GEAP-3191

MASTER

METAL-WATER REACTIONS: III

FUEL ELEMENT STRESSES DURING A NUCLEAR ACCIDENT

By

K. M. Horst

G. E. Class I

July 24, 1959

GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

SAN JOSE, CALIFORNIA

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Prepared for the U. S. Atomic Energy Commission
under
Contract No. AT(04-3)-189
Project Agreement No. 1
May 1, 1958

July 24, 1959

ATOMIC POWER EQUIPMENT DEPARTMENT
GENERAL ELECTRIC COMPANY
San Jose, California

Printed in the U.S.A.
Price: $0.75 Available from the Office of Technical Services,
Department of Commerce, Washington 25, D. C.

UNCLASSIFIED

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TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

I. ABSTRACT

The thermal and pressure stresses in metallic plate and rod type nuclear fuel elements during a nuclear accident severe enough to produce a metal-water reaction have been reviewed. The initial reactor periods required to cause these severe accidents are in the range of 4 to 12 milliseconds. Transient temperature and power data during the nuclear accidents are identical to the most severe accident for each type of fuel that is presented in the preceding report of this series.

The result of the stress review presented in this report indicates that the metallic plate type fuel will swell from internal fission gas pressure, the rate of gas generation, and the magnitude and exposure to elevated central temperatures. Pulsed irradiation type experiments on sample fuel elements are desirable to determine the extent and consequences of this swelling including determination of the mechanism of fuel element failure and if dispersion occurs, measurement of the size of the particles.

The stress review for the clad oxide rod type fuel, indicates that the oxide core will expand more than the Zircaloy cladding during severe reactor accidents producing cladding strains in the range of one to two percent. Experimentally, 7 to 10 percent strain is required to rupture irradiated Zircaloy tubing indicating that the cladding should not fail during the reactor transients considered. However, localized areas of the cladding may experience damage during fabrication, loss of ductility or strain concentra-

tions sufficient to crack the cladding. There is no evidence to indicate that the cladding will be pulverized into small enough pieces to accelerate a metal-water reaction nor to modify the analytical models used to evaluate the core conditions during the accidents considered. Pulsed irradiation type experiments on fuel specimens simulating the conditions reviewed would be desirable to evaluate the transient material and mechanical properties of the fuel elements.

II. INTRODUCTION

The overall safety of a properly operated nuclear reactor is dependent, in part, on the consequences of improbable, but conceivable, severe accidents. It is desirable that the fission products be contained within the fuel cladding even during such an accident. In the case of water moderated and cooled reactors, these accidents may involve chemical reactions of water with any of the common reactor structural materials such as aluminum, zirconium, uranium, and stainless steel. Laboratory studies have indicated that rapid energetic metal-water reactions cannot be expected to occur unless the metal is molten and finely dispersed (1)[@*]. There are two basic mechanisms that can lead to these conditions in a water cooled and moderated reactor: a severe nuclear accident and a loss of coolant. This report is concerned with the integrity of the fuel cladding during severe nuclear startup accidents.

The purpose of this study is to show which forces, or resultant stresses, in the fuel and cladding are sufficient to cause failure during a severe reactor transient. The sources of stress considered are:

1. Internal pressure from fission product gases.

2. Relative expansion of fuel and cladding.

3. Thermal gradients within the materials.

This approach is considered for two types of fuel elements:

1. Plates - metallic fuel and cladding.

2. Rods - oxide fuel and metallic cladding.

The results of this analysis should help in forming a concept of how the metal may be dispersed and if the dispersion is likely to take place before or after melting. The fuel temperatures during the severe reactor transients were obtained from the preceding report in this study, "An Evaluation of Nuclear Excursions in Light Water Reactors" by J. I. Owens, GEAP-3178 (Reference 3).

@Numbers in parentheses designate Bibliography.

*While ignition of solid metal has been observed in oxygen and its mixtures with inert gases (2), including steam, the rapid self-accelerating reaction of metal with water vapor at temperatures below the melting point has not been reported.

III. ANALYSIS OF METALLIC FUEL PLATES

A typical cross-section of a plate type fuel element is shown on Figure 1. The cladding materials considered are aluminum and zircaloy, and the fuel is made of uranium metal alloyed in a matrix of metal similar to the cladding. The fuel is bonded with the cladding to form an integral unit.

Fission product gas pressure and thermal gradients across the fuel plate are the two primary sources of stress, during a rapid increase in reactor power, that could influence the failure of the element.

A. PRESSURE STRESSES

Fission Product Gas Generation

Measured at standard conditions, approximately 4.3 cubic centimeters of gas are produced when one percent of the atoms are fissioned in a cubic centimeter of U-235 metal (4). For comparison, the experimentally determined quantity of gas generated in several types of fuel is shown in Table I for typical reactor exposures.

TABLE I

VOLUME OF FISSION GAS GENERATED FOR TYPICAL REACTOR EXPOSURE

Fuel type	U-Zr Alloy	U-Al Alloy	UO_2
Initial Enrichment,%	93	93	1.6
Weight Fraction of U in fuel,%	4	17	88
Percent of U-235 burned	50	12	53
Cc of gas (STP) per cc of fuel	3.0	2.4	2.7

At 10,000 MWD/ton exposure the UO_2 rod type fuel is about one atom percent fissioned. While the highly enriched flat plate type fuel has a larger percentage of the uranium burned during the lifetime of the fuel element, the uranium is diluted in a metallic matrix of structural material so that the number of atoms fissioned per unit volume of fuel is similar to the other types. The quantity of gas produced ranges from 2 to 3 cc. (at STP) of gaseous fission products per cc. of fuel*. The fission gases thus produced are believed to escape from within the fuel lattice by diffusion. Rapid diffusion

*For the longer fuel exposures which may be achieved in the future, the constants developed here will require modification.

