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FLEXIBLE VSR'S AND
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SUMMARY REPORT
FLEXIBLE VSR'S AND VSR
CHANNEL SLEEVE DEVELOPMENT PROGRAMS

INTRODUCTION

The purpose of this report is to summarize the results of development programs which have evaluated a number of vertical rod channel sleeving materials and provided flexible vertical rods, acceptable for both interim use before rod channel sleeving, and for subsequent use in sleeved channels. With the exception of planned in-reactor performance tests on the selected "best" sleeving materials and a nominal amount of recommended rod material evaluation, these programs can be considered complete.

BACKGROUND

Graphite distortion caused vertical rod system problems have, for the past several years, been a cause for concern within Irradiation Processing Department. These problems, most prevalent at the K, C, and DR Reactors, have to date caused only minor operational efficiency losses. However the progressive nature of the graphite distortion phenomenon coupled with the high degree of reliability required of nuclear safety systems, has demanded a concerted effort toward development of acceptable long term solutions to these problems. A number of solutions ranging from continual vertical rod channel maintenance to elimination of the vertical rod system by substitution of equivalent control are in the process of being evaluated.

Vertical safety rod system problems have in the past consisted of three types of events. First, overall channel curvature and/or individual graphite bars have caused binding of the vertical rods and an increase in the length of time required for rod insertion. Secondly, rod insertion has in isolated cases been prevented by the broken ends of graphite bars which move into and block the channel. The third event, which more properly concerns operation of the Ball 3X system, results from localized gaps and openings which have appeared between adjacent graphite bars. Balls can become trapped in these gaps and crevices when the Ball 3X system operates either inadvertently or intentionally during planned functional testing.

Installation of vertical rod channel sleeves has been proposed as one method of alleviating, on a long term basis, vertical rod operational problems caused by graphite moderator distortion. This proposal has the dual advantages of providing containment of 3X balls within the rod channel in the event of Ball 3X system operation and continued utilization of the major portion of currently installed vertical rod system components. (i.e. winches, shock absorbers, step plugs, safety circuitry, etc.). A sleeve material which will resist the inclination of graphite bars to move into the rod channel is required and a sleeve column design with the capability of following continued gross moderator distortion is needed. Installation of vertical rods, with sufficient flexibility to satisfy speed of insertion criteria as channel distortion progresses, is an additional requirement for acceptable long term rod performance in channels which are sleeved.

Suggested interim action and development programs necessary to provide acceptable sleeve materials and flexible vertical rods have been outlined elsewhere.[1] Briefly, in addition to continued data collection and analysis activities, these programs provided for the following:

1. Development of equipment for channel repair to assure interim rod serviceability until sleeve installation could be accomplished by planned project action.

2. Development of equipment for renovating or enlarging both the channels through the top cast iron thermal shields and the rod channels through the graphite moderator.

3. Design and development of flexible vertical rods for procurement and use at any reactor requiring them on an interim basis. These rods were to be acceptable for continued operation in channels which have been sleeved.

4. Development of channel sleeving materials and design of acceptable sleeve column segments.

Equipment for interim channel repair and maintenance to assure rod serviceability has been available for some time, and development of equipment for enlarging vertical rod channels has progressed satisfactorily. Channel enlargement demonstrations have been conducted both on channel mock-up facilities and on one channel at C Reactor. Equipment for similar operations at the K or other reactors can be developed. This report is concerned primarily with the results of the third and fourth development programs listed above.

SUMMARY

To provide temporary relief of vertical rod binding problems at C Reactor, a flexible C Reactor size (3 inch O.D.) rod was designed, fabricated by modifying an existing spare rigid rod, tested in a mockup facility and installed for further in-service performance evaluation in #35-C VSR channel. Operating experience has shown acceptable performance over the past 21 months since installation.

Modification of smaller ($2\frac{1}{4}$ inch O.D.) rigid rods from spare rod assemblies, for flexible operation at DR Reactor provided some relief of rod binding problems while design tests were being completed and procurement of "Universal" rods was being initiated.

Tests of alternate control materials in the 305 Test Reactor, for utilization as additional neutron absorbers within BDF Reactor size ($2\frac{1}{4}$ inch O.D.) boron stainless steel rods, verified the use of B_4C filler material in the design of a Universal flexible rod applicable at all existing Hanford reactors. These rods require a minimum of spare parts inventories. Design and testing of accessory equipment completed the Universal rod development effort, except for further

 [redacted]

consideration of optimum rod sheath materials, and replacement rods are being procured for interim use before channel sleeving. A universal flexible rod was installed in a graphite test sleeve containing channel (#28) at C Reactor. Performance since installation in August 1963 has been acceptable.

High strength polycrystalline graphite is the material best suited for vertical rod channel sleeves from both a cost and availability standpoint. The material recommended for this application has compressive strength greater than 10,000 psi., flexural strengths on the order of 4600 psi, and apparent density greater than 1.8 g/cc. Nuclear grade, polycrystalline, graphites usually have mechanical strengths of about half the above values and densities on the order of 1.7 g/cc. VSR channel sleeves, fabricated from high strength graphite should be capable of resisting normal crushing loads on the order of 6000 lbs. Maximum in-reactor sleeve loads attributed to continued graphite distortion are not expected to exceed 3000 pounds after ten years of service and average sleeve loads are expected to be in the 200 to 500 pound range.

Development efforts have produced sleeving materials with a wide range of strength and deflection characteristics. Of the materials considered, aluminum oxide has the highest load resisting capacity. This material, when fabricated with uniform 1/2 inch walls to sleeve dimensions applicable at C, DR or H Reactors will resist external crushing loads in the 20,000 to 30,000 pound range. Fabrication problems associated with production of sleeves with non-uniform walls, such as believed necessary at the K Reactors, limit consideration of aluminum oxide in the K Reactor sleeving application. In addition to fabrication problems, a significant economic penalty would result from added enrichment to compensate the loss of reactivity associated with installation of aluminum oxide sleeves in any reactor.

Sleeves with the capability of large deflection under load before sleeve collapse can be produced by impregnating various pyrolized substrate materials with pyrolytic carbon. Hemp, jute cloth, twine and cheesecloth are some of the substrate materials evaluated. These materials, which are acceptable from the standpoint of having no associated reactivity loss, are not competitive in cost comparisons with other materials.

Pyrolytic carbon impregnation of polycrystalline graphite materials has also increased, by 40 to 100%, the strength of the base materials. However, development efforts to date have indicated that a relatively weak, low density (1.37 g/cc) substrate material is required to allow maximum pyrolytic carbon penetration. Although significant strength increases were obtained, the final sleeve material does not compare favorably with other high density, high strength polycrystalline graphite materials.

The performance of polycrystalline graphite and aluminum oxide sleeves have been evaluated in mock-up tests. While these materials were both found to be acceptable from rod impact and sleeve wear considerations, aluminum oxide eventually abrades the rod. None of the materials evaluated will withstand internal pressures developed by events such as rod insertion in a channel partially filled with 3X balls. This type of sleeve load, while unlikely, would require sleeve removal and replacement. In the event that localized sleeve failure occurs, either from isolated events such as described above, or extremely high external sleeve load transmitted through an adjacent tube block containing a stuck, ruptured fuel element, it is desirable that sleeve segment removal be as easy as possible. Graphite sleeves have obvious advantages from the standpoint of sleeve removal, when compared with ceramic sleeve materials.

Procurement of test quantities of high strength polycrystalline graphite sleeves, for in-service performance verification at C Reactor, is planned. Acceptable performance of these sleeves at C Reactor should provide an adequate basis for procurement of similar material sleeves for use at the K Reactors.

DISCUSSION

I. Flexible Vertical Safety Rods

Flexible vertical safety rod development has included three separate but interrelated efforts; flexible C Reactor size rods, flexible BDF Reactor size rods for DR Reactor and "Universal" flexible rods.

Since rod binding problems attributable to channel distortion were first noted at C Reactor, and since operational spare rod inventories were limited, initial development was directed toward provision of a flexible C Reactor size (3" nominal O.D.) rod. A flexible rod joint, similar to that previously used for BDF Reactor size ($2\frac{1}{4}$" nominal O.D.) rods, was designed for adaptation to the existing spare C Reactor rigid rod assemblies. An operational spare rigid rod was modified for flexible operation and subjected to a number of design tests in a mockup in 195-D Building.[2] These tests included measurement of rod flexibility, determination of insertion time versus rod displacement when inserted in a (simulated) distorted channel and observations of general rod operability after greater than 500 cycles of operation.

After completion of mockup tests a second C Reactor rigid rod was modified for flexible operation and installed in #35-C channel for in-reactor evaluation.[3] The channel selected for trial installation was one in which a BDF size rigid rod, previously installed to alleviate rod binding, was no longer meeting rod insertion time requirements. The C Reactor size flexible rod operated satisfactorily in this channel for about 14 months after which time rod binding was again noted. The flexible rod was removed and re-installed subsequent to channel broaching. This rod has continued to perform satisfactorily an additional 7 months to the present time.

An evaluation of the equipment requirements for BDF size flexible VSR installation at DR Reactor was conducted. This study led to installation of a BDF size flexible rod in one channel at DR Reactor.

Accessory equipment design requirements for step-plug sleeve inserts, guide cables, etc., were forwarded to D-DR Maintenance Engineering for further action in modifying rods from the limited spare parts inventories available at that time. Installation of smaller rods both at DR and C Reactors has provided temporary relief of rod binding problems in the interim time required for Universal Flexible rod procurement.

Simultaneously with the above interim corrective measures to assure continued acceptable VSR performance, a program was undertaken to evaluate the relative merits of various neutron absorbers designed for insertion within the hollow BDF size rod sheath. The materials included consideration of B_4C powder, Samarium oxide in an aluminum oxide matrix (NPR - 3X balls) and B_4C - graphite compound similar to the material currently used in K Reactor vertical rods. The measured worth of the various materials were determined relative to a section of K Reactor rod, by pile period measurements in the 305 Test Reactor. These tests indicated that an increase in nuclear effectiveness of approximately 20% could be obtained (relative to the K Reactor rod) by packing boron carbide inside a BDF Reactor size hollow vertical rod sheath fabricated from type 304 Boron stainless steel.[4] An analysis of these test results[5] showed that, in terms of total reactor control strength, the proposed BDF size rod with B_4C filler material would also provide an acceptable replacement rod for C Reactor.

The above tests and analysis of test results indicated that, with the exception of rod length and other relatively minor accessory equipment requirements, the same basic flexible rod design could be used at all the currently operating Hanford reactors; hence the designation "Universal" flexible rod. Modification of the flexible rod joint design previoulsy used for BD and F Reactors allowed greater flexibility without loss of joint integrity.[6]

Prior to ordering replacement quantities of universal flexible rods, a sintered B_4C compact rather than B_4C powder was specified as the rod filler material. Also, to verify in-service universal rod performance in sleeved vertical rod channels, a universal rod was fabricated from a BDF size spare rod assembly and installed in mid 1963 in the first sleeve containing channel (#28) at C Reactor.[7] Because of the unavailability of the sintered B_4C compact at the time, powdered B_4C was used as rod filler material in this rod.

The basic rod design has now been adapted to all reactors requiring replacement rods, accessory equipment design has been completed, and procurement has been initiated on interim replacement quantities of the required equipment. Final testing, to assure nuclear acceptability of each heat of boron steel used in rod fabrication, is currently in progress.

