OKB «Gidropress»
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Russian practice of RPV integrity assessment under PTS conditions

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INTRODUCTION

In this paper the approach used by Gidropress (main designer of Russian WWER reactors) for RPV integrity assessment is presented. Recently performed calculations for RPVs of Novoronezh NPP, units 3 and 4, are used as an example of practical application of this approach.

The calculations have been performed on the base of Russian regulatory requirements [1,2], at the same time the recommendations of IAEA Guidelines for PTS assessment [3] was also taken into account.

The scope of the work includes:

- Analysis of real state of NPP systems and PTS selection
- Analysis of material behavior including results of templets investigation
- Thermal hydraulic calculations
- Structural analyses for the leading transients
- Development of supplementary measures to reduce the risk of RPV fracture

The reactor pressure vessel drawing is presented in Fig. 1.

SELECTION OF THE LEADING SEQUENCES FOR THERMOHYDRAULIC ANALYSIS

The selection of leading sequences is performed on the basis of the list of design initial events which was developed for the WWER reactor plant safety analysis [4]. The experience of the other countries designing and operating PWR reactor plants and recommendations of IAEA given in the draft of Guide [3] are taken into account. The following groups of initial events should be extracted as potential unfavourable consequences for reactor vessel integrity:

1. **Decrease in reactor coolant inventory (LOCA):**

   - Inadvertent opening of one pressurizer safety valve.
   - Primary pipeline breaks:
   - Rupture of instrumentation pipelines;
   - Rupture of pipelines connected the primary side equipment units;
   - Rupture of main coolant pipelines.
   - Leaks from the primary to the secondary side of the SG:
   - SG tube rupture;
   - Primary collector leaks up to cover lift-up.

2. **Increase in reactor coolant inventory:**

   - Inadvertent actuation of ECCS during power operation;
   - Incorrect operation of make-up and blowdown primary system;
   - Pressurizer level control system malfunctions.

3. **Increase in heat removal by the secondary side:**

   - Inadvertent opening of one SG safety or relief valve or turbine bypass valve;
   - Spectrum of steam system piping break inside and outside of containment;
   - Rupture of feedwater pipeline;
   - Feedwater system malfunctions that decrease feedwater temperature;
   - Feedwater system malfunctions that increase feedwater flow rate;
   - Secondary pressure regulator malfunctions that increase steam flow rate.

4. **Accidents with the RPV cooling from outside:**

   - break of the biological shield tank.

The compilation of the list of initial events for each reference NPP is usually carried out using the engineering judgement and taking into account design features and performed modifications.

Proceed from the above mentioned list of initial events and taking into account real status of both plants the following accidents are considered for Novoronezh NPP, units 3 and 4:

1. Cold leg leak Du 32
2. Cold leg leak Du 20 (compensated break)
3. Hot leg leak Du 20 with later isolation of damage section
4. Inadvertent opening of one pressurizer safety valve
5. Inadvertent opening of one pressurizer safety valve with reclosure
6. Primary-to-secondary leaks Du 38 and 108
7. Primary blowdown collector break (Du 64 from lower plenum)
8. One SG steam line break
9. One SG steam line break with stuck open pressurizer safety valve
10. Inadvertent opening of one SG safety valve
11. Inadvertent opening of one BRU-A

The feature of these units is that HPIS pumps are connected into two hot legs.

The accident with the RPV cooling from outside is not considered, because of the construction of the biological shield tank is designed so that in case of seal failure water from the biological shield tank do not hit on the reactor vessel outside surface.

All accidents are considered for real status of both nuclear plant units taking into account available protections and interlockings. Also possibility of loss of off-site power and low residual heat level are taken into account in the most conservative way.

The examination of calculational results shows that the following consequences are the most important from the point of view of RPV integrity:

1.	One SG steam line break.
2.	Isolable primary leaks
3.	Inadvertent opening of pressurizer safety valve with closure at any time

Such accidents are characterised by high primary pressure due to HPIS pumps operation and low coolant temperature in downcomer.

Accident with steam line break is characterised by cold plum generation due to great heat removal in damaged SG (see Fig. 2). Isolation valve on steam line from emergency SG starts to close at 50 s after DG connection to the reliable power supply section (closing time is equal to 140 s). Two HPIS pumps start to operate at 177 s.

Reclosure of PRZ SV (flow rate capacity 10,5 kg/s) is considered in 1 hour after beginning of accident (see Fig. 3). Two HPIS pumps start to operate at 120 s.

Coolant temperatures at downcomer for different primary leaks are presented in Fig. 4.

DETERMINATION OF STRESSES IN THE VESSEL WALL

Stresses in the vessel wall in the region of grinding-out are determined with 3-D model by FEM . The computer code TACT was used for stress calculations. The isoparametrical curvilinear finite elements are used with 20 nodes and quadratic interpolating functions. Discrete model comprising 2810 finite elements is presented in Fig. 5. With regard for symmetry the 1/2 part of the reactor is simulated. Single grinding-outs in welds No 4 and No 5 are considered.

In the calculation the residual stresses in the welds were also taken into account. Distribution of residual stresses through the weld thickness is assumed in the form of:

$$\sigma_z = \sigma_\theta = 60 Cos(2\pi x/S)$$

where σ_z σ_θ - axial and circumferential stresses, **x** - thickness coordinate, **S** - vessel wall thickness.

PROCEDURE FOR EVALUATION OF THE ALLOWABLE CRITICAL BRITTLE FRACTURE TEMPERARURE

Evaluation of brittle fracture resistance of the RPV at the design stage is performed in accordance with the former Soviet Union "Standards for Strength Evaluation of Components and Piping of Nuclear Power Plants", PNAE G-7-002-86 [1]. The same approach is usually used for RPV residual lifetime evaluation for units under operation.

The evaluation is performed on the basis of the linear elastic fracture mechanics. The main characteristics of the materials used in the calculation are static fracture toughness, K_{IC}, and the critical brittle fracture temperature T_k as function of operational history (with respect to the material degradation). Change in material properties in the course of operation is taken into account by means of introducing the shifts of initial critical brittle fracture temperature T_k due to different operational effects (radiation embrittlement, thermal ageing, fatigue damage) in the calculation.

RPV resistance to brittle fracture during a particular plant state is considered to be ensured if, for all defect sizes up to the postulated quarter wall thickness size defect, the following condition is met:

$$K_I < [K_I]_i$$

where K_I is the intensity factor and $[K_I]_i$ is the allowable value of stress intensity factor for the plant state considered, i.e.:

$i = 1$ for normal operating conditions,
$i = 2$ for operational occurrences and hydraulic tests,
$i = 3$ for accident conditions.

Statistically evaluated lower envelope of all available experimental data is taken as the K_{IC} temperature dependence. Allowable stress intensity factors $[K_I]_i$ are obtained from the K_{IC} by applying safety factors:

- for normal operating conditions
$n_k = 2$, $\Delta T = 30$ °C,
- for operational occurrences and hydraulic tests
$n_k = 1.5$, $\Delta T = 30$ °C,
-for accident conditions
$n_k = 1.0$, $\Delta T = 0$ °C.

The n_k is a safety factor with respect to fracture toughness values and ΔT is a safety factor with respect to calculated crack tip temperature. The allowable stress intensity curve is obtained as a lower envelope of two curves, the first of which is obtained by dividing the K_{IC} by n_k and the other one by a horizontal shift of the initial curve by ΔT. The recommended temperature dependencies of $[K_I]_i$ for different RPV materials are given in the applicable standard [1].

