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EVALUATION OF THE PTS POTENTIAL IN A WWER-1000 FOLLOWING A STEAM LINE BREAK

M. Beghini, F. D'Auria, G.M. Galassi, E. Vitale

Università degli Studi di Pisa
Dipartimento di Costruzioni Meccaniche e Nucleari
Via Diotisalvi 2, 56100 Pisa - Italy

fax ..39-50-585.265

ABSTRACT

A qualified nodalization for WWER-1000 is available at DCMN (Dipartimento di Costruzioni Meccaniche e Nucleari) of University of Pisa that is suitable for running with the thermohydraulic system code Relap5/mod3.2. The nodalization consists of about 1400 hydraulic nodes and more than 5000 mesh points for conduction heat transfer. The four loops of the NPP are separately modelled.

Detailed information about the plant hardware has been got from contacts with Eastern Organizations in Bulgaria, Russia and Ukraine. The qualification of the nodalization has been achieved at a steady state level utilizing a procedure available at DCMN and at a transient level on the basis of operational (planned) transients performed in the Bulgarian Kozloduy-5 NPP and of the unplanned transient occurred at the Ukrainian Zaporosche NPP (April 1995). Data measured in steam generators have also been utilized.

The nodalization has been widely applied to the analysis of accident scenarios in WWER-1000, including Large Break LOCA, Small Break LOCA, ATWS, Loss of Feedwater and Station Blackout.

The present activity aims at evaluating the potential for PTS (Pressurized Thermal Shock) following a steam line break accident. The thermalhydraulic results were employed as input for a parametric Fracture Mechanics analysis based mon conservative hypothesis of the shape and localization of a pre-existing defect. Stress analysis evidenced the effect of partial cooling of the vessel and gave some general indications of the risk for unstable crack propagation under the simulated PTS conditions.

1. INTRODUCTION

The safety of nuclear power plants constitutes a global concern on the planet. This justifies interchanges of competences and information between Western and Eastern Countries in the nuclear technology area. Specifically, applications of advanced best estimate system codes to the evaluation of safety margins of Russian type WWER plants, are part of such a context.

Considering the experience gained in the validation and use of the Relap5 and Cathare codes, e.g. (D'Auria and Galassi, 1990a, D'Auria and Galassi, 1990b, Bernard and D'Auria, 1990) including applications to WWER related phenomena, e.g. (Mavko et al., 1995) (D'Auria et al., 1995) and the scientific cooperation with Energoproekt of Sofia (BG), the decision was taken to develop a WWER-1000 detailed nodalization. In the initial stage of the activity, Relap5/m3.2 (Fletcher and Schultz, 1995) code was selected. The reference plant hardware information is included in (Kolev et al., 1992), (Hinovsky, 1995) and (US DOE, 1987) nominal operating conditions and Emergency Core Cooling Systems are also described into detail in the same reports.

The nodalization was set up in the frame of a cooperation also involving the Italian Licensing Authority (ANPA), the Vendor Ansaldo and the University of Roma 'La Sapienza', e.g. (Gatta and Mastrantonio, 1995), (D'Auria et al., 1996) and (Cerullo et al., 1996) the last one also including the qualification of the Kozloduy transient data (see below).

The resulting nodalization consists of more than one thousand nodes; among the other things, each of the four loops of the plant is modelled independently. The qualification of the nodalization was attained in the frame of the activity documented in the paper by (Aprile, 1996) utilizing previously defined criteria, (Bonuccelli et al., 1993) and the plant transient data described by Cerullo et al., 1996 and by Kolochko, 1995.

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Previous activities have been documented by Aprile et al. 1996 and by F. D'Auria et al. 1997a and F. D'Auria et al. 1997b dealing with the evaluation of safety of WWER-1000 from the thermalhydraulic side. LOCA (Loss of Coolant Accident) and transients without the loss of integrity of the primary loop, are separately addressed in the two papers above. The main outcomings can be summarized as follows:

- ECCS (specifically SIT conditions) are well designed for Large Break LOCA but indadequate performance may happen following Small Break LOCA;
- peak cladding temperature in the hot rod (alone) has been found to overpass safety limits; however, initial neutron flux distribution was not known with sufficient approximation;
- ATWS transients originated by large break in the secondary side of steam generators (similar to the one here considered) may cause severe damage to the plant;
-accident management recovery procedures based upon the depressurization of the secondary side and/or upon the injection of auxiliary feedwater are extremely effective in recovering primary side owing to the large area for heat transfer available in the steam generators.

The present paper deals with the evaluation of the PTS (Pressurized Thermal Shock) potential in the RPV (Reactor Pressure Vessel) of the WWER-1000. PTS may occur in different accident scenarios involving simultaneous pressurization and cooling of the RPV. Different transients originated in the secondary side of the steam generator may cause primary side pressurization, while RPV cooling may be the result of cold water injection (mostly HPIS = High Pressure Injection System) or of cooling of the primary fluid following anticipated events in the steam generator. Without the availability of a finalized PSA (Probabilistic Safety Analysis), the selected transient is one of those having the potential for PTS. The origin is a large break at the top of one steam generator; scram occurs at 120% core power and all plant systems are assumed to be available (essentially, pump trip, steam generators isolation, pressurizer PORV actuation).

In order to make meaningful the analysis, the downcomer has been split into two parts : on connected with the single broken and the other connecte with the three intact steam generators. This has been done both from a hydraulic and a structural mechanic point of view. This hypothesis produced a not axisymmetric cooling of the belt line of the vessel . Both in the downcomer and in the vessel wall, temperature was considered to be not depending on the angular position within each of the two parts.

Experimental and theoretical researches have been conducted in the recent years at the DCMN (see for instance Vitale (1989), Beghini and Vitale (1989), Beghini, Vitale and Milella (1991), Vitale and Beghini (1991), Beghini, Bertini and Vitale (1992)) on the structural behaviour of vessel during PTS accident. In particular, numerical methods were developed and applied for Fracture Mechanics analysis which allow facing these typically multiparametric problems in an accurate and efficient way. Such an approach was applied to this PTS scenarious giving indication of the risk for a brittle rupture of the vessel during the transient.

2. OUTLINE OF THE WWER-1000 PLANT

The Kozloduy WWER-1000/320 plant, shown in Fig. 1, and documented in the papers (Kolev et al., 1992), (Hinovsky, 1995), (US DOE, 1987), and (Gatta and Mastrantonio, 1995), basically consists of:

* a pressurized water reactor vessel of 3000 MWt rated power with 163 hexagonal fuel assemblies, which are open (ventilated type) to permit cross flows among the bundles. The fuel is UO2 in annular pellet form, clad in zirconium-niobium alloy. There are 10 banks of control rods without fuel followers, located in 61 fuel assemblies;
* inner radius and wall thickness in the belt line region 2068 mm and 199 mm respectively ;
* four primary loops equipped with horizontal tubes steam generators (Fig. 2) and main circulation pumps of centrifugal type;
* one turbogenerator with 1500 rpm and 1000 MW electric power;
* an emergency core cooling system including three High Pressure Injection Pumps, four accumulators (SIT = Safety Injection Tank) and three Low Pressure Injection Pumps;
* two independent feedwater loops connected with the four steam generators; each loop includes a turbine driven pump and piping connecting the feedwater line to four different locations in each steam generator; several valves are part of the system. The two feedwater loops feed a common header upstream the four steam generators.

3. RELAP5 NODALIZATION

A detailed nodalization was developed considering theguidelines of the code User Manual and the criteriaby(Bonuccelli et al., 1993). The sketch of the noding scheme can be seen in Fig. 3; information about adopted code resources can be drawn from Tab.1. The vessel and the piping do not represent any particular challenge for the code being quite similar to the corresponding ones in the Western type PWR. However, the bypass flow paths configuration inside the vessel is different: in the case of the WWER-1000 the upper part of the downcomer is not directly connected with the upper head: instead, a bypass flow path exists from the lower plenum to the upper head through the control rods housing and guide tube. The active core has been modelled adopting four parallel stacks (three core regions and one hot rod) characterized by different linear power.

