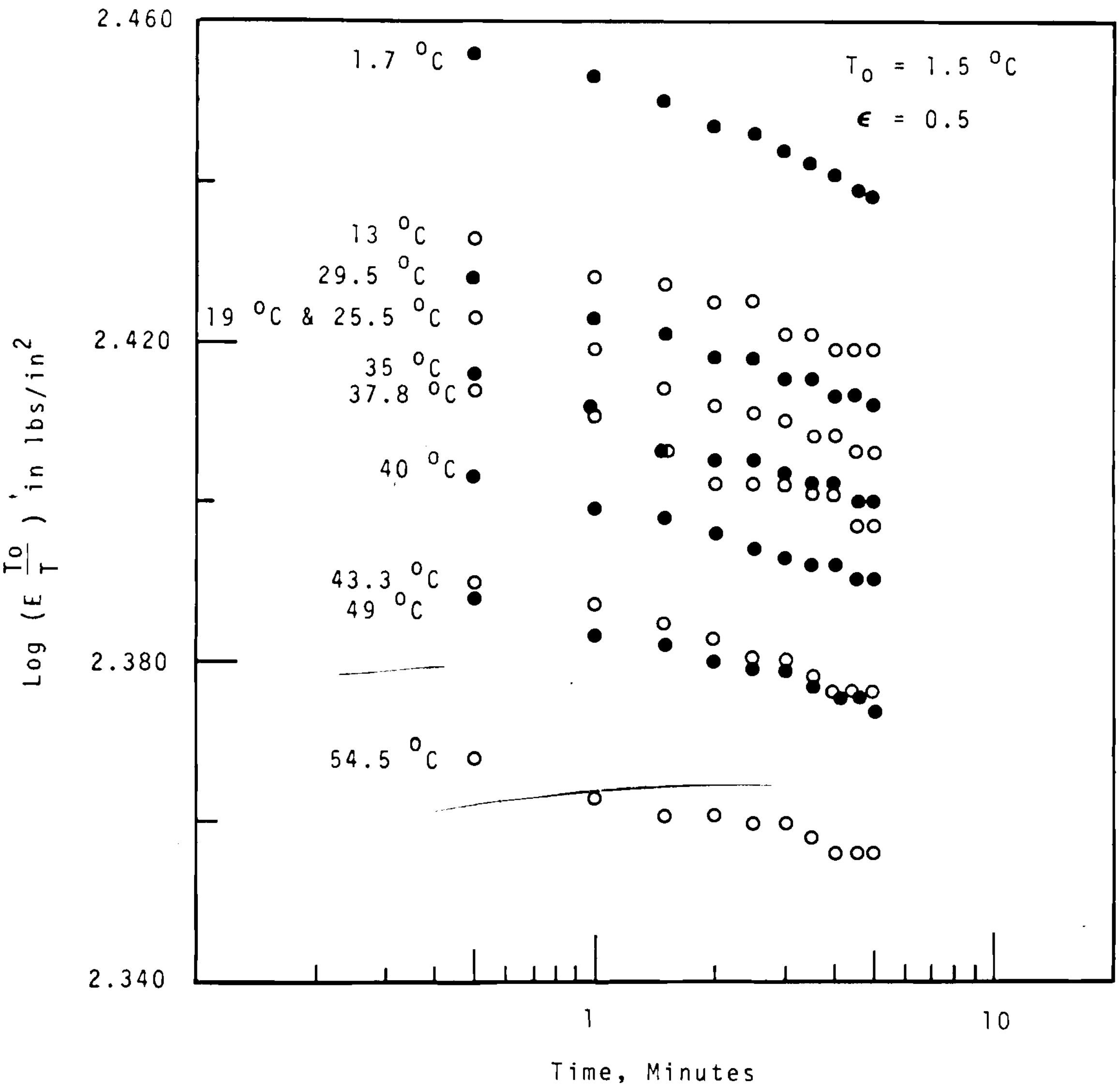

FIGURE 4. Stress Relaxation Silastic 2096

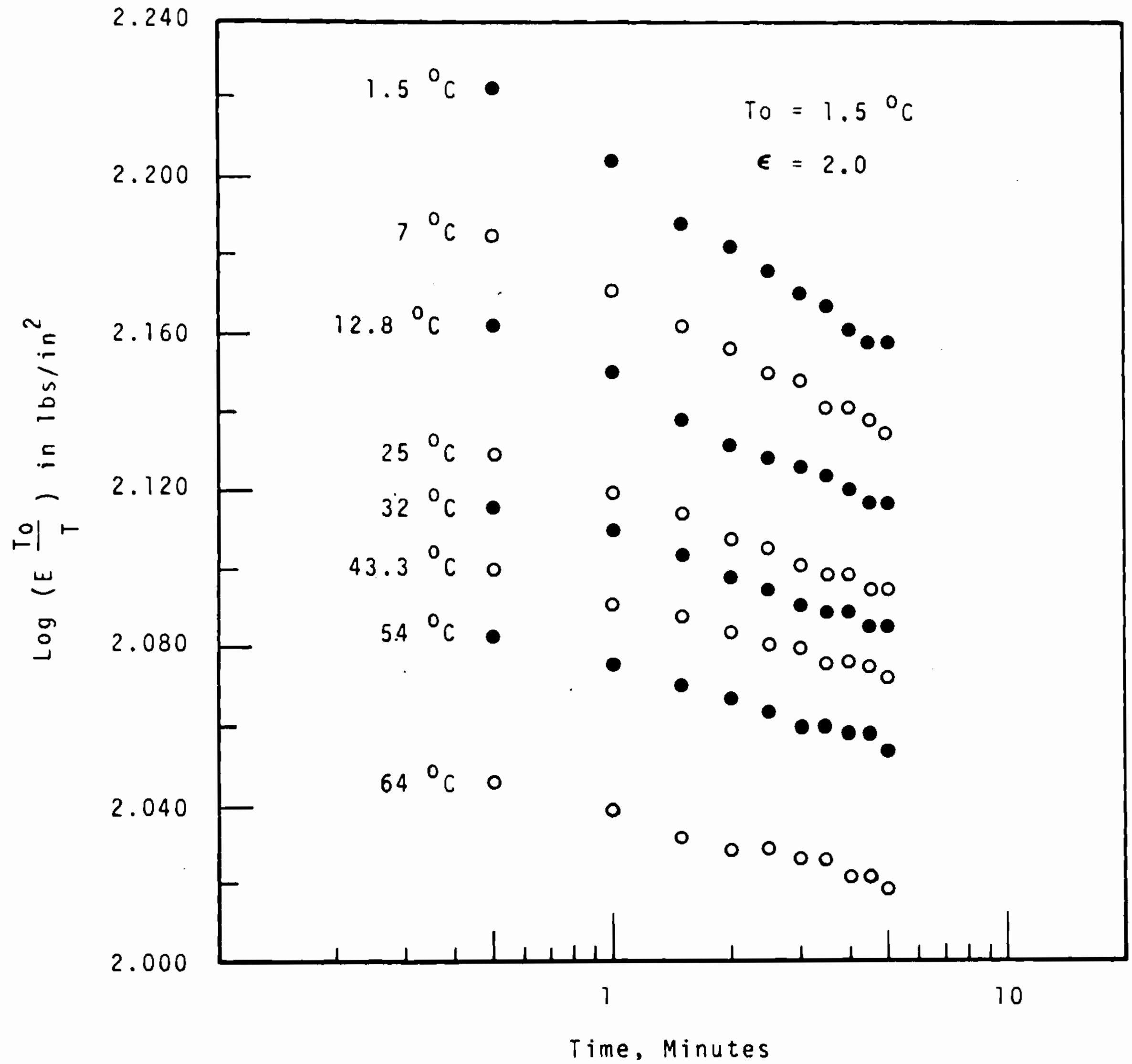