Surface semi-elliptical cracks are postulated and with depth up to a=0.25 S (where S is the vessel wall thickness) and with aspect ratio a/c = 2/3. Stress intensity factor K_I, is determined using a formula given in the Standard [5], which takes into account real distribution of stresses in the defect depth. Mechanical as well as thermal and residual stress components are taken into account.

Comparing calculated loading path in terms of K_I values of the whole set of postulated defects with temperature dependencies of allowable values of stress intensity factors $[K_I]_i$, a maximum allowable critical brittle fracture temperature T_{ka}' for the analysed PTS sequence is obtained. The lowest of these temperatures for the whole set of analysed PTS sequences is taken as the maximum allowable critical brittle fracture temperature T_{ka}.

This temperature is then compared with the critical brittle fracture temperature T_k of the analyzed vessel. Based on this assessment, decisions on further operation, annealing, etc, could be made.

PROCEDURE FOR EVALUATION OF ALLOWABLE DEFECT SIZE

Evaluation of allowable defect size is performed in accordance with the procedure "Method for evaluation of allowability of defects in materials of components and pipings in NPPs during operation", M-02-91 [2].

This procedure is in principle divided into three main parts.

1. Defects, found during ISI are schematised using conservative approach, i.e. equivalent defect diameter obtained from ultrasonic tests is transformed into fatigue like crack with the same surface area but with semiaxis ratio a/c equal to 0.5 for internal defects (subsurface) and to 0.4 for surface defects, respectively. Detailed rules and formulas for evaluation of closely spaced defects or their groups are also given. All defects are defined as located in the plane perpendicular to the RPV surface as well as to the principal stresses.

2. Possible growth of the given defect due to operational loading is calculated. The standard provides the coefficients of Paris law for RPV materials. This calculated defect growth for the whole remaining lifetime is added to the initially schematised defect sizes

3. Calculation of defect allowability is then performed using a complex approach, including linear elastic fracture mechanics, elastic-plastic fracture mechanics as well as theory of plasticity.

The following safety factors for different operational conditions are used:

- for normal operating conditions
$n_k = 3$, $\Delta T = 30$ °C,
- for operational occurrences and hydraulic tests
$n_k = 2$, $\Delta T = 20$ °C,
-for accident conditions
$n_k = 1.4$, $\Delta T = 10$ °C.

Values of K_I are calculated both for the crack deepest point as well as for the intersection with the surface for a surface defect or the closest point to the inner surface for an internal defects. These values are then compared with allowable values of fracture toughness derived from static fracture toughness K_{IC} used in the standard [1].

The following procedure based on requirements of the Standard M-02-91 [3] was used for evaluation of allowable defect size in RPVs of units 3 and 4 of Novovoronezh NPP.

The spectrum of postulated defects of different sizes of the most hazardous type in the most stressed zone of the weld is selected. In the present calculation such defects are considered to be the surface semi-elliptical cracks of different depths with ratio of semi-axes being **a/c=0,4** [2], located in the pole of grinding-out (the zone with maximum stress concentration).

For each calculated crack the temperature boundaries of the zones of brittle, quasi-brittle and plastic mechanisms of failure are determined as per the criteria of /1/. Hereat as the temperature dependence $K_{IC}=f(T-T_K)$ the curve $[K_I]_3$ is used for the welds of steel 15Kh2MFA from [1].

For each calculated time moment of the considered conditions the stresses in the region of calculated cracks are determined and, using methods of [4] the values of K_I are calculated for two points of the crack front: the point at maximum depth and the point in the region of the crack front outgoing to the free surface. Hereat the crack is located in the plane perpendicular to the direction of action of maximum tensile stresses.

For the crack front points indicated above by metal temperature the character of failure is determined (brittle, quasi-brittle or plastic) and the permissible value of $[K_I]$ with regard for the character of failure and margins for the conditions of the category considered (NOC, OO, AS). The permissible value of $[K_I]$ is determined by formula presented in [2].

By means of comparison of K_I and $[K_I]$ values for all time moments of all considered conditions the maximum depth of the permissible crack $[a]_c$ is determined from the condition of $K_I = [K_I.]$

By means of calculation of the crack kinetics [2] the depth of the initial crack **[a]** is determined, which may extend to the size of $[a]_c$ due to cyclic loads for four years of operation (time interval between ISI).

On the basis of recommendations of [5] on schematisation of defects, detected in the course of in-service inspection, the minimum areas **[F]** of surface defects and subsurface (located close to the surface) defect are determined which shall be schematised by the surface semi-elliptical crack of **a= [a]** depth and ratio of semi-axes **a/c=0,4**.

Subsurface defects of larger depth of occurrence in accordance with [5] are schematised by subsurface elliptical cracks. As the values of K_I for the subsurface cracks with the same field of stresses is considerably lower than for the surface cracks, and also taking into account the fact that stresses under emergency cooldowns drop quickly with increasing the distance from the inner surface into the depth of thickness, the permissible area of such defects will be considerably larger than the permissible are of surface and subsurface defects determined above.

RESULTS OF CALCULATIONS FOR THE LEADING TRANSIENT «STEAM GENERATOR STEAMLINE BREAK»

Results of calculation T_{ka} using the procedure from PNAE G-7-002-86 [1] for welds No 4 and 5 are presented in Table 1. Figs 6-7 give the results of calculation K_I, the position of the curve of static fracture toughness K_{IC} is also shown there at $T_k = T_{ka}$.

Table 1

Results of calculation $T_{\kappa a}$ using the procedure PNAE G-7-002-86 for the conditions «Break of steam generator steamline»

Section	Residual stresses	$T_{\kappa a}$ for weld No 5 °C	$T_{\kappa a}$ for weld No 4 °C
grinding-out	60 MPa	163	176
grinding-out	no	-	183
without grinding-out	60 MPa	169	182
without grinding-out	no	-	190

Permissible sizes of defects in weld No 4 are determined depending on T_k with regard for cyclic extension for the preceding 4 fuel cycles (interval between non-destructive examination of the weld metal). The calculations were performed both with the use of safety factor n_k=1.4, ΔT=10°C [2], and without these factors. In the latter case for obtaining the permissible defect the safety factor 2 should be applied for sizes of the calculated defect. In accordance with c IAEA recommendations the least of two defects, obtained in such a way, shall be assumed. It follows from the results obtained that application of safety factors of n_k=1.4, ΔT=10°C is ,in the given case, more conservative.

Analysis of sensitivity of the results to different factors has also been performed. The specific care was paid to effect of aspect ratio of the crack It follows from the results of calculations that if the requirements for sensitivity of ultrasonic inspection system are worded in terms of equivalent area of defect (in accordance with the Russian standards), then the use of relation of semi-axes of the postulated crack **a/c=0.4** (in accordance with [2]) is more conservative, than that assumed in Western countries **a/c=0.3**, because it results in somewhat less equivalent area of the permissible defect.

RESULTS FOR ISOLABLE PRIMARY LEAKS

As it follows from the analysis of results of thermohydraulic calculations in case of leak isolation the rapid primary pressure increase takes place to the value of the head of the emergency makeup pumps. At the same time with the pressure increase the gradual rise of coolant temperature in the reactor pressure chamber begins due to loss of cold water supply from the emergency makeup pumps.

The calculations for these conditions were made in the form of determination of permissible pressure in the reactor vessel depending on the time of the process and depending on the coolant temperature in the reactor pressure chamber that allows to determine the limiting conditions of the mode which could be laid down in the basis of elaboration of the instructions for the operator or an algorithm of operation of automatic system of the reactor vessel protection against cold overpressure.

