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THERMAL REACTOR SAFETY RESEARCH IN SWEDEN

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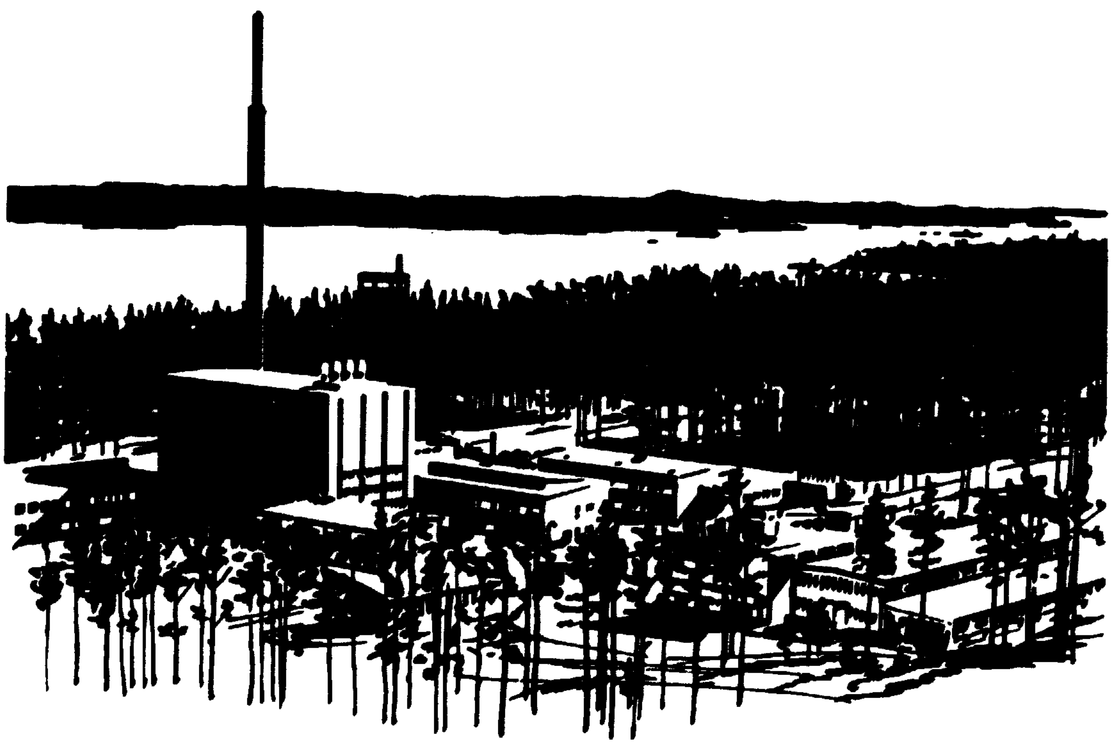

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ABSTRACT

Sweden benefits in many ways from the reactor safety research work performed in other countries. Its own activity is complementary to this effort but a certain fraction is oriented towards safety issues which are intimately related to the special design of the ASEA-ATOM boiling water reactor design.

Through the availability of the decommissioned Marviken reactor plant Sweden has been able to pay a leading rôle in integral containment experiments with international participation. Joint efforts with other countries are now devoted to defining new large scale experiments to be performed in the unique Marviken facility.

The largest portion of the safety research program in Sweden is performed by Studsvik Energiteknik AB but various universities, consultant firms and research institutes are also involved. In addition, large efforts are made by the reactor vendor ASEA-ATOM but this work is not included in the present report.

The overall annual budget is at present between M$7 and 8 with three governmental authorities as the main financing bodies.

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1. BACKGROUND

As is well known light water reactors have been chosen for the Swedish nuclear power program. There are at present five boiling water reactors (BWR) and one pressurized water reactor (PWR) in operation and four other reactors - two BWRs and two PWRs - are technically ready or close to be ready to start operation. The boiling water reactors are of the ASEA-ATOM design while Westinghouse is the vendor of the pressurized water reactors

The Swedish view of nuclear safety agrees with common international apprehension and practice. For example, American regulations and standards often serve as a basis for judgements and decisions made by the Swedish safety authorities.

The Swedish nuclear power program includes some components of a closed fuel cycle. Fuel for boiling water reactors is manufactured by ASEA-ATOM. Provisions are being discussed for intermediate storage of spent fuel and final deposition of radioactive waste. Safety research is carried out in Sweden concerning these parts of the cycle. However, such matters have been judged to lie outside the scope of this article and will consequently not be mentioned in this report.

To a large extent Sweden can benefit from reactor safety research carried out in other countries. Much effort is spent on following up work done abroad, interpreting the information received, and adjusting the results to Swedish conditions. The contribution that our country can make to the overall knowledge in the field is limited to areas for which our resources in equipment and manpower are particularly well suited. In addition, a large portion of the Swedish safety research program is set up to deal with questions related to the special design of the ASEA-ATOM reactors.

By research agreements of various kinds with many foreign organizations - from general information exchange agreements to direct collaboration in research projects - the possibilities to obtain and evaluate results from research activities abroad are greatly enhanced.

The joint sponsoring by several countries of large safety projects within our country is a prerequisite for the possiblity to perform such experiments. The Marviken projects may here serve as an example.

The fact that Sweden has an independent vendor (ASEA-ATOM) of BWR systems has lead to the safety work being concentrated on such systems. Almost all of the early effort related to loss-of-coolant accidents (LOCA), for instance, dealt with problems relevant to BWRs. The same was true for the first series of containment experiments performed at the Marviken facility. However, there are many questions concerning PWR's which are gaining importance. In consequence, much of the present activity has a bearing on PWR systems.

The LOCA problem still plays a dominant rôle in the Swedish research program. With the availability of the large US computer programs RELAP, TRAC and others the theoretical effort has, however, shifted towards verifying and possibly modifying such codes rather than to developing new codes. Studies of fuel and canning behaviour under various conditions have always played an important part of the research activities and continue to do so. The volume of materials research, on the other hand, tends to decrease. Further, large scale experiments on conventional containments are not likely to come about but the interest for alternate concepts is growing and may result in new theoretical and experimental activities. Analyses of failures of various components, such as valves of all kinds, give rise to research projects with high priority. Finally, activities related to the man-machine problem, human performance and overall risk analysis are steadily growing in importance.

2. ORGANIZATION

Up to the early 1970's the Swedish Government financed most of the safety research and development work in Sweden by direct grants to AB Atomenergi (now Studsvik Energiteknik AB). AB Atomenergi, in collaboration with ASEA and the utilities, formulated the research program and was responsible for its execution. With the foundation of the ASEA-ATOM company, and the concentration on light water reactors as the preferred choice for power reactors the system for funding safety research gradually changed to the prevailing one.

Between 1972 and 1974 a special committee set up by the Ministry of Industry existed under the name of The Swedish Board for Reactor Safety Research. It was instructed to initiate projects relevant to reactor safety and environmental problems. The funding came through a special fee from the utilities, and the total budget for this three year period was 5.6 M$ (25 MSwcrs). Later research departments were created within The Swedish Nuclear Power Inspectorate (in 1975) and The National Institute for Radiation Protection (in 1976). The responsibility for the ongoing projects was transferred along with the funds to these two governmental authorities. Approximately two thirds of the research volume was considered to belong to the Nuclear Power Inspectorate and one third to the Radiation Protection Institute. A few projects were taken care of by the interim National Council for Radioactive Waste. In 1975 The National Board for Energy Source Development, to some extent responsible also for reactor safety research, was founded. The safety research of the Nuclear Power Inspectorate is financed through a special fee from the utilities with construction permits for power reactors. The Radiation Protection Institute and the Energy Source Development Board, on the other hand, receive their funding from the government. From Table 1 the annual budgets for reactor safety research including problems with relevance to environmental protection in Sweden can be seen. For the country as a whole an undisclosed amount for the safety-oriented work done by ASEA-ATOM and by the utilities on their own expense should be added.

Table 1

Annual budget for Swedish reactor safety research, including environmental protection research, in M$

Governmental authority \ Fiscal year	1975/76	1976/77	1977/78	1978/79	1979/80 (proposed)
Nuclear Power Inspectorate	1.6	3.1	4.0	4.7	5.4
Radiation Protection Institute	-	0.4	0.5	0.6	0.8
Energy Source Development Board	3.7	2.2	2.2	1.9	1.8
Total	5.3	5.7	6.7	7.2	8.0

In giving priorities to research areas and individual projects The Nuclear Power Inspectorate and The Radiation Protection Institute are mainly guided by their needs as regulatory and inspecting bodies. Each authority has at its side a research council that assists and advices them on project priorities. The research sponsored by the Energy Source Development Board is more directed towards basic safety research and the development of improved safety systems.