FIGURE 5. Stress Relaxation NORDEL Ethylene-Propylene

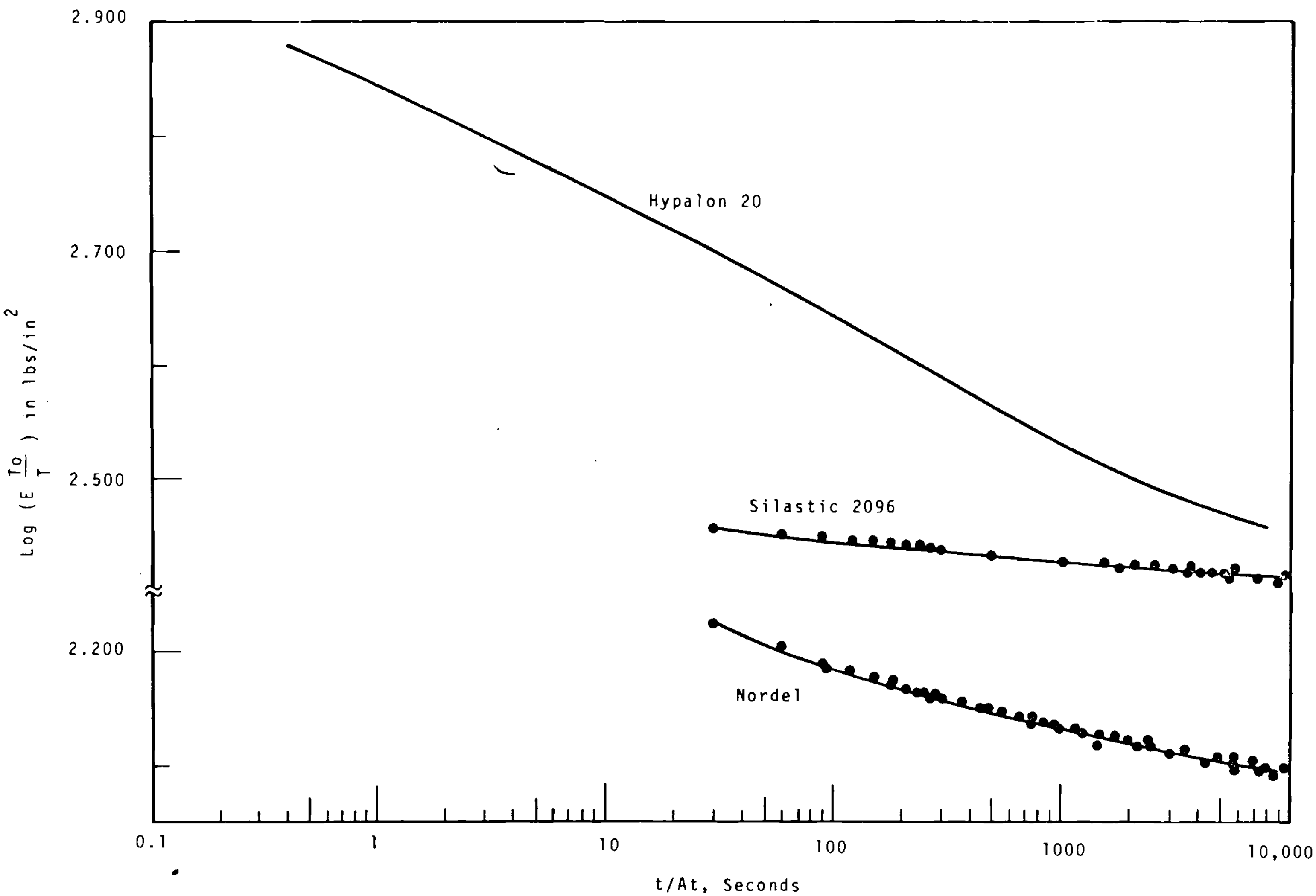

FIGURE 6. Stress Relaxation