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2nd INTERNATIONAL SEMINAR ON

Standards and structural analysis in elevated temperature applications for reactor technology

ENEA - ENEL

Venice, Italy, October 15 - 17, 1986

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2nd INTERNATIONAL SEMINAR ON

STANDARDS AND STRUCTURAL ANALYSIS IN ELEVATED TEMPERATURE APPLICATIONS FOR REACTOR TECHNOLOGY

TITLES OF THE PAPERS

Deutschland

D.1 – Elastic Creep-Fatigue Damage Evaluation Method: Application to SNR 300.
A. Angerbauer, H. Meyer-Reumers, A. Sapci

D.2 – Lifetime Prediction and Strain Calculation for Materials Subjected to High Temperature Deformation at Non-Stationary Stress and/or Temperature Loading.
M. Boček.

D.3 – Methods of Life-Time Prediction for High Temperature Components . Using Structural Design Codes and Inelastic Analysis.
K. Bieniussa, E. Bodmann, M. Dette, H. Nickel, F. Schubert.

D.4 – Life-Time and Creep Ratcheting Calculation of Two Typical HTR-Components.
K. Bieniussa, G. Breitbach, H.H. Over, H.J. Penkalla, F. Schubert, H.J. Seechafer.

D.5 – The Deformation and Fracture Behaviour of Tubes Under Multiaxial Loading Conditions at Temperatures Above 800°C.
G. Breitbach, K. Franske, H.J. Penkalla, F. Schubert.

D.6 – Engineering Approach to Elastò-Plastic Strain States Induced by Thermal Loading.
Führing.

France (cont'd)

F.7 - Dynamic Buckling of Structure Loaded by Seismic
A. Combescure, A. Hoffmann, E. Homnan

F.8 - Design Rules for Piping Components Against Excessive Deformation and Elastic Collapse. Application to Pipe Bends Experiments.
D. Acker, F. Touboul.

F.9 - Fatigue Simplified Elastic-Plastic Analysis.
D. Autrusson, D. Acker, A. Hoffmann.

F.10 - Review of Analysis Methods to Prevent Thermal Buckling.
D. Moulin, A. Combescure, D. Acker.

F.11 - Buckling of Thin Cylinders Under Lateral Pressure and Cyclic Thermal Axial Gradient.
N. Waeckel, J.F. Jullien.

Italy (cont'd)

I.5 - CEC Elasto Plastic-Creep Benchmark Studies Relevant LMFBR Structural Components.
F. Corsi, A. Terzaghi, F. Donarelli.

I.6 - Life Evaluation of a Nozzle Charged by Low Primary Loads and Severe Cyclic Thermal Shocks.
F. Corsi, G.M. Giannuzzi, P. di Giamberardino, B. Taddia.

I.7 - Relevant Mechanical Characteristics of an AISI 316 Steel Plate for High Temperature Structural Integrity Evaluations.
D. D'Angelo, V. Regis.

I.8 - Application of a Load-Resistance-Factor-Design (LRFD) Format for the Design of High Temperature Components.
G. Ferro, J. Van Dyck, G.M. Manfredini.

I.9 - Post welding residual stress analysis in PEC reactor fuel element upper grid.
S. Barilli, A. Finelli, P. Neri,etc.

I.10 - AISI 316 Weldments Under Cyclic Loads at High Temperature.
E. Manfredi.

I.11 - Defect Allowable Size Assessment in PEC Core Element Welds.
P. Neri, P. Pierantozzi.

I.12 - Application of Elastic Follow up Evaluation Criteria on Simple Geometries.
G. Stretti, F. D'Amico.

Session Nº	Date	Time	Session Title	Papers
	10.15	9.30-10.30	Registration of participants	
	10.15	10.30-11.00	Coffee break	
I	10.15	11.00-13.30	Opening address - Overview of national practices in design	F.4; D.9; GB.1; I.15
	10.15	13.45-14.30	Lunch	
II	10.15	14.30-15.45	Material characterization for E.T. design	I.7; I.10; D.5; V.1
III	10.15	16.00-17.45	Design criteria and general methodologies	F.2; I.2; GB.2; D.6; I.8
IV	10.16	9.00-10.45	Life prediction procedures	D.3; D.4; I.6; D.2
	10.16	10.45-11.00	Coffee break	
V	10.16	11.00-13.30	Stress analysis of components	D.1; I.3; D.7; D.8; I.1; F.3; I.16
	10.16	13.45-14.30	Lunch	
VI	10.16	14.30-15.45	Non-elastic analysis	GB.4; F.8; I.5; F.6
	10.16	15.45-16.00	Coffee break	
VII	10.16	16.00-17.30	Simplified analysis in design	GB.5; F.5; I.12; F.9
VIII	10.17	9.00-10.45	Core element design requirements	F.1; I.13; I.11; I.9
	10.17	10.45-11.00	Coffee break	
IX	10.17	11.00-13.30	Buckling and fracture aspects	I.14; GB.3; F.7; I.4; F.10; F.11
IX	10.17	13.30-13.45	Conclusion remarks	
	10.17	13.45-14.30	Lunch	
	10.17	14.30-16.00	Round Table on: "A Common Set of Standards for the Next European FBR Power Station"	
	10.17	16.00	Closure	

A METHODOLOGY TO ANALYZE THE STRUCTURAL BEHAVIOUR AT HIGH TEMPERATURE OF HEAVY WATER MODERATOR REACTOR PRESSURE TUBES

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A methodology to analyze the high temperature behaviour of the Pressure Tubes of Heavy Water Moderated Natural Uranium Fuelled Nuclear Plants has been developed.

The high temperature conditions for these components is expected as a consequence of the partial break of a feeder, which can cause a complete stagnation of the primary coolant, accompanied by a relevant high pressure state.

Each Pressure Tube, made in Zircaloy-2, is located inside a relevant Calandria Tube, also made in Zircaloy-2, whose functions is to separate the Pressure Tube from the moderator. A separation between the two tubes has furtherly to be guaranteed in order to avoid embrittlement phenomena. $CO2$ gas fills the gap between these two tubes.

Due two above described accident, two different but related modes of rupture can start:

a) a high speed creep mechanism, which can expand the Pressure Tube up to going into contact with the relevant Calandria Tube;
b) a bursting mechanism due to simultaneous high pressure and reduced mechanical characteristics.

The contact may be eventually localized or circumferentially generalized. In both cases the creep mechanism can be accelerated and the burst pressure furtherly reduced. Also circumferential thermal gradients can arise. A complete analysis of this situation can requires either analytical or experimental approach.

The analytical approach is presented in this paper.

This methodology has been successfully applied to an Italian Heavy Water Moderated Nuclear Plant.

A METHODOLOGY TO ANALYZE THE STRUCTURAL BEHAVIOUR AT HIGH TEMPERATURE OF HEAVY WATER MODERATOR REACTOR PRESSURE TUBES

INTRODUCTION

This paper considers the mechanical behaviour of Heavy Water Reactor Pressure Tubes when subjected to the thermal transients caused by a partial feeder break, in presence of flow stagnation.

Heavy Water Reactor Pressure Tubes have the function to support and position the fuel bundles and contain the primary coolant.

Usually Pressure Tubes consist in a central portion made in Zircaloy-2 facing the fuel bundles and in lower and upper steel end-fittings connected to the relevant pressure tube by means of rolled joints and to the primary lines by means of mechanical joints. The inlet line is usually indicated as the "feeder" and the outlet line as the "riser", referring to vertical Pressure Tubes.

Each Pressure Tube is located inside a relevant Calandria Tube (CT) surrounded by the low temperature heavy water moderator. These Calandria Tubes have either the function of containing the moderator or the function of thermal insulating the Pressure Tubes. To reach this goal a gap between each couple of tubes has to be guaranteed during each steady operational condition. The contact between the Pressure Tube and its relevant Calandria Tube has to be avoided in order to prevent embrittlement phenomena of the pressure tube itself. The only mechanical contact admitted is by means of two garter springs which have the task to reduce the span lenght of the Pressure Tube.