The largest part of the Swedish reactor safety research program is carried out by Studsvik Energiteknik AB. Other Nordic research centers are also involved. Work is performed by universities and special research institutes, for example The Research Institute of National Defence. Several consultant firms are engaged as well as the reactor vendor ASEA-ATOM. The funding of safety research in Sweden is shown graphically in Fig 1.

3. RESEARCH ACTIVITIES

The Swedish research program comprises efforts in all the main areas normally associated with LWR nuclear safety. In summarizing the work we have followed an approach where we start with the innermost barrier against release of radioactivity, the fuel element, then go on to the primary circuit and conclude with the containment. Problems related to releases of radioactivity through the containment are also dealt with. A section on accident analysis and risk assessment is furthermore included. Short summaries of each main area are given together with highlights from particular projects.

THE FUEL ELEMENT

The integrity of the first barriers - the oxide fuel and the cladding - against release of radioactivity is of great importance for the normal operation of a reactor as well as for accident conditions. Several large irradiation programs with international participation - Inter-Ramp, Over-Ramp etc - are under way at the material testing R2 reactor at Studsvik. In these projects phenomena related to pellet cladding interaction (PCI) are studied. Ramp tests are performed at various burnup levels under experimentally well defined conditions with the prime objective to find the limiting conditions for safe operation [1].

In the Inter-Ramp program a set of 20 BWR-type fuel rods are base-irradiated under long-term cyclic conditions in the R2 reactor, where they are subsequently ramped at the same fast rate to pre-set terminal power levels and hold for 24 hrs or until failure [2]. The failure incidence is seen to be surprisingly systematic showing a sharp failure transition, ordered times to failure and a consistent fission gas release, which abruptly increases at about the power level of the failure incidence (420 W/cm). Non-penetrating incipient cracks are found in rods ramped to just below the failure threshold and are coincident with the onset of plastic deformation for the fuel cladding [3].

The Over-Ramp program involves fast power rampings of PWR fuel rods base-irradiated to burnups between 10 and 30 000 MWd/tU in commcercial power reactors.

These two programs are followed by a few others. From a safety point of view the Super-Ramp program attracts special interest as it deals with the PCI performance at high and extended high burnup levels.

Experimental studies of <u>fission product behaviour</u> within intact fuel rods have been in progress for many years. In one series of experiments, the axial and radial distributions of fission products within the fuel and on the inner clad surface after irradiation at a well defined linear heat rating have been studied using gamma spectrometry, beta autoradiography and ceramography. Measurements have been performed after only a few days cooling, permitting investigations of tellurium and iodine in addition to the longer-lived barium, cesium and ruthenium fission products.

This program is generating new data of the type shown in Fig 2. The fuel rod designated as S176-3 had been irradiated to a significant burnup in the Ågesta reactor, followed - after a substantial cooling period - by a 5 day irradiation in the Studsvik R2 test reactor. Shortly after the irradiation, axial gamma scanning was performed and 2 mm thick fuel discs were removed at three locations along the rod.

In the figure, the results of diameter gamma scanning and beta autoradiography are compared. Pronounced migration of Te-132, I-131, I-133 and Ba-140 can be observed. The movement of Ba-140 to the fuel centre, and the formation of an enriched band of Te-132 were unexpected. Measurements of Cs-134 and Cs-137 were performed later after some months cooling. The results are being correlated with the observed fuel structures after irradiation and the radial temperature distributions calculated from the two power levels to which the fuel was subjected.

In other experiments, the volatility of fission products accumulated in the clad-fuel gap and the outer fuel zone at temperatures up to 1200 ^{o}C is being studied in order to define the possible contribution from fission product vapour pressure to the fuel rod internal pressure during a loss-of-coolant accident. The results from these two series of experiments up to about summer 1976 have been published [4].

A program is coming to an end in which fission product leakage from defect fuel rods is studied. Measurements of fission product release from fuel irradiated in the Oskarshamn, Barsebäck and Halden reactors are compiled and analyzed. Other aspects of the program are measurements on loop water samples from the Studsvik R2 test reactor.

The measurements of the iodine release from defect test fuel elements in the Halden Boiling Water Reactor have been completed and reported [5]. The release fraction, f, of ^{131}I was shown to be a function of the average fuel centre temperature, T_c (^oC), according to the equation

$$f = 3.7 \cdot 10^{-6} \cdot \exp(4.7 \cdot 10^{-3}\ T_c).$$

Various models have been used in the analysis of the release data for iodine and for gaseous fission products [6].

The measurements of fission product release in Swedish power reactors have yielded release fractions and release coefficients for a number of fission products. As an example the release fraction f for ^{131}I was determined as $3.6 \cdot 10^{-4}$, $7.5 \cdot 10^{-4}$ and $4.7 \cdot 10^{-4}$ for the Oskarshamn I BWR reactor for the years 1975, 1976 and 1977, respectively.

Measurements of the increased leakage of fission products in connection with shutdown have been made in Oskarshamn I [7]. The measured integrated release of ^{131}I in a spike has been compared with predicted release, based on a model of General Electric.

In a new project with ASEA-ATOM and Studsvik as participating parties further studies are performed and a model is being developed by which levels of radioactivity in the primary circuit due to this phenomenon can be estimated.
A new project is further being planned with the aim to measure directly fission gas contents in power reactor fuel at different burnups. The work will be carried out at Studsvik in collaboration with ASEA-ATOM.

The growing interest for taking fuel to higher burnups has initiated new R&D projects. At Studsvik the release of gaseous fission product at extended burnup is being studied on fuel irradiated in various reactors.

Experimental work related to the canning material is also being done. A particular problem of concern is zircaloy creep in steam during a slow temperature transient which would result from a LOCA in a Swedish BWR with external circulation loops. Experiments have been performed during which the strain was measured as a function of time during a simulated slow transient in steam. These data are used as a basis for modelling creep in steam. A few related projects have been performed where the effects of oxide on the outside and fission products on the inside were studied.

On the theoretical side much of the activity is concentrated to the implementation and use of the well-known codes GAPCON-THERMAL-2, MOXY-EM and TOODEE-2. Work is also carried out on models for describing the migration of fission products in the fuel. The models are based on the experimental results on radial fission product redistribution mentioned above. Barium and iodine motion has been studied [8, 9].

THE PRIMARY CIRCUIT

A large fraction of the total safety related work falls under this heading. The different ongoing projects belong in a more or less pronounced way to one of the categories: measures for accident prevention, systems for accident mitigation, accident analysis,and measures to reduce the consequences of an accident.

Materials Research.

The main efforts in this area have been devoted to the mechanical and metallurgical properties of materials for pressure vessels and piping. Several projects on fracture mechanics are going on including studies of fracture toughness, corrosion fatigue, stress corrosion crack growth rates, dynamic analysis,probabilistic safety evaluations,flaw detection,and ageing of material properties.

The fracture mechanics work is mainly concentrated to the determination of the "upper-shelf" fracture toughness of the pressure vessel materials (A533B) by the J-integral method in the temperature region 20-350 $^{\circ}$C. An important result of this work was the demonstration of the influence of the strain rate on the temperature at which the minimum in fracture toughness occurs (cf fig 3 from [10]).The work was continued on the welded bonds and the heat-affected zones of A533B steel plate. The results obtained on these materials show [11] that the temperature-dependence of fracture toughness is similar to that of A533B plate, although the level of fracture toughness is generally higher than for the plate material. The conclusions are that the heat-affected zone and the welded bonds are susceptible to strain ageing as well. During the current year similar determinations of fracture toughness will be made on A508 steel. The results should be available in 1980.

The corrosion fatigue work is concentrated on the effects of BWR-environment on the fatigue properties of A316 type of stainless steels [12].The work was started for the purpose of gaining more knowledge of the phenomenon of environmental enhanced crack growth rate, knowledge that could be used for improving the design of BWR reactors. The ultimate goal is to quantify the effect of small environmental changes on the crack growth rate and to study different material conditions. Recently the work has been extended to include studies of the pressure vessel steel A533B.

Measurements of residual stress distributions through large weldments have been carried out at Studsvik since 1973. Stress distributions at different levels through the specimen are determined experimentally and the stress system is then reconstructed mathematically using a method developed in collaboration with the Department of Strength of Materials at the Royal Institute of Technology, Stockholm. Results previously obtained [13]indicate that large residual stresses can exist in a thick weldment even after a post-weld heat treatment at 620 ^{0}C for 5 hours.

Current research at Studsvik includes the effect of post-weld treatments and residual stresses introduced in reparing welds as well as the estimation of the residual stresses in a nozzle, using a combination of experimental and mathematical techniques.

It was found in[10,11] that ageing of material properties is of particular interest. This area of research has been taken up at the Swedish Institute for Metals Research and a study of strain ageing in welded vessel material (A533B) has recently been reported [14]. The work continues with the aim of measuring the fracture toughness of reactor vessel material aged in different ways.