Steam generator geometry imposes drastic changes in noding philosophy with respect to what is suitable for the U-tubes steam generators. As "symmetry axis" for the secondary side (and consequentely for the primary side), a line was taken connecting the center between the two collectors with the average (virtual) position of the centers of the horizontal tubes with the aim to separate (in the nodalization) the hot and the cold sides of the tubes. In this way, the secondary side of the steam generators was divided into three zones: a) the hot zone including the hot collector, and the hot 1/2 parts of the tubes; b) the cold zone including the cold collector and the cold 1/2 parts of the tubes; c) the downcomer region gathering different zones of the secondary side: essentially, the regions where downflow is assumed (e.g. the boundaries and the location of feedwater nozzles). Fours positions were distinguished for the feedwater nozzles as results from Fig. 3. The nodes subdivision in the primary side was a consequence of the above: more nodes are placed on the hot side of the tubes; six axial levels lumping six groups of tubes, can be noted.

The Emergency Core Cooling System (ECCS) consists of three High Pressure, four Low Pressure Injection Systems (HPIS and LPIS, respectively) and four accumulators (SIT, Safety Injection Tank).

These are directly connected with the downcomer and with the upper plenum of the main vessel. All the ECCS have been modelled together with the AFW (Auxiliary Feedwater System) and the relief valves at the top of the pressurizer and in the secondary side of steam generators (PORV and SRV).

3.1 Nodalization qualification

The nodalization qualification process consists of two main steps, concerned with the steady state and the transient level, respectively. In the former, two sub-steps can be identified, see in the report (Bonuccelli et al., 1993): development of the nodalization and achievement of the steady state.

a) steady state level

Criteria like those recommending the maximum node length, the number of mesh points necessary to simulate the conduction heat transfer in the fuel rods and across the steam generators tubes, the connection between neighbouring nodes, etc. have been used. The first sub-step concludes with the check that various dimensions of the nodalization coincide, within an acceptable error, e.g. see (Bonuccelli et al., 1993), with the design or reference values. An example of this is given by Fig. 4 that shows the correspondence between the volume and the height of the secondary side of one steam generator.

The second sub-step deals with the achievement of the steady state. Typically 100 s are considered sufficient, running the code in the transient mode. Again, criteria dealing with the time derivative and the absolute values of relevant time trends are available and have been matched (in the last case, design or reference values are used for defining the acceptable error). Some results are shown in Tab. 2.

b) transient level

The NPP transients adopted for qualifying the nodalization at the 'on-transient' level are outlined in Tab. 3. It can be noted that the Kozloduy transients were 'planned' operational transients (Hinovsky, 1995) while an unplanned event occurred in Zaporosche in April '95, (Kolochko, 1995). In this last case, the intervention of three High Pressure Injection Systems and the accumulators was sufficient to recover the NPP in a few tens of minutes. The available NPP data were qualitatively and quantitatively well predicted by the nodalization , (Aprile, 1996).

The nodalization was also qualified through the comparison of results from the same (i.e. same sequence of events and properly scaled boundary conditions) small break LOCA transient in a Western type PWR and in related experimental facilities, Tab. 3, see also (D'Auria et al., 1996).

4. APPLICATION OF THE RELAP5 NODALIZATION TO THE ANALYSIS OF A TRANSIENT RELEVANT TO THE PTS

4.1 Framework of the activity

The main purpose of system codes is to perform accident analyses in order to confirm the design choices also in relation to the planning of Emergency Operating Procedures, e.g. see Aksan et al. 1996 and D'Auria et al. 1995.

Before giving details related to the transient of interest, it seems worthwhile to show the matrix of accident sequences already studied: the related results are reported in the already mentioned papers.

A matrix of accidents was fixed, essentially consisting of simplified sequences of events, without having in mind the results of probabilistic safety studies, as already mentioned. Rather, the selected events, put a challenge to the code and the nodalization, include a wide range of phenomena, and make possible the comparison with the results of similar transients in Western PWR.

Almost all the selected scenarios can be classified as "beyond DBA - before core melt"; namely "beyond DBA" testifies that not all the Emergency Systems (ECCS) were assumed to operate as from the respective design, and "before core melt" indicates that the analysis is stopped, eventually, when any fuel cladding surface temperature reaches the licensing limit. On these bases, the matrix in Tab. 4 was built up.

4.2 Large Steam Line Break with Scram

The transient here considered (VV-SLB-05) is the counterpart of the one listed as 6b in Tab. 4. An ATWS situation is considered in the test scenario 6b. Relevant boundary conditions for the test scenario are included in the same Tab. 4.

The main result related to the above test 6b is shown in Fig. 5, where the core power history is reported and compared with the same quantity characterizing scenarios 6a, 6c and 6d. It is shown that the lack of scram brings the plant to a dangerous situation for break areas larger than about 0.5 m^2.

The VV-SLB-05 has the same boundary conditions and relevant imposed sequence of events as the test 6b, with the exception of the insertion of the control rod that is assumed when the neutron flux achieves 120% of its nominal steady state value.

Significant results are shown in Figs. 6 to 10 that depict the thermalhydraulic scenario. As a consequence of the break, isolation of the steam generators occurs with consequent secondary side pressure rise (Fig. 6). Scram limits the core power to values of the order of 150% of nominal value (Fig. 8), keeping the DNBR (Departure from Nucleate Boiling Ratio) above the unity value also due to initial fluid temperature lowering and mass flux increase through the core (Figs. 4 and 5). In a few minutes primary pressure reaches the pressurizer PORV (Pilot Operated Relief Valves) opening set point owing to average primary coolant heat up (Figs. 6, and 9). Pressurizer level also increases as a consequence of the above (Fig. 7). Pumps operation has been assumed in order to maximize the potential for PTS.

4.2.1 Information relevant to PTS

In order to make more realistic the data base necessary for calculating the PTS potential of the considered scenario, it was necessary to introduce changes to the nodalization described in the Chapt. 3. Specifically, the three hydraulic nodes in the upper part of the downcomer, connected with the cold legs were split into two parts with flow area equal to 1/4 and 3/4 of the total value, respectively. The former was connected to the broken steam generator cold leg and the latter was connected with the three intact steam generators cold legs. The RPV wall structures associated with the hydraulic nodes were subdivided in the same way. Thirty meshes were considered for conduction heat transfer inside the RPV wall.

The temperatures of the meshes associated with the cooled part of the downcomer are are given in Figs. 11 and 12: thermal gradients of the order of 70 K are calculated in the RPV wall thickness. The temperatures of the uncooled part of the downcomer (3/4 of the total) remain almost unchanged and for simplicity are not shown.

All this information is used as input to the structural mechanics calculation described in the following chapter.

5. FRACTURE MECHANICS ANALYSIS

5.1 Basic assumptions

During previous studies, the suitability of the Weight Function (WF) technique for solving Fracture Mechanics (FM) problems under thermal transients was generally verified. The WF coupled with the nominal stress (i.e. the stress in the uncracked structure) allows evaluating FM parameters straightforwardly by a simple integration. This approach is very efficient from a computational view point and can produce accurate results as compared to complete 3-D Finite Element (FE) analyses .

For a typical vessel geometry several WFs have been proposed both for 2-D and 3-D schemes. The present analysis was focused on the evaluation of the potentiality of the particular thermal transient to produce an unstable crack propagation. Therefore it was decided to limit the number of parameters making some conservative hypotheses about the location and shape of the crack. It was assumed that the temperature distribution obtained by the thermal analysis through the wall thickness could be applied to the belt line region where the most severe irradiation effect is expected. A pre-existing surface defect was assumed in the inner side of the wall. Crack was considered virtually infinite in axial direction and having constant through-thickness depth and was located just under the nozzle where cold water is injected. This allowed assuming a simple 2-D scheme for FM analysis which can be considered reasonably conservative. Indeed, real defects (commonly associated with vessel welds) have limited extensions and can be embedded in the wall.

Nominal stress was determined by assuming for the vessel a 2-D scheme of an infinite long cylinder made by homogeneous material (cladding was neglected) which is cooled in a region having angular extension equal to 90°. Within the cooled region the temperature distribution was assumed independent to the angular position and given by the thermal hydraulic analyses (Fig.11). A cubic spline interpolation was applied to obtain a continuous function between the nodes. For the remaining part of the cylinder a linear through-thickness temperature was imposed.

