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ASSESSMENT OF MARGINS WITH RESPECT TO PRESSURIZED THERMAL SHOCK FOR THE 3 LOOP PLANTS OF THE FRENCH PROGRAM

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Abstract

At the beginning of 1982, FRAMATOME and EDF embarked on a two and a half year program on pressurized thermal shock (P.T.S.). The objectif of the program is to demonstrate that present and older reactor vessels of the French Nuclear program exhibit adequate Safety Margins with respect to Fracture, under all normal, upset, emergency and faulted conditions, and to provide recommandations of feasible plant specific modifications, both technically and economically that could be needed to guarantee the existence of such safety margins. The program is divided in three principal phases :

Phase 1 - Transient analysis

Phase 2 - Fracture Mechanics Analysis

Phase 3 - Assessment of Margins and recommandations

Phase 1 is completed and brings some important conclusions on the potential severity, with respect to P.T.S., of events occuring when the plant is at intermediate or cold conditions, on the importance of the primary flow rate which controls the degree of mixing in the case of cold water injection, and, of the importance of operating procedures and operator action. Phase two is presently in progress and should provide interesting information on the influence of cladding, flaw geometry and material characteristics.

1. INTRODUCTION

A very large analytical and experimental effort is presently dedicated to the Pressurized Thermal Shock (PTS) problem in the United States, where a number of relatively old Reactor Vessels were built using weld materials with high content in residual elements such as phosphorus and copper, and are predicted to undergo significant damage (toughness degradation) from the nuclear core neutron bombardment, throughout the plant design life. For these vessels, the materia toughness degradation combined with the postulated occurence of a severe P.T.S transient could result,at some time during p ant life, in a significant reduction of the safety margins with regard to the R.P.V fast fracture risk.

In France, all the reactor pressure vessels built, including the older ones, have low content in phosphorus and copper, in the base as well as in the weld materials. Table 1 presents the Cu, P, Ni content values, the initial and the calculated end of life (E.O.L) inside surface RT_{NDT} values of the reactor vessel beltline base material (A 508 Cl.3) for the reactor pressure vessels of the French program. The maximum values in copper and phosphorus content are respectively 0.10 and 0.014 and calculated E.O.L RT_{NDT} values are below 70°C.

Table 2 presents the same data for the reactor vessel beltine weld material. The maximum values of Cu and P content are 0.125 and 0.019 respectively and the maximum calculated E.O.L RT_{NDT} is 139°C for the Bugey 5 reactor vessel. From CP 1 (First unit of the 900MWe Contrat Programme) the specified Cu and P content values have been tightened, resulting in calculated E.O.L inside surface RT_{NDT} values below 45°C. These results indicate that none of the reactor vessels built in France have a material RT_{NDT} predicted to reach the NRC screening criteria value of 167°C (300°F) throughout the entire life of the reactor. Consequently, the P.T.S issue is not considered in France as acute a problem as it is in the "United States". However, at the end of 1981, Framatome and E.D.F have estimated that an important work remained to be done in this area and at the beginning of 1982 a joint FRA-E.D.F program was started. The reasons that led FRA and E.D.F to embark in such a program were threefold :

a) Methodology

The first reactor vessel fracture mechanics analyses were performed by Framatome between 1975 and 1977 according to ASME code methodology. Since then, the state of the art has progressed significantly and, today, recognizes the importance of parameters such as stainless steel cladding, sub-surface flaws and elastic-plastic behavior.

b) Materials

The amount of material data available has increased very significantly from the time the first analyses were made, for instance on :

- the influence of Ni on irradiation embrittlement
- the J-Δa resistance curves from room to operating temperatures
- the Cu, P, S segregation effects in R.V forged rings
- the crack growth rate data (da/dN).

c) Transients

The determination of standard design transients was made in the past mainly taking into consideration core cooling requirements rather than pressurized thermal shock effects on the reactor vessel . For instance, transients assumed to occur at intermediate condition between hot stand-by and cold shutdown, or at cold condition when the fluid is monophasic, are generally more severe with respect to P.T.S, than when assumed to occur at full power or hot stand-by. This is specially the case when, in a transient scenario involving safety injection, it is postulated that all primary pumps are shut-down and natural circulation, in one or more primary loops breaks down, resulting in a complete loss of flow. The cold safety injection water is then assumed to enter the R.V down-comer without mixing with the hot primary water and to cause a severe thermal shock on the vessel wall.

Consequently, the prime objective of the joint FRA-E.D.F. program is to perform, first, a comprehensive analysis of possible pressurized thermal shock transients, second, a fracture mechanics analysis of the R.V beltine, utilizing the transients previously defined, the latest methodology and material data, in order to verify that the safety margins specified in the French design and construction code (RCC-M) are met for all conditions (upset, emergency, faulted), throughout the design life of the reactor. If the safety margins were found to be, in some cases, unacceptable, recommandations will be made to reduce the fluence on the vessel, through optimized fuel management and/or to make system or operating procedure changes.

2. PROGRAM DESCRIPTION

The FRAMATOME-E.D.F joint program was started at the beginning of 1982 and should be terminated in mid 1984.

This program includes three main phases :

Phase 1 consists in a thorough analysis of pressure and temperature transients that the R.P.V beltine could undergo during plant operations.

Phase 2 is the Fracture Mechanics Analysis utilizing the most up to date methodology and material data.

Phase 3 estimates the safety margins available during normal, upset, emergency and faulted conditions.
If for some conditions, the margins are found to be inadequate, recommandations will be made regarding system changes, procedure modifications, fuel management.

Table 3 indicates the time schedule attached to the 3 phases of the program. Phase 1 is almost completed and phase 2 is in progress.

2.1. Transient analysis (Phase 1)

As already mentioned before, the available standard design transients have been defined at 100% power. From the standpoint of R.P.V fast fracture risk, it is most likely that limiting transients are those which could occur from hot stand-by condition or from condition between hot standby and cold shutdown (intermediate), rather than transients which could occur from 100% power. Also, the temperature variations described are average values in the cold and hot legs, and in some comditions (S.I actuation for example), are not representative of the fluid temperature in the R.P.V down-comer. It is the reasons why a very significant part of the program has been dedicated to transient analysis.

The transient analysis includes the following tasks :

- Determination of overcooling and/or overpressurization transients not included in the standard design transient list.

- Determination of probability of occurence for the transients defined in previous task and classification of the transients in normal, upset emergency, faulted categories.

- Selection of transients for the Fracture Mechanics Analysis.

2.1.1. Determination of transients

The determination of transients is performed as a function of the plant condition at the time the initiating event occurs. Three plant conditions are considered :

- From -100% Power to hot standby (included)
- Intermediate two-phase conditions. The plant is cooling-down from hot stand-by with a steam bubble in the pressurizer and the RHR is not yet connected.
- Cold conditions. The plant continues to cool-down, the RHR is connected and the primary fluid is in single or two-phase conditions.

2.1.1.1. At power and hot stand-by conditions.

Since only severe cooling and/or overpressurization are of significance regarding RPV failure, two type of scenarios only are considered :

- Partial[(1)] or total[(2)] loss of flow with safety injection actuation (inadvertent or not)
- Rapid cooling of the primary fluid with safety injection actuation and partial[(1)] or total[(2)] loss of flow.

Table 4 presents the transients considered.