The lower envelope coolant temperature curve for all considered transients was used for this calculation (presented in Fig. 4).

Calculation of the permissible pressure was performed both as per the criteria of PNAE G-7-002-86 [1], and with the use of methods M-02-91 [2]. In the latter case the calculation was made for the postulated crack of 5mm depth with the use of safety factors of n_k=1.4; ΔT=10°C that is more conservative than using the postulated crack of 10mm depth with the following using of safety factor 2 for the size of reliably detected defect.

Results of the calculation are given in the form of graphs in Fig 8.

Specific attention was paid to the transient «Inadvertent opening of Prz safety valve». According to IAEA recommendation a possibility of safety valve closing at any time moment of the accident was considered. Results of the calculation for this transient are given in the form of graphs in Fig 9 .

It follows from the results presented in Figs 8-9 that the repeat pressure increase in the reactor vessel, cause by leak isolation, could result in severe situation from the viewpoint of reactor vessel brittle strength at reaching the high values of $\mathbf{T_K}$ of weld No 4 material if closing of the valve takes place at low temperature of the primary coolant. Special measures are to be taken on prevention of the reactor vessel cold overpessure. Such measures may be operator's interference or introduction of automatic system of the reactor protection against cold overpressure.

Operator could make an attempt to isolate the leak without the risk of hazardous overpressure of the reactor vessel only till the primary coolant temperature goes down the definite value depending on the current ductile-to-brittle transition temperature of the reactor vessel material.

The relevant requirements have been included into instructions for operator.

MAIN CONCLUSIONS OF ANALYSES PERFORMED

• The conditions "Break of steam generator steamline" is the leading accident governing the reactor vessel service lifetime. Permissible ductile-to-brittle transition temperature of weld No 4 material $\mathbf{T_{Ka}}$, determined for the worst conditions (with regard for stress concentration in the region of grinding-outs after taking templets, with regard for residual stresses in the weld) is $\mathbf{T_{Ka}}$=176°C for such mode.

• The primary leaks, in case of their isolation (by operator's actions or due to spontaneous closing of Prz safety valve after its failure to fit) need special care. The repeat increase of the reactor vessel pressure, caused by leak isolation, could result in severe situation from the viewpoint of the reactor vessel brittle strength at reaching the high values of $\mathbf{T_K}$ of weld No 4 material, if isolation of leak is performed at low temperature of the primary coolant. The special measures are to be taken on prevention of cold overpressure of the reactor vessel.

• To provide for safe operation of the reactor vessels during the design service life it is necessary to apply the system of non-destructive examination of the reactor vessel that allows to detect reliably the small crack-type surface and subsurface defects in the vessel cylindrical part (including circumferential welds).

It should be noted that the same approach was used by Gidropress for integrity assessment of Kozloduy NPP, unit 1, under leading PTS conditions. Independent calculations had been performed for this RPV by Siemens and Westinghouse using Western approaches. The final results obtained by all three companies (in terms of allowable transition temperature $\mathbf{T_{ka}}$) were very close. The $\mathbf{T_{ka}}$ values were obtained within the range of 173-180°C.

REFERENCES

1. Norms for strength calculation of equipment and pipelines of nuclear power plants, PNAE G-7-002-86, Moscow, Energoatomizdat, 1989.

2. Methods for detection of permissible defects in the metal of equipment and pipelines during NPP operation, M-02-91, Moscow, 1991.

3. Guidelines on the reactor pressure vessel PTS assessment for WWER nuclear power plants (DRAFT), WWER-SC-157, IAEA, August 1996.

4. Требования к содержанию отчета по безопасности АС с реакторами типа ВВЭР, ПНАЭ Г-01-036-95, Госатомнадзор России, 1995.

5. Methodical recommendations. Power-generating plants. Calculations and tests for strength. Calculation of stress intensity factors, MR 108.7-86, NPO TZNIITMASH, Moscow, 1986.

REACTOR PRESSURE VESSEL (B-179)

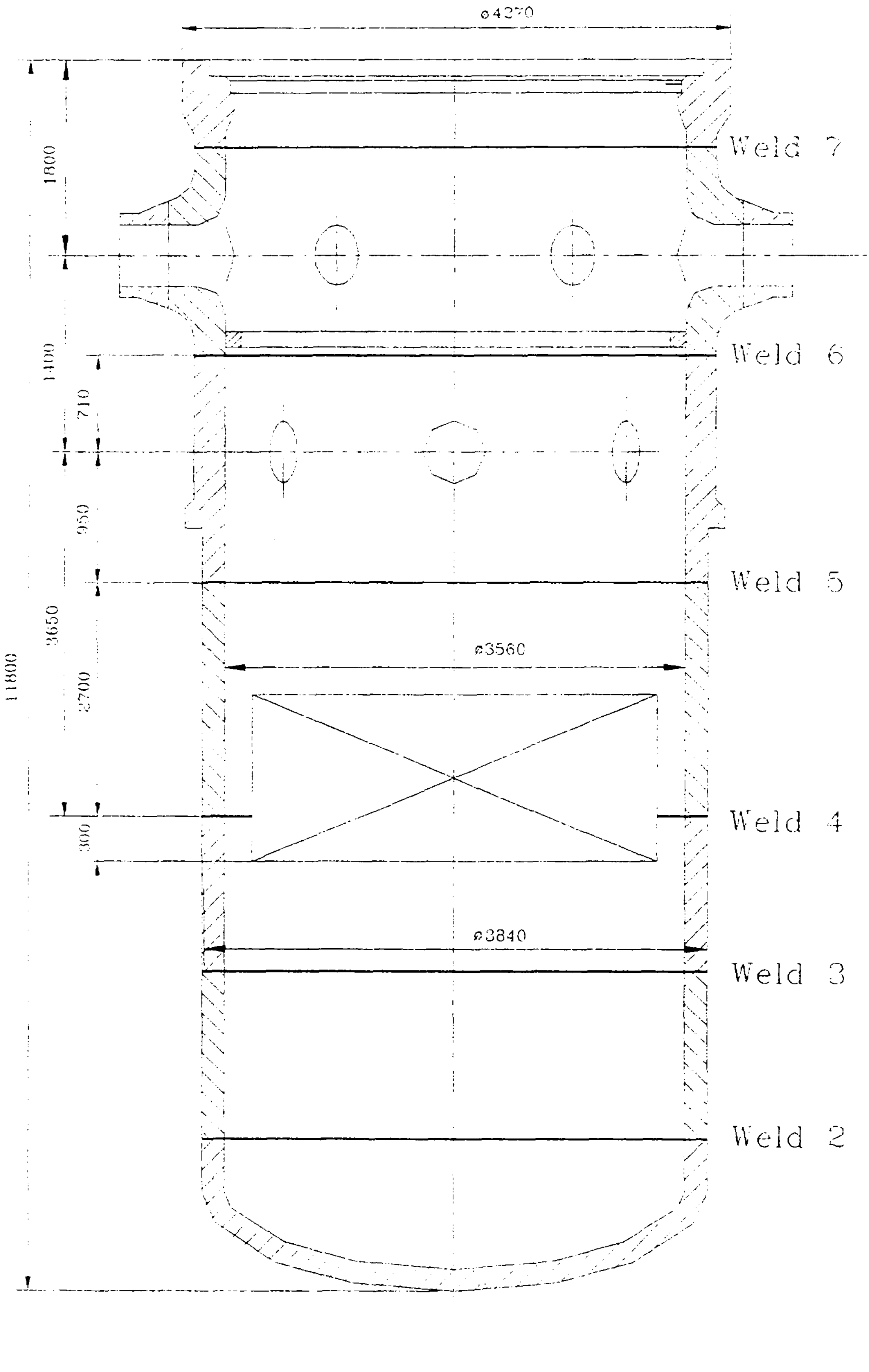


Fig 1.

