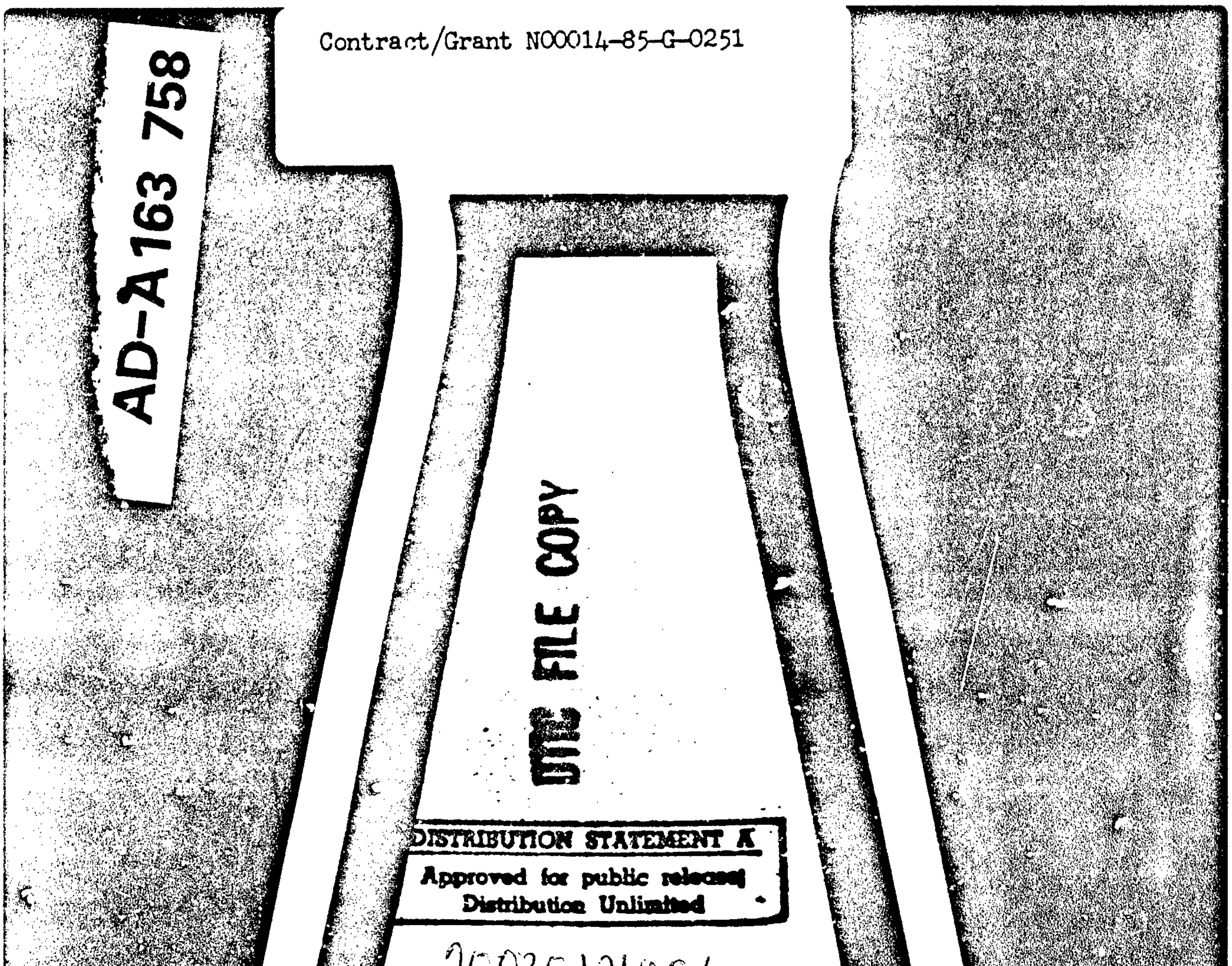
Maintenance Welding in Nuclear Power Plants/III

AWS

November 6-8, 1985
Hyatt Regency Hotel
Knoxville, Tenn.

CONFERENCE PROCEEDINGS



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MAINTENANCE WELDING IN
NUCLEAR POWER PLANTS III

Maintenance Welding in Nuclear Power Plants III was held at Knoxville, Tennessee, November 6-8, 1985. The purpose of this conference was to provide participants with the current developments and trends in this important segment of the welding industry and to help attendees understand and obtain an international perspective of these trends from a global roster of speakers and participants.

The American Welding Society is indebted to the Program Advisory Board which was co-chaired by Joseph Danko and Philip Flenner, and to the speakers and authors who provided the technical papers for these proceedings.

Finally we want to acknowledge the attendees whose participation and probing questions contributed immeasurably to the success of this conference.

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CONTENTS OF PROCEEDINGS

STEAM GENERATOR AND PIPE REPLACEMENT OF GERMAN LIGHT WATER REACTORS

by

Alexander Hümmeler*
Dieter Pellkofer**

1 Servicing Activities in the Power Plant Area

An ever increasing number of nuclear power plants is operating safely and satisfactorily today in the whole world. The plants are subjected to systematic in-service inspections at regular intervals; these inspections make it possible to detect and repair any defects present in time. Innovations and improvements of the plant safety or economic efficiency, however, also require more or less extensive activities to be performed in existing plants. For this reason, a special servicing department dealing with repair and maintenance activities, in particular in radiation areas, has been developed for nuclear power plants. The following is a description of KWU's engineering and planning potential and the state of development of the mechanized inspection techniques using actually performed activities as an example.

2 Planning is the Most Important Precondition

The preparation phase is one of the most important phases in the replacement of a large component or piping system in a nuclear power plant. It normally starts with the elaboration of a feasibility study. This procedure will be explained using the steam generator replacement performed in Obrigheim nuclear power plant in 1983 as an example. The reason for the replacement of this steam generator has already been given in /1/. The feasibility study has already been performed on the occasion of the order for the new steam generators in 1972.

Thus it was possible to start with the formation of a working team as a first step immediately upon receipt of the order for the actual steam generator replacement. This team consisted of employees with appropriate experience which were chosen from various engineering departments and were provided with an organisational structure suitable for their task and with appropriate job descriptions and responsibilities. Figure 1 shows the organisational diagram for the steam generator replacement in Beznau, now in its preparatory phase.

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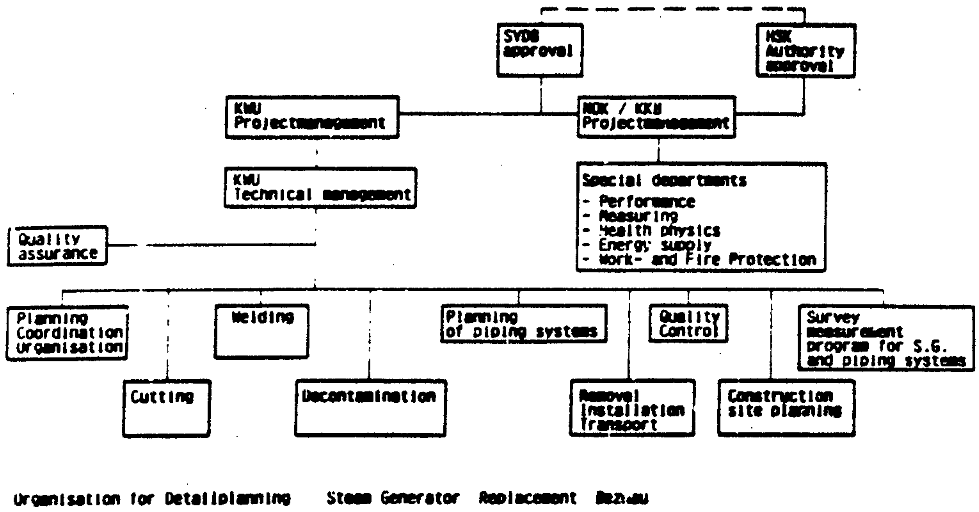


Fig. 1

After the determination of the concept, the essential work to be performed was the planning of the transport routes and the removal of interfering components. The feasibility study had already shown that the plant concept made it possible to move and replace the steam generators as complete units without any changes required to the containment (Figs. 2 and 3).

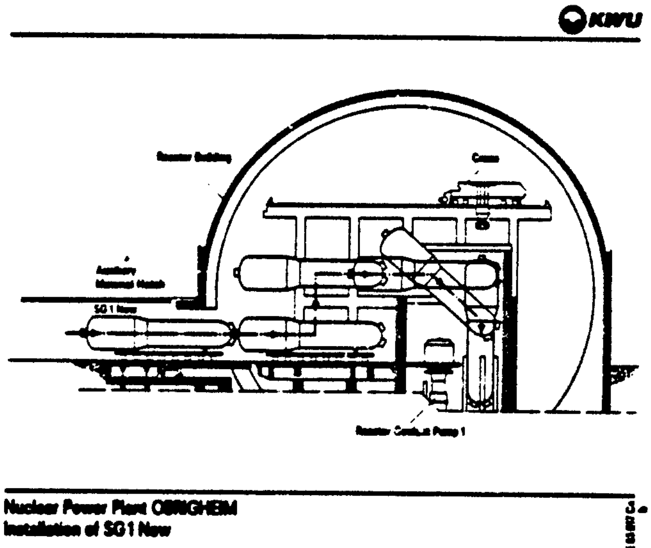
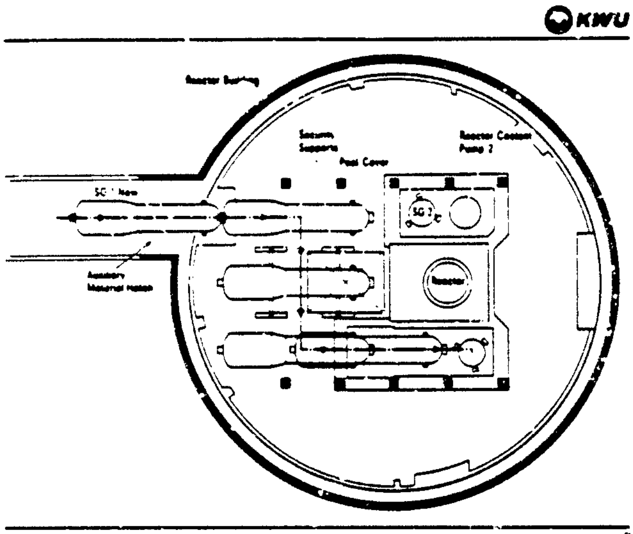


Fig. 2



Nuclear Power Plant OBRIGHEIM
Installation of SG1 New

Fig. 3

The equipment airlock had to be removed and replaced temporarily by a larger assembly airlock. The lifting capacity of the reactor building crane was sufficient to transport the steam generators. In addition to the crane, a special lifting beam had to be installed in order to lift the steam generators high enough and to bring them into the horizontal position. To protect the fuel assembly storage pool against fouling and parts falling down, the fuel pool was covered by a layer of steel girders placed side by side which, in turn, were covered additionally by steel plates for the duration of the assembly time. Furthermore, a heavy-load drop stop was installed above the pool which protected the fuel assembly storage pool against a hypothetical drop of the steam generator and crane failure while crossing the fuel assembly storage pool.

