SECONDARY CYCLE DESIGN CONSIDERATIONS FOR REDUCTION OF REACTOR TRANSIENTS FREQUENCY

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1. INTRODUCTION

For many years, the attention of those concerned with nuclear power plant safety has been concentrated on systems and components required for the safe shutdown of the plant, specially under severe accident conditions such as a reactor coolant pipe rupture. As a consequence, a large number of standards, regulations and guides have been issued covering the so called nuclear safety class systems and components while the non-nuclear safety items were left to the scope of conventional power industry standards and practices, where economy and availability have been the primary concerns.

Most of the secondary cycle systems and components of pressurized water reactor (PWR) plants are considered to be not essential to the reactor safety and are, therefore, classified as non-nuclear safety class /1/. However, since the secondary cycle function is to utilize heat from the reactor coolant during normal operation to generate power and to dissipate this heat during initial phase of plant shutdown and plant transients, any major disturbance in its operation will produce heat imbalances which cause transients in the reactor that might lead to scram, according to the degree of severity.

The importance of these facts could be recognized since the WASH-1400 /2/, the classic risk analysis study, defined the loss of coolant accident(LOCA) and transients as the two broad types of situations with a potential of leading to a reactor core melt. The term "transient " in WASH-1400 refers to any one of the plant conditions that would require the reactor to be shutdown. But it was the Three Mile Island accident that showed in a dramatic way the importance of non-nuclear safety related components and the sequence of non-critical events to the safety of the plant as well as the importance of the ability of the reactor control and protection systems to shutdown the reactor with undue radiological risk to the health and safety of the public. This fact has shifted part of the attention of safety and risk analysis from the most severe accidents with low probability of occurrence and from the safety related systems designed to mitigate the consequences of such accidents, into smaller accidents with higher probability of occurrence and into non-safety related systems whose malfunctions could lead to these accidents.

If, besides the classical single failure criteria /3/, other risk assessment criteria such as common mode failure (CMF) and anticipated transients without scram (ATWS) are adopted /4/, the importance of these smaller events becomes more evident. Both the WASH-1400 and the German Risk Study Summary /5/ had rated the small reactor coolant pipe break accident as the one of having the highest probability of causing a reactor core melt, among the anticipated accidents analyzed. The non-closure of the pressurizer power operated relief valve of TMI Unit 2 has the same effect as a small reactor coolant pipe break accident. It should be remarked that the type of electromagnetic operated relief valve that failed at TMI is rated as having a 2% probability of sticking in the open position per operation and this problem had been consistently overlooked until that accident, because it was classified as a non-nuclear related item /4/.

This paper calls the attention to the importance of secondary cycle design and operation to the safe operation of PWR nuclear power plants.

2. PLANT TRANSIENTS

The anticipated plant transients considered relevant to the plant safety are required to be includedin the safety analysis reports. These transients are classified according to ANS-51.1 in four categories depending on the expected frequency of occurence:

. Condition I - normal operation
. Condition II - incidents of moderate frequency
. Condition III - infrequent incidents
. Condition IV - limiting faults

Many of the anticipated plant transients in a PWR plant are caused by secondary cycle component failure and/or system malfunction /1,6,7,8/, such as

. loss of main feedwater
. loss of condenser cooling
. turbine trip
. loss of external load
. station blackout
. excessive heat removal due to feedwater system malfunction

. steam generator tube leaks
. main steam line break
. feedwater line break
. single failure of a control component.

Although the reactor control and protection systems and the containment are designed, analyzed and tested to handle these plant transients with a high degree of reliability, maintaining the radioactivity releases to the environment and to the plant operators within the limits prescribed by the codes (as low as is reasonably achievable) /9/, an increase of transients challenging the safety systems obviously will increase the risk of a safety system failure /10/.

The reduction of frequency of reactor transients caused by secondary cycle disturbance can be achieved by improvement of systems design, increase in component reliability, improvement of maintenance procedures and plant operator training.

Both system design and component reliability can be improved by establishing intermediate grades between the presently required for safety and non-safety systems as suggested by the NRC /3/ and by analyzing carefully the operating experience of hundreds of plant years of PWR commercial plants available to the public via documents such as Licensee Event Reports (LER). One lesson learned from analysis of these reports is that valves are more responsible for safety related events than any other plant component /12/.

3 SECONDARY CYCLE DESIGN CONSIDERATIONS

As discussed in the previous section, improvement in pressurized water reactor safety can result from upgrading of secondary cycle systems and components design.

Some considerations will be made below for two specific areas:

. system design and component performance
. special aspects of mechanical design of components.

3.1
System design and component performance

In the attempt of designing secondary cycle systems of nuclear power plants, the analyst should be more concerned in preventing disturbances than for a conventional power plant. Computer simulation techniques widely used to analyze the behaviour of safety related systems should be extended to the most important secondary side systems to evaluate their designs and component performance under upset conditions. The results of this simulation can show plant safety and availability could be improved. A good example of this technique is show in the work of Rovnak et al. /11/ where the behaviour of the condensate, feedwater and heater drain pumps of Millstone Unit 2 was analyzed to investigate the effect of condensate or heater drain pump trip on the operation of the feedwater pumps.

A brief discussion will follow for the main systems of the secondary side focusing on some specific points.

3.1.1
Condenser cooling system

One critical problem in condenser cooling system of nuclear power plants is condenser tube leakage, specially when seawater is the cooling medium, since the contaminants of the cooling water can cause severe corrosion problems in the steam generator tubes. Besides the fact that condenser tube leaks have been one of the major causes of plant unavailability in the USA /10/, the consequent steam generator tube failure is very serious with respect to plant safety since it is a case of LOCA.

Titanium condenser tubes have been successfully used for seawater applications /13/, showing an excellent performance with respect to corrosion by seawater. The condensate polisher units have also proven to be a good way to decrease steam generator tube failure rate due to the fact that they remove the seawater leakage impurities within a limited leakage rate. With the removal of these leakage impurities the steam generator tube denting can be avoided since the main cause of the phenomenon is the reaction of chlorides of seawater with copper of the secondary side and carbon steel of the steam generator tube sheets.

Another problem that should not be overlooked is the potential of corrosion existing at the steam side of the condenser air removal section in plants that use All Volatile Treatment (AVT) for condensate and feedwater chemistry control /14/. This problem can become serious, specially when high copper alloys are used as condenser tube materials.

With respect to system operation, it is important to analyse the system transients to verify if the pressure surges due to sudden pump stoppage will not cause pressures exceeding the design limits. Since the condenser cooling system normallly operates under syphon with the condenser at the highest circuit elevation, the pressure downsurges can cause the water column separation and the subsequent column reattachment which generates damaging pressure spikes that can rupture the condenser /15/.

Considering also that the modern nuclear units are designed with rated electrical power exceeding 1000 MWe and that a nuclear unit has a condenser cooling load approximately 60% higher than a fossil fuel plant of equal power output /16/, the cooling water flow required by the nuclear unit is much larger than the one required by conventional power plant.

Therefore, the condenser cooling system transients should be more closely analyzed in nuclear plants, paying attention also to problems associated with the sensitivityofpropeller type pump to head fluctuations. It must be born in mind that a condenser cooling system failure, and the associated loss of condenser vacuum, result in a turbine trip with the steam dumpsystem blocked, thus leaving the main steam safety and relief valves as the only means of handling the steam produced in the steam generators during reactor shutdown.

3.12
Condensate and Main Feedwater Systems

The analysis of seventeen years of PWR plant operating experience in the USA /10/ has shown that the major causes of reactor scram are failures in the condensate and main feedwater systems; they cause an average of 2.75 scrams per plant year. This is almost twice as much as the second major source of reactor scram which is the turbine generator system failure with 1.50 scrams per plant year. It is to be noted that the 2.75 scrams amount to 38% of the average 7.24 total scrams per plant year.
According to the German Risk Study Summary, failure of main feedwater supply has a probability of causing a core melt (3 x 10^{-6} per reactor year) comparable to medium reactor coolant pipe break (2 x 10^{-6} per reactor year).

An analysis of Westinghouse reactors subjected to a complete loss of main feedwater accident without reactor scram, which is considered to be the accident which causes the maximum overpressure in the primary system, is presented by Rose and cooper /17/.

Therefore , one sure way to improve reactor safety is to decrease the frequency of failures in the main feedwater supply.