Nominal stress was evaluated by an analytical model based on the theory of curved beams in generalize plane strain (Boyley and Weiner 1960) and either temperature distribution and internal pressure were assumed as loading conditions.

Stress Intensity Factor was determined by means of a WF recently proposed by (Verfolomeyev and Hodulak (1997)) on the bases of accurate parametric FE evaluations produced by Bryson and Dickson (1993). Given the temperature distribution, the accuracy in the SIF evaluation can be estimated in a few percent.

A surface crack is usually filled by the coolant. This was considered by adding the inner pressure to the nominal stress in the Stress Intensity Factor (SIF) evaluation (direct effect of pressure on the crack edges). However, the tip temperature was assumed not affected neglecting the refrigerating effect of the small quantity of liquid penetrating the crack.

5.2 Main results

Hoop stress due to thermal loading only is shown in Fig. 13 for several time instants during transient. Maximum value is attained after about 20 seconds and the peek level (170MPa) is comparable to the stress due to primary pressure. After about 40 sec. thermal stress becomes compressive at the inner surface.

If the temperature distribution produced by cooling a limited part of the vessel is applied to the whole cylinder, the result is usually considered to be conservative. In order to asses the level of conservatism, the nominal stress for the present thermal distribution was applied to sectors having different extensions. In Fig. 14 maximum thermal stress is plotted versus the angular extension of the cooled region (assuming the same temperature distribution for any cooled zone). It is worth noting that the hypothesis is conservative only if the cooled sector is lower than about 140°. In the examined case (cooled sector equal to 90°) thermal stresses were only a few percent lower than those evaluated during a complete cooling.

In Fig. 15 SIF evolutions produced by thermal transient was calculated for a few crack depth including relatively shallow (a=5,10 mm), mean size (a=40,50 mm) and deep (a=100mm) defects. Negative values indicates that thermal loading tends to reduce the stress at the crack tip. The combined effect of thermal and pressure loading (total SIF) is shown for the same crack lengths in Fig. 16 . By comparing Fig. 15 and 16, it can be concluded that thermal loading is more important for shallow cracks as the SIF is doubled during the PTS transient. On the contrary, for deep cracks the SIF increment during the transient tends to be a relatively small fraction of the regime value (due to pressure).

This is a general conclusion for a PTS analysis which usually indicates relatively shallow defects as the cause of the highest probability of unstable propagation. The condition of propagation for a 5 mm crack can be examined in Fig. 17 where curves of fracture toughness are reported too. The ASMR KIc reference function was adopted with a few RTNDT value. It can be observed that a very high transition temperature (RTNDT=270°C) needs to promote brittle facture. Similar conclusions were obtained also for other defect sizes.

This observation seems to indicate the examined PTS transient is potentially dangerous for relatively shallow cracks (5-15 mm depth) for a material in a very severe embrittled condition. In these cases, if an unstable propagation occurs, the probability for the crack to pass through the wall is very high as the SIF is strongly increasing with crack depth.

More complete analyses could be obtained by the proposed methodology (also including critical crack depth diagrams and probability estimates) having specific values of fracture strength of the material in the real conditions of the plant. A complete 3-D approach including the limited size of the defect can be made by adopting a suitable WF such as that recently proposed by Beghini, Bertini and Gentili (1997).

6. CONCLUSIONS

Conclusions related to the present study can be distinguished at two levels including the thermalhydraulic system behaviour and the stress response of the RPV walls

From the thermal qualified WWER-1000 plant nodalization suitable for Relap5/mod3.2 was developed in the present framework: steady state and transient level qualification criteria were adopted and matched. Relevant conclusions from previous analyses can be summarized as:

* the Large Break LOCA behaviour of the WWER-1000 appears good from the safety viewpoint with low value for the Peak Cladding Temperature (PCT <1000 K); the hot rod represents an exception possibly originated from inadeguate knowledge of steady state core power distribution;

* Small Break LOCA without HPIS cause unacceptable rod surface temperatures (see below);

* operating conditions for accumulators appear well suited for Large Break LOCA; in the case of small break LOCA the small gas volume prevents complete liquid discharge into the primary loop;

* increase in the gas volume space of accumulator largely improves their performance, though an optimization process for selecting this quantity has not been performed.

* AM procedures based on delayed actuation of AFW (secondary side feed) and steam generator depressurization (better secondary side bleed) are very effective and should be considered for the Emergency situations in WWER-1000;

* natural circulation performance of the WWER-1000 systems appear suitable from the safety point of view. provided sufficient heat sink is available for steam generators;

* a long time is available to operators (more than 1 hr) before core uncovery occurs in the case of steam generator tube ruptures, even in situations where relatively large break areas are considered (i.e. up to 20 tubes, see Tab. 4);

* the introduction of a PWR like neutron kinetic model made it possible to perform ATWS calculations. In the case of LOFW, the increase of the primary system temperature keeps the core power at the decay value starting from about 100 s after the initial event: the consequent slow boiloff scenario does not lead to core uncovery (DNB) for at least 20 mins. Limited spurious injection of cold unborated liquid does not causes unacceptable consequences. In the case of steam line break, the capability of the system to withstand the consequences largely depends upon the considered break areas (Fig. 5 and Tab. 4)..

The main result from the system analysis is the demonstration that scram is able to control the reactivity rise consequent to a large break in one steam line, keeping the power to values not endangering the core integrity.

The achieved conclusions should be verified on the basis of a proper uncertainty analysis (e.g. adopting the methodology described by (D'Auria et al., 1995); a better knowledge of all relevant boundary conditions would also improve the reliability of the results.

From the Fracture Mechanics point of view a model based on the Weight Function was applied for analysing the transient. By adopting some conservative hypotheses on the shape and position of the defect a parametric analysis was performed giving the following main conclusions :

* cooling a 1/4 of the vessel does reduce the thermal stress as compared to the complete cooling but thermal stress is no more than a few percent lower. The conservatism of the axisymmetric loading hypothesis is not excessive in this case ;

* Thermal Shock increases the Stress Intensity Factor more significantly for shallow cracks than for deep defects as for those pressure loading is prevailing ;

* temperature reduction of the crack tip region is not particularly severe and a high level of material embrittlement is required for promoting an unstable crack propagation.

More complete analysis could be performed if specific data about material fracture properties and typical shape, size and location of defects were available.

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PARAMETER	NPP KOZLODUY
NUMBER OF NODES	1102
NUMBER OF JUNCTIONS	1148
NUMBER OF HEAT STRUCTURES	1028
NUMBER OF MESH POINTS	5590
NUMBER OF CORE ACTIVE STRUCTURES	42
NUMBER OF CONTROLVARIABLES	1107
NUMBER OF TRIPS	58

Tab. 1 - Nodalization resources.

VVER1000 PLANT	UNITS	EXP	NEW NOD
Primary side pressure	MPa	15.7	15.7
Hot Leg Temperature	°C	320.1+3.5	322
Cold Leg Temperature	°C	289.8+2	287.2
Primary Side Mass	Kg	240000	235100
Prz level	m	8.77	8.75
Core Power	MW	3000	3000
Reactor Flow	Kg/s	15704	15450.6
SG Power	MW	750	748
Secondary Side Mass	Kg	160000	159900
Feedwater Temperature	°C	220	220
Secondary Side Pressure	MPa	6.28	6.27

Tab. 2 - Nodalization qualification at steady state level: calculated and measured boundary condition values.

TRANSIENT	PLANT	TYPE	DESCRIPTION	INITIAL POWER	INITIAL PRESSURE
KZ-1	Kozloduy-5	planned	1Pump trip	72.	15.7
KZ-2	Kozloduy-5	planned	2Pumps trip	54.	15.7
KZ-3	Kozloduy-5	planned*	Partial loss of feedwater	72.1	15.8
ZAP	Zaporosche	unplanned	PORV stuck open	1.0**	18.8
PWR	Krsko	calc.°	Small Break LOCA	100.	15.