The preparation time available from the date the decision to replace the steam generators was made until the actual replacement was relatively short. The following major activities had to be performed within this time:

- Preparation of detailed plans with personnel requirement and radiation exposure in mrem
- Licencing procedure
- Design and construction of a storage hall of a surface area of about 250 m² for the old steam generators
- Fabrication of eight new sections for the reactor coolant line and of various piping sections for the main steam and feedwater lines

- Integration of the new steam generators into the existing system because of their higher output rating
- Development of a suitable decontamination procedure (electro-polishing) for the reactor coolant lines including the manufacture and testing of manipulators and shielding
- Personnel training on a model of scale 1:1 by simulating the spatial conditions in the steam generator channel head and testing of the necessary equipment, mainly grinding and cutting equipment
- Manufacture and testing of the lifting and tilting device
- Construction and furnishing of a canteen for about 900 fitters for the duration of the overhaul
- Initiation of the radiological protection measures and enlargement of the access to the controlled area

Another important step in project processing was the elaboration of the planned time schedule which is shown in a simplified form in Fig. 4 (white bars).

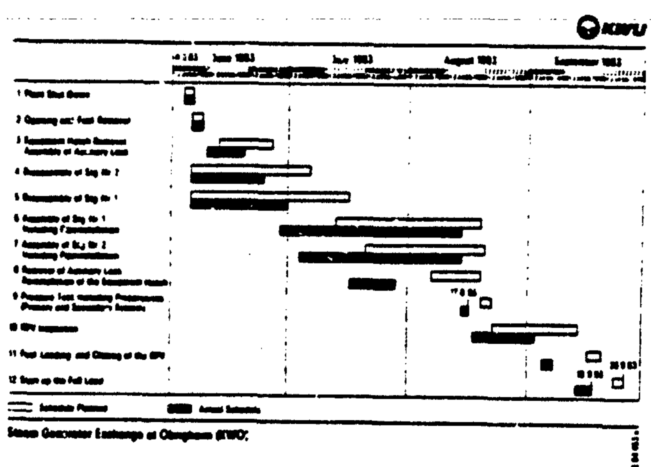


Fig. 4

For the purpose of this paper, and for better understanding, only the cross-section macrograph plan of the reactor coolant line shown in Fig. 5 will be given as an example out of the variety of detailed plans. During the performance of the welds, the weld sequence and the resulting welding shrinkage played an important role. To ensure stress-free fitting of the newly fabricated parts into the existing piping system, a specific weld sequence had to be followed. For the welding of internally clad ferritic pipes by means of remote-controlled welding machines which will be described in detail later, a special welding procedure had been developed /2/. Figure 5 shows the metallographic section of a "Cardo-weld".

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Welding position: 1G

Path A–B		Path C–D	
Punch location	HV1	Punch location	HV1
Weld metal	214	Cladding	239
	185		239
	205		255
	210		235
	214		270
	229		270
	229	1. FLC–layer	293
	229		324
	234	2. FLC–layer	251
2. FLC–layer	201		260
	201		280
	201		225
	201		226
	205		216
1. FLC–layer	268		215
	287		200
	297	Weld metal	223
	290		229
Cladding	251		226
	238		224
			224
		HAZ	255

One–Side Welding of Clad Pipes (TIGW–Mechanized):
Hardness (HV1) of FLC–Area

Fig. 5

A major item in the preparation activities was the training of personnel on models, this training was carried out for all essential activities.

The software time consumption in hours is shown in Fig. 6 for all the planning and preparatory work.

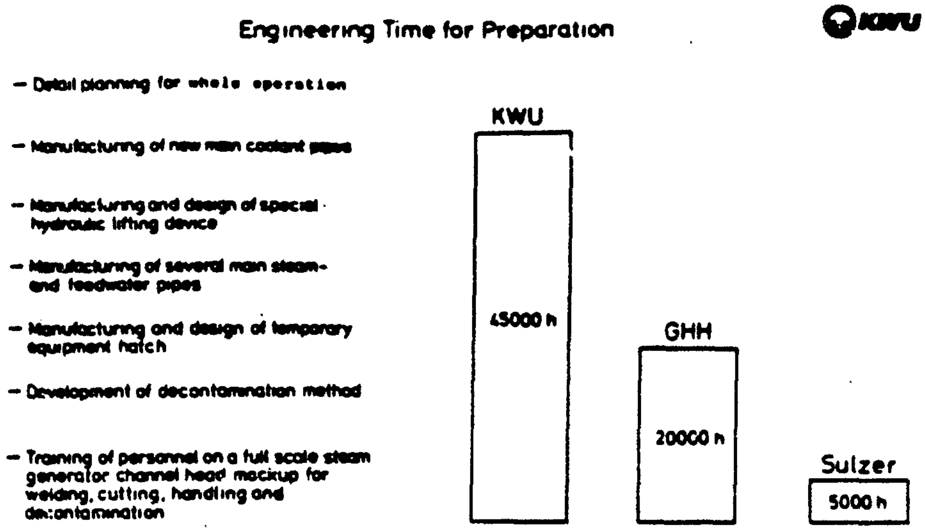


Fig. 6

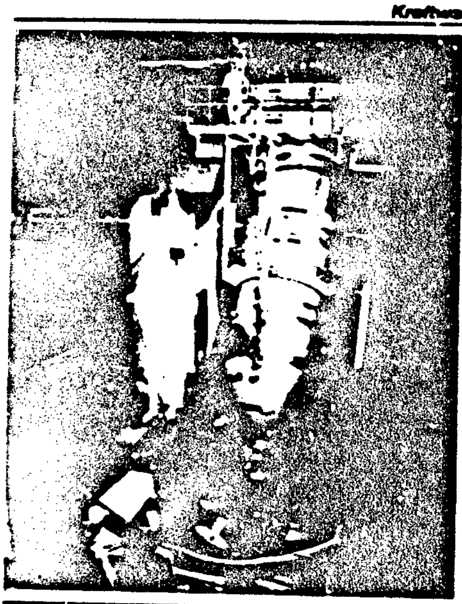
The concept described above for the replacement of components is also valid for piping, for example for replacement of recirculation piping. The planning and preparation activities at the Santa Maria de Garona nuclear power plant (a boiling water reactor plant in Spain) were performed along the same lines.

3 State-of-the-art Equipment is Required
(Performance and Equipment)

It is not possible to deal with all the details of the performance of the project in this publication. For this reason, only a few special devices can be presented.

3.1 Transport

If it is possible to transport the heavy components easily within the containment, the feasibility study has passed the test. But personnel training, experience and proper functioning of the auxiliary equipment also contribute considerably to the successful performance of the work. Figure 7 shows the Obrigheim steam generator while being lifted from its installation position. It was gradually lifted further up to the



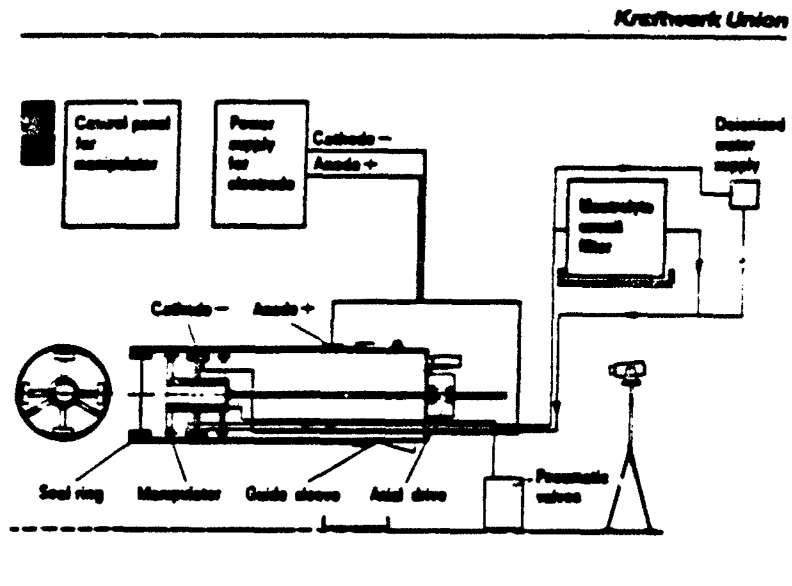
Lowering of the Steam Generator

Fig. 7

building crane by means of the hydraulic traverse and tilted by a second crane having a low lifting capacity. Arrangement and design of the retaining straps and their relation to the centre of gravity proved to be correct. Twelve hours and 45 minutes were required to perform this work for the first time and eleven hours and 25 minutes for the second time. This shows that it was worthwhile to plan the work in detail.

3.2 Decontamination and Shielding

The use of efficient decontamination procedures enables personnel radiation exposure to be reduced considerably today. One of the procedures that was developed further by KWU is the electropolishing process.



Decontamination of Primary Coolant Pipes
Electropolishing, Process System

Fig. 8

Figure 8 shows the schematic diagram of the process. The decontamination factor achieved in the primary system piping during the steam generator replacement was greater than 200. Most of the equipment is mechanized and remote-controlled. Figure 9 shows the use of the electropolishing equipment for decontamination. The amount of radioactive waste produced during this procedure is extremely small.

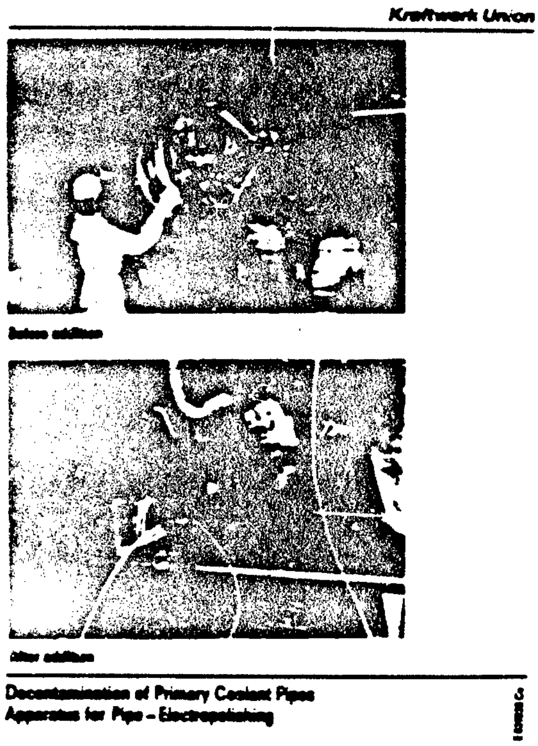


Fig. 9

Special austenitic plugs have been developed as radiation shielding in the reactor coolant line. These plugs consist of individual modules which are interconnected by means of a rope. After cutting the pipe, they are placed behind the section to be decontaminated. During welding and annealing, the austenitic plug and a thermal insulation plug remain in the piping. The entire system is then pulled out of the pipe by means of a recovery rope as shown in Fig. 10.

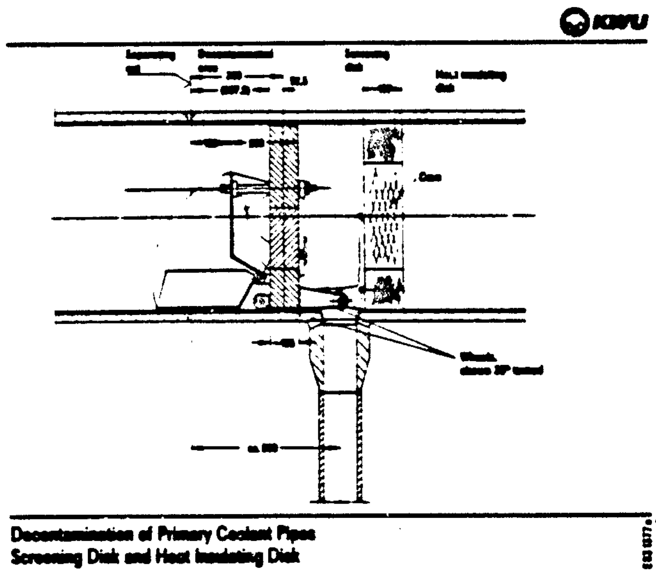


Fig. 10

The plug will separate into individual pieces in a controlled manner so that these can be removed from the pipe through an opening which is smaller than the pipe diameter. This means that the weight of only one module at a time will have to be handled.

