SCANDINAVIAN PCRV DEVELOPMENT: PRESENT STATUS AND PLANNED WORK, 1973–75

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SUMMARY

At the 1st SMIRT Conference, the papers H5/3 and H6/7 described the first phase of Scandinavian development work in PCRVs for BWRs. The papers reported tests in the elastic range, resp. insulation tests performed at Studsvik, Sweden, on a large scale integral model of the proposed design of a PCRV for a BWR. The second phase of development consisted of a study to assess the feasibility of BWRs with PCRVs from the technical and economic points of view as well as to locate and enumerate aspects of the Scandinavian design that were considered to require practical verification.

The reference design of a 900 MWe BWR with PCRV which was the basis of the feasibility study was not significantly different from the design presented at Geneva 1971. The phase (ii) vessel has, like the phase (i) design, a removable lid, a cold liner with removable insulation, integrated main circulating pumps and a conventional totally enclosing pressure suppression containment. The important new features introduced during phase (ii) have been steam and water penetrations in the barrel of the vessel, external hoop prestressing and a proposal for liner ventilation.

The experimental programme of phase (ii) consisted of the simulation of various extreme accidents such as the sudden loss of insulating gas and depressurisation resulting from a steam rupture. The programme concluded with a cold pressure test to 2.5 times the design pressure (215 ata).

The phase (ii) study confirmed the feasibility of the Scandinavian PCRV design for a BWR and proposed a phase (iii) programme of tests of a verifying nature on a number of points in order to provide the detail evidence to support an investment in a nuclear application. This proposed verification programme consists of tests on the liner ventilation system, failure tests in special rigs on the lid design and perforated bottom slab, vibration tests on the insulation casing and radial pipe assemblies, manufacturing tests on the bottom slab, corrosion and radiolysis tests on the insulation as well as design of an integral model of the phase (ii) vessel design.

This paper is mainly concerned with the technical results of the experimental work and the feasibility study performed during the second phase of development. It also reports on the present status of the various experimental tasks that have been undertaken in the verification programme.

1. Introduction

At the first SMiRT Conference in 1971, three papers described the first phase of Scandinavian development work on PCRVs for BWRs [1, 2, 3]. The papers reported on structural tests in the elastic range and insulation tests performed at Studsvik, Sweden, on a 1:3.5 scale integral model (Fig 1) of the proposed design of a PCRV for a 900 MWe BWR. The present paper reports briefly on the results of the second phase of PCRV development in Sweden.

The experimental programme of phase II consisted once again both of structural and insulation tests. The structural tests included the mapping of force distribution in the lid-locking struts as well as cold over-pressure tests up to 215 bar, i.e. 2.5 times the design pressure. Various extreme accident situations were simulated such as the sudden loss of insulation gas and the depressurization resulting from a steam pipe rupture. An analysis was made from the technical and economic points of view to assess the feasibility of BWRs with PCRVs. This was based on a reference design that was produced for a 900 MWe BWR. The second phase also included the identification and enumeration of such aspects of the Scandinavian design as were considered to require practical verification.

2. Phase II Tests on the Studsvik Model

The pressure and temperature conditions in the model vessel during both phases I and II are shown in Fig 2.

2.1 Structural Tests on Vessel

The second phase of the Scandinavian model vessel testing was started with an investigation of the influence of strut prestress on the force distribution in the lid locking struts.

With a very simple prestressing arrangement, it was possible to prestress the struts to a level corresponding to a lid lift-off-pressure of about 40 bars. From this it is concluded that even complete prestressing of the struts should be possible if lid lift-off is not desirable in an actual design.

Force distribution at some different pressures is shown in Fig 3.

At the end of the phase II experimental period, the vessel was pressurized in stages to a maximum pressure of 215 bars which is equivalent to 2.5 times the design pressure.

In the following, the behaviour of the vessel is illustrated at a few characteristic points.

The strains on the liner at the middle of the vessel are shown in Fig 4. The strains are mainly linear up to a pressure of about 160 bar. This indicates that the vessel is what may be called "concrete-elastic" up to this pressure in this part of the vessel.