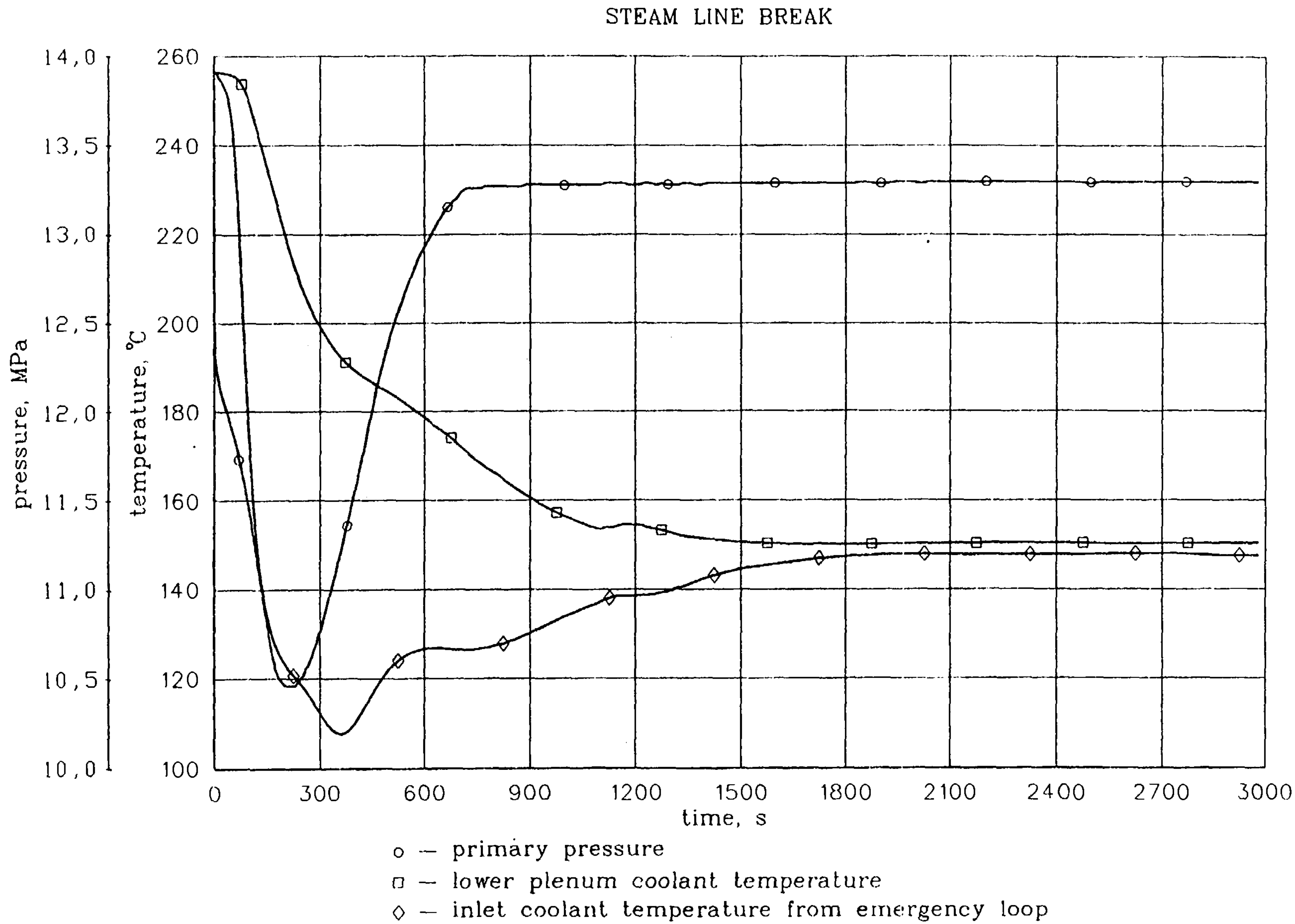
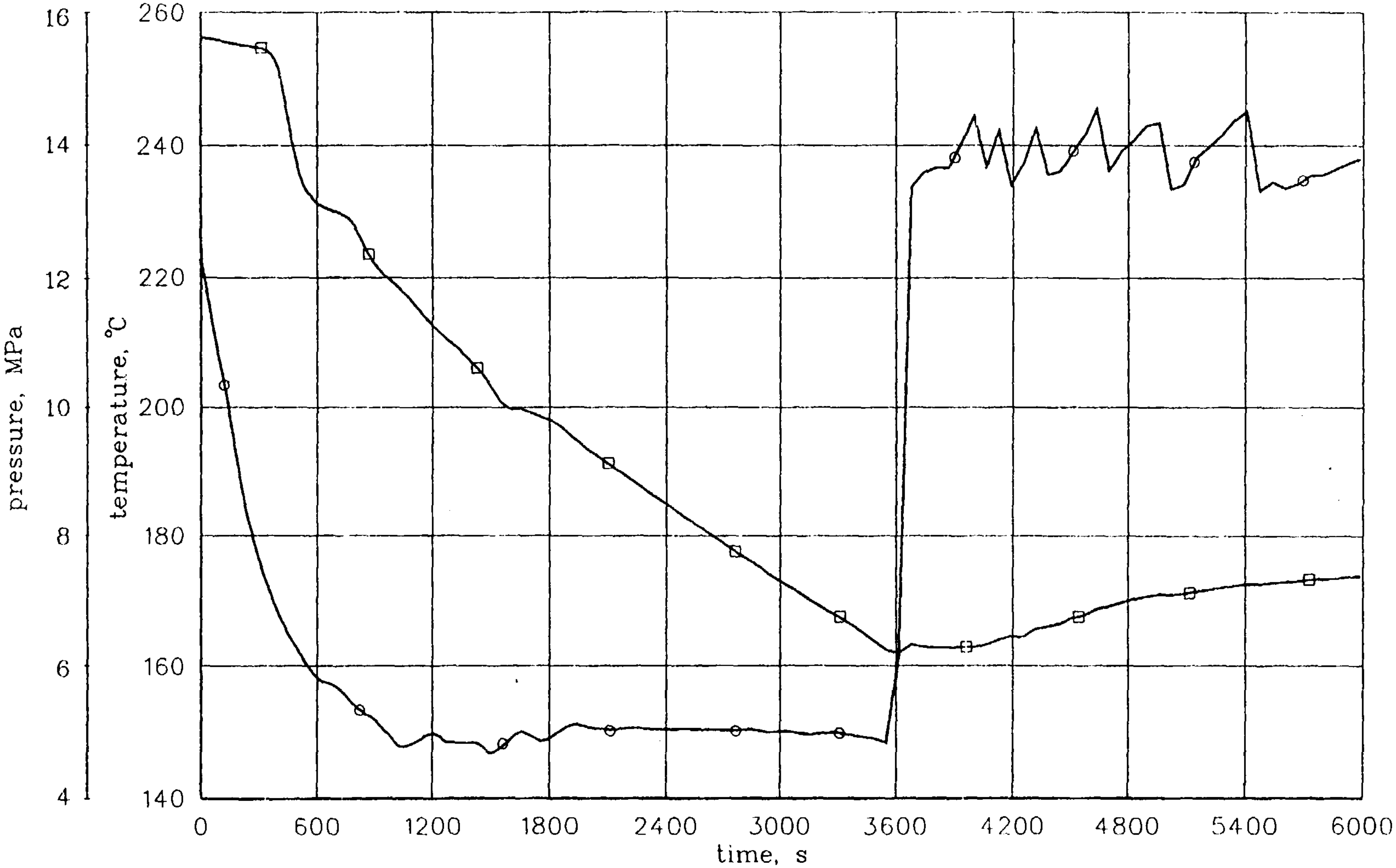