3.3 Cutting of Pipes

Remote-controlled equipment monitored via video cameras were used for cutting the pipes. Figure 11 contains a diagrammatic sketch of this equipment and the various types available. This

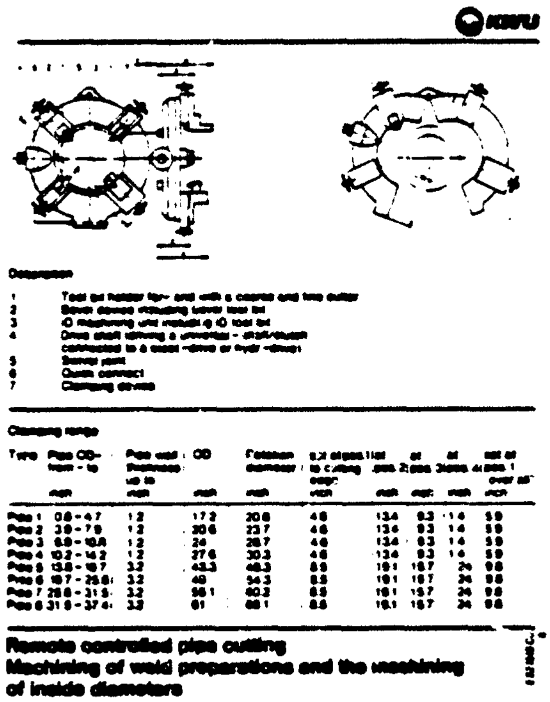


Fig. 11

equipment was tried and tested during the cutting of the reactor coolant lines in Obrigheim, but also in a variety of other applications such as feedwater and main-steam line replacement in the nuclear power plants of Würgassen, Brunsbüttel, Isar and Philippsburg /3/.

3.4 Weld Preparation
(Calibration, centering and machining)

The weld to be performed on the pipes can be prepared by means of the cutting equipment. However, a special system developed for the machining of pipe and elbow end faces is of course more suitable. This KWU pipe machining system consists of an equipment package which can be used in situ because it is of low weight and can be handled easily, but which is also suitable for use in the workshop.

The pipe machining system comprises:

- a pipe expanding device for adaptation of the inner diameter and for calibration (Fig. 12)

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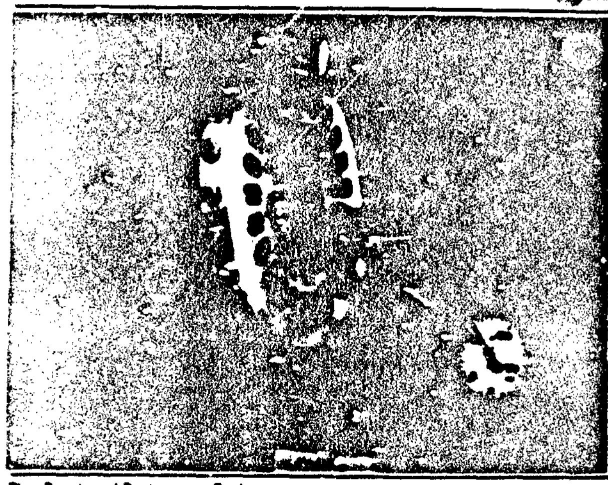
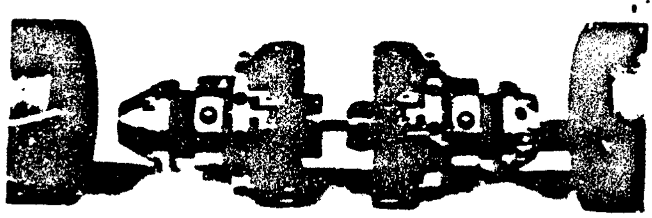


Pipe Repair and Replacement Tools
Pipe Expansion Device (DN [illegible] inch Pipe)

Fig. 12

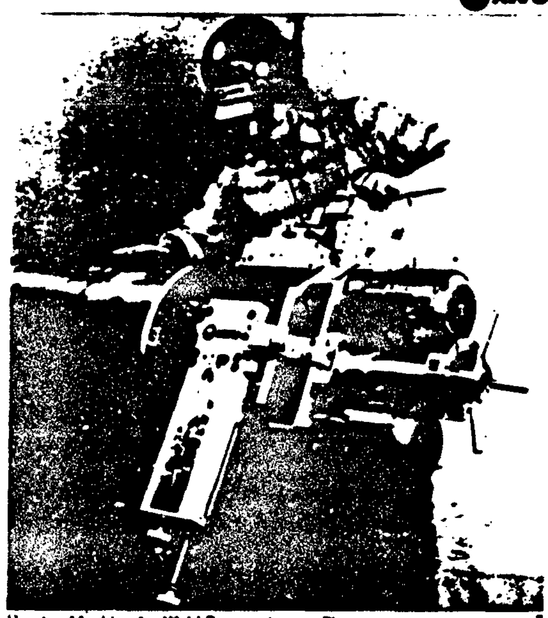
- a pipe centering device for centering, elimination of any out-of-roundness present at pipe ends and for hydrogen-nitrogen mixtures supply (Fig. 13)

- a turning equipment for pipe end machining (Fig. 14).



Pipe Repair and Replacement Tools
Pipe Centering Device

Fig. 13



Turning Machine for Weld Preparation on Pipes

Fig. 14

This pipe machining system has been designed primarily for use in pipes of diameters from DN 200 to DN 800. Figure 15 represents a schematic diagram of the preparation of ends of

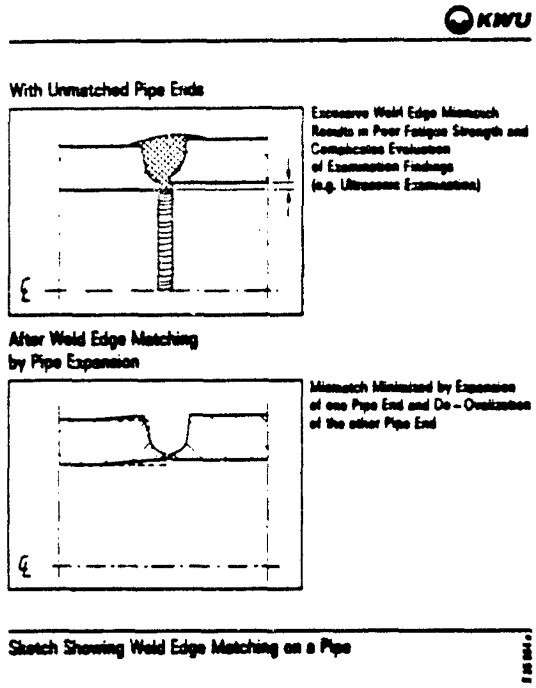


Fig. 15

a pipe weld joint. As can be seen from the diagrams, this pipe machining system enables the ends of the weld joints to be optimally adapted and machined. The equipment is driven hydraulically via a hand pump.

The pipe centering device enables exact centering and easy adjustment of the root opening. It is mainly used for pipes and elbows the ends of which reveal too great an out-of-roundness (ovality). The pipe centering equipment makes it possible to press pipe ends into a nearly circular shape up to a wall thickness-to-diameter ratio of about 5 % and to retain this shape during the welding procedure. If necessary, a pipe alignment deviation of up to 5° can be adjusted. After completion of the welding procedure, the equipment is detensioned by remote-control and removed from out of the pipe as a complete unit.

The centering equipment which can be split into two halves is suitable for the connection of additional devices for machining the welding edge and can also be used together with the pipe expansion device.

3.5 Welding

To achieve the present state of remote-controlled welding of pipes, extensive development activities in the fields of process and equipment engineering had to be performed in several steps:

3.5.1 Welding Procedure

The welding procedure used for mechanized welding of permanently installed pipes is the pulsed GTAW welding procedure with a cold filler electrode which has already been used successfully for several years. It has been proved to be difficult to increase the relatively low melting rate of this welding procedure /4/. For this reason, the volume of the weld joint was reduced and a weld with an extremely narrow root opening developed (Fig. 16, /5/). For the welds to be performed during the steam generator

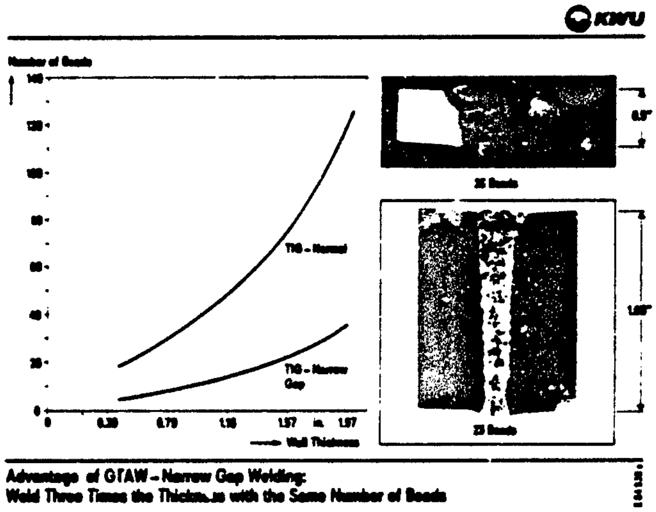


Fig. 16

replacement in Obrigheim, however, the welds were still prepared in the conventional way. The first use of the narrow root opening welding procedure for the welding of pipes on site was in 1984 in the Brokdorf nuclear power plant after qualification has been obtained for ferritic pipes from the German authorized inspector.

For the replacement of the recirculation loop in the nuclear power plant of Santa Maria de Garona, the procedure qualification was obtained for austenitic weld joints in accordance with the ASME Code. The procedure has proven to be efficient in practice.

3.5.2 Orbital Welding Machines

The use of the various generations of orbital welding machines by KWU started with commercial devices. For the replacement of the steam generators in Obrigheim, however, orbital welding heads designed by KWU have already been used. Figure 17 shows the type series used at present. It starts at approx. 4 mm

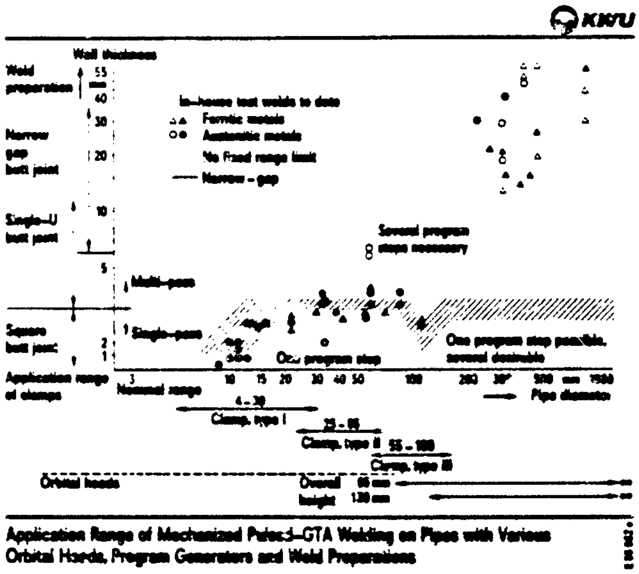


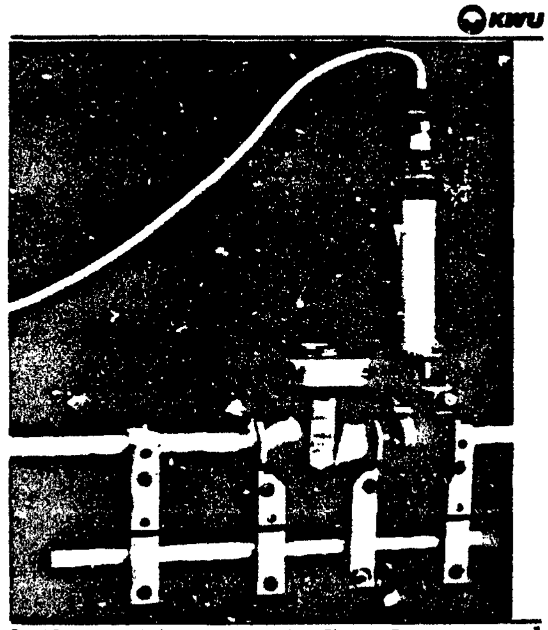
Fig. 17

diameter, with a minimum height of the welding head of 25 mm; this means that the clear distance between two pipes or any other stop need not be more than 25 mm. KWU has designed and manufactured its clamping-type welding heads on the basis of the experience gained in practice so that they have the following advantages:

- the centering system always ensures centric tensioning
- the welding head is always vertical to the pipe surface
- the electrode supply system is located on the welding head (no additional device required)
- the applicable diameter range is larger than that for comparable commercial clamp-type welding heads
- overall size is reduced
- use on flanges, fittings and tees is ensured.