Above this pressure there is a successively increasing rate of strain due to the fact that the concrete is cracking, and the loading is principally carried by the steel members of the structure. The measured strains do not completely return to zero after unloading from the maximum pressure 215 bar, but the vessel is still in the "steel-elastic" range as the prestressing cables are well below their yield stress. The zero shift may be due to the fact that the cracks do not close completely.

The highest strain measured at all in any structural part is the meridional strain in the corner between the bottom slab and the vessel wall as shown in Fig 5. As is seen on this figure, the measured strain exceeds $4 \cdot 10^{-3}$. Thus yielding is taking place. When unloading, the strain varies linearly. As the linear strain change is about $3 \cdot 10^{-3}$, which is higher than the yield strain, there will be compressive stresses in the liner after unloading.

Deformations along the whole cylindrical outer surface of the vessel are shown in Fig 6, which clearly demonstrates the progressive character of deformations in the middle part of the vessel under increasing overload conditions.

The overpressure tests demonstrated the progressive (and reversible) strain and deformations in the cylindrical part of the vessel. The behaviour of the lid, top part of vessel and bottom slab was still linear up to 215 bar.

Since the start of the testing of the model, more than three years have passed now (March 1973). Besides measurements at different special tests according to the scheme in Fig 2, there have been periodical readings made even at other times with the intention of studying long term characteristics such as creep in the concrete and relaxation in the tendons.

Fig 7 illustrates the strain variation in the hoop direction in the middle of the vessel. The strain increase is about 30 % of initial strain due to prestressing. This change has taken place mainly during the first year of duration of the experiment. After this period, the strain is almost at steady state. The abovementioned strain change is much lower than would be anticipated from uniaxial creep data on concrete.

The total loss of prestress after three years is about 5 % in hoop tendons and about 4 % in axial tendons.

2.2 Transient tests on thermal insulation system and liner

The most severe thermal load which the liner could be exposed to is the hypothetical failure of the lid torus seal or a steam pipe penetration rupture. This could result in a loss of insulation gas and flooding of the insulation gap with water at saturation temperature. Within some minutes the liner could be heated up to nearly saturation temperature, whereas the concrete vessel essentially remains cold, a thermal load situation very similar to the normal load in vessel designs with a "hot liner".

The most severe transient for the insulation structure itself is the sudden depressurization following a hypothetical main steam line rupture.

After the successful steady state insulation tests [2] it was found desirable to expose liner and insulation system to accident conditions as well. These tests could be performed after slight modifications of instrumentation and auxiliary systems. A multipoint recorder connected to 12 selected thermocouples was used for transient temperature recording. Pressure and water levels were continuously recorded with a multichannel UV recorder.

For the simulation of a loss of insulation gas accident a penetration pipe in the top of the insulation gap (Fig 1) could be utilized. The inner diameter of a throttle disc in the blow off pipe outlet was successively increased during the test series which resulted in increasing flooding rates in the insulation gap (Fig 8). In the last experiment without throttle disc the insulation gap was flooded within 48 seconds. During the flooding phase the steam dome pressure dropped from 70 to 50 bar (Fig 9). After the insulation gap was flooded, the depressurization rate decreased until the steam dome water level after about 450 seconds reached the lower casing edge. Then steam was discharged from the steam dome into the insulation gap and blow off pipe and the depressurization rate again increased significantly.

For safety reasons the liner coolant systems were drained during the experiments and did not contribute as a heat sink. During the flooding phase (Fig 10) the liner temperature rose steeply. After flooding the temperature rise was less steep and it took about 8 - 9 minutes until the liner was heated up to almost saturation temperature. After this point the temperature curves turned and followed the decreasing steam dome saturation temperature. The temperature situation at the turning point is shown in fig 11.

As the liner of the model vessel was not originally designed for 'hot liner' conditions, it was decided to limit the maximum liner temperature during the tests to about 200°C (205°C on fig 10). Inspection after the test revealed that the liner was able to withstand the tested temperature loads without buckling or cracks and the model vessel could be used for further testing without limitations. Anchorage of the liner to withstand the operating temperature of the reactor a limited number of times is to be the subject of a study during phase III.