* unplanned sequence of events

** estimated

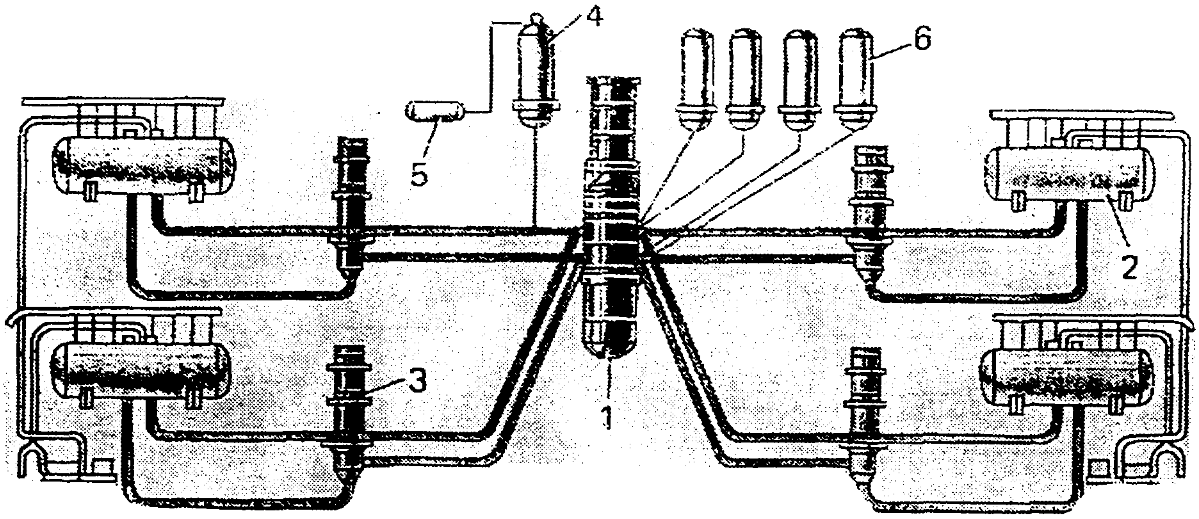
° with the support of experimental data in Test facilities

Tab. 3 - Transients utilized for the 'on-Transient' qualification of the Relap5/mod3 WWER-1000 nodalization.

N°	NPP TRANSIENT	TYPE	SG SIGNIFICANT CONDITIONS	HPIS	ECCS LPIS	SITS	RCP	AM PROCEDURE
1a	VV-SGT-01	SGTR 1 tube	Isolation at scram (AFW off)	off	in CL at 2.15 MPa	4 at 5.9 MPa	off	-
1b	VV-SGT-03	SGTR 5 tubes	Isolation at scram 30k/hr (intact SG)	off	in CL at 2.15 MPa	4 at 5.9 MPa	off	-
1c	VV-SGT-04	SGTR 20 tubes	Isolation at scram 40k/hr (intact SG)	off	in CL at 2.15 MPa	4 at 5.9 MPa	off	-
2a	VV-LFW-01	LOFW	Isolation at scram AFW delayed	off	off	4 at 5.9 MPa	on	AFW
2b	VV-LFW-02	LOFW	Isolation at scram AFW delayed	off	off	4 at 5.9 MPa	off	AFW
3	VV-PTR-01	LOFA 4 pumps	Isolation at scram off	off	off	off	off	-
4a	VV-SBP-00	PORV Stuck Open	15K/hr after scram AFW off	at 8.9 MPa	in CL at 2.15 MPa	4 at 5.9 MPa	off	-
4b	VV-SBP-01	PORV Stuck Open	15K/hr after scram AFW off	off	in CL at 2.15 MPa	4 at 5.9 MPa	off	-
4c	VV-SBP-02	PORV Stuck Open	15K/hr after scram AFW off	off	in CL at 2.15 MPa	4 at 5.9 MPa	off	SIT* init. cond
4d	VV-SBP-03	PORV Stuck Open	SG SRV opening after dryout AFW off	off	in CL at 2.15 MPa	4 at 5.9 MPa	off	SG BLEED
5a	VV-LFW-A1	LOFW ATWS	Isolation	off	off	4 at 5.9 MPa	on/off	-
5b	VV-LFW-A2	LOFW ATWS	Isolation	off	off	4 at 5.9 MPa	on	HPIS unborated
6a	VV-SLB-01	SLB/ATWS BA=0.0022 m^2	Isolation	off	off	4 at 5.9 MPa	on	-
6b	VV-SLB-02	SLB/ATWS BA=0.98m^2	Isolation	off	off	4 at 5.9 MPa	on	-
6c	VV-SLB-03	SLB/ATWS BA=0.5m^2	Isolation	off	off	4 at 5.9 MPa	on	-
6d	VV-SLB-04	SLB/ATWS BA=0.1m^2	Isolation	off	off	4 at 5.9 MPa	on	-

*N_2/H_2O ratio changed from 1/5 to 7/5

Tab. 4 - List of performed RELAP5 calculation of WWER-1000 transient scenarios: sequences of assumed main events.



1 - reactor pressure vessel;
2 - steam generator;
3 - primary coolant pump;
4 - pressurizer;
5 - quench tank;
6 - emergency core cooling accumulators

Fig. 1 - WWER-1000 NPP: Primary system

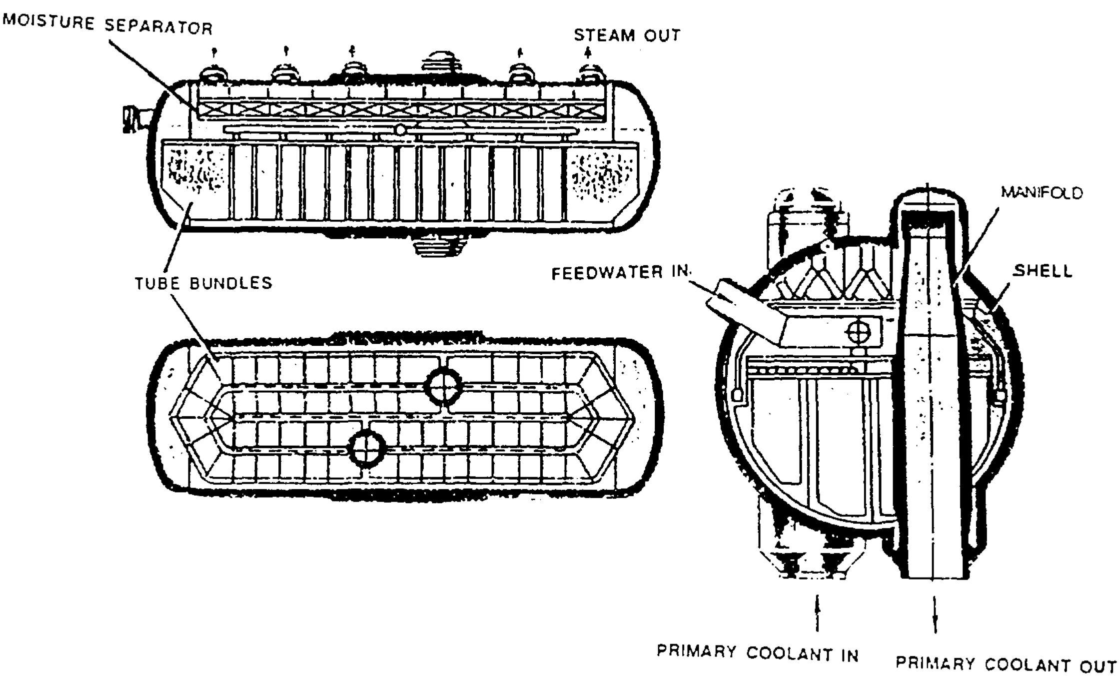


Fig. 2 - WWER-1000 NPP: Steam Generator detail.

