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PERIODIC INSPECTION OF PRESSURE VESSELS

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Periodic Inspection of Pressure Vessels

This conference was sponsored by the Pressure Vessels section of the Applied Mechanics Group and was held in London. 354 delegates registered to attend. The planning panel consisted of Dr R. W. Nichols (chairman), Mr R. L. J. Hayden, Mr G. P. Smedley and Mr B. Watkins.

Six sessions included: (1) 'Requirements for inspection' (papers C42, C56, C48, C26); chairman, Professor Y. Ando. (2) 'Inspection practice in differing applications' (papers C29, C59, C53, C52, C51); chairman, L. J. Chockie. (3) 'Inspection practice in nuclear plant' (papers C55, C41, C44, C31); chairman, R. D. Wylie. (4) 'Equipment for reactor pressure vessel inspection' (papers C57, C25, C46, C43, C47, C27); chairmen, Dr T. Broome and Dr R. W. Nichols. (5) 'Future developments—acoustic emission and other techniques' (papers C28, C30, C40, C45, C49, C58); chairmen, Dr S. H. Bush and B. Watkins. (6) 'Other developments and general aspects' (papers C50, C54, C60); chairman Dr R. W. Nichols.

A cocktail party was held in the evening of Wednesday 10th May in the ballroom of the St Ermin's Hotel, Caxton St, London, S.W.1.

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C25/72 THE NUCLEAR REACTOR VESSEL INSPECTION TOOL USING AN ULTRASONIC METHOD

T. YAMAGUCHI* Y. FUKUSHIMA* S. KIHARA* T. ENDO* Y. YOSHIDA*

This paper describes the outline of an in-service inspection tool using an ultrasonic method for the P.W.R. type of nuclear pressure vessels, which is now under development. This equipment consists of the mechanical components of the tool, the control panel to drive it, and data acquisition apparatus. The tool is designed to be adjustable to reactor vessels of various sizes. Mitsubishi Heavy Industries Ltd (M.H.I.) has now completed the fabrication of most parts of the tool, and are planning to carry out a mock-up test in the near future to check the operational function of the tool and the detectability of the defects.

INTRODUCTION

IN THE nuclear industry, in order to ensure the protection of the public from any foreseeable accidents, the safety of all components is absolutely essential. In a nuclear power plant especially, periodic inspection is required even after operation commences, to secure the safety of the vessel and to prevent, in advance, the occurrence of any accident.

In operation, a nuclear reactor vessel becomes radioactive after being irradiated by neutrons. Therefore, depending on their positions, some parts of the vessel are inaccessible. For this reason, the inspection of such parts must be remotely performed.

To meet the above requirements, M.H.I. began several years ago to research and develop this inspection method. As a result, development is almost completed of an inspection method and its remote control tool. Furthermore, it was scheduled recently to perform a mock-up test utilizing a model vessel.

OUTLINE OF A NUCLEAR REACTOR VESSEL

The outline of the major dimensions and welded joints of the typical P.W.R.-type reactor vessels which are now under design, or have already been constructed, is shown in Fig. 25.1. The materials used are nickel–chromium–molybdenum forged steel for the upper shell, shell flange, closure head flange, outlet and inlet nozzles, and safety injection nozzle; and manganese–molybdenum–nickel plate for the middle shell, lower shell, closure head, and lower head. In addition, forged austenitic stainless steel (A.S.M.E. SA-182 F316, F304) is used at the safe-end of each nozzle. On the inner surface of the vessel, the overlay cladding equivalent of austenitic stainless steel type 304 is performed and is finished by grinding. The total thickness is approximately 5·5 mm.

In order to secure fully the safety of the nuclear reactor vessel in its fabricating process, extraordinarily strict non-destructive inspection of various sorts are performed. The kind of inspection method used for each component in fabrication is shown in Fig. 25.2, each welded joint being inspected by various methods, such as ultrasonic, radiography, magnetic particle, and liquid penetrant, after the welding is completed or after the overlay welded surface is finished.

GENERAL SYSTEM REQUIREMENT

An inspection method which is applicable to the in-service inspection of a nuclear reactor vessel must satisfy the following conditions (1)†.

(1) It should be capable of non-destructive, volumetric inspection, which can be carried out from one side of a vessel.

(2) It should be capable of automatic and remotely controlled measurement.

(3) It must have appropriate sensitivity resolution and good repeatability of data.

(4) It should be insensitive to irradiation from radioactive materials.

(5) It must have an electrically processible data output.

(6) It should be capable of rapid operation and analysis.

The ultrasonic inspection method is the only one that satisfies all these conditions.

The ultrasonic inspection methods adopted were normal beam, angle beam, and pitch–catch methods, which are shown in Table 25.1. The search unit is composed of lead–zirconate–titanate transducers, which are 20 mm in diameter with a frequency of 1 MHz (for angle beam and pitch–catch methods) and 2·25 MHz (for normal beam method). The effect of radioactive irradiation, for lead–zirconate–titanate transducers, as shown in Fig. 25.3, is hardly observed regarding the loss of sensitivity within the irradiation dosage $<1{\cdot}7\times10^{7}$ r.

Inspection is performed from within a vessel. The inner

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† *The reference is given in Appendix 25.1.*

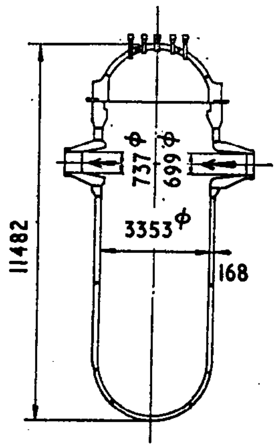
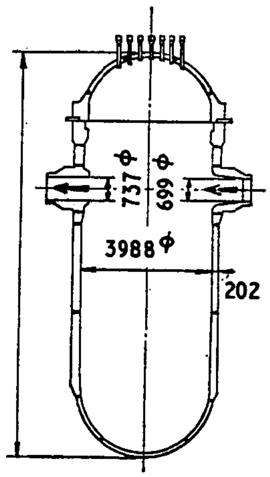
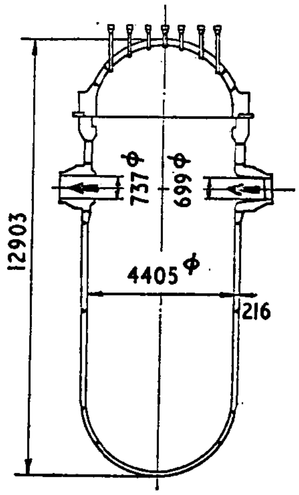
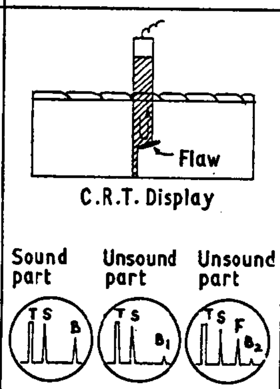
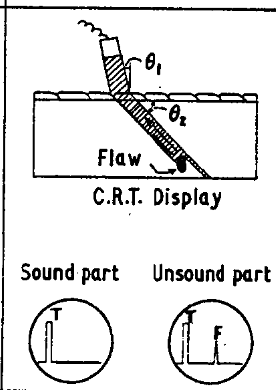
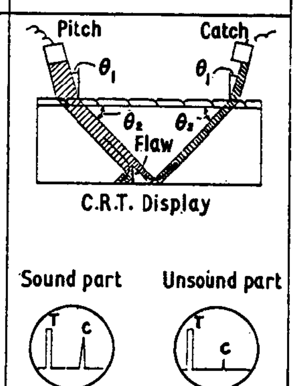
Plant	Kansai Mihama N°2	Kansai Takahama N°2	Kansai Ohi N°1
Number of Loop	Loops 2	Loops 3	Loops 4
Electrical output	500 MWe	826 MWe	1170 MWe
Dimension			