A sketch of a typical vertical Pressure Tube and the relevant Calandria Tube is shown in Figures 1 and 2.

DESIGN CRITERIA AND MATERIAL BEHAVIOUR

These components are usually designed referring to the ASME Section III "Boiler and Pressure Vessel" Code, but because of the peculiar material used for the Heavy Water Reactor Pressure Tubes it is not possible to directly use the ASME III stress limits for all the service levels to be analyzed.

The main difference with respect to ASME III steels is the Zircaloy-2 reduced strain-hardening capability, in particular after radiation exposure. The Zircaloy-2 material presents an Ultimate strenght/Yield strenght ratio less than 2, while the ASME III steels usually have a ratio equal to or greater than 2. This aspect causes that the direct application of ASME III limits above the yield strenght of Zircaloy-2 gives a lower safety margin than for the usual ASME materials. As a consequence of that, in order to have safety margins not lower than those given by ASME III Code, "ad hoc" limits have to be set up for the level C and D service conditions, since for these levels the ASME allowables are greater than the yield strength. In Figure 3 the allowables defined for the material we are discussing about are listed. It may be observed that the ultimate strength is the controlling parameter , so the stress limits are set up based on this value. As a cosequence the Primary Membrane Design Stress Intensity for faulted conditions is never larger than 70% of the Ultimate Strength.

The Design Stress Intensity, Sm, has to be defined taking into account the anisotropy of this material, whose anisotropic behaviour is shown in Figure 4 (see ref. [1]). The variation of Zircaloy-2 properties vs. temperature are also shown in Figure 4 (see ref. [1]).

The above discussion refers to the standard stress verification for emergency and faulted condition based on using Primary Stress Intensity , calculated by linear-elastic methodologies, to be compared with the suitable Design Stress Intensity. The alternative procedures for faulted conditions, outlined by ASME III Appendix F, can be used with no modification of the allowable limits, because they are based on the ultimate strength which is the controlling parameter for this material.

ACCIDENTAL SCENARIOS WHICH CAN GIVE A HIGH TEMPERATURE STATE FOR THE PRESSURE TUBES

The pressure tubes can experience a very high temperature state, with temperature very higher than the design one, when subjected to the pressurized thermal transient following a very particular size partial feeder break accident.

Such an accident represents a faulted condition for the involved pressure tube, thus the level D service limits have to be applied in studying its effects on that tube.

Following the most conservative approach, a complete stagnation of the coolant has to be postulated as a consequence of this accident. This stagnation is the cause of a sudden increase of the coolant temperature, while the pressure remains pratically unchanged with respect to the normal operating value, at least in correspondence to the highest temperature values.

Typical temperature vs. time and pressure vs. time curves associated to such an accident are shown in Figure 5. The scenario highlighted has a very high potential of impairing the involved pressure tube, because three different but related failure modes can start:

1) the bursting pressure can be reached;
2) thermal creep phenomena can occur;
3) local instability can occur.

FAILURE MODES

A pressure tube is depending with respect to the following failure modes:

- static collapse (bursting);
- buckling (elastic and/or plastic);
- creep;
- non-ductile fracture;
- fatigue.

The creep phenomena usually considered are only those related to the effects of the neutron radiation, because these tubes usually operate at moderate temperatures, where no thermal creep phenomenon can occur.

As before said, the accident above described can start three of these failure modes.

The static collapse can be experienced if the applied pressure can become larger than the burst pressure of the tube. The examined situation has a high potential for static collapse because the high temperature causes a very large decrease in Zircaloy-2 mechanical properties (see Fig. 4) and the pressure acting on the component remains pratically at the normal operating value. For static collapse, however, pressure is not the only load to be considered, even if it is the predominant one.

The high temperature state starts a high speed thermal creep mechanism which together with other analogous phenomena (radiation creep, thermal expansion, pressure itself) and depending on time duration of this state can expand the pressure tube up to going into contact with the relevant Calandria Tube.

Due to creep the pressure tube becomes weaker because its inner diameter increases and thickness decreases (see Figure 6 and 7). For this reason the bursting pressure decreases itself so enhancing the possibility for static collapse.

The creep phenomena if very lasting in time can lead to failure in an independent manner with respect to pressure effects. Furtherly the contact between calandria and pressure tubes may be dangerous for the integrity of the calandria tube itself and can start local instability phenomena which can be a further cause of failure.

The other modes of failure usually considered for this component (buckling, neutron creep,non-ductile fracture, fatigue) are not involved by this accident so they are not considered in this analysis. Based on the above considerations, an analytical methodology to assess the integrity of such tubes in front of the considered accident has been developed. In the flow-chart of Figure 8 the main items and their logical connections of this methodology are presented.

METHODOLOGY

In Fig. 8 the activities to be performed in order to analyze this situation are shown. Essentially three kinds of analyses are carried out depending on the three involved failure modes.

In order to qualify the component against this accident all these analyses have to be satisfied. In practice it is possible to show that the local instability analysis can be performed in different ways depending on the possibility to demonstrate that the gap existing between the pressure and the calandria tubes is greater than zero during the entire transient.

To perform all the above analyses, stylized transients (see Figure 10) conservatively derived from the postulated ones (see Fig. 5) are used. On the point of view of local plastic instability analysis the involved parameters are the circumferential and the longitudinal stresses, entering the diagram shown in Figure 9. This approach is felt to be yet valid in the examined condition if the constraint of the tube is unchanged with respect to the normal operating conditions. If a contact is experienced the determination of the state of stress requires a very particular methodology and also the usual verification approach may be no more valid. So it is very fundamental to check if contact may be realized or not, in order to decide if the standard plastic instability methodology is still valid.

Due to the complexity of the problems, an analyisis of the local behaviour of the two tubes come into contact couldn't be only an analytical one but could require also experimental studies.

So it results to be very fundamental to verify if the contact could be experienced or not by the tubes, if the contact has the characteristic to be very lasting in time and if the contact is very marked.

If the contact is such to not anticipate any contacts (global or local) between the two tubes, as said before no local instability problems are expected, so experimental studies studies result to be unnecessary. Also it experiments are necessary if moderate contacts for a very short time are found or the contact has an oscillatory nature.

Local instability problems is expected to start if a state of permanent contact is obtained. Such a permanent situation can turn an initial local contact into a global one so weakening an entire beltline of the tube, which can experience instability phenomena.

The only analytical approach is herein presented because when the methodology has been applied, it has been possible to prove that no contact is anticipated for the two tubes, due to the postulated accidental transient.

STATIC COLLAPSE APPROACH

The first objective is to study the state of stress induced by the considered temperature and pressure transients and compare it with the allowables.

The aim of this evaluation is to determine the safety margins against rupture during the LOCA transient, considering all the loads acting on the tube besides pressure, and the variation vs. time of the mechanical characteristics.

Due to the severity of the situation, more sophisticated methods than the simple elastic analysis have to be used. In particular the stress ratio method has been felt to be practicable.

Even if the reference code doesn't give a specific formulation for this method in the case of piping or components subjected to pressure (see Appendix A-9000 of ASME III Code, ref.[3]),the stress ratio method has been applied treating the pressure effects as given by a normal force and considering the maximum pressure stress (the circumferential one) overposed to the other stresses (axially oriented).

The stress ratios have been calculated during the transient (one time per second) using each time the state of stress given by the relevant pair of temperature and pressure and the allowable stresses determined with reference to the actual temperature. No matter in determining the allowable limits for all the loads but pressure, because they are indicated by the Code.