Ultrasonic testing (UT) is the most widely used technique for inspection of weldments in reactor pressure vessels in Sweden. There are special problems connected with the testing of austenitic stainless steel and inconel weld materials. A project was started in 1973 at Tekniska Röntgencentralen in Sweden in collaboration with Southwest Research Institute in the US with the aim to clarify these difficulties and to analyze and develop new ultrasonic techniques for the testing of such materials. Three phases, entailing literature search, weld sample fabrication and qualification, and testing with conventional ultrasonic methods have been reported [16]. In the final phase of the program, to be reported in the beginning of 1980, some special techniques for ultrasonic testing are being investigated.

For pressurized objects which are difficult to inspect because of their position in the plant a special projection principle for ultrasonic testing with registration on-line is being discussed. The apparatus [17] has the added advantage that it can be used with remote control in high level radiation environment. The use of UT probes with a focussing device is a way to increase the accuracy of detecting defects. These two techniques are being investigated by the Swedish Plant Inspectorate and the reports are planned for the Spring of 1980.

Work related to Loss of Coolant Accidents.

The larger part of the Swedish LOCA oriented activities is concentrated to Studsvik. Much of the work is performed within the frame-work of the Nordic cooperative agreement

Theoretical work on probabilistic elastic-plastic fracture mechanics has been carried out for some years at the Department of Strength of Materials at the Royal Institute of Technology in Stockholm. There are at present several interrelated projects which are intended to generate data regarding stress distribution, statistical distribution of properties of materials, and flaw concentration. University departments, consultant firms and the governmental Swedish Plant Inspectorate are involved in these activities. In particular,various improved techniques are studied for the detection of flaws in pressurized components, especially the reactor vessel.

In the Barsebäck 2 boiling water reactor plant a special design of the feed-water nozzle has been used in order to minimize temperature induced stresses and the likelihood for ensuing cracks. A project under the leadership of ASEA-ATOM has been performed entailing measurements of the temperature distribution. It seems clear that the new design in conjunction with a special operational procedure has gre. reduced the risk for severe stresses.

For many years considerable effort was devoted to prestressed concrete pressure vessel technology to be adapted for use in BWR design [15].Advanced model testing was made and the general technique is well understood. Due to lack of prospects for a full scale realization in a long time the continued effort is, however, now restricted to a minimum.

NORHAV. Through this joint effort Sweden, Finland, Denmark and Norway have also been able to make an agreement with the US Nuclear Regulatory Commission (NRC) through which the results of the LOFT project become available to the Nordic countries.

On the theoretical side most of the effort is concentrated on the use and application to various problems of the codes RELAP4, MOXY and TOODEE being part of the US WREM package. Complementary work is done by AB Fjärrvärme, which is a consultant firm to the Nuclear Power Inspectorate in ECC matters. The TRAC code has further been installed at Studsvik, but it will take a long time before it will be used routinely. An improved version of RELAP4/Mod 3 has been used for extensive studies of the ASEA-ATOM Forsmark 1 reactor with internal main circulation pumps. Some modifications of RELAP4 have been required to represent special features of the reactor, for instance the internal pumps. Extensive parametric studies have been performed including calculations on small as well as large breaks. The calculations show that there will be no uncovery of the fuel for any of the break sizes and as illustrated in Fig 4 (reproduced from [18]) the increase in the fuel cladding temperature is predicted to be small.

To obtain proficiency in the use of the WREM codes Studsvik has further participated quite actively in the CSNI standard problem calculations, an efficient form for small scale international cooperation.

Moreover, Studsvik is participating in the development of the reflood codes NORCOOL-I and -II. The work is performed at Risö and the NORCOOL-I code is practically completed. It is now being tested against experimental data including results from reflood experiments at Studsvik. Some related work on two-phase systems and ECC is done at the Department of Reactor Technology at Chalmers Institute of Technology.

LOCA related experiments have mainly been performed in two loops at Studsvik, the GÖTA and FIX loops. The test programs are summarized in Fig 5.

In the GÖTA loop a large effort has been devoted to studying the efficiency of spray cooling using an electrically heated 64 rod fuel bundle. The purpose of the experiment has been to determine such quantities as convective heat transfer coefficients, rewetting times for the channel and rods, the influence of pressure, power level and spray flow on cooling properties,and the dependence of internal spray distribution on cooling. The analysis of these experiments has recently been concluded. The test rod bundle was so distorted, however, that it was not meaningful to try to evaluate the heat transfer coefficients. Nevertheless, an important quantitative result has been obtained, namely that the accumulated heat was removed by the cooling system despite the distortion of the bundle.

The experiment with 64 heated rods has been complemented with two less extensive series on a cluster with one and four adiabatic central rods, respectively. This type of cluster resembles more closely to-day's BWR fuel.

The spray cooling studies in the GÖTA loop have been complemented with investigations of the influence of counter current flow limitation (CCFL). Steam at various rates has been introduced at the bottom end of the cluster and its influence on the spray cooling has been studied. The interaction between the steam and the spray cooling causes characteristic oscillations in the bundle pressure drop. At low steam velocities two periods of CCFL occur corresponding to the rewetting of the channel walls and of the whole bundle. Complementary experiments on the influence of different geometries have been made at the Department of Reactor Technology at the Royal Institute of Technology.

The experiments at the spray cooling loop were concluded by a series of reflood experiments [19]. It is against the results of these experiments that the NORCOOL-I code now is being tested.

Another large scale experiment has been performed under simulated loss of coolant conditions. The study was made in the FIX loop at Studsvik and aimed at determining time to dry-out (t_{d-o}) under various transients such as simulated top and bottom breaks, pump trip etc. The effect on t_{d-o} of a possible interaction between a hot channel and neighbouring channels was of particular interest. In the FIX loop, the hot channel is represented by an annular single channel in communication with a larger channel. The latter consists of an electrically heated 36 rod cluster. The experiments have been completed and the results have been compared with RELAP calculations. The agreement theory-experiment is generally acceptable as seen from Table 2 and Fig 6. No effect of parallel channel interaction was observed.

The FIX loop is now being modified for a new set of experiments. The purpose of the new phase is to generate additional experimental data on heat transfer during the whole blowdown phase up to the point in time when the spray cooling starts. For these experiments it is even more important than for the FIX I measurements that the loop in all relevant parts is a correctly scaled system. Large efforts have been devoted to this problem and extensive supporting calculations have been performed using the ASEA-ATOM LOCA code system GOBLIN. To be able to extend the measurements into the post dry-out region directly heated rods of a new design have been developed at Studsvik. With these rods measurements can be performed up to rod surface temperatures of the order of 800 C. At present the experimental facility with its instrumentation is being built. The actual experiments are planned to start in 1980.

In several projects the safety analysis codes of ASEA-ATOM such as the GOBLIN program are verified against experiments. For proprietary reasons only certain parts of the results are generally available.

An experiment to determine the energy release from fission fragments is going on at Studsvik. The aim of the measurements is to provide data for improving the accuracy of the decay heat curve. The evaluation of the experiments is done in close collaboration with related projects in the US. The previous phase of the Swedish work concerned determination of the energy distribution and the total energy of gamma radiation emitted in the time interval 10 to 1500 s after thermal fission in U-235 [20]. The gamma measurements have been followed by similar studies of the beta decay which will be reported in the Autumn of 1979. During the third phase of the project the beta decay from Pu-239 is to be measured.

Plant Surveillance.

The noise of process signals can be analysed to yield information about the plant status, e.g. resonance peaks in the power spectrum caused by vibrations, control loop instabilities etc. By observing deviations from the normal noise pattern incipient failures can be detected at an early stage.

Development of noise analysis methods for surveillance and early fault detection is carried out in collaboration with the utilities in a comprehensive program comprising both BWR and PWR applications. Measurements have been performed in all Swedish nuclear power plants currently in operation. From the analysis and interpretation of these measurements a basis has been obtained for the design of an on-line surveillance system using noise analysis patterns [21].

As an example of the sensitivity of noise methods two normalized auto power spectral density curves of the nuclear power signal from a boiling water reactor are displayed in Fig 7. One of the curves shows a sharp and easily detectable resonance at 0.17 Hz, which was traced to unsuitable control system performance and could be eliminated by appropriate adjustments of control parameters.

A study on methods and problems connected with the use of computers in safety related systems in nuclear power plants has been running since the beginning of 1977. The goal of this work is to find criteria and recommendations for the design of computer systems in applications where high reliability and safety is demanded. The Studsvik work is concentrated on two areas, specification and verification of software. The work in these areas is performed in close contact with the laboratories at Harwell, Risö, Karlsruhe and Garching.

THE CONTAINMENT

Containment studies have for a long time played an important rôle in the Swedish safety research program. This is primarily due to two factors - firstly, the importance of the containment as the outermost barrier against uncontrolled release of radioactivity and secondly, the availability in Sweden of a full scale facility, the decommissioned Marviken nuclear plant, for such research.