Finally the capability of the insulation structure to withstand sudden depressurization was tested by simulations of steam pipe ruptures. For these tests the steam pipe in fig 1 was used and equipped with a specially designed valve allowing for instantaneous opening of the full pipe cross section to the surrounding atmosphere.

Before starting the tests, normal BWR thermal conditions (70 bar/285°C) were established. The transients were initiated by sudden opening of the steam pipe valve. Fig 12 shows the pressure-history of two depressurization tests, one with nitrogen filled insulation gap and the other with water-filled insulation gap. The tests are considered representative of the conditions resulting from a steam line failure on a nuclear prototype. The insulation structure withstood the transients without any damage or detectable deterioration of insulating properties.

3. Phase II Technical Studies

3.1 Reference design

The reference design of a 900 MWe BWR-plant with PCRV which was the basis of the phase II feasibility study is not significantly different from the earlier design presented at the 1971 Geneva Conference [4]. The main features are retained, such as: the removable top-closure, a cold liner separated from the removable insulation, integrated main circulating pumps and a totally enclosing, pressure-suppression type of containment (Fig 13).

Novel design features introduced during phase II are:

- location of the major primary-system penetrations including tubes for live steam, feedwater, shut-down and emergency core cooling in the cylindrical barrel of the vessel (Fig 14).
- application of the pre-load wire-winding technique for the hoop prestress.
- introduction of a channel system in the concrete structure as a protective means against potentially severe consequenses of liner leakage.

The core design, performance characteristics and general plant lay-out of the Swedish BWR-plant Forsmark I served as reference plant for the feasibility study.

3.2 Design analysis

Generally the design criteria for the PCRV-structure were based upon the ASME Code draft [5] with additional requirements concerning those design features, which are specific to the Scandinavian design and not included in the Code draft, such as: removable top-closure, free-standing thermal insulation, and liner leakage protective system.

The structural design analysis, based upon finite-element computer codes, confirmed the implementation of serviceability and structural safety requirements (Fig 15). Experimental verification of local safety factors are included in the phase III programme.

3.3 Safety aspects

An essential safety feature is the location of all major pipe penetrations at a level above the reactor core. Thereby the consequences of large pipe ruptures in the bottom part of the vessel are eliminated and the reactor core can be kept flooded during all postulated accidents.

The radial penetrations are designed as double-tube arrangements (Fig 16) in order to minimize stresses due to differential thermal movement and to facilitate periodic inspection and potential repair. Flow-restricting devices are installed to limit the loss-rate of reactor coolant in case of outer sleeve failure.

As a protective means against structural overload in case of liner leakage a system of closely spaced channels has been introduced in the vessel wall. Performance testing of this system will be included in phase III.

The application of external hoop prestress has been favoured due to improved total cost and surveillance possibilities.

3.4 Technological and economic evaluation

As part of the feasibility study a cost comparison was made on a plant-to-plant basis by reference to the Forsmark I station, including all major components, systems and structures which are affected by the choice between the steel vessel and PCRV-alternatives. The analysis was based upon extensive studies on component fabrication, construction techniques and plant construction schedules, carried out jointly by the utilities, reactor supplier and atomic energy organizations with assistance from civil engineering companies and manufacturers of heavy steel equipment.

The process analysis revealed no problems, which cannot in principle be solved at the present stage of technological development and by existing resources in Scandinavia. The results of the cost comparison indicated, even with the relatively low unit output of 900 MWe, a marginal saving on direct costs and a potential reduction in overall plant construction time for the PCRV-plant.

4. Planned Programme of Verification

The phase II technical studies confirmed the feasibility of PCRVs for BWRs. The study report also recommended a programme of experiments, test fabrications and design work.

In parallel with this programme, it is planned to prepare a tender for a BWR with a PCRV so that all the information that is necessary for the placing of an order of a nuclear plant of this design can be made available.