For pressure it has been deemed that the burst pressure with a safety margin should be used. The safety margin has been assumed as by ASME III Code; so the value 0.7 has been used. The burst pressure has been determined by means of the following relationship (see ref. [2],[4] and [5]):

$$Ps = Su\ (0.25/(n+0.227))\ (e/n)^{n}\ (2\ t/Di)(1-t/Di)$$

where:

- Su = ultimate strenght
- n = strain hardening ratio
- e = Neper number
- Di = inner diameter
- t = thickness

Before performing any calculation, the simple check to determine if the pressure is itself able to failure the tube has to be carried out. So the pressure acting inside the tube during the tranient has to be compared with the burst pressure at each instant. This comparison gives the residual margins against failure and thus the residual ability of the components to withstand more loads besides pressure.

In determining Ps the irradiated and unirradiated conditions have to be assumed, since both mechanical (due to radiation hardening of the material) and geometrical (due to neutron creep phenomena) characteristics change. An upper bound may be found not considering the hardening effects for Su but using the irradiated value for "n" (lower than that for unirradiated naterial). Also the effects of the thermal creep which cause a further change of the geometrical properties have to be considered.

In Figure 11 is shown the variation of the burst pressure against the applied pressure versus time, during the styilzed transient presented in Fig. 10. In Figure 12 is shown the variation of Stress Ratio versus time obtained in ths case when the methodology has been applied (for CIRENE Heavy Water Reactor). Having the Stress Ratio always under 1.0 during all the transient indicate that no static collapse problem ise expected.

DEFORMATION APPROACH

This approach is based essentially in determining the uniform cirumferential expansion of the tube due to all the effects which it experiences.

The effects usually taken into account are (in the elastic field):

- neutron creep;
- thermal expansion;
- pressure.

At the actual temperatures the thermal creep becomes very important and, depending on the duration, is the controlling effect. All these causes have to be evaluated to determine the axisymmetric expansion of the tube. The relationship used are the following:

a) for neutron creep (see ref [1]):

$$\varepsilon_{\vartheta}^{n.c.} = \sigma_{\vartheta} \cdot K \cdot \phi \, (T-160)\, t$$

where:

$\varepsilon_{\vartheta}^{n.c.}$	= circumferential strain due to neutron creep	
ϕ	= fast netron flux (E > 1 MeV)	$[n/cm^2\ s]$
σ_{ϑ}	= loop stress	[psi]
T	= temperature	[C]
K	= constant = $4 * 10^{-27}$	
t	= time	[hour]

b) for thermal expansion :

$$\varepsilon_{\vartheta}^{N.C} = \alpha \Delta T$$

where:

α = coefficient of thermal expansion [s^{-1}]

ΔT = difference between present and initial temperature

c) for pressure (only in elastic field) :

$$\varepsilon_{\vartheta}^{P} = P * R /(s E)$$

where:

P =	internal pressure	[MPa]
R =	inner radius	[mm]
s =	thickness	[mm]
E =	modulus of elasticity	[MPa]

d) for thermal creep (see ref [7]):

$$\dot{\varepsilon}_{\vartheta} = A \sigma^{n} e^{-(Q/RT)}$$

where:

A = creep constant
n = strain hardening ratio
Q = activation energy
R = gas constant

The variation of circumferential deformation vs. time is so found (see Figure 13). It has to be compared to the allowable strain, which it has been assumed specifically for this material to be 3 %, only for the verification against rupture (see ref. [1] and [8]).

If the comparison is satisfactory , it is furtherly necessary to check the residual gap between the pressure and the calandria tubes (considering also the expansion of this last tube). If this gap is greater than zero it is possible to state that no local problem should arise. A confirmation of that may be reached, at this time, only when it is possible to exclude also any local contacts.

In order to perform this analysis the real geometry (technological deflections) of the pressure and the calandria tubes have been considered. All the loads applied by the supporting structure (the Reactor Assembly) are considered and the deflection of the two coupled tubes is determined, evaluating the minimum residual gap between the tubes (in such an evaluation all the above parameters are obviously considered). In this manner if a final gap larger than zero is found any local instability problem can be completely excluded. So any other verification except that usually carried out for plastic instability is unnecessary.

In the worst case (pair of Pressure and Calandria Tubes having the most reduced initial gap) the curve of Figure 14 it has been obtained, where the variation vs. time of the gap during the described accidental transient supposed to occur at the end-of-life condition is presented.

PLASTIC INSTABILITY

This verification may be performed in two steps. The first one is that normally used and consists in determining the σ_θ and σ_ℓ acting on the tube and entering in the diagram of the Figure 9. Since also the residual stresses have to be considered , this verification is controlled by particular zones as that of the rolled joints, where the technological procedure is such to froze a high residual stress state.

Also during the considered transient these zones are those more stressed, except in the case a contact is obtained. In this case, however, as already said, a different approach (experimental besides analytical) should be pursued.

APPLICATION

The methodology above described has been successfully applied to the CIRENE Nuclear Power Plant, evaluating the behaviour of the pressure tube subjected to the feeder break with a flow stagnation transient.

It has been demonstrated :

1) the ability of the tube to withstand all the loads and the pressure during the transient;

2) the creep strain (due to both neutron and thermal creep) is less than the allowable;

3) the calandria and the pressure tubes don't come into contact also locally;

4) the plastic instability verification has been conducted using the Figure 9 diagram; no further analysis are necessary for this stand point.

In particular, to evaluate the local final residual gap, the results of a peculiar code to select and optimize the Pressure and the Calandria tubes to be coupled, taking into account their geometrical characteristics, besides all the loads and displacements applied, and examining all the possible configurations, have been used.

CONCLUSIONS

A peculiar method to evaluate the effects of high temperature pressurized transients of H.W.R. Pressure Tubes has been described, in order to consider all the failure modes which can occur in this situation. The methodology has been successfully applied to the CIRENE Nuclear Power Plant

REFERENCES

[1] C.R.N.L. Engineering Manual - Section 5 - Subsection 3
Zirconium Alloys Part 2, rev. 1 - Mechanical properties and corrosion resistence.

[2] Integrity Assessment for a Zircaloy-2 Pressure Tube with Fretting Corrosion Defects.
F. Alicino - G. Zampini
5th International Symposium on PVP - San Francisco - 1984

[3] ASME Boiler and Pressure Vessel Code, Section III - Nuclear Power Components

[4] W.R.C. Bullettin n. 95 - PVRC Interpretative Report of Pressure Vessel Research

[5] W.R.C. Bullettin n. 40 - Design of a Thin-Walled Cylindrical Pressure Vessel Based Upon the Plastic Range and Considering Anysotropy

[6] J. Marin - Mechanical Behaviour of Engineering Materials
Prentice Hall - New Jersey - 1962

[7] K.M. Rose and E.D. Hindle - Diameter Increase in SGHWR Zircaloy Cans Under Loss of Coolant Accident Conditions
A.S.T.M. STP 633