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(1) Partial loss of flow : All primary pumps tripped. Natural circulation maintained.

(2) Total loss of flow : All primary pumps tripped. Loss of natural circulation

2.1.1.2. Intermediate two-phase conditions.

$$24 \text{ bars} < P < 155 \text{ bars}$$
$$160°C < T_{Mean} < 284°C$$

The scenarios considered are similar to those considered in 2.1.1.1. and the transients are also listed in Table 4.

2.1.1.3. Cold conditions.

The scenarios considered here are the following :

- Misfunctioning of RHR
- S.I actuation (inadvertent or not)
- Unbalence between charge and discharge of primary water
- Loss and recovery of primary flow in conjunction with cold water injection.

Table 5 presents the list of transients considered at cold conditions.

2.1.1.4. Conclusions.

Some important conclusions may be drawn from the transient determination :

a) Some upset transients described in the Standard Design Transient document and assumed to occur from full power or at hot stand-by could become more severe regarding RPV pressurized thermal shock if assumed to occur from intermediate conditions because of the lower fluid temperature reached during the transient.
For example, FIG.1 and 2 show the temperature and pressure transients corresponding to inadvertent S.I actuation with full flow at hot stand-by and intermediate conditions respectively. Although the temperature variations and final pressures are comparable in both cases, the final temperature is more than 100°C lower in the case of FIG.2.
Generally, transients assumed to occur at hot stand-by are envelope of the same transients occuring at full

power, and transients postulated to occur at intermediate conditions are usually more severe than when postulated to occur at hot stand-by, because of lower fluid temperatures reached.

b) The severity of primary fluid cooling in the down-comer (and consequently, the severity of the thermal shock on the RPV wall) caused by safety injection greatly depends on the flow rate in the primary loops.

FIG.3 presents the temperature and pressure variations for inadvertent safety injection at hot stand-by and with partial loss of flow, i.e assuming all the three P.P tripped at time zero but natural circulation maintained. In this case, the temperature variation (286°C $\longrightarrow$ 219°C) is larger than in the case with full flow (286°C $\longrightarrow$ 255°C as shown in FIG.1.

FIG. 2 and 4 also compare the same transients assumed to occur at intermediate conditions and again show a more severe temperature variation in the case with partial loss of flow. FIG.5 presents again the same transient but assuming complete loss of flow (no natural circulation). This transient would, obviously, cause an extremely severe pressurized thermal shock on the R.P.V. However, the probability of loosing natural circulation is, in this case, low and the probability of occurence for the transient in FIG.5 is estimated to be around 10^{-9}/year. Therefore, this transient is not to be considered as a design transient(*). It is also worth noting that the severe thermal transient is only seen by the pressure boundary located between the S.I nozzles and the R.P.V, and the average fluid temperature variation in the loop is considerably smaller. In an operating plant, the Primary Circuit Temperature monitoring system utilizes thermo-couple located in the main loop, upstream of the S.I nozzles and, in case of S.I actuation, the recorded temperatures could be significantly different from the fluid temperature in the down-comer.

(*) Generally, for a given initiating event, if additional assumptions are made which render the transient more severe, the probability for the transient to occur will decrease with it's increasing severity.

c) <u>Transients postulated to occur at cold conditions (RHR connected) could result in severe pressurized thermal shocks of the R.P.V beltine.</u>
According to the probability of occurence calcultations, these transients should be considered as emergency and faulted conditions.
The principal system configurations related to cold conditions are presented in table 6. FIG.6 and FIG.7, for example, present the pressure and temperature transients corresponding to an inadvertent opening of a RHR regulation valve.

d) <u>Operator action and operating procedures are of paramount importance regarding the severity of a transient resulting from a given initiating event.</u>
For example, in FIG.7, the operator is assumed to restart the primary pumps 30 minutes after the beginning of the transient,resulting in a minimum fluid temperature of 90°C. In fact, the operator will certainly take action much earlier and the minimum fluid temperature reached will be higher. If the operator is assumed to take no action at all, the fluid temperature would eventually get down to 20°C.

2.1.2. <u>Determination of the transient probability of occurence.</u>

The determination of a transient resulting from a postulated scenario implies that the probability of occurence of such transient is calculated in order to estimate if it has to be considered in the design and in which operating condition (Upset, Emergency, Faulted), it falls.

Table 7 summarizes the main assumptions made to perform the probability calculations.

Tables 8 to 12 present the estimated probabilities of occurence of the transients considered in tables 4 and 5 and their classification in normal, upset, emergency and faulted categories.

2.1.3. Selection of transients for the Fracture Mechanics Analysis.

The basis for the determination of the transients, as described in paragraph 2.1.1., is their "a priori" potential significance regarding the reactor vessel integrity problem. However, it is quite obvious that, after performing the detailed Fracture Mechanics analysis, some of these transients will in fact appear to have little impact on the reactor vessel integrity and may be eliminated, in order to limit the extent of the Fracture Mechanics Analysis

The transient selection includes the following steps :

- Select significant transients using a screening criteria based on a simplified Fracture Mechanics Analysis.

- For each category (Upset, Emergency, Faulted), define envelope transients.

2.1.3.1. Transient selection based on screening criteria

The screening criteria are based on a simplified Fracture Mechanics Analysis. Several fictitious transients are used in this analysis, as shown in Figure 8. The parameters considered are :

The final fluid temperature : T_F

The material E.O.L. RT_{NDT} : $T_F - RT_{NDT}$

The fluid temperature variation : $T_I - T_F$

The fluid temperature variation rate : $\dfrac{T_I - T_F}{\Delta t}$

The fluid pressure : P

The simplified Fracture Mechanics Analysis uses the fictitious transients and the following features :

- Continuous longitudinal surface flaws postulated
- Cladding not included
- LEFM analysis with plastic zone correction
- Instability reached when $K_I \geqslant K_{IC}$ at EOL
- K_{IC} : reference Fracture toughness curve (R.C.C.M.)
- EOL RT_{NDT} predicted using R.G. 1.99 formula

The analysed transients are considered insignificant if a postulated flaw smaller or equal to 25 mm does not become unstable at EOL.

The results of the simplified analysis show that the most significant parameter is $T_F - RT_{NDT}$. The other important parameters are $(T_I - T_F)$ and the fluid pressure P. The temperature variation rate $(T_I - T_F)/\Delta t$ was found to have very little influence.

Consequently, the screening criteria are defined as follows :

$$\text{If} \quad \left\{ \begin{array}{r} P \leqslant 15 \text{ MPa} \\ \Delta T \leqslant 180° \text{ C} \\ T_F - RT_{NDT} > 10° \text{ C} \end{array} \right. \quad \text{then } a_c > 25 \text{ mm}$$

$$\text{If} \quad \left\{ \begin{array}{r} P \leqslant 4 \text{ MPa} \\ T_F - RT_{NDT} > 0{,}3\ \Delta T - 70° \text{ C} \\ \Delta T > 70° \text{ C} \end{array} \right. \quad \text{then } a_c > 25 \text{ mm}$$

The influence of taking into account the presence of the cladding on the above screening criteria is presently analysed and may result in some changes in the formula.

2.1.3.2. Determination of Envelope transients

The transients selected using the previous criteria are classified in categories (Upset, Emergency, Faulted) according to their probability of occurence and, for each category, one or more envelope transients are determined as follows :

- The envelope transients cover transients having similar pressure and temperature variations and probability of occurence.