Fig. 25.1. Principal dimensions of typical reactor vessels

surface of a vessel is not entirely smooth. In fact, there exists a valley construction on the bead overlapped section which is a hazard relative to the penetration of an ultrasonic beam. Therefore an immersion method was adopted. This is a method in which an ultrasonic beam is transmitted by keeping a certain distance between a transducer and a surface of the vessel wall to be inspected. One-third of the thickness of the wall to be inspected is usually taken as a water path distance.

Although the cladding surface is finished by grinding, there is still comparative roughness on the bead-overlapped part, i.e. the valley part, as well as a metallurgical change. These items result in scatter of the transmitted energy and, therefore, give an effect on echo height. Consequently, the main endeavour of the M.H.I. researchers' study was to improve the accuracy of inspection by minimizing the effect of cladding relative to the volumetric inspection of a reactor vessel.

Table 25.1. Flaw detecting method

Method	Normal Beam	Angle Beam	Pitch-Catch
Principle			
Remarks	T: Transmit echo S: Surface echo F: Flaw echo B: Back echo	θ_1: 19° (Incident angle in water)	θ_2: 45° (Refracted angle in steel) C: (Transmission echo).

Areas to be inspected by this tool are to include those where volumetric inspections are required by the A.S.M.E. Code Section XI. Areas to be inspected and the ultrasonic inspection method which is to be adopted for each area of the vessel are shown in Table 25.2 and Fig. 25.4. In order to promote precision of inspection, normal beam and pitch-catch methods are simultaneously used on the circumferential and longitudinal welds of the vessel (angle beam method is also applicable).

AUTOMATIC REMOTE-CONTROL TOOL

This tool is composed of a mechanical operation component which works underwater in a nuclear reactor vessel, and a control panel to operate this component; this is shown in Fig. 25.5. The construction of the mechanical operation component enables it to be adjusted to reactor vessels of various sizes, and it is remotely operated with high precision and reliability on the areas to be inspected.

The control panel is the operation unit by which the mechanical component is driven, and it is capable of manual and automatic operation. Adequate consideration is given regarding water pressure strength, resistance to radioactive irradiation, and corrosion resistance for the main component. Simultaneously, a special design to maintain the rigidity of the tool has been devised in order to promote driving precision. This tool is also designed to permit transportation in a container, and to give easy disassembly and assembly.

Configuration and function

The tool is composed of the following elements.

Turning table equipment

It consists of rail, turning frame, and driving mechanism. The rail is laid on the upper surface of a pressure-vessel flange. The turning frame with driving mechanism is placed on the rail, and it is rotated on a rail driven by three rollers.

Elevation mechanism

It consists of the elevator, guide frame, elevation driving mechanism, and guide frame support mechanism. The elevator runs vertically along the guide frame whose upper edge is fixed on the turning table.

Manipulator

There are five series of manipulator driven by electric and pneumatic power. Each manipulator is equipped with ultrasonic transducers of various kinds, as shown in Fig. 25.6. The manipulator series 1 is provided on the turning table; series 2, 3, and 4 on the elevator; series 5 at the lower edge of the guide frame. Each of them is operated in accordance with the particular areas to be inspected.

Automatic remote-control equipment

It consists of a control panel and a position indication panel. The control panel is placed on an operation deck. By command from this panel, transducers on a manipulator are moved circumferentially, vertically, and radially to scan the surfaces to be inspected automatically.

Flaw detection

It is important that this tool be able to detect the position of a flaw as well as its presence. By making an initial setting

Table 25.2. Detailed descriptions on inspection areas of the reactor vessel

Areas subject to inspection	Actually inspectable areas		Ultrasonic testing method
	Length (L), mm	Width (B), mm	
(A) Flange ligaments between stud holes	Entire circle	Flange width	Normal beam method (No. 1)
(B) Vessel to flange welds	Entire circle (except stud hole)	Wall thickness	Normal beam method (No. 2)
(C) Shell circumferential welds	Entire circle	Weld metal width+2x (wall thickness)	Normal beam method and pitch-catch method (possible to use angle beam method)
(D) Shell longitudinal welds	Entire length of each weld	Weld metal width+2x (wall thickness)	Normal beam method and pitch-catch method (possible to use angle beam method)
(E) Outlet and inlet nozzle attachment welds to vessel	Entire circle	Upper shell wall thickness (except a part of inlet nozzle)	Normal beam method (No. 1)
(F) Pipe to safe-end welds	Entire circle	Weld metal width+2x (wall thickness)	Normal beam method and angle beam method (No. 2)
(G) Outlet and inlet nozzle to safe-end welds	Entire circle	Weld metal width+2x (wall thickness)	Normal beam method and angle beam method (No. 2)
(H) Inner radius section of the inlet nozzle-to-vessel juncture	Entire circle	Inner radius section	Normal beam method
(I) Shell to lower head welds	Entire circle (except internal support attachment)	Weld metal width+2x (wall thickness)	Normal beam method and angle beam method
(J) Meridional welds in lower vessel head	10% of each longitudinal weld (upper part)	Weld metal width+2x (wall thickness)	Normal beam method