The main items of the verification programme are described briefly below:

a) Liner ventilation system: Existing codes and draft codes for PCRVs already take into consideration the effects of pressure permeation of concrete. As mentioned earlier, the Scandinavian design takes this concern one stage further and proposes the installation of a venting system in the concrete immediately behind the liner for the prevention of gas pressurization of the bulk of the vessel wall. The phase III experiment proposes the building of a model vessel with the possibility of simulating liner leaks of various configurations and determining the effectivity of the system as envisaged for the prototype vessel. Preparatory tests on methods of measuring gas pressure in concrete have already started.

b) Failure tests on lid design and bottom slab: As these parts of the vessel are designed with an inherently higher factor of safety than that of the barrel of the vessel, it is not possible to test them to failure on the Studsvik model. It is planned to test these details on special rigs where the barrel is made sufficiently over-strong so that failure with increasing pressure load should occur in the lid and the bottom slab respectively. The lid failure tests are already under way in Denmark and an interim report is presented at this Conference [6].

c) Pneumatic failure tests on vessel models: A series of such experiments has been carried out at the Norwegian Technical University, Trondheim [7]. It is planned to execute the next two models (Nos. 5 and 6) as part of the Scandinavian development project on PCRVs for BWRs. One of these will be fitted with a removable lid. The technical specifications of these models are under preparation.

Apart from the tests mentioned above, the verification programme will also include fabrication tests on the steel structure for the bottom slab, tests on a simplified bottom slab thermal insulation, vibration tests on the insulation casing with its associated flexible penetration pipes as well as corrosion tests on the stainless steel vessel insulation. The Studsvik model will be subjected to thermal shock and very high overpressure (>2.5 x design pressure) tests.

The phase III programme will also include complementary design work on vessel details, as well as the design of an integral model incorporating such design details as steam outlet penetrations. It is planned to build and test this model only after a nuclear prototype has been committed. In addition to the above design tasks, it is also planned to review the existing norms and codes of practice for PCRVs in order to clarify what modifications, if any, need to be incorporated for use on LWR concrete pressure vessels of the Scandinavian design.

5. Conclusions

To summerise the results of phase II:

- the experimental programme demonstrated the satisfactory behavior of the vessel at very high over-pressure (215 ato = 2.5 x design pressure). The function of the insulation system has also proved satisfactory under extreme accident conditions.
- the technical studies have confirmed the feasibility of concrete pressure vessels for BWRs and resulted in a reference design for a 900 MWe BWR with a PCRV, based on the core and general plant layout of the Swedish State Power Board's Forsmark I plant. A cost comparison on a plant-to-plant basis between the PCRV alternative and a steel vessel plant of the Forsmark I type resulted in a marginal advantage in direct costs for the PCRV design. The comparison also showed a potential reduction in over-all plant construction time for the PCRV plant.

- The phase II report recommends the practical verification of a number of aspects such as the local factors of safety for the lid and the bottom slab as well as the function of a liner ventilation system. Work on a number of items of the verification programme has already started. It is planned that the programme will be completed by mid - 1975.
- A tender for a BWR with a PCRV will be prepared in parallel with the execution of the verification programme. This should ensure that all information necessary for the commitment of a nuclear prototype should be available by autumn 1975.

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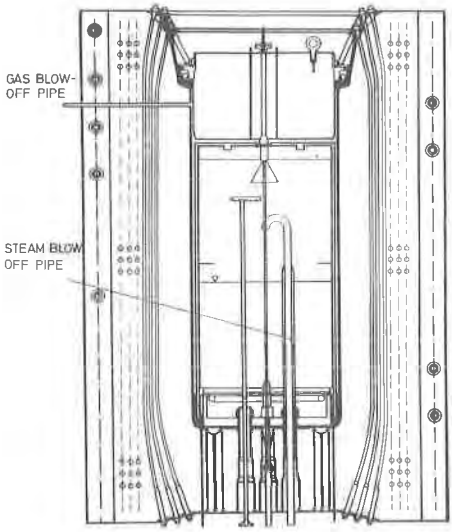