The Marviken Experiments.

The full scale safety experiments in the Marviken plant started in 1972. The first set of experiments dealt with the response of the pressure-suppression (PS) containment to simulated ruptures in pipe systems connected to the pressure vessel. The experiments aimed at providing information about the influence of such factors as the amount of energy in water and steam in the pressure vessel, the size and location of the break, the wetwell temperature etc on the pressure build-up and the temperature rise in the drywell and wetwell and on the mechanisms involved in the blowdown processes.

This first series of blowdown experiments called CRT-I (Containment Response Test) [22] was performed as an international project with participation from the Nordic countries, FRG and the US. The results of the CRT-I experiments emphasized the need for further blowdown tests (CRT-II) [23] to get more detailed information about the phenomena involved. More specifically, the CRT-II experiments aimed at providing knowledge about the influence of the following factors on pressure oscillation in the containment:

- Total vent pipe area
- Submergence depth
- Pool surface area
- Pool mass (varying pool level and pool surface area)
- Vent flow path geometri
- Pool temperature (due to heating during blowdown)
- Vent mass flow rate
- Mixture compositions in vents (fraction of air).

A supplementary test which was performed in conjunction with the CRT program was the measurements of impact loads on structures in and above the pool.

The experimental phase of the project was started in February 1976 and was completed in late autumn the same year. The project was a joint undertaking with Sweden, Denmark, Finland, Norway, Japan, France, FRG, the Netherlands and the US as participating parties.

The experimental phase of a third multinational project at Marviken has recently been concluded. The objective of this project (Critical Flow Tests, CFT) was to obtain data for critical flows in short pipes having diameters up to 500 mm. Data were collected at subcooled and low-quality saturation fluid conditions in the range that would be valid in a full scale reactor loss-of-coolant accident. Fig 8 shows the test nozzle location and the flow paths in the test building.

The t..st series included about 25 blowdowns with a maximum water-steam discharge rate of 15000 kg/s. Fig 9 [24] shows as an example pressure profiles in the nozzle test section and rupture disc assembly from pressure measurements at the nozzle wall. The pressure profiles are presented as pressure ratios formed by dividing the measured pressures by a pressure approximately equal to the stagnation pressure at the nozzle entrance. The pressure ratio corresponding to the saturation pressure associated with the stagnation temperature is included as a point on each profile for comparison purposes. Pressure profiles corresponding to times when the stagnation conditions were subcooled all indicate that the pressure at the test section entrance drops below the saturation pressure and that a zone of pressure recovery exists at the upstream end or the test section extending as far as 0.6 m downstream of the test section entrance. These results are thought to be indicative of non-equilibrium thermo-dynamic conditions and perhaps a separation of the flow. The zone of pressure recovery is followed by the zone of gradually decreasing pressure caused by viscous losses. The pressure recovery zone is not present in the pressure profile when the stagnation conditions are two phase already at the inlet of the nozzle.

Analysis of Experiments and Theoretical Work.

Containment problems have further been studied in the TESTA facility - a simplified small scale model of the Marviken PS containment - and in a special model at Studsvik. The TESTA experiments, which dealt with pressure oscillation measurements, have been terminated while the experiments at Studsvik are under way. In the latter ones the level rise due to the rapid introduction into the wetwell of non-condensable gas during blowdown is studied as well as the load on the ceiling separating the wetwell and the drywell.

In parallel with the experimental work a computer code development and verification program has been undertaken in cooperation with ASEA-ATOM. A containment response code COPTA is now available and has been tested against selected experiments. COPTA treats the containment as a multicompartment volume and includes a complete model for the reactor tank. It also contains a semitheoretical level rise model for the water level in the wetwell.

A fairly extensive evaluation effort is underaken to analyse the results from the Marviken CRT-II experimental program. The objective of this project is,among other things,to examine the experimental results searching for interaction between pressure spikes at different vent pipes and to systematize the influence of different test parameters on pressure oscillations and pressure spikes. Efforts are also made to find a model for interpretation of the Marviken data so that they can be applied to a given BWR containment.

The analysis of certain parts of the containment during a blowdown involves very complicated calculations. This is particularly true for such structures which contain a large amount of steel reinforcement. A project has therefore been initiated in which a consultant firm to the Nuclear Power Inspectorate in collaboration with several University departments is going to apply a finite element method for such analyses. Results from this work will not be available until 1980.

The containment is likely to be subjected to heavy loads during a blowdown using the safety relief valves. Such tests have been performed in the Forsmark 1 plant by ASEA-ATOM. The analysis has been concluded recently and all non-proprietary information will be published shortly.

Evaluation work of the CFT experiments has started in Studsvik. In a first phase the experimental results are being compared with calculations using RELAP4.

The performance of isolation valves during a LOCA is a crucial question for the integrity of the containment. These large components have not been tested under realistic accident conditions. Studsvik is participating in the German HDR* program in order to obtain results from isolation valve experiments. The results are to be analysed and applied to the type of valves installed in the Swedish nuclear power plants. This work will be carried out by Studsvik in collaboration with ASEA-ATOM, Swedish consultant firms and utilities, and the Finnish research center VTT. The project will last for at least two years.

Designing for Earthquake Loads.

Although Sweden and its neighbouring countries are seismically relatively stable, earthquakes have happened in this region. Measurements have been commenced by The Research Institute of National Defence in order to increase the knowledge of Swedish seismic activity and to elucidate its connexion with geologic and tectonic structures. The aim is to improve the present calculations of structural loads due to accelerations at the Swedish nuclear power plant locations.

17 detecting stations are to be built in Sweden and 4 more in Denmark will be connected to the system. Some of the stations are already operable, and data are being collected. This phase will continue for another three years, and final results are planned for the end of 1981.

*) HDR = Heiss Dampf Reaktor project

New Containment Concepts.

In Sweden, an Urban Siting Study [25] was performed in 1969-1974 and published shortly before the Rasmussen Study (WASH-1400). An important result of both studies was the observation that very large accident releases of radioactivity with adverse health effects on the population in the vicinity of a nuclear power plant would occur only as a result of a core meltdown in combination with extended containment rupture. The rupture could be caused by overpressure, melt through, missiles from steam explosions or a combination of these factors.

Work is going on in several countries on new containment design that would reduce the probability for radioactivity release in case of a core melt and mitigate the consequences of such an accident. At Studsvik an effort is devoted to elucidating the conditions and possibilities for making such changes in existing or future BWR containments that would considerably reduce the consequences of a core melt. In parallel with the design studies small scale experiments have begun on filtering systems using water sprayed rock columns.

Tests of Sensors for Steam and Water Leakage.

Functional tests on switches used for indication of steam or water leakage in a nuclear power station have been performed at Studsvik under well defined conditions. High temperature steam was introduced with a fast opening valve into a large clean hall with concrete walls. A large number of switches and indicators were placed at different distances from the valve and the function of the switches and indicators was studied under various conditions [26].

The proper function of these switches, which are pressure or temperature sensitive devices,is very important from the safety point of view. The test program included checks of the computing methods which are used for estimating the pressure and temperature distribution and their transient behaviour. The intention was the verification of the principles for choosing the appropriate number and locations of the switches.

FISSION PRODUCT RELEASE, TRANSPORT AND REMOVAL.
RADIATION EXPOSURE IN THE ENVIRONMENT

The behaviour of certain fission products such as xenon, elemental iodine and methyl iodide within a BWR PS-containment for LOCA conditions was studied as a part of the first series of the full-scale experiments at the Marviken plant [27]. Investigations of high temperature release processes from pellets have been performed as have studies of iodine removal by sprays. Limited theoretical studies of fission product release and behaviour under severe accidental conditions (core melt) are under way.

For normal operation the behaviour of fission and activation products is analysed within several projects covering fission product distribution within the pellets and fuel elements, release from fuel, transport in various reactor systems and removal in filters. The releases to the coolant are analysed and compared with available models. The major normal pathways of gaseous radionuclides and their characteristics in BWR and PWR systems have been mapped. Poisoning of charcoal filters and methods for testing efficiencies are studied.

Computer programs have been developed at Studsvik and extensively used for calculation of radiation doses as a result of releases to the environment, during normal operation as well as in accident situations. Planned improvements in the code package concern methods for treatment of variable weather ander the release period, ground and sea contamination, wet deposition and infiltration in soil.

The different codes for the local, regional and global atmospheric dispersion have been checked in the overlapping zones and linked together with a file describing the actual population distribution, thus facilitating the calculation of collective dose commitments.

The models for hydrological dispersion and exposure are in a stage of intense development. The BIOPATH program treating transport in local, regional and global ecosystems, has been under development and application especially for waste storage analysis. Certain transfer coefficients of importance are studied in special field experiments.