- The probability of occurence attributed to the envelope transients is equal to the sum of the probability of occurence of each individual transient enveloped.

2.2. Fracture Mechanics Analysis (Phase 2)

The envelope transients selected in 2.1.3.2. will be used in the Fracture Mechanics Analysis. Two important parameters in the analysis are the material E.O.L. RT_{NDT} and the postulated flaw geometry. Variations of these two parameters will be considered in order to estimate their influence on the results.

2.2.1. RT_{NDT}

In tables 1 and 2, the content in residual elements (Cu,P,Ni) and calculated E.O.L. inside surface RT_{NDT} of the beltline materials were presented for all the reactor vessels of the French PWR plants. For the base materials, the Cu, P, S content values considered are the laddle values multiplied by a segregation factor which is a function of distance from the inside surface of the vessel wall. At the cladding-base metal interface, the segregation factors are as follow :

$$P = 1.19\ P_L$$

$$Cu = 1.10\ Cu_L$$

$$S = 1.3\ S_L$$

Also, for the base materials, the initial RT_{NDT} values considered are those determined at 1/4 T location minus 15° C to account for the toughness variation from the 1/4 t location to the inside surface location.
The shift in RT_{NDT} due to irradiation is calculated using the R.G 1.99 formula and the formula established by FRAMATOME.[1]
The RT_{NDT} gradient through the vessel wall is calculated using a DPA fluence attenuation through the wall.

In order to cover the wide range of E.O.L. RT_{NDT} obtained, four different combinations of E.O.L. fluence, RT_{NDT} shift calculation formula and Cu, P, Ni content are considered, for both the base and weld materials. These sets of combinations are presented in table 13.

2.2.2. Flaw geometries

The presence of the cladding will be accounted for and three inside surface flaw geometries will be considered, as follows :

- Longitudinal infinite flaws
- Longitudinal semi-elliptical flaws (1 x 6 aspect ratio)
- Continuous circumferential flaws

In the case of semi-elliptical flaws, instability will be considered on the crack front, at the inside surface, at the cladding-base metal interface and at the deepest point of the crack. Sub surface flaws will also be considered.
Accurate stress intensity factor solutions[2] determined by influence functions will be used for each specific flaw geometry and loading.

(1) B. HOUSSIN and Al. - Radiation embrittlement of PWR Reactor vessel weld metals : nickel and copper Synergism effects
presented at the A.S.T.M. 11 th conference on radiation embrittlement - A.S.T.M. STP 782 (1982) pp. 392 - 411

(2) J. HELIOT, R.C. LABBENS and A. PELISSIER-TANON "Semi-Elliptical Cracks in a cylinder subjected to stress gradients "Fracture Mechanics A.S.T.M. STP 677 (1979) pp. 342 - 364

2.3. Determination of Safety Margins and Recommandations (Phase 3)

The results of the Fracture Mechanics Analysis will indicate if the Safety Margins required by the R.C.C.M. code criteria can be met for all operating conditions (Upset, Emergency, Faulted).

If, for some conditions, the margins are found to be inadequate, recommandations will be made with respect to system and/or procedure modifications, fluence reduction (fuel management). The various material parameter combinations analysed (Table 13) will allow to estimate the benefits obtained through the possible modifications regarding R.V. fluence reduction. Also, the benefits resulting from possible system and/or procedure changes will be estimated through the parametric aspects of the transient analyses.

3. CONCLUSION

The objective of the joint FRAMATOME - E.D.F. program is to demonstrate that adequate safety margins regarding R.P.V. rupture exist for all the reactor vessels of the French nuclear program under all operating conditions and to identify and propose plant specific modifications that may be needed to guarantee such safety margins.

The program which started at the beginning of 1982 is presently under way and should be terminated in the middle of 1984.

The first phase of the program consisting in a thorough transient analysis is completed and has brought some important conclusions described in detail in paragraph 2.1.1.4., revealing

a) the potential severity (regarding R.P.V. pressurized thermal shock) of events occuring when the plant is at intermediate or cold conditions.

b) The importance of the primary flow rate which controls the degree of mixing in case of safety injection, especially the complete loss of flow where natural circulation breaks-down and mixing is limited.

c) the importance of operating procedures and/or operator action.

The Fracture Mechanics Analysis phase will be completed at the end of 1983 and should provide interesting information on the role of cladding, flaw geometry and material properties.

TABLE 1.

R.P.V. BELTLINE BASE MATERIAL CHARACTERISTICS

VESSEL		EOL FLUENCE X 10^{19} (N/CM^2)	CHEMICAL ANALYSIS (INCLUDING SEGREGATION FACTOR)			$(RT_{NDT})I$ (°C)	$(RT_{NDT})F$ RG 1.99 (°C)
			P	Cu	Ni		
FSH	1	4.8	0.013	0.09	0.80	- 32	44
	2	4.8	0.012	0.09	0.73	- 32	38
BG	1	5.0	0.014	0.06	0.69	- 37	34
	2	5.0	0.014	0.06	0.73	- 22	49
	3	5.0	0.011	0.04	0.76	- 22	31
	4	5.0	0.011	0.06	0.76	- 12	41
CP	1	5.1	0.012	0.07	0.72	- 12	48
	2	5.1	0.011	0.06	0.71	- 27	27
	3	5.1	0.012	0.08	0.73	- 22	38
	4	5.1	0.011	0.08	0.75	- 17	37
	5	5.0	0.012	0.08	0.74	- 27	32
	6	5.0	0.012	0.07	0.77	- 22	37
	7	5.0	0.012	0.06	0.73	- 17	42
	8	5.0	0.012	0.09	0.71	- 22	49
	9	5.0	0.011	0.07	0.72	- 17	36
	10	5.1	0.012	0.09	0.75	- 22	50
	11	5.1	0.012	0.09	0.76	- 17	55
	12	5.1	0.012	0.09	0.74	- 27	45
	13	5.1	0.012	0.08	0.72	- 17	43
	14	5.1	0.013	0.10	0.72	- 32	59
	15	5.0	0.010	0.10	0.72	- 12	59
	16	5.0	0.014	0.09	0.72	- 17	67
	17	5.1	0.012	0.10	0.72	- 22	63
	18	5.1	0.010	0.09	0.73	- 22	38
	19	5.0	0.011	0.09	0.74	- 17	48
	20	5.0	0.011	0.09	0.71	- 22	43
	21	5.1	0.011	0.09	0.76	- 12	54
	22	5.1	0.011	0.07	0.74	- 7	47

TABLE 2.