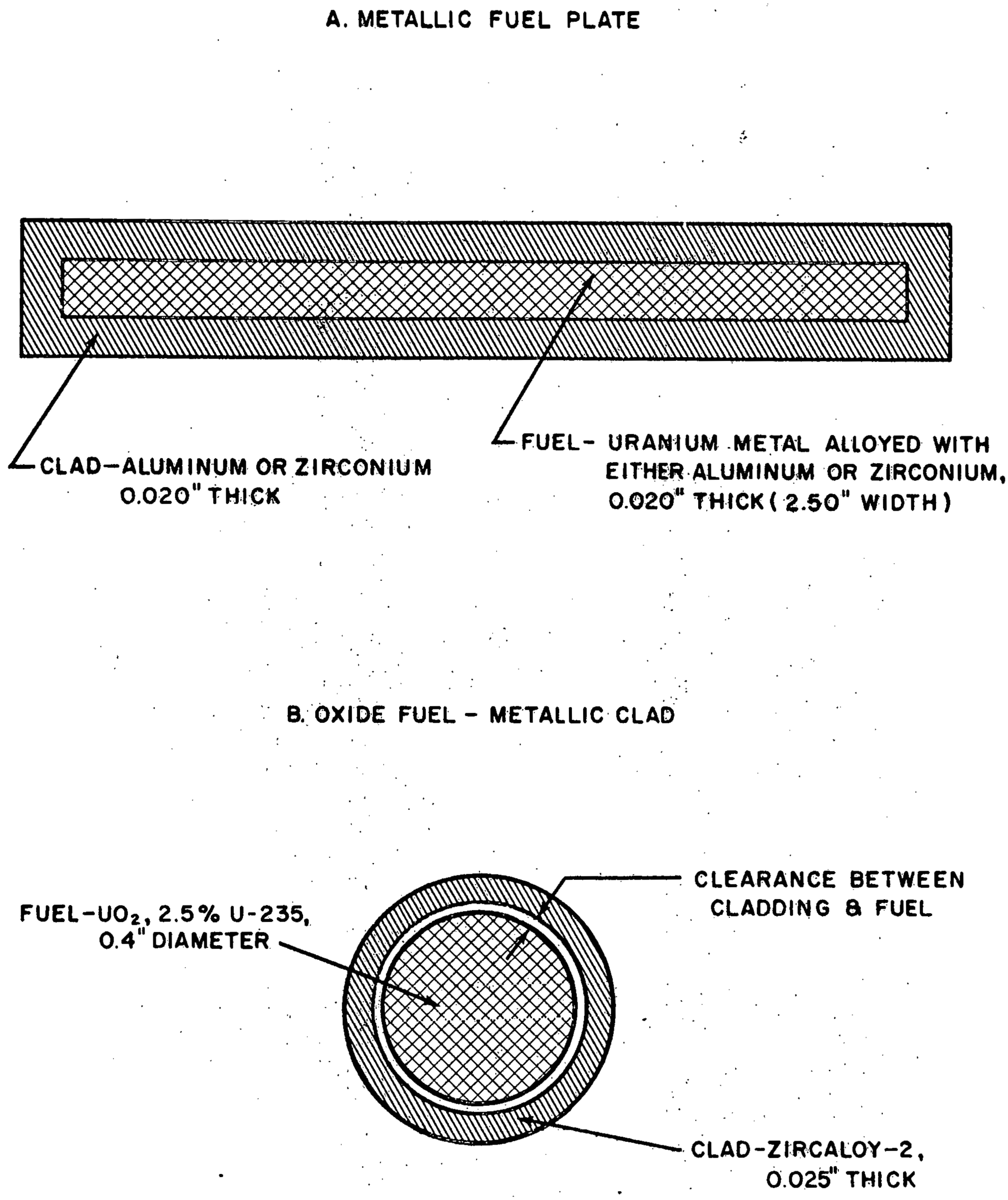

FIGURE 1: CROSS SECTIONS OF TYPICAL METALLIC PLATE & OXIDE ROD FUEL ELEMENTS (NOT TO SCALE).

rates of a gas in metal are limited to those materials which are soluble in the matrix metals, which is not the case with the fission product rare gases. The ease with which a gas diffuses through a metal lattice is, also, dependent upon the sizes of the gas atoms relative to the lattice dimensions. Reynolds (5) concluded that it is not surprising to find the diffusion of the fission product gases from metallic fuel to be small since the rare gases are so inert chemically, and the atomic diameter of xenon (about 4Å) is significantly greater than the lattice spacing of most metals (approximately 2 to 3Å). Since only a small fraction of the fission product gas is evolved from the fuel during irradiation at temperatures below about two-thirds the absolute melting point ("Tammann Temperature"), it is assumed in the present study that all fission gas is retained within the fuel material, prior to the initiation of the treatment.

Gas atoms trapped in the metal lattice would be held by atomic forces while atoms locating in voids and grain boundaries would form gas pockets. It is assumed in this report that all gas is in spherical bubbles, and that the surface tension and elastic strength of the material are sufficient to contain the resulting gas pressure.

Gas Pressure at Steady-State Operation*

Churchman and Barnes investigated the swelling of irradiation uranium during subsequent out-of-pile heating (6). The specimens in this study were of natural uranium metal irradiated in the NRX to a burnup of 0.4 percent uranium at temperatures below 300°C. The quantity of gas in these samples, released upon melting, was found to be 2 to 3 cc. of rare gas (at standard conditions) per cc. of metal. The specimens were heated to 1000°C and the amount of swelling produced was observed at various temperatures. The results of this work are shown in Figure 2. Negligible swelling of the metal was observed below 500°C (within the observational accuracy of ± 0.5 percent). This means that if the potential gaseous fission product atoms did generate gas, it would be contained in a very small volume (5×10^{-3} cc. of gas volume per cc. of fuel, or less), during normal reactor operation.

Below 50 atmospheres the pressure-volume relationship would be approximated by the perfect gas laws: however, Enderby (7) reports that the pressure-volume relation is approximated reasonably well over a wide range of pressure above 500 atmospheres by taking p-v as a linear function of p. Utilizing the quantity of gas generated per volume of fuel (Table I) and the pressure-volume relation for xenon at 200°C as reported in Reference (7), the equivalent gas pressure was obtained (see Figure 3). While krypton and xenon are the primary gases formed during the fission process, the atom ratio of xenon to krypton is approximately 6; therefore, considering the gas as all xenon introduces no significant error in the analysis.

Having arrived at a pressure - volume relation for the fission product gas, the next step is to determine the size and number of bubbles in the fuel. Churchman and Barnes (6) investigated a sample of metallic uranium that was heated for a number of hours at 800°C. The bubble density was measured to be approximately

* Steady-state operation here refers to operation at constant power and constant temperature. It is recognized that the quantity of gas is continually increasing with operating time; therefore, results will be specified for a given amount of fuel burned.