Potential sources of problems in the main feedwater system are the instabilities that occur during transients caused by load changes, pump trips, control valve dynamic instability and piping failure.

Fast load changes cause level oscillation in the heater drain tank which can result in variations of the heater drain pump flow to the feedwater pump. A trip of one condensate or heater drain pump results in a momentary decrease in the pressure at the suction of the feedwater pumps /11/ which could cause the pump to cavitate before the standby condensate or heater drain pump is brought into normal operation. To cope with this situation and avoid frequent feedwater pump trip, a delay should be foreseen in the feedwater pump low suction pressure trip control. Also, the feedwater pump specification should include the requirement of the pumps being able to operate under cavitating conditions, during a short period of time, without damage. Automatic controls to put the standby pump into service are desirable to reduce to length of time that the feedwater pumps would operate under cavitating conditions. Standby pumps are convenient means of decreasing the severity of transients due to pump failure. The selection of the condensate pump discharge pressure, in system configurations as shown in Fig.1, also deserves a careful analysis. If high pressures are specified to increase the margin for cavitation at the feedwater pump suction the design pressure for the condensate system downstream of the pumps will be correspondingly high which increases the equipment cost. On the other hand, if a lower pressure, and consequently lower margin for the feedwater pump suction cavitation, is selected for the condensate pump discharge to decrease the condensate system equipment cost, the result will be a condensate-feedwater train that is more susceptible of being subjected to frequent feedwater pump trips and more demanding to the reactor protection systems. Into this line of analysis , it must not be forgotten the condensate and feedwater pump requirements during steam dump conditions, when the heater drain pumps trip and the condensate pumps alone must supply the feedwater pumps with the required flow to keep are steam generator level above the low-low value that causes reactor scram.

The considerations of the above paragraphs refer to "solid" condensate and feedwater systems, i.e., those systems without a feedwater tank between the condensate and feedwater pumps.

Fig.1 shows the simplified condensate and feedwater systems for Angra Unit 1 and Millstone Unit 2. It can be seen that while in Angra Unit 1 spare pumps are provided for condensate , feedwater and heater drain pumps (3 x 50% capacity pumps, one standby), in Millstone Unit 2 only the condensate pumps have spare. In this last case, the failure of a heater drain pump is compensated by the startup of the standby condensate pump.

Feedwater control valve instabilities at low flows have been observed to cause serious transients /18/ with consequent reactor shutdown due to steam generator level fluctuations, reaching low-low level , or steam flow-feedwater flow mismatch. This type of problem can be overcome by a proper selection of feedwater control valve trim or by the use of a smaller size bypass control valve that takes over the feedwater flow control at low flow conditions.

Besides the possibility of forcing a reactor scram, these feedwater system transients are accompanied by pressure pulses (water hammer) producing strong pipe vibrations that can cause pipe failure if they are not taken into consideration in the piping analysis and support design.

Piping and other equipment failures add to the above list of causes of condensate and feedwater system malfunctions. This type of failure can be either a consequence of an upset condition in the system, such as a water hammer, or of a design deficiency resulting in excessive vibration, erosion, corrosion, etc. An example of this type of problem is the event sequence that happened in the North Anna Unit 1 due to feedwater heater tube leakage /19/. The subject of piping and equipment failure will be further discussed in section 3.2 herein.

3.1.3
Main Steam System

Since the main steam system operates at the highest temperature and at one of the highest pressure levels of the secondary side, one could be led to believe that it would be the main cause of reactor scrams, However, probably due to its relative simplicity and small quantity of active components, it is rated below many other secondary side systems in causing reactor scrams. Nevertheless, with respect to plant unavailability the main steam system is second only to the turbine generator system, for american plants /10/.

downstream of the main steam isolation valves and is, therefore,a non-nuclear safety system.

The most severe reactor transients caused by main steam system malfunctions are due to primary coolant excessive heat removal resulting from abnormally high steam releases that can be caused by a steam dump, safety or relief valve sticking in the open position or by an actual line rupture. The steam dump , safety and relief valves are specified to limit the maximum flow through one stuck open valve to a value that is safe for the reactor operation,even for the highest expected main steam header pressure. An accident that resulted in a stuck open steam dump valve, and its consequences , is discussed by Verna /19/.

The consequences of a large main steam pipe break are much more severe than for a stuck open valve and could be damaging if steps are not taken in the design to protect the adjacent equipment from pipe whip, jet impingement and compartment pressurization effects.

3.1.4
Other systems

Other secondary side systems such as feedwater heater drains, instrument air, fire protection, component cooling, extraction steam, etc., also play important role in the normal operation of the plant with some degree of relationship with reactor safety.

The turbine generator system has a very intimate connection with the reactor control system since the latter is always trying to match the reactor power output with the turbine load demand. The system is also responsible for most of the outage time in USA plants but, as this system design is almost totaly in the manufacturer's scope, it will not be discussed here.

The emergency feedwater system major function is to provide an alternate and reliable source of feedwater to the steam generator in emergency conditions. During normal power operation this system is inactive and therefore an analysis of its effect as a source of reactor transients is beyond the scope of this paper.

The safety related portion of the main steam system is separated from the non-safety related one by main steam isolation valves, located just downstream of the containment penetrations. The main steam isolation valve main function is to limit the uncontrolled flow from the steam generators caused by an eventual pipe rupture, inside or outside the containment, thus protecting the containment against overpressure and maintaining the reactor heat sink. The valve and piping arrangement should be such that the uncontrolled flow of more than one steam generator is prevented by valve closure . The fail safe position of these valves is closed and, therefore, redundant control and power channels should be provided to decrease the probability of reactor transients due to accidental valve closure caused by power supply failure or spurious control malfunctions.

The main steam system upstream of the isolation valves, including the steam generators, is protected against overpressure by safety valves. Power operated relief valves can be provided with lower set pressure to assist the safety valves during main steam system transients. The design pressure of the main steam system should be selected taking into account these transients and avoiding frequent opening of the self actuated safety valves. The safety and power operated relief valves, besides protecting the steam generators from overpressures, provide also means for removing heat from the reactor coolant system when the condenser is not available.

Since the nuclear steam supply system (NSSS) cannot take large steps of load changes, a steam dump (turbine bypass) system is provided. The steam dump system operation is very important to decrease the frequency of reactor transients by absorbing the excess load when reactor-turbine power mismatch occurs due to large external load reductions. The maximum flow of main steam that can be dumped directly to the condenser or to the atmosphere varies from plant to plant and the size of external load reduction step that can be accomodated without reactor trip varies accordingly. For the Unit 1 of Angra, the steam dump system is designed to discharge up to 85% of the full power flow into the condenser and therefore the plant can take an external load rejection from 100% power to the plant auxiliaries load without reactor trip or steam release to the atmosphere. This is possible due to the fact that the NSSS can take a 10% step of load reduction and the plant auxiliaries use approximately 5% of the turbogenerator output at full power. The steam dump system and/or power operated relief valves are also used during plant startup, hot standby, shutdown and plant tests to relieve the small amount of steam produced by the steam generators. The steam dump system is located

3.2
Special aspects of mechanical design of components

Other means of decreasing the probability of system failure is the improvement of component reliability. The design of nuclear safety components is subjected to a large number of codes and regulations requiring a sophisticated analysis both for loading conditions and acceptance criteria. In general, however, most of the phenomena causing the loading and failure modes in the nuclear safety components are also present in non-nuclear components.

The difficulties involved in the analysis of several important phenomena tend to hinder a detailed study of them. An often arbitrary increase in the safety factors substitutes a more careful analysis. This procedure not always leads to a conservative approach.

On the other hand the scientific and technologycal achievements can not be disregarded or reserved for components that presently are considered as critical. As it was pointed out above, the safety class of non-nuclear components, as far as they can be the origin of a sequence of events that would eventually cause a reactor core melt, is very questionable.

In the sequel some important phenomena which are many times disregarded in the design of non-safety related equipment will be commented. Also, these statements should not be construed to imply that these tecniques should be indiscriminately used in the design of secondary side components.

3.2.1
Hydrodynamic - structure interaction

Most of the reported piping failures in PWR plants are atributed to vibration /21/, which are partly caused by rotating equipment and partly caused by the fluid flow. Two points will be focused in this section, namely the flow induced vibration and the effect of pulse propagation due to short time perturbations in the fluid flow such as the closure of stop valves.