ACCIDENT ANALYSIS AND RISK ASSESSMENT

The methodology for risk assessment used in WASH-1400 has been extensively studied and implemented for reactor safety work in Sweden. As the Swedish BWR systems differ from that of the model plant design of the Rasmussen report, fairly thorough analyses have been made of the Barsebäck 2 plant (external main circulation pumps) and the Forsmark 3 plant (internal circulation pumps). In fact, two independent studies were made of the Barsebäck plant, one by Studsvik Energiteknik AB and one by the American consultant firm MHB Technical Associates. The two studies showed no large difference in the assessment of the core melt probability despite MHB's critical attitude to nuclear energy. The severe consequences calculated by the American firm were debated. The consequences for various cases calculated in the Studsvik study were compatible with those of the Rasmussen report. A comparison of the two studies was presented at the Brussels meeting on Nuclear Power Safety, 1978 [28].

The WAM computer code used in WASH-1400 and further developed by EPRI has been installed at the Studsvik computer and is being used for detailed safety analysis of complex systems. In general, development of the Rasmussen methodology is given high priority in Sweden particularly in the light of the Lewis Committee findings. The matter of common cause failures will be given special attention. In addition, work is being done to systematize information on component failures in order to obtain a better data base for risk assessment.

It is clear that the knowledge of the behaviour of the reactor system during a core melt is incomplete. In order to furnish a better basis for the accident analysis and emergency planning core melt studies have been done in Sweden mostly based on work done at Sandia in the US and at Karlsruhe in the FRG. There are at present indications that Sweden should participate in experimental work related to core melt behaviour.

Another area to which much attention is paid nowadays is the man-machine problem and human performance. One special aspect of this is how to design the control room for a nuclear plant to give maximum reliable operation. A joint Nordic project was started in 1976 - supported by the Nordic Council of Ministers and by national authorities - to perform a comprehensive control room design study. The project, which will continue till the end of 1980, is divided into four parts

- Operator task analysis at two nuclear power plants
- Control room design with regard to human decision making and information processing
- Methods for evaluation and influencing human reliability in large safety related systems
- Methods for operator training - basic education and retraining.

In several projects the operators' working conditions have been studied. The ergonomics of the control room are being studied by the consultant firm LUTAB in cooperation with the personnel involved. One task is to try to eliminate special accident-prone situations and improve the interface between man and machine. Several part-reports are ready and a summary report will be published in the Autumn of 1979.

CONCLUDING REMARKS

When this is written (summer 1979) the future of nuclear energy in Sweden is uncertain. A referendum in spring 1980 will be decisive for whether new reactors will be allowed to start operating or not. A no to nuclear power in Sweden would mean that the six reactors now in operation could be in service for a maximum period of 10 years. Irrespective of the outcome of the referendum, however, the near term safety research effort would be a natural prolongation of the ongoing program. The following concluding remarks can be made about that program

- A large part of the program aims at defining the degree of conservatism in the licensing procedure
- Basic safety related issues are attacked as well as those of direct interest for the regulatory bodies
- BWR issues are still dominating but PWR problems are gaining in importance
- Well defined separate effect tests are usually preferred to large integral experiments. This is due partly to the large costs involved in performing the latter, partly to the difficulties in drawing conclusions of general validity from integral experiments
- Some safety issues require large internationally funded projects. Sweden will continue to promote cooperative projects that can be performed in Sweden and to take part in selected projects in other countries to the extent that its resources allow
- The research program is set upp to deal with issues at all safety levels with some preference for projects related to accident prevention measures
- The research program is partly intended to provide the basis for backfitting considerations, the objective being to increase the safety of older plants to the standard of more modern ones.

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Table 2

Times for the onset of dryout and flashing.
Comparison between experimental and calculated values

Case	Event	Time, seconds	
		Experiment	RELAP calc.
Bottom break, diam. 7.3 mm (Top break, diam. 15 mm)	Dryout	-	-
	Gap flashing	5	5-6.5
	Bottom flashing	-	-
Pump trip	Dryout	2.3-2.4	2-2.4
	Gap flashing	4.7	4.2
	Bottom flashing	5.3	8.1
Top break, diam. 24 mm	Dryout		4.5
	Gap flashing	2.5	2.8
	Bottom flashing	3.7	5.5
Top break, diam. 29 mm	Dryout	2.9	3.0
	Gap flashing	1.3	1.7
	Bottom flashing	3.0	3.6