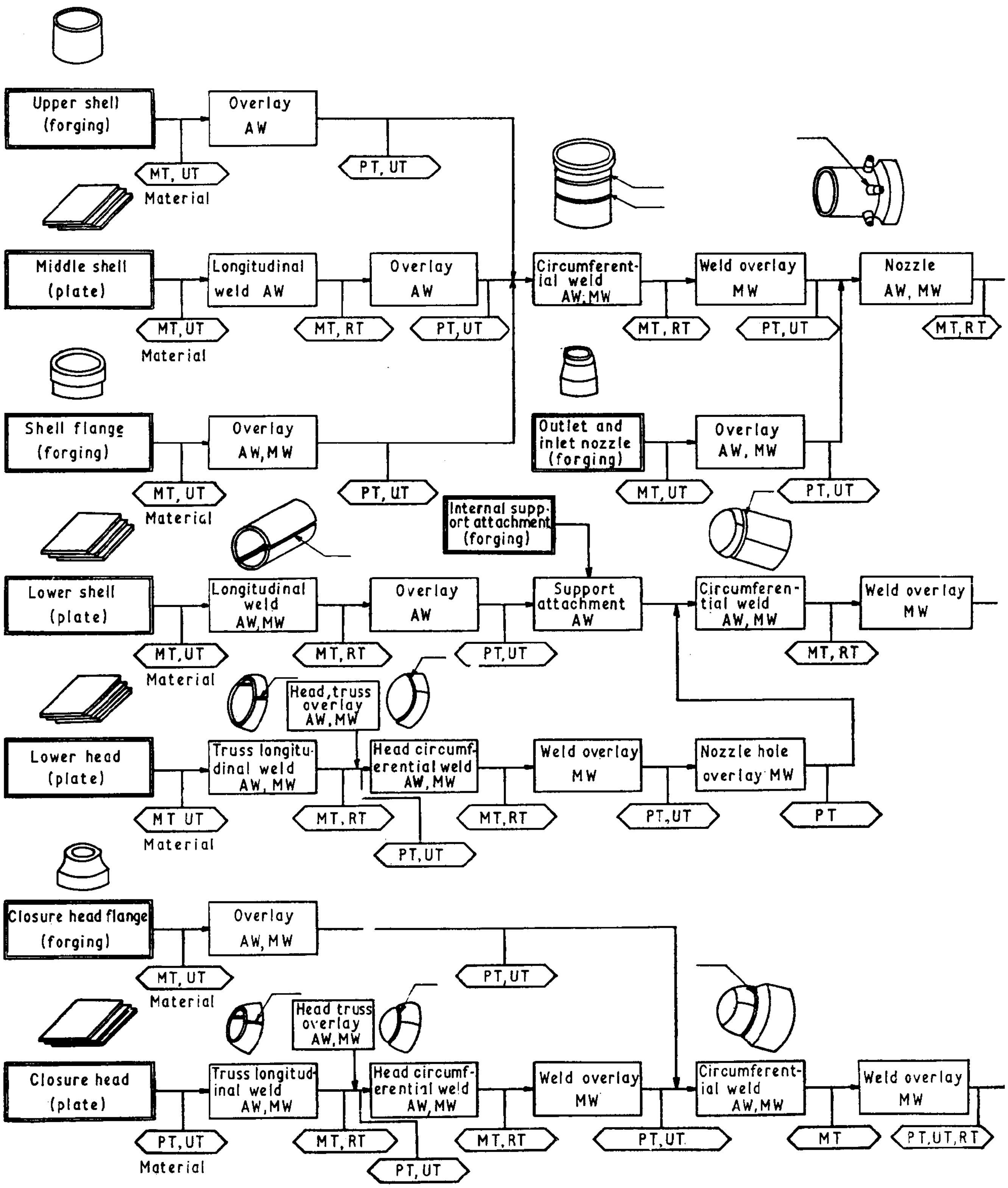
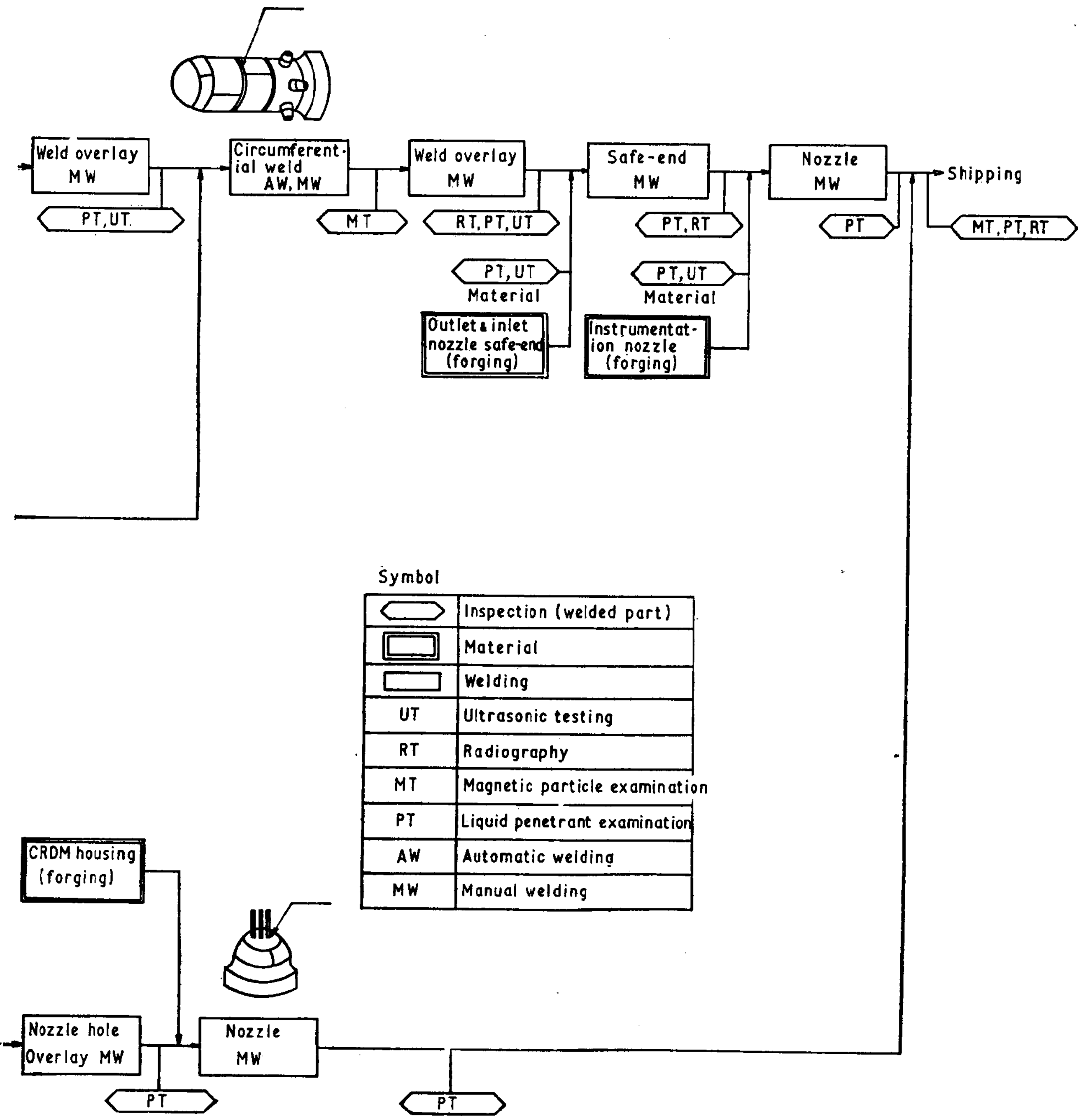
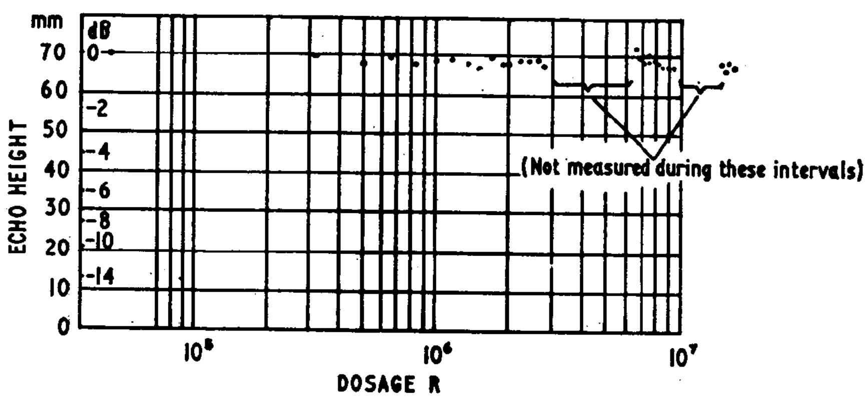