R.P.V. BELTLINE WELD MATERIAL CHARACTERISTICS

VESSEL		EOL FLUENCE X 10^{19} (N/CM^2)	CHEMICAL ANALYSIS			(RT_{NDT}) I (°C)	(RT_{NDT}) F RG 1.99 (°C)
			P	Cu	Ni		
FSH	1	4.8	0.018	0.120	0.17	- 20 (1)	138
	2	4.8	0.013	0.100	0.14	- 20 (1)	83
BG	1	5.0	0.013	0.100	0.14	- 20 (1)	85
	2	5.0	0.019	0.076	0.43	- 20 (1)	98
	3	5.0	0.015	0.125	0.09	- 20 (1)	129
	4	5.0	0.019	0.113	0.53	- 20 (1)	139
CP	1	5.0	0.011	0.035	0.67	- 47	21
	2	5.0	0.011	0.035	0.67	- 42	26
	3	5.0	0.011	0.035	0.67	⩽ - 52	⩽ 16
	4	5.0	0.011	0.035	0.67	⩽ - 52	⩽ 16
	5	5.0	0.011	0.035	0.67	⩽ - 52	⩽ 16
	6	5.0	0.011	0.035	0.67	⩽ - 52	⩽ 16
	7	5.0	0.014	0.050	0.59	- 47	39
	8	5.0	0.011	0.056	0.73	- 27	41
	9	5.0	0.011	0.056	0.73	- 27	41
	10	5.0	0.011	0.056	0.73	- 27	41
	11	5.0	0.014	0.050	0.59	- 47	39
	12	5.0	0.014	0.050	0.59	- 47	39
	13	5.0	0.011	0.056	0.73	- 27	41
	14	5.0	0.011	0.056	0.73	- 27	41
	15	5.0	0.011	0.056	0.73	- 27	41
	16	5.0	0.011	0.056	0.73	- 27	41
	17	5.0	0.011	0.056	0.73	- 27	41
	18	5.0	0.012	0.020	0.72	⩽ - 52	⩽ 22
	19	5.0	0.012	0.020	0.72	- 42	32
	20	5.0	0.010	0.040	0.71	- 42	20
	21	5.0	0.008	0.017	0.62	- 42	7
	22	5.0	0.007	0.040	0.78	- 47	2

(1) Conservative estimate

TABLE 3.

PROGRAM SCHEDULE

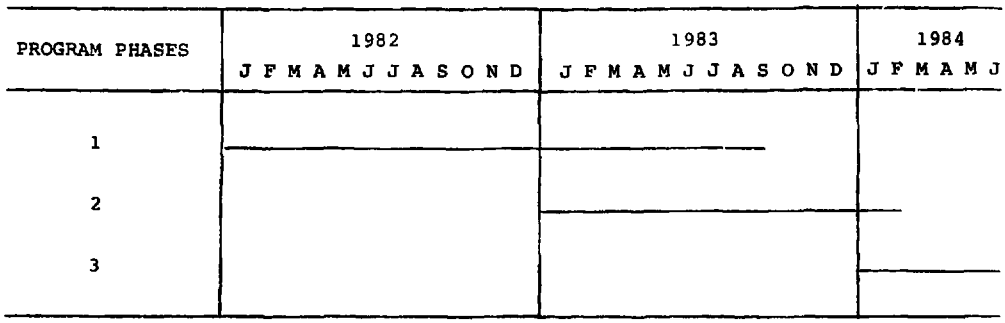

TABLE 4 - TRANSIENTS CONSIDERED (100% POWER - HOT STAND-BY INTERMEDIATE TWO-PHASE CONDITIONS)

- Inadvertent S.I.

  a) - with full flow
  b) - with partial loss of flow
  c) - with total loss of flow

- Excess of steam let-down with S.I.

  a)
  b)
  c)

- Excess of feedwater with S.I.

  a)
  b)
  c)

- Pressurizer break with S.I.

  a)
  b)
  c)

- Inadvertent discharge of accumulators (Intermediate conditions only)

  b)
  c)

TABLE 5 - TRANSIENTS CONSIDERED - COLD CONDITIONS

- Normal shutdown[1] of primary pump
  a) Normal pressure
  b) Inadvertent overpressurization

- Abnormal shutdown of primary pump and restart in accordance with procedure[2]
  a) Normal pressure maintained when P.P. restarted - RCS in two-phase condition between 130° C and 180° C
  b) Overpressurization caused by P.P. restarted with primary temperature between 70° C and 130° C

- Abnormal shutdown of primary pump and restart without action on RHR flow

- Abnormal shutdown of P.P. with overpressurization and restart in accordance wi·:h procedure

- Inadvertent opening of a RHR regulation valve

- Inadvertent closure of the RHR by-pass regulation valve

- Inadvertent safety injection actuation at single phase conditions with RHR connected

- RHR or CVCE break - Normal application of procedure (Temp. IHX $>$ 150° C)

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(1) Normal P.P. shutdown when Temp. $\leqslant$ 70° C, 25 bars $\leqslant$ P $\leqslant$ 27 bars

(2) The procedure calls for reduction of RHR flaw rate

TABLE 6 - SYSTEM CONFIGURATIONS AT COLD CONDITIONS

- Reactor shutdown
- $T < 180°$ C , 25 bars $\leq P \leq 30$ bars (abs)
- Two phase conditions for $T \geq 130°$ C [(1)]
- Single phase conditions for $T < 130°$ C [(2)]
- RHR connected to RCS for $T < 160°$ C
- For $T > 70°$ C, at least one P.P. operating
- For $T < 70°$ C, all P.P. shutdown, RHR pumps operating

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(1) Minimum authorized temperature = 120° C

(2) Operating procedures allow to maintain single phase conditions up to 180° C

TABLE 7 - PRINCIPAL ASSUMPTIONS IN PROBABILITY CALCULATIONS

- Sequence of events considered
- Probability of initial plant conditions function of operating time period
- Related events considered
- $f = \lambda T$

  f = frequency of event occurence of period of time T

  λ = components/systems failure rates
- Operator factor[1] taken into account for :
  . Lack of action or late action
  . Increase in component failure probability due to human error (based on E.D.F experience)

(2) Related probability based on wash 1400 data

TABLE 8.

TRANSIENT OCCURRENCE PROBABILITIES

100 % POWER

TRANSIENT	PROBABILITY OF OCCURRENCE (/ YEAR)	CATEGORY
Inadvertent S.I.		
b - W. Partial loss of flow	$4\ 10^{-5}$	Faulted
c - W. Total loss of flow	$(3 \times 10^{-6},\ 2 \times 10^{-8})$	[Faulted, NC[1]]
Excess of steam let-down with S.I.		
a - W. full flow	6×10^{-2}	Upset
b - W. Partial loss of flow	7×10^{-6}	Faulted
c - W. Total loss of flow	$(6 \times 10^{-7},\ 4\ 10^{-9})$	NC
Excess of feed water with S.I.		
a -	$8\ 10^{-5}$	Faulted
b -	$9\ 10^{-9}$	NC
c -	$(5\ 10^{-10},\ 5\ 10^{-12})$	NC
Pressurizer break with S.I.		
a -	$2\ 10^{-2}$	Upset
b -	$2\ 10^{-6}$	Faulted
c -	$(2\ 10^{-7},\ 10^{-9})$	NC

(1) NC = Not considered

TABLE 9.

TRANSIENT OCCURRENCE PROBABILITIES

(HOT STAND-BY)

TRANSIENT	PROBABILITY OF OCCURRENCE (/ YEAR)	CATEGORY
Inadvertent S.I.		
a -	$2\ 10^{-2}$	Upset
b -	$2\ 10^{-6}$	Faulted
c -	$(2\ 10^{-7},\ 10^{-9})$	NC
Excess of steam let-down with S.I.		
a -	$3\ 10^{-3}$	Emergency
b -	$4\ 10^{-7}$	NC
c -	$(3\ 10^{-8},\ 2\ 10^{-10})$	NC
Excess of feed water with S.I.		
a -	$2\ 10^{-2}$	Upset
b -	$2\ 10^{-6}$	Faulted
c -	$(2\ 10^{-7},\ 10^{-9})$	NC
Pressurizer break with S.I.		
a -	$2\ 10^{-3}$	Emergency
b -	$3\ 10^{-7}$	NC
c -	$(2\ 10^{-8},\ 10^{-10})$	NC

TABLE 10.