FIG.1 PCRV MODEL VESSEL

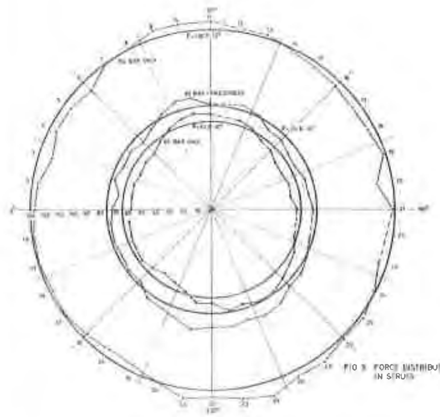


FIG 3 FORCE DISTRIBUTION IN STRUTS

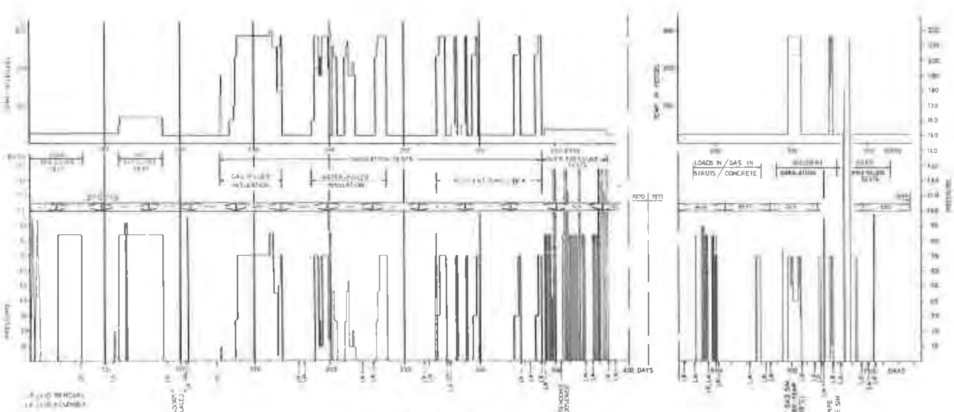


FIG.2 HISTORY OF TESTS PERFORMED

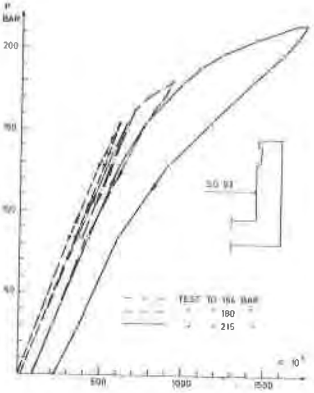


FIG 4 HOOP STRAINS ON THE LINER

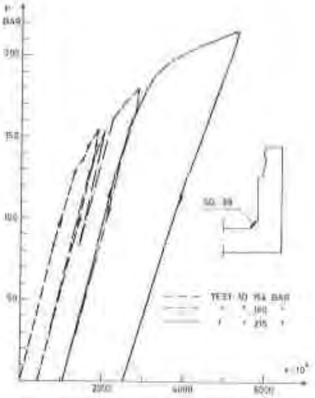


FIG 5 MERIDIONAL STRAINS AT THE INTERNAL CORNER