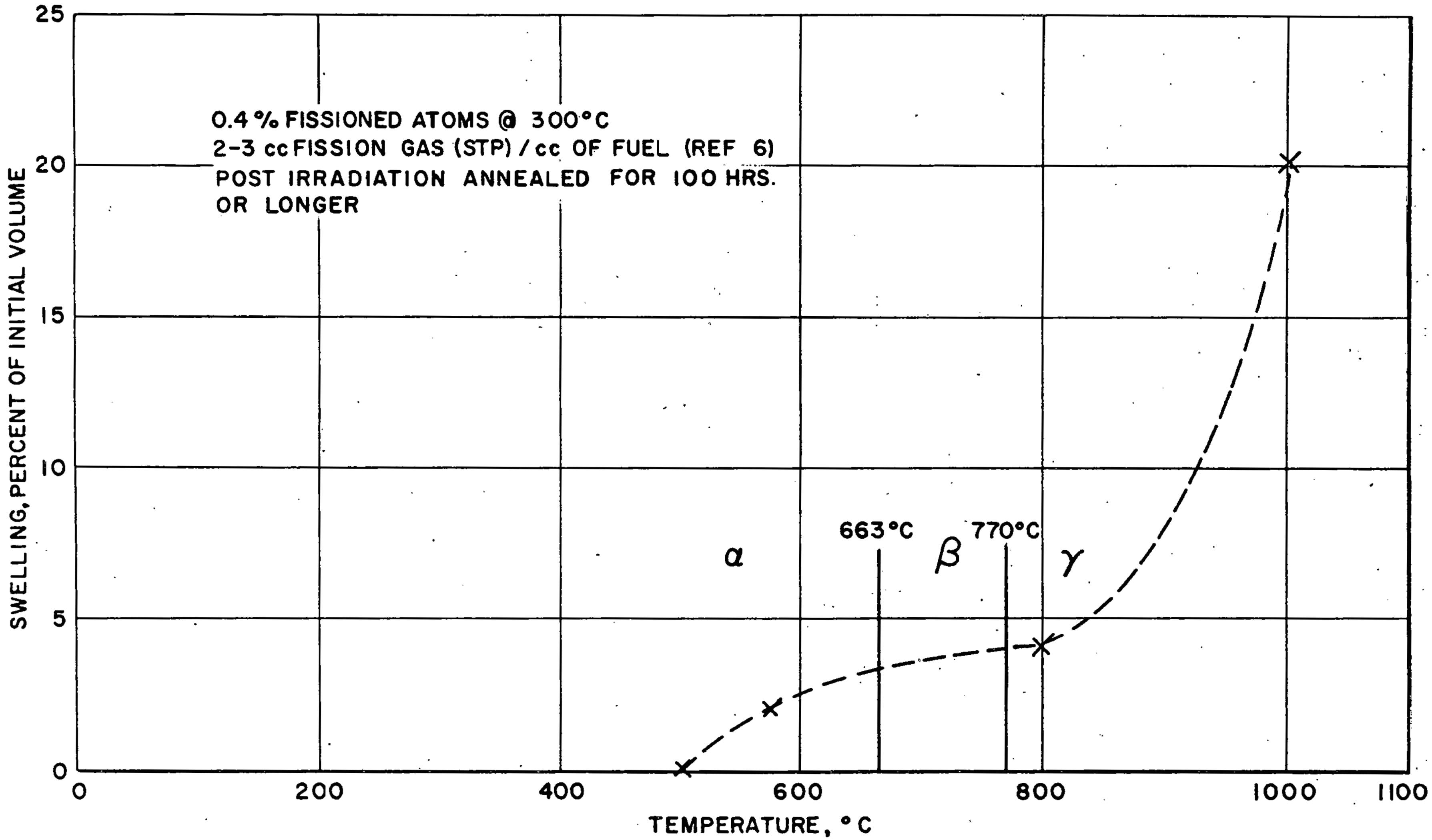

FIGURE 2: SWELLING OF NATURAL URANIUM DURING POST IRRADIATION HEATING

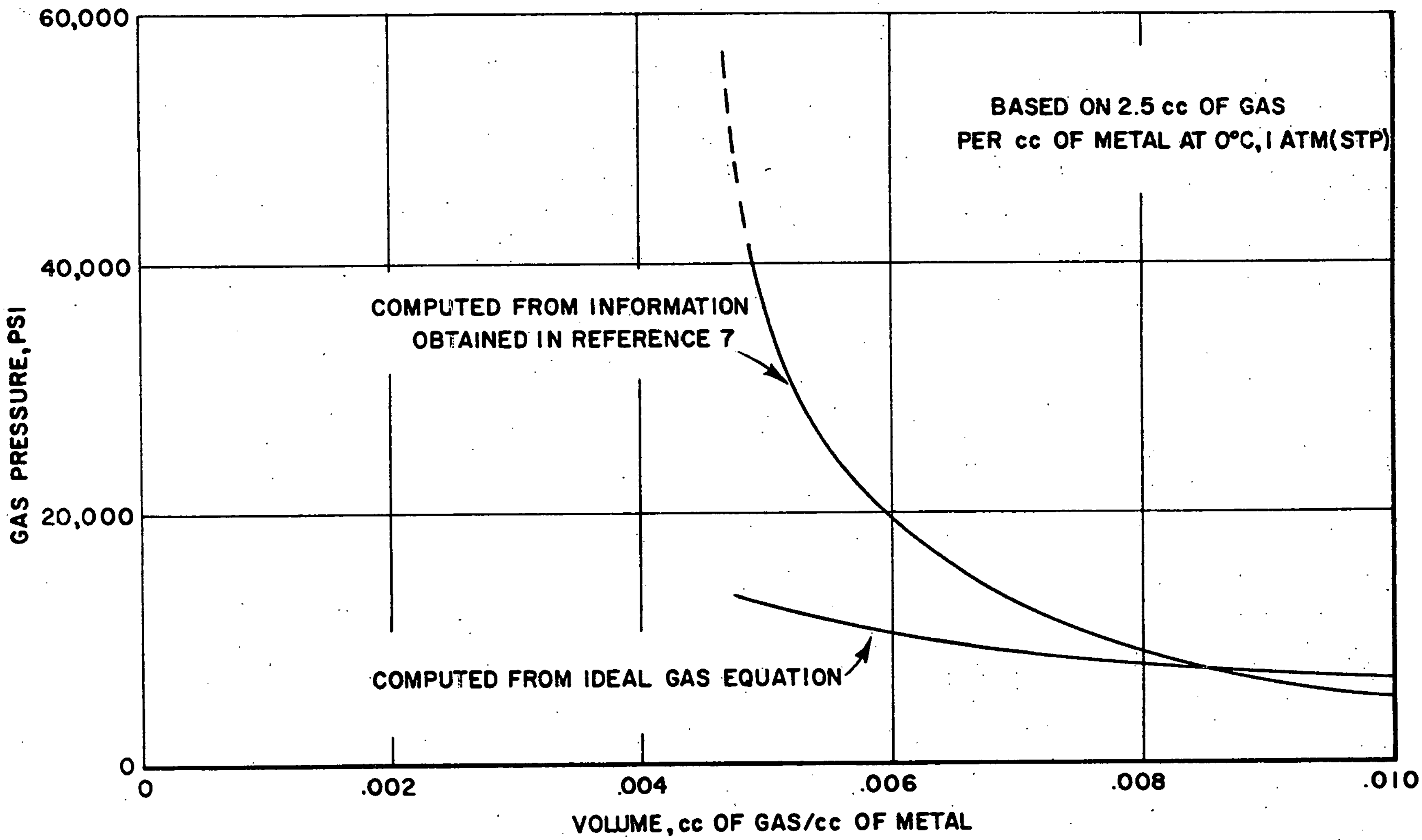

FIGURE 3: PRESSURE - VOLUME RELATION FOR XENON AT 200°C

10^{13} bubbles per cc. of fuel with a mean radius of 1000 Ångstroms (10^{-5} cm). This accounted for over sixty percent of the observed swelling. A smaller number of bubbles with larger diameters made up the remaining volume. The corresponding volumes for a range of bubble sizes and density are conveniently shown in Figure 4. Willis (8) postulates that the equilibrium bubble radius is 50 Ångstroms. Assuming 10^{13} bubbles per cc. of metal, a bubble radius of 50 Ångstroms would produce pressures in excess of 100,000 psi within the cavity. As will be shown in the following paragraphs, this is in excess of the material strength and swelling should result; however, no swelling was observed. If the gas pockets were in fact as small as Willis has postulated, a bubble density considerably greater than the experimentally observed 10^{13} would be required.

From these parameters, and the physical properties of the metal, it is now possible to determine the stress which the material surrounding the bubble can withstand without gross distortion; and this stress value in turn, is readily convertible to a figure for the internal pressure within the gas pocket.

The validity of this concept can then be checked by comparison of the pressure found in this way with that previously computed from the equation of state of the gas.

The metal surrounding each bubble is considered to act as a thick spherical pressure vessel; therefore, the maximum circumferential stress, σ_θ, in the fuel is related to the internal gas pressure P by the following equation (Reference 9) ---

$$\sigma_\theta = (P/2) \left[(r_o/r_i)^3 + 2 \right] \Big/ \left[(r_o/r_i)^3 - 1 \right]$$

where r_i and r_o are the internal and external radii respectively. The maximum radial stress, σ_r, is quantitatively equal to the internal pressure. For any given radius ratio, r_o/r_i, the higher of these two types of stress determines whether the material will yield. By equating σ_θ to σ_r, it readily follows that for r_o/r_i less than the cube root of four the circumferential stress dominates; for r_o/r_i greater than this value (that is about 1.6) the maximum pressure that can be contained within the elastic limit will be determined by the radial stress.* The values for the yield strength of U-Al and Zr alloy fuel at 200°C are approximately 7,000 psi and 45,000 psi respectively (10).

Since the size of the bubble is so small, the effect of surface tension must be taken into account in determining the pressure which can be contained. The pressure increment due to this is given by --

$$\Delta P = 2 \gamma / r_i$$

where γ is the surface tension of the metal. This constant for uranium at 800°C is approximately 1000 dyne/cm. (6). Assuming this value at 200°C, the pressure increment due to the surface tension ranges from 3000 psi for a bubble radius of 100 Ångstroms to 30,000 psi at 100 Å. By adding the surface tension and elastic yield strength, the internal pressure which can be contained in a given bubble radius is obtained.

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* The maximum stress theory was utilized in this analysis and is believed to be sufficiently accurate. Somewhat more accurate results could be obtained with the maximum distortion energy theory.