STEAM LINE BREAK
pressure, MPa
temperature, °C
14,0
13,5
13,0
12,5
12,0
11,5
11,0
10,5
10,0
260
240
220
200
180
160
140
120
100
0
300
600
900
1200
1500
1800
2100
2400
2700
3000
time, s
o — primary pressure
□ — lower plenum coolant temperature
◇ — inlet coolant temperature from emergency loop

Fig 2

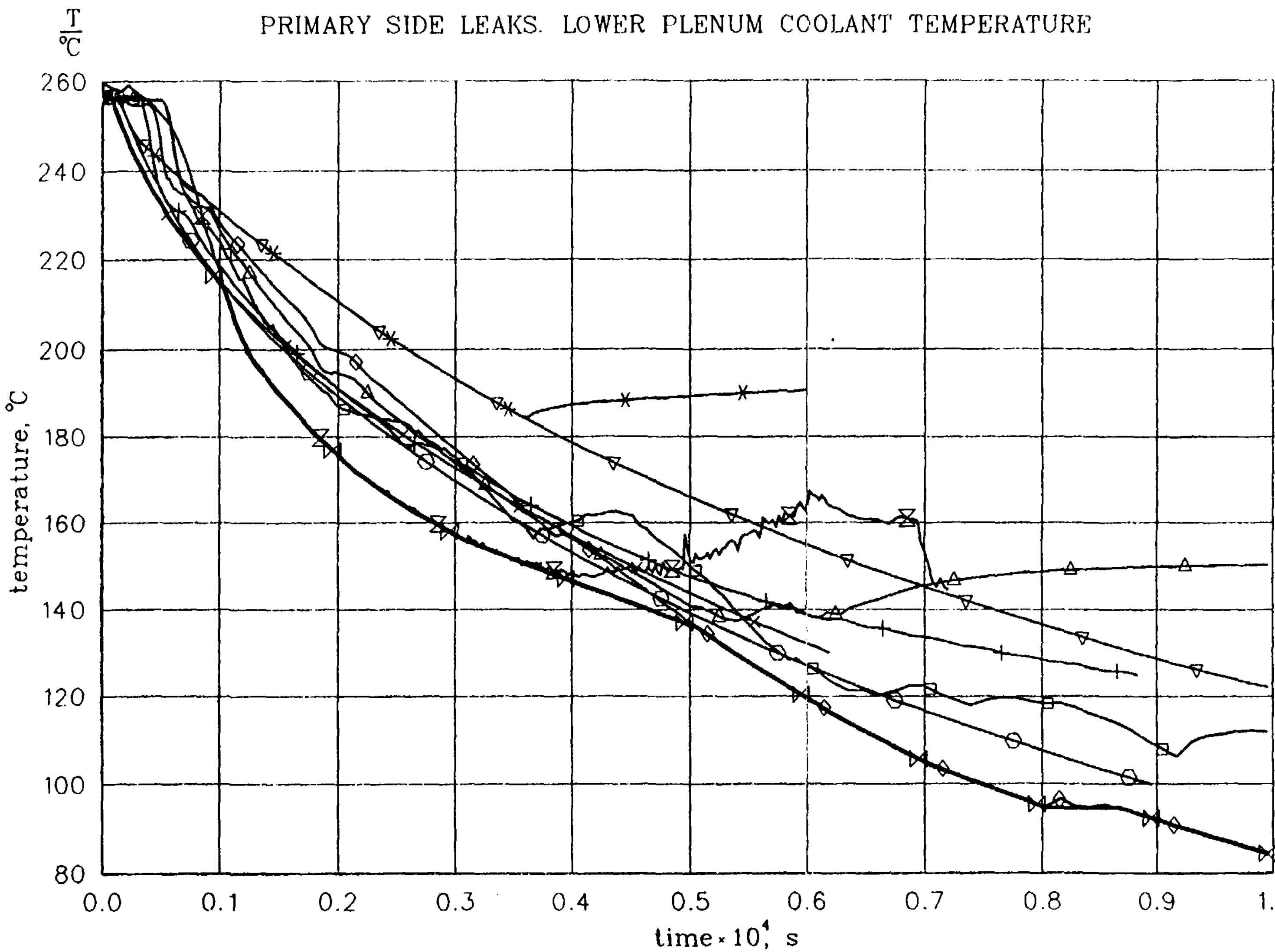
INADVERTENT OPENING OF PRZ SV WITH RECLOSURE



o — primary pressure
□ — lower plenum coolant temperature

Fig 3

PRIMARY SIDE LEAKS. LOWER PLENUM COOLANT TEMPERATURE



□ – cold leg leak Du 32
◇ – cold leg leak Du 20
△ – hot leg leak Du 20
▽ – inadvertent opening of PRZ SV
✳ – inadvertent opening of PRZ SV with reclosure
× – primary-to-secondary leak Du 108 without LOOP
+ – primary-to-secondary leak Du 108 with LOOP
○ – primary-to-secondary leak Du 38
⧖ – primary side blowdown collector break
⋈ – envelope curve