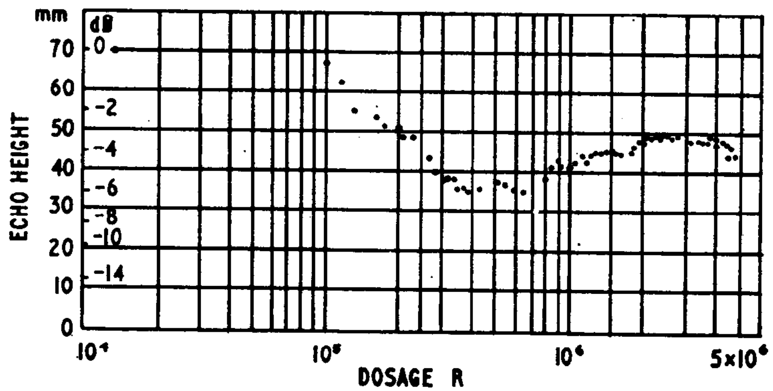
Fig. 25.2. Outline of process for non-destructive



examination during fabrication of reactor vessel



Lead-Zirconate-Titanate transducer (Nº3 3SIL20D)
(Dose rate $5\cdot94\times10^{3}$ R/min, continuous flaw detecting during gamma irradiation)



Lithium-Sulphate transducer (Nº1 57A2688)
(Dose rate $5\cdot94\times10^{3}$ R/min, continuous flaw detecting during gamma irradiation)