Master Curves at To = 1.5 °C

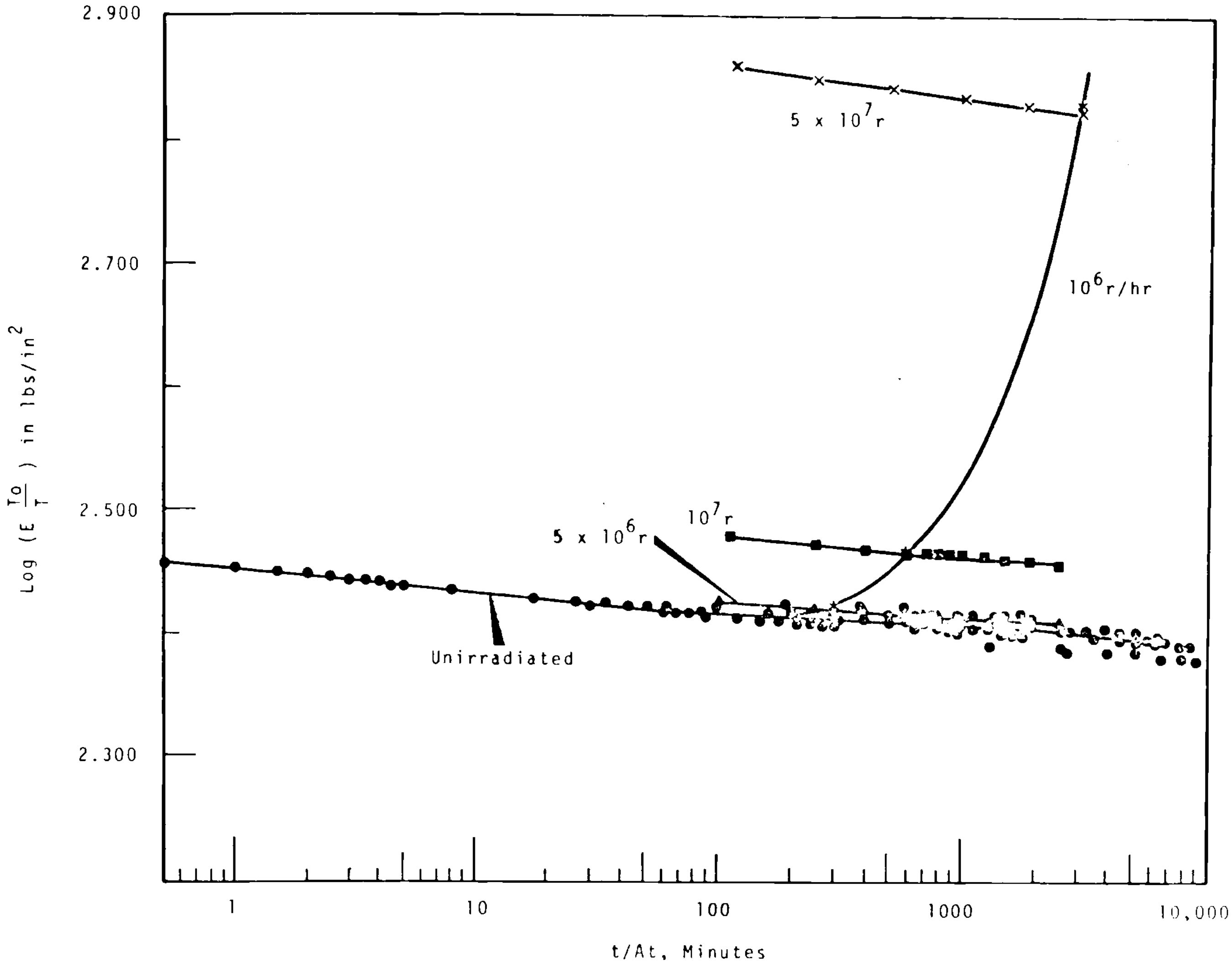

FIGURE 7. Effects of Radiation on Stress Relaxation of Silicone