Fig 4

RPV of Novovoronezh 3,4
Finite element mesh

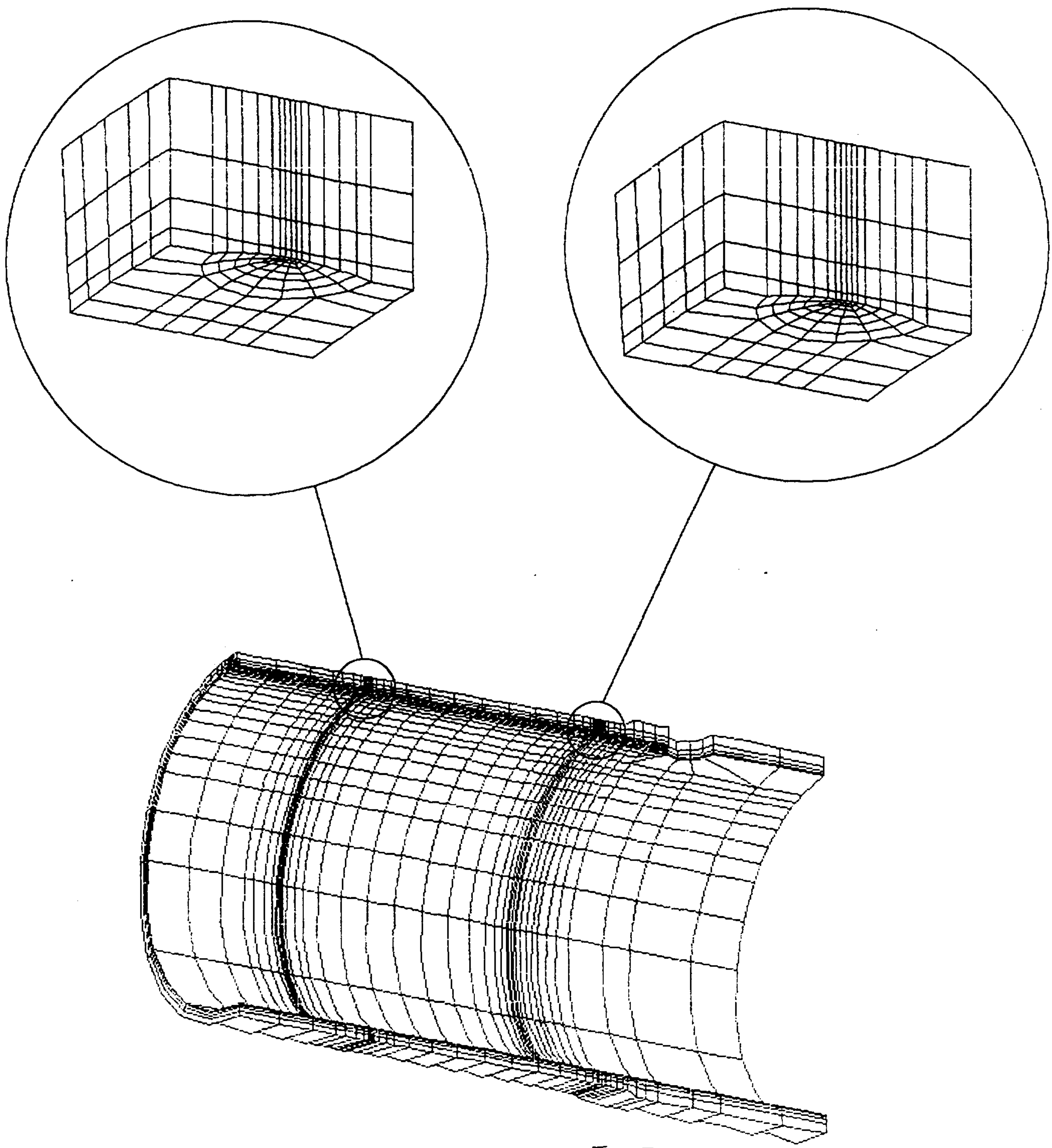


Fig 5

Stem generator steamline break.
Weld 4. Gringing section. Residual stress 60MPa.
Axial surfase cracks with a/c=2/3.
Deepest point of crack front

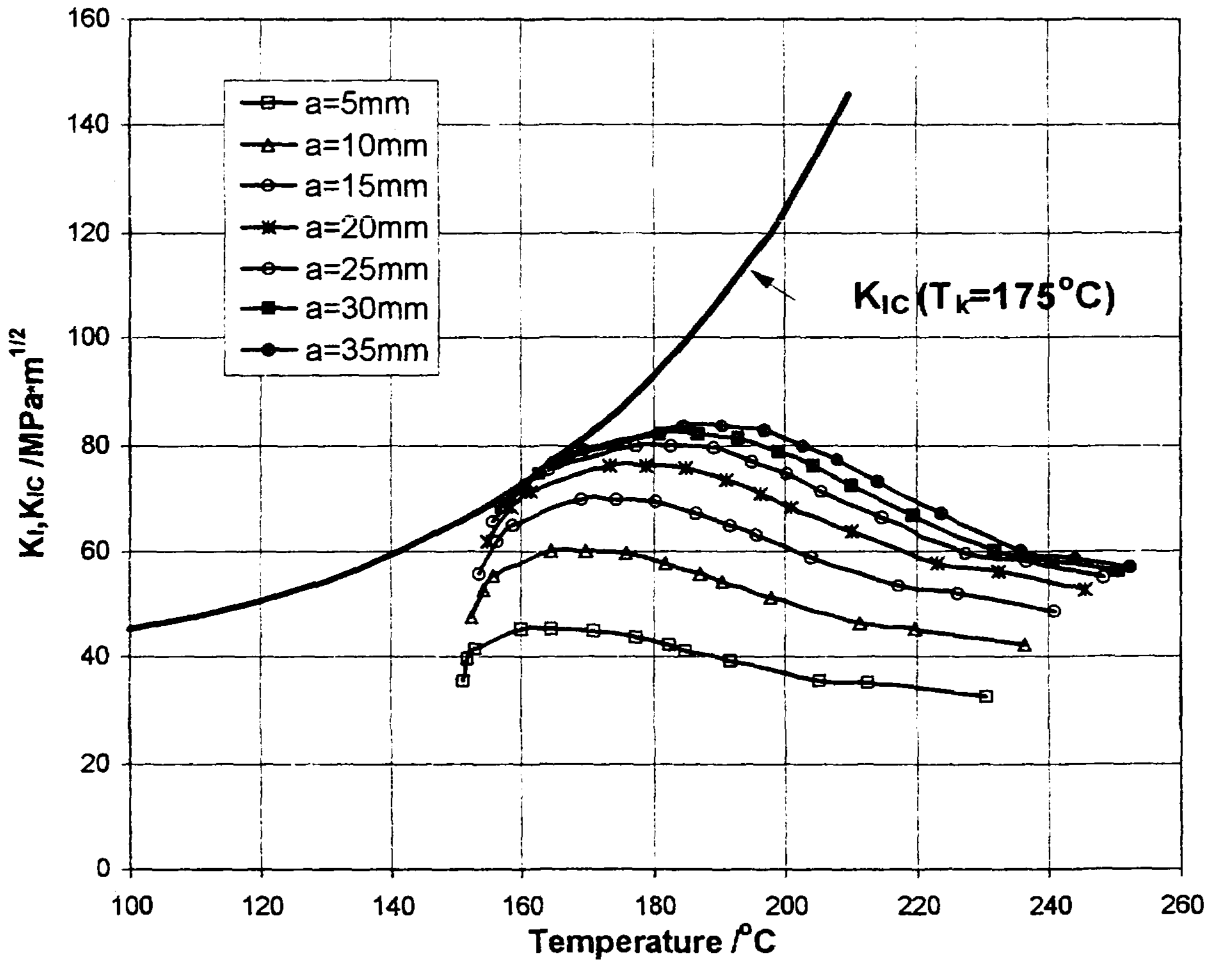


Fig. 6

**Stem generator steamline break.
Weld 5. Gringing section. Residual stress 60MPa.
Axial surfase cracks with a/c=2/3.
Deepest point of crack front**