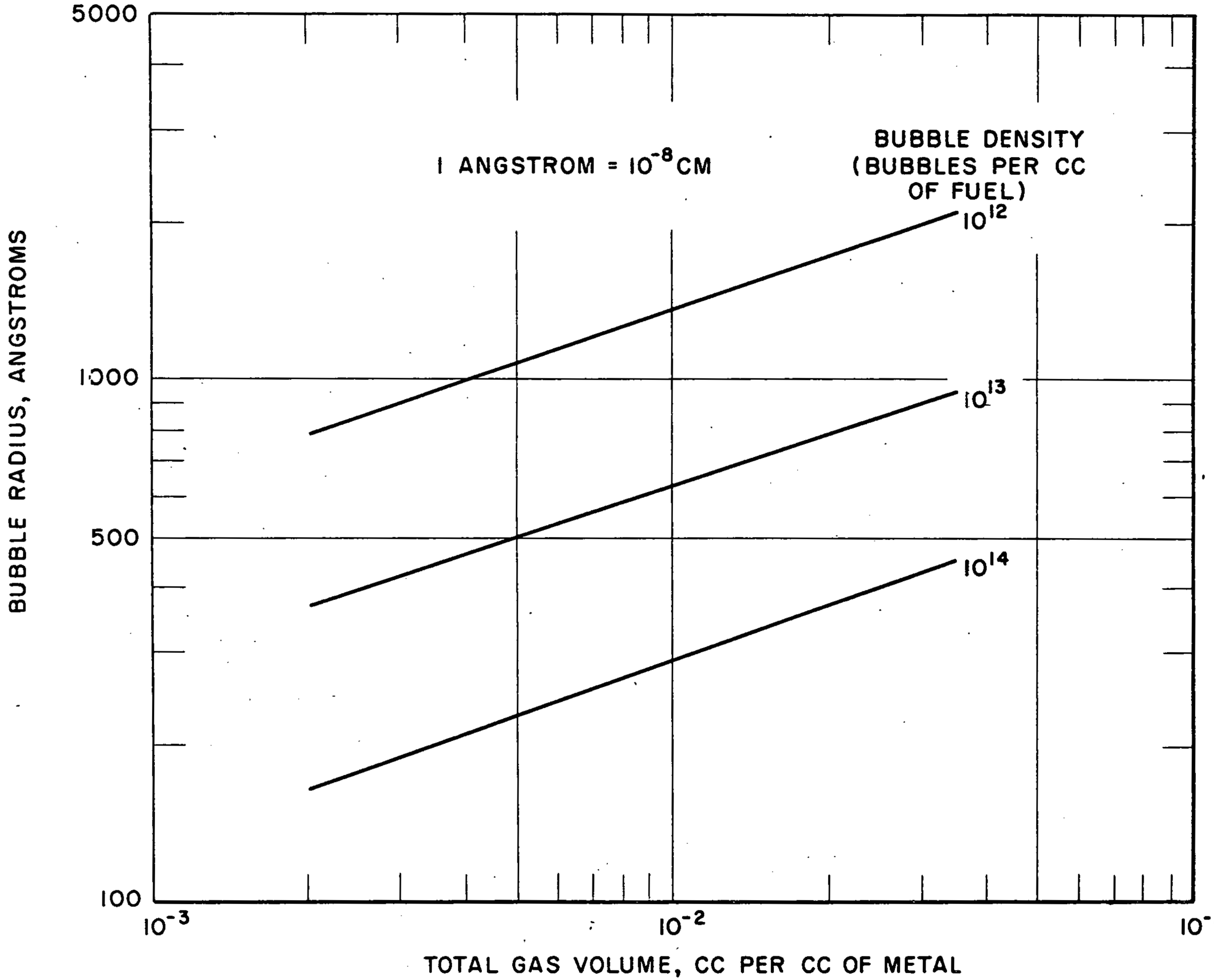

FIGURE 4: RELATION BETWEEN SIZE, NUMBER, AND VOLUME OF BUBBLES

The results shown on Figure 5 indicate that within the range of 100 and 1000 Ångstroms bubble radius and for less than 10^{14} bubbles per cc. of fuel, the bubble pressure is approximately 10,000 - 50,000 psi for the U-Al alloy plate. The wide range is due, in part, to the large effect which the surface tension has on the pressure contained by the sphere (see Figure 5). This pressure is to be compared with that calculated from the equation of state assuming 10^{13} bubbles per cc. of fuel, as observed by Churchman and Barnes (6) and a bubble radius of approximately 500 - 600 Ångstroms. The corresponding gas pressure is approximately 10,000 - 50,000 psi as shown on Figure 3.

In a similar manner, the pressure that can be contained by the U-Zr alloy is calculated to about 50,000 - 60,000 psi (Figure 6). This again corresponds to the pressure computed assuming a density of 10^{13} bubbles per cc. of fuel and a radius of about 500 Ångstroms.

Thus, based on the preceding analysis the bubble pressure under steady-state conditions is 10,000 - 50,000 psi for the U-Al fuel and 50,000 - 60,000 for the U-Zr fuel. The corresponding swelling, assuming no initial cavities in the fuel, is in the range of 0.5 to 0.8% according to Figure 3. This compares favorably with the swelling of 1% or less obtained in Reference 7 and 11 with somewhat different analytical models.

Pressure Increase During a Reactor Transient

Having estimated the internal gas pressure under steady-state conditions, this section of the report will consider the effect of a reactor transient on the gas pressure and material strength. Several excursions have been considered for both the U-Al and U-Zr plate elements; the method of analysis and results are reported in Reference 3. The variation of the metal temperatures with time for a reactor period which produces molten metal is shown on Figure 7 for the U-Al and U-Zr plate.

Note that the temperature in both materials exceeded the melting point within a duration of approximately 100 ms*. Even making the assumption that the material strength remained constant, the volume would increase by a minimum of 1 to 2% in order to maintain the same pressure as at the steady-state conditions.

Several of the references (11,12) have indicated the change of volume during post-irradiation annealing of metallic fuel elements. Generally, the duration of the annealing was long compared to the short transients of a fraction of a second studied in this report. Test parameters such as fission rate and normal reactor operating temperature have been observed to have an effect on the volume change during the post-irradiation annealing. They