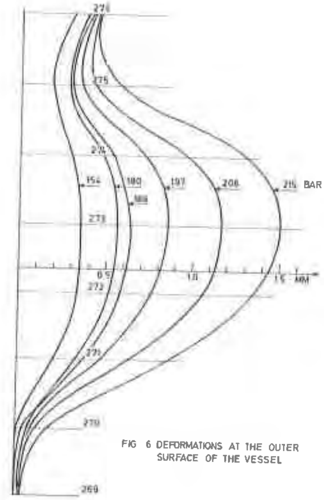


FIG 6 DEFORMATIONS AT THE OUTER SURFACE OF THE VESSEL

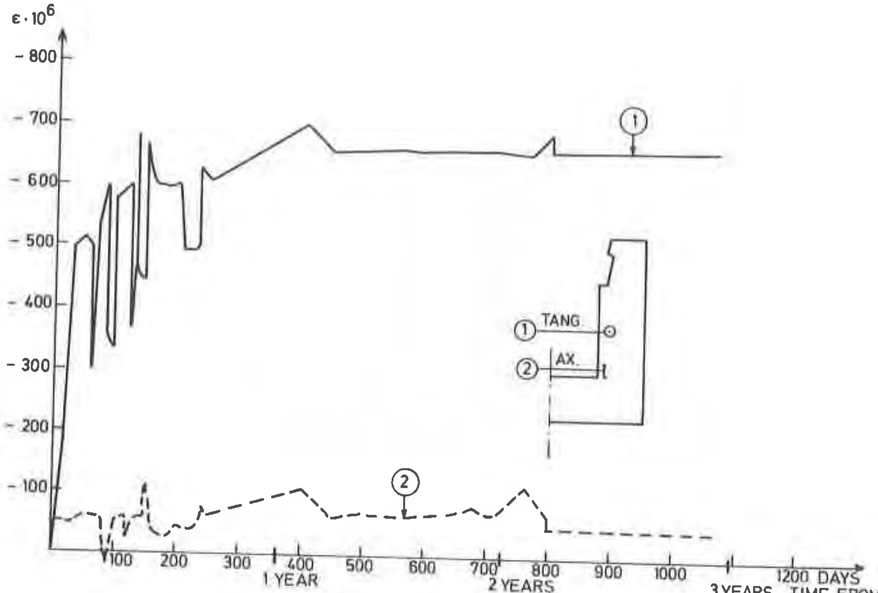


FIG. 7 STRAINS IN THE VESSEL WALL

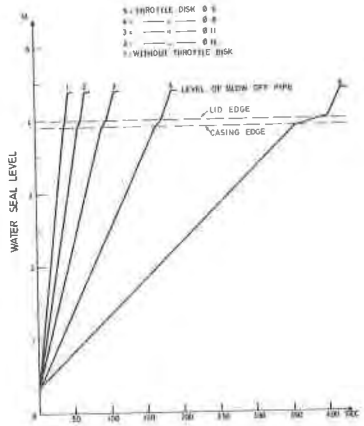


FIG 8 LOSS OF INSULATION GAS PARAMETER STUDY: WATER SEAL LEVEL DURING FLOODING

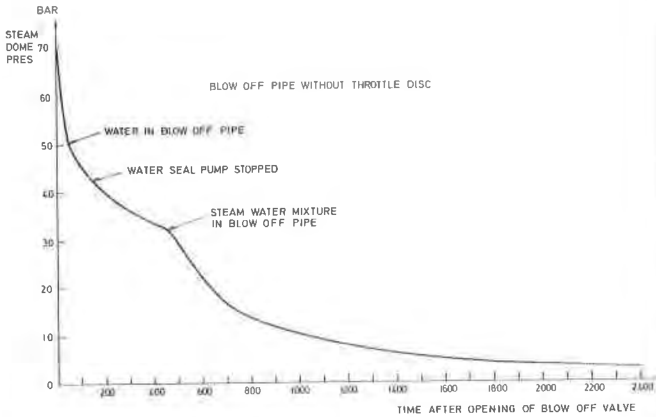


FIG 9 LOSS OF INSULATION GAS: STEAM DOME PRESSURE