[8] Application Related Phenomena in Zirconium and its Alloys
A.S.T.M. STP 453

Figure 1

LONGITUDINAL SECTION FOR THE CIRENE PRESSURE AND CALANDRIA TUBES

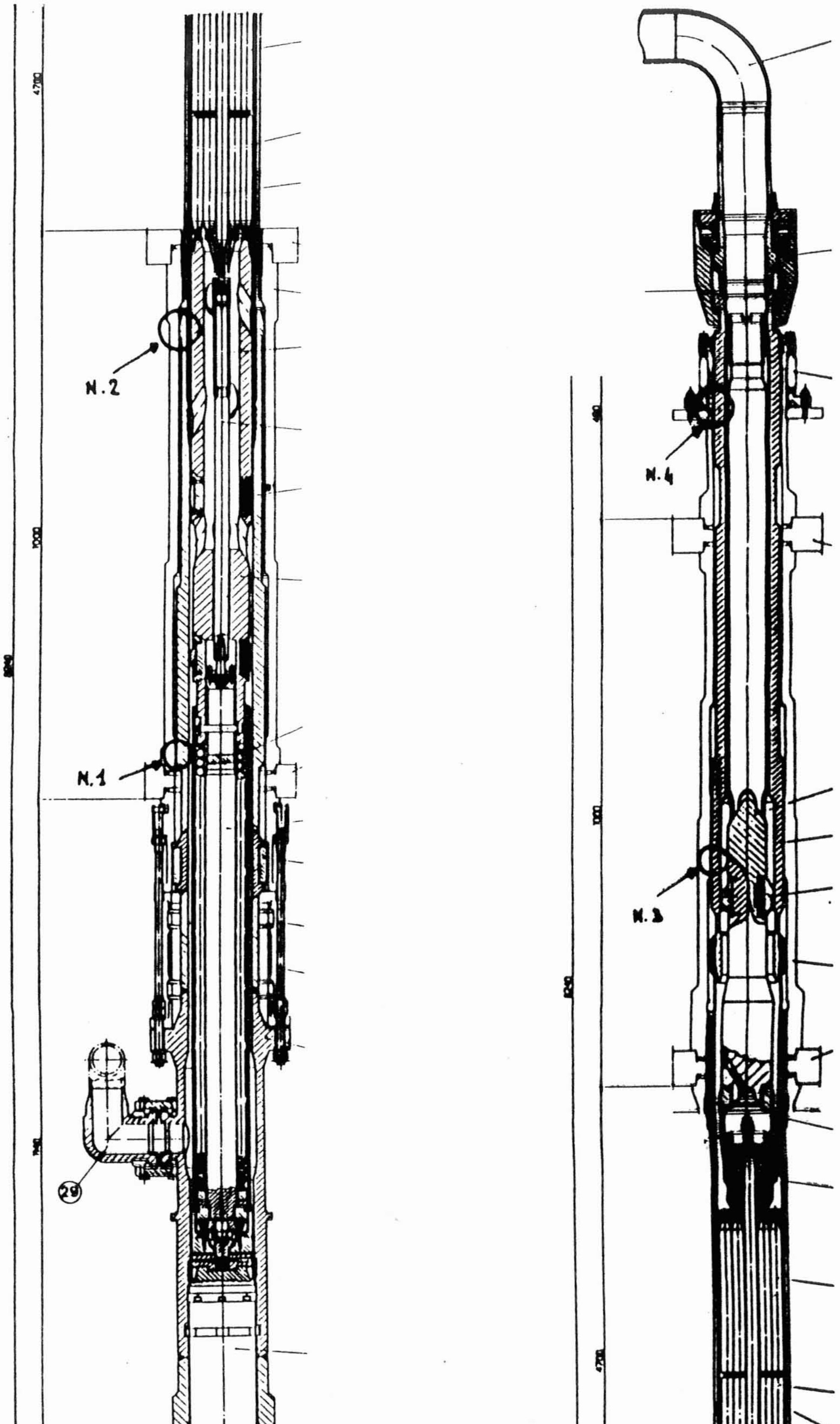


Figure 2

TRANSVERSAL SECTION FOR A TYPICAL H.W.R.
PRESSURE AND CALANDRIA TUBES

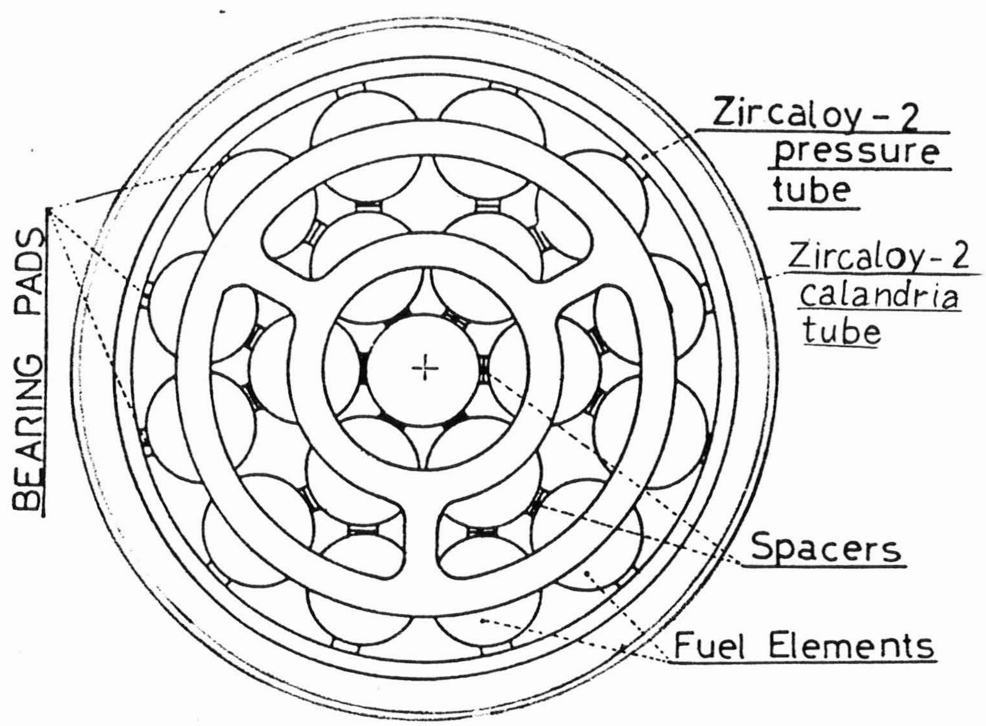


Figure 3

ALLOWABLE LIMITS FOR EMERGENCY AND FAULTED CONDITIONS

Level C Service Limits	Level D Service Limits
$P_m \leq 1.5\ S_m$	$P_m \leq 2.0\ S_m$
$P_L \leq 2.25\ S_m$	$P_L \leq 3.0\ S_m$