* The abbreviation "ms" is used for milliseconds

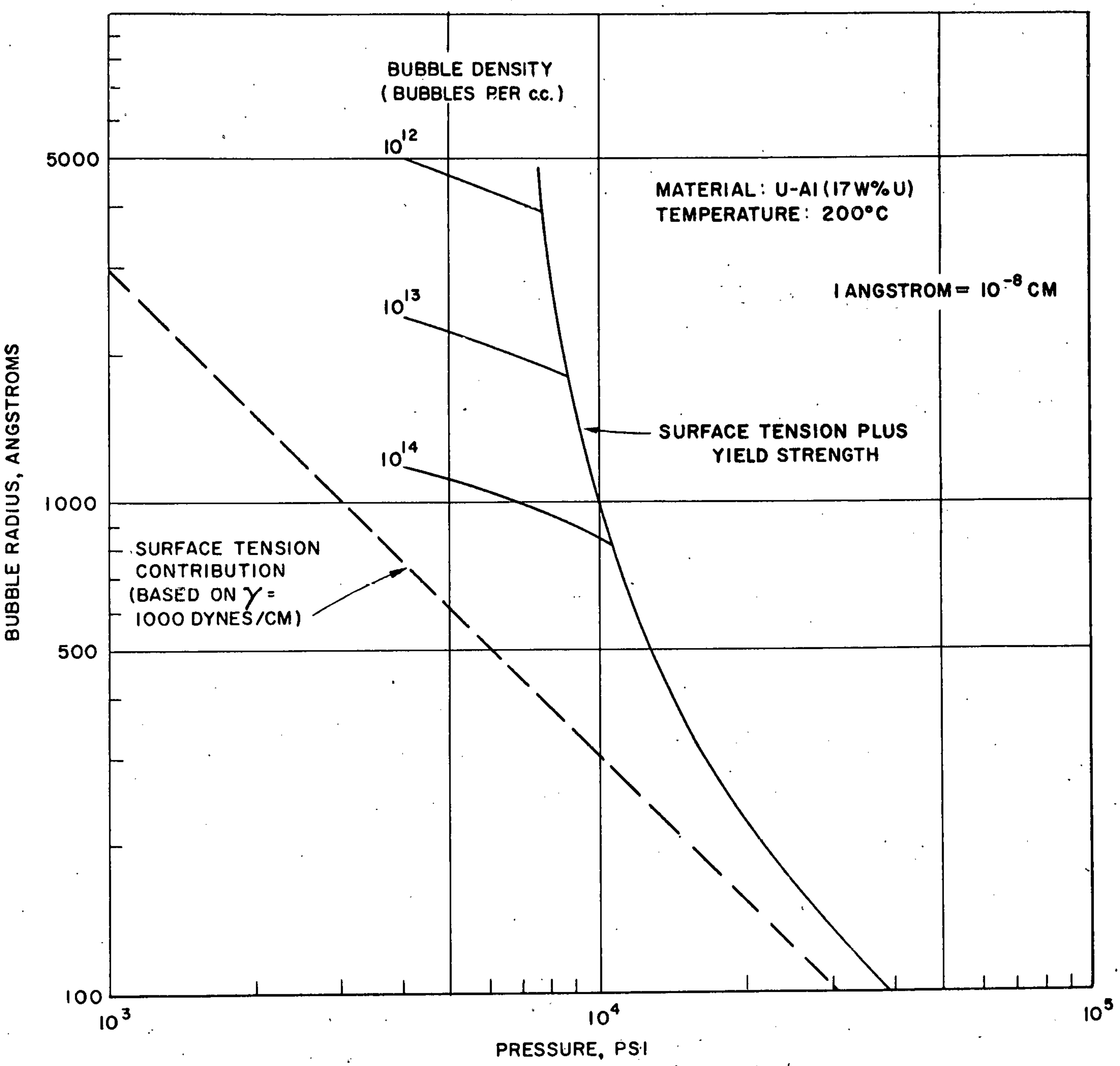

FIGURE 5: PRESSURE CONTAINED BY A SPHERICAL SHELL OF U-Al

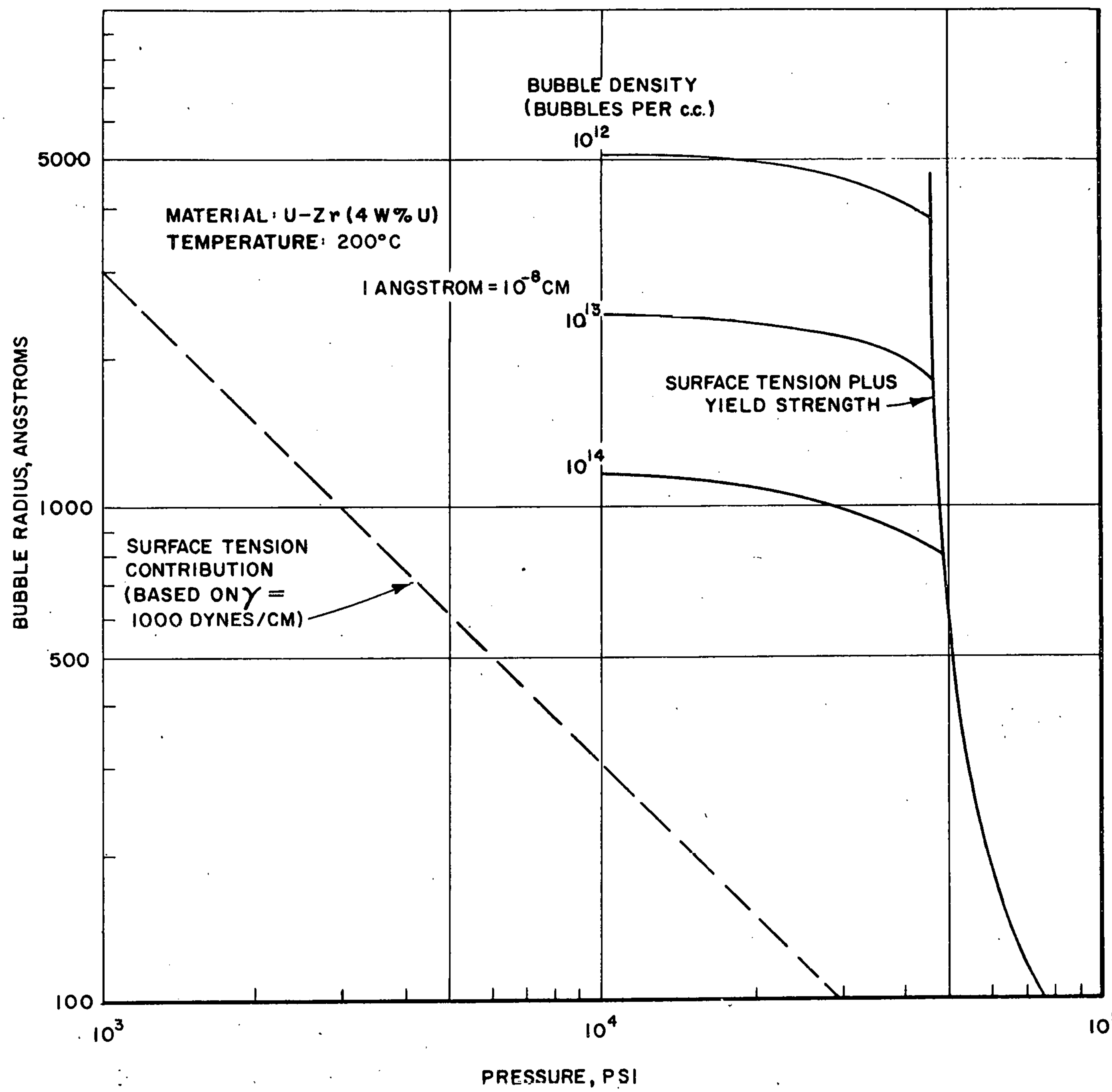

FIGURE 6: PRESSURE CONTAINED BY A SPHERICAL SHELL OF U-Zr

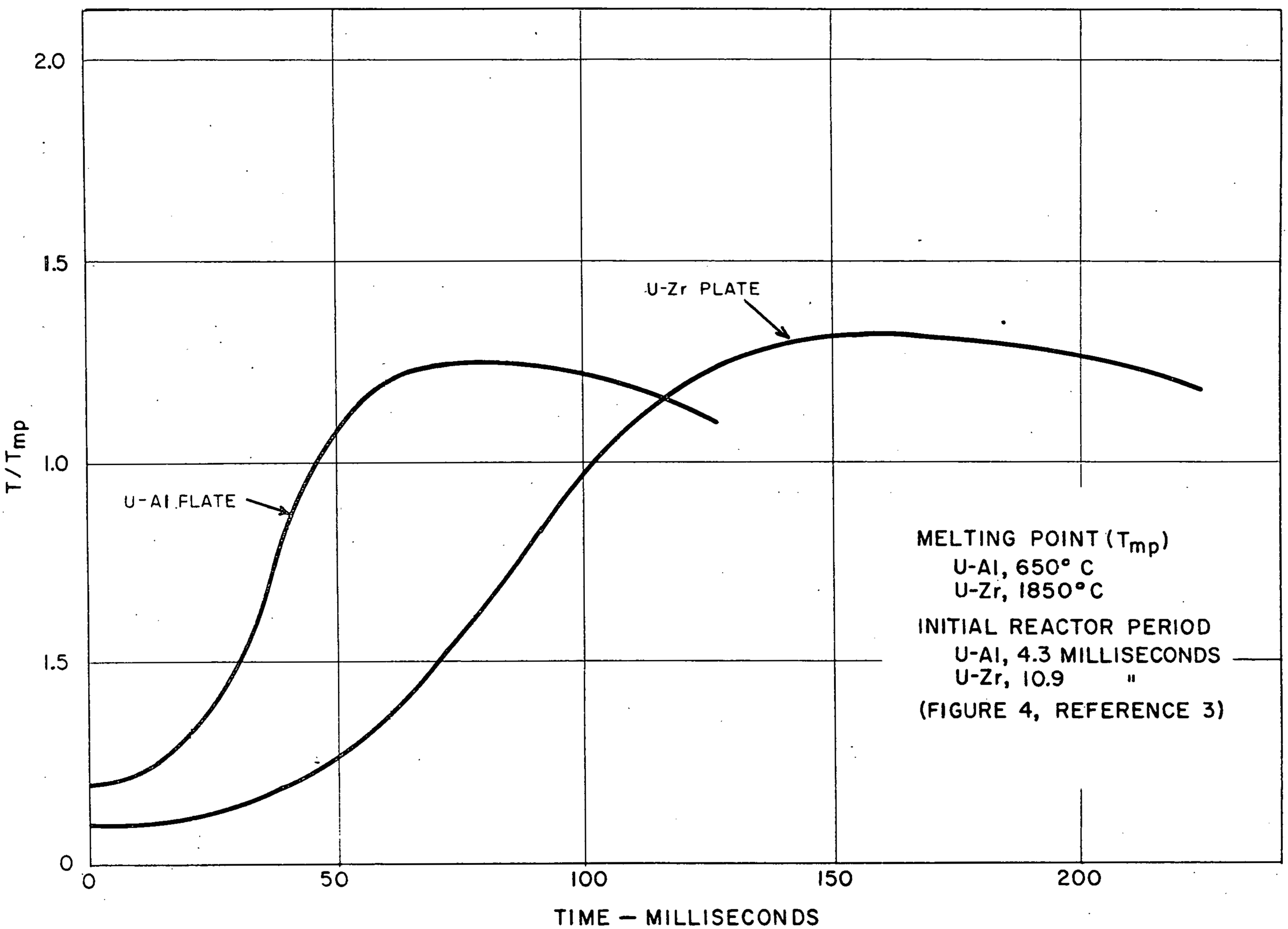

FIGURE 7. RATIO OF MAXIMUM TO MELTING TEMPERATURE IN PLATE FUEL ELEMENT DURING A NUCLEAR ACCIDENT

may also have a similar effect during severe in-pile transients. Also, there is the indication that during normal operating conditions, the temperature of fuel has an important bearing on the quantity of fission products in the form of gas bubbles. Thus, in addition to understanding the strength and behavior under dynamic loading of fuel element materials at temperatures approaching their melting point, the irradiation history of the fuel will have some influence on its behavior during the transient conditions.

In addition to the many uncertainties inherent in the computation, the behavior of solids under the dynamic conditions of rapid load application (approaching shock loading) is still relatively unexplored. If a static model is applied to this case, that is, if the behavior of the system is assumed to be the same as would be observed with slow load changes, then it would be concluded that the metal would undergo gross volume changes and distortion, at the very least. It is not, however, possible to say, by either model, that the metal would or would not disperse into many small fragments at any given time in the transient history.

B. THERMAL STRESSES

The thermal stresses in the element resulting from the temperature gradients across the fuel plate can be approximated by the following expression found in Timoshenko and Goodier (13) -

$$\sigma = \alpha E \Delta T / (1-\nu)$$

Where σ = thermal stress, psi
α = coefficient of thermal expansion, in/in/°C
ν = Poisson's Ratio
E = modulus of elasticity, psi
ΔT = difference between maximum and average temperature in the wall, °C

Because of the higher thermal conductivity of the U-Al alloy, the ΔT for this would be expected to be small; however, the uranium-zirconium alloy has a lower conductivity and a temperature gradient of approximately 650°C occurs soon after the excursion begins. A gradient of this magnitude produces a tensile stress of the order of 50,000 psi which must be superimposed on the local stresses in the spherical shell surrounding each gas bubble. The combined effect of a thermal stress and local stresses around each of the many small gas bubbles during these rapid temperature transients is a difficult problem to evaluate. So many simplifying assumptions would be necessary to solve such a problem that the calculated results would be of only limited validity. Data from pulsed neutron experiments with irradiated fuel would provide a basis for an improved analytical model and give information on the effects of a rapid nuclear excursion on plate type fuel elements.