Since the outside diameter of the universal flexible rod is less than both existing C and K vertical rods, the universal rod is particularly well suited for application in sleeved channels. The smaller diameter rod allows slightly greater latitude in sleeve wall thickness selection and also provides greater free area within the sleeve for 3X ball flow around an inserted rod. The life expectancy of the smaller diameter rod is also expected to be greater since greater sleeve column curvature can be experienced before reaching the limits of rod flexibility.

Since the rods are suited for use after channel sleeving has been accomplished, all rods which are procured on an interim replacement basis now, need not be replaced by any sleeving project. A major advantage of the universal flexible rods, not possessed by previous rod orders for particular reactors, is the uniformity and interchangeability of components and the smaller permissible spare parts inventories that need be maintained, i.e., a replacement rod for B Reactor for example, can be used at K Reactor simply by installing the appropriate head assembly and adding additional rod segments to attain the required length.

The universal flexible rod sheath material consists of boron containing type 304 stainless steel. This material has also been used in all previous Hanford vertical rods which specified that the nuclear poison material be integral with the rod sheath. Although operating experience has indicated the type 304 boron stainless material to be a serviceable rod material, the boron addition causes low ductility and fabrication techniques which result in relatively high rod costs when compared to rods which do not require the nuclear poison to be integral with the rod sheath.

Recognizing some of the mechanical property limitations of type 304 boron stainless rod materials, a previously incompleted program to determine the suitability of a vacuum melted, type 430 boron stainless steel for this application was reactivated early in 1963.[8] The results of tests performed on extruded tubular sections of the type 430 boron stainless material[9] show this material to be, in many respects, superior to the type 304 boron stainless sheath material. These tests compare machineability; weldability, tensile strength, (under ambient, heat cycled and hot conditions) impact resistance, flattening tests and hardness of the two materials. In each test the type 430 boron steel appeared to be better suited than the type 304 boron steel for the vertical rod application. Since the above tests were all conducted on an experimental heat of type 430 material which was obtained several years ago, a realistic current price for this material is not available. The remaining work contemplated as part of the type 430 boron stainless steel development effort consists of requesting bids from various manufacturers on the cost of fabricating several different quantities of vertical rods from the type 430 material. Evaluation of these bids will determine whether or not future orders for replacement rods will specify type 430 rather than type 304 stainless sheath material. Comparison of the costs of the two rod materials will also be considered in a decision regarding the necessity for in-reactor evaluation of the type 430 material on a one or two rod basis.

II. Sleeve Material Development

Sleeve Requirements

Dimensional changes in the graphite moderator resulting from long term exposure to neutron irradiation, and the unique service requirements the sleeves must fulfill, complicate selection of an adequate channel sleeving material and development of an acceptable sleeve design. A segmented sleeve column, which will conform to rod channel curvature as moderator distortion progresses, is required.

The requirements for an acceptable rod channel sleeve can be stated as follows:

1. The sleeve must withstand temperatures on the order of 500 C - 1000 C for a number of years.

2. The sleeve must have the capability of withstanding impact loading from rods weighing up to 500 pounds which are driven into the reactor and may reach velocities of 1500 feet per minute. The impact loads on the liner can be reduced by use of segmented or flexible vertical rods.

3. The sleeve must resist forces on its exterior surface. These forces, which may result from thermal expansion of the moderator bars, or some combination of vibration, gravity forces, thermal expansion and irradiation caused dimension changes in the moderator, may be relatively high or extremely low.

4. The liner material must have a low absorption cross section so that significant reduction of rod effectiveness will not result from liner installation or so a significant economic penalty for enrichment is not required for reactor operation.

In addition to the above requirements that apply primarily to the liner material, a number of design restrictions must also be considered. Spacing of process tubes containing fuel elements effectively limits the maximum outside cross sectional sleeve dimensions. The necessity for adequate free area within the liner, to allow long term rod operation as the sleeve column continues to distort, places a restriction on maximum permissible liner wall thickness. The length of individual liner segments must also be restricted to allow the liner column to curve as moderator distortion progresses without opening gaps between adjacent column segments. The usual design requirements for a material which can be fabricated to close dimensional tolerances and which can be obtained at a reasonable cost must also be considered.

Material Evaluation

Evaluation of alternate materials for application as uncooled channel liners was originally initiated as a result of horizontal rod operating difficulties experienced at the K Reactors. These operating problems were caused by migration of 3X balls from the vertical rod channels to the horizontal rod channels. The nuclear properties of five different materials were reviewed to determine the maximum temperature which might be attained by each material when used as an uncooled sleeve, the effect each would have on the reactivity of the reactor, and the effect of each on the horizontal control rod system.(10) The five materials considered were: Inconel, a 300 series stainless steel, Zircaloy-2, aluminum oxide and zirconium silicate. The mechanical and corrosion properties of these materials at the expected temperatures were not considered by the initial evaluation.

All of the materials, with the exception of Zircalloy-2, would be expected to withstand the reactor environment satisfactorily. Table I summarizes the results of the original analysis of sleeve materials for horizontal control rod application at a K Reactor. The economic penalties associated with Inconel and stainless steel sleeves ($1,130,000 and $780,000 respectively per reactor per year) are seen to be quite significant when wall thicknesses sufficient to provide reasonable structural integrity are considered. The zircaloy-2 and NPR Process tube sleeves, while requiring acceptance of only nominal economic penalties ($22,000 and $41,000 per year, respectively) were not expected to withstand reactor environmental conditions.

The suitability of zircaloy-2 was measured by estimating its corrosion rate in the reactor gas.(11) It was assumed that the oxidizing agents in the gas (O_2, H_2O, CO_2 and CO) reacted with the zircaloy-2 HCR sleeves uniformly throughout the reactor and did not react with other structural materials in the stack. It was further assumed that the reaction of each of these oxidants obeyed one law i.e., a combination of a cubic and linear expression.(12) Under these conditions the corrosion rate was estimated to be ten mils per day.

Assuming that the oxidizing agents react uniformly over the total surface area of the HCR sleeves, and that the quantities of the oxidizing agents are: CO_2 1.95%, O_2 0.13%, CO 0.39% and H_2O 0.06%, a gas flow rate of 450 cubic feet per minute can corrode zircaloy-2 at a rate of only one mil per day.

In view of the fact that the corrosion rate of all or some of the K Reactor HCR sleeves, if made of zircaloy-2 would be of the order of one to ten mils per day, zircaloy-2 was considered unsuitable for this service.(11) A later analysis of zircaloy-2 for K Reactor VSR channel liners, based principally upon the rate at which the material would become saturated with oxygen and nitrogen, showed an estimated service life expectancy no greater than six months.(13) The material was again judged unsuitable on the basis of estimated service life.

When vertical rod binding as a result of graphite distortion became an operational problem at C Reactor, channel sleeving was again the suggested course of action which appeared to have the most merit and efforts toward development of an acceptable sleeve design were intensified.

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Beryllium, because of its highly desirable nuclear characteristics, was one of the first candidate sleeving materials considered for possible rod channel sleeve application. Both beryllium metal and beryllium oxide were evaluated. Beryllium metal was rejected because of poor corrosion resistance, unsatisfactory mechanical properties, toxicity and high cost. Beryllium oxide was rejected primarily on the basis of cost and toxicity considerations.

Beryllium Metal

Although beryllium metal has served successfully both as a moderator and reflector material, its utilization in structural applications has been limited by the following considerations:[14][15]

1. Unsatisfactory mechanical properties; particularly the lack of ductility at room temperature, which results in elongations of less than 2 percent. The transition temperature from brittle to ductile behavior is in the range of 200 to 300 C. The lack of ductility is a serious problem during fabrication.

2. Susceptibility to radiation damage. Except for the fuel materials, beryllium is probably the only reactor material that forms gas within the metal lattice under irradiation. The volume of helium formed in beryllium is less than that of the fission gas released in uranium at the same irradiation level, but under certain conditions, it is sufficient to destroy the structural properties. At low temperatures (500 C or below), helium will form only small bubbles in the metal lattice. As the temperature increases to a 700 C range, larger bubbles will accumulate by diffusion, and swelling will take place up to a 30 volume percent increase at 1000 C. Ductility decreases can be expected in heavily irradiated beryllium at temperatures of about 700 C.

3. High cost and toxicity. The cost of beryllium powder or bead in 1959 was approximately $70 to $100 per pound. Fabricated material would of course be considerably higher than the cost of powder or bead.

   Because of the toxic effect on humans, the AEC has recommended strict criteria for the fabrication and handling of beryllium. Beryllium metal is the least dangerous form, but Be_2C is extremely hazardous.

4. Lack of satisfactory fabrication methods. Fabrication methods are apparently still in the developmental stage. In 1959 lengths of tubing with .25 inch wall and 4 inch OD were reported to have been extruded in experimental lots. Powder metallurgy techniques were developed and slip casting was successfully demonstrated at the Stevens Institute of Technology late in 1960.[16] Also, the possibility of producing hollow, thin walled shapes by drain casting and sintering was shown. However, it was not believed that these methods were sufficiently advanced for commercial production of channel liners.

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5. Lack of corrosion resistance in CO_2 at the reactor operating temperatures. The behavior of beryllium in CO_2 at 600 C and above is not fully understood; it apparently was not considered by the British to be satisfactory in view of their decision to employ stainless-steel-clad fuel elements in the AGR and to restrict the use of beryllium to experimental channel loops. In arriving at this decision, progress on fabrication development was considered to be slow but satisfactory, the irradiation resistance of beryllium was considered to be adequate, and the major difficulty was considered to be the lack of corrosion resistance. The reaction of sintered beryllium in a nitrogen atmosphere has also been investigated at temperatures from 650 to 925 C.(17) Little corrosion was found at less than 650 C but larger reactions occurred above 700 C.

Beryllium Oxide

Beryllium oxide is generally stable and inert toward most materials. At high temperatures however, beryllium oxide does react slightly with carbon, and reacts significantly with water vapor.(18)

As mentioned previously, beryllium and its compounds are toxic. This toxicity provided an additional basis, other than cost, for the original rejection of beryllium oxide as a candidate sleeveing material. The following factors were influential in the decision to limit consideration of beryllium oxide for the sleeving application.

1. All operations involving dust of fumes from beryllium must be done in carefully designed shops where exhaust facilities are available. Although toxicity problems would undoubtedly be handled during part fabrication by the vendor supplying the material it was believed that additonal care over and above that normally required for work with irradiated materials, could easily be required subsequent to installation.

   It was originally believed that a slow beryllium-carbon reaction could occur at the reactor operating temperatures. At a later date, it was determined that Be_2C formation would be unlikely.

   The product of beryllium and water vapor is beryllium hydroxide in a stable gaseous form. At temperatures as low as 1000 C, concentrations of beryllium hydroxide a factor of nine greater than the limit of two micro-grams per cubic meter recommended for health reasons can be expected.(19) It was postulated that the occurrence of water leaks, and defective gas seals, coupled with localized positive reactor pressures could necessitate monitoring procedures to assure that acceptable atmospheric concentrations of beryllium were not exceeded.

   Considering the possibility that any material selected for trial installation might well have to be removed prior to final sleeve material selection, it was judged that the additional care required and delay resulting from provision of adequate assurance of personnel safety, were also factors weighing against the material.