Figure 4

VARIATION vs. TEMPERATURE OF LONGITUDINAL AND CIRCUMFERENTIAL PROPERTIES OF ZIRCALOY-2 MATERIAL

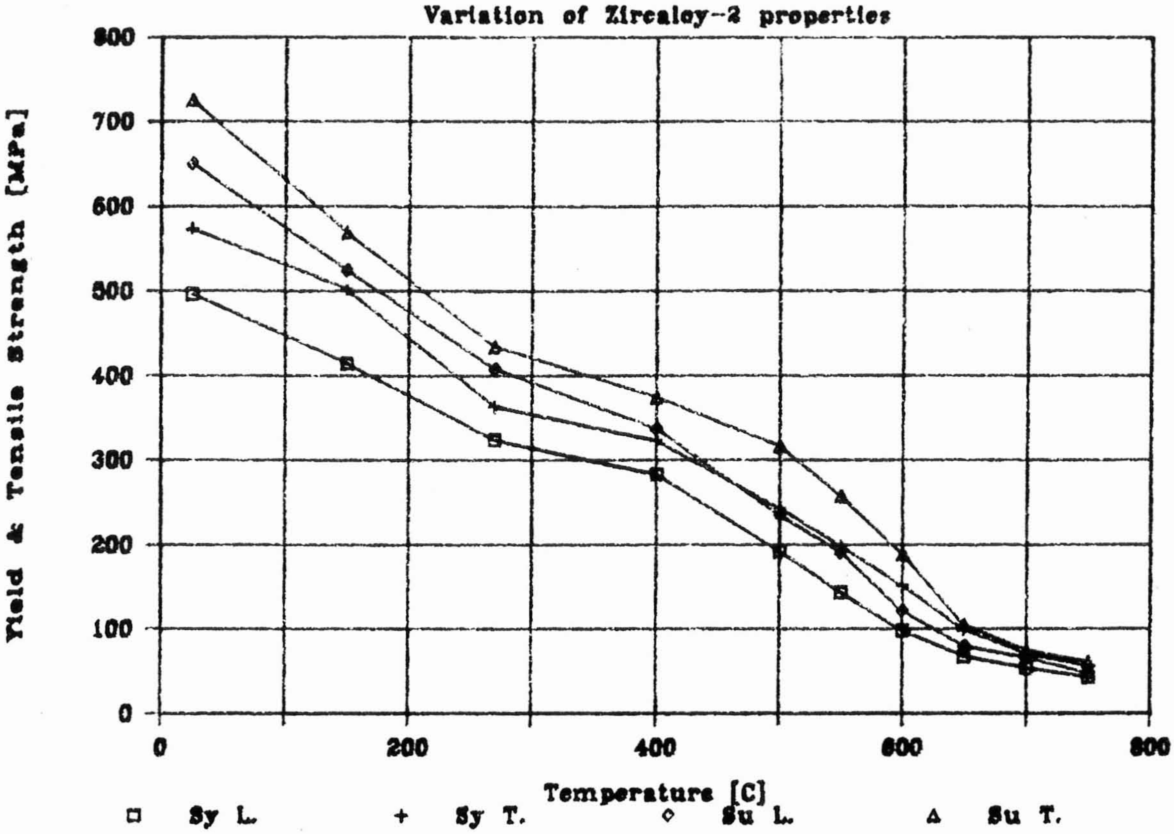


Figure 5

TYPICAL TEMPERATURE AND PRESSURE vs. TIME VARIATIONS DUE TO A PARTIAL FEEDER BREAK