IV. ANALYSIS OF ZIRCALOY CLAD OXIDE RODS

A typical oxide fuel rod, Figure 1, is constructed of UO_2 pellets housed in a metallic cladding. The UO_2 has a low tensile strength, is very brittle, and cracks radially under even small tensile strains. Forces which have an important effect on the behavior of a fuel rod during a reactor transient are:

1. Expansion forces from the relative expansion of the UO_2 and cladding during a temperature excursion.

2. Internal pressure from the fission gas release.

3. Thermal gradients within the cladding.

A. EXPANSION STRESS

In considering the behavior of a fuel element of this type during a nuclear transient, three facts are important:

1. After irradiation at high heat flux and temperature, the UO_2 pellets are generally cracked into many pieces. The fragments vary in size and shape, and even at ambient temperature, they tend to be in close proimity to the tube wall (14, 15).

2. Sintered UO_2 has a coefficient of thermal expansion about twice as great as that of Zircaloy-2 (10).

3. The temperature rise in the UO_2 during a nuclear transient is quite large, while the ΔT in the zirconium alloy cladding is very small (3). For example see Figure 8.

When a transient occurs in a fuel element of this type, the thermal expansion of the constituents of the assembly causes a load to be exerted on both the UO_2 and the Zr-alloy cladding in times of the order of 10 to 20 ms. In time intervals as short as this, there will be little opportunity for the cracked and fragmented UO_2 to become re-oriented and for the stresses produced by the temperature pulse to be relieved. Thus, a rather large strain can be expected in the clad surrounding the UO_2.

The magnitude of this cladding strain is determined on the assumption that the expansion of the UO_2 produces a swelling force which deforms the jacket with no appreciable deformation (or compacting) of the UO_2. The following expression describes approximately the net radial deformation of the cladding.

$$e = (\alpha r \Delta T)_{fuel} - (\alpha r \Delta T)_{clad} - \delta$$

Where e = radial deformation
α = coefficient of thermal expansion
ΔT = temperature change during transient
r = radius
δ = initial clearance between cladding and fuel