Special assembly aids have been developed for welding by means of clamp-type welding heads; they have to comply with the following requirements:

- Easy centering and orientation of the pipe ends of several welding points simultaneously.
- Uncomplicated preassembly of piping sections in situ.
- Adaptability to the various fittings, tees, valves etc.



Small Pipe Assembly System – Welding Head Clamping Device

Fig. 18

Figure 18 shows the use of clamp-type welding heads and their assembly aid. For example, this system enables the preassembly of units destined for radiation areas to be performed to the exact dimensions.

For the orbital welding heads developed by KWU for large-diameter pipes, the tensioning system has been designed such that not only the welding head but also the grinding equipment for grinding the weld joint top, the milling device for machining areas to be repaired and the automatic UT testing equipment can be clamped onto the same guide ring (cf. the following sections). This ring, which is quickly to assemble, can be aligned by means of a simple auxiliary device.

The welding heads are designed for 100 % remote control and are characterized by a high precision of the guides and motions.

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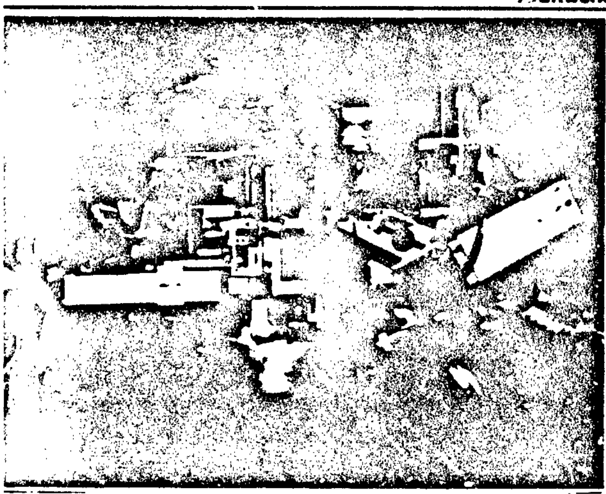


Orbital Welding Method
Orbital Welding Head for External Pipe Welding

Figure 19 shows the relatively large orbital welding head as used for the steam generator replacement in Obrigheim.

The present state of development of a narrow root opening welding torch at a height of only 130 mm is shown in Fig. 20. Each of the two heads is equipped with two video cameras.

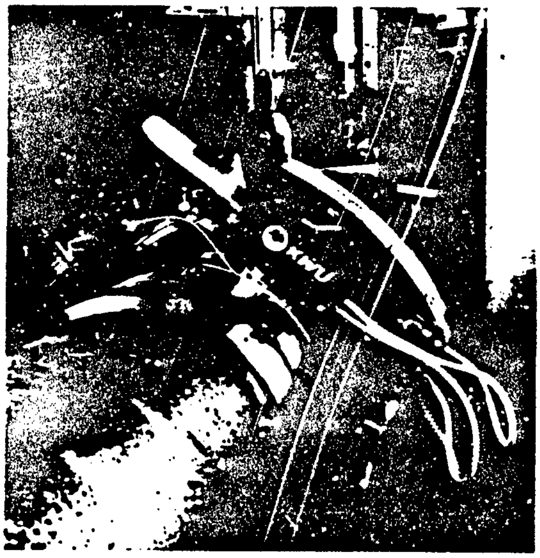


Fig. 20 Orbital Welding Head for GTA Narrow-Gap Welding with TV Monitoring

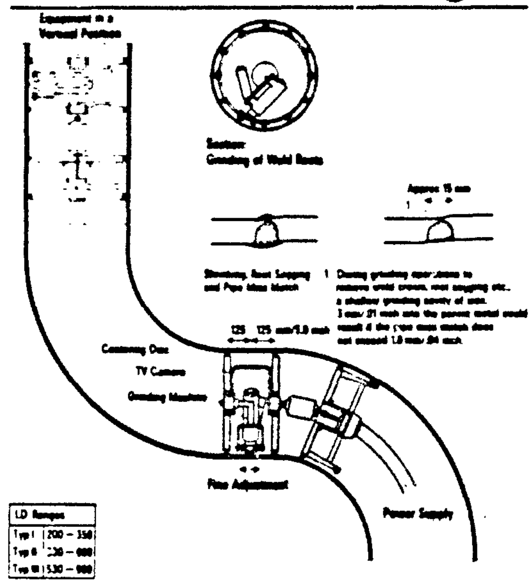
The last figure clearly shows the two video cameras on the welding head, which are not bigger than a cigarette packet. This latest generation of the so-called CCD cameras was developed especially for welding /7/. It is possible with this system to suppress the brightness of the arc at any position in a controlled manner and to brighten the surrounding field that has to be observed. This new system will be used first for the recirculation loop replacement in Spain already mentioned before. The devices for grinding the joint top, milling any areas to be repaired and for ultrasonic testing will be adapted to this system to an overall height of 130 mm.

Because of the considerably shorter welding times, the GTAW narrow opening welding procedure will probably render it possible to achieve an economic breakthrough in the conventional field and for new fabrications.

3.5.3 Special Welding Machines

To eliminate failures in austenitic pipes due to stress corrosion cracking, e.g. in the recirculation loop, a manipulator developed by KWU was modified. This system is remote-controlled and can be passed through pipes and elbows like a worm (Fig. 21). It consists of two components, the tool carrier and the transport system.

KWU



Self-propelling Pipe Inside Grinding Machine
for Removing of the Weld Root Sagging with TV-control

Fig. 21

In addition to a welding head monitored via a video camera, the tool carrier can also be equipped with a grinding head, a camera, a milling tool or an isotope for the radiographic examination. It is planned to use these machines for grinding or milling out the areas affected by stress corrosion cracking and subsequently to make up for the missing material by means of a suitable surfacing weld. The internal welding machine shown in Fig. 22 is connected to a programmable high-quality power source with which it is possible to build up any desired number of beads automatically.

KWU

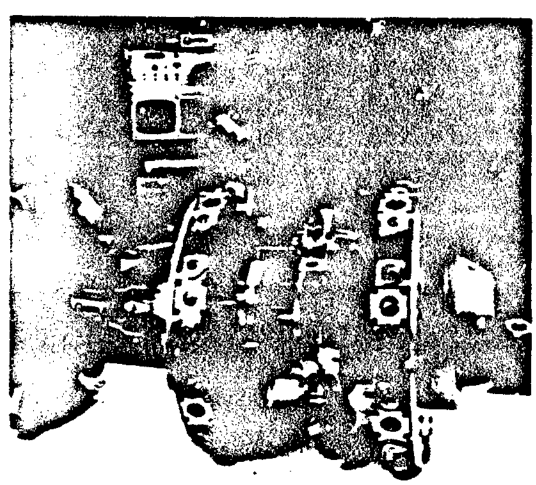


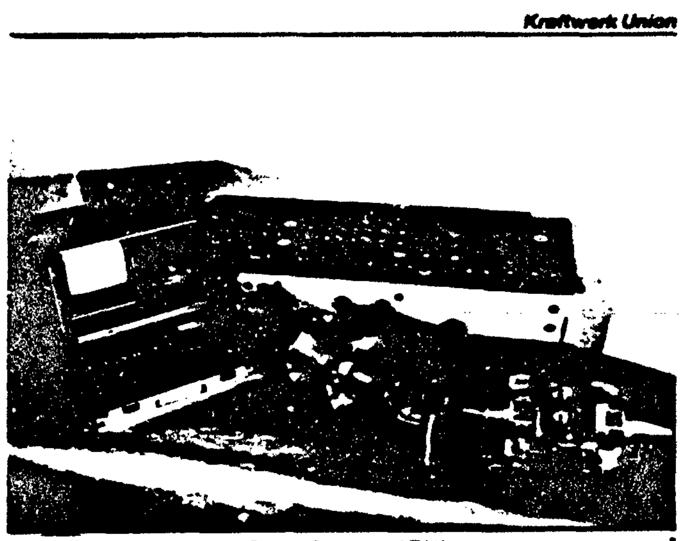
Fig. 22

In-Pipe Operations
Welding Machine

Other remote-controlled special welding machines which operate on the basis of the pulsed GTAW welding procedure and which are used for piping and vessel construction, have already been discussed in /7/.

3.6 Grinding

The grinding of pipes has already been discussed on several occasions. Figure 23 represents the pipe manipulator mentioned



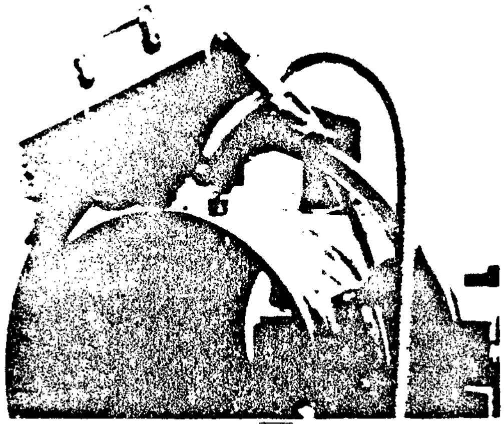
In-Pipe Grinding Machine with Control Console and TV Unit

Fig. 23

before which is equipped with a grinding disc. Over and over again, this system has proven to be efficient for grinding misaligned weld edges and poor manually-welded roots.

An external welding machine to be fixed to the guide ring of the welding head can be seen in Fig. 24. In addition, it can be equipped with an extraction system for the grinding dust. This system has already been successfully tried and tested during the steam generator replacement in Obrigheim. Compared to manual grinding, the grinding times are considerably shorter and the surface quality obtained is improved at the same time.

Kraftwerk Union



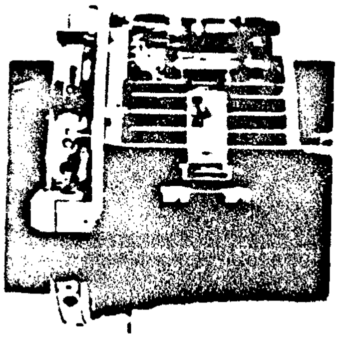
Circumferential Belt Grinder for Pipe-OD-Grinding

Fig. 24

2.7 Ultrasonic Testing

Figure 25 represents a search unit for ultrasonic testing which, in this case, is mounted on the guide ring of the GTAW narrow opening orbital welding head. It is being considered to install the guide ring permanently and to use it further for in-service inspections. This would facilitate exact localization and reproducibility of the findings at a later date.