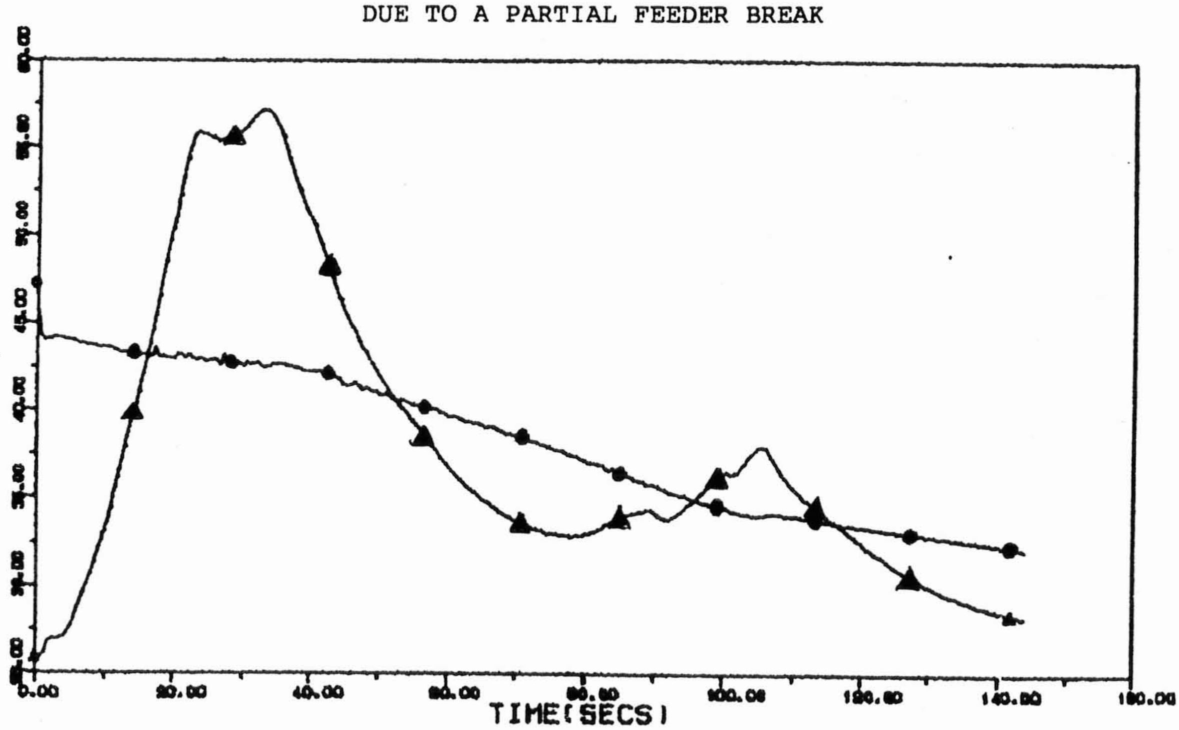


Figure 6

INNER RADIUS VARIATION vs. TIME

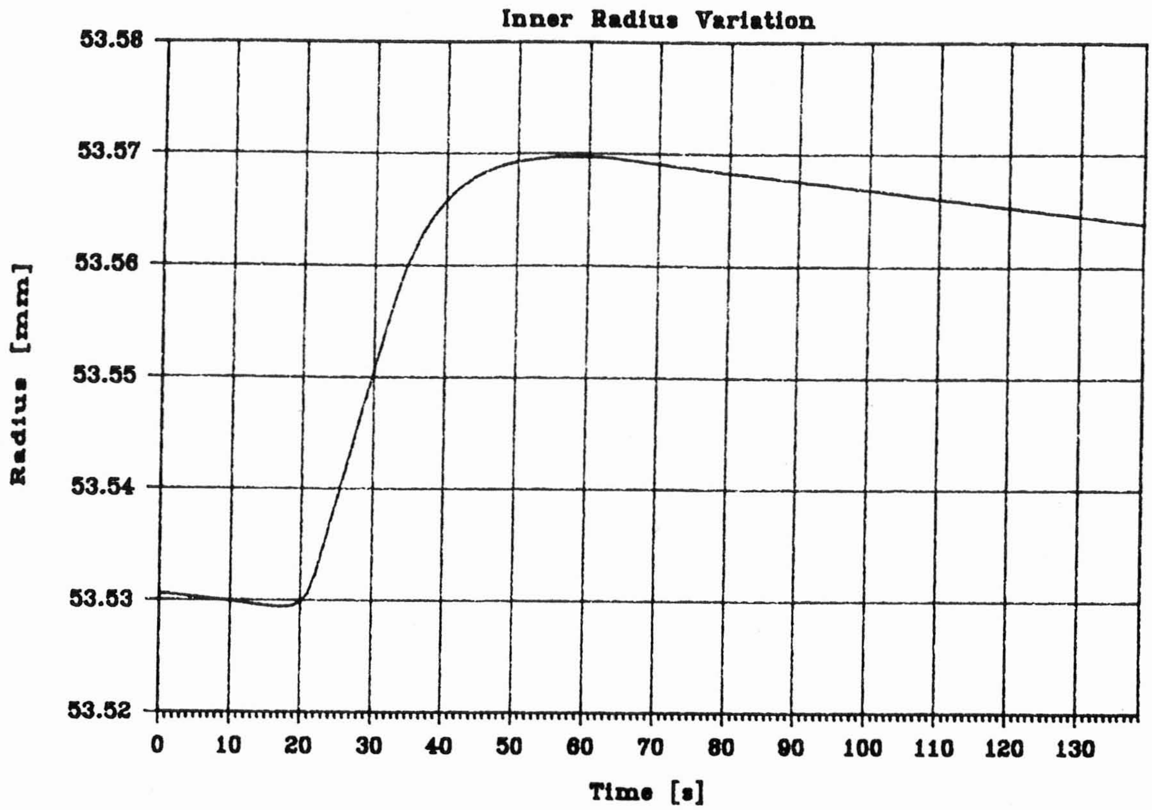


Figure 7

THICKNESS VARIATION vs. TIME

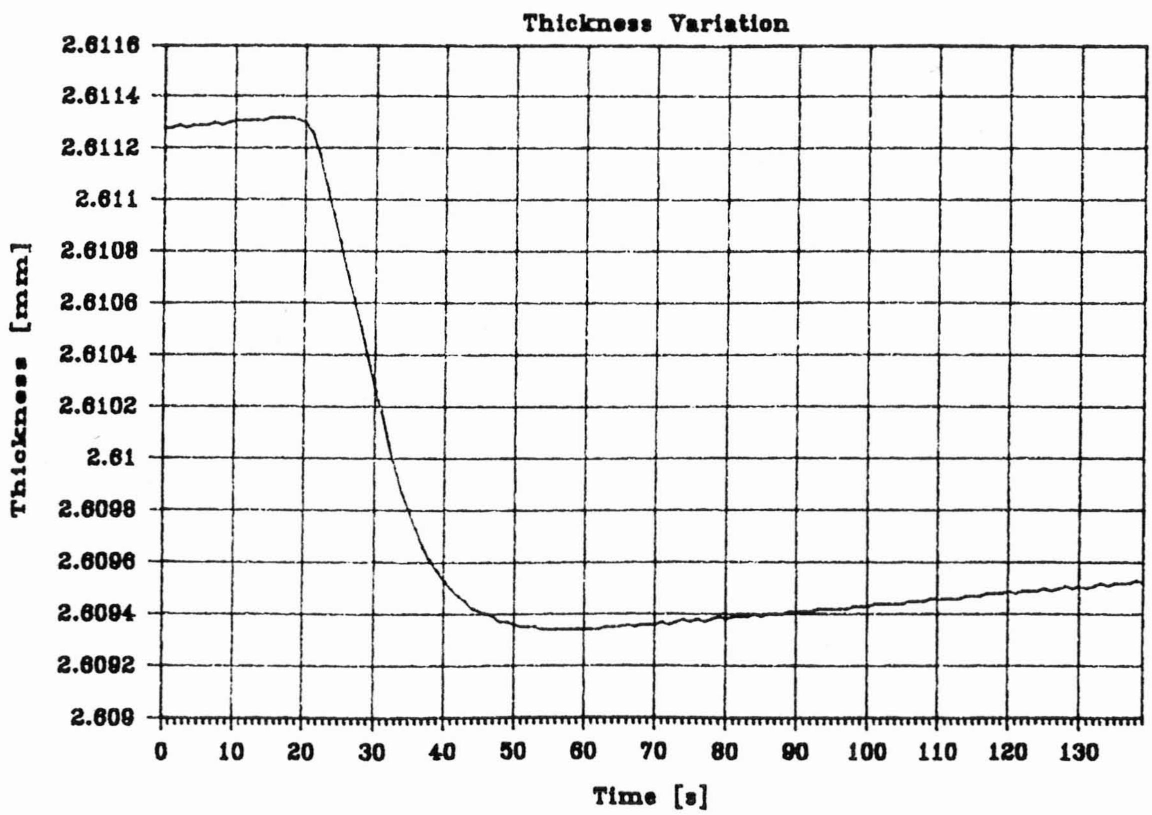
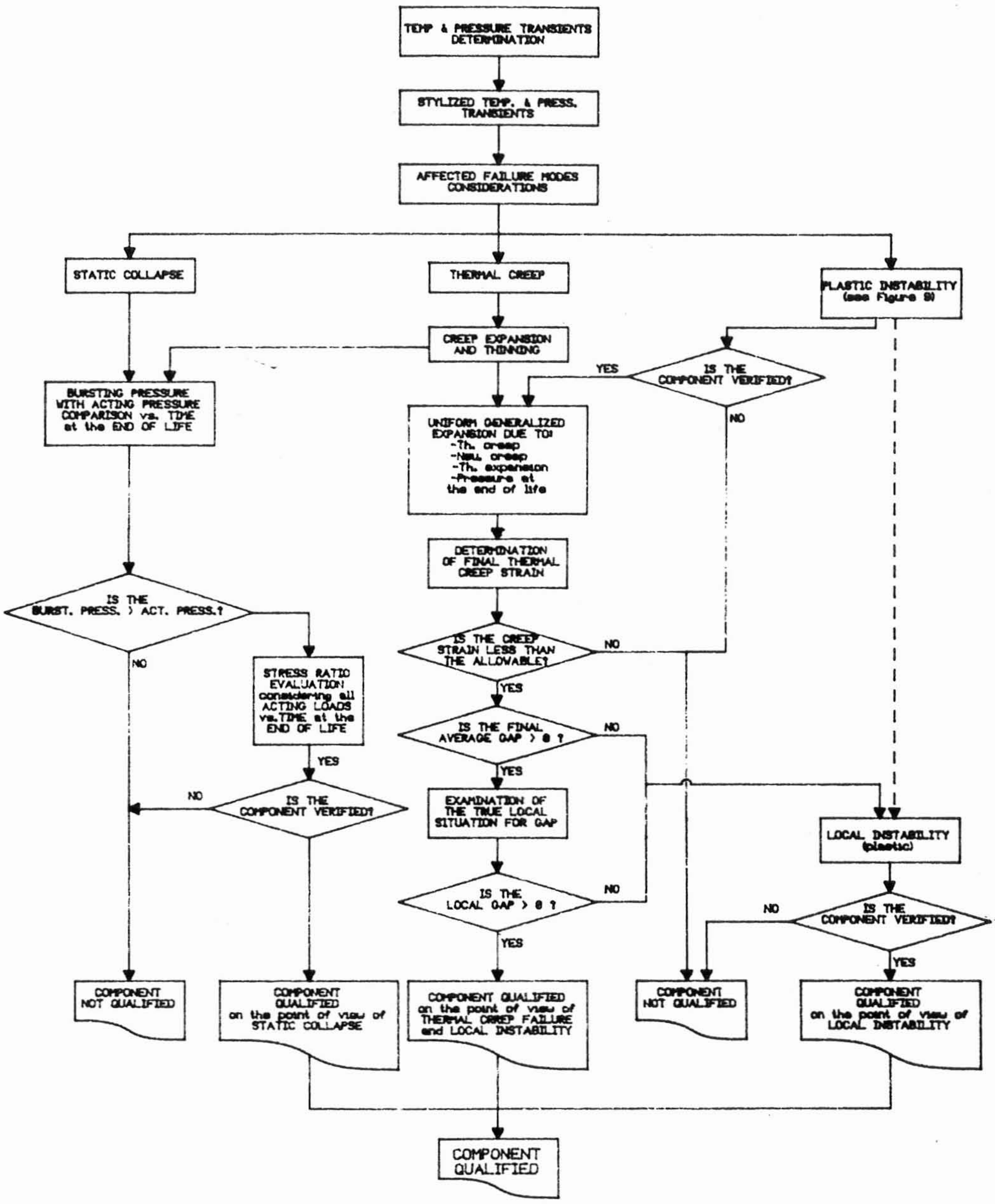