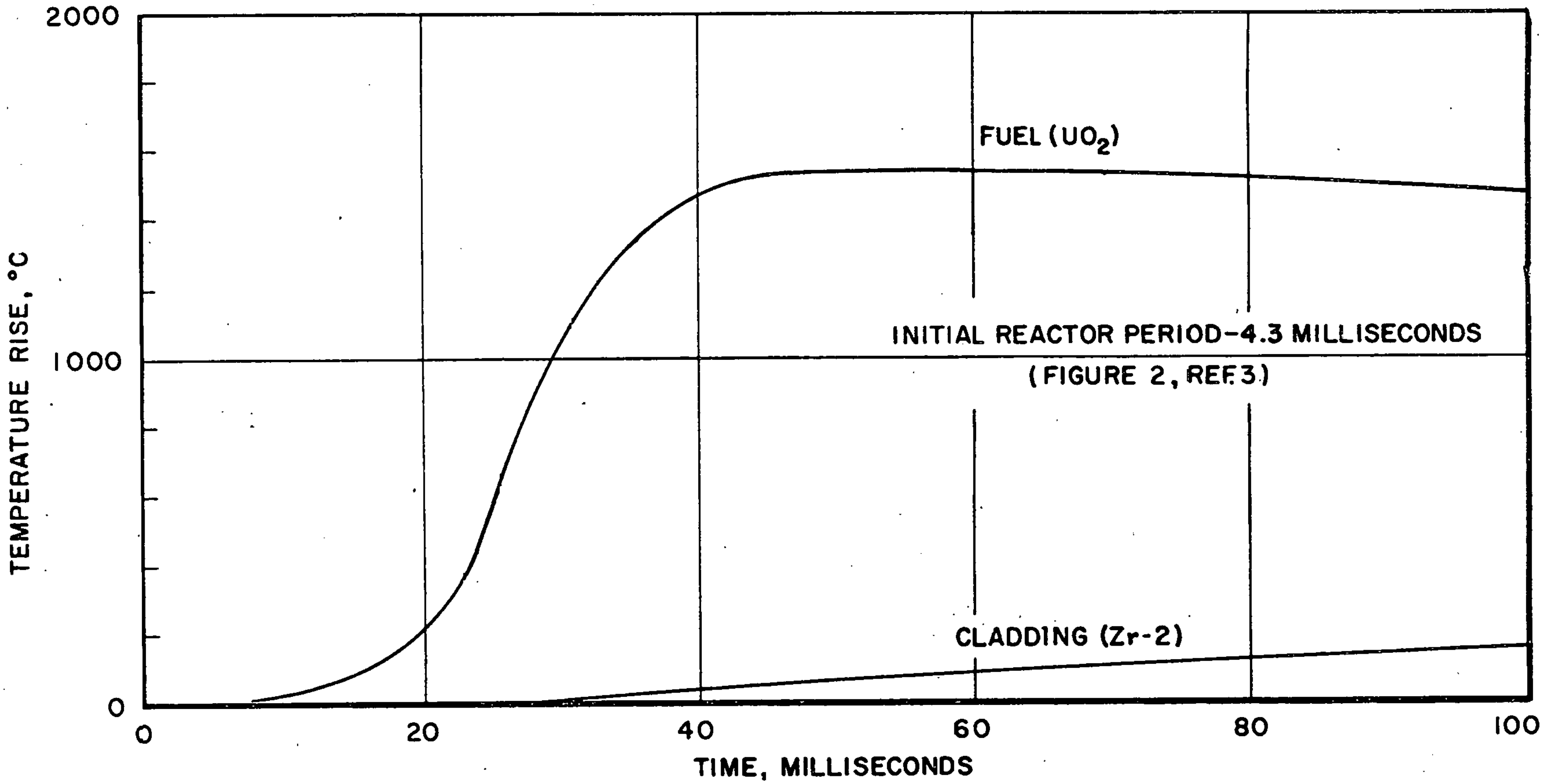

FIGURE 8: TEMPERATURE RISE IN AN OXIDE ROD DURING A NUCLEAR ACCIDENT

The net radial strain is shown on Figure 9 for reactor periods in the range 4 to 10 ms. For the shortest period shown on this curve (4.3 ms.) and with no initial clearance between cladding and fuel, the maximum strain in the cladding is less than two percent. The strain to rupture for unirradiated zircaloy-2* at 200°C is approximately twenty percent (16). Neutron irradiation and absorption of hydrogen and oxygen will tend to embrittle the zirconium alloys thereby reducing the strain to fracture. For example, neutron irradiation reduces the strain to failure from 20 percent to approximately 7 percent after an integrated fast flux of about 3×10^{20} nv (17, 18). During normal operations, hydrogen would embrittle only a few tenths of a mil on the outside surface of the zirconium wall leaving the remaining material essentially unaffected. Thus, the two percent strain calculated above is apparently insufficient to rupture the irradiated material. Also, note that the calculated two percent strain is based on the assumption that no densification of the fuel material occurs. As the temperature of the UO_2 is increased, with shorter periods, the ceramic will become more plastic and some compacting may occur thereby decreasing the amount of strain in the cladding. Initial clearance between cladding and UO_2 will also reduce the net strain, therefore, periods less than 4 ms. apparently will be required to rupture the cladding as a result of volume expansion of the UO_2. There is considerable doubt whether a thermal reactor of this type could ever accidentally achieve such a short period.

B. PRESSURE STRESSES

Some measurements and calculations of fission gas release from irradiated fuel elements have been made (14). The values range from zero to nearly one hundred percent of the fission gas present, depending on the heat flux, geometry, temperature distribution, total exposure, and density of the fuel. The gas which is released exerts a pressure stress on the cladding. The pressure is dependent on the temperature and the void volume available from, for example, the porosity in the uranium fuel, end gaps, and clearances between the UO_2 and the cladding.

According to the excursion studies in Reference 3, the temperature of the UO_2 would increase to as much as 1500°C during the course of a reactor transient. Calculation of a typical design indicates that after an increase of 1500°C above the initial temperature (100°C), the internal pressure will be approximately 1500 - 2000 psi. The resulting wall stress, assuming a 1000 psi external system pressure, is about 6000 psi, which is well below the yield point of zircaloy-2 at 200°C (the studies indicate the increase in cladding tem-

*With 10 - 15 percent cold work.

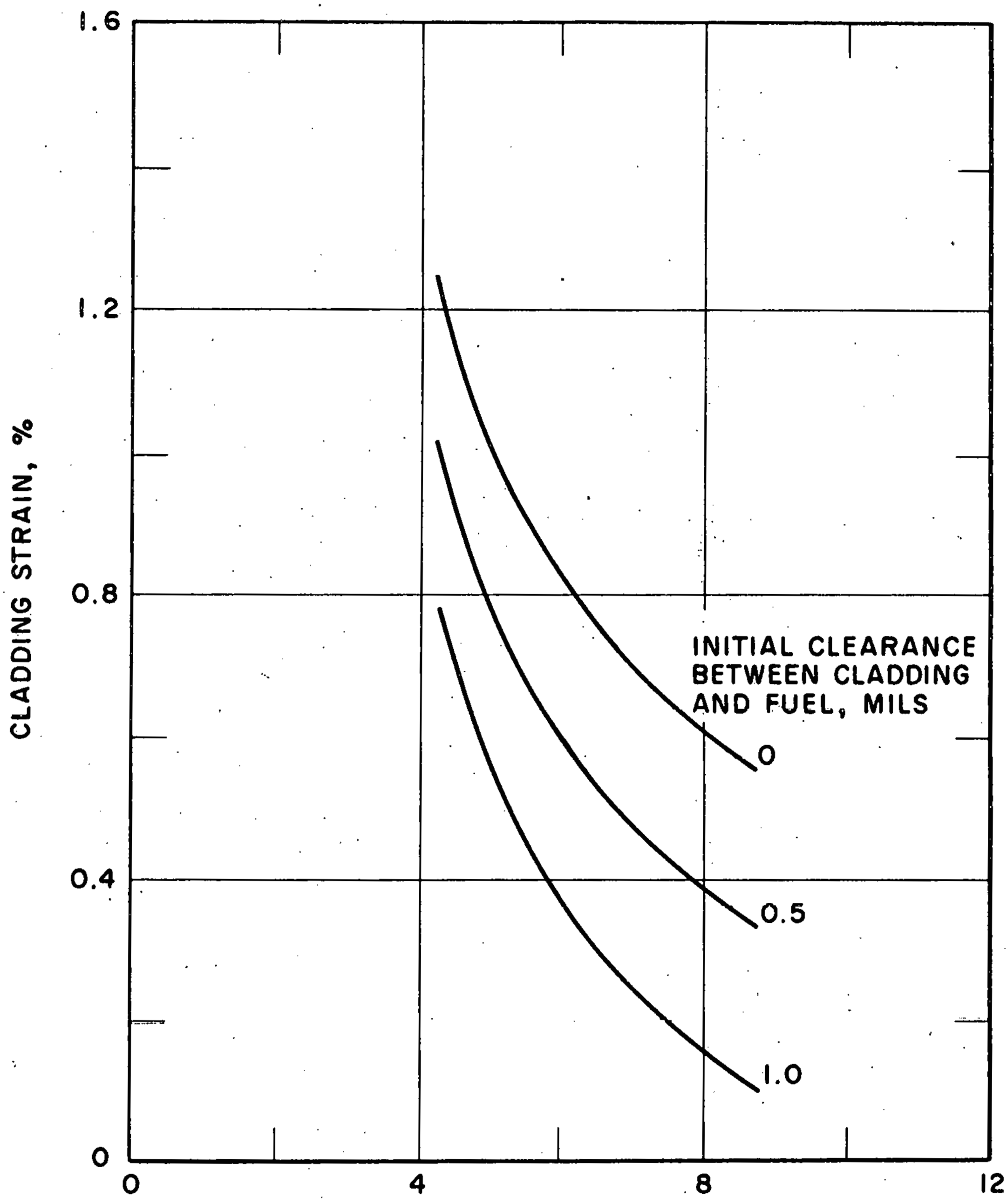

FIGURE 9: CLADDING STRAIN AS A RESULT OF TEMPERATURE INCREASE IN OXIDE FUEL AND METAL CLADDING

perature to be less than 100°C above ambient). This pressure calculation was based on the release of approximately 30 percent of the generated fission gas from the UO_2 after 10,000 MWD/ton fuel exposure.