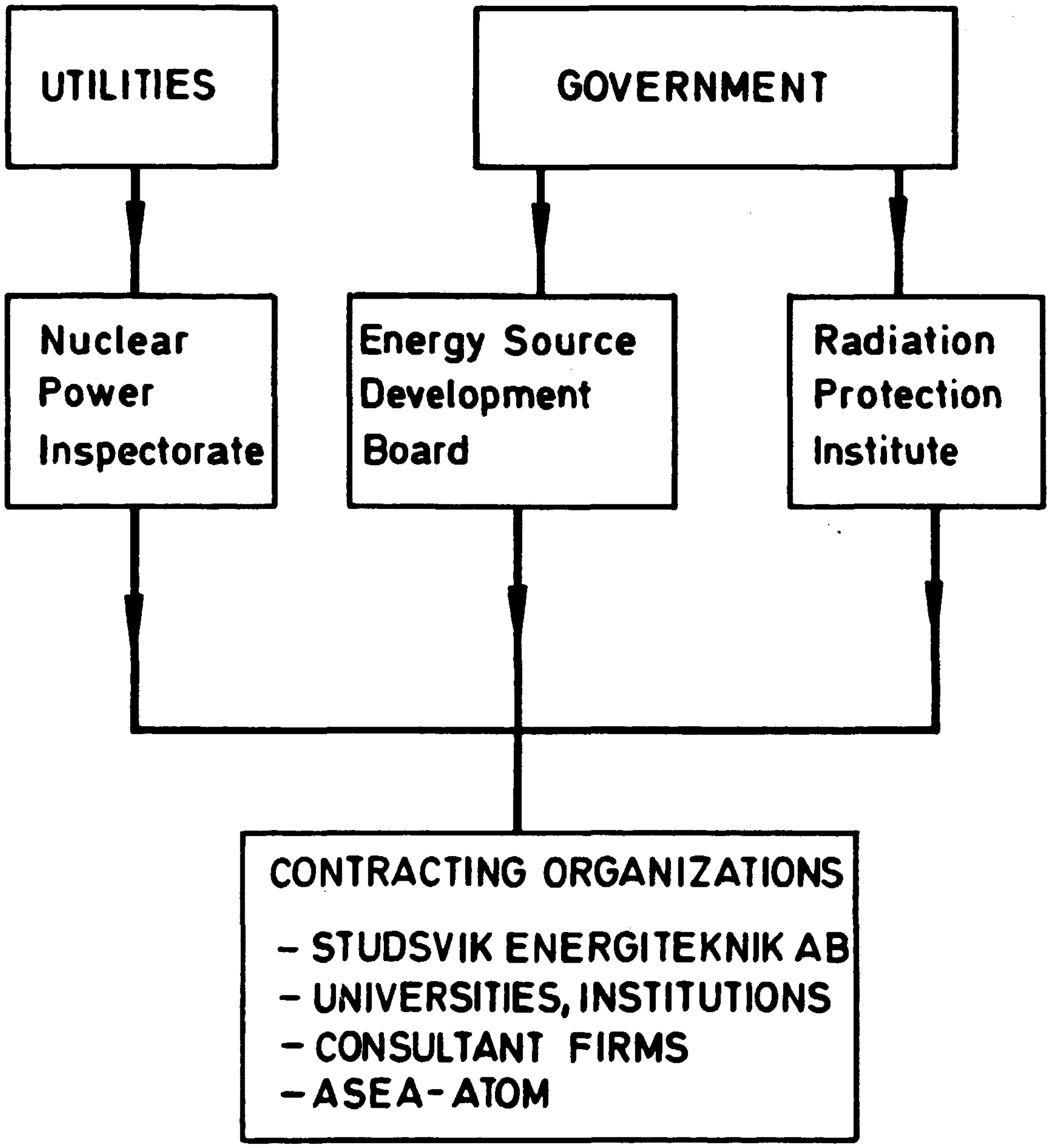

Fig 1: Flow scheme for the safety research funding in Sweden

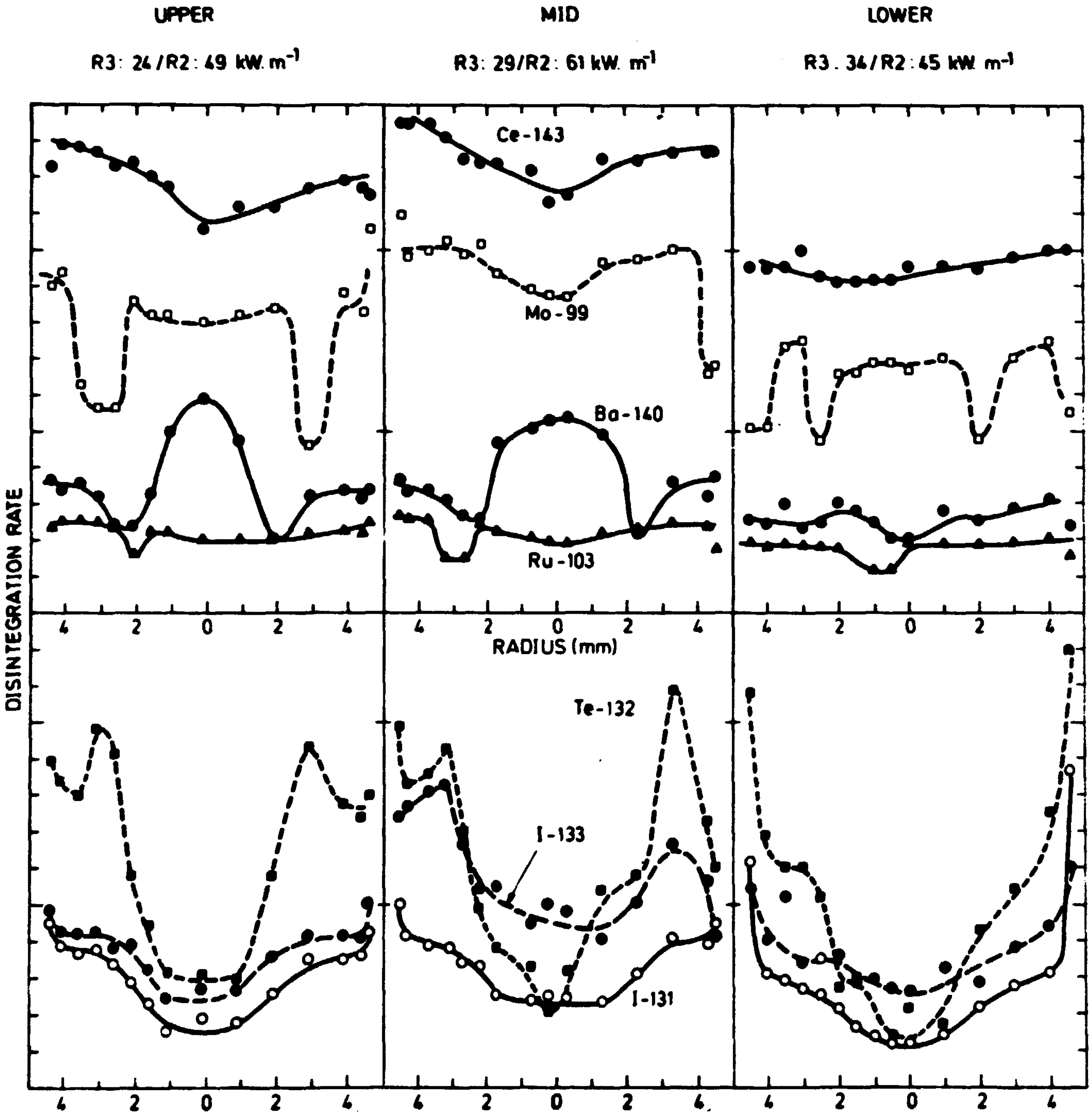

Fig 2: Radial fission product distribution [1]

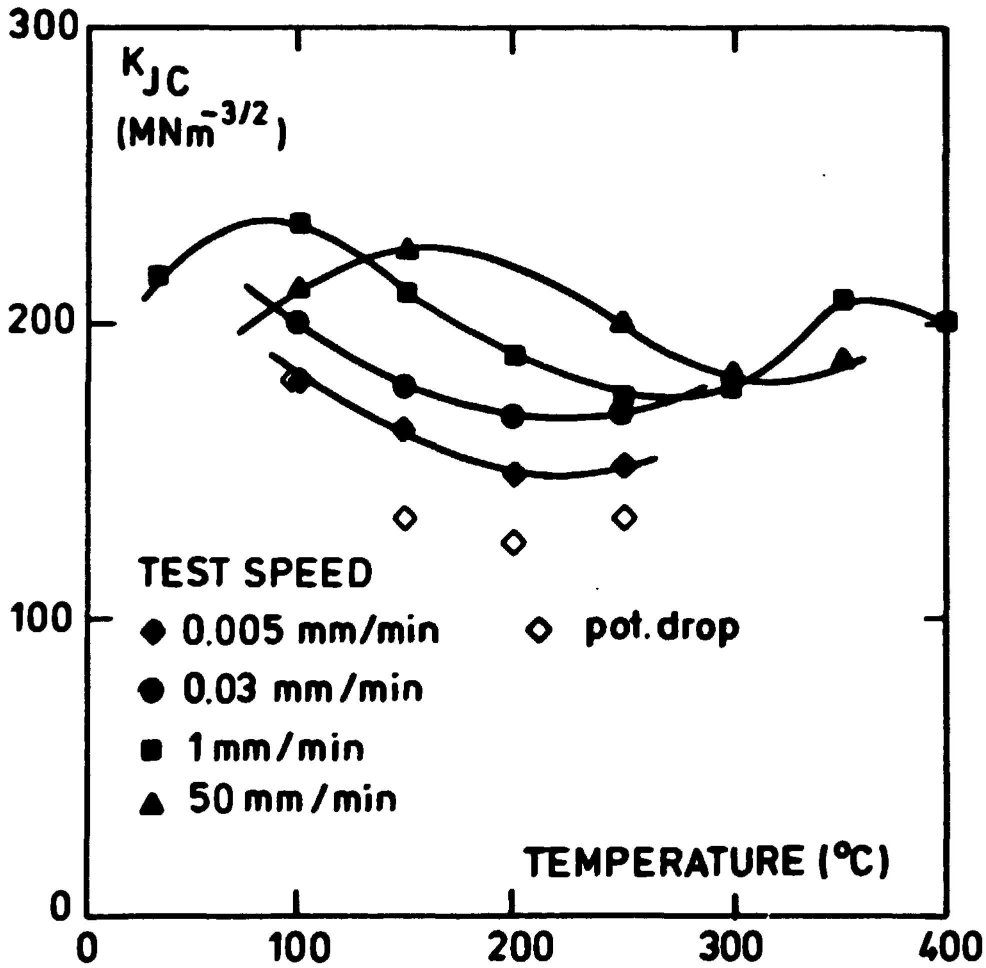

Fig 3: K_{JC} versus temperature for A5333 B steel at different strain rates (after Östensson [4])