Fig. 25.3. Effect of gamma radiation on the sensitivity of search unit

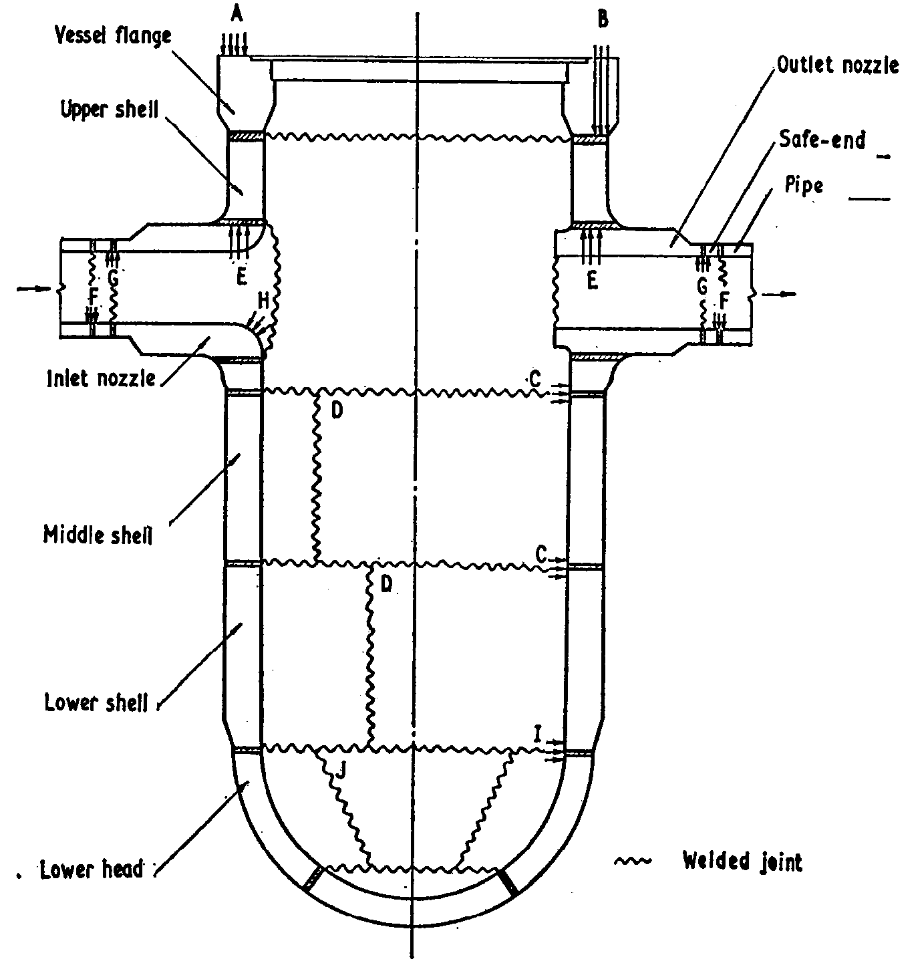


Fig. 25.4. In-service inspection areas

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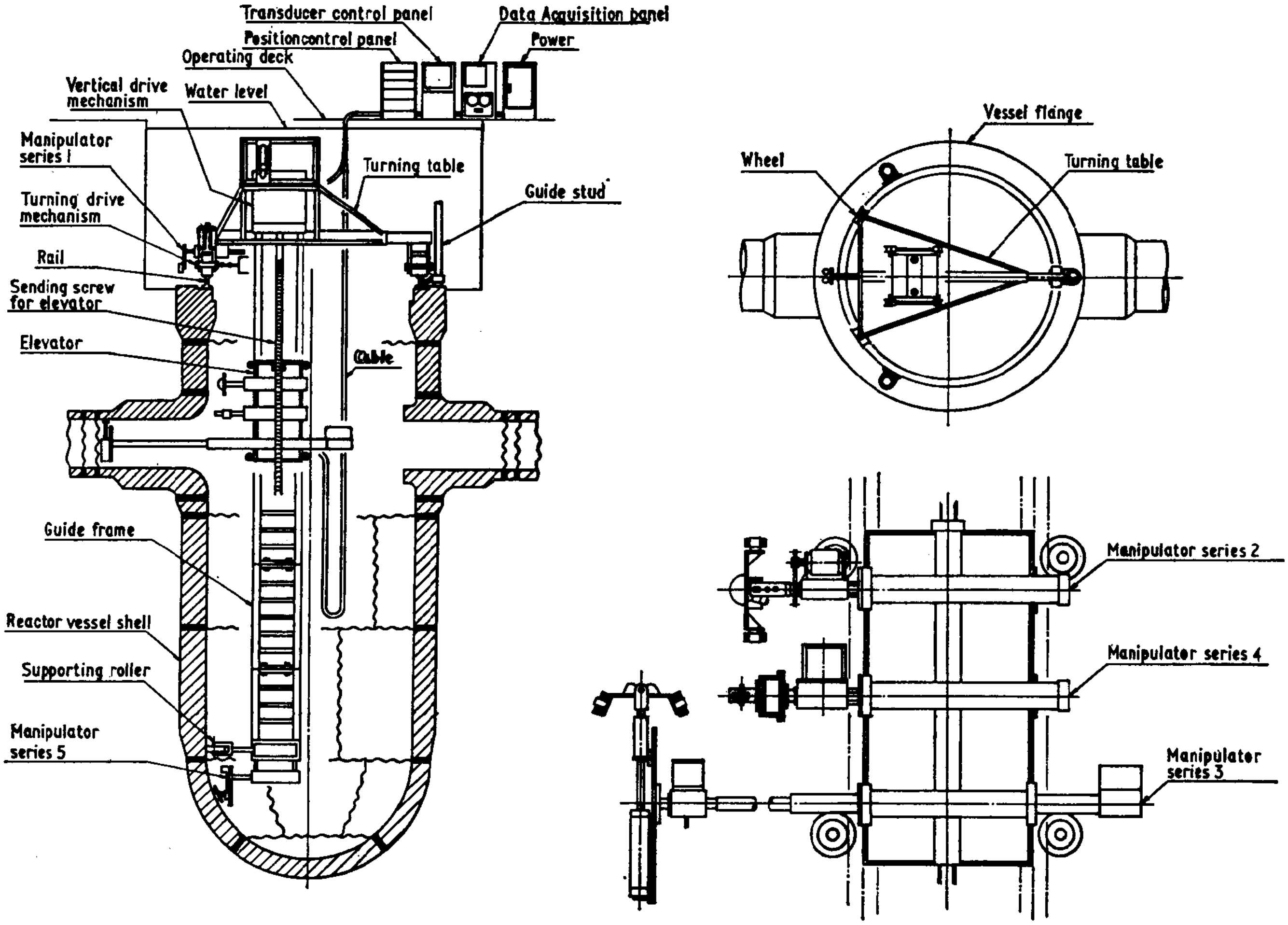

Fig. 25.5. Schematic of inspection tool

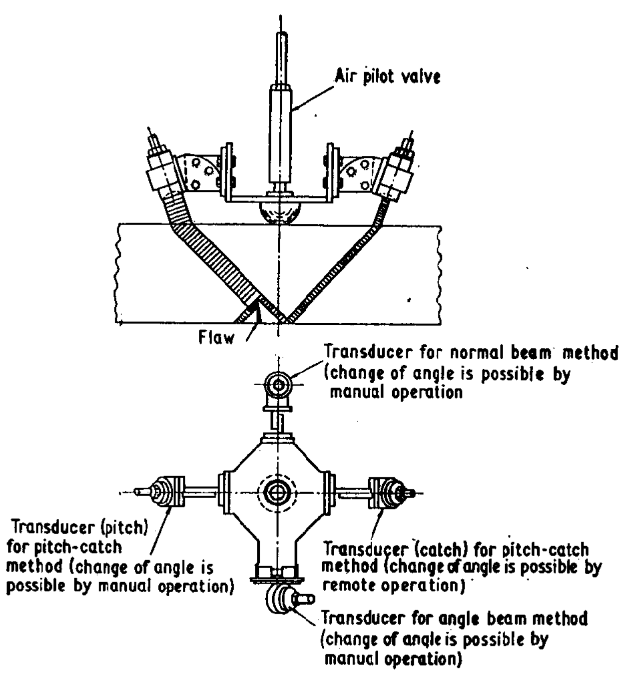

Fig. 25.6. Inspection assembly

position of the tool on a vessel at a reference point, the position of the inspection area is continuously monitored at the control panel. This position-indication signal is simultaneously recorded as an address information in a data acquisition unit.

Dimensions of tool

Because the inner diameter and depth of a reactor vessel differ in accordance with capacity, the method of installing this tool on a vessel changes in each case. This tool is applicable within the following range by adjusting each component element in its assembly process.

Inner diameter of vessel	3300–4420 mm
Maximum depth (from vessel flange surface to shell to bottom head weld)	8400 mm
Nozzle inner diameter	690–900 mm
Nozzle depth (from vessel inner surface)	1130 mm

In addition, the major dimensions and weight of this tool are:

Length × width × height	4430 × 1760 × 10 800 mm
Weight	About 3·5 ton

Scanning precision and speed

The scanning precision and speed of the search units furnished on this tool are:

Angle of circumferential rotation	±0·1°

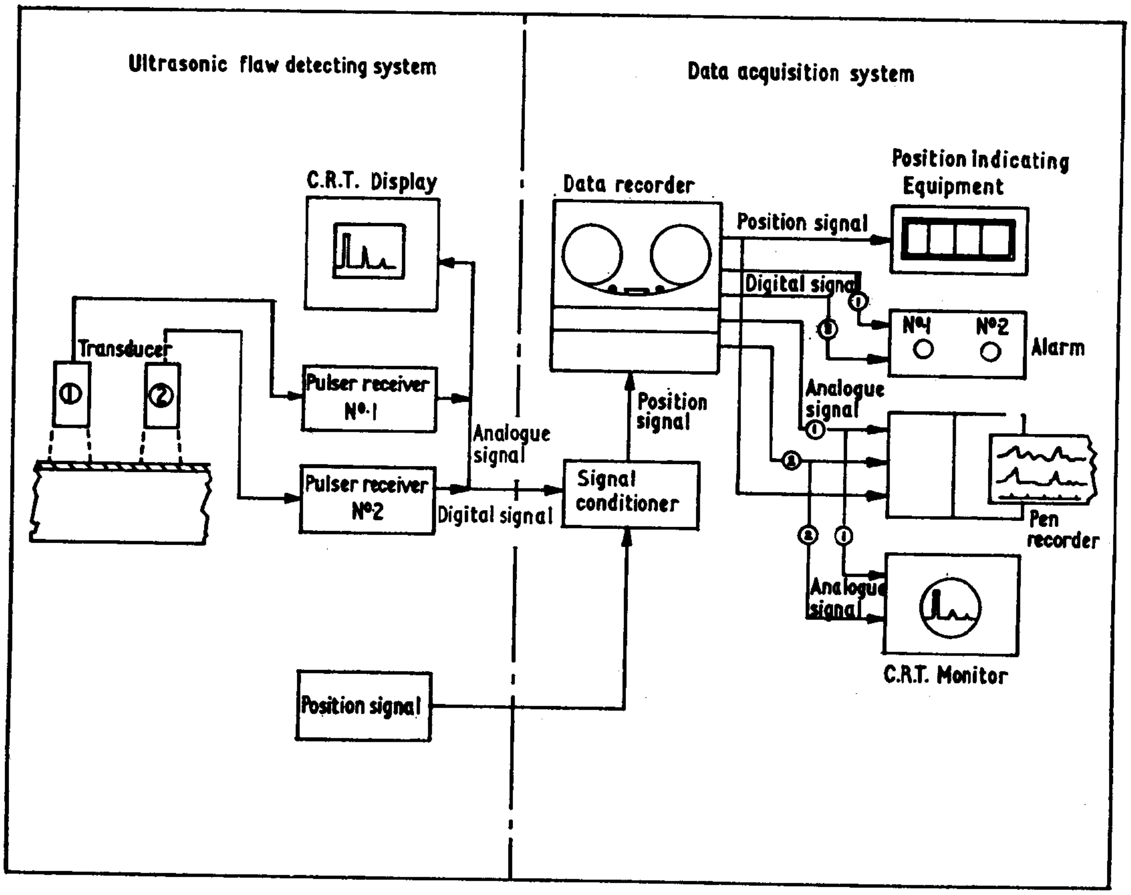

Fig. 25.7. Block diagram of the data acquisition apparatus

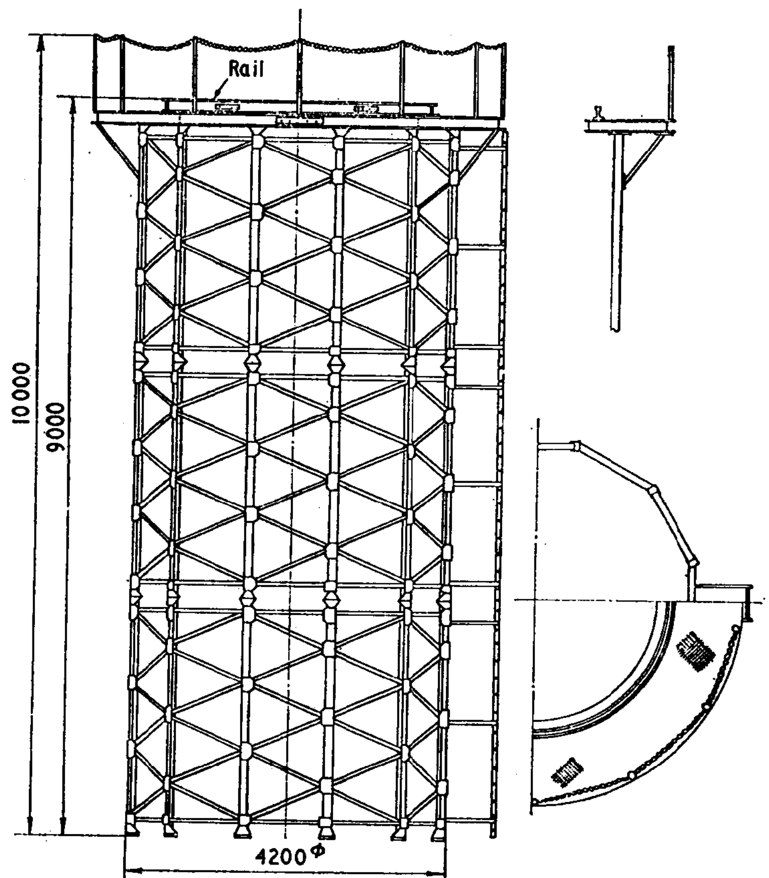

Fig. 25.8. Assembly stand

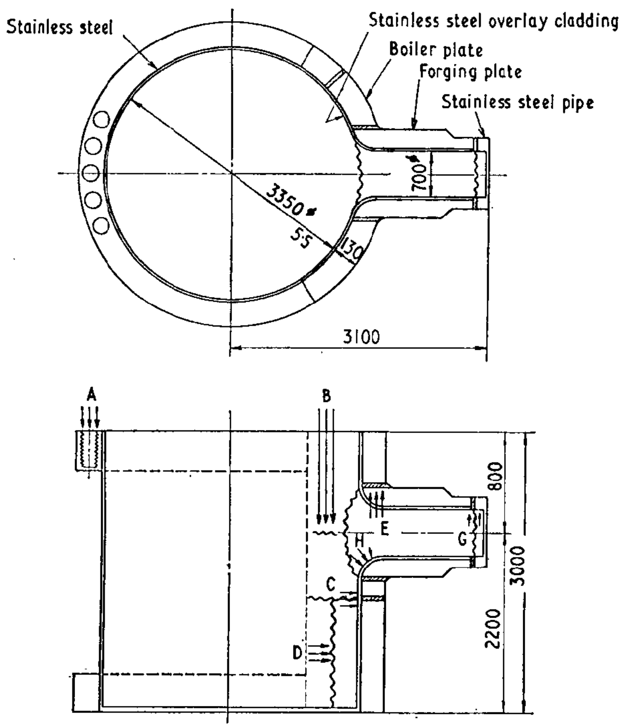

Fig. 25.9. Model vessel for the mock-up test

Vertical position	±3 mm
Lateral position	±2·5 mm
Angle of transducer unit rotation (when transducer is rotated 90°)	±0·1°
Scanning speed	max. 100 mm/s

DATA ACQUISITION APPARATUS

Outline of the system

An overall configuration of the data acquisition apparatus for this tool is shown in Fig. 25.7. Two sets of transmitter/receiver are provided. They are the electronic amplifier to transmit and receive ultrasonic beams. The pulse-echo height received is observed by a cathode-ray tube display. Pre-set voltage is provided at the data recording parts to eliminate the effect of noise echo by a cladding on a surface to be inspected, and to record only a flaw signal which exceeds this pre-set voltage. The flaw signal is recorded by a pen recorder as well as being magnetically recorded using a data recorder. A signal conditioner is provided to adjust the signal voltage level to be recorded by a data recorder. An alarm signal will be emitted if a flaw signal over the pre-set level exceeds an acceptable limit. In addition, the data recorder is recording the presence of a flaw and its position simultaneously.

Position signal

Although the search unit is 20 mm in diameter, the inspecting position will be displayed every 1 mm of its movement. The position is measured as a distance (rotation angle, vertical position, radius) between the reference point and the inspected part, which is recorded and displayed digitally on a control panel. In order to detect the co-ordinates of the moving distance to each direction, the signals are emitted by a shaft-encoder (DECIHI-CODE, Pulcen), which is installed on a driving screw to drive search units.

Recording unit

A data recorder is capable of the simultaneous recording of all signals from the transducers, position, scanning direction, and audio signal for recording. The signals recorded by a data recorder are easy to retrieve. The pen recorder records continuously the signals exceeding the pre-set voltage to inspect visually the presence of flaws macroscopically.

FIELD ASSEMBLY

In order to perform the inspection of a nuclear reactor vessel using this tool, an assembly stand on which this tool is built must be set in advance. Adjustment of the controlling function of each manipulator and the calibration of sensitivity of each transducer are also performed in this state. The stand for a field assembly is constructed of the frames shown in Fig. 25.8, which are easy to assemble and disassemble. The size is approximately 5000 mm in diameter and approximately 9000 mm in height. Therefore a space is required wide enough for the installation of this stand near the nuclear reactor when it is assembled in the field.