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2. The strength characteristics of aluminum oxide and beryllium oxide are comparable as indicated below:

	Coors Porcelain Co. (1960) Data	Beryllium Corp. (1960) Data
Tensile Strength (psi)	17,000 - 18,000	17,500
Compressive Strength (psi)	210,000	200,000
Flexural Strength (psi)	42,000 - 45,000	25,000

The strength characteristics are apparently slightly higher for the more recently produced beryllium materials, but the strength of aluminum oxide is of the same order of magnitude and should be less costly. The cost estimate for pure or reactor grade BEO, obtained on the basis of half-shell liners previously considered for application in the K Reactor HCR channels, was in the range of $887.00 to $1,056.00 for each twelve inch section of liner. These values were obtained from National Beryllia Corp. on a low bid basis for 680 and 34 part quantities respectively. On this basis, sufficient sleeves to line 20 VSR channels would cost between $580,000 and $697,000. Aluminum oxide sleeves should be obtained for about $150 per foot or about $100,000 per 20 VSR channels, if this material were selected for final installation.

Since ceramics appeared from these initial studies to possess merit as candidate sleeving materials, considerable emphasis was placed on this type of material in subsequent analyses. It soon became apparent however, that the available literature was deficient with regard to the effects of long term neutron and gamma radiation on the physical properties of many ceramics. A thorough search of the literature back to 1943 yielded references to only ten reports.[20] The nature and quantity of data available showed that physical and mechanical properties of ceramics changed under irradiation but these data would not permit an estimation of the rate of change of properties relative to a K or C Reactor.

The most promising candidate sleeving materials appeared to be polycrystalline graphite, aluminum silicate, aluminum oxide, and silicon carbide. These materials were selected because of their potential ability to withstand the temperature and radiation environment to which they would be subjected and because of their low cost and low neutron absorption cross sections relative to some other ceramics.

Irradiation Damage Tests

Recommendations[20] that candidate ceramic sleeve material samples be irradiated in a flux field similar to in-service sleeve conditions prior to acceptance led to selection of three materials for this purpose. Compression and flexural test specimens of aluminum oxide, silicon carbide and aluminum silicate were ordered for placement in a test channel at C Reactor for long term irradiation. Later analysis of aluminum silicate showed that this material was extremely brittle and, on that basis, it was rejected as a candidate sleeving material. The aluminum silicate samples were, however, on order and were easily included in the irradiation damage tests although the material was no longer under consideration for channel sleeving.

Graphite boats containing the various ceramic specimens were placed in the test channel in December 1961. Starting six months after initial exposure, representative samples have been periodically discharged and destructively tested. Current planning provides for continuation of these tests until late 1964.

To date, the test specimens have attained exposures on the order of 4100 MWD/AT, i.e. 4.0 $(10)^{20}$ nvt; $E > 0.18$ Mev. Irradiation temperature has been 550 C to 600 C. While considerable scatter is evident in values at which individual samples failed, the following generalizations can be made.

Aluminum Oxide

It appears that the material flexural strength which was initially 25,000 psi, passed through a maximum of 27,500 psi at about 2000 MWD/AT. The percent change in 6 inch specimen length (ΔL) also reached a maximum of + 1.4 at about 2000 MWD/AT, and later decreased to about + 0.3 at 4000 MWD/AT. The compression strength tests have not indicated any significant trends to date. Although the available data are insufficient to predict behavior at higher exposures, it is expected that the material strength will not be greatly reduced due to irradiation effects.

Silicon Carbide

Modulus of rupture values have remained essentially constant at about 5000 psi over the range of exposures to date. Average compressive strengths have remained between 14,000 psi and 16,000 psi throughout the exposure period. Percent change in 6 inch specimen length increased to + 1.7 at 2000 MWD/AT and then decreased to about + 0.3 at 4000 MWD/AT.

Aluminum Silicate

Of the three ceramics, the aluminum silicate has exhibited the greatest dimensional instability under irradiation. The percent change in 6 inch specimen length has increased to more than + 3.5 at 4000 MWD/AT, with no indication of a reversal in this trend. Modulus of rupture values have remained at approximately twice the initial 3300 psi value since an exposure of 1500 MWD/AT. was reached. The compressive strength has remained at approximately 40,000 psi over the range of 1500 to 400 MWD/AT. However, unirradiated material compressive strength values are not currently available for comparison.

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Reactivity Loss Calculations

Reactivity losses associated with each of the above materials were calculated on the basis of full vertical rod channel sleeving at C Reactor.(21) Three postulated liner designs, with cross sectional dimensions as follows, were considered in the reactivity loss calculations.

Design #1. A 1/4 inch wall liner installed in the existing (straightened) 4 3/16 inch square channels. The resultant opening would be 3 11/16 inches square.

Design #2. A 1/2 inch wall liner installed in a rectangular opening which had been straightened and enlarged to 4 11/16 inches side to side and 5 3/16 inches front to rear. The resulting opening would be 3 11/16 inches by 4 3/16 inches.

Design #3. Similar to case 2 but with one inch thick end walls to provide additional sleeve strength. The channel would be enlarged to 4 11/16 inches side to side and 6 3/16 inches front to rear. The resultant opening would be 3 11/16 inches by 4 3/16 inches.

The pertinent information from these reactivity loss calculations is presented in Table II, based on an order of magnitude estimate of $50,000 per mk-year direct losses caused by the sleeves.

In addition to the economic penalties associated with ceramic sleeve installation, the effect of the various channel sizes on Ball 3X system and vertical rod strength were also evaluated.(21) The analysis showed that liners with dimensions as in postulated sleeve design #1 would result in about a 10% reduction in Ball 3X system strength and approximately a 6% reduction in local control strength of the rod, compared to about 5% and 3% respective reductions if either sleeve design #2 or #3 were used. On this basis, sleeves with cross sectional dimensions approximating those in design #2 were considered best.

G.E.L. Studies

Concurrent with the above studies, the assistance of the General Electric General Engineering Laboratory was requested, under contract ATH-IP-1-62, for performance of a number of studies on the graphite distortion problem program. Specifically, the work requested of GEL considered three approaches to the problem. First, a review of materials (both metallic and non-metallic) which could be used as channel liners was requested. Secondly, consideration would be given to long term solutions, as opposed to "temporary fix" type solutions, which could still be accomplished with essentially the existing safety rod system. The third approach would be conception of alternate systems to achieve the function of the safety rods, but which might be more compatible with the distorted graphite moderator. It was also intended that the GEL effort would be an independent evaluation of the problem to verify the sleeving approach to long term problem solution and to determine whether additional approaches existed which would warrant consideration.

In the GEL investigation, only one material, other than those already under investigation, warranted consideration as a VSR channel sleeve. This material was pyrolytic graphite.

It was also the major conclusion of the GEL study that the channel sleeving approach was the best probable "temporary fix" type of solution to vertical rod operational problems.[22] Sleeving was believed to be a temporary solution to the problem for two reasons. First, there was no information at that time which would establish whether the strength of a particular sleeve would be sufficient to resist further deformation of the rod channel over an extended time period. Secondly, it was suggested that sleeving could be considered a permanent solution only if sleeve installation were accompanied by key cutting to remove the major means of force transmittal in the stack. A conceptual design of a key cutting tool, which embodied the general features of a saber saw and could be operated in each VSR channel, was presented in the final GEL study report. Selective key cutting to remove paths of force transmission throughout the moderator was judged to require a prohibitive amount of reactor outage time because of the number of keys which would require cutting. Further consideration of this concept was curtailed.

The decision to direct subsequent development efforts toward provision of the strongest sleeve possible, and evaluate performance by in-reactor testing, was made. A correlative effort toward estimating minimum values of in-service loads which would act on the sleeves, by installation of expandable force blocks, was also envisioned. This latter course of action was developed through the design and design testing phases of the program but was suspended prior to in-reactor installation. Installation of test sleeves of the various candidate sleeving materials was considered to have a higher priority than force block installation and subsequent analytical evaluation of graphite stack forces raised considerable question as to the necessity for force block installation.

As a result of the GEL recommendation to evaluate pyrolytic graphite materials, designs were prepared and procurement of test quantities of sleeves from this material was initiated. Due to inherent material fabrication problems, the most judicious sleeve configuration of pure pyrolytic graphite utilizes an elliptical cross section. Bids received on pyrolytic graphite sleeves 18 inches long in elliptical cross section configurations quoted unit prices ranging from $2370 to $3000 for quantities of 16 sleeves. The extremely high unit cost of elliptical cross section pyrolytic graphite sleeves provided the basis for cancellation of this order. Ten sleeves of pyrolytic graphite plate were then obtained at an approximate unit cost of $800, for mockup tests to compare this material with other candidate sleeve materials.

 [redacted]

Concurrent with these efforts to obtain pyrolytic graphite sleeves, design and procurement of test quantities of silicon carbide, aluminum oxide and polycrystalline graphite sleeves was initiated. Acceptable bids (less than $150/ft. of sleeve) were received and procurement orders for test sleeves of each material were placed.

Fabrication problems were encountered during production of both ceramic sleeves. After unsuccessfully trying to develop a method of silicon carbide sleeve fabrication, the vendor requested cancellation of the order. A number of trial sleeve fabrication techniques were tried over an eight month period before this order was cancelled.

Fabrication problems were also encountered by the manufacturer of the aluminum oxide sleeves. However, these problems were resolved and the order for 48 sleeves, 6 inches long, was received nine months after the purchase order was placed.

Mockup Tests[23]

Evaluation of candidate VSR channel sleeve designs, with regard to impact and wear resistance, was accomplished by installing test sleeves in a channel mockup facility in 195-D Building and observing the extent of sleeve damage as a function of rod drops. As indicated above, the materials evaluated by mockup tests were aluminum oxide, polycrystalline graphite (1.7 g/cc density) and pyrolytic graphite plate sleeves. Two grades of polycrystalline graphite, 817 RYL and TSX, were evaluated.

The mockup tests had three objects:

1. To determine the extent of damage to various liner materials proposed for sleeving C Reactor VSR channels when subjected to the following rod operations:

    a. Full rod drop in a misaligned channel.

    b. Full rod drop into a channel partially filled with 3X balls.

2. Determine the interference between the vertical rod and the 3X balls when a simultaneous ball and rod drop occurs in a sleeved channel.

3. Demonstrate the feasibility of overboring and lining vertical rod channels in the reactors by conducting a trial sleeve installation in a mockup simulating reactor channel distortion conditions.

Two degrees of channel distortion were simulated in the mockup. First, the channel was given a lateral displacement of 1 3/4 inches in a 3 foot length of channel as measured from an original channel vertical centerline. The second degree of distortion was 2½ inches lateral displacement over the same interval.

The tests conducted with a rod tip similar to that used on currently installed VSR's at C Reactor resulted in breakage and damage to the sleeves in both the 2½ inch and 1 3/4 inch distorted channels. Modifications of the rod tip, so that it has a rounded bullet nose configuration, greatly extended the life of the liners.

The grade 817 RYL polycrystalline graphite and the aluminum oxide sleeves remained intact for 1000 rod drops, under both degrees of channel distortion, when the modified rod tip was used. The test was stopped after 1000 rod drops since (assuming a constant channel curvature) this number of rod insertions is comparable to about 25 years of reactor usage. The aluminum oxide sleeves did however, abraid the vertical rod surface.. This abrasion became progressively worse as the test continued.

The vertical rod with the modified tip wore a hole through the side of one pyrolytic graphite liner after 883 rod drops in the channel which was distorted 1 3/4 inches.

On those tests in which the rod was dropped into a channel partially filled with 3X balls, liner failure was observed on the aluminum oxide and both the 817 RYL and TSX graphite materials. When the balls were removed the broken liner pieces fell into and completely blocked the channel. The sides of the pyrolytic graphite plate sleeves remained essentially intact when the rod was dropped into the ball filled channel but almost all of the assembly pins were broken.