TRANSIENT OCCURRENCE PROBABILITIES
(INTERMEDIATE TWO-PHASE CONDITIONS)

TRANSIENT	PROBABILITY OF OCCURRENCE (/ YEAR)	CATEGORY
Inadvertent S.I.		
a -	$6\ 10^{-3}$	Emergency
b -	10^{-6}	Faulted
c -	$(10^{-7},\ 7\ 10^{-10})$	NC
Inadvertent discharge of accumulators		
b -	$6\ 10^{-11}$	NC
Excess of steam let-down with S.I.		
a -	10^{-3}	Emergency
b -	$3\ 10^{-7}$	NC
c -	$(2\ 10^{-8},\ 10^{-10})$	NC
Excess of feed water with S.I.		
a -	$6\ 10^{-3}$	Emergency
b -	10^{-6}	Faulted
c -	$(10^{-7},\ 7\ 10^{-10})$	NC
Pressurizer break with S.I		
a -	$3\ 10^{-4}$	Emergency
b -	$7\ 10^{-8}$	NC
c -	$(7\ 10^{-9},\ 4\ 10^{-11})$	NC

TABLE 11.

TRANSIENT OCCURRENCE PROBABILITIES

COLD CONDITIONS-RHR CONNECTED

TRANSIENT	PROBABILITY OF OCCURRENCE (/ YEAR)	CATEGORY
Normal shut-down of P.P.		
a) Normal pressure	3	Normal
b) Inadvertent overpressurisation	10^{-4}	Emergency
Abnormal shut-down of P.P		
a) Two phase	$9\ 10^{-3}$	Upset
b) Single phase and P.P. restart in accordance with procedure	$7\ 10^{-2}$	Upset
Abnormal shut-down of P.P and restart W.O. action on RHR flow	$2\ 10^{-3}$	Emergency
Abnormal shut-down of P.P W. overpressurization and P.P. restart in accordance with procedure	$2\ 10^{-3}$	Emergency
Inadvertent opening of a RHR regulation valve	$4\ 10^{-5}$	Faulted
Inadvertent closure of the RHR by pass regulation valve	$2\ 10^{-9}$	NC

TABLE 12.

TRANSIENT OCCURRENCE PROBABILITIES

COLD CONDITIONS - RHR DISCONNECTED

TRANSIENT	PROBABILITY OF OCCURRENCE (/ YEAR)	CATEGORY
RHR or CVCS break. Normal application of procedure		
$T_{II} > 150°C$		
Break on CVCS	10^{-5}	Faulted
Break on RHR	$4\ 10^{-6}$	Faulted
$T_{I} < 150°C$		
Break on CVCS	10^{-3}	Emergency
Break on RHR	$3\ 10^{-4}$	Emergency
RHR or CVCS break. Abnormal application of procedure ($T_{I} > 150°C$)		
Break on CVCS	10^{-7}	NC
Break on RHR	$4\ 10^{-8}$	NC

TABLE 13.

MATERIAL PARAMETER COMBINATIONS

	$\Delta\ RT_{NDT}$ FORMULA	FLUENCE (E.O.L.) x 10^{19} n/cm^2	Cu (%)	Ni (%)	P (%)	$(RT_{NDT})_I$ (°c)	$(RT_{NDT})_F$ (°c)
WELD MATERIAL	RG 1.99	5.0	0.113	0.530	0.019	- 20	140
	RG 1.99	3.8	"	"	"	"	120
	FRA	5.1	0.07	1.2	0.014	"	80
	FRA	5.1	0.06	0.73	0.011	"	50
BASE MATERIAL	RG 1.99	5.1	0.1	0.83	0.014	0	110
	FRA	"	"	"	"	0	100
	FRA	"	0.08	0.85	0.008	0	70
	FRA	"	"	0.83	0.011	- 20	60