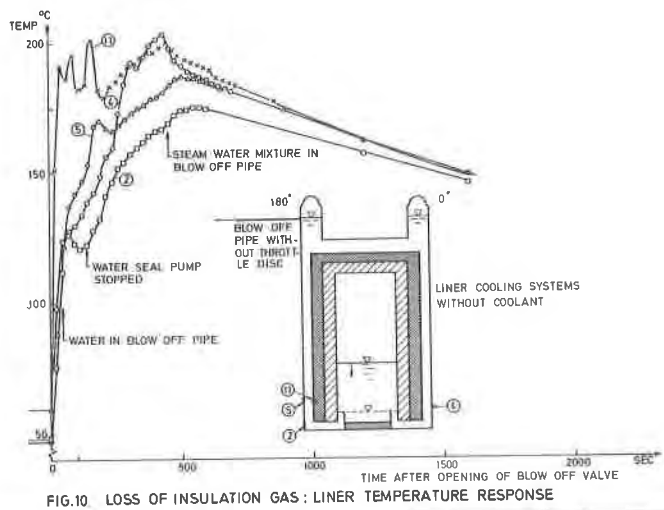


FIG.10 LOSS OF INSULATION GAS: LINER TEMPERATURE RESPONSE

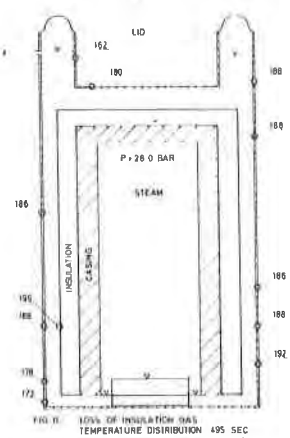


FIG 11 LOSS OF INSULATION GAS TEMPERATURE DISTRIBUTION 495 SEC AFTER TRANSIENT START NO THROTTLE-DISC
DATE 71-10-19 TIME 11 09

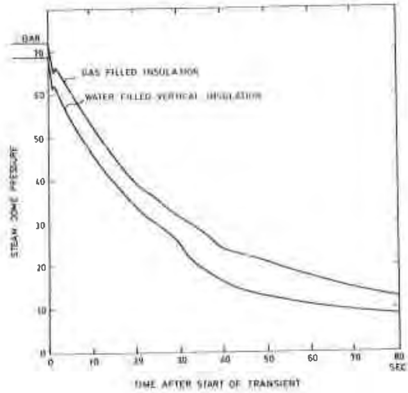
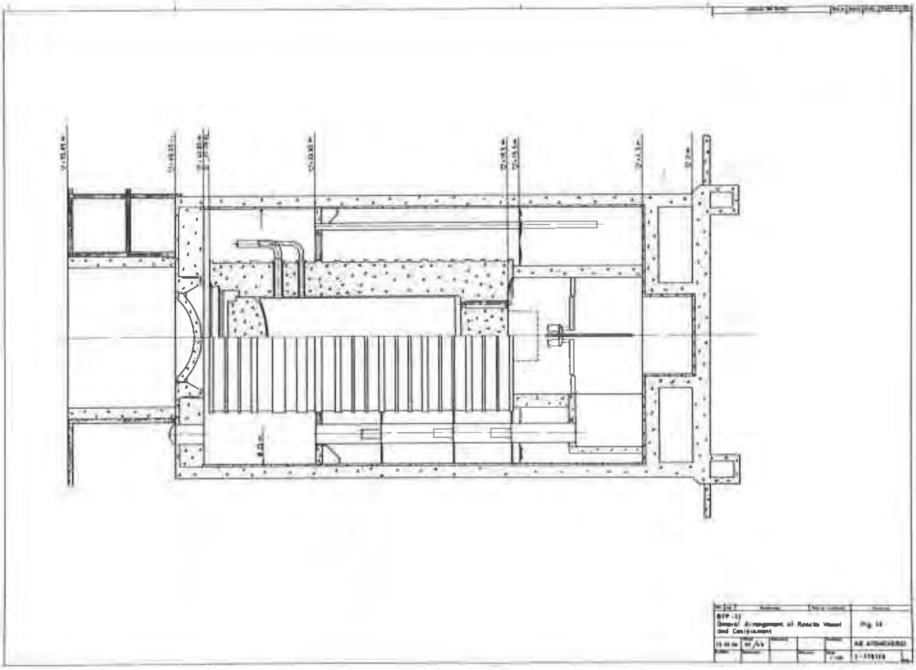