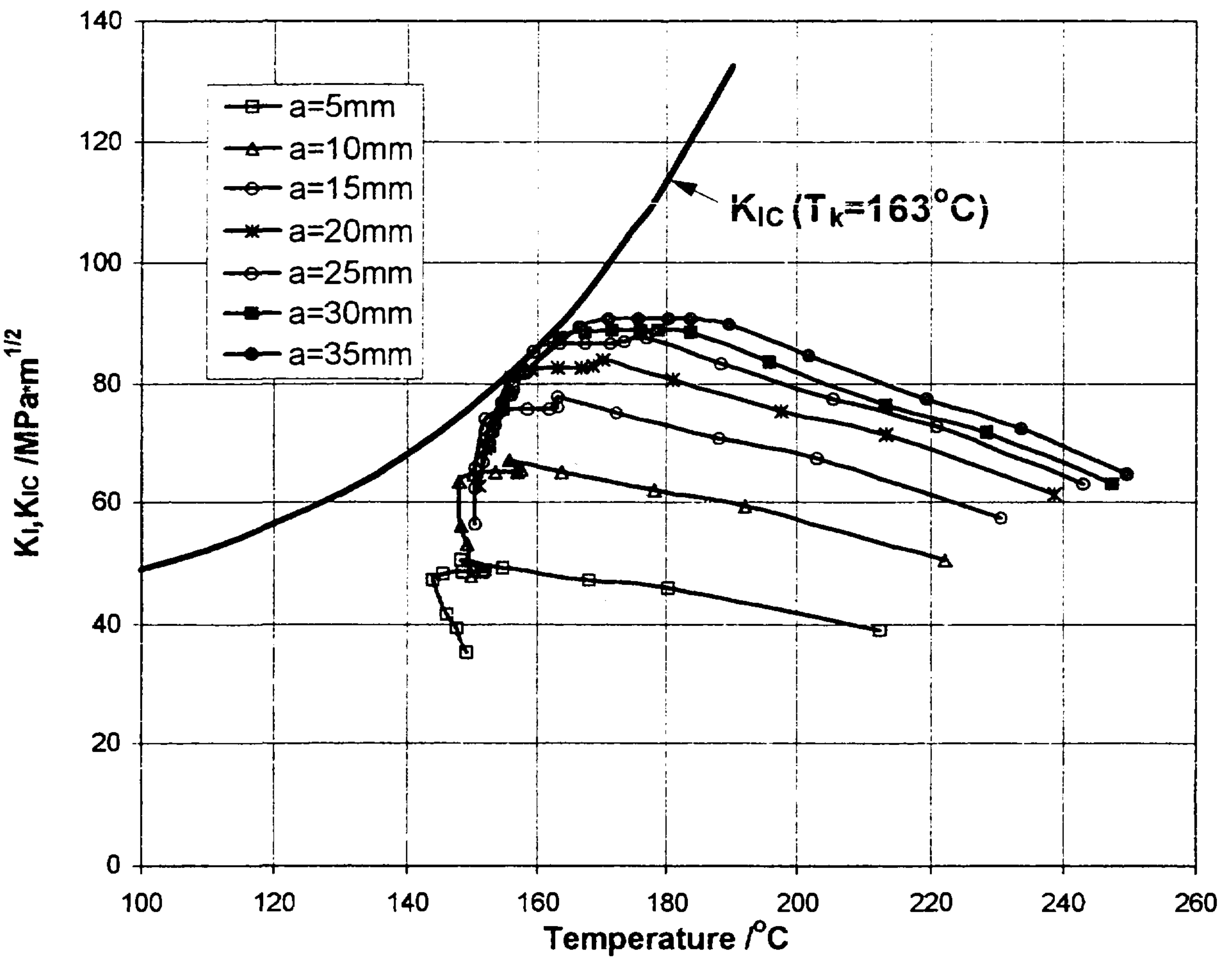
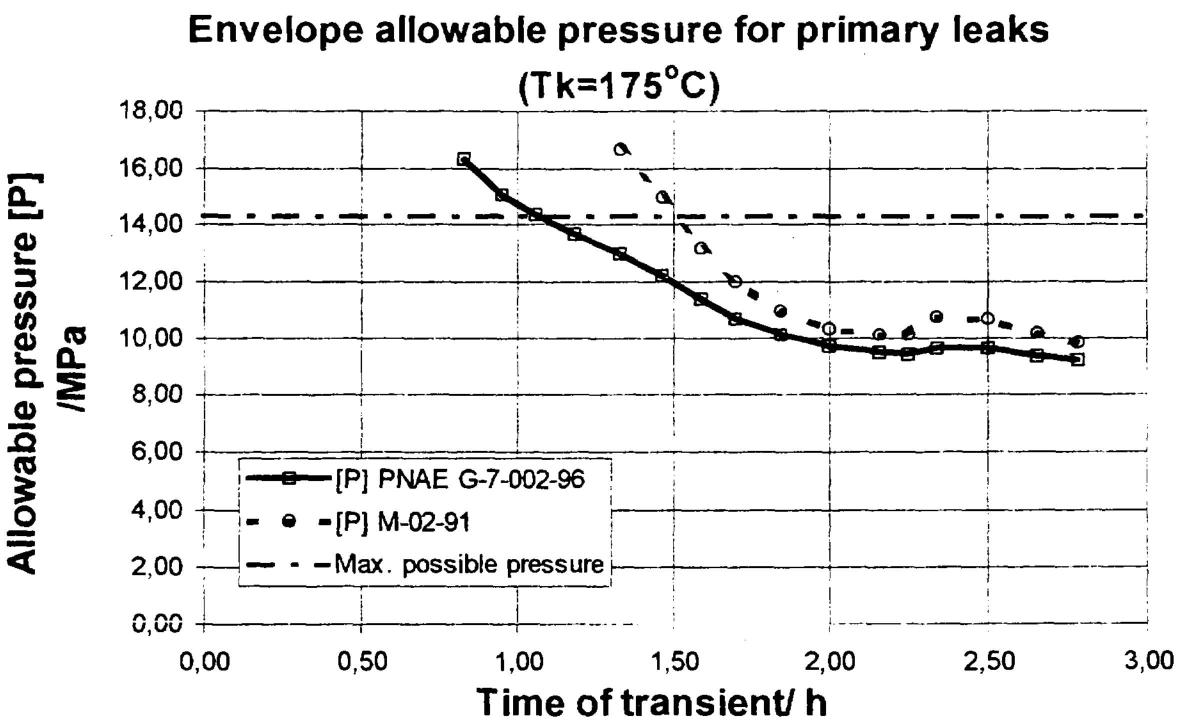
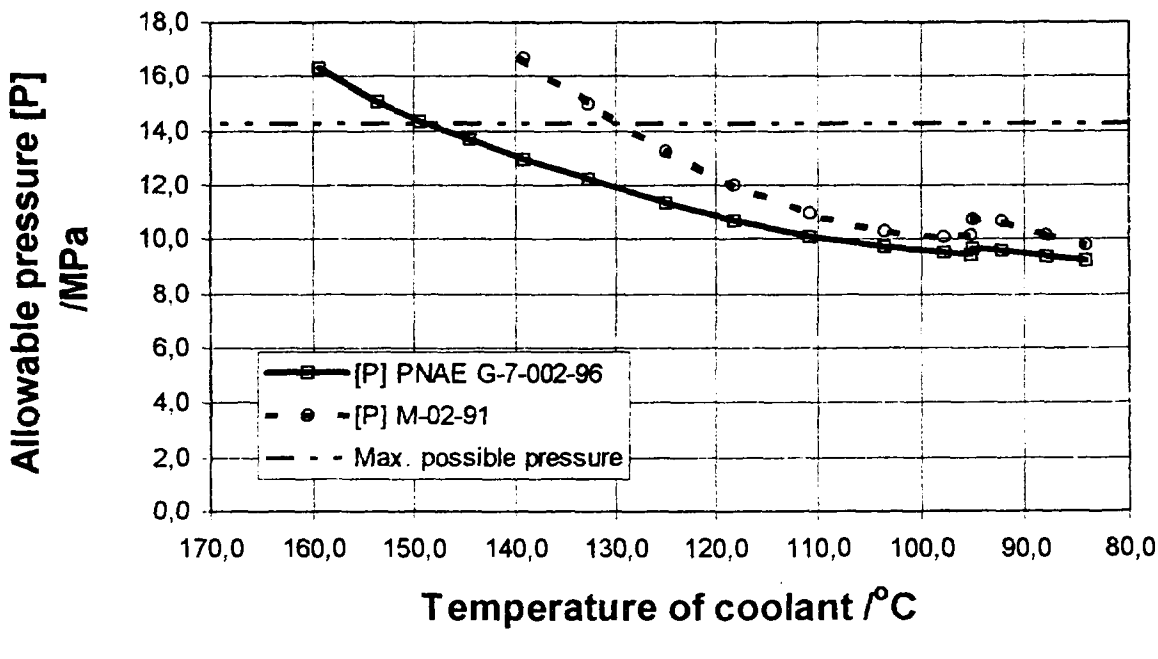