It is well known that under certain circunstances, the flow of a fluid through a pipe can lead to dynamic instability or flutter. This phenomenon can be observed when the flow velocity through the nozzle of a hose, left free to move, reaches sufficiently high values. A large number of theoretical and experimental papers have been published recently, most of them dealing with straightpipes /22 - 25/. The linearized governing equation can be obtained from the principle of virtual work to give /23/:

$$EIY^{IV} + (p_t + p_f)Y'' + p_f(2V_oY' + V_o^2Y'') = 0 \tag{1}$$

The dynamic coupling between the fluid flow and the pipe displacement is determined by the last terms in equation (1). Clearly $2p_fV_oY'$ and $V_o^2Y''p_f$ represent respectively the Coriolis and centrifugal accelerations. The stability analysis of equation (1) can be performed through the behavior of the associated eigenvalues, obtained with solutions of the type:

$$Y = \phi(x)\, e^{\lambda t} \tag{2}$$

The stability chart is shown in Fig.2 for the case of a horizontal simple supported run of pipe. From this chart it is apparent that above a certain critical velocity the pipe presents unstable behavior. This instability is determined by the sign of the real part of λ. When the real part of λ is positive the solution is unstable (buckling or flutter).

Although it is true that in general, for straight pipes, the critical velocities can reach very high values, which are not likely to occur in normal systems, it is possible that for very flexible spatial configurations, the critical velocities could have lower values. Unfortunately there is no stability analysis reported for spatial systems. Field tests have confirmed that for some systems large amplitude vibrations occur, caused apparently by the fluid flow. In these cases, rearrangement of support frames or the introduction of additional supports conveniently located can minimize the problem. In the case of systems which are not accessible, or for which the flow velocity under abnormal conditions reaches values above the velocities used during the pre-operational tests, a careful analysis must be performed.

Another important coupling between the flow and pipe occurs when a pressure pulse moves along the fluid. This is the case when a stop valve closes almost instantaneously. The closing time for turbine stop valves, for instance, can be as short as 0.15 sec. The pressure pulse travels down the pipe generating in elbows and area reductions forcing functions of very short durations. The analysis is carried out in two steps. First the thermohydraulic analysis must be performed in order to obtain the time history of the pressure at some critical sections. With this data set a dynamic analysis of the piping system must be performed. Both, for the thermohydraulic analysis and structural dynamic analysis, there are special computer codes available. Recently it has been developed at the

Argonne National Laboratory, a sophisticated code that takes into account the shell behavior (ICEPEL) including plastic deformation /26/. Even for the simplified analysis of the pipe as an one-dimensional structure, the effect of these short time pulses, like steam hammer or water hammer, is significant. The loads on thesupports, due to the closure of a turbine stop valve in a 30" diameter steam line can reach values as high as 30 tons. A typical history of the force at a section of steam line due to a steam hammer is shown in Fig.3.

3.2.2
Pipe whip

Equipment and piping protection requirements impose a careful analysis of possible location for pipe rupture. Portions of the main steam and feedwater lines are located inside the containment vessel and other nuclear island buildings. Since these are high energy and large size lines they must be analyzed with respect to the consequences that their rupture can cause in the equipment nearby, besides the transients in the feedwater or main steam systems themselves. This analysis is an interdisciplinary task involving stress analysis, elastic-plastic fatigue considerations, crack propagation and thermohydraulic analysis. Presently, there is no sufficient information to set up a reliable systematic procedure in a standard form. Therefore, conservative criteria have to be used to postulate sections in a piping system which will be considered potentially as the most favorable for rupture to occur /27/.

It follows that after rupture the surrounding equipment and pipes must be protected from serious damage. The most critical case associated with pipe rupture is the guillotine type of rupture. It is necessary to design special supports toabsorb the kinetic energy induced in the pipe by the resulting jet force (Fig.4). Several models have been developed to represent the elastic-plastic dynamic response of the pipe and restraint. The main scope of this analysis is to design a proper support at a convenient location such that most of the kinetic energy can be dissipated through plastic deformation of the restraint.

Silva and Bevilacqua /28/ have shown that very simplified models, such as rigid-plastic beams, or a mass-spring model can lead to non-conservative results, if an appropriate correction factor is not used. Comparison of several models /28-31/ has shown no substantial difference in the results. In the work of Guerreiro et al./29/ the influence of the internal pressure

has been taken into account in the elastic-plastic law. An approach using directly the moment-curvature constitutive relation was developed by Maneschy /31/.
Fig.5 through 7 show some results where the influence of the restraint rigidity, the gap value and force intensity are depicted.

One of the most important conclusions in this analysis is that the position of the supportplays an important role. The energy absorbed by the restraint is much more affected by its position relatively to the point of application of the blowdown force than by its stress -strain characteristics.

3.2.3
Thermal stresses and strains

The inelastic behavior of pipe, pipe components and equipment associated with thermal cycle has been growing in importance since the compromise between economy and safety was sharpened in the design of nuclear power plants. This effect is particularly important in components subjected to fast temperature transients and where dissimilar material welds (austenitic-ferritic) are present. One question that has recently been deserving the attention of design engineers is the ratcheting occuring in pipes and components when submitted to thermal cycles. Essentially the thermal ratcheting is the accumulation of plastic strain during successive thermal cycles. Consider for instance the simple truss shown in Fig.8.
Assume that the load P is such that the bar 1 is loaded beyond its yield point. The bars are considered as elastic-plastic. Now if the temperature of the bar 1 is subjected to a thermal cycle ranging from T_0 to T_1, while the bars 2 remain at a constant temperature, these will eventually yield and after the first cycle the vertical displacement of P will increase of δ_p. Fig.8 shows the evolution of this situation and it is easily seen that the structure does not shake down.

For pipes, valves and pressure vessels the temperature gradient across the wall, caused by thermal transients, can lead to ratcheting. Kalnins and Updike /32/ show Bree-diagrams for a bi-axial stress state for some σx/σy ratios. Fig.9 shows the case of equal bi-axial stresses.

The temperature through the wall thickness was assumed with a step distribution. It was shown that the Muller, Bree, Brugreen theory developed for calculating time independent thermal ratchet strains in a shell subjected to a steady equal bi-axial stress state of mechanical loading and a cyclic thermal gradient through the wall thickness may be used conservatively for determining the shake range of a pressure vessel subjected to unequal bi-axial stress state, provided that the stress is rep ced by the equivalent stress in the calculations.

A similar type of analysis has been performed for beams where the constitutive equation relates directly curvate vs. moment, rather than strain vs. stress. It was shown that also for this case thermal ratchet occurs.
A simplified analysis for thin shells, including the effect of thermal stress, is reported by Tonarelli and Ticozzi /33/. The ratcheting effect is investigated.

The ASME B&PV Code Section III 1979 Summer Addenda has included special requirements limiting stress intensities to a level that allows the pipe to shake down when submitted to a thermal cycle, for class 1 components.

It has been pointed out /28/ however that, even if the structure shakes down, the plastic deformation occurring during the first cycle can be excessively high. This can result in unacceptable distortions in the equipment.

Another potential cause of failure are residual stress in pipes and components due to welding or other fabrication processes.

Welding residual stresses in piping are of fundamental importance, since welding is the most common means of joinning piping. Also, it has been established that more than half of pipe failures occurs at the welds /21/.

The residual stresses are generated by temperature transients and temperature dependent material properties. Recently, analytical models have been developed to deal with this problem and experimental investigation carried out to provide relevant data to verify computational results /35,36/.

Although the mechanical models for simulations of residual stress in piping systems need further improvement, it is possible to anticipate with a reasonable degree of accuracy the circunferential and longitudinal stress distribution for one or two-pass welds. For a larger number of passes Fig. 10 shows the results reported by Rybibki and Stonesifer /35/. For these cases the symmetry assumption taken for the analytical model seems to introduce some margin of error.

3.2.4
Creep

High intensity loads acting permanently on structural members generate a continuous deformation process known as creep effect. This phenomenon is intensified when the structure is subjected to high temperatures.

Experiments in metallic materials /37/ have shown the existence of three stages (Fig.11).

It is clear that, under the action of an external load, the structure can undergo excessive strains after a certain time interval. The design life of the equipment plays then an important role in the design . In the work of Taroco and Feijoo /38/, one can find a very good description of the creep and viscoplastic behavior for homogeneous materials. The solution of a thick walled tube under internal pressure is shown in Fig.12 for the secondary creep, governed in accordance with the Odqvist constitutive law:

$$D= \frac{3}{2} K \, T_e^{n-1} \, S$$

where: D is the strain rate tensor, T_e is the effective stress tensor $T_e= (3/2\ S.S)^{1/2}$, S is the deviator stress tensor, K and n are material constants. It is seen that the stress distribution is substantially affected by the creep influence, as compared with the elastic solution
n= 1.