MOCK-UP TEST

A mock-up test is planned as a utilization test in the near future. The vessel model for the test has a cylindrical shape with a length of 3000 mm and an inner diameter of 3350 mm, as shown in Fig. 25.9. A part of the wall thickness is constructed of a size comparable to the real vessel, where a nozzle of an actual inner diameter is attached. On the inner surface, cladding of stainless steel is performed. Thus, the model simulates the real vessel. A flaw of proper size (either artificial weld flaws or machined defects) is made on a part of this model which corresponds to the area to be inspected, as shown in Table 25.2. In this manner, the operational function and flaw detectability of this tool is planned to be examined.

CONCLUSIONS

M.H.I. initiated the development of the Mitsubishi nuclear reactor vessel ultrasonic inspection tool several years ago. At present, the greater part of it having been already completed, a utilization test is planned using a vessel model for mock-up testing.

In order to secure the higher safety factors required of a nuclear reactor vessel, the completion of this in-service inspection tool is urgently required, in the development of which M.H.I. is making its greatest effort, with the collaboration of each division of the various related fields.

APPENDIX 25.1

REFERENCE

(1) Gross, L. B. and Johnson, C. R. 'In-service inspection of nuclear reactor vessels using an automated ultrasonic method', *Mater. Eval.* 1970 **28** (No. 7), 162.

C26/72 DEVELOPMENT OF IN-SERVICE INSPECTION SAFETY PHILOSOPHY FOR U.S.A. NUCLEAR POWER PLANTS

S. H. BUSH* R. R. MACCARY†

The American Society of Mechanical Engineers' (A.S.M.E.) Section XI Code on 'In-service inspection of nuclear reactor coolant systems' represents the joint efforts of the United States Atomic Energy Commission's regulatory organization and of the nuclear industry. Efforts initiated in 1967 culminated in a code in 1970. Significant features of this code are: (1) the concept of designing the system to permit inspection and possible repairs; (2) the requirement of a complete examination prior to start-up to serve as a base line for future examinations; (3) the acceptance of new inspection systems or techniques more amenable to remote applications, provided such systems can be validated; and (4) the establishment of inspection periods and level of inspection for given components or sections of components based on the concepts of relative probability of degradation of the various portions of the systems and the significance of such degradation to the safety of the reactor system.

The general safety philosophy that led to the development of the Section XI Code is described, together with a general description of the code coverage. This code represents a significant departure from A.S.M.E. policy in that it deals with the operational phase of a pressure system compared to the design, fabrication, and construction stage.

Some consideration is given to future revisions in the code and their significance to safety and to reactor operation. It is felt that this code corrects one of the major reservations inherent in nuclear systems where great care was given to design and fabrication but no analogue existed to the regular inspections typical of fossil-fuel plants during their operation.

INTRODUCTION

THE PRACTICE of periodic inspections of pressure-retaining components of fossil-fuelled power stations is well established as essential to a high degree of availability of the power station. Although the nature and extent of the inspections have, in a large measure, evolved from several decades of operating experiences of many power stations, no formalized codes for conducting these inspections had been developed in the United States. Rather, a variety of rules have been imposed by the enforcement authorities having jurisdiction at the power station sites. Industry codes focused principally upon the development of recommended rules for the care of power boilers (1)‡, with relatively limited attention being given to other pressure-retaining components such as piping, pumps, and valves.

The design arrangements of pressure-retaining components in fossil-fuelled power stations have always considered the need for ready access to all components to facilitate inspections, maintenance of equipment, repairs, and replacement of components as required. However, with the development of nuclear power plant designs, system designers initially believed that periodic inspections of the reactor coolant pressure boundary would be impracticable due to the radioactivity of the system. Consequently, very limited attention was given to the needs for in-service inspection in early nuclear power plant designs, so systems were not provided with adequate access for inspection of many components.

The system designers assumed that in-service inspections would be unnecessary, provided the nuclear power system components were designed and constructed to higher quality standards than those applied to fossil-fuelled power stations. The development and publication of the first edition (1963) of the A.S.M.E. Boiler and Pressure Vessel Code, Section III, 'Nuclear Vessels' (2) was a first step in response to this need for high-quality standards of vessel components in nuclear power stations.

DEVELOPMENT OF AN IN-SERVICE INSPECTION CODE

As the number of nuclear power stations in service increased, operationally induced defects of components requiring repairs also increased. The U.S. Atomic Energy Commission (A.E.C.) recognized as early as 1966 that the enhanced quality standards applied in the design and construction of components of the reactor coolant pressure boundary did not justify the omission of a planned programme of periodic in-service inspections. Therefore, the regulatory division of U.S. A.E.C. began to develop criteria for the in-service inspection of nuclear reactor

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* *U.S. A.E.C. Advisory Committee on Reactor Safeguards, Battelle Memorial Institute, Pacific Northwest Laboratory.*

† *U.S. Atomic Energy Commission, Washington, D.C., U.S.A.*

‡ *References are given in Appendix 26.1.*

coolant systems; the U.S. A.E.C. also encouraged nuclear industry code-writing organizations to further up-grade the level of quality standards for all pressure-retaining components of nuclear power stations.

A comparable effort on development of an in-service inspection code by the U.S. nuclear industry began about the same time, eventually leading in late 1967 to a joint A.E.C.–industry co-operative code development programme under the auspices of the American National Standards Institute (A.N.S.I.) N-45 Committee, and with the sponsorship of the A.S.M.E. An initial 'Draft code for in-service inspection of nuclear reactor coolant systems' was published in October 1968, followed by the publication of the 1970 edition of the A.S.M.E. Boiler and Pressure Vessel Code, Section XI, 'In-service inspection of nuclear reactor coolant systems' (3). In addition, the A.S.M.E. appointed a standing committee, the A.S.M.E. Subgroup on In-service Inspection, responsible to the A.S.M.E. Subcommittee on Nuclear Power (Section III Code). Participation by members of the U.S. A.E.C. Advisory Committee on Reactor Safeguards and the Division of Reactor Standards on these committees should assure the continued joint co-operation with industry in further developments and improvements of the rules for in-service inspection.

A.E.C. ADOPTION OF IN-SERVICE INSPECTION CODE

The U.S. A.E.C. formally accepted several A.S.M.E. codes, including Section XI, in its recently published (1971) 'Codes and standards for nuclear power plants' (4), which includes adoption of the rules of the in-service inspection code as mandatory requirements to be met by applicants for licences to build and operate nuclear power stations in the United States. The rules of the A.S.M.E. Section XI Code also fulfil the requirements of A.E.C. General Design Criterion 32 (5) with respect to periodic inspection and testing of important areas and features of components which are part of the reactor coolant pressure boundary.

LOSS-OF-COOLANT ACCIDENT 'ENVELOPE'

In the early stages of development of the rules for in-service inspection, the regulatory division of the U.S. A.E.C. was faced with the major task of identifying those pressure-retaining components of light-water cooled and moderated nuclear power plant systems [i.e. pressurized water (P.W.R.) and boiling water reactor (B.W.R.) plants] most important to safety. Components whose malfunction or structural failure could potentially impair the safe continued operation of the nuclear power system merited continued surveillance throughout the service lifetime of the plant. In-service inspection should determine the prevailing structural integrity of the components so that necessary corrective measures could be taken on a timely basis.