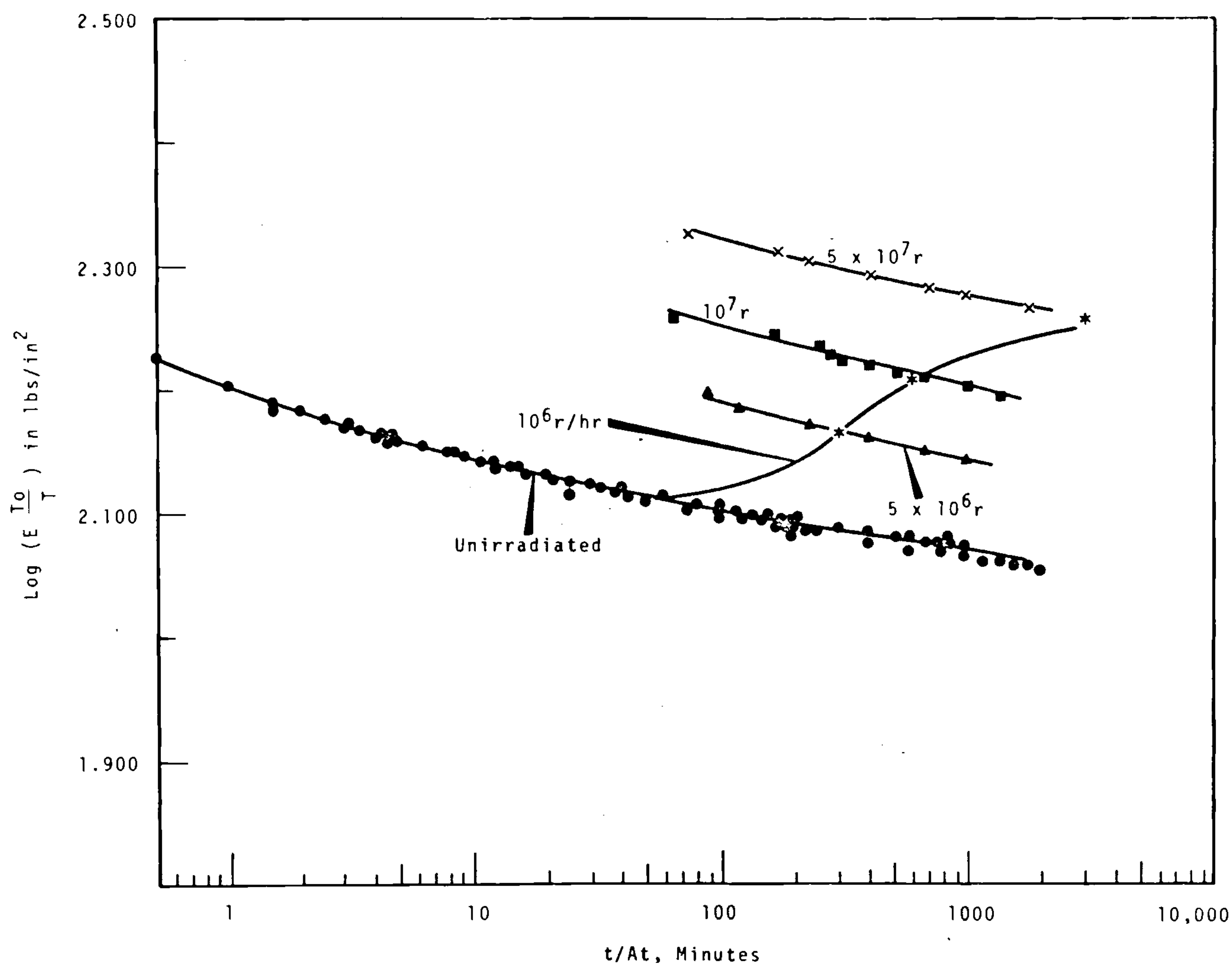

FIGURE 8. Effects of Radiation on Stress Relaxation of Ethylene Propylene