Figure 8
HIGH TEMPERATURE ANALYSIS FOR H.W.R. PRESSURE TUBES
METHODOLOGY FLOW-CHART



Dot lines indicates non-mandatory analysis

Figure 9

ALLOWABLE STATE OF STRESS CURVES FOR PLASTIC INSTABILITY ANALYSIS

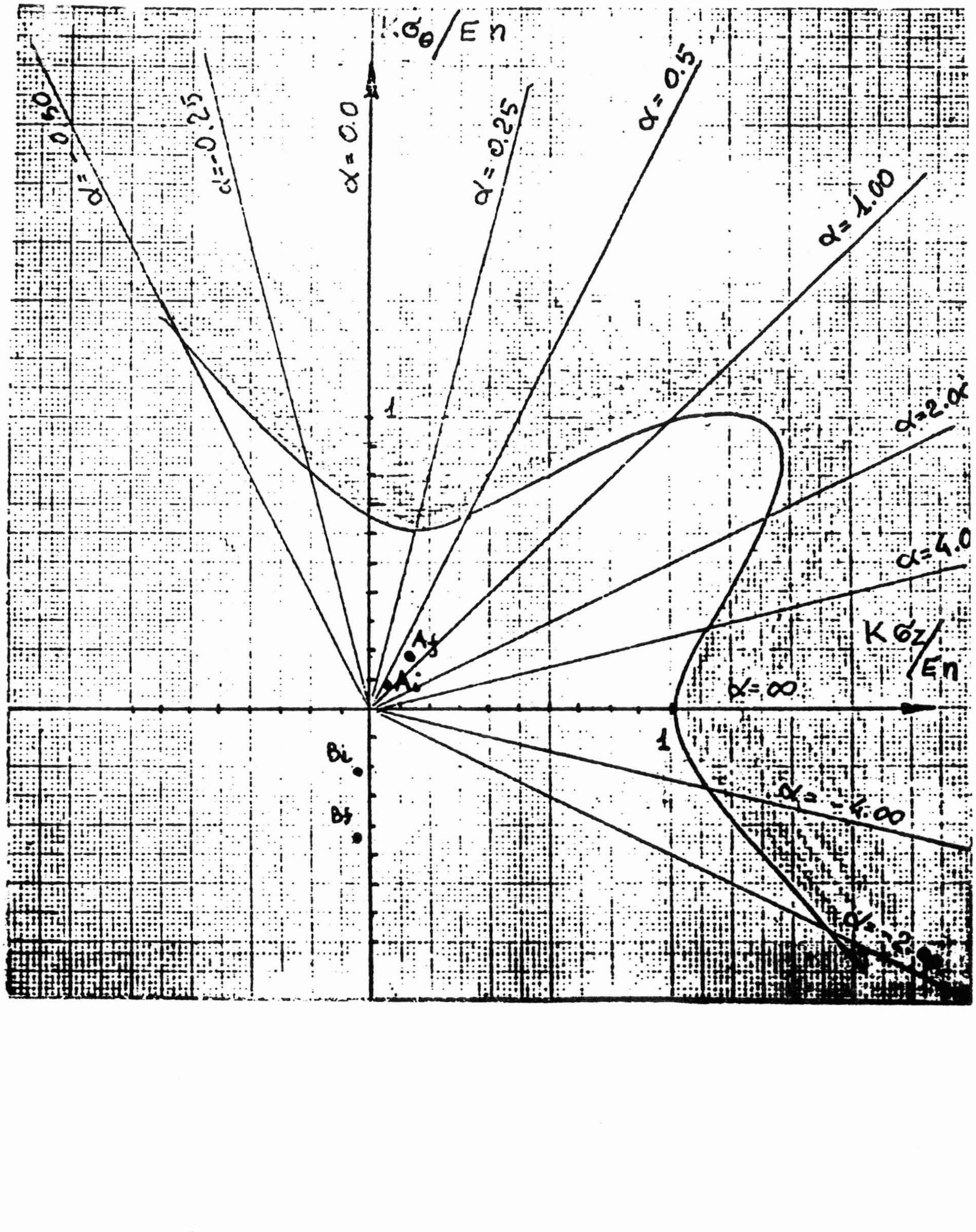


Figure 10

STYLIZED TRANSIENTS

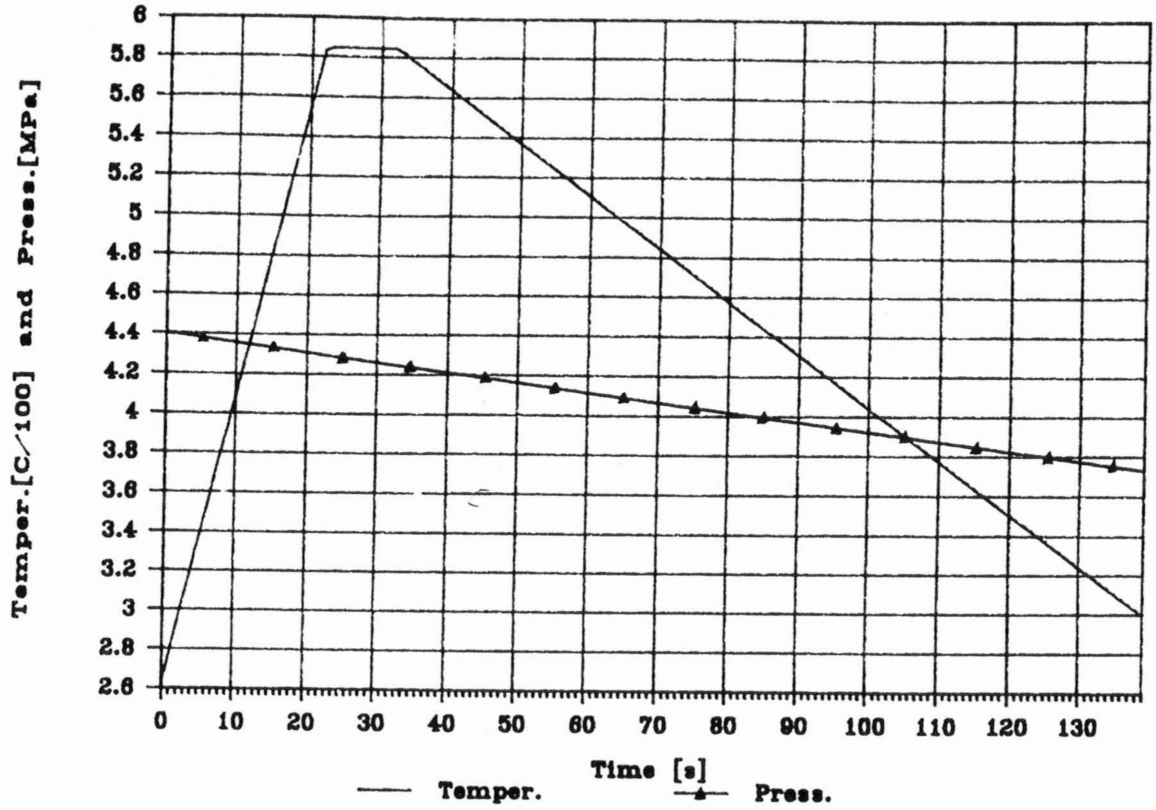


Figure 11