KWU



Features:

- constructed in modules
- four interchangeable assembly groups: circular track, main carriage, slide arm, axial slide
- scan coverage by combined motions of the main carriage around the track and the axial slide perpendicular to the track
- easy adaptation of pipe diameters from 214 mm through 950 mm (8.4 to 37.5 inch)
- up to 20 independent pre-programmed scannings

Technical data:

overall height from pipe surface	70 mm	(2.75 inch)
total weight	about 25 kg	(55.1 lbs)
manipulator axial length		
equipped with arm "160"	350 mm	(14.0 inch)
equipped with arm "220"	415 mm	(16.5 inch)
maximum scanning speed	100 mm/sec	(4 inch/sec)
maximum number of US-search units	5	

accessories: adapter for examination of pipe attachment, nozzle and nozzle socket welds

Pipe Weld Scanner

Fig. 25

4 The Results are Convincing

4.1 Planning

It has been shown that it is possible to plan the replacement of heavy components in a nuclear power plant more or less exact in advance. The planned time schedule in Fig. 4 is almost completely in compliance with the actual time schedule (dark bars). The minor deviations, which are hatched in the time schedule, have to be compensated by a professional site management. In the case of the steam generator replacement in Obrigheim nuclear power plant it was even possible to achieve a reduction of the time schedule of one week at a total time required of 13 weeks. The actual times required for individual activities confirm and justify the necessity and requirement of in-depth planning and preparation.

4.2 Performance

- The use of efficient decontamination procedures and shielding measures enables personnel radiation exposure to be reduced considerably. Figure 26 shows how it was possible to reduce

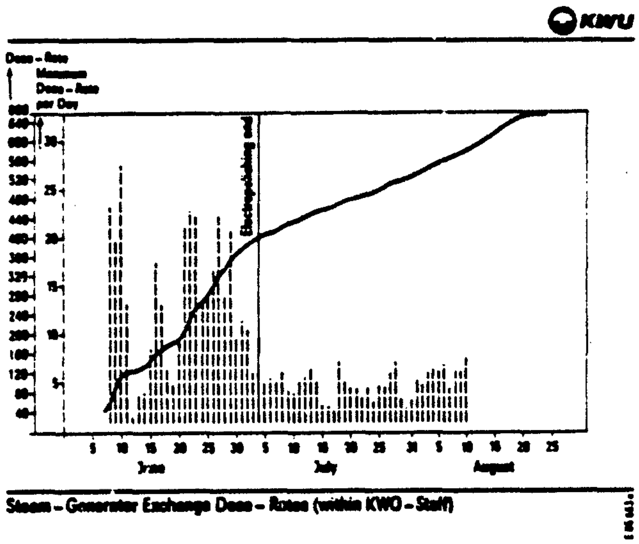


Fig. 26

  the initially high radiation exposure in mrem by means of the corresponding measures despite the increase in the number of people employed and the increase in the working hours (Fig. 27).

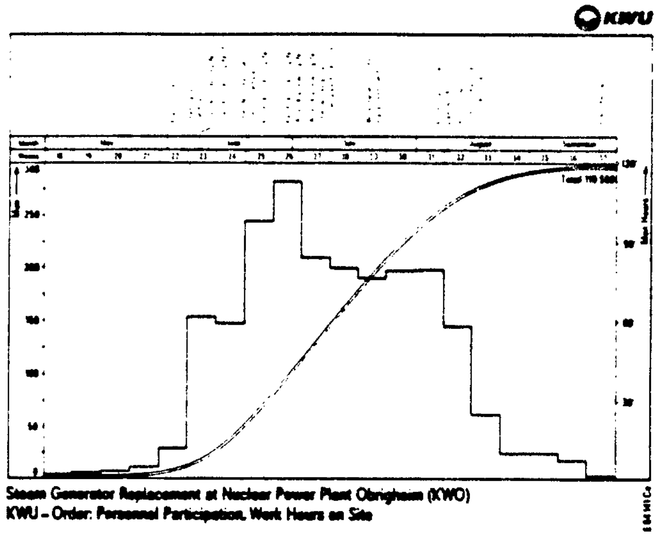


Fig. 27

- The use of highly-developed piping assembly and welding systems makes it possible to reduce the residence time of the personnel in the systems considerably. At the same time, a high quality level is achieved. It is possible to eliminate any factors that cannot be exactly planned in advance such as impermissible misalignment of the edges or out-of-roundness of the pipe ends. The implementation of weld repairs, if any, requires only a slight amount of additional work.

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IN-SITU RETUBING OF STEAM GENERATORS

BY

S. ROY

BABCOCK & WILCOX CANADA LTD.

SEPTEMBER 1985

1. INTRODUCTION

* CANDU PHWR system incorporates recirculating steam generators very similar to those in Pressurized Water Reactor (PWR) systems. There are differences in design, these are:

(i) All CANDU plants excepting Bruce 'A' & 'B' have integral preheaters.

(ii) Higher circulation ratio is employed to minimize corrosion.

(iii) Tubing material is now standardized as Incoloy 800, although Monel 400 as well Inconel 600 have been used successfully.

Twelve steam generators which were damaged and subsequently retubed in-situ were for three CANDU 600 MWe plants in Gentilly. Quebec; Point Lepreau, New Brunswick; and Cordoba, Argentina. The damaged tube condition occurred during a manufacturing heat treatment process.

The original steam generators were designed and manufactured in compliance with ASME Section III Class I requirements with the support of a full quality assurance program. The same code classification and extent of the quality assurance program was maintained for the rebuilding project.

* Canadian Deuterium Uranium (CANDU) - Pressurized Heavy Water Reactor.

2. DEFINITION OF THE DAMAGE

The damage sustained by the steam generator tubing can generally be described as indentations. These were later termed "dings" to differentiate from the tube denting phenomenon. Tubes were "dinged" at support plate locations in the straight leg portion of the tube bundle as well at the U-bend supports. Dinging produced localized tube wall indentations and some local reduction in the inside diameter of the tubing. Figure 1 illustrates dinging of tube.

2.1 DISCOVERY OF THE DAMAGE

The defects were discovered while carrying out eddy current examination of tubing because of defects introduced during tubing manufacture. The tubing defects for which the eddy current examination was being carried out were unrelated to the dings produced by heat treatment.

2.2 CAUSE OF DINGING

All steam generators were given a two stage heat treatment. The first stage heated the lower part consisting of the tubesheet, primary head and a length of the vessel including the shroud, tube supports and tubing. The second stage heat treatment consisted of heating the steam drum with its internals as well as a section of the secondary shell below the closure weld. This section included the U-bends and a length of the bundle incorporating a number of the tube support plates. Heat treatment was carried out by placing the appropriate parts of the vessels in a gas fired furnace without any gas circulation within the vessels. Although the heating and cooling rates complied with the applicable code requirements, large axial and radial temperature differentials (of the order of 150° C between the tube bundle and the secondary shell) developed. This differential induced relative axial strain between the shroud and the tie-rods in the interior of the tube bundle. The axial strain caused the edges of the support plates to move up or down relative to their centers resulting in a dished shape. Figure 2 shows the support plate rotation during heating and cooling.

2.3 INSPECTION

The damage was assessed by eddy current examination using an internal probe calibrated to measure external damage. The assessment program involved Babcock & Wilcox Canada as well as * AECL, the plant designers and the owners of the plants.

Data was collected on the tube dinging. Bowing and bending of tubes were also found. As a diagnostic measure, the flatness of the support plates was also established by eddy current examination as well as direct profile measurement of the top most plates. Not all tubes were examined rather, the investigation was carried out to a point where the cause of the tube damage was clearly established and the necessity for repair agreed.

Inspection results showed that approximately twenty-five percent of the tubes were damaged, the peripheral tubes sustaining the most severe dinging. The damage depth was generally less than 0.5mm but a small number of tubes were shown to have damage depths as high as 1.0mm.

The support plate dishing was highest in the top most plates. The angle of dishing exceeded the criteria established for tube lock up for approximately twenty-five percent of the tubing. The results of eddy current examination of Point Lepreau steam generators are shown in Figure 3. Bending and bowing of tubes were also seen in both hot and cold leg tubes, the severest cases occurring at the bundle periphery.

* Atomic Energy of Canada Ltd. are the designers of CANDU plants and were also the purchasers of the steam generators.

3. DECISION TO REBUILD

The decision to rebuild or retube the steam generators were taken jointly by Babcock & Wilcox Canada, AECL and the plant owners. The decision was based on prudence and conservatism and not based on the inspection results because these did not provide any acceptance or rejection criteria. Criteria based on stress levels could not be used either because although the ding locations could have very high stresses, ASME codes do not rule out such localized stresses.

Parallel investigations were instituted examining fretting and corrosion aspects. Fretting tests were inconclusive but corrosion tests did show that dings may accelerate penetrations of the tube walls during service.

Options examined prior to the decision were:

1. Plugging the tubes based on acceptance criteria generated by inspection.

2. Replacement of damaged tubing only.

3. Complete retubing.

The joint decision was to completely retube the steam generators to ensure reliability.

It was also rationalized that plugging of up to 20% of tubes may significantly affect steam generator performance by causing severe primary side maldistribution or tube vibration. The possibility of further tube lock up during the operation could not be ruled out either. Further, the economic penalty of station derating and that of a lengthy repair after the plant had gone into operation was judged to be very high.

The factors which influenced the decision in favour of "in-situ" repair were as follows:

1. Repair at the manufacturing facilities would have required structural re-work to the reactor building to remove and re-install the vessels causing significant project delay.

2. Reasonable access at site at that stage of construction allowed simulation of the necessary environment of key operations.

4. SEQUENCES OF DISASSEMBLY & REBUILDING

The processes of disassembly and rebuilding were distinct, each requiring engineering effort of diverse natures. The first of a kind nature of the undertaking was recognized by all parties involved and instigated the launching of a major engineering effort. Much of the engineering effort was expended in developing the new procedures while design and manufacturing proceeded in parallel. The details of the key first of a kind processes are discussed in the following sections. However, it should be noted that the disassembly process also required developmental support.

4.1 DISASSEMBLY: (See Figure 4)

The disassembly process consisted of the following steps:

(a) The internal shroud was cut manually.

(b) The vessel shell was cut just below the steam generator/drum weld by a formed cutter. The drums were moved and stored on a specially designed structure. The formed cutter also left a prepared weld edge to be utilized during the rewelding of the shell.

(c) The U-bend sections of the tubes were removed by abrasive cutting.

(d) The core tubes were removed by cutting them just above the tubesheet and pulling upwards. The cut was made with a one-revolution tube cutter. Specially rigged crane was utilized for pulling of these tubes.

(e) The remaining tubes and shroud were cut and removed leaving approximately 150mm long tube stubs beyond the secondary face of the tubesheet.

(f) The tube stubs were removed by first pulling the tubes to relax the rolled joints and necking them. This was followed by machining of the tube to the tubesheet seal welds. The pull force limitation was established by finite element analysis incorporating the tubing and tubesheet overlay properties. Integrity of the tubesheet overlay was confirmed by carrying out liquid penetrant examination following the tube stub pulling.

4.2 REBUILDING SEQUENCE (See Figure 5)

The following were the key steps in the rebuilding process of the steam generators:

(i) Install factory manufactured shroud sections containing baffle plates and tube supports.