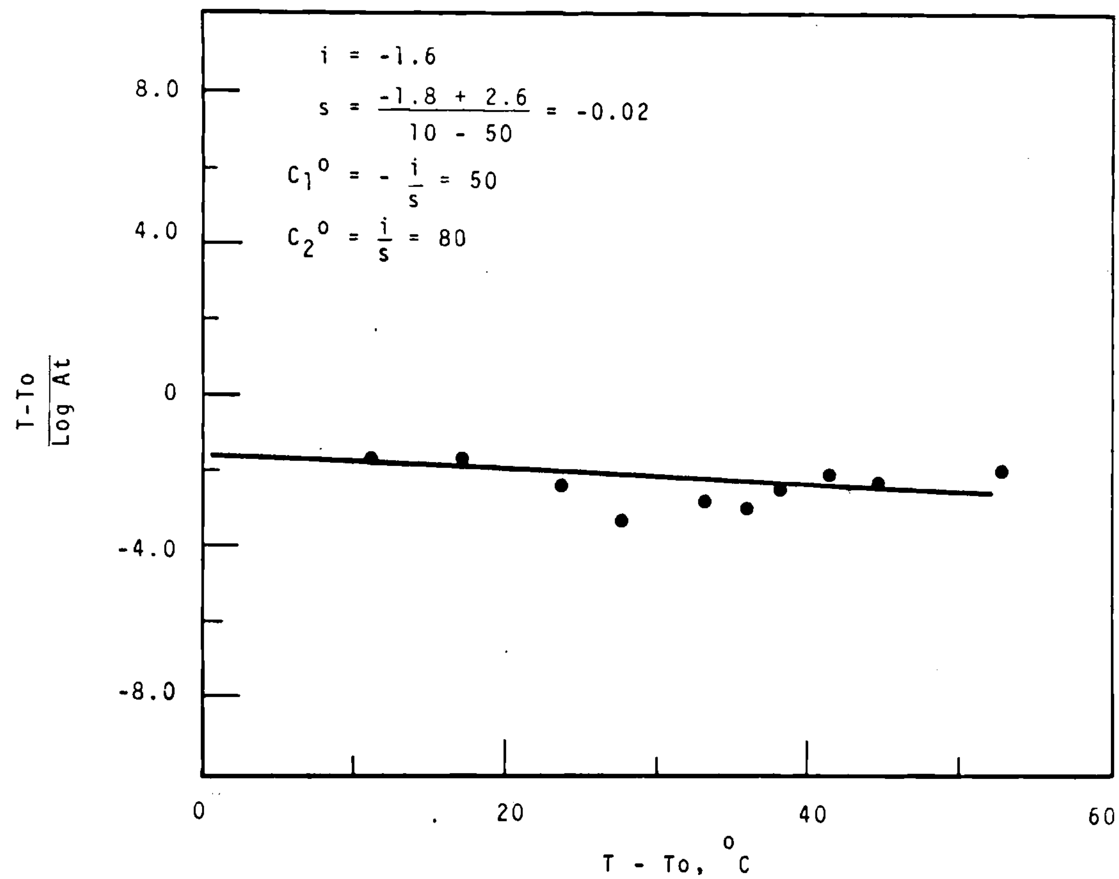

FIGURE 9. Determination of Constants of WLF Equation for Silicone