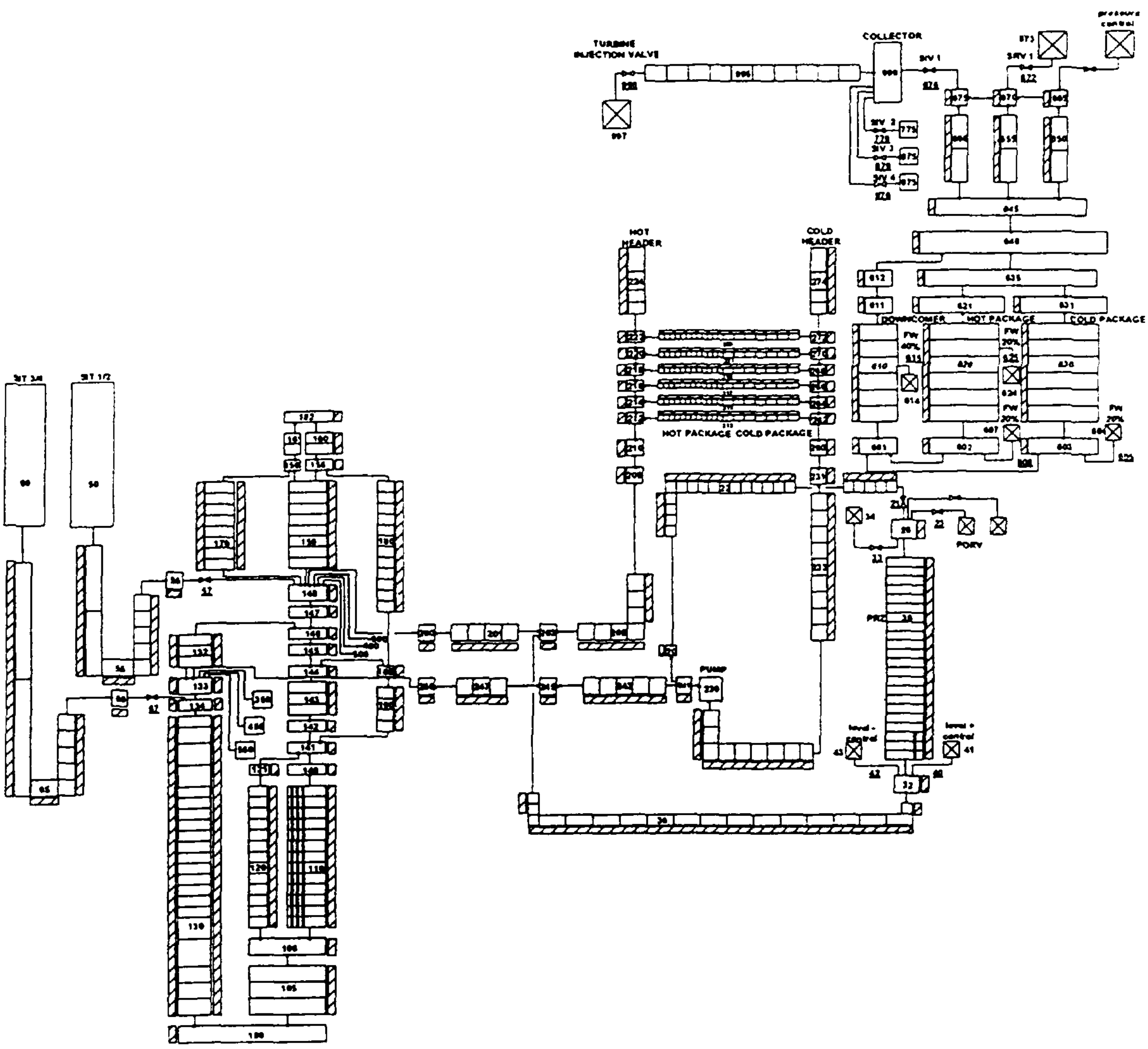


Fig. 3 - WWER-1000 NPP: Noding scheme for Relap5/mod3.2

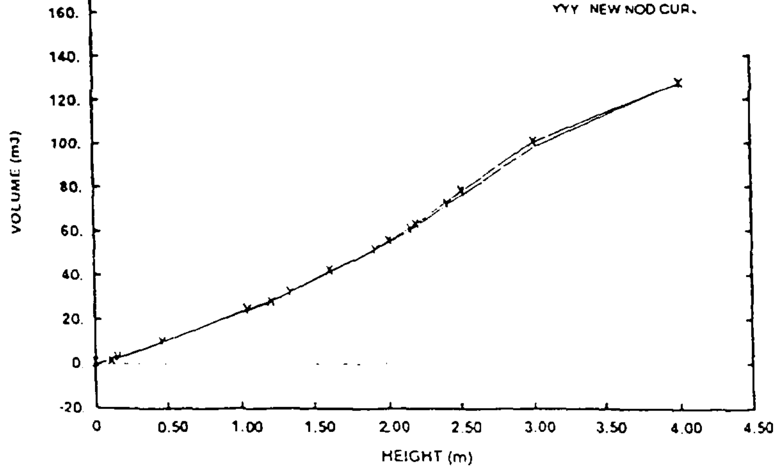


Fig. 4 - Nodalization qualification at steady state level: calcualted and measured volume/height curve for SG.

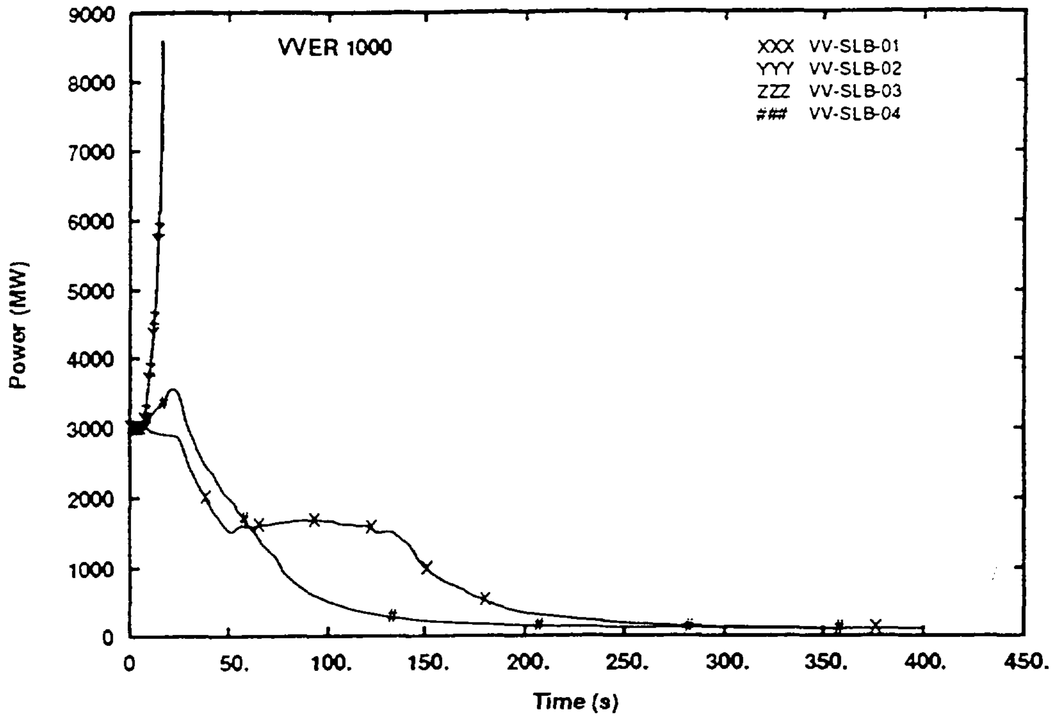


Fig. 5 - WWER-1000 SLB/ATWS: core power (list of transients in Tab. 4).