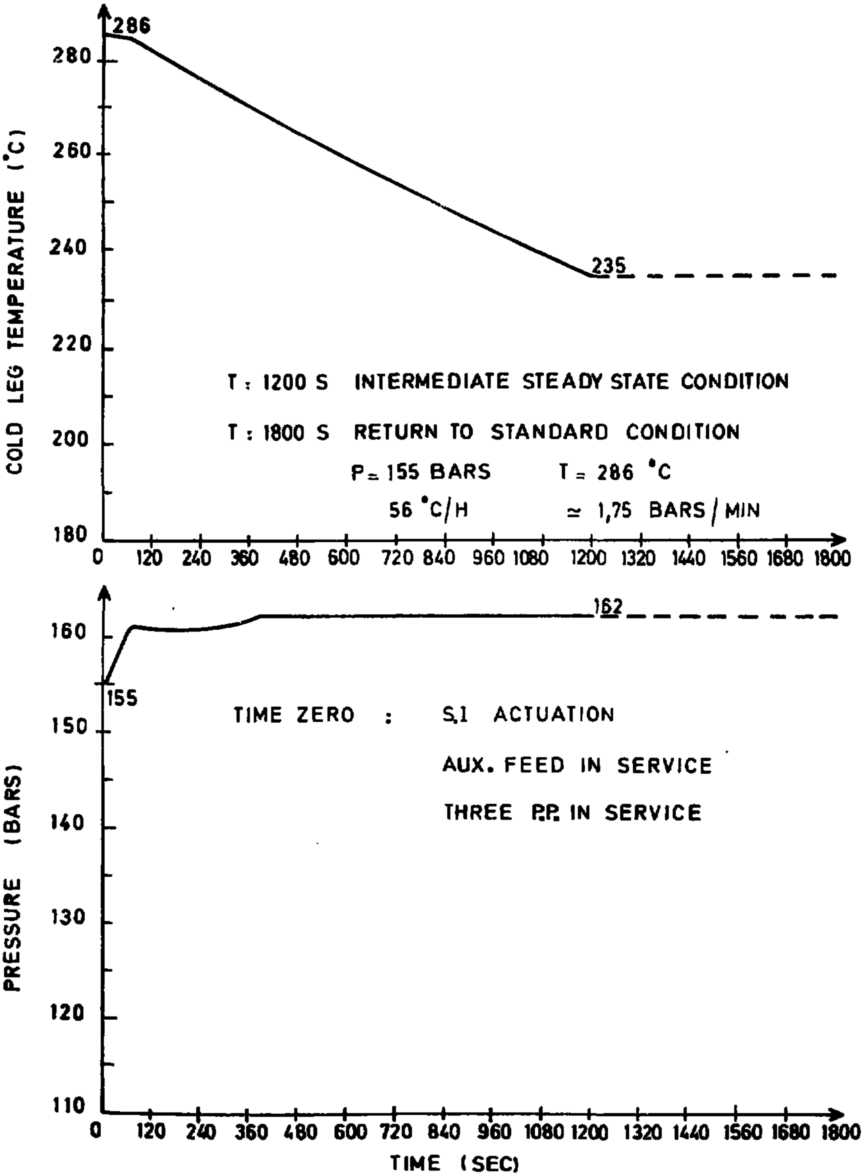


FIG : 1 INADVERTENT S.I WITH FULL FLOW AT HOT STAND-BY

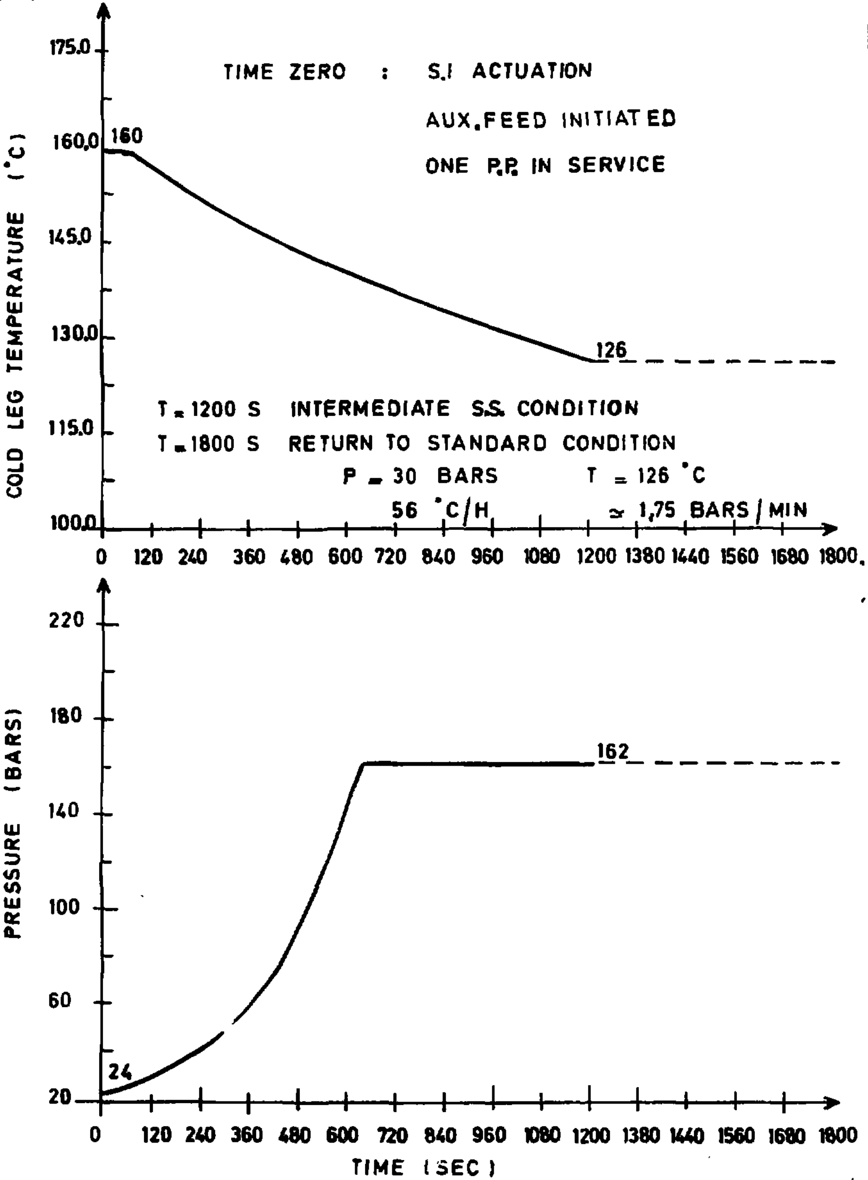


FIG: 2 INADVERTENT S.I WITH FULL FLOW AT INTERMEDIATE CONDITIONS (TWO-PHASE)

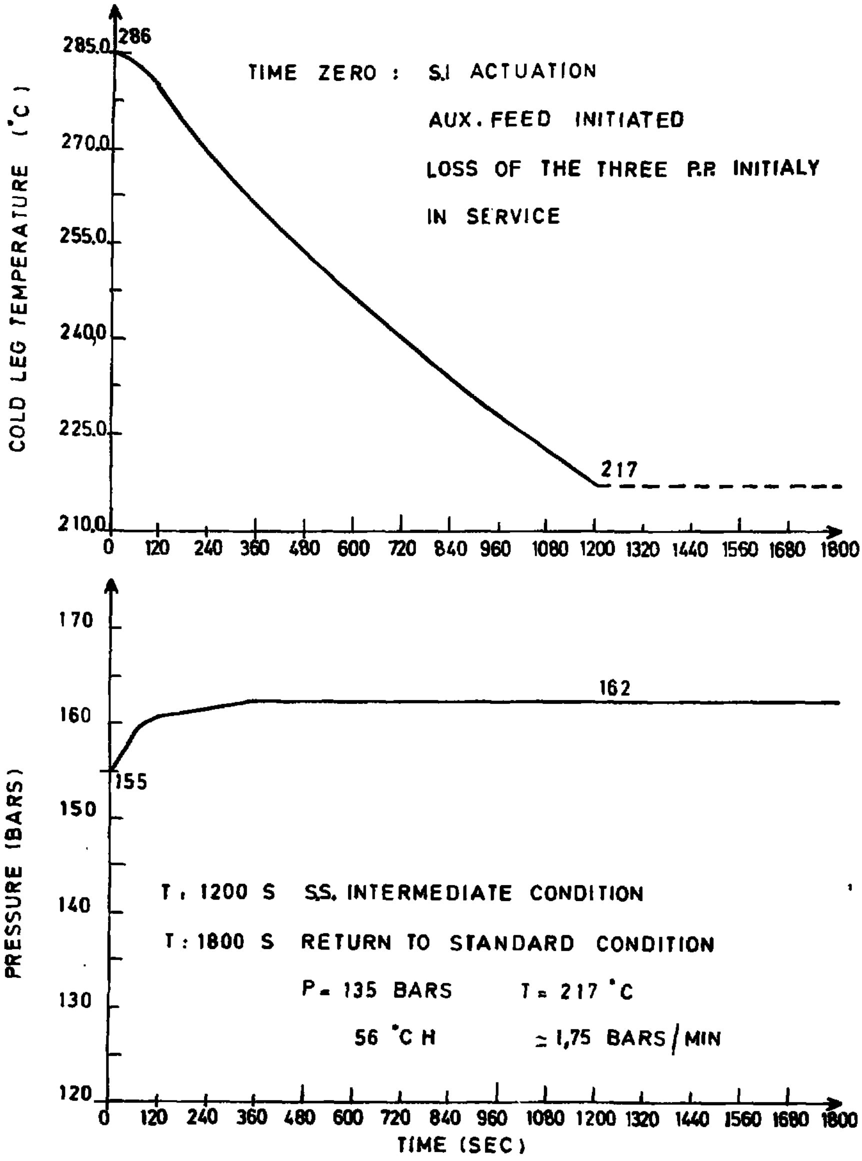
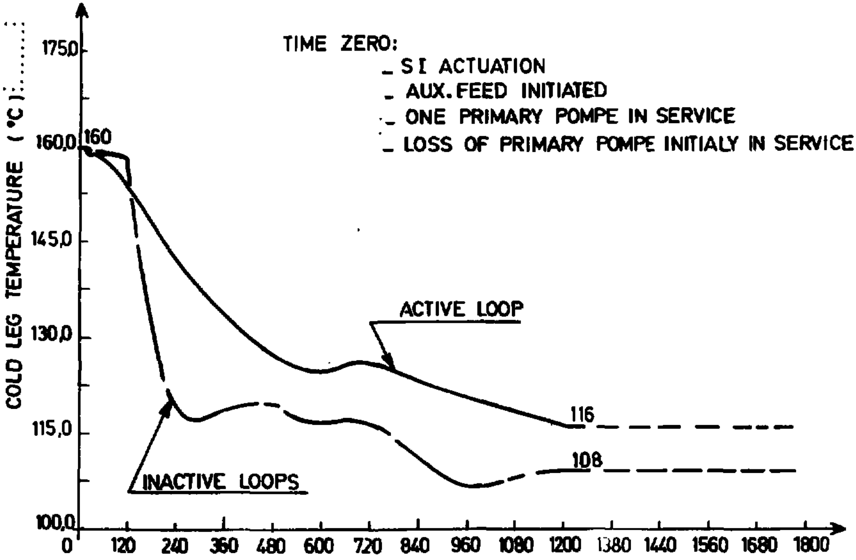