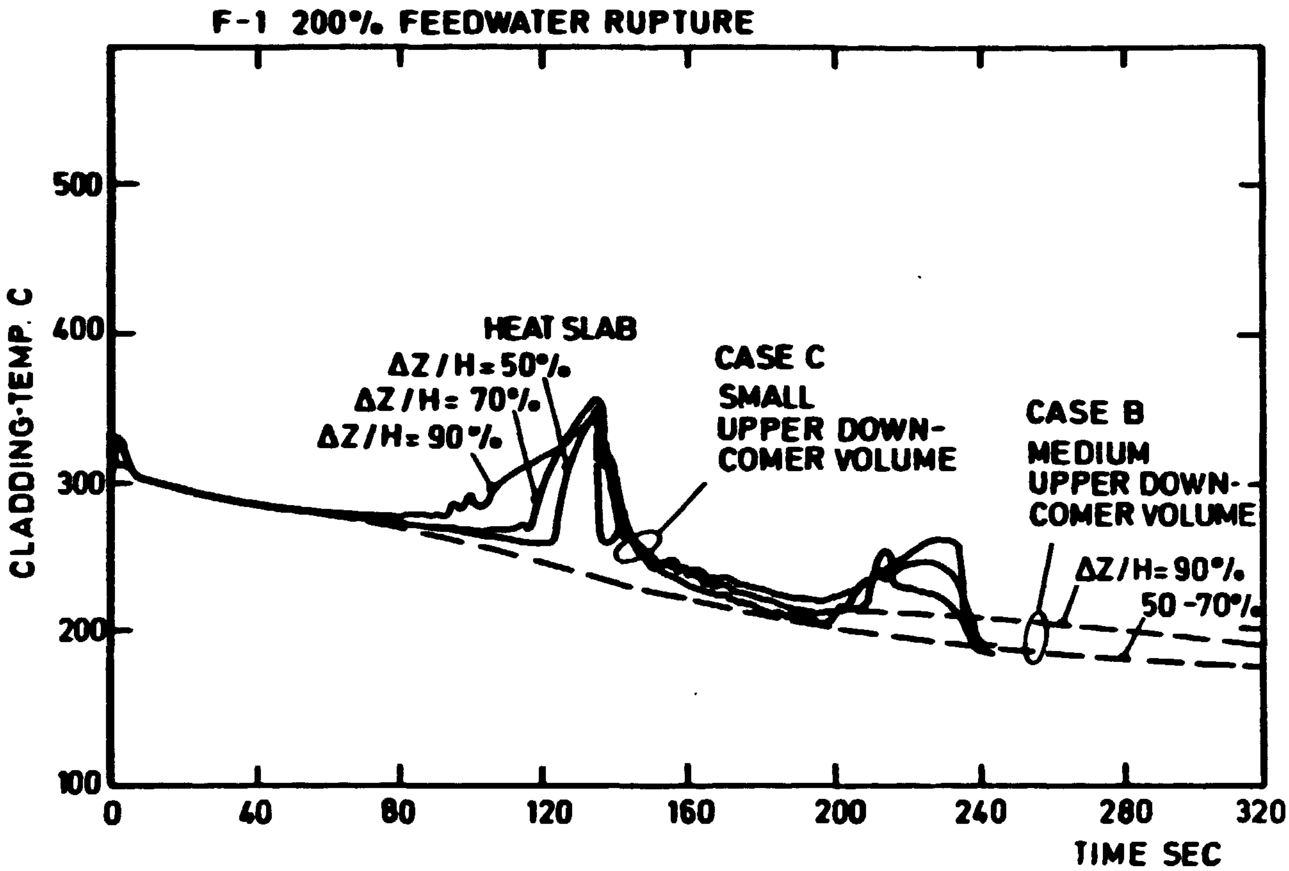

Fig 4: Maximum cladding temperatures for different downcomer nodalization at three levels of the channel. RELAP4 calculations for the ASEA-ATOM Forsmark 1 reactor [10]

GÖTA LOOP

	SPRAY COOLING		
No. of heated rods	64	63	60
Exp. year	1973-77	1976-77	1977

COUNTER CURRENT FLOW LIMITATION
64
not heated
1976 - 77

REFLOOD
63
1977 - 79

FIX LOOP

FIX I

TRANSIENT DRY OUT EXP

DETERMINATION OF TIME TO DRY OUT

1974 - 77

FIX II

TRANSIENT DRY OUT AND POST DRY OUT EXP

DETERMINATION OF HEAT TRANSFER DURING BLOW DOWN

1977 -

STUDSVIK/K2-79/193
1979-09-25

Fig. 5 LOCA RELATED LOOP EXPERIMENTS AT STUDSVIK.

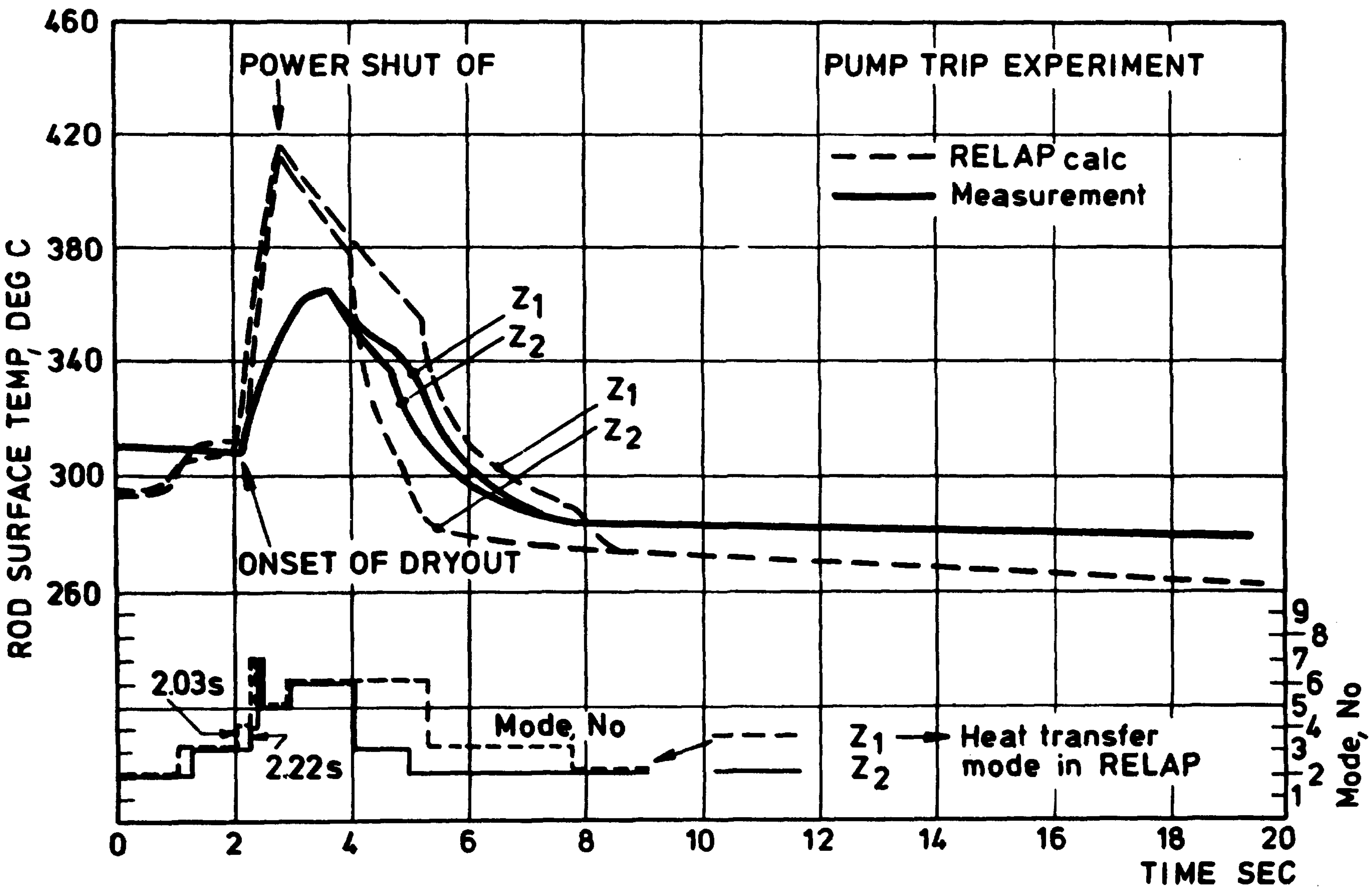

Fig. 6 ROD SURFACE TEMPERATURE AT TWO LEVELS

STUDSVIK ENERGITEKNIK AB

STUDSVIK/K2-79/193
1979-09-25

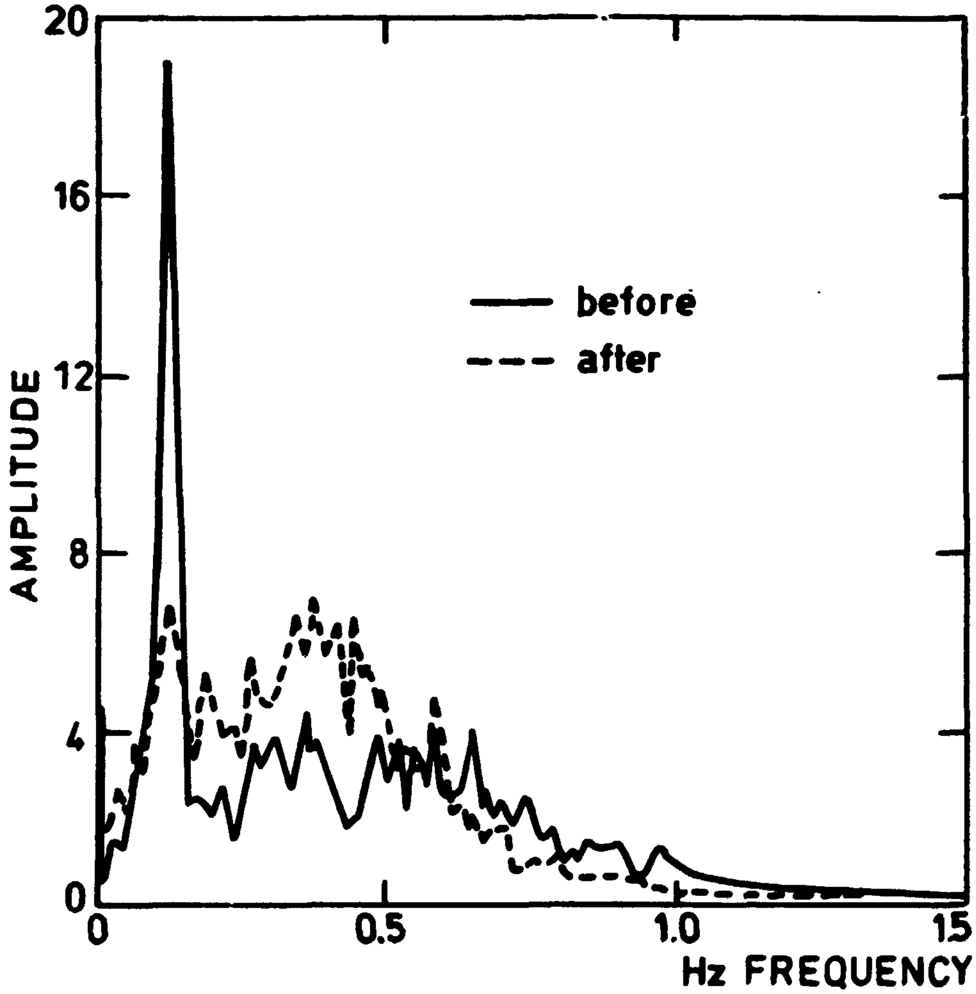

Fig 7: Normalized auto power spectral density for in core detectors (APRM) before and after control parameter adjustment [13]