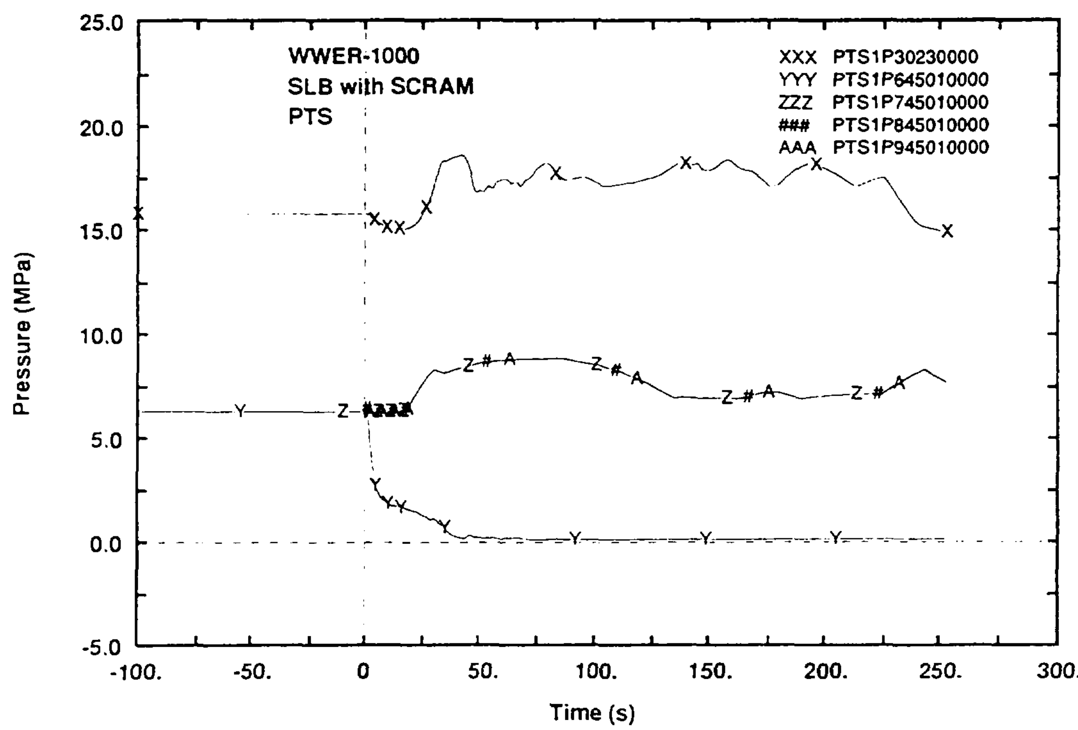


Fig. 6 - WWER-1000, SLB, PTS calculation: primary and secondary side pressures.

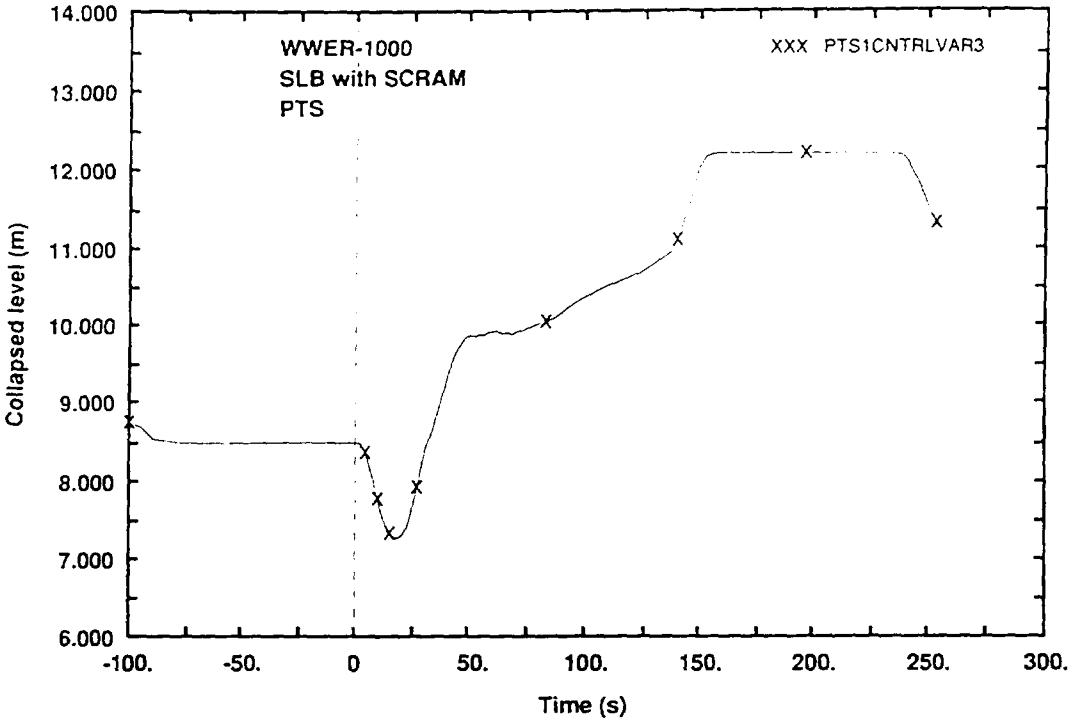


Fig. 7 - WWER-1000, SLB, PTS calculation: presssurizer level.

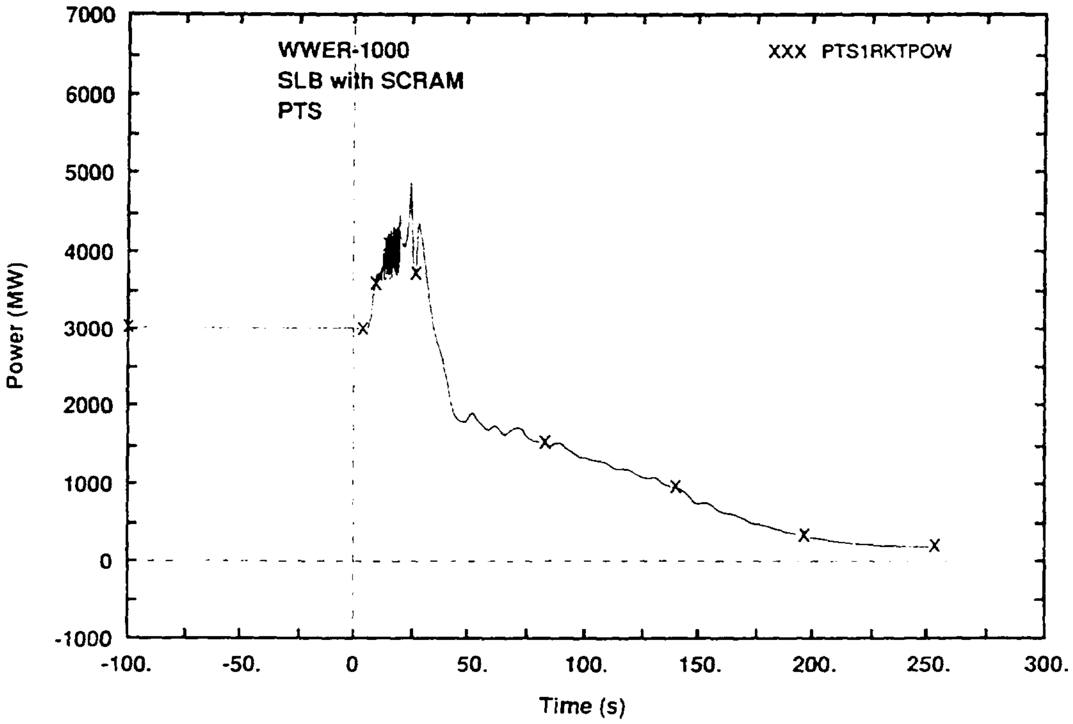


Fig. 8 - WWER-1000, SLB, PTS calculation: reactor power.