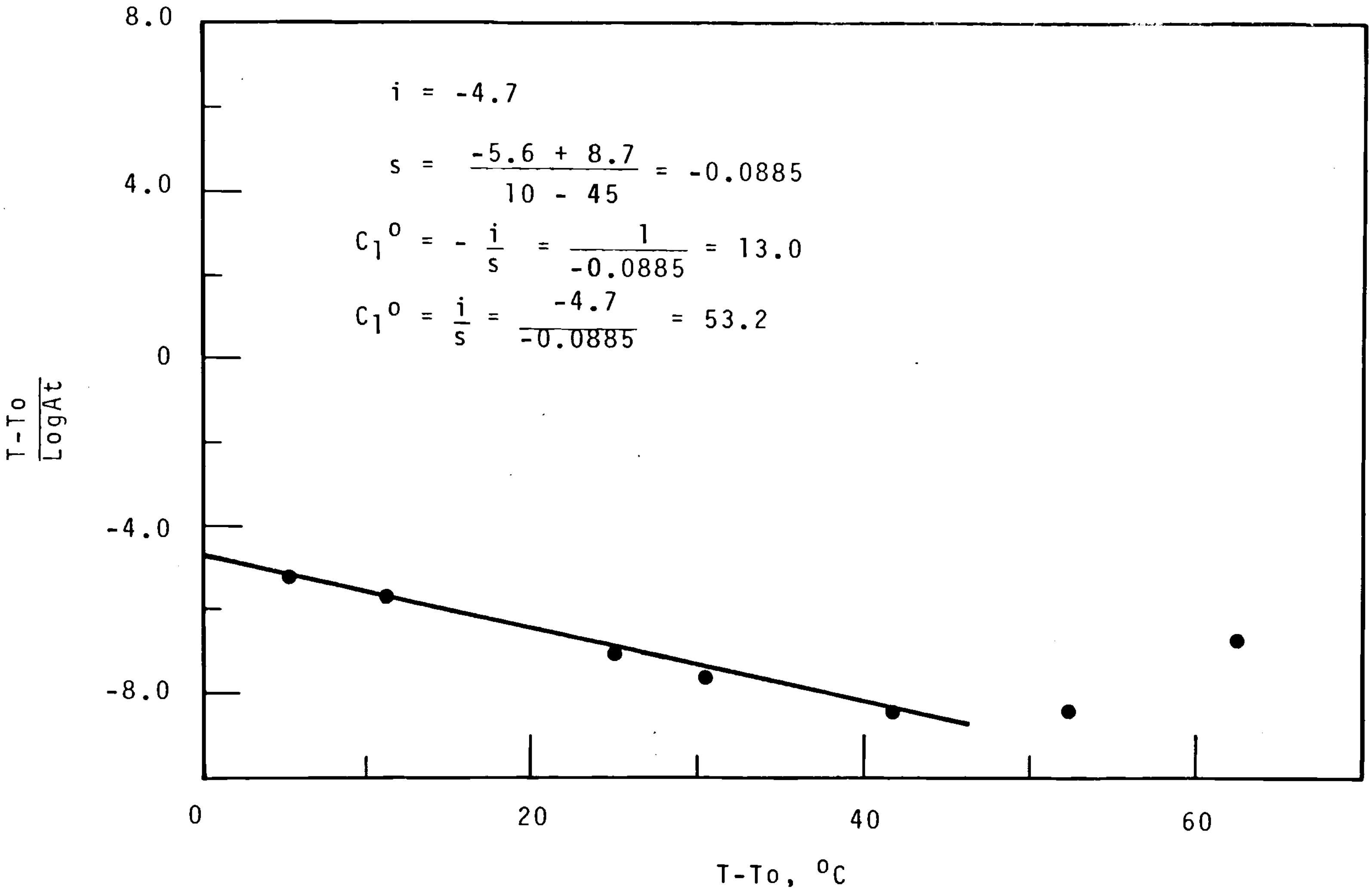

FIGURE 10. Determination of Constants of WLF Equation for Ethylene Propylene