No significant interference between the rod and the 3X balls was noted when these control elements were dropped simultaneously.

A trial installation of graphite sleeves in an overbored VSR channel mockup verified sleeve installation tooling design and demonstrated the feasibility of on-reactor sleeve installation.

The mockup tests verified the ability of the polycrystalline graphite and aluminum oxide sleeves to withstand rod impact loads until the limits of rod flexibility are reached. From the standpoint of rod wear, these tests also showed the advantages of using graphite rather than ceramic sleeves. The aluminum oxide materials seriously abrade the rod surface under continuous usage and would undoubtedly necessitate more frequent gas seal replacement if the channels were sleeved with this material.

Sleeve Strength Tests

Development contracts and direct purchase of graphite and pyrolytic graphite products from several vendors has provided a number of alternate carbon materials which have been evaluated for the vertical rod channel sleeving application. Methods of evaluation have included simple flexural strength tests, flexure, shear and crushing load resistance tests on samples fabricated to the sleeve dimensions of concern. The mechanical strength tests of the alternate developmental sleeves provided comparative data upon which selection of the best material for in-reactor service evaluation could be based.

 [redacted]

The lattice spacing and resultant vertical rod channel size at C Reactor allows consideration of channel sleeves with uniform wall thicknesses greater than that permissible at the K Reactors. The greater wall thicknesses result from the ability to enlarge existing channels to larger dimensions. Since the channel size is similar at C, DR and H Reactors, any sleeve which is acceptable at C Reactor will also be applicable at DR and H, if it is assumed that the external sleeve loading (stack forces) are comparable at these reactors. The latter assumption appears to be a reasonable one.

C Reactor Materials

Nuclear grade polycrystalline graphite and aluminum oxide are the materials which have been tested for application at C Reactor. Because of its availability from surplus material remaining on plant after NPR moderator stack construction, grade TSX graphite was selected for the first installation in a test channel at C Reactor . Vertical rod channel sleeves of this material were fabricated in 17-3/8 inch lengths. At the time of test channel sleeve installation it was determined that channel distortion had progressed sufficiently to prevent insertion of 17 3/8 inch long sleeves in the enlarged channel. The test sleeves were therefore cut in half, permitting easy installation of the 8 11/16 inch long sleeve segments.

In addition to the grade TSX graphite sleeves, test sleeves were also fabricated from grade 817 RYL graphite. The 817 RYL is a 1.7 g/cc density nuclear grade graphite with slightly less stringent purity requirements than those required for production of TSX graphite. The strength characteristics of the TSX and 817 RYL graphites are presumed to be comparable.

The type of load applied to sleeves of each material considered for C Reactor application is shown in Table III, together with the number of samples tested. The load was applied to each specimen through graphite blocks similar in size to those which would bear on the sleeve during in-reactor service.

The aluminum oxide liners were, as expected, the strongest of the selected materials in resisting external loading. Minimum forces causing failure were greater than 20,000 pounds when crushing loads were applied to the weakest plane of the sleeve, i.e. the side to side sleeve dimension. Failure loads were on the order of 30,000 pounds when the crushing loads were applied in the strongest plane of the sleeves, i.e. at 90° to the weakest plane of the sleeve or in the sleeve front to rear dimension. Figure 1 indicates typical load-deflection characteristics of the aluminum oxide sleeves.

The polycrystalline graphite sleeve strengths varied from 3,000 pounds to 8,500 pounds depending primarily upon the type of applied load. The

17 3/8 inch 817 RYL sleeves under flexure type loading failed at about 5,000 pounds when loaded in the weakest plane of the sleeve and at approximately 8,500 pounds when loaded at 90° to the weakest plane of the sleeve. (Figure 2).

When loaded in shear, the 8 11/16inch long TSX sleeves failed at loads on the order of 4,000 pounds and 6,000 pounds when loaded respectively in the weakest and strongest planes of the sleeve. Direct crushing loads caused failure of these sleeves at about 3,000 pounds. (Figure 3).

K Reactor Materials.

The size of the K Reactor VSR channels and the more restrictive amount of graphite that can be removed during channel enlargement before sleeving, limits the number of acceptable channel liner materials even more than at other reactors. Aluminum oxide for example, because of the difficulty encountered in fabricating rectangular sleeve forms, is believed to be unacceptable for application at the K Reactors. The degree of permissible channel enlargement at the K Reactors coupled with sleeve inside cross sectional area requirements severely limits maximum wall thickness of any sleeves considered for the K Reactors. Also, considerations of long term vertical rod serviceability dictate use of a sleeve with non-uniform wall thickness. High shrinkage during firing of aluminum oxide parts in the sizes of concern make fabrication of uniform wall thickness sleeves difficult and sleeves with non-uniform wall thickness practically impossible with current technology.

For these reasons, development efforts for K Reactor VSR channel sleeves have been directed primarily toward increasing the strength and/or deflection characteristics of carbon materials. The recent development of methods of impregnating polycrystalline graphite with pyrolytic carbon provided an early indication that the strength characteristics of the base materials could be significantly increased by this method.

Two types of polycrystalline graphites have been evaluated as suitable base or substrate materials for subsequent pyrolytic carbon impregnation. First, a 1.57 g/cc density graphite, National Carbon Co. grade AGSR, was used. Secondly, a 1.37 g/cc density graphite, produced by Great Lakes Carbon Company in a limited quantity for experimental evaluation at HAPO several years ago, was used. The low density material had a reasonably open pore structure and was therefore believed to be more likely to obtain a greater degree of pyrolytic carbon impregnation. This presumption proved to be correct, and substrate material weight gains from about 9 to 14 percent were obtained on sleeve forms fabricated from the 1.37 g/cc density graphite. The increase in strength resulting from pyrolytic carbon impregnation varied from approximately 40 to 96 percent depending upon the degree of impregnation. While strength in creases on the order of 100 per cent are not insignificant, it was believed that strength increased by a factor of three or four might be obtained. Complete penetration of the low density substrate was obtained on those sleeve specimens which experienced the larger percentages of weight gain and strength increase. However, when considering that the 1.37 g/cc substrate material consists of approximately 40% void volume (compared to theoretical graphite density), it is apparent that weight gains of 10 to 15% do not approach complete coating of voids with depositions of pyrolytic carbon.

Table IV shows the type of load, sleeve material and number of specimens fabricated to K sleeve dimensions. Comparison of the strength characteristics of the untreated polycrystalline graphites with the pyrolytic carbon impregnated substrate materials is indicated by the load versus sleeve deflection curves on Figure 4. It will be noted that the strength of the 1.57 g/cc density material, with a weight gain of only 2.2 percent is comparable to the lower density graphite with a 12.3 percent weight gain. The relatively high strength of the 1.57 g/cc density graphite is attributed primarily to the higher initial density of this material. A secondary contributing factor to the observed higher strength can be attributed to a 10 to 15 mil surface coating of high strength pyrolytic carbon which was deposited on the surface of the 1.57 g/cc density graphite sleeves before the deposition process was interrupted. The pyrolytic carbon impregnated graphite sleeves (Table IV) failed in the 2500 to 3500 pound load range when crushing loads were applied in the weakest plane of the sleeve.

Composite materials formed by impregnating graphite felt, carbonized hemp, carbonized jute cloth, twine and cheesecloth with pyrolytic carbon have also been evaluated as candidate rod channel liner materials. While infiltration of graphite felt, carbonized hemp, burlap or jute, cheesecloth and twine increase the load resisting capacity of these materials significantly, the end composite materials do not compare favorably with polycrystalline graphite materials from a strength standpoint. The most significant characteristic of the composite materials appears to be their ability to deflect under load without complete sleeve collapse. The pyrolytic carbon impregnated hemp particularly, after initial material cracking, failed a thread or string at a time. Sleeve deflections on the order of 0.250 inches were attained before complete liner collapse. Values of load at failure ranged from about 450 to 1800 pounds for the composite materials (Figure 5.)

Materials which deflect large amounts prior to collapse or complete liner failure might provide acceptable rod channel liners. The ability of a liner material to deflect or even fail by localized cracking without complete liner collapse could be suitable for installation in channel locations where applied loads result from thermal expansion or where forces can be expected which are within the strength range of the material. Figures 5 and 6 show the load-deflection characteristics of the pyrolytic carbon impregnated graphite felt, hemp, jute and other composite substrate materials.

Flexural Strength Tests

In addition to the tests discussed above which were performed on fabricated sleeve forms, a number of flexural strength tests have been conducted on samples of high strength polycrystalline graphites.

[redacted]

The test specimens were fabricated from four inch square coupons in thicknesses of 3/8 and 1/2 inch. The material designation, description and number of specimens tested are shown in Table V. The data from these flexure strength tests were compared with similar data obtained from specimens fabricated from the pyrolytic carbon impregnated graphite materials.

Table VI summarizes the results of the flexural strength tests performed on the nine different graphite materials listed in Table V. Since the average flexural strength of most nuclear grade graphite is between 1970 psi and 2400 psi., it is evident that all but one of the materials listed in Table VI are stronger than the usual reactor graphites. However, there are several serious limitations on availability and fabricability of some of the stronger materials.

The H-243 material with a flexural strength of 4861 psi, while having the highest strength of the materials tested, is still developmental in nature. This material is not available in production quantities. The price would obviously not be competitive with slightly lower strength commercially available graphites.

The "PT" designated materials, particularly PT-0176 and PT-0177 containing silicon, are also developmental from the standpoint of sleeving applications. Additional development contracts would be required to formulate economical sleeve fabrication techniques if these materials were selected for further evaluation. A penalty for compensating enrichment would also have to be accepted if either the PT-0176 or PT-0177 sleeving materials were installed in-reactor. The penalty results from silicon contents on the order of 19 to 23 weight percent in these materials.

The H-205-85 material, with a flexural strength of 4635 psi (Table VI) is about 50% stronger than the strongest pyrolytic carbon impregnated, 1.37 g/cc density, polycrystalline graphite. Although the tested H-205-85 material was a molded graphite, informal contact with the manufacturer has indicated that sleeve stock can be extruded from the same grade of graphite with essentially the same strength characteristics. The price of fabricated sleeves, of the extruded material, with non-uniform wall thickness as contemplated for the K Reactors, should be available at prices of about $100 per lineal foot if orders are placed for production quantities of channel sleeves.

Current planning provides for procurement of test quantities of the H-205-85 (or similar strength material) sleeves for in-service performance evaluation at C Reactor. Acceptable performance at C Reactor should provide an adequate basis for procurement of similar material sleeves for the K Reactors. High strength graphite sleeves with acceptable nuclear purity can be obtained.

Sleeve Material Costs

Exact cost comparisons between the various candidate sleeving materials are not available at this time, particularly on a production lot basis.

 [redacted]

Many of the materials which were evaluated do not have sufficiently attractive mechanical properties to warrant requests for cost estimates on fabricated sleeve assemblies . Approximate cost comparisons can however be obtained by examining the prices paid for various quantities of fabricated sleeve parts, developmental composite material sleeves, and test coupons. Table VII indicates the material configuration , quantity obtained and unit price for some of the various candidate materials considered.

In Service Sleeve Loads [24]

The ability of candidate sleeving materials to withstand normal rod impact loading resulting from rod insertion has been demonstrated by mockup tests. However, until recently, values of estimated forces likely to be applied to the external surface of the installed sleeve column were not available.