BURSTING PRESSURE VARIATION vs. TIME DURING THE TRANSIENT

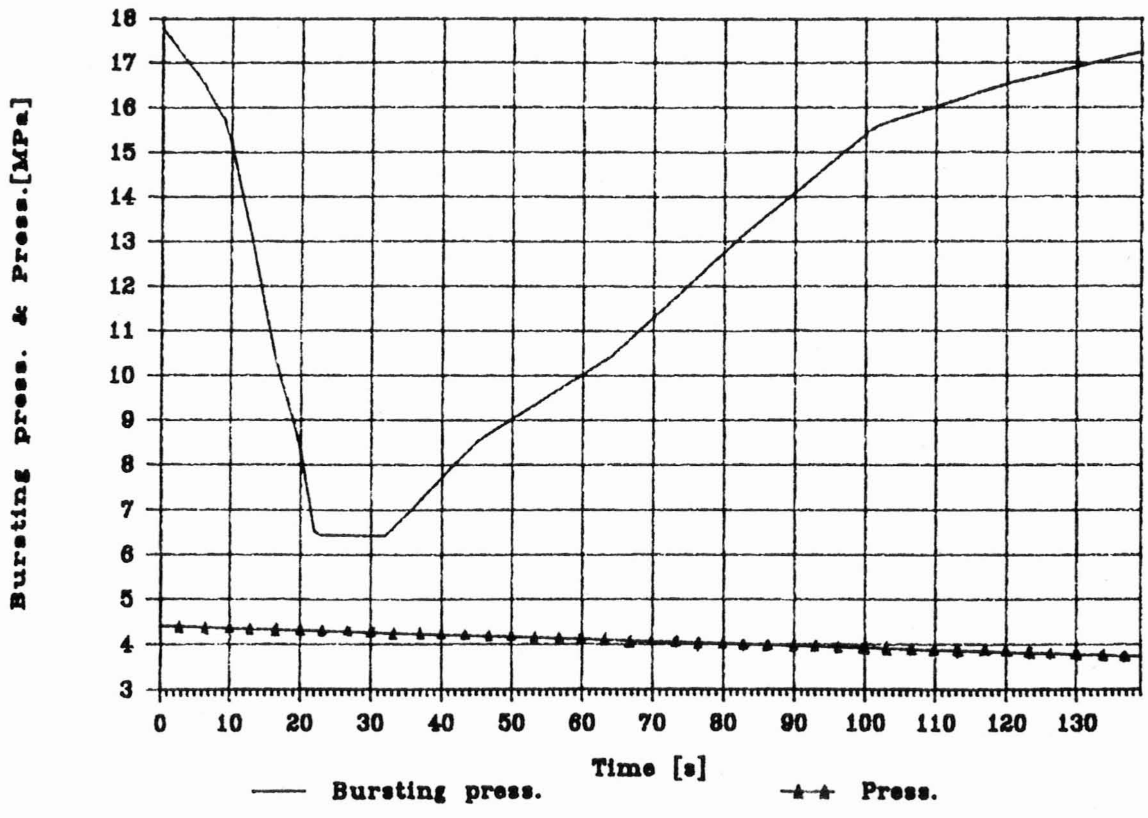


Figure 12

STRESS RATIO VARIATION vs. TIME DURING THE TRANSIENT

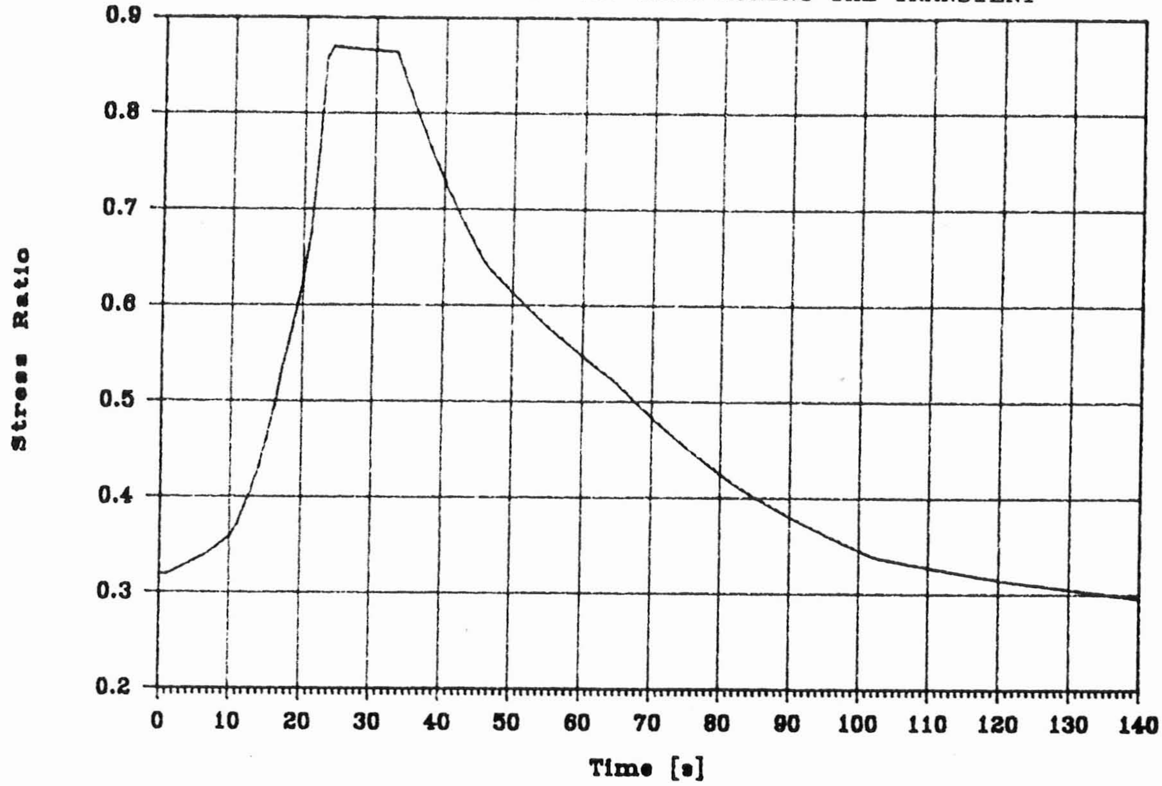


Figure 13

CIRCUMFERENTIAL STRAIN VARIATION vs. TIME DURING THE TRANSIENT

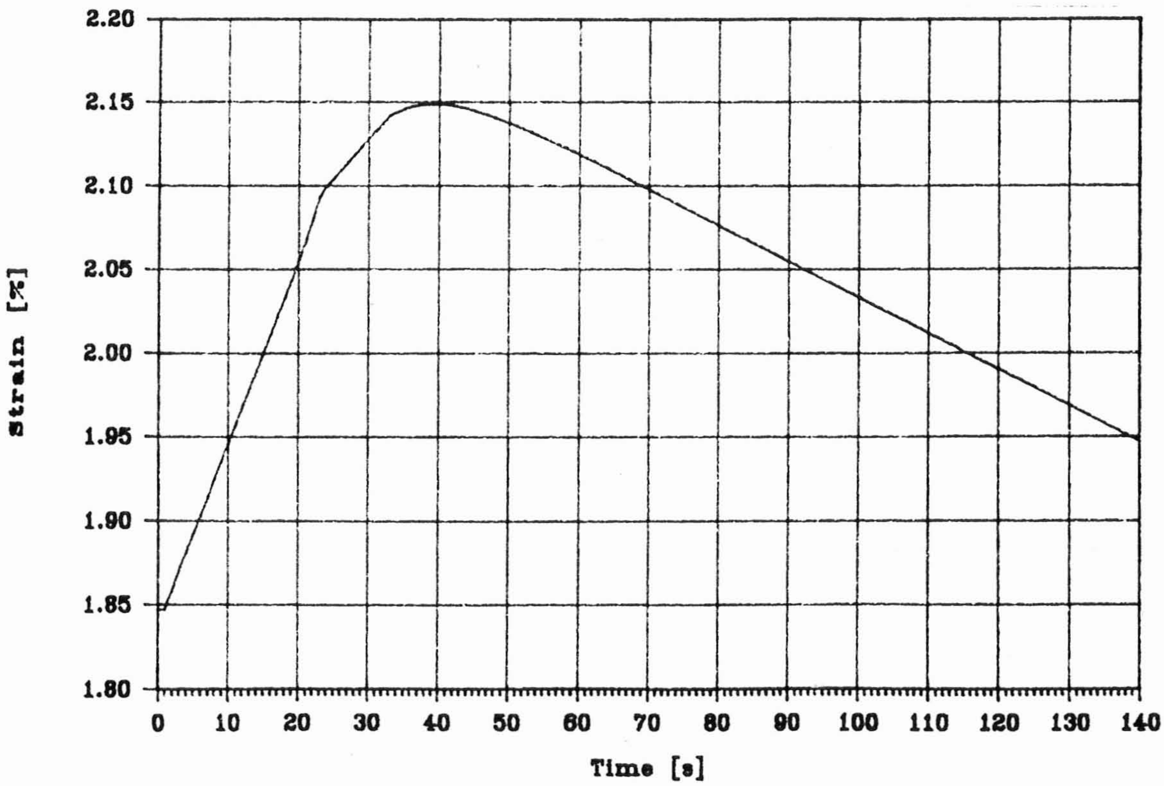


Figure 14

MINIMUM GAP VARIATION vs. TIME DURING THE TRANSIENT