Although the structural integrity of reactor primary coolant systems of P.W.R. and B.W.R. plants was generally accepted as essential to the operation of power plants without undue risk to the health and safety of the public, criteria specifically defining the boundary had not yet been formulated at that time.

REACTOR COOLANT PRESSURE BOUNDARY

A comparative study was undertaken to delineate the reactor coolant pressure boundary of typical P.W.R. and B.W.R. plants. The system boundary criteria which evolved from this study identified for the first time the loss of reactor coolant accident 'envelope' of the system. More specifically, the reactor coolant pressure boundary envelope included all those piping runs interconnecting components of the system whose postulated failure could result in loss-of-coolant accidents which would require the isolation of all systems penetrating the primary reactor containment and the operation of engineered safety feature systems to mitigate the consequences of such postulated failures.

The boundary definition, as currently specified in paragraph IS-120 of the A.S.M.E. Section XI Code (3), identifies all components of the system (e.g. pressure vessels, piping, pumps, and valves) which must be designed to the highest quality standards (6), and the number of such components subject to in-service inspection under the rules of the articles of the A.S.M.E. Section XI Code. Figs 26.1 to 26.4 of typical P.W.R. and B.W.R. plants prepared by the U.S. A.E.C., which graphically delineate the reactor system pressure boundary as well as associated systems, were included in the code for the purpose of general guidance. In effect, the boundary encompasses the reactor coolant system, and portions of systems connected thereto, located within the confines of the primary reactor containment structure up to and including the outermost isolation valves in system piping penetrating the containment.

Identification of the components (and their overall system arrangement) within the reactor coolant pressure boundary provided the initial bases for the development of appropriate in-service inspections. The primary problem associated with the inspection of reactor coolant pressure systems was the need to examine components in radiation fields where access by personnel would not always be practical. This difficulty, totally unlike conditions in fossil-fuelled power stations, posed the greatest challenge in the development of in-service inspection rules.

INSPECTION ACCESSIBILITY

During the initial efforts to develop a meaningful programme of in-service inspection, both the U.S. A.E.C. staff members and the A.N.S.I. N-45 Committee recognized the restrictive space allowances and encumbrances, such as the structural and concrete members surrounding components of nuclear power stations, which would seriously interfere with any contemplated inspection, and, in some areas, preclude any examination other than with special mechanized and remotely-operated equipment developed for such purposes.

To remedy this situation, the designers of nuclear power stations had to be made aware of the need to change the prevailing design philosophy from one of 'restricted accessibility' to one of adequate 'in-service inspection accessiblity'. This intent is now clearly reflected in the general provisions for access requirements specified under Article IS-140 of the A.S.M.E. Section XI Code. These accessibility requirements, as applied to recently designed power plants, have substantially influenced the current design of components, the component supporting arrangements in the reactor coolant systems, and the spatial

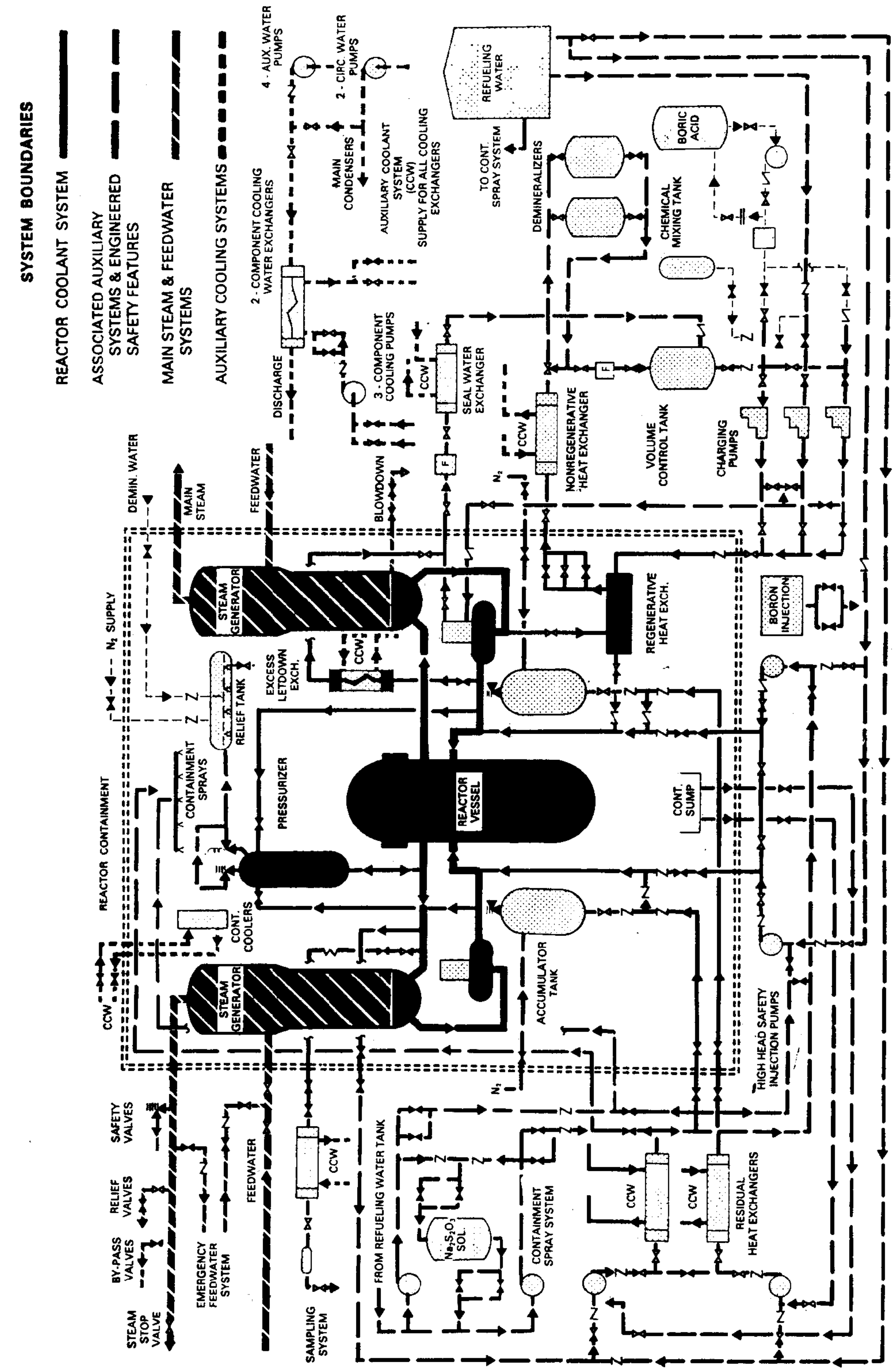

Fig. 26.1. Typical P.W.R.-1 plant

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