A graphite distortion model, which explains both qualitatively and quantitatively much of the observed results of graphite distortion, has been proposed.[24] This distortion model has provided the necessary theory which allows maximum expected sleeve loads to be calculated. The following conclusions are pertinent.

1. The keys placed around the VSR channels are the major force transmitting components in the moderator stack. These keys can no longer be intact, because of the degree and type of observed distortion which has occurred to date.

2. The maximum load which can currently be applied to an installed sleeve is on the order of 1200 pounds and average loads are in the 200 pound range. Continuing moderator distortion can increase the maximum applied load to about 3000 pounds during the next ten years.

3. When VSR channels are sleeved, it would be well to sleeve as many channels as possible at one time. Then the keying action of the sleeves will not allow any particular sleeved channel to pick up a disproportionate share of the total load.

The results of sleeve strength tests previously discussed, show that 1.7 g/cc density polycrystalline graphite should provide an acceptable sleeving material. However, because of the higher material strength characteristics attainable with 1.8 g/cc density graphite and also due to the inexact nature of the calculated maximum applied loads, a 1.8 g/cc graphite is judged best for channel sleeves. The expected slightly higher cost for the stronger material, compared to a 1.7 g/cc density graphite, is offset by the greater assurance of material integrity obtained.

Because of the unavailability of a clearly defined rod channel distortion rate and the unknown degree of straightening which can be obtained by channel enlargement, the length of time that sleeving and flexible rod installation will provide an acceptable solution to rod operating problems is difficult to establish. However, if it is assumed that maximum overall lateral channel curvature progresses at a rate of 0.2 inches per year, i.e. half the approximate top center of stack vertical distortion, an estimate of the term of adequacy can be based on the rod flexibility. This type of analysis shows that the flexible vertical rods should be adequate for at least ten years in some of the more severely distorted channels. If the degree of rod flexibility becomes a limiting factor in the future, additional flexibility may be obtained by reworking the joints between adjacent segments of a troublesome rod. Before increasing the flexibility of any rod a simple exchange or transfer of the troublesome rod to a less distorted channel may be all that is required, since considerable variations in rod flexibility can result from the design tolerances on the rod joint components.

As indicated previously, flexible vertical rod and uncooled VSR channel sleeve development programs can be considered essentially complete. The economics of this proposed solution must, however, be compared with alternate courses of action which are currently being studied. Partial channel sleeving, revaluation of cooled metallic sleeve concepts, and study of alternate types of control systems are included in the work which has yet to be completed. An outline of these and other options has been documented[25] previously.

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TABLE I

EFFECT OF SLEEVING 20 HCR
CHANNELS AT A K REACTOR

Material	Sleeve Wall Thickness (Inches)	Approximate Production Loss $/Year	Effect on HCR Strength	Max. Estimated Temp. °C
Inconel	0.12	$1,130,000	Large Loss	805
	0.06	570,000	Loss	755
	0.03	290,000	Small Loss	730
Stainless Steel	0.12	780,000	Large Loss	790
	0.06	400,000	Loss	745
	0.03	200,000	Small Loss	725
Zircaloy-2	0.12	22,000	Small Effect	790
	0.06	11,000	Small Effect	745
	0.03	6,000	Small Effect	725
NPR Process Tube	0.25	41,000	Small Effect	830
Zirconium Silicate	0.15	17,000	Small Loss	725
Aluminum Oxide	0.15	37,000	Small Loss	740

TABLE II

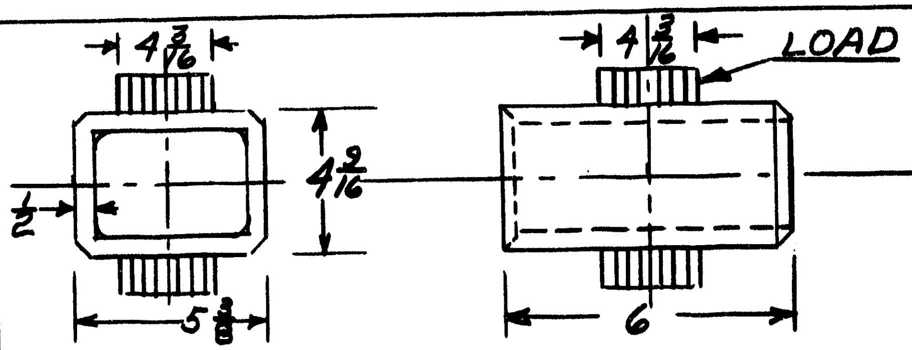
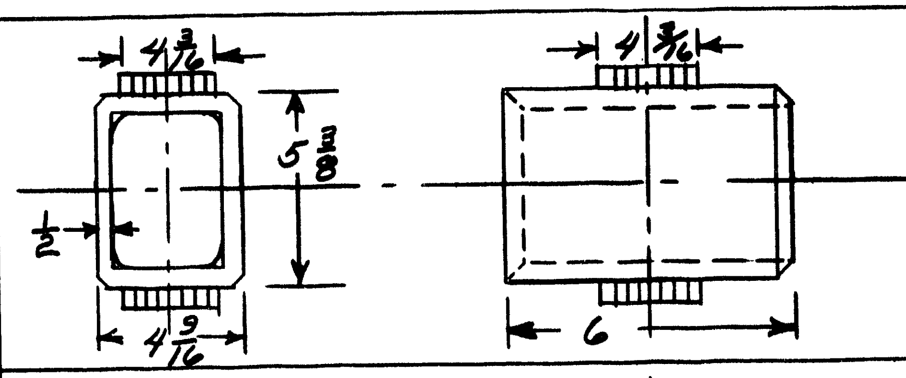
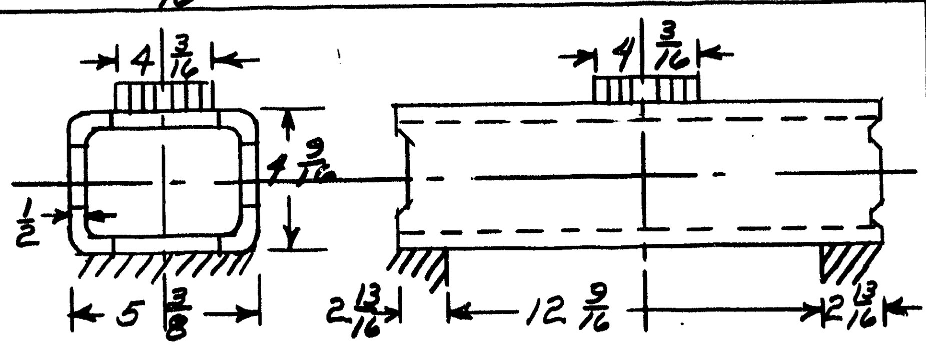
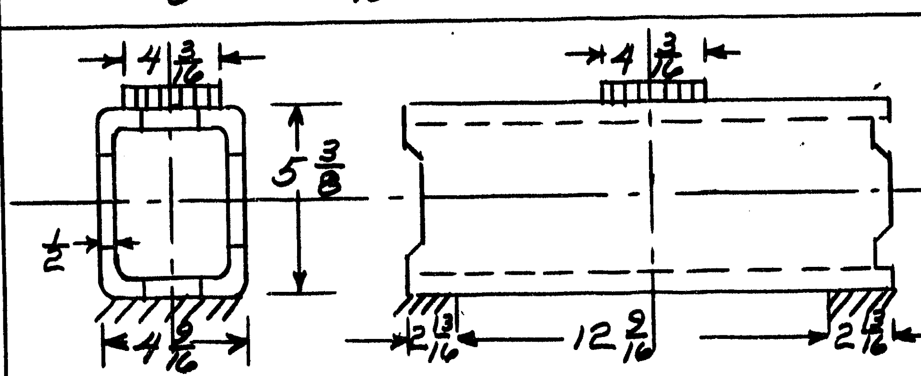
CALCULATED REACTIVITY LOSSES FROM
SLEEVING 45 VSR CHANNELS AT
C REACTOR

Sleeve Material	Cost-Dollars/Year		
	Design #1	Design #2	Design #3
Graphite	0	0	0
Aluminum Silicate	$125,000	$300,000	$450,000
Aluminum Oxide (85%)	175,000	400,000	600,000
Silicon Carbide			
6.5% Nitrogen	400,000	875,000	1,350,000
1.4% Nitrogen	175,000	400,000	600,000

[redacted]

[redacted]

TABLE III
SLEEVE LOADING - C REA CTOR MATERIALS

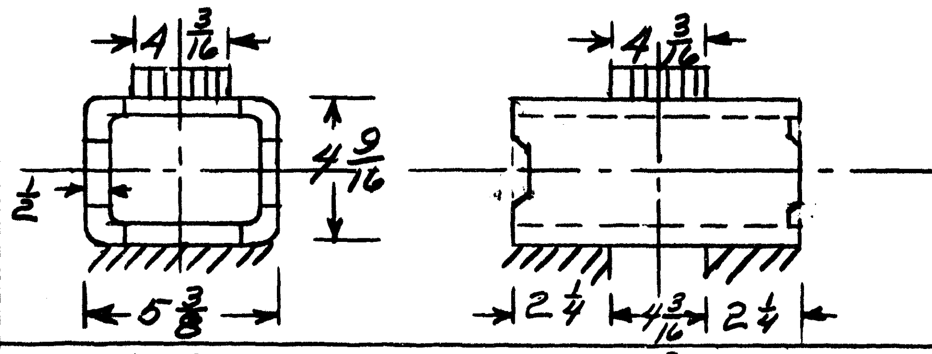
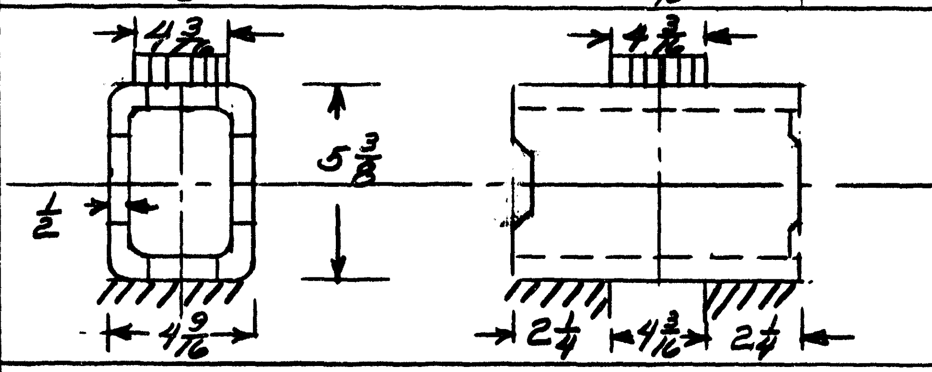
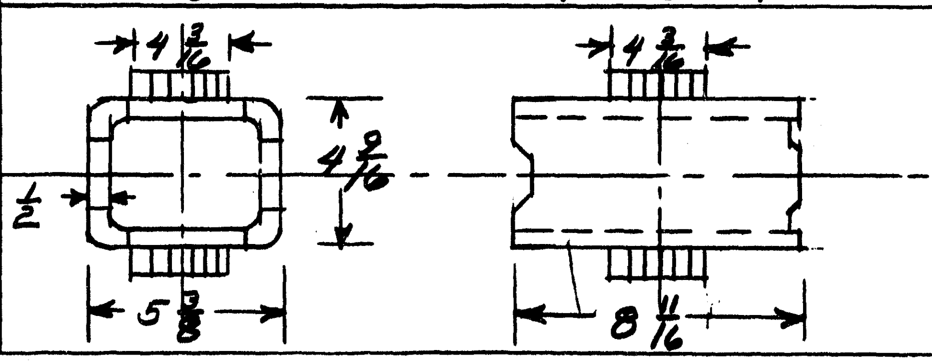
<table>
<tr><th>Material</th><th>Test Results Shown on Figure</th><th>Sleeve Dimensions and Type of Load</th><th>Number of Sleeves Tested</th></tr>
<tr><td>Al_2O_3</td><td>1-curve A</td><td></td><td>3</td></tr>
<tr><td>Al_2O_3</td><td>1-curve B</td><td></td><td>3</td></tr>
<tr><td>817 RYL Graphite
TSX Graphite</td><td>2-curve A</td><td></td><td>4
1</td></tr>
<tr><td>817 RYL Graphite
TSX Graphite</td><td>3-curve B</td><td></td><td>4
1</td></tr>
</table>