FIG 12 DE-PRESSURIZATION TEST STEAM DOME PRESSURE AFTER SIMULATED STEAM PIPE RUPTURE



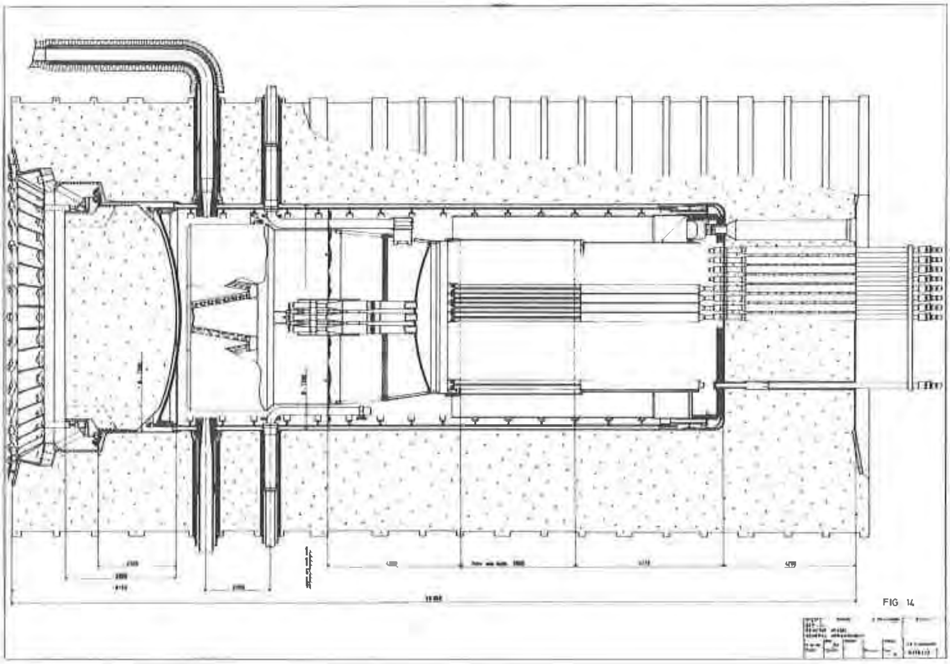


FIG 14

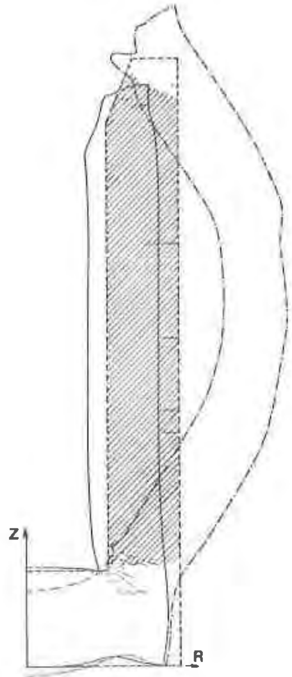


Fig 15

VESSEL DEFLECTION AND CRACK PATTERN
$P = 212\ KP/CM^2 = 2.50\ P_d$

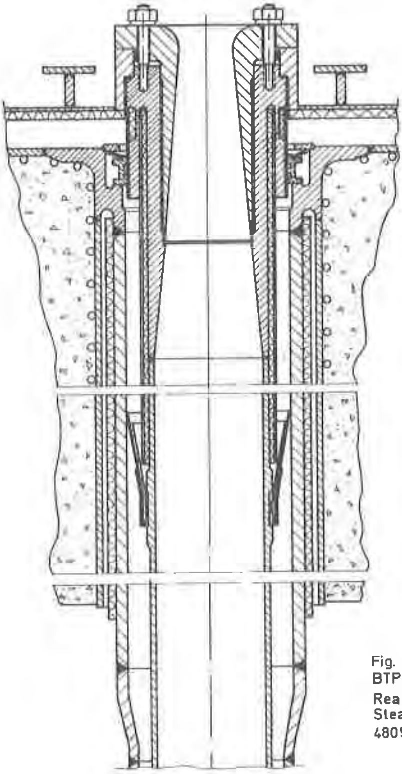


Fig. 16.
BTP - II
Reactor Vessel
Steam Outlet
480505