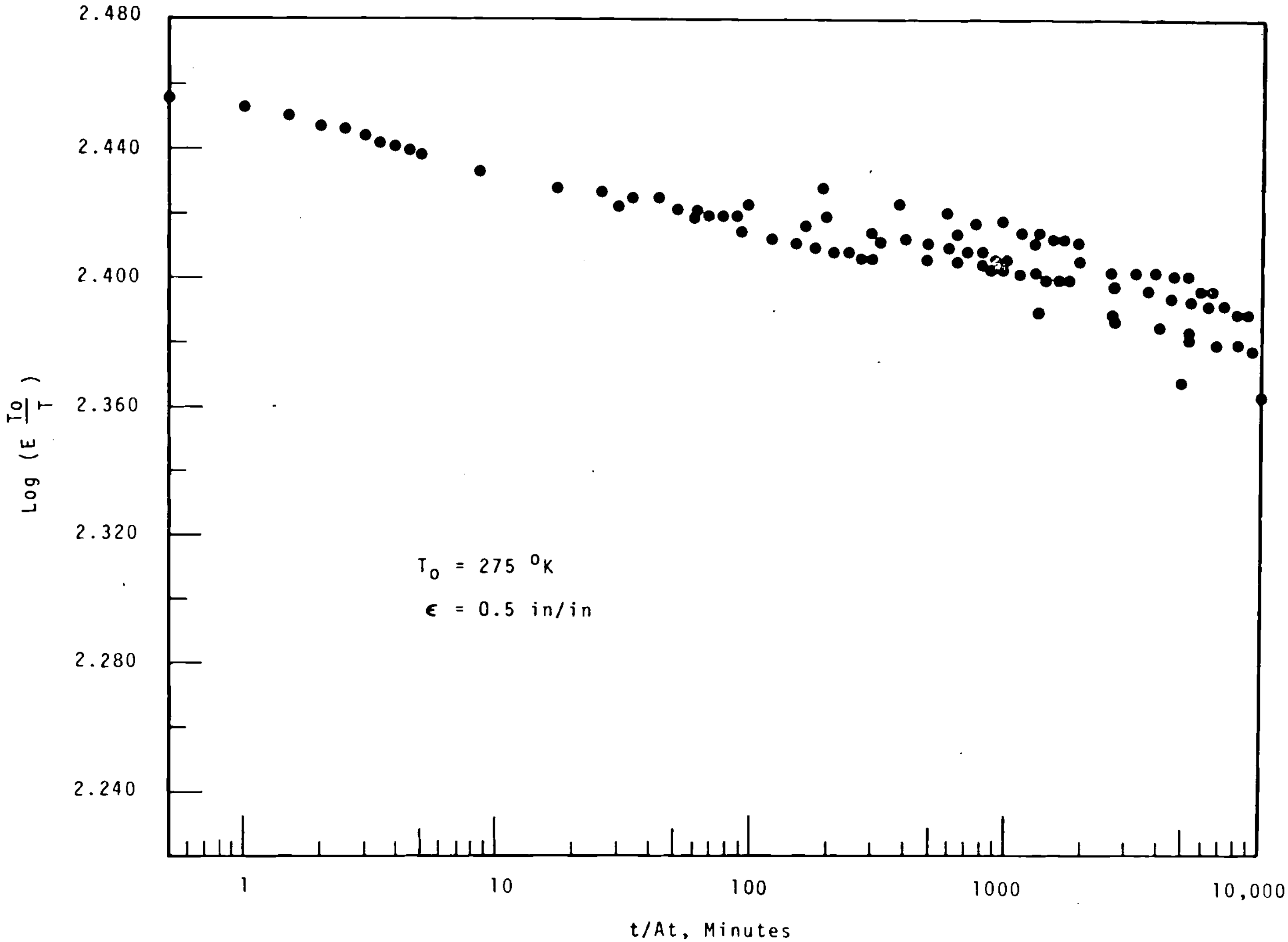

FIGURE 11. Master Stress Relaxation Curve for Silicone

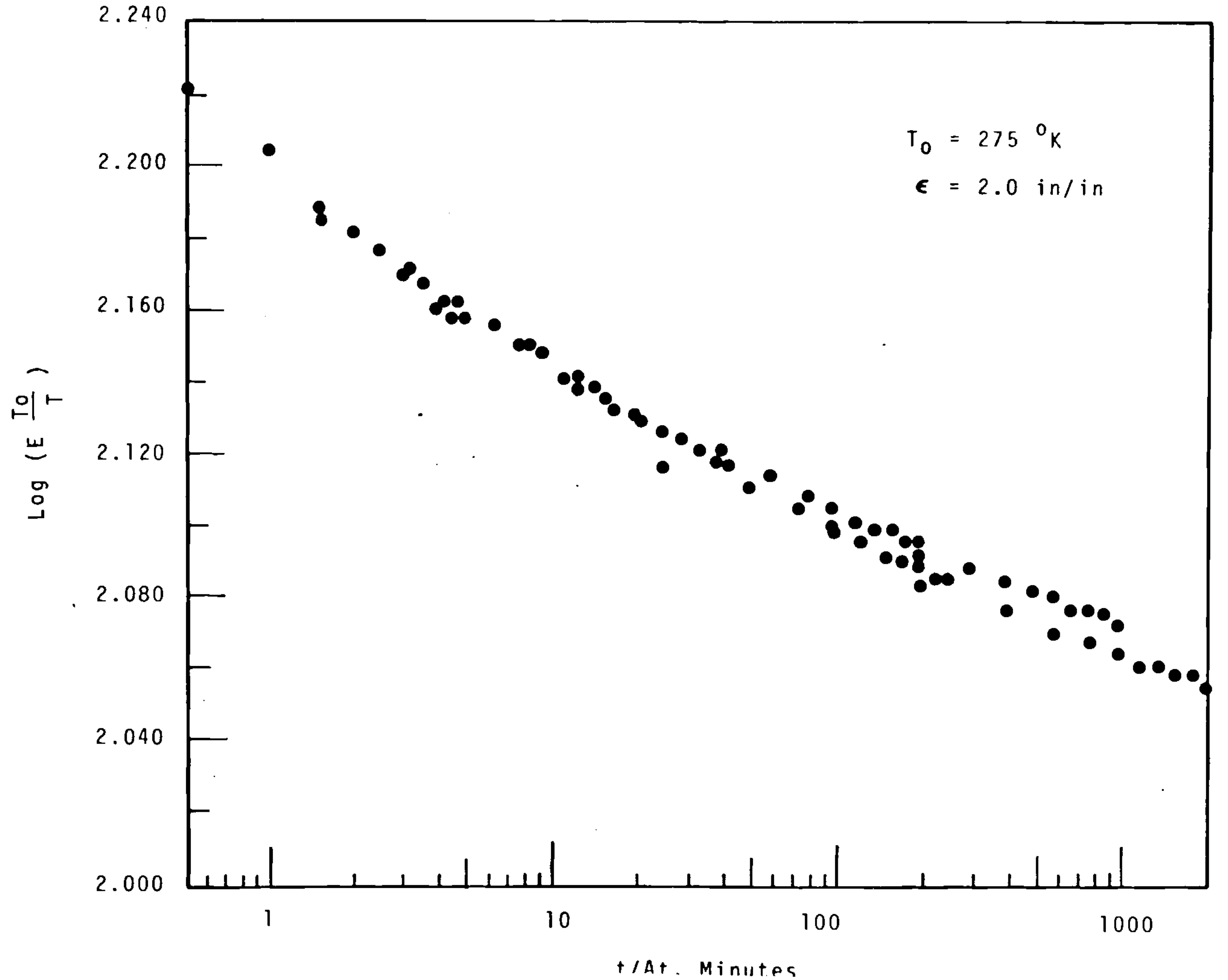

FIGURE 12. Master Stress Relaxation Curve for Ethylene-Propylene

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