End View Side View

DECLASSIFIED

[redacted]

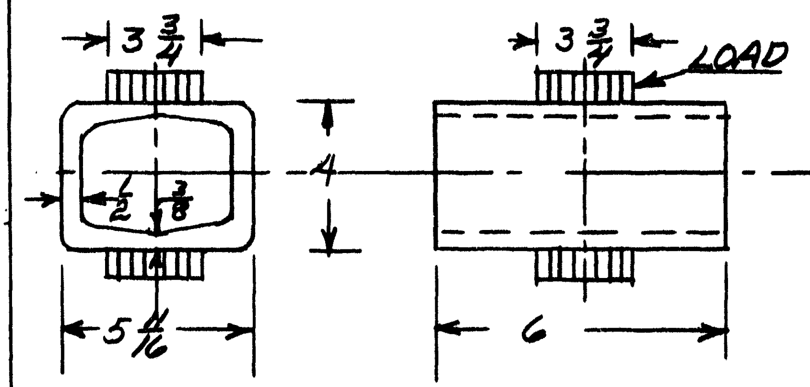
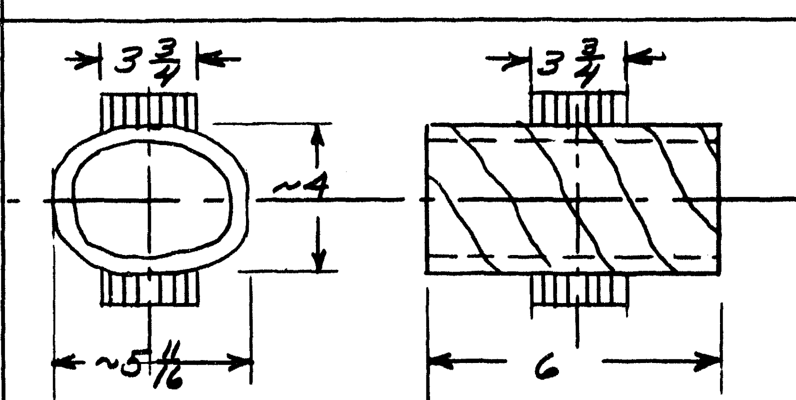
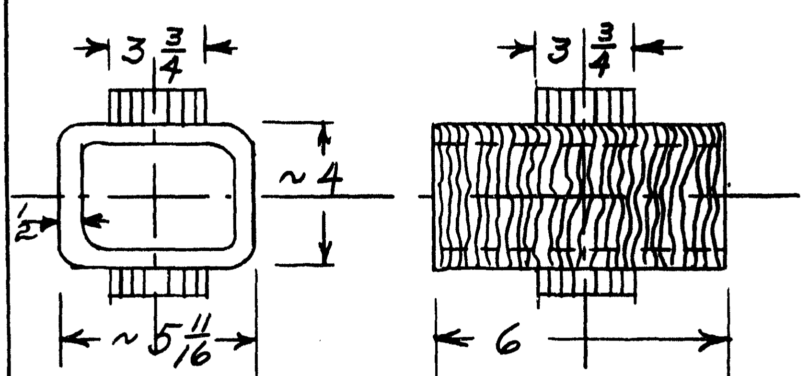
TABLE III (Cont'd.)

Material	Test Results Shown on Figure	Sleeve Dimensions and Type of Load	Number of Sleeves Tested
TSX	3-curve A		3
TSX	3-curve A		3
TSX	3-curve B		4

End View Side View

 [redacted]

TABLE IV
SLEEVE LOADING - K REACTOR MATERIALS

Material	Test Results Shown on Figure	Sleeve Dimensions and Type of Load	Number of Sleeves Tested
Untreated Graphite			
AGSR (1.57 g/cc)	4-curve A		3
817 RYL (1.7 g/cc)	4-curve B		5
1.37 g/cc density	4-curve C		5
PyC - Graphite			
AGSR 2.2% Wt. Gain	4-curve D		2
1.37 g/cc 9.8% Wt. Gain	4-curve E		2
1.37 g/cc 12.3% Wt. Gain	4-curve F		2
1.37 g/cc 13.9% Wt. Gain	4-curve G		2
PyC Graphite Felt	5-curve A 5-curve B		2 2
PyC - Hemp	5-curve C		1
PyC - Jute	5-curve D		1
PyC - Twine	6-curve A		1
PyC - Cheesecloth	6-curve B		1
PyC - Burlap	6-curve C		1

Note: Approximate cross section dimensions resulted from substrate deformation during PyC impregnation on all except PyC Graphite and untreated specimens.

TABLE V

TEST SPECIMENS FOR
FLEXURE STRENGTH COMPARISON

Material Designation	Description	Number and Size of Specimens
PT-0175	Mascerated Graphite Cloth Filler with two carbonaceous impregnation treatments	6-3/8 x ½ x 4" 6-½ x ½ x 4"
PT-0176	Laminated Graphite Cloth Filler with Silicon treatment	6-3/8 x ½ x 4" 6-½ x ½ x 4"
PT-0177	Mascerated Graphite Cloth filler with Silicon treatment	6-3/8 x ½ x 4" 6-½ x ½ x 4"
H-205-85	Bulk molded, 1.82 g/cc density, high strength polycrystalline graphite	6-3/8 x ½ x 4 6-½ x ½ x 4"
H-243	Bulk molded, 1.86 g/cc density, high strength polycrystalline graphite	6-3/8 x ½ x 4" 6-½ x ½ x 4"
AGSR (#1 & #5)	PyC Impregnated, 1.57 g/cc density polycrystalline graphite - 2.2% wt. gain	6-3/8 x ½ x 4" 6-½ x ½ x 4"
Experimental (Low Density) Graphite #6,#7	PyC Impregnated, 1.37 g/cc density, extruded polycrystalline graphite - 9.8% wt. gain	6-3/8 x ½ x 4" 6-½ x ½ x 4"
Experimental (Low Density) Graphite #9,#10	PyC Impregnated, 1.37 g/cc density, extruded polycrystalline graphite - 13.9% wt. gain	6-3/8 x ½ x 4" 6-½ x ½ x 4"
Experimental (Low Density) Graphite #11,#12	PyC Impregnated, 1.37 g/cc density, extruded polycrystalline graphite - 12.3% wt. gain	6-3/8 x ½ x 4" 6-½ x ½ x 4"

[redacted]

TABLE VI

GRAPHITE MATERIALS
FLEXURAL STRENGTH TEST - SUMMARY

Graphite Material	Average Modulus of Rupture (PSI)
PT-0175	3392
PT-0176	3835
PT-0177	2940
H-205-85 - 1.82 g/cc graphite	4635
H-243 - 1.86 g/cc graphite	4861
PyC Impregnated, 1.57 g/cc graphite, 2.2% wt. gain	2828
PyC Impregnated, 1.37 g/cc graphite, 9.8% wt. gain	2376
PyC Impregnated, 1.37 g/cc graphite, 12.3% wt. gain	2951
PyC Impregnated, 1.37 g/cc graphite, 13.9% wt. gain	3086

[redacted]

TABLE VII

APPROXIMATE SLEEVE MATERIAL COST COMPARISON

Material	Material Configuration	Quantity	Unit Price
Pyrolytic Graphite Plate	Elliptical cross section sleeve, 18" long; uniform 1/4" wall; dimensions applicable at C Reactor	16	$ 2,370.00
Pyrolytic Graphite Plate	Rectangular cross section sleeve, 12" long; uniform 1/4" wall; dimensions applicable at C Reactor	10	750.00
Aluminum Oxide	Rectangular cross section sleeve, 6" long; uniform 1/2" wall; dimensions applicable at C Reactor	70	71.75
PyC Impregnated Polycrystalline Graphite	Rectangular cross section sleeve; 6" long; non-uniform wall; dimensions applicable at K Reactor	11	935.00
PyC Impregnated, Hemp, Jute, Twine, etc.	Developmental Type rectangular cross section sleeve; 6" long; non-uniform wall; dimensions applicable at K Reactor	8	1,915.00
PT-0175	4" x 4" x 3/8" thick coupon 4" x 4" x 1/2" thick coupon	1 1	300.00 300.00
PT-0176	4" x 4" x 3/8" thick coupon 4" x 4" x 1/2" thick coupon	1 1	360.00 380.00
PT-0177	4" x 4" x 3/8" thick coupon 4" x 4" x 1/2" thick coupon	1 1	340.00 340.00
817 RYL Poly-crystalline Graphite	Rectangular cross section sleeve; 18" long; uniform ½" wall' dimensions applicable at C Reactor	16	103.39

[redacted]