Assuming 100 percent release, the internal pressure would rise to 5000 - 6000 psi with a corresponding stress of approximately 35,000 psi. This stress exceeds the yield strength (about 30,000 psi at 200°C for Zircaloy-2) but is less than the stress developed at 2 percent strain from thermal expansion. Thus, the pressure stress even with 100 percent gas release is not likely to strain the cladding beyond the value resulting from the thermal expansion.

Another conceivable mechanism for the failure of the cladding during a severe nuclear excursion is due to fragmentation of the UO_2 or "internal explosion" of the many small gas cells. As previously discussed, the state of the potential gaseous fission products (atoms) that are retained in the fuel element during normal operating conditions is not known. It is thought that at low temperatures, the gas atoms remain dispersed throughout the structure, but upon heating, diffuse to voids and flaws in the material where high pressures are formed. Undoubtedly the rapid rate of gas formation, gas release and temperature rise of the fuel during the excursion will highly stress the structural matrix forming the small gas cells. A possible example of this type of "explosive fragmentation" is the experimentation with Thorianite, a naturally occurring mineral containing helium atoms, which, upon heating began to disintegrate at 700°C and exploded catastrophically at 950°C (19).

It thus appears that there is a considerable doubt as to how a metal jacketed oxide fuel would behave during a nuclear excursion. Power pulsed experiments under the aforementioned conditions with irradiated fuel would seem to be the best way to resolve these questions.

C. THERMAL STRESS IN CLADDING

As described in Figure 8, the cladding temperature remains near the initial conditions during the transient. The calculated heat flux during the excursion with a 4.3 ms. period is about 120,000 BTU/hr-ft^2. The temperature gradient across a 30 mil zircaloy wall at this heat flux would be about 30°C. Utilizing the following expression for thermal stress in a long hollow cylinder with thin walls (13) -

$$\sigma = \alpha E \Delta T / (1-\nu)$$

Where σ = circumferential stress, psi
α = coefficient of thermal expansion, per °C
E = modulus of elasticity, psi
ν = Poisson's Ratio
ΔT = difference between maximum and average temperature across the wall

the calculated value is approximately 2000 psi. Thus, the thermal stress which arises in the wall material during the transient does not constitube an important contribution to the failure of the cladding.

V. CONCLUSIONS

Metallic Plates

A. Assuming the fission product gas to be contained in many small bubbles within the fuel by surface tension and the yield strength of the material, the internal gas pressure at the end of the fuel element's "life" and during normal operation is calculated to be in the range of 10,000 - 50,000 psi for the U-Al fuel and 50,000 - 60,000 psi for the U-Zr fuel. These pressures are based on 2.5 cc. of fission gas per cc. of metal and an operating temperature of 200°C, and correspond to a bubble density of 10^{13} bubbles per cc. of fuel and a bubble radius of 500 Ångstroms.

B. During a reactor temperature excursion severe enough to melt the fuel plate, the surface tension and yield strength decrease rapidly while the internal fission product gas expands. Considerable expansion or possibly dispersion of the fuel is expected to take place prior to melting. Experiments that will expose irradiated plate fuel to severe nuclear excursions are recommended to establish the mode of failure and the extent of the dispersion.

Oxide Rod

A. The combined strain resulting from the relative expansion of the fuel and cladding, and a thermal gradient across the cladding during a nuclear excursion of 4 ms. is approximately 2%. Since the strain to rupture irradiated Zircaloy-2 at 200°C is in the range of 7 to 10 percent, it is apparent that the cladding in general, will not rupture during the reactor excursions considered.

B. Localized areas of cladding may experience higher strains due to surface imperfection, corrosion, embrittlement and non-uniform pressure distribution within the cladding.

C. Due to the uncertainties in the behavior of the fuel during a severe nuclear excursion and the localized conditions described above, transient nuclear experiments with irradiated fuel and embrittled cladding are recommended to establish fuel element integrity.

VI. BIBLIOGRAPHY

1. Higgins, H. M. and Schultz, R. D.: The Reaction of Metals With Water and Oxidizing Gases at High Temperatures, IDO-28000, (April, 1957).

2. Littman, F. E., Church, F. M., and Kinderman, E. M.: The Reactions of Titanium With Oxygen and Steam, Trans. of American Nuclear Society, p 156, (June, 1958).

3. Ownes, J.: An Evaluation of Nuclear Excursions in Light Water Reactors, GEAP-3178, (June, 1959).

4. Holden, A. N.: Physical Metallurgy of Uranium, Addison-Wesley, (1958).

5. Reynolds, M. B.: Diffusion of Fission Krypton From Metallic Uranium, Nuclear Science and Engineering 1, p 374, (1956).

6. Churchman, A. T. and Barnes, R. S.: Swelling and Inert Gas Diffusion in Irradiated Uranium, Section 3 - Swelling of Irradiated Uranium on Subsequent Out-of-Pile Heating, Geneva Paper 81, (1958).

7. Enderby, J. A.: Plastic Flow and the Swelling of Gas Bubbles in Uranium, IGR-R/R 198, (1956).

8. Willis, A. H.: Sources of Fuel Element Instability, Geneva Paper 616, (1958).

9. Roark, R. J.: Formulas For Stress and Strain, McGraw-Hill Book Co., 3rd Edition, p 276, (1954).

10. Etherington, H.: Nuclear Engineering Handbook, McGraw-Hill Book Co., 1st Edition, pp 10-18, (1958).

11. Wyatt, L. M., Enderby, J. A., and Foreman, A. J. E.: Swelling and Inert Gas Diffusion in Irradiated Uranium, Section 5 - Calculation of Swelling, Geneva Paper 81, (1958).

12. Johnston, W. V.: The Effects of Transients and Longer - Time Anneals on Irradiated Uranium - Zirconium Alloys, KAPL-1965, (1958).

13. Timoshenko, S. and Goodier, J. N.: Theory of Elasticity, McGraw-Hill Book Co., 2nd Edition, p 399, (1951).

14. Eichenberg, Frank, Kisiel, Lustman, and Vogel: Effect of Irradiation on Bulk UO_2, WAPD-183, (1957).

15. Imhoff, Cook, Pennington, and Spalaris: Experimental Programs in the Vallecitos Boiling Water Reactor, (Presented at annual meeting of the American Nuclear Society, June 17, 1959 - to be published).

16. Power Reactor Technology, 1, p 36, (February, 1958).

17. Reactor Technology Report No. 8, Metallurgy, KAPL-2000-5, (1959).

18. Howe, L. M. and Thomas, W. R.: The Effect of Neutron Irradiation on the Tensile Properties of Zircaloy-2, AECL-809, (March, 1959).

19. Barnes, R. S. and Greenwood, G. W.: The Effect of Gases Produced in Reactor Materials by Irradiation, Geneva Paper 29, (1958).