Fig. 7

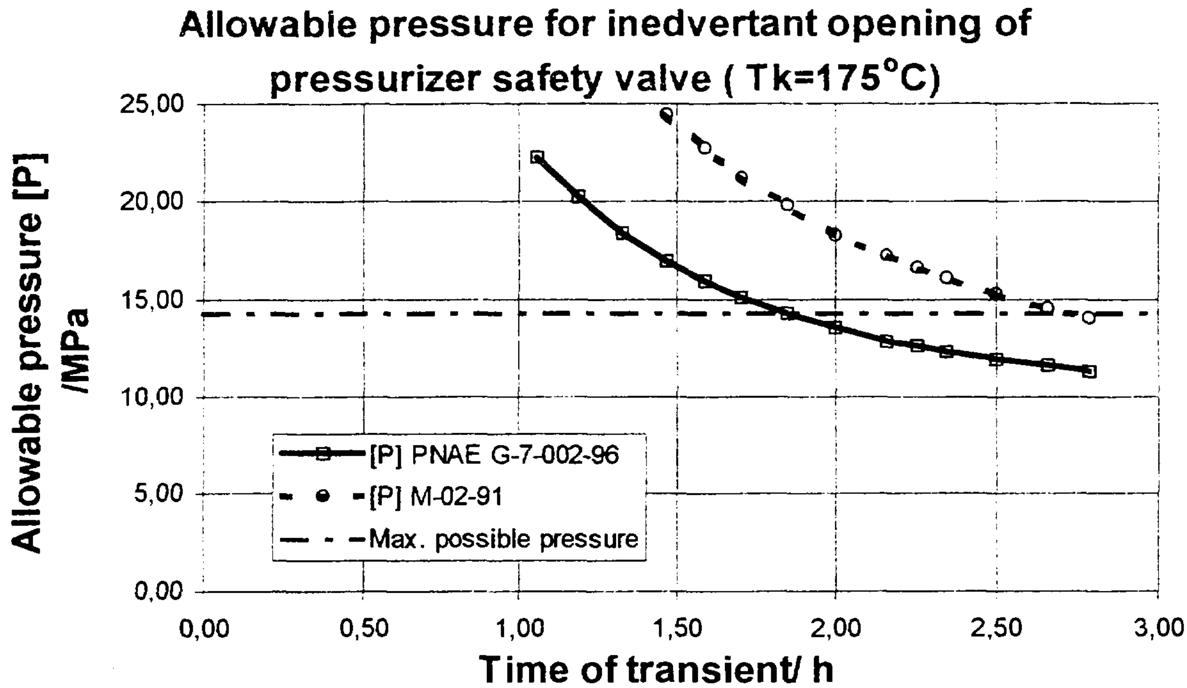


Envelope allowable pressure for primary leaks
(Tk=175°C)
Allowable pressure [P] /MPa
18,00
16,00
14,00
12,00
10,00
8,00
6,00
4,00
2,00
0,00
0,00
0,50
1,00
1,50
2,00
2,50
3,00
Time of transient/ h
[P] PNAE G-7-002-96
[P] M-02-91
Max. possible pressure



Allowable pressure [P] /MPa
18,0
16,0
14,0
12,0
10,0
8,0
6,0
4,0
2,0
0,0
170,0
160,0
150,0
140,0
130,0
120,0
110,0
100,0
90,0
80,0
Temperature of coolant /°C
[P] PNAE G-7-002-96
[P] M-02-91
Max. possible pressure

Fig. 8


Allowable pressure for inedvertant opening of pressurizer safety valve (Tk=175°C)
Allowable pressure [P] /MPa
Time of transient/ h
25,00
20,00
15,00
10,00
5,00
0,00
0,00
0,50
1,00
1,50
2,00
2,50
3,00
[P] PNAE G-7-002-96
[P] M-02-91
Max. possible pressure

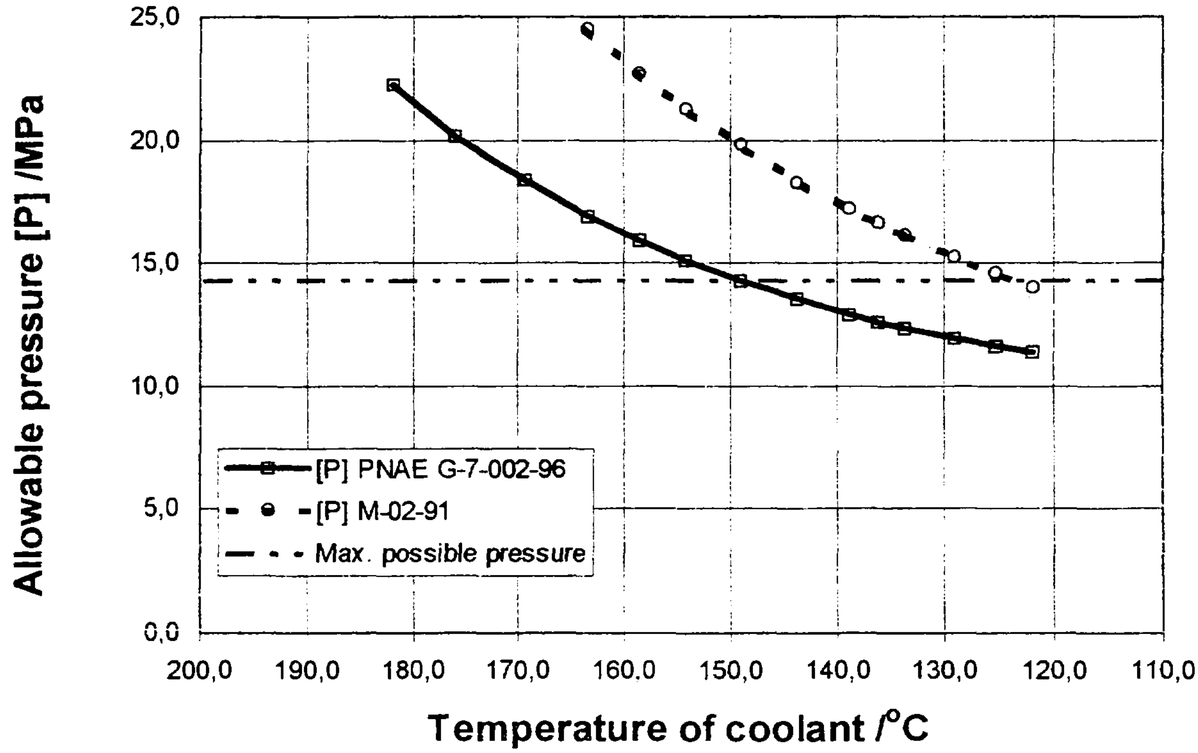

Allowable pressure [P] /MPa
Temperature of coolant /°C
25,0
20,0
15,0
10,0
5,0
0,0
200,0
190,0
180,0
170,0
160,0
150,0
140,0
130,0
120,0
110,0
[P] PNAE G-7-002-96
[P] M-02-91
Max. possible pressure

Fig. 9