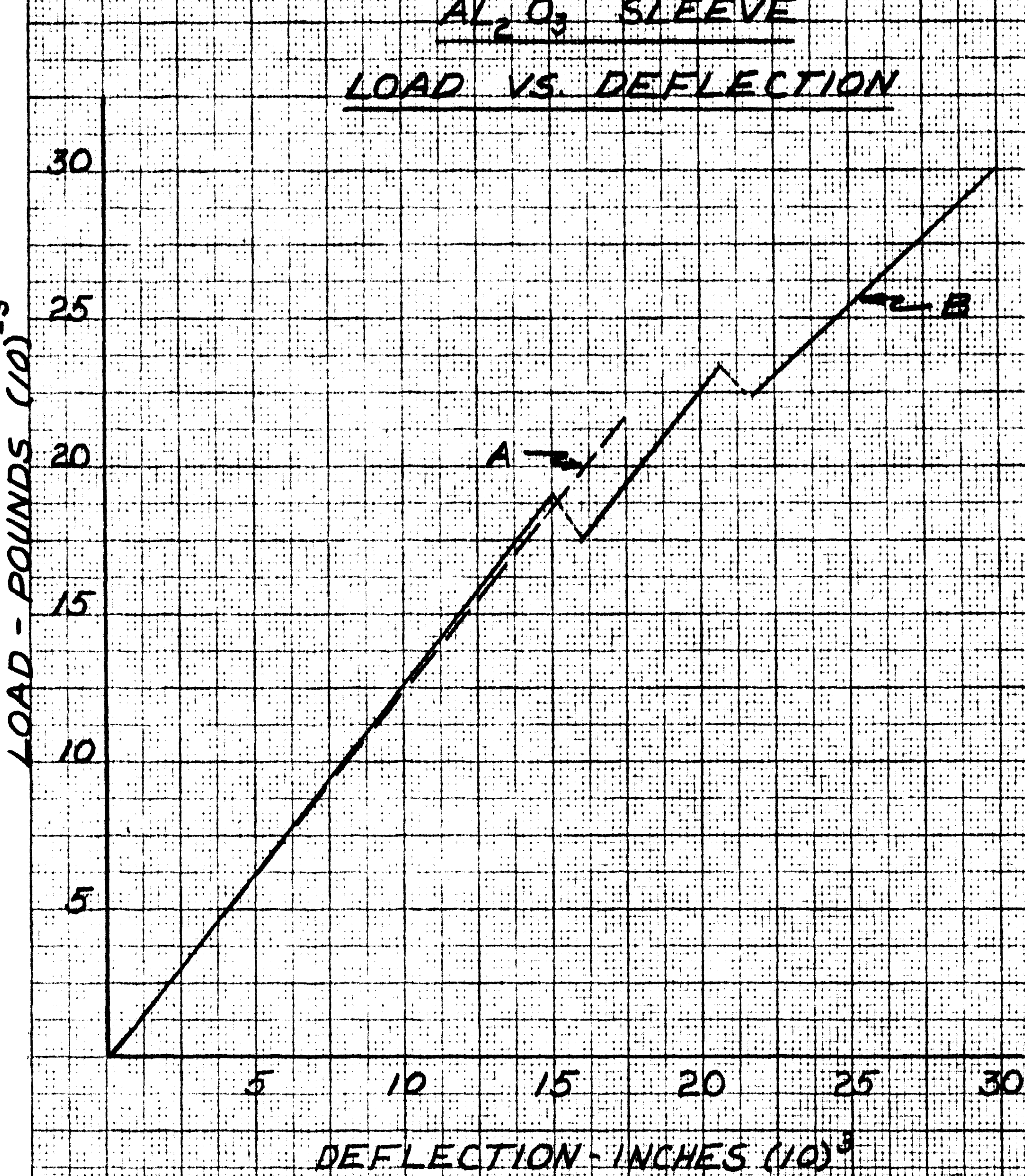

AL2 O3 SLEEVE
LOAD VS. DEFLECTION
LOAD - POUNDS (10)-3
5
10
15
20
25
30
DEFLECTION - INCHES (10)3
A
B

FIGURE 1.

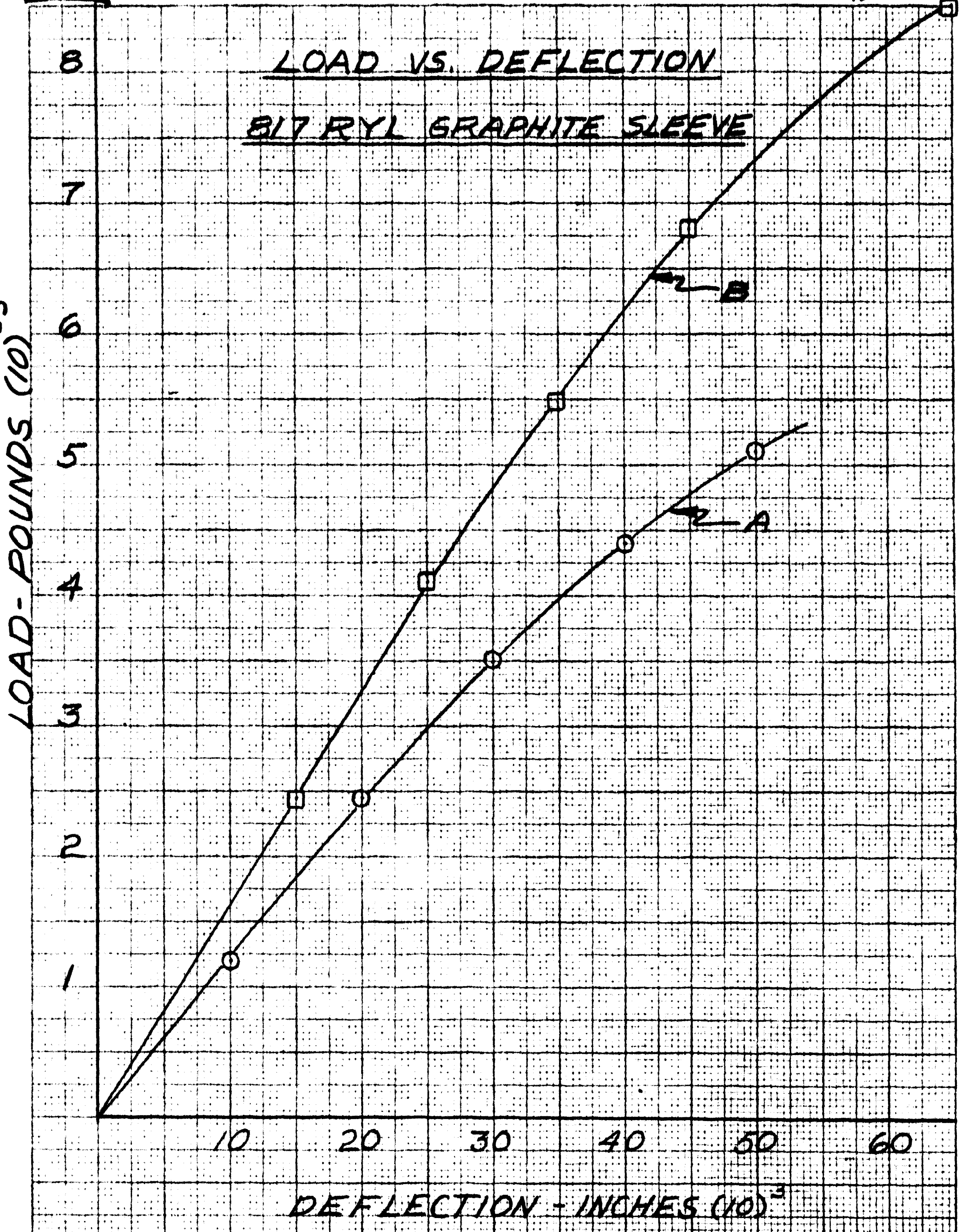

LOAD VS. DEFLECTION
817 RYL GRAPHITE SLEEVE
LOAD - POUNDS (10)^-3
DEFLECTION - INCHES (10)^-3
1
2
3
4
5
6
7
8
10
20
30
40
50
60
A
B

FIGURE 2

LOAD VS. DEFLECTION

TSX - GRAPHITE SLEEVE

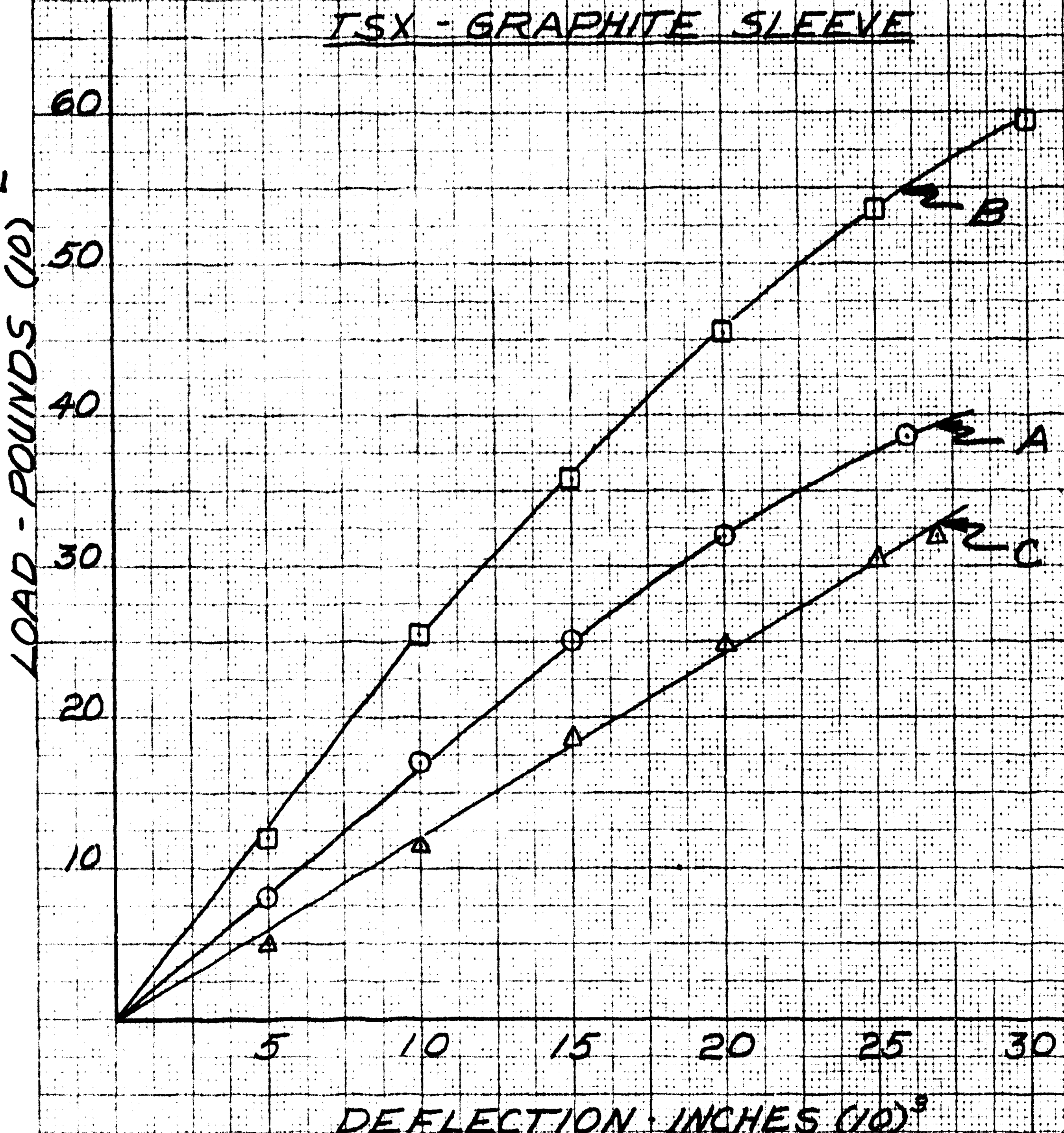


FIGURE 3.