(ii) Install tubes and tack expand tubes into the tubesheet holes.

(iii) Bake out tubesheet at 120°C for 48 hours as the final cleaning process and complete seal welding. A preheat of 37°C was maintained.

(iv) Reinstall steam drum and complete closure weld.

(v) Stress relieve the closure weld.

(vi) Perform helium leak testing of the seal welds at 414 Kpa (60 psig).

(vii) Tubes were expanded at the secondary face of the tubesheet by hydraulic expansion. Prior to installation of the tubing, the tubesheets were thoroughly cleaned and one hundred percent of the tubesheet holes were conditioned. The tube hole conditioning was carried out to remove any axial scoring, so that presence of these would not reduce the length of the leak path to the seal weld. All axial scores of 0.075mm depth or greater were removed from the upper 75mm of the tubesheet. At the primary face no score was accepted within 3mm from the edge to ensure weld quality.

5. FIRST OF A KIND PROCESSES

Incorporation of first of a kind processes was necessitated by the uniqueness of the problem. Processes were researched by literature survey and evaluated by actual visiting (of the facilities where these were being utilized) by joint teams of representatives of Babcock & Wilcox Canada, AECL and the owners of the plants. In all cases, following the selection, further process of evaluation and proving out was undertaken. Majority of these undertakings were carried out by Babcock & Wilcox, however, AECL also performed independent and parallel engineering and developmental activities. In some cases parallel paths were taken in order to select more expedient or reliable processes.

As part of the proving out exercise all parameters of a new process were tested. Following the finalization of the parameters, familiarization was carried out. Then the process was verified in mock-ups which simulated site conditions. Simulation of site conditions was considered vital so that Babcock & Wilcox fabricated four mock-ups of the Steam Generator primary head. Three of these were sent to the plant sites and one was retained in the Cambridge facilities. Qualification samples, whenever applicable, were produced in the mock-ups incorporating conservative degree of difficulties to simulate site conditions. Qualification samples were submitted for approval and evaluated to agreed acceptance/rejection criteria.

In many cases, processes or process parameters had to be modified to suit the conditions. Extreme care was taken regarding field application even though the process activity or the apparatus had successfully undergone the qualification exercise. Initially limited approval was given in order to gain adequate sample size and confidence. Gradually limited approval was turned into production methods.

As noted earlier because of the uniqueness of the problem, developmental effort was necessary for both phases, ie. those of disassembly and rebuilding. Some of the key activities which required developmental effort are listed below.

1. One revolution tube cutter.

2. Seal weld removal, machine tool & cutter.

3. Tube stub removal method.

4. Hydraulic expansion of the tube in tubesheet hole.

5. Flush welding of the tubes to the tubesheet.

It should be noted that some of the processes incorporated in the in-situ retubing have been utilized and been proven. However, these were certainly first of a kind processes as far as CANDU steam generator design and manufacture are concerned. Hence their application coupled with the application under site condition, rather than shop environment, necessitated this degree of evaluation.

6. DEVELOPMENTAL EFFORT

As notes earlier, developmental efforts were necessary in a number of aspects of this project - including disassembly. New methods of manufacture (ie. the shroud was sectionalized) were established, as well as some new designs were incorporated eg. coned tube support plates, threaded tie rods etc. However, the most significant part of the development work was devoted to two items, namely tube to tubesheet welding and tube to tubesheet expanded joints. These two aspects will be described in some detail in this section.

6.1 TUBE TO TUBESHEET WELDING

The established requirement of CANDU Steam Generators with respect to tube to tubesheet welds is for a fillet weld having an average "Minimum Leak Path" (MLP) of 1.25t where 't' is the average wall thickness of the tubing. Figure 6 illustrates the fillet weld geometry. The fillet welds were produced in the works by an automatic TIG process in a downhand configuration with the axis of the steam generator vertical. During the initial stages of the retubing program it was decided that the effort required to develop fillet welds suitable for in-situ application would be substantial and inconsistent with the overall project schedule. An alternate weld geometry ie. flush weld was chosen because it was considered to be faster and less sensitive to variations in parameters such as hole diameter than fillet welding. Nevertheless, developmental effort to establish acceptable, reliable and repeatable flush tube to tubesheet welds was still significant.

Firstly, new welding machines were identified and bought, weld parameters and requirements were agreed to, followed by a lengthy process of proving out the welding machine, operators and the process by extensive block work. The target minimum MLP (minimum leak path) for the flush weld was established to be 0.8t. Following development welding in blocks, large qualification samples were produced. The qualification samples utilized identical tubesheet material (ie. of the same heat), tube samples from the actual tubing (ie. a mix from three tube suppliers had to be incorporated) and were performed in a mock-up of the primary head. The qualification samples were sectioned by Babcock & Wilcox, as well as by AECL and statistically analyzed to generate warning and control limits.

The target figure established for this kind of welds was one leak per 10,000 welds on the first helium leak test. This target was achieved, also the site quality control blocks consistently indicated that higher than required minimum leak path was being achieved. This aspect of the retubing program was the most difficult, risk-laden and controversial one but was accomplished with success.

6.2 HYDRAULIC EXPANSION

All CANDU Steam Generators traditionally employed two mechanically rolled joints between the tube and tubesheet. One was located above the tack expanded region approximately 50mm from the primary face of the tubesheet. The other joint was located at the secondary face of the tubesheet. Both joints were approximately 50mm in length. The mechanical rolling was controlled by limiting the degree of wall thinning and additionally received stress relief due to vessel heat treatment. As a result susceptibility to Stress Corrosion Cracking (SCC) was not a concern.

For the retubing program, the problem was to find a method of producing low residual stress tube to tubesheet joints without heat treatment, since in-situ heat treatment was impractical. The area of interest was the transition zone between the expanded and the unexpanded areas. The effort to find a way to expand was carried out in parallel with corrosion testing of stressed specimens and was successful.

The technique selected was hydraulic expansion because it produces low residual stresses as well as for the convenience afforded to for use in the restricted access of a steam generator primary head.

Once the process was selected, parallel efforts were launched by AECL and Babcock & Wilcox developing the tooling for two methods. The process utilized was that developed by AECL, although the bladder method of Babcock & Wilcox was utilized in a limited manner to correct non-conforming expansions.

Figures 7 and 8 illustrates the comparison between the hydraulically and mechanically expanded joints. A schematic diagram of the equipment used is shown in Figure 9.

7. CONCLUSIONS

Although the retubing exercise did not produce any scientific breakthrough, it demonstrated that in-situ retubing is a practical venture. In a radio-active environment the task would be far more difficult and costly and is judged impractical. Some design features of the rebuilt steam generators have been incorporated in the later design of steam generators. However, the major achievement was that of planning a major first-of-a-kind undertaking concurrently at three sites involving assembly and mobilization of people, material and new techniques to accomplish the job. The rebuilt steam generators at the three plants have been operating successfully at sustained high reactor power levels since 1983 when Point Lepreau went into commercial operation. The capacity factor for Point Lepreau for the year of 1984 was 89.74% and overall lifetime capacity factor for the same station has been recorded at 77.54%.

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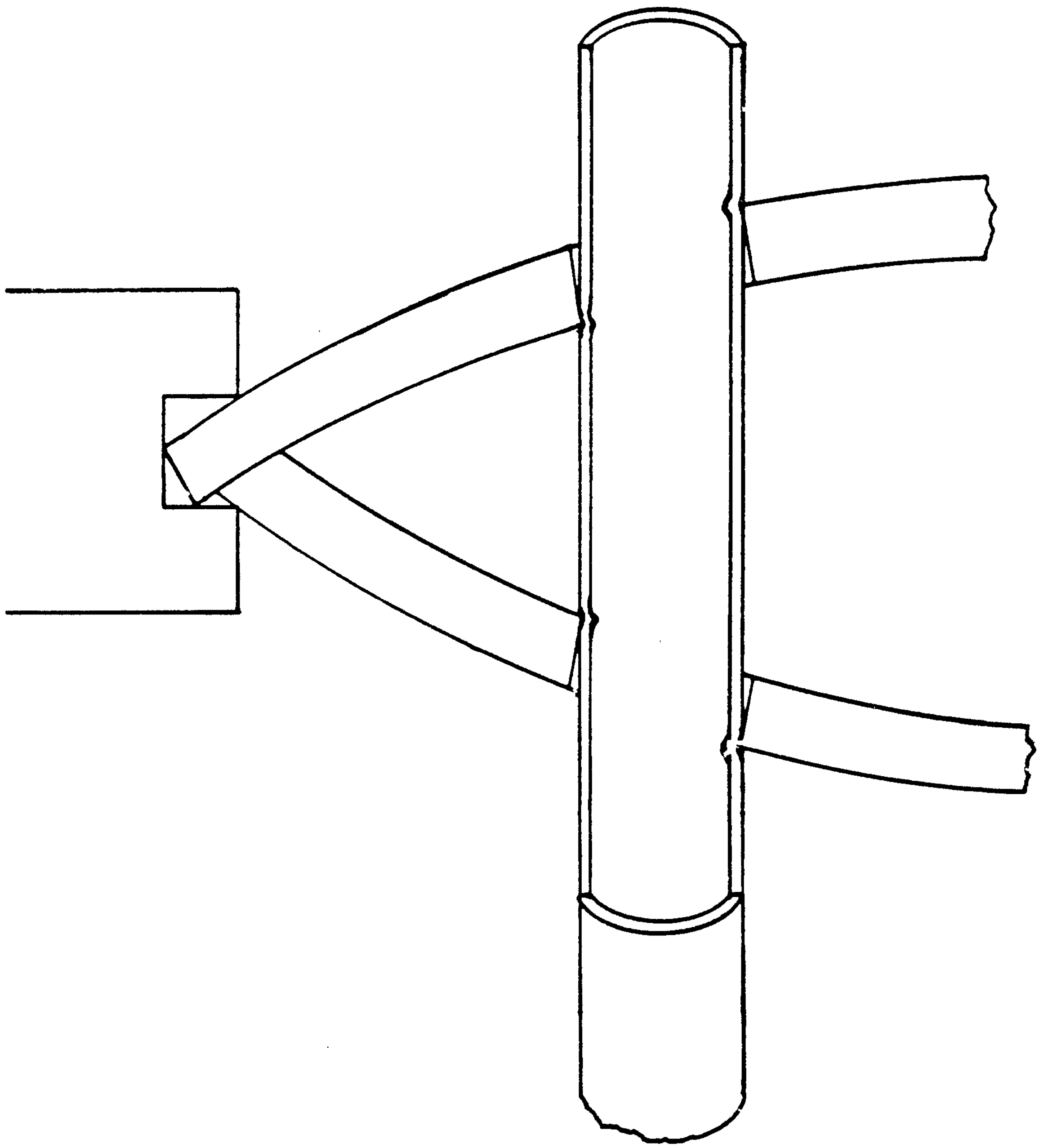


FIGURE 1. DINGING OF TUBE

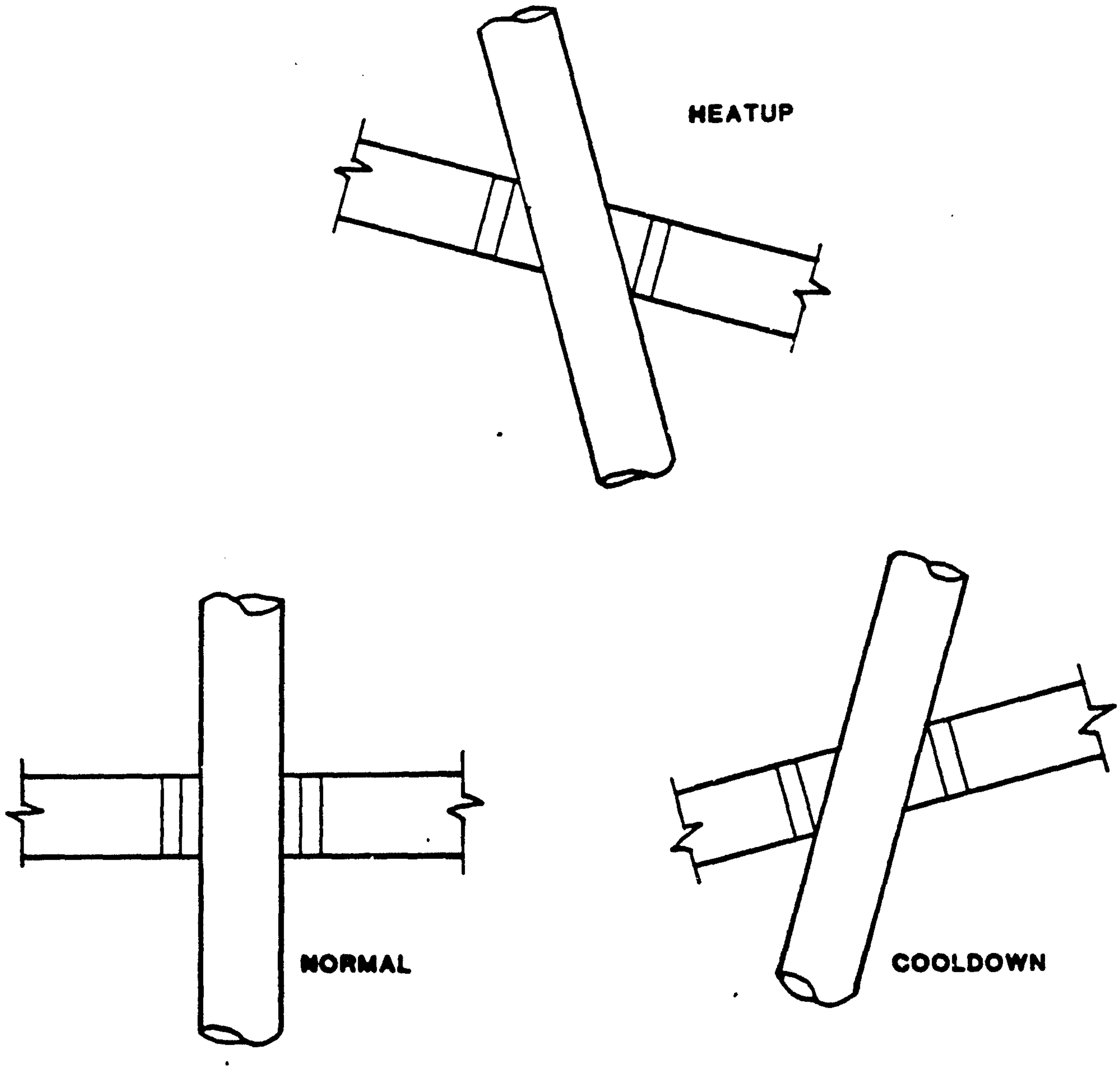


FIGURE 2 ROTATION OF SUPPORT PLATES

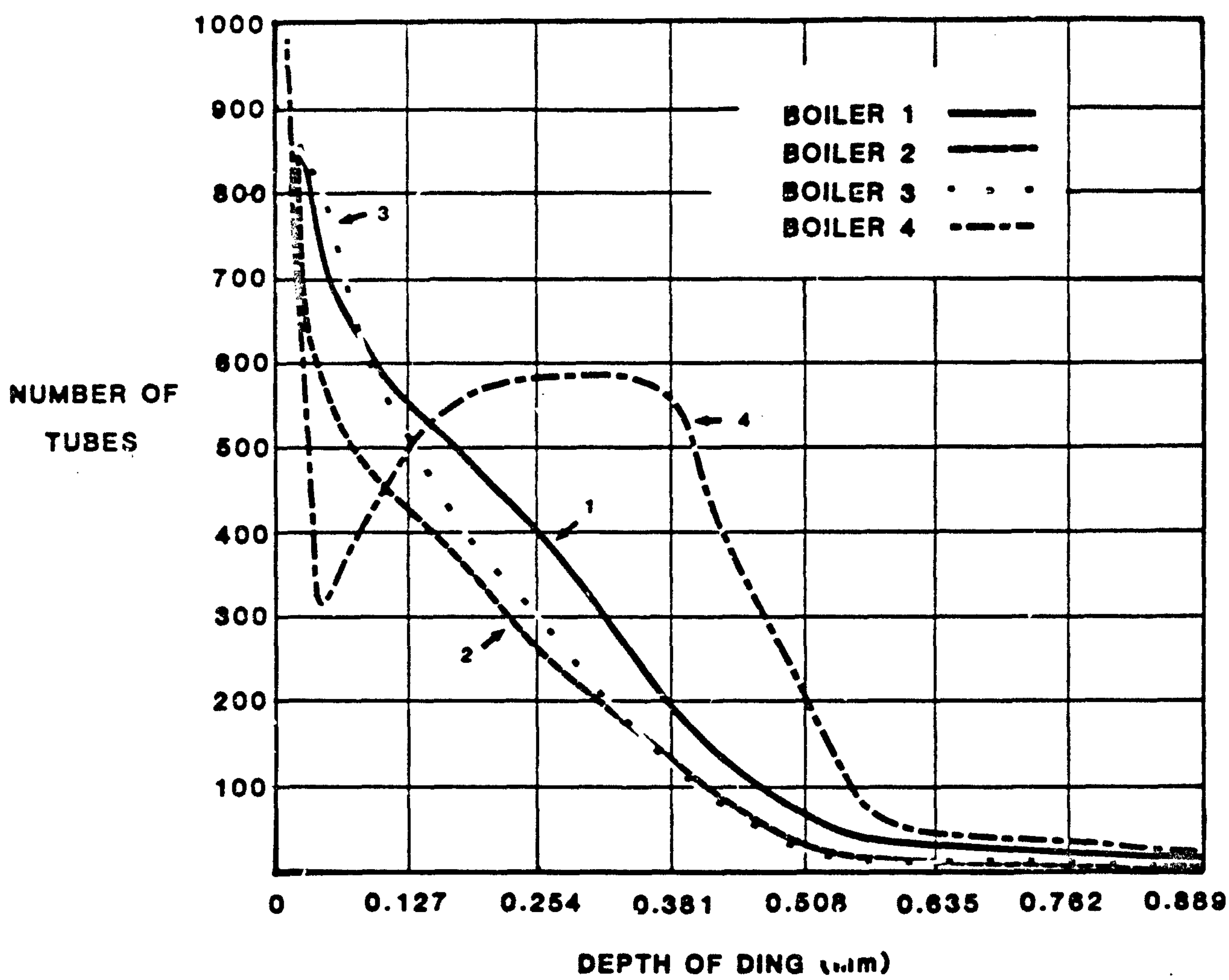


FIGURE 3 POINT LEPREAU DING DATA

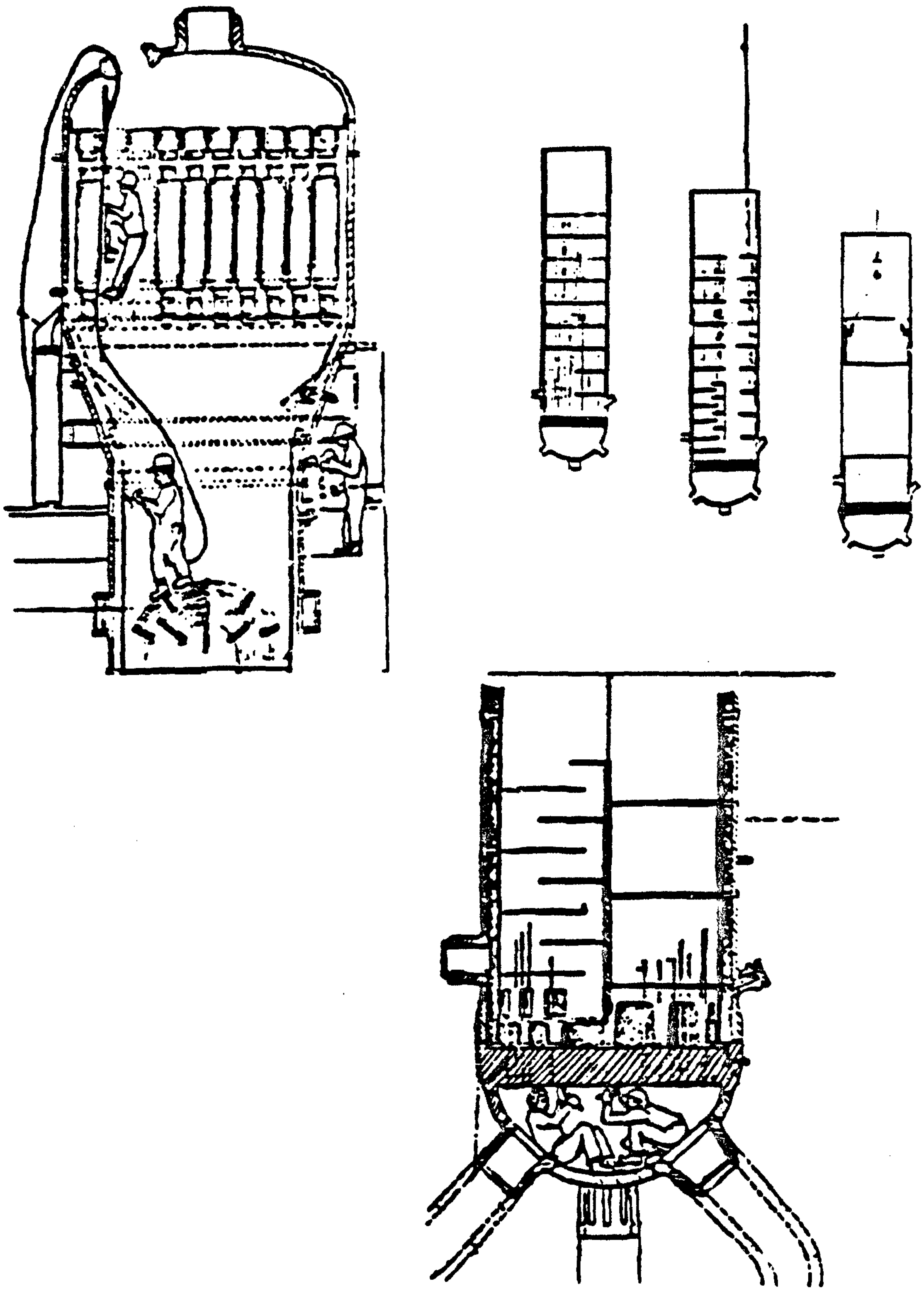


FIGURE 4 STEAM GENERATOR DISASSEMBLY

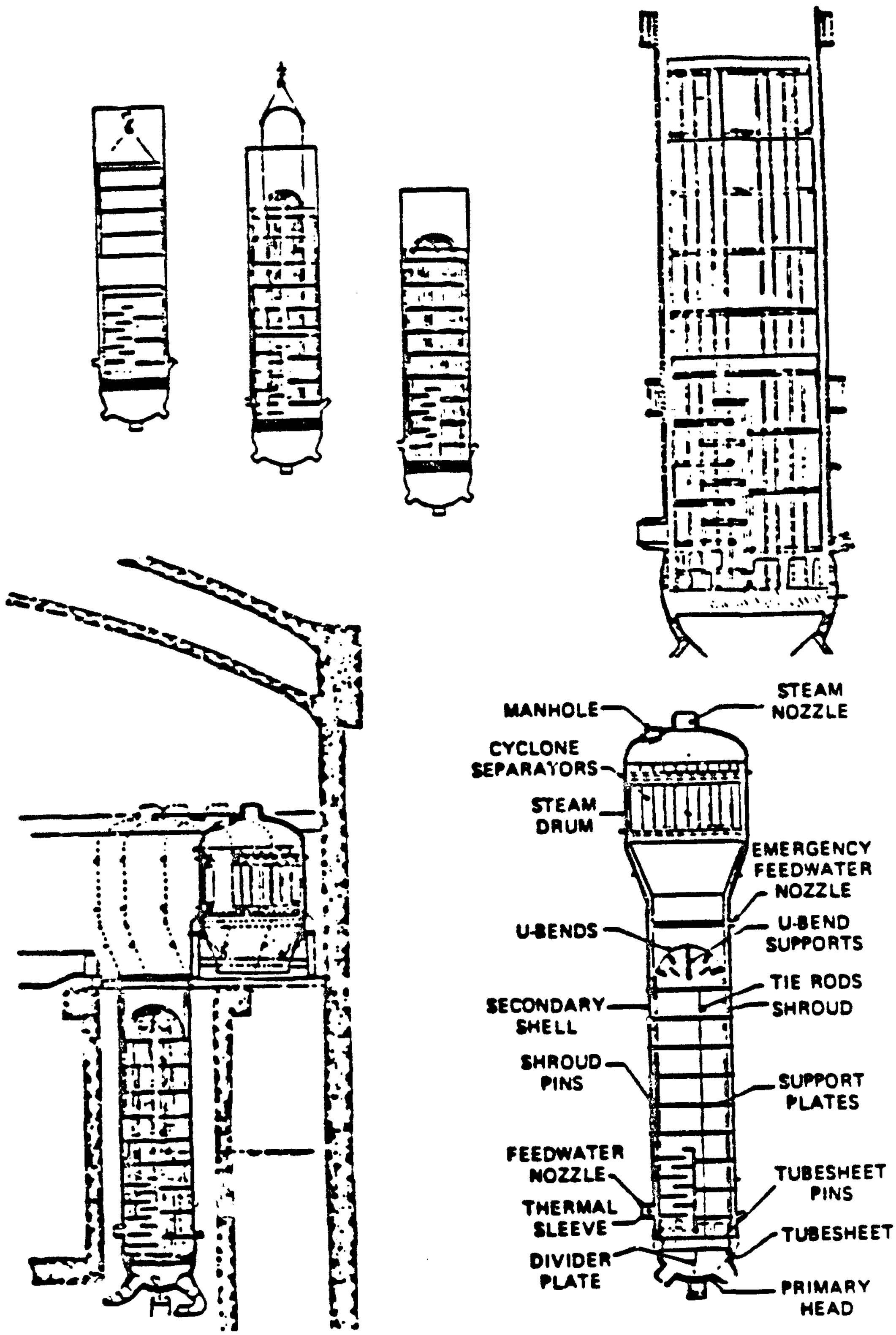
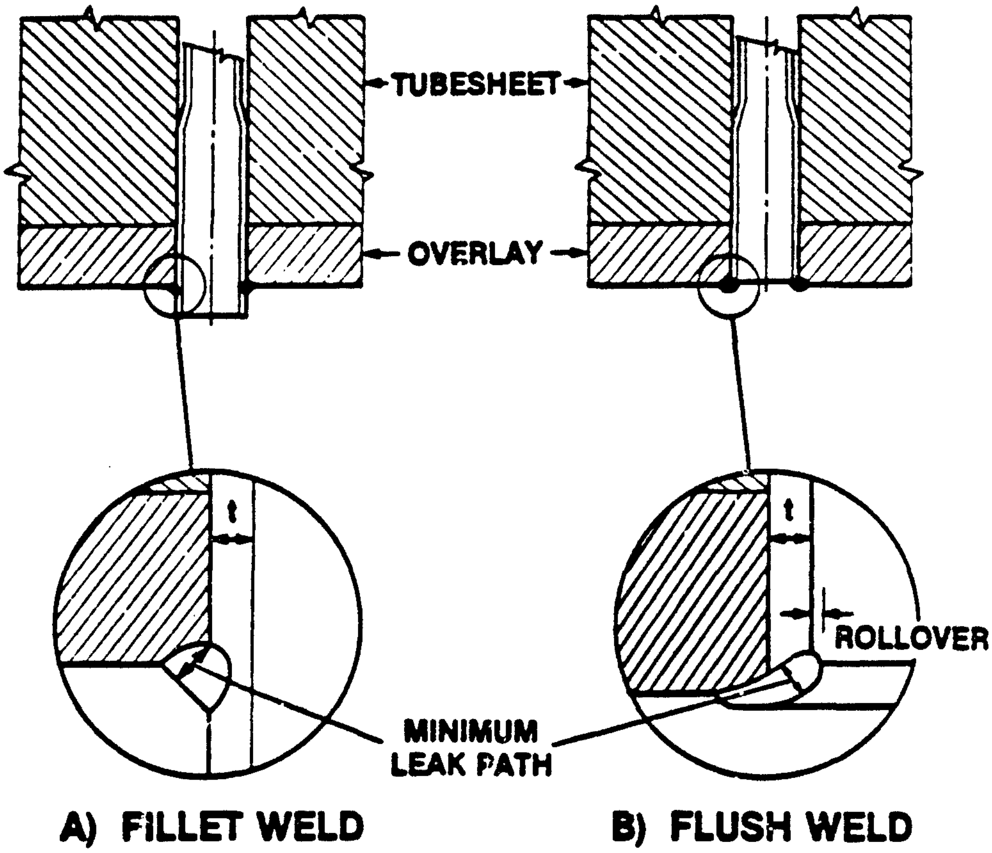


FIGURE 5 STEAM GENERATOR REBUILD SEQUENCE



WELD DETAILS

	FILLET WELD	FLUSH WELD
MIN. LEAK PATH	1.0t	0.8t
AVERAGE LEAK PATH	1.25t	0.95t
ROLLOVER	0.0	0.175-0.35 mm

FIGURE 6 COMPARISON OF WELD GEOMETRIES

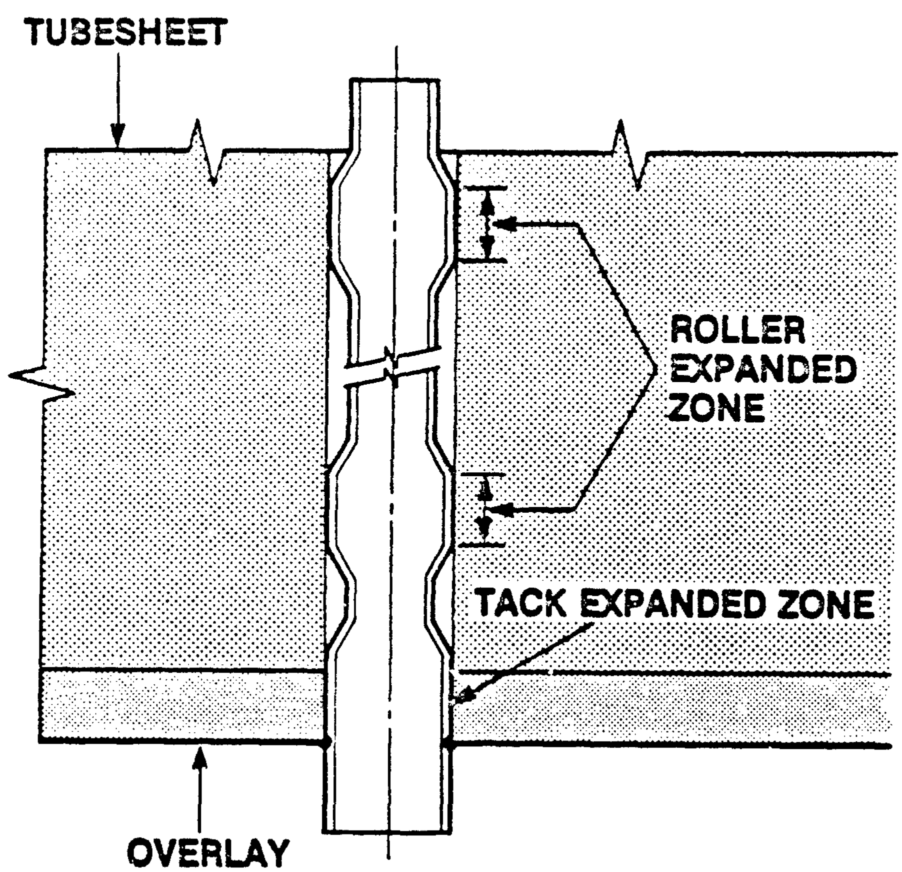


FIGURE 7 TUBE TO TUBESHEET JOINT DETAILS

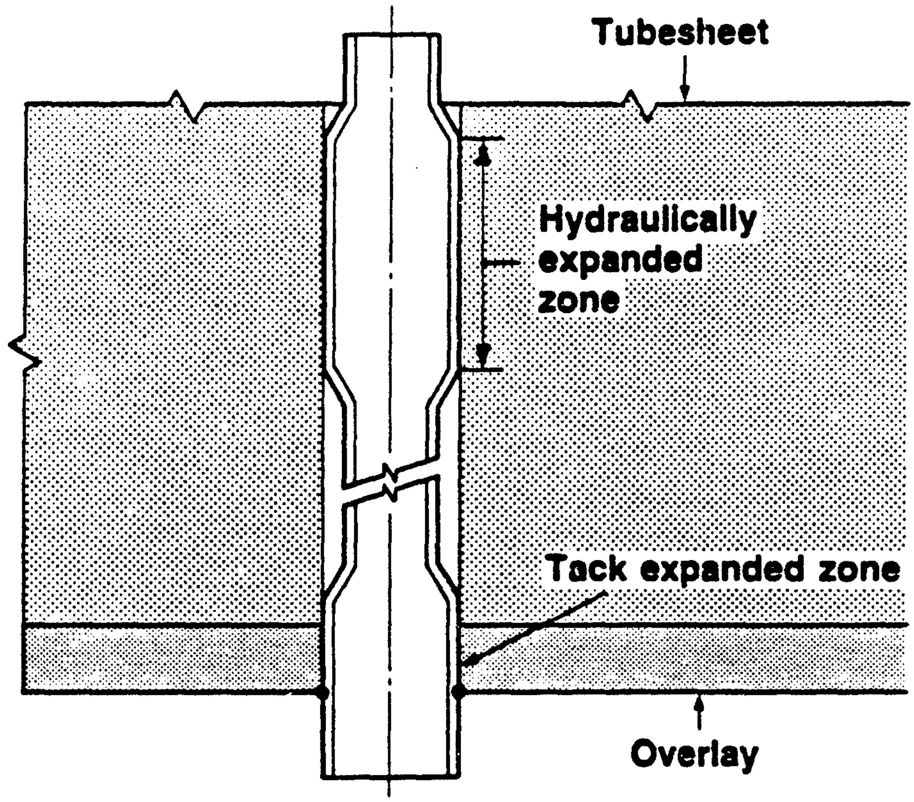


FIGURE 8 TUBE TO TUBESHEET JOINT DETAILS