Hole cuttings:

1 Floor beneath vessel

2 Drywell floor

3 Transport channel floor

4 Transport channel wall

5 Front wall

6 Cables passages

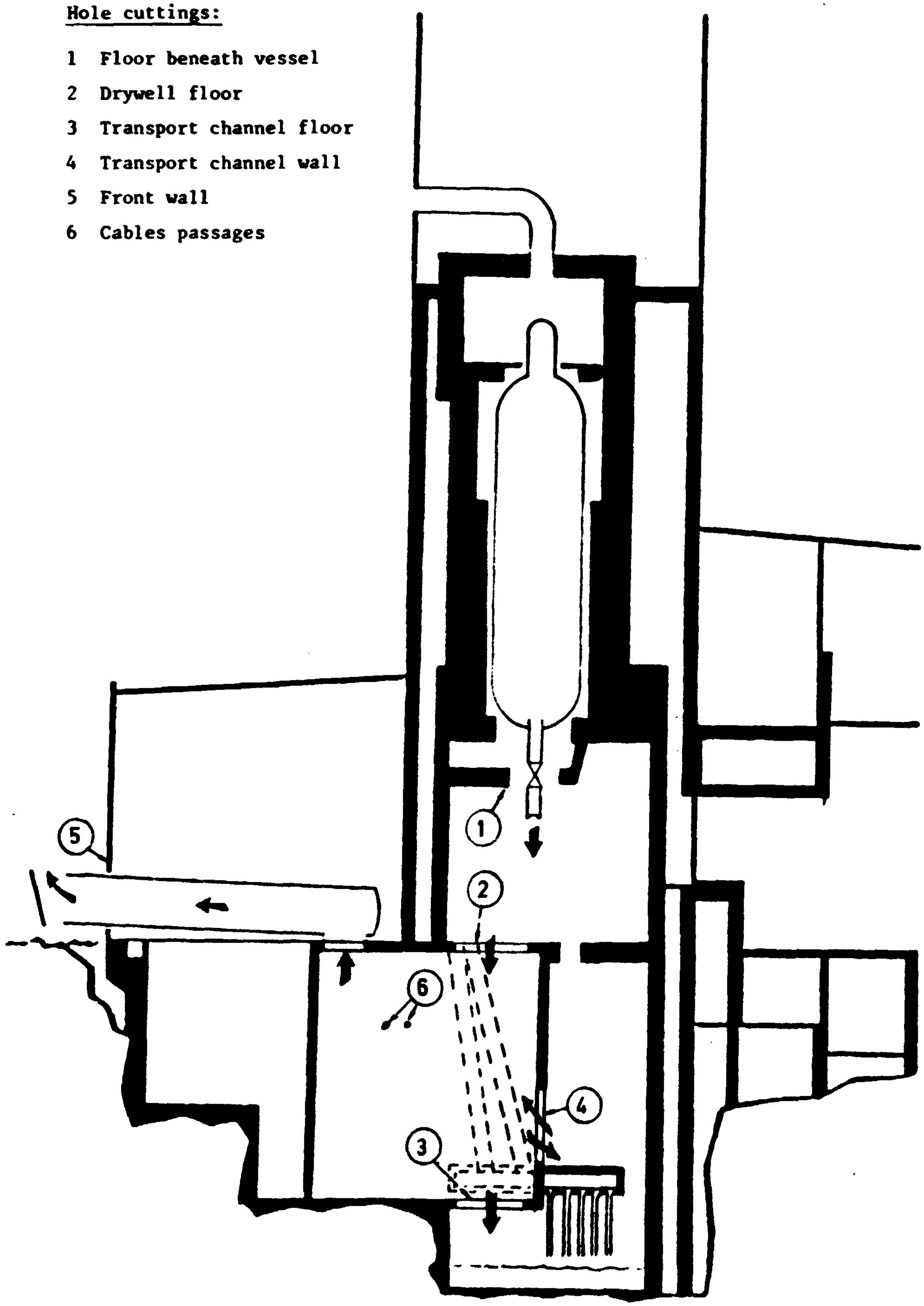

Fig 8: Marviken Critical Flow Test assembly and exhaust system

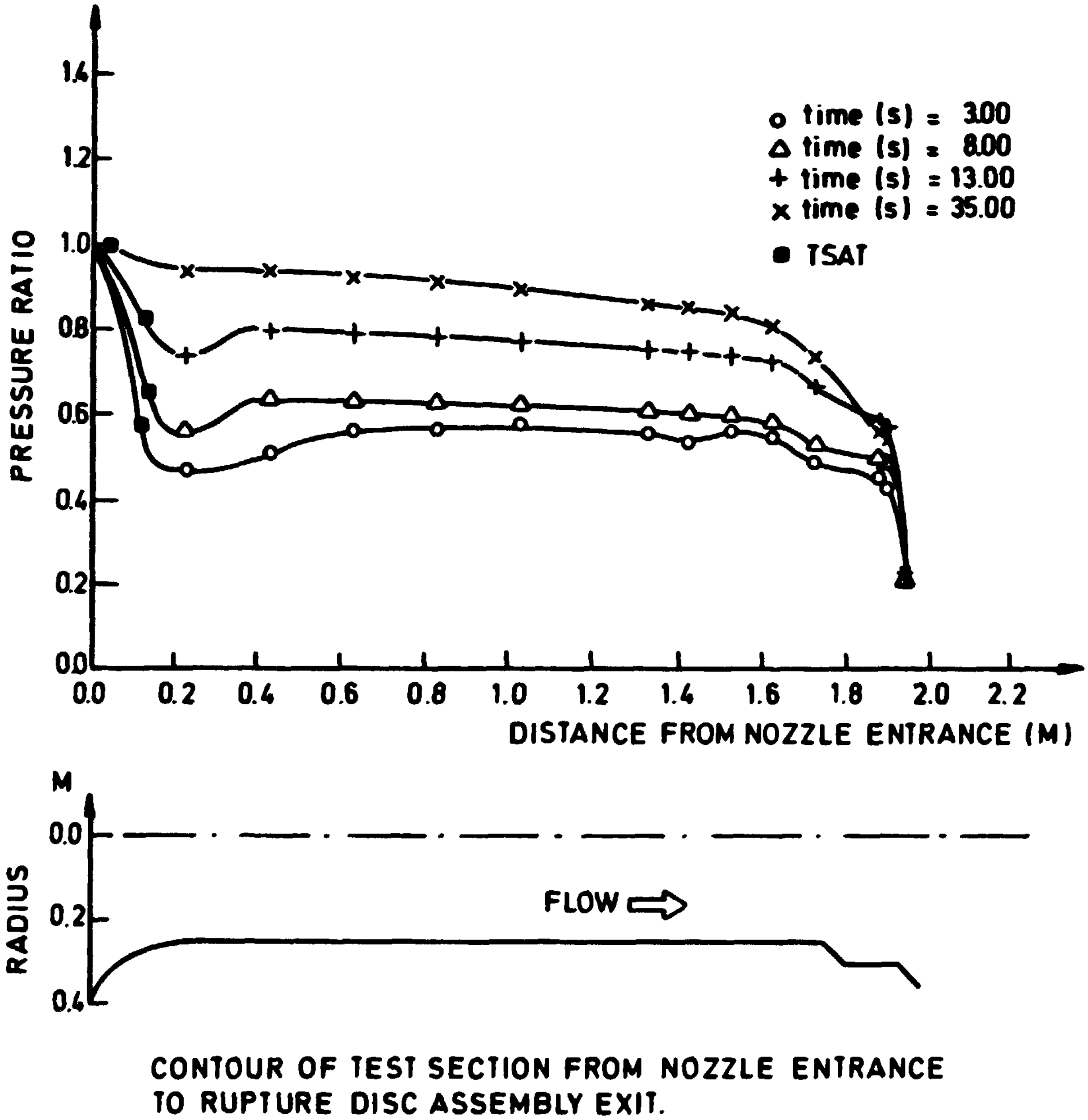

Fig 9: Nozzle pressure profiles at various times during blowdown [24]

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