FIG: 3 INADVERTENT S.I WITH PARTIAL LOSS OF FLOW AT HOT STAND-BY



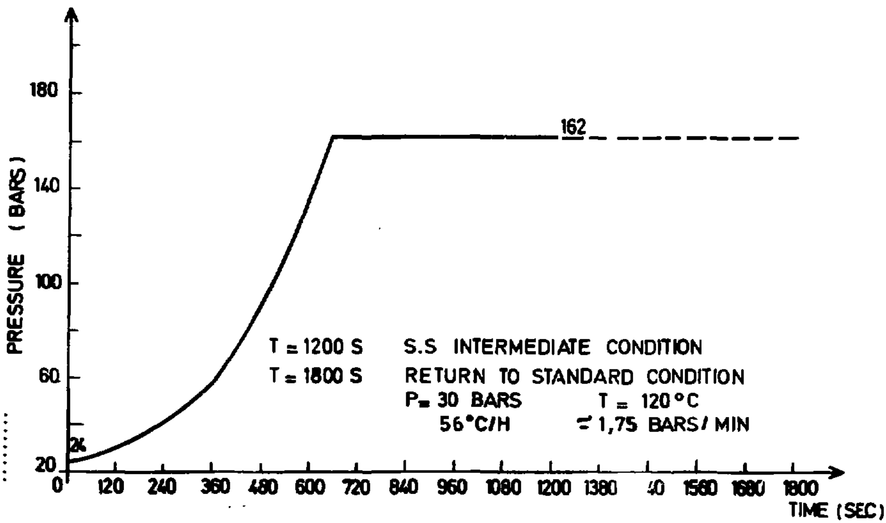


FIG: 4

_ INADVERTENT S.I WITH PARTIAL LOSS OF FLOW AT INTERMEDIATE CONDITIONS

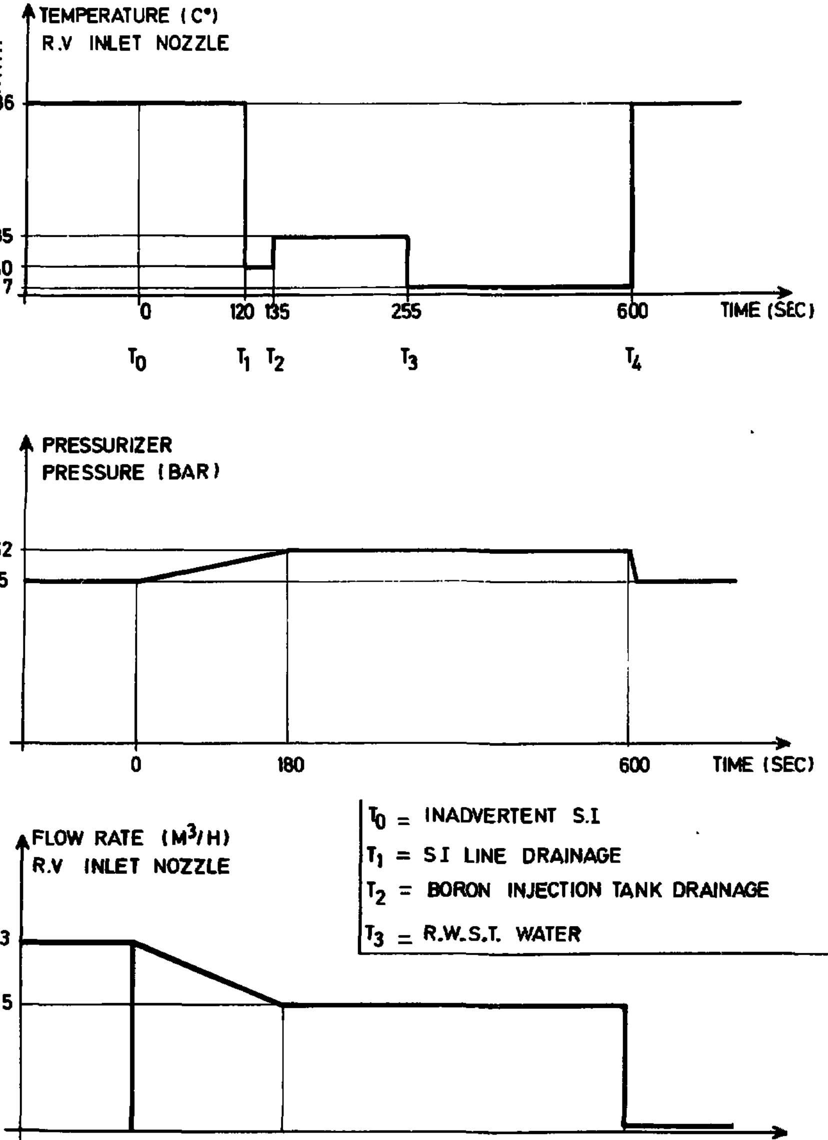


FIG: 5 _
INADVERTENT S.I WITH TOTAL LOSS OF FLOW AT HOT-STAND-BY

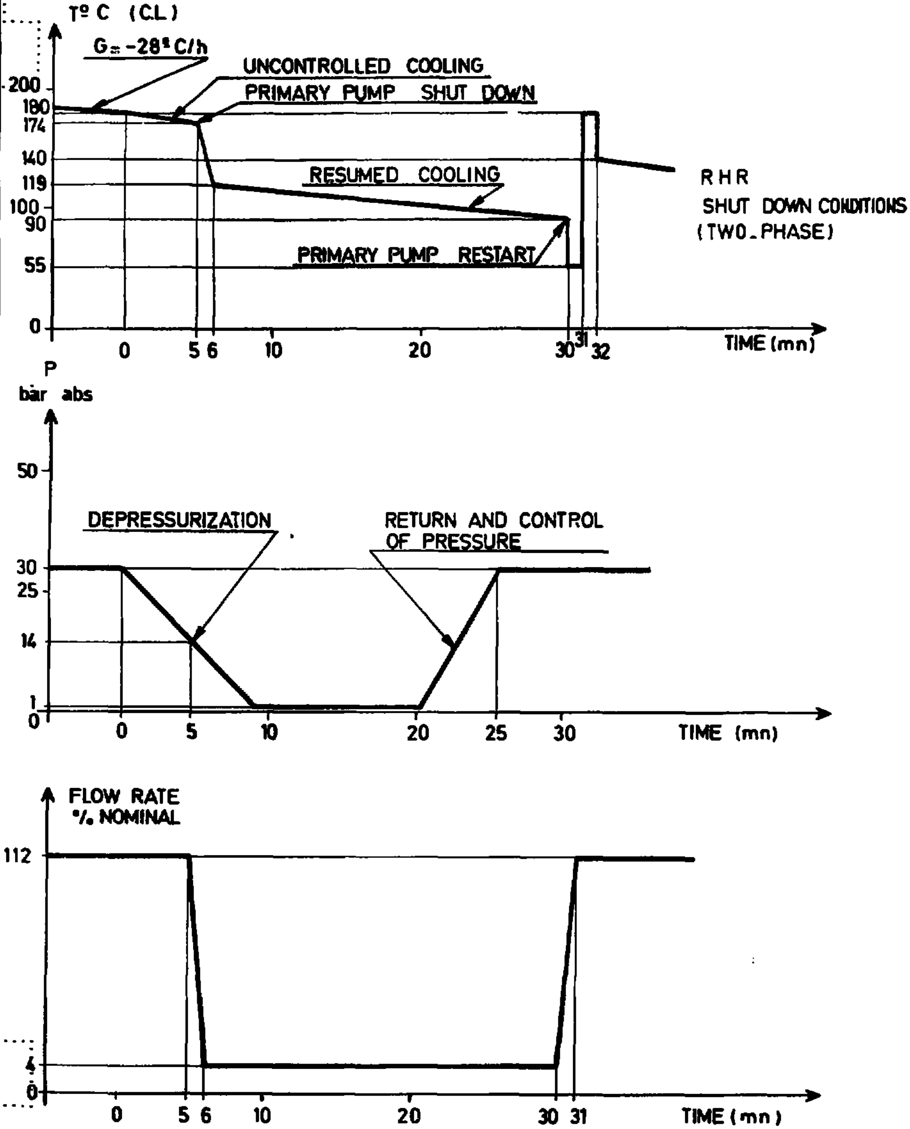


FIG : 6

INADVERTENT OPENING OF A RHR REGULATION VALVE AT TWO_PHASE COLD CONDITION