[redacted]

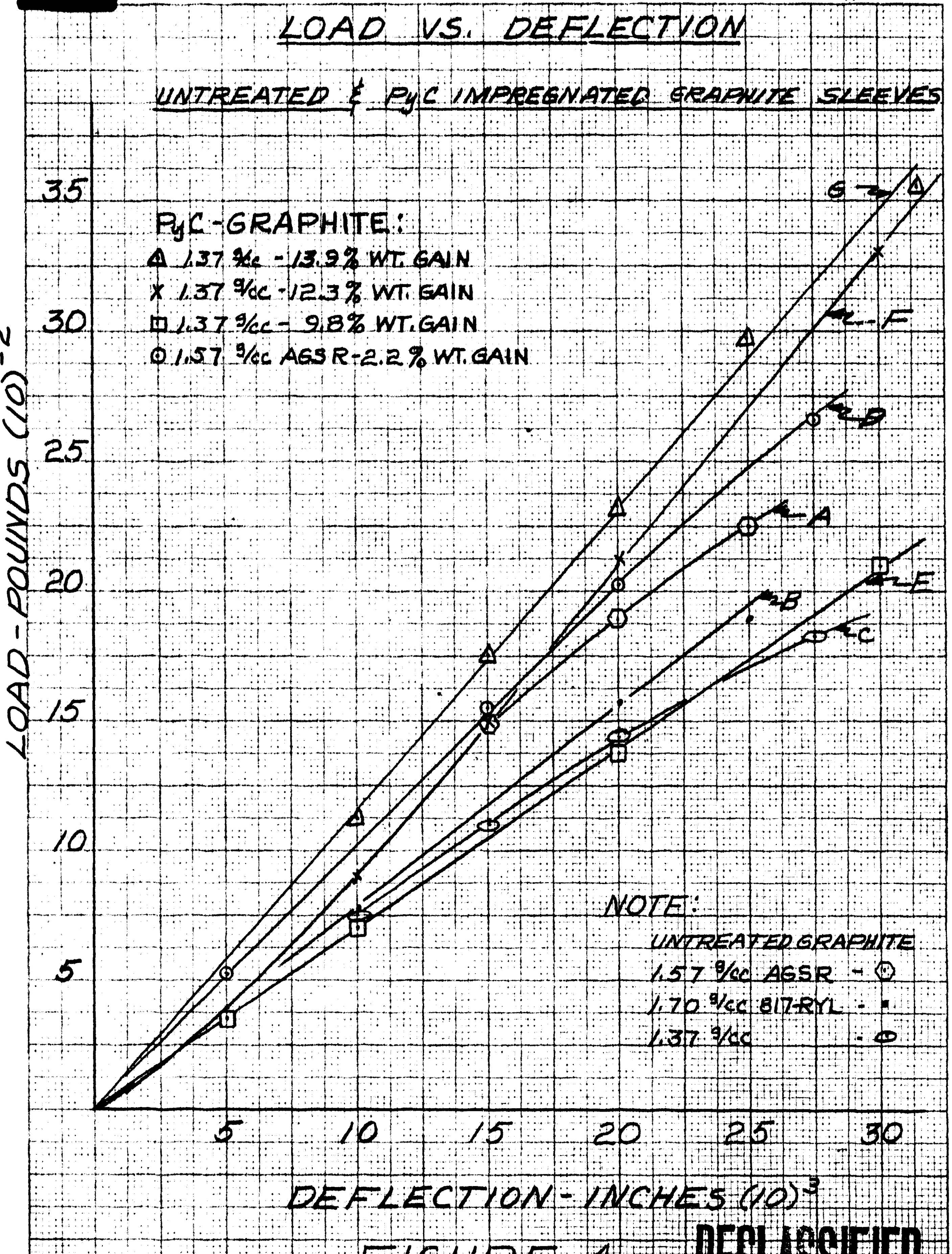

LOAD VS. DEFLECTION
UNTREATED & PyC IMPREGNATED GRAPHITE SLEEVES
PyC-GRAPHITE:
△ 1.37 g/cc - 13.9% WT. GAIN
× 1.37 g/cc - 12.3% WT. GAIN
□ 1.37 g/cc - 9.8% WT. GAIN
⊙ 1.57 g/cc AGSR - 2.2% WT. GAIN
NOTE:
UNTREATED GRAPHITE
1.57 g/cc AGSR
1.70 g/cc 817-RYL
1.37 g/cc
G
F
D
A
B
E
C
5
10
15
20
25
30
35
LOAD-POUNDS $(10)^{-2}$
DEFLECTION-INCHES $(10)^{3}$

FIGURE 4.

LOAD VS. DEFLECTION

PyC-IMPREGNATED HEMP, JUTE & FELT

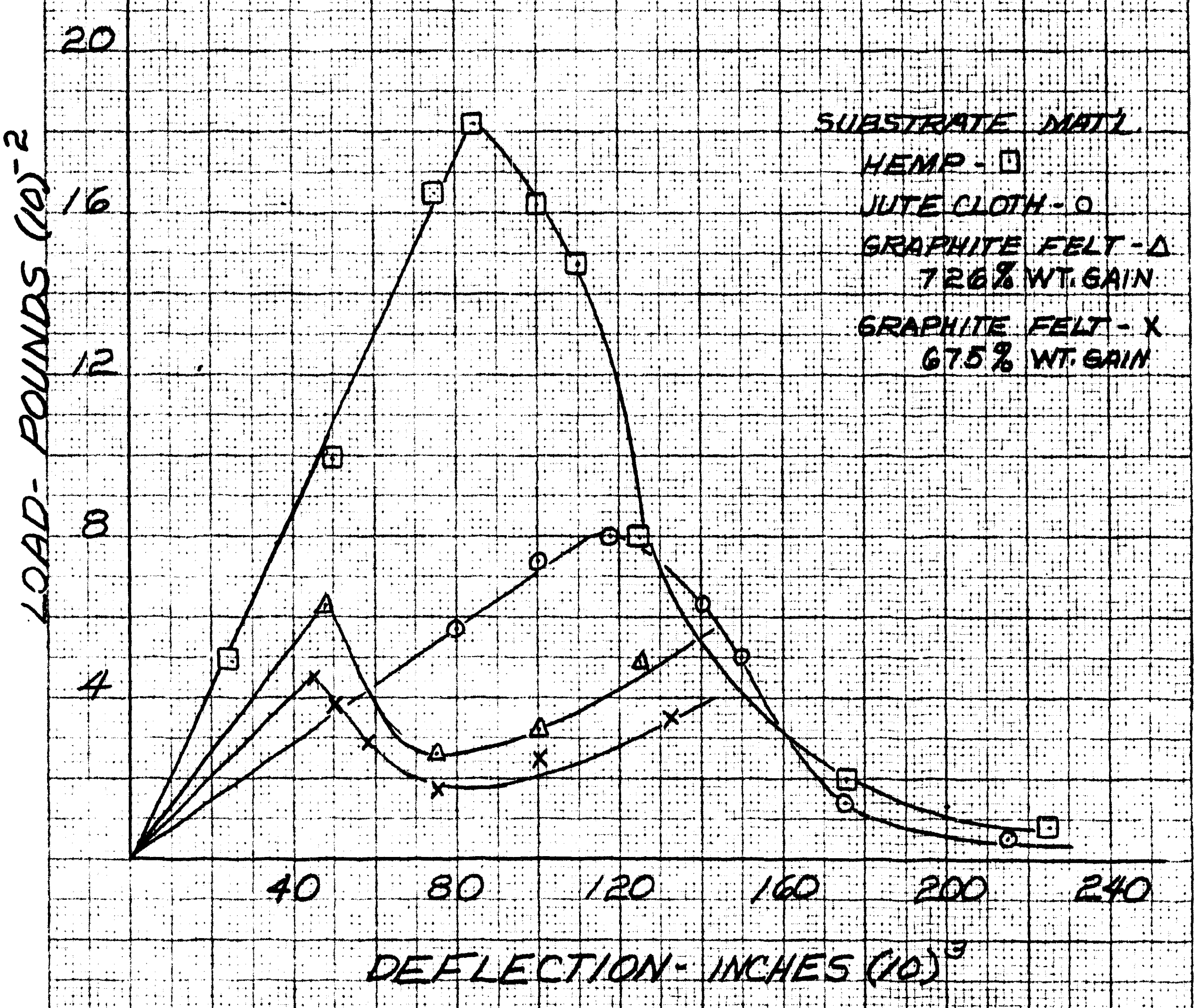


FIGURE 5.

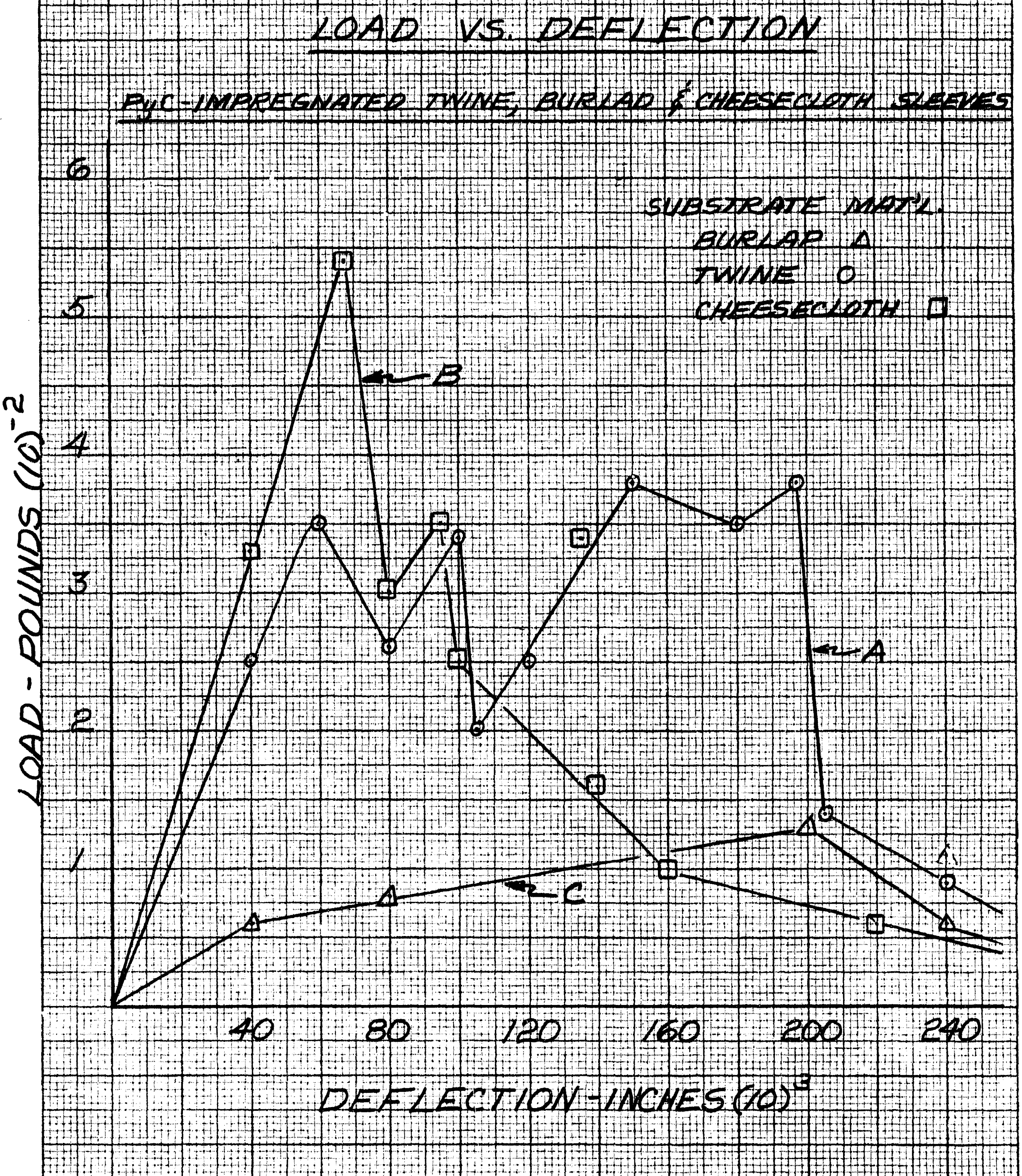

LOAD VS. DEFLECTION
PyC-IMPREGNATED TWINE, BURLAP & CHEESECLOTH SLEEVES
SUBSTRATE MAT'L.
BURLAP △
TWINE ○
CHEESECLOTH □
B
A
C
1
2
3
4
5
6
40
80
120
160
200
240
LOAD - POUNDS (10)^-2
DEFLECTION - INCHES (10)^3

FIGURE 6.