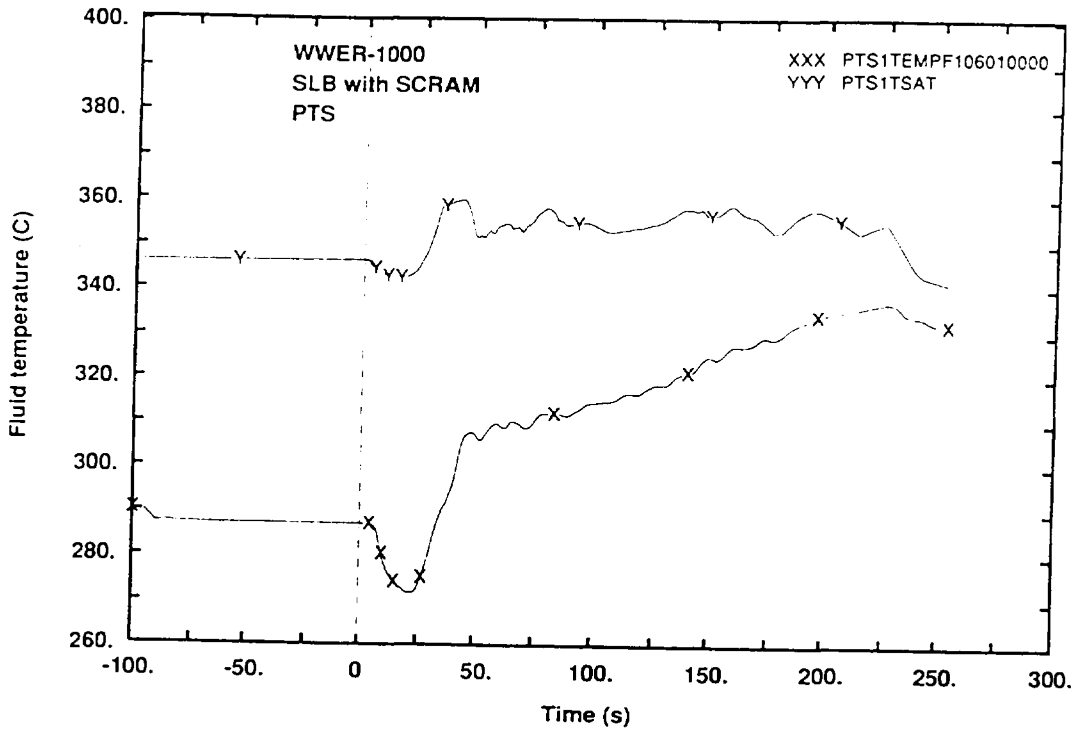


Fig. 9 - WWER-1000, SLB, PTS calculation: fluid temperature in the LP compared with saturation temperature.

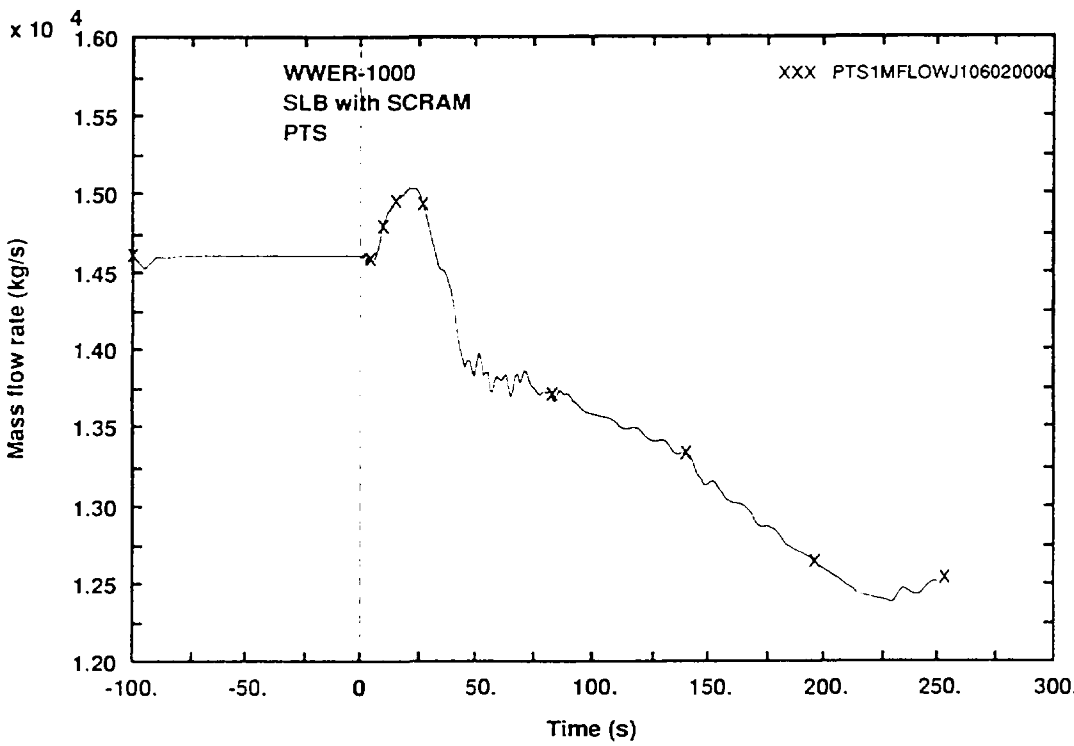


Fig. 10 - WWER-1000, SLB, PTS calculation: mass flow rate at core inlet.

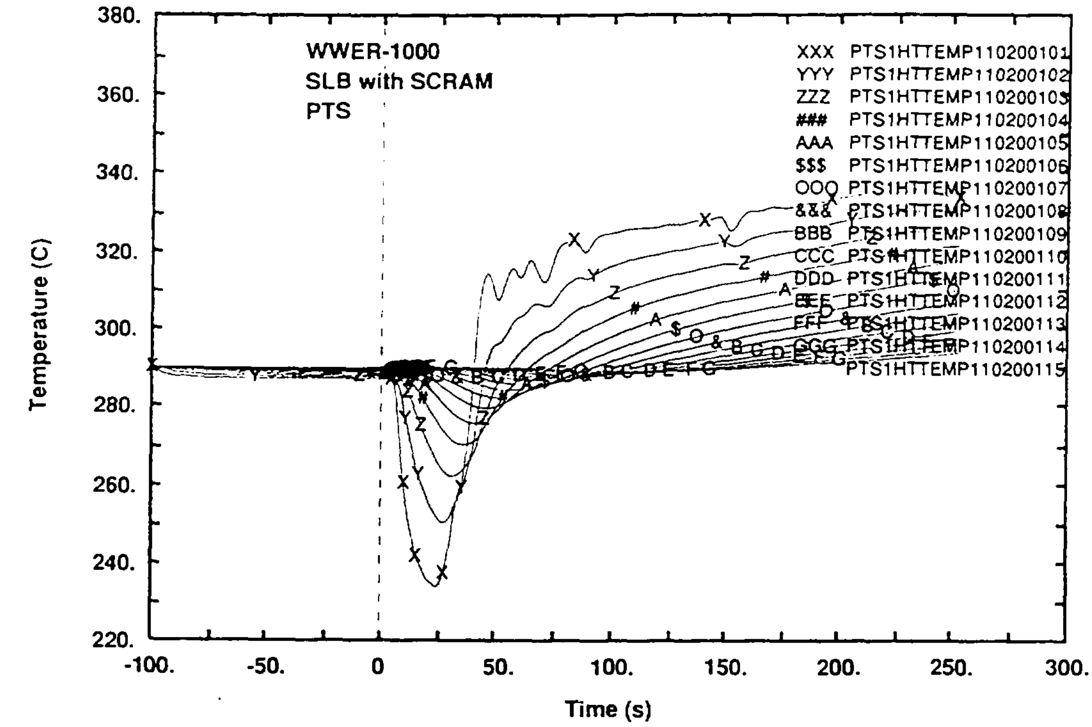


Fig. 11 - WWER-1000, SLB, PTS calculation: temperature distribution inside the vessel wall (meshes from 1 to 15), affected SG side.

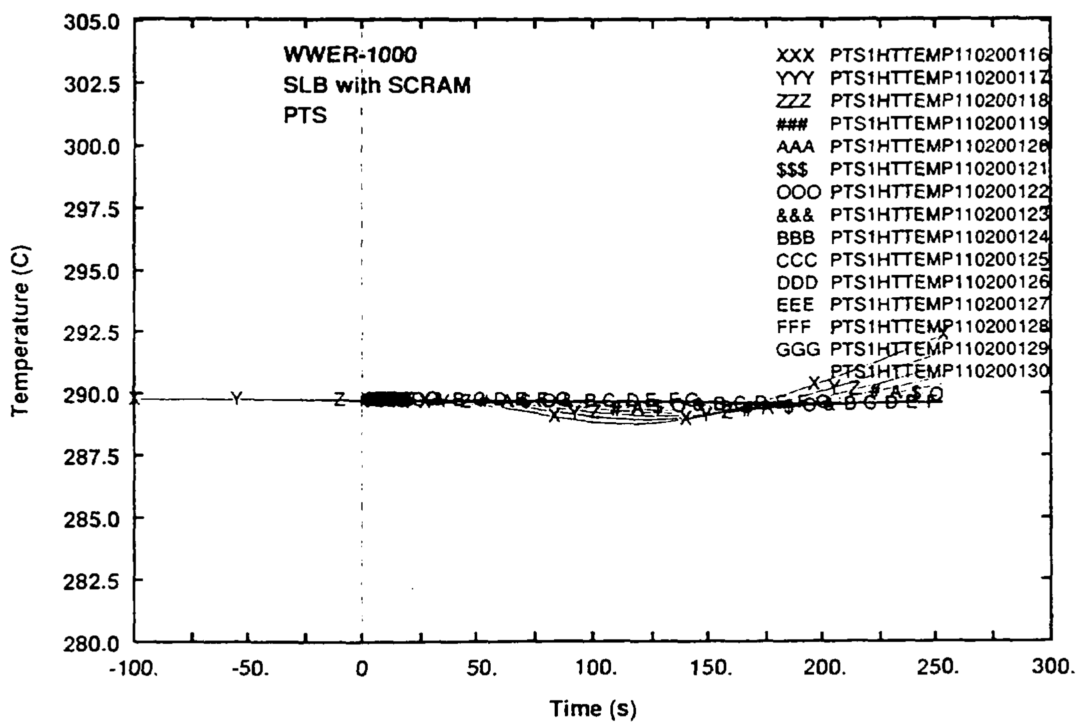


Fig. 12 - WWER-1000, SLB, PTS calculation: temperature distribution inside the vessel wall (meshes from 16 to 30), affected SG side.