FRAMATOME - Tour Fiat - 1, Place de la Coupole - COURBEVOIE (Hauts-de-Seine)
12 FRAM 0373 08/79

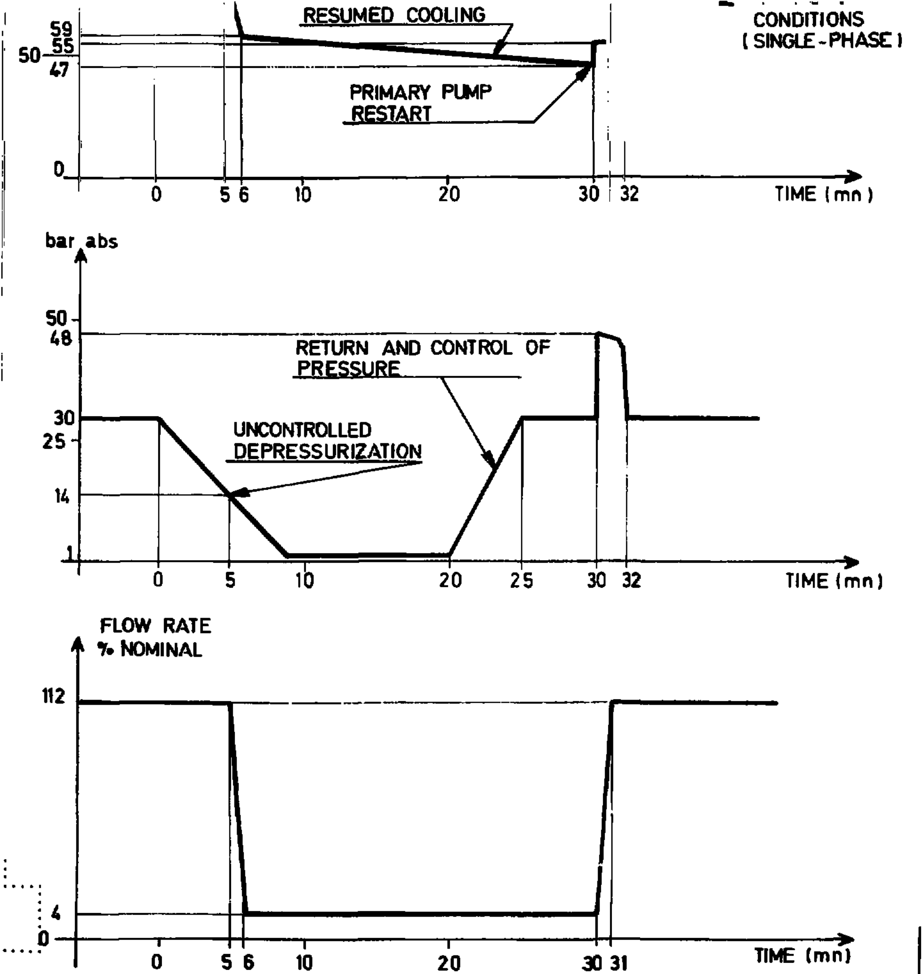
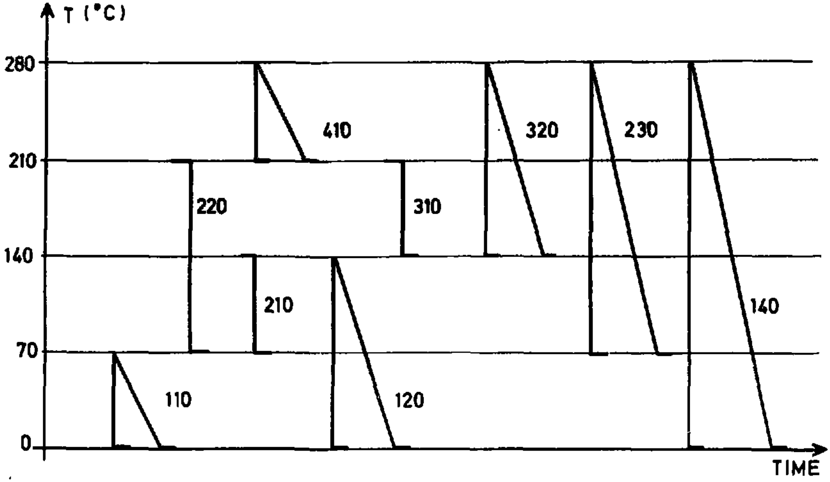


FIG: 7 _

INADVERTENT OPENING . OF R H R REGULATION VALVE AT SINGLE PHASE COLD CONDITION



PRESSURE :

- 15 MPa FOR ALL ADDITIONAL CASES
- 4 MPa FOR CASES 110 - 120 - 140 - 210 - 220 - 230 - 320 - 410
- 0 MPa FOR CASES 120 140

FIG. 8 -
FICTITIOUS TRANSIENTS USED TO DERIVE THE TRANSIENT SCREENING CRITERIA