4
Summary

In this paper it has been pointed out that the secondary cycle systems should not be considered of secondary importance to the pressurized water reactor safety , and to suggest that some of the advanced design and analysis techniques, used for nuclear safety related components, be also used - with discretion - for secondary side non-nuclear components.
Improved system design and more reliable components will unquestionably decrease the frequency of challenges to the reactor protection and post trip heat removal systems and, consequently, will decrease the probability of accidental core melting.

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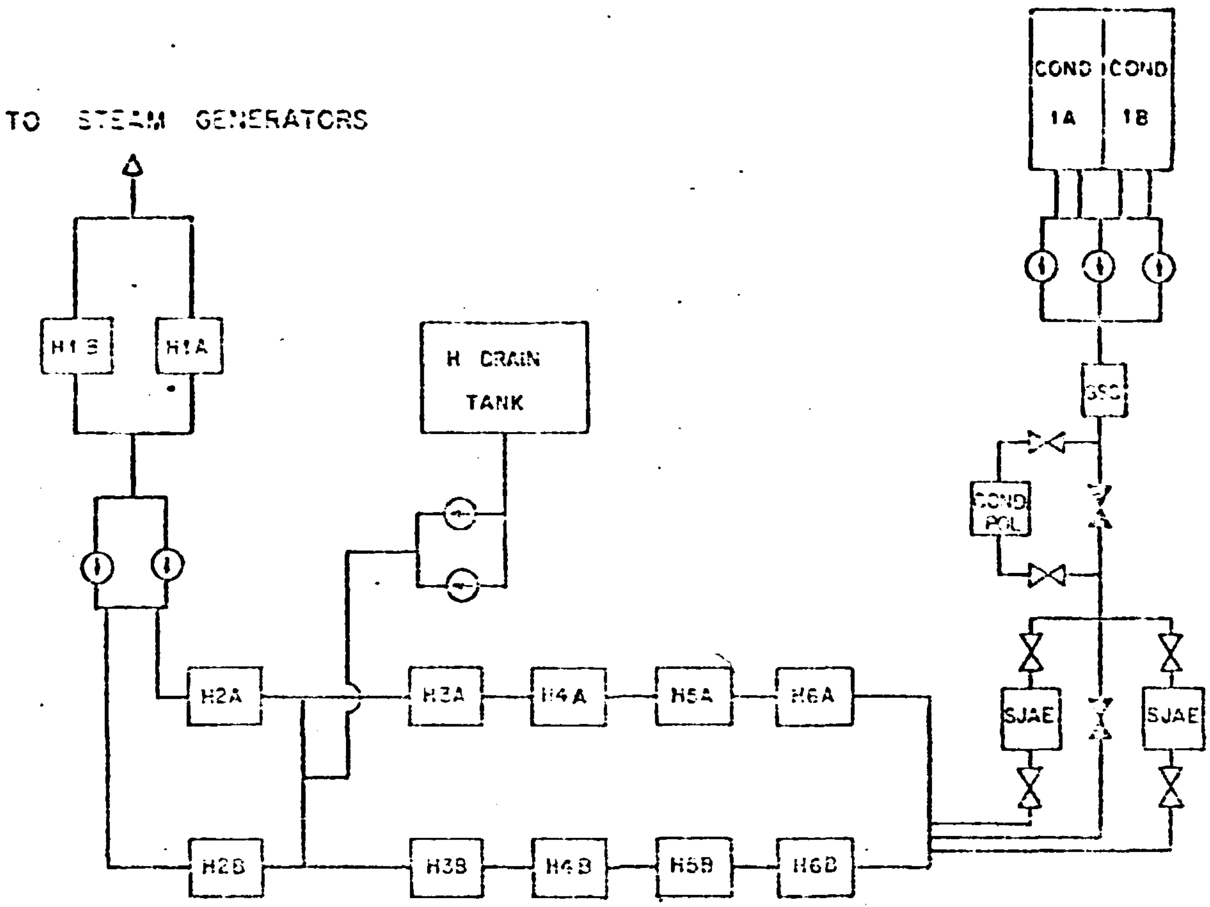
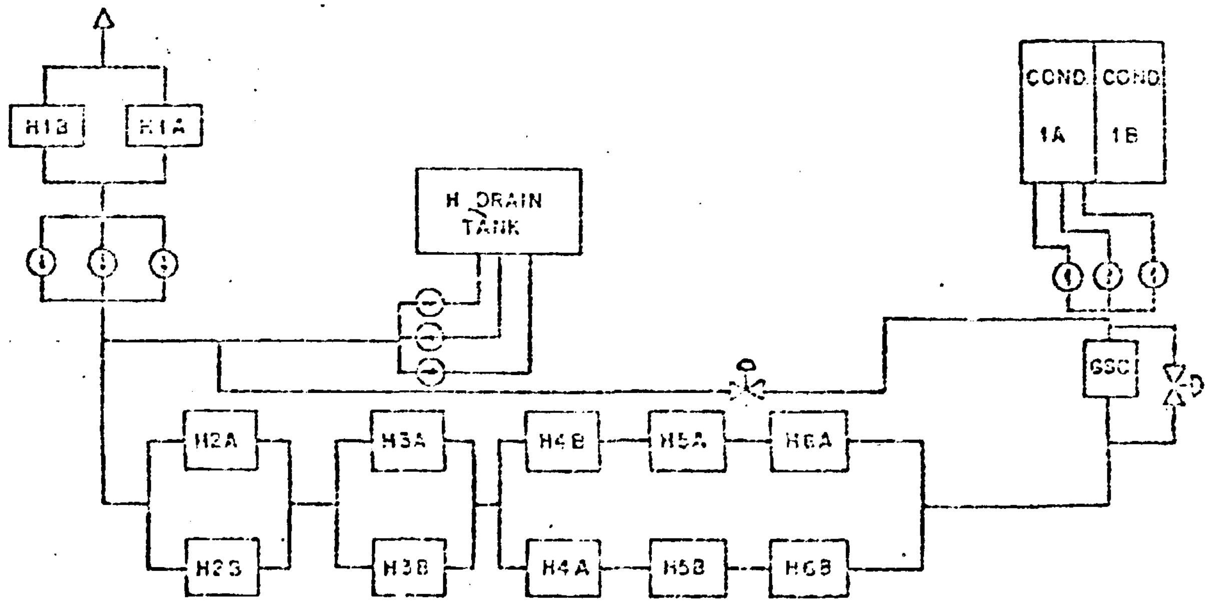


MILLSTONE 2



ANGRA I

FIG. 1 - DIFFERENT CONFIGURATIONS OF FEEDWATER AND CONDENSATE SYSTEMS (SIMPLIFIED)

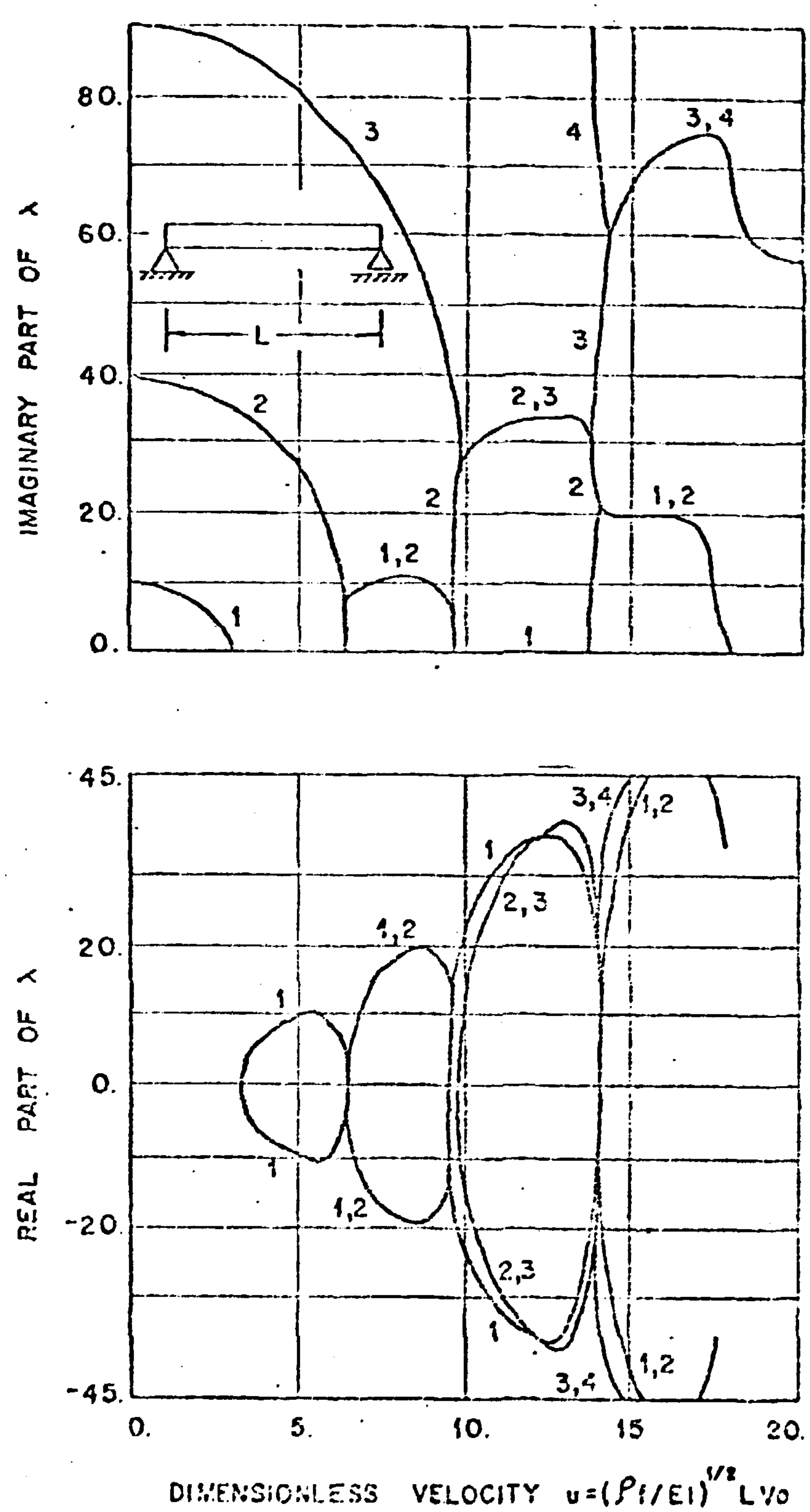


FIG. 2 - DYNAMIC STABILITY OF SIMPLE SUPPORTED PIPE SPAN CONVEYING FLUID

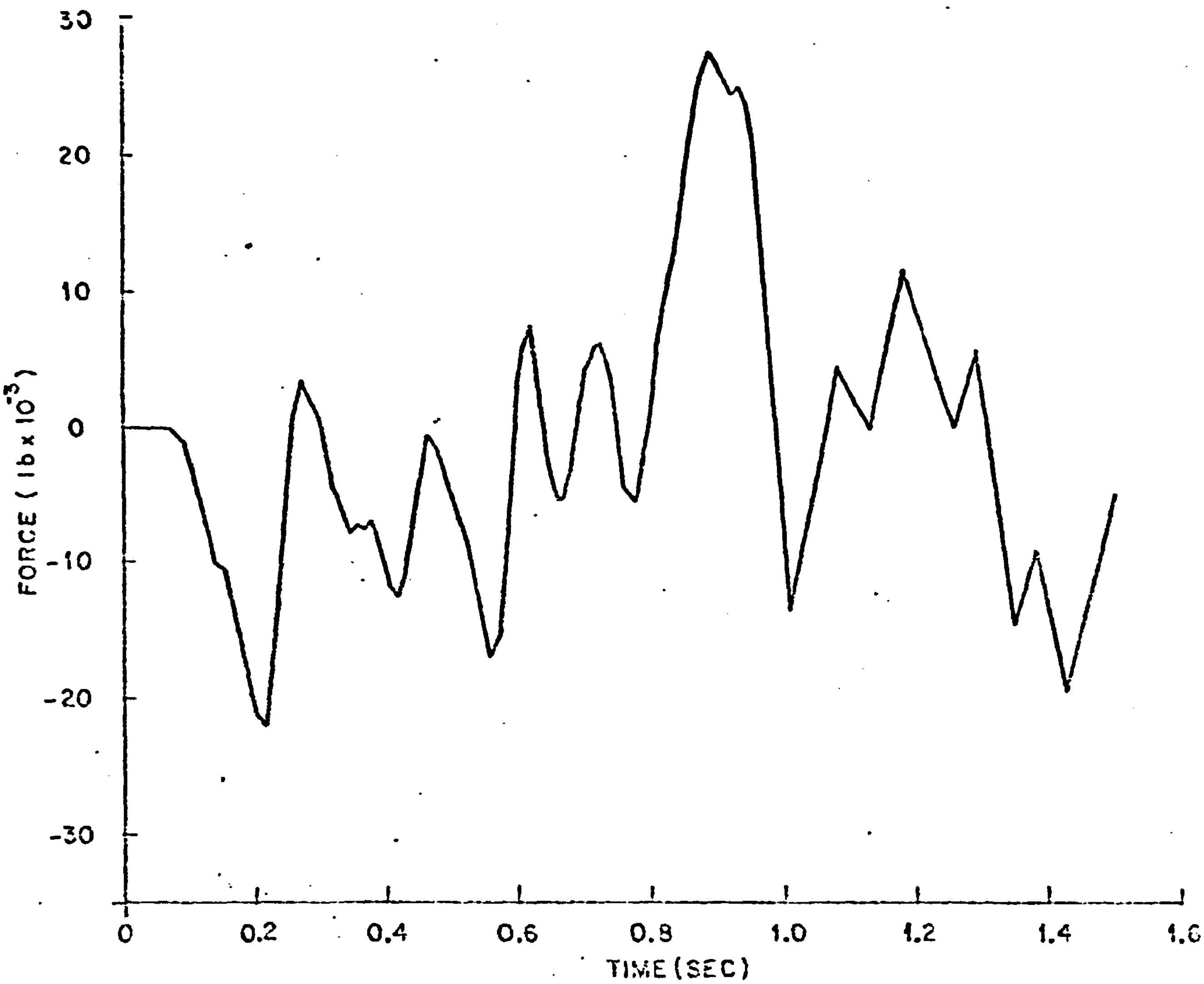
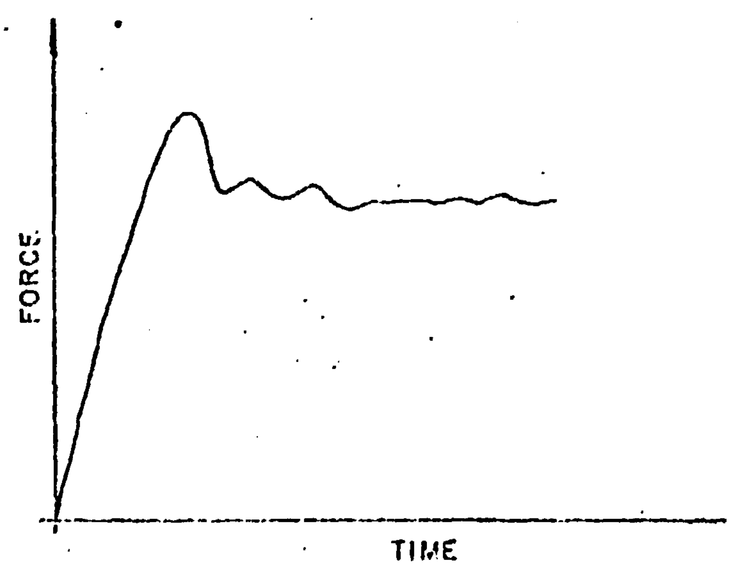


FIG. 3 - FORCE TIME HISTORY DUE TO STEAM HAMMER



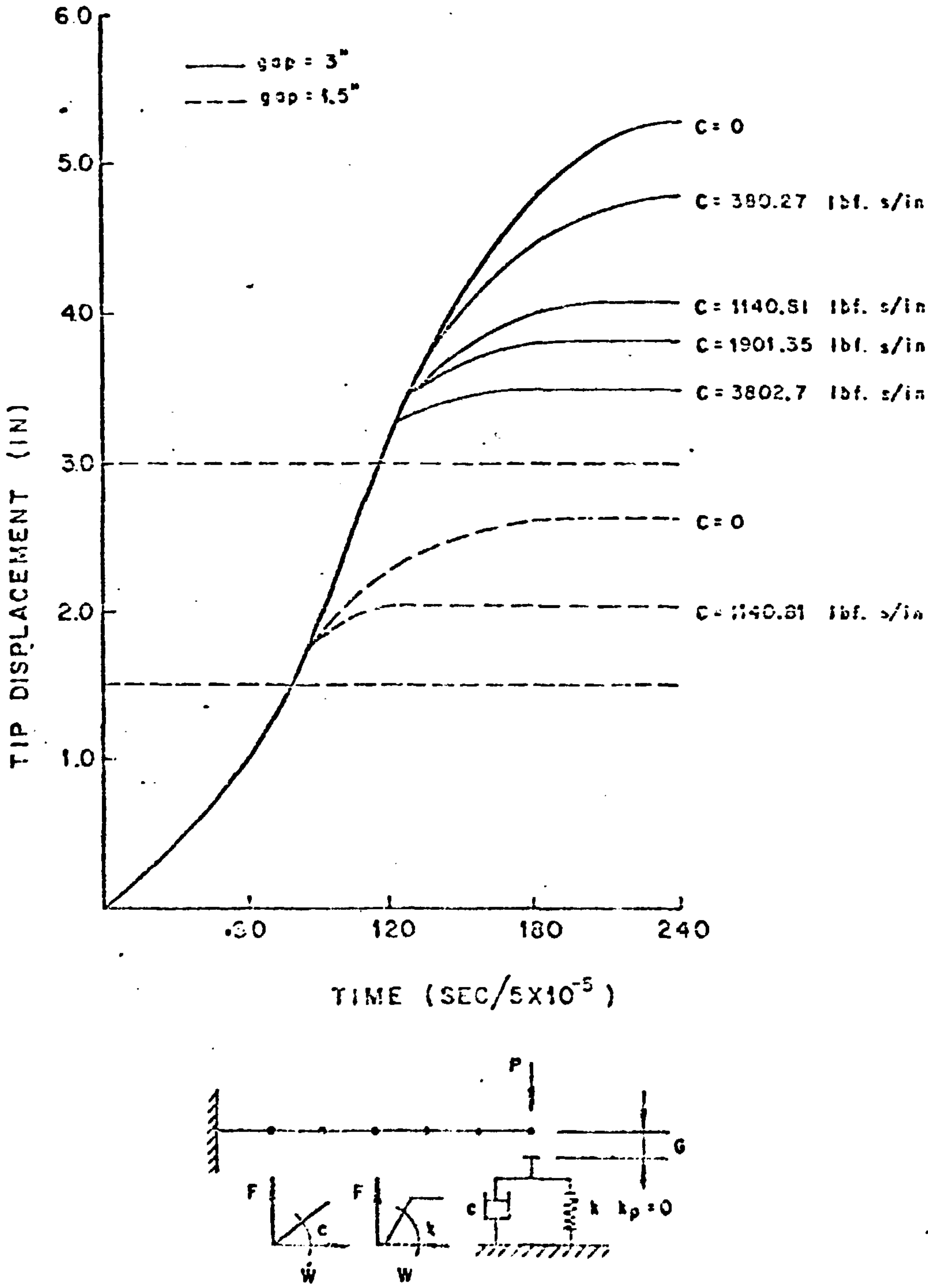


FIG.5 - INFLUENCE OF GAP AND RESTRAINT DAMPING ON TIP DISPLACEMENT.

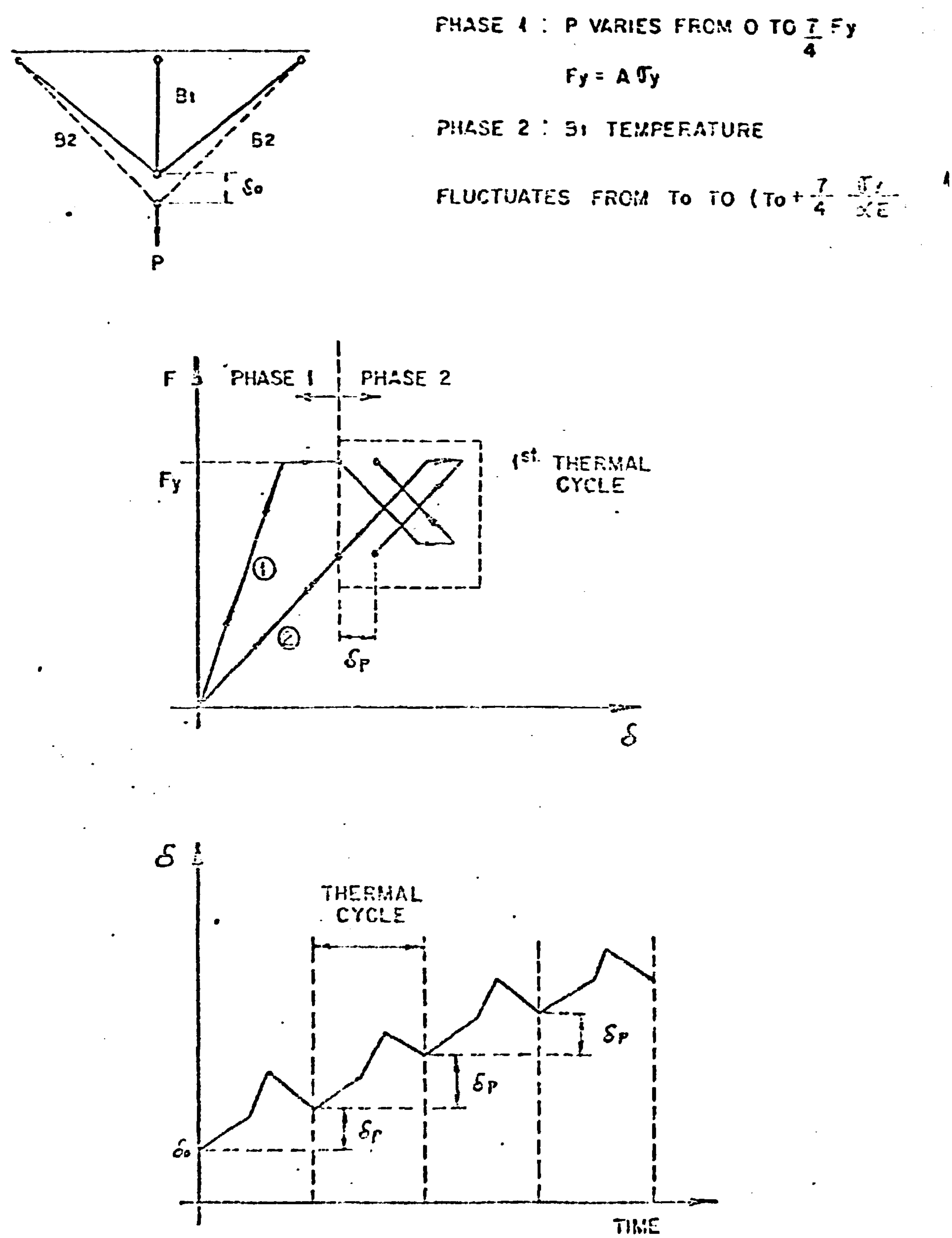


FIG 8 - STRAIN ACCUMULATION DUE TO
THERMAL RATCHETING

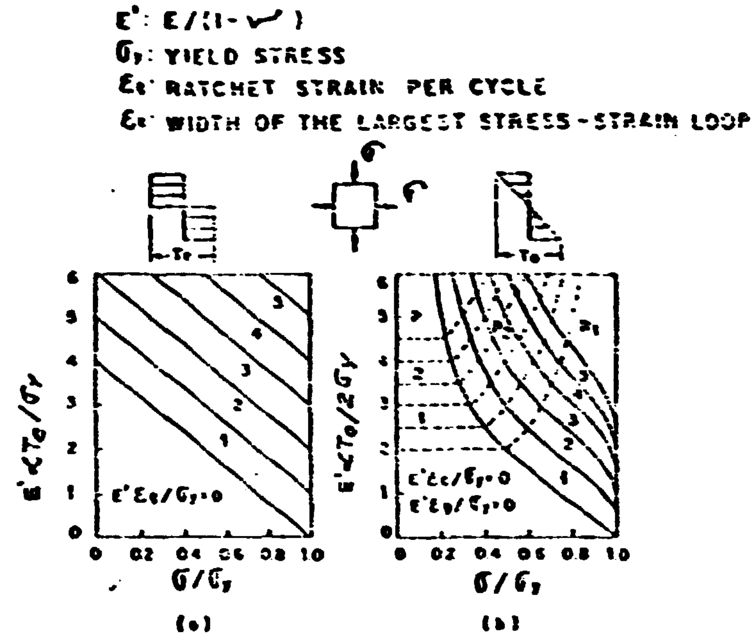


FIG. 9 - THERMAL RATCHET STRAINS FOR EQUAL BIAXIAL STRESSES
(a) STEP TEMPERATURE DISTRIBUTION
(b) LINEAR TEMPERATURE DISTRIBUTION

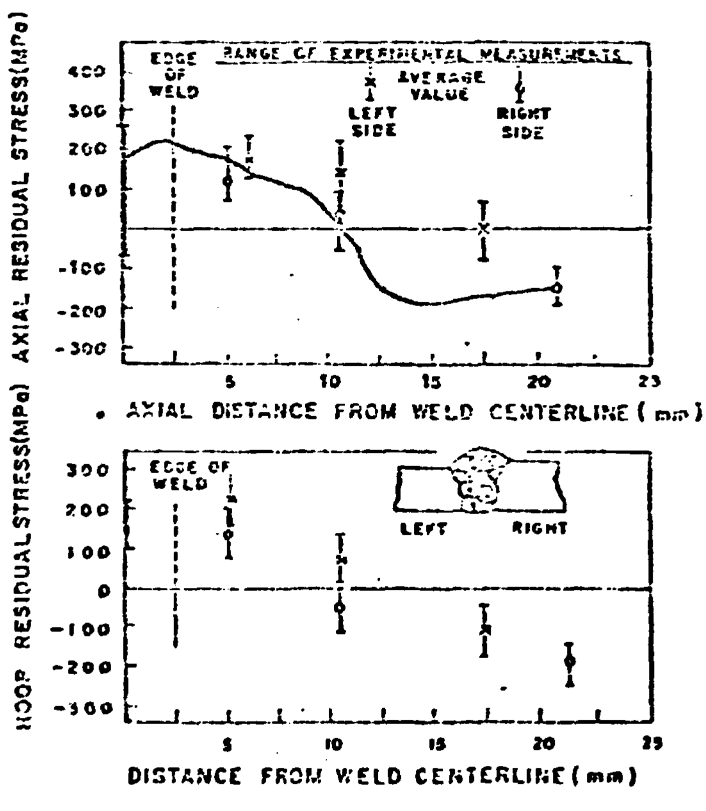


FIG.10 - COMPARISON OF CALCULATED AND EXPERIMENTALLY DETERMINED RESIDUAL STRESS FOR THE INNER SURFACE OF THE SEVEN-PASS WELD.

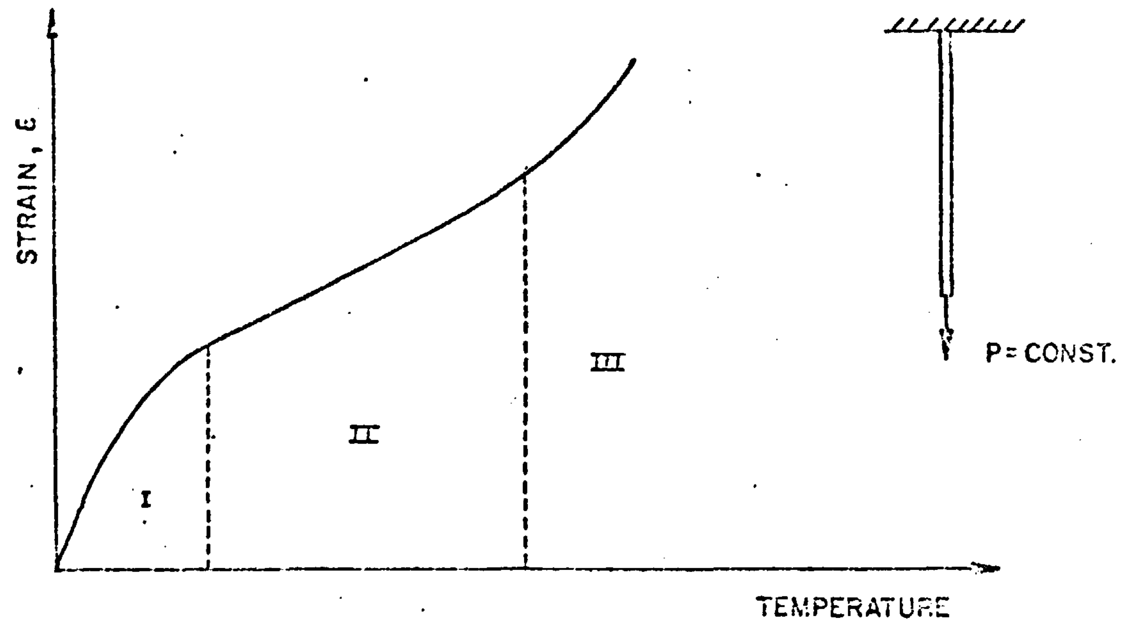


FIG. 11- TYPICAL METALLIC MATERIAL BEHAVIOR UNDER CONSTANT LOAD

I - PRIMARY CREEP
II - SECONDARY OR STATIONARY CREEP
III - TERCIARY CREEP

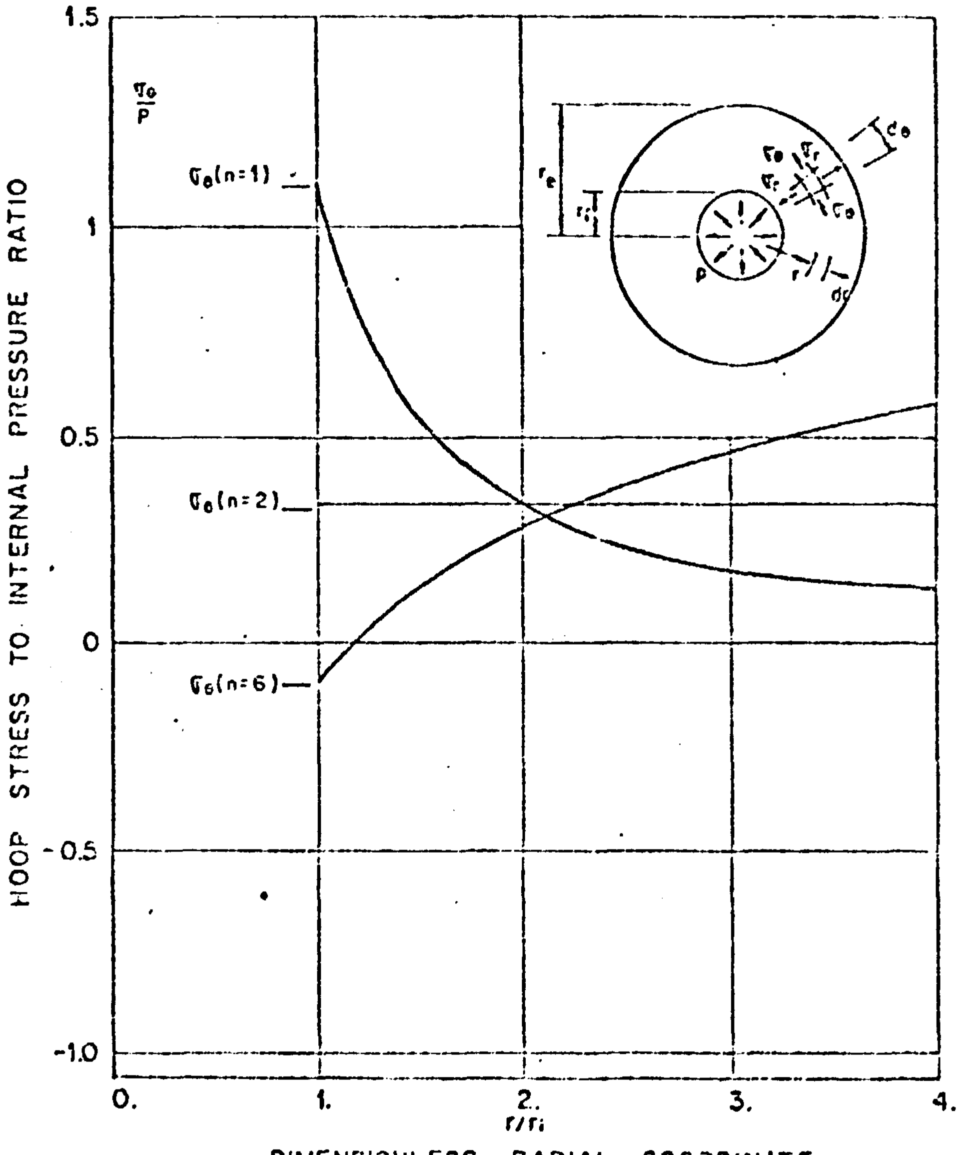
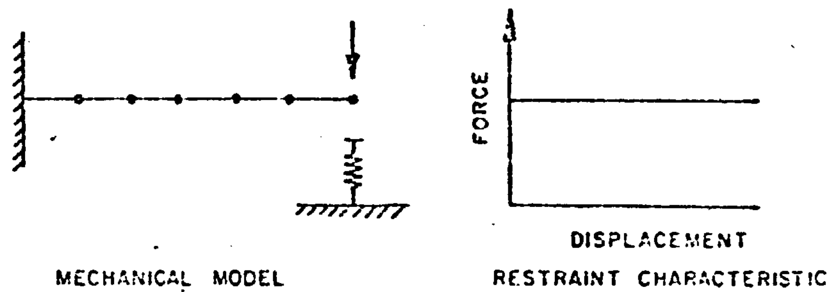


FIG. 12 – HOOP STRESS VARIATION THROUGH THE WALL FOR DIFFERENT CREEP PARAMETERS.



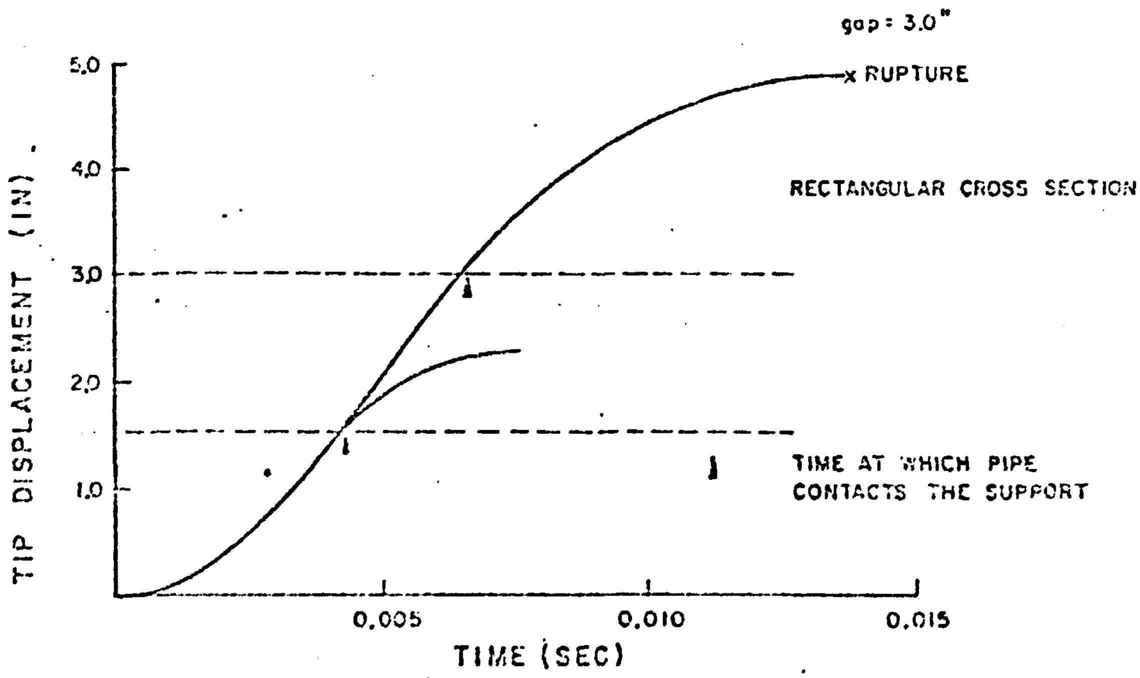


FIG. 7 — TIME HISTORY OF THE TIP DISPLACEMENT FOR A RIGID - PLASTIC RESTRAINT

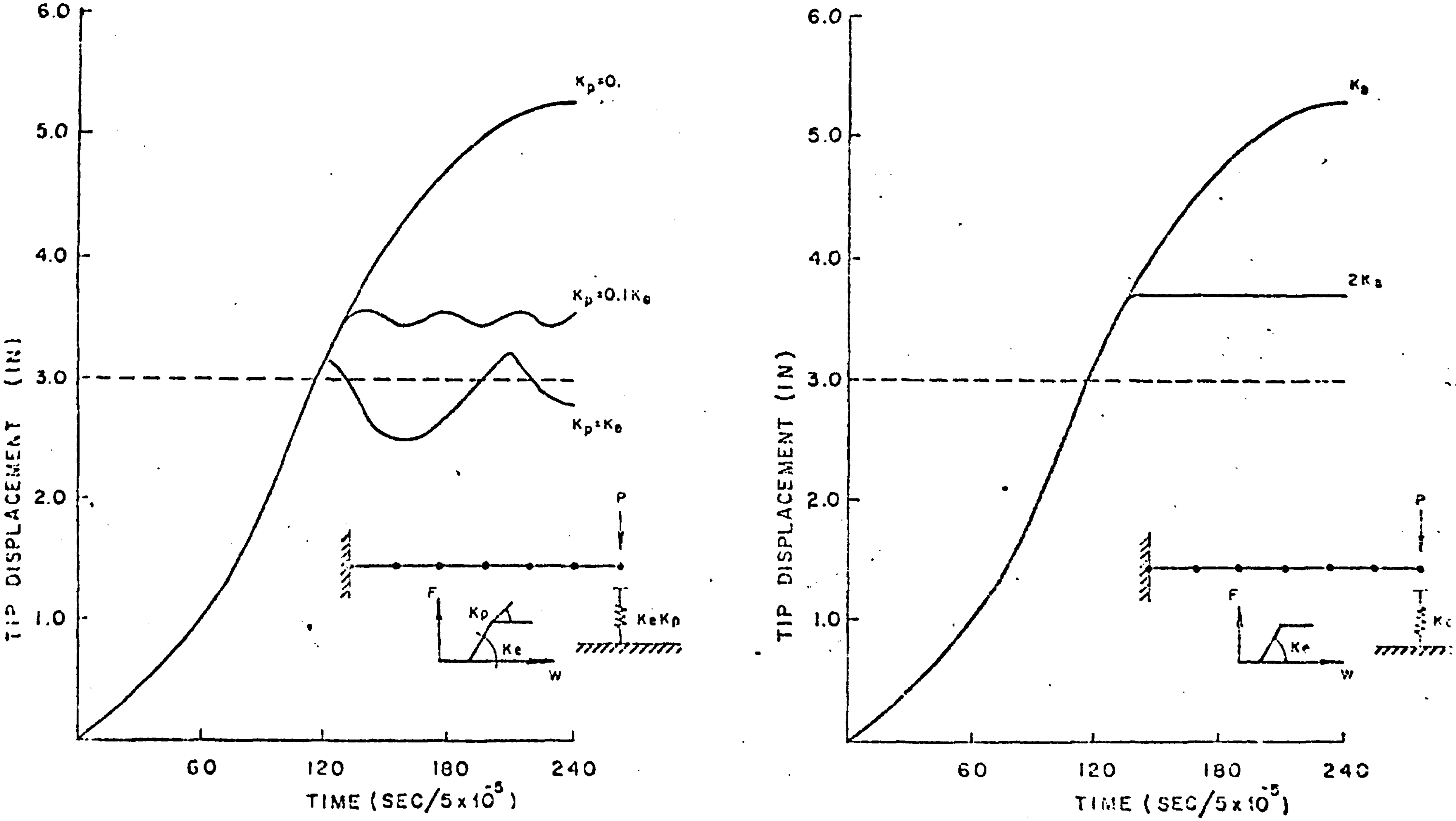


FIG.6 — INFLUENCE OF RESTRAINT STIFFNESS