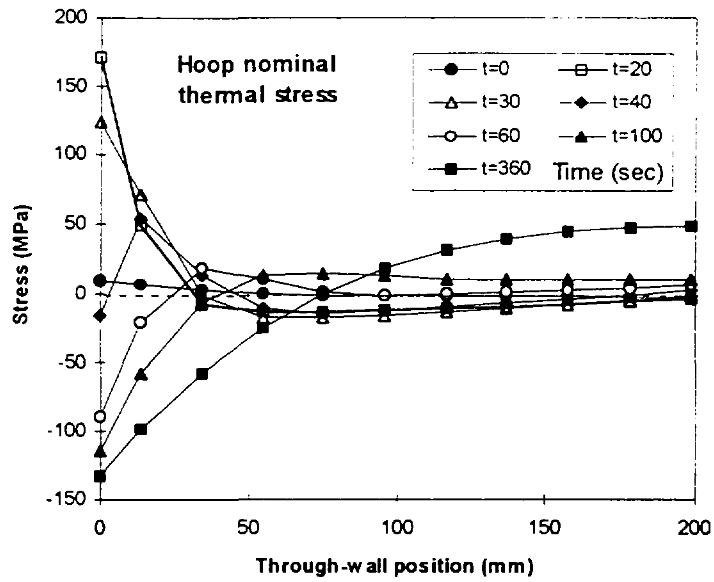


Fig. 13 - Thermal stress distributions in the cooled zone of the belt-line by assuming the colled part equal to 1/4 of the cylinder.

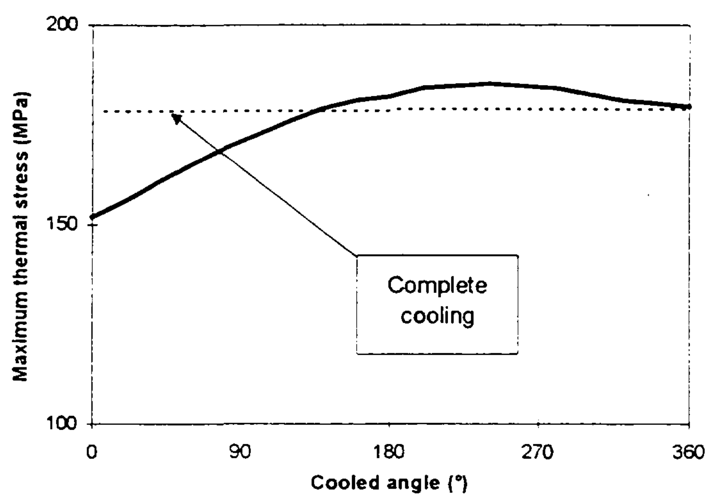


Fig. 14 - Maximum thermal stress in a cylinder by imposing in a sector the through-wall temperature distribution equal to that evaluated at 20 sec. in the analysed transient.

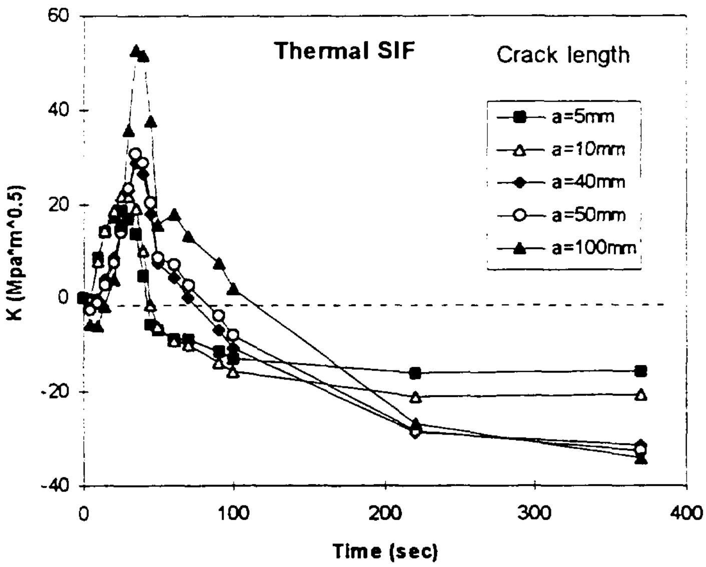


Fig. 15 - SIF produced by thermal loading only during the PTS transient for some crack lengths .

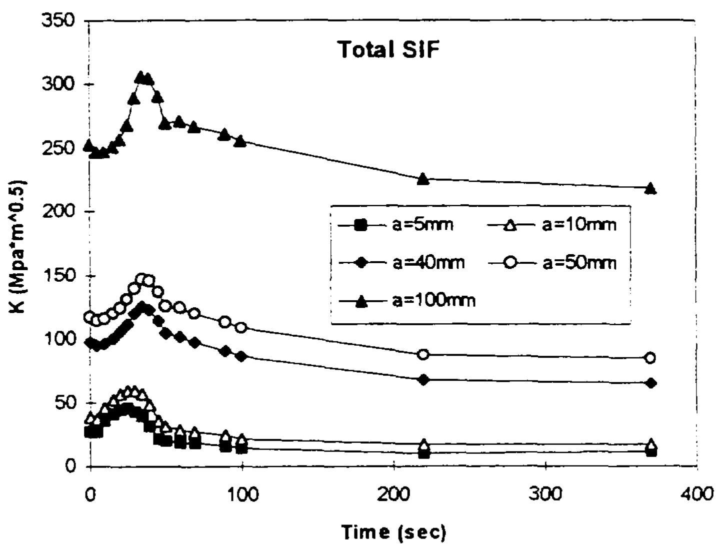


Fig. 16 - SIF during the PTS transient for some crack lengths under the effect of pressure and thermal loading.

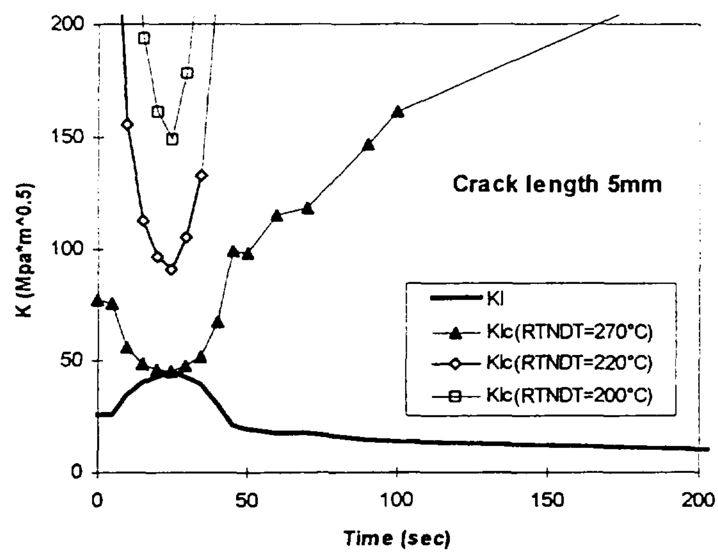


Fig. 17 - Comparison of SIF and fracture toughness for a shallow crack (a=5mm) during the PTS transient for three values of the brittle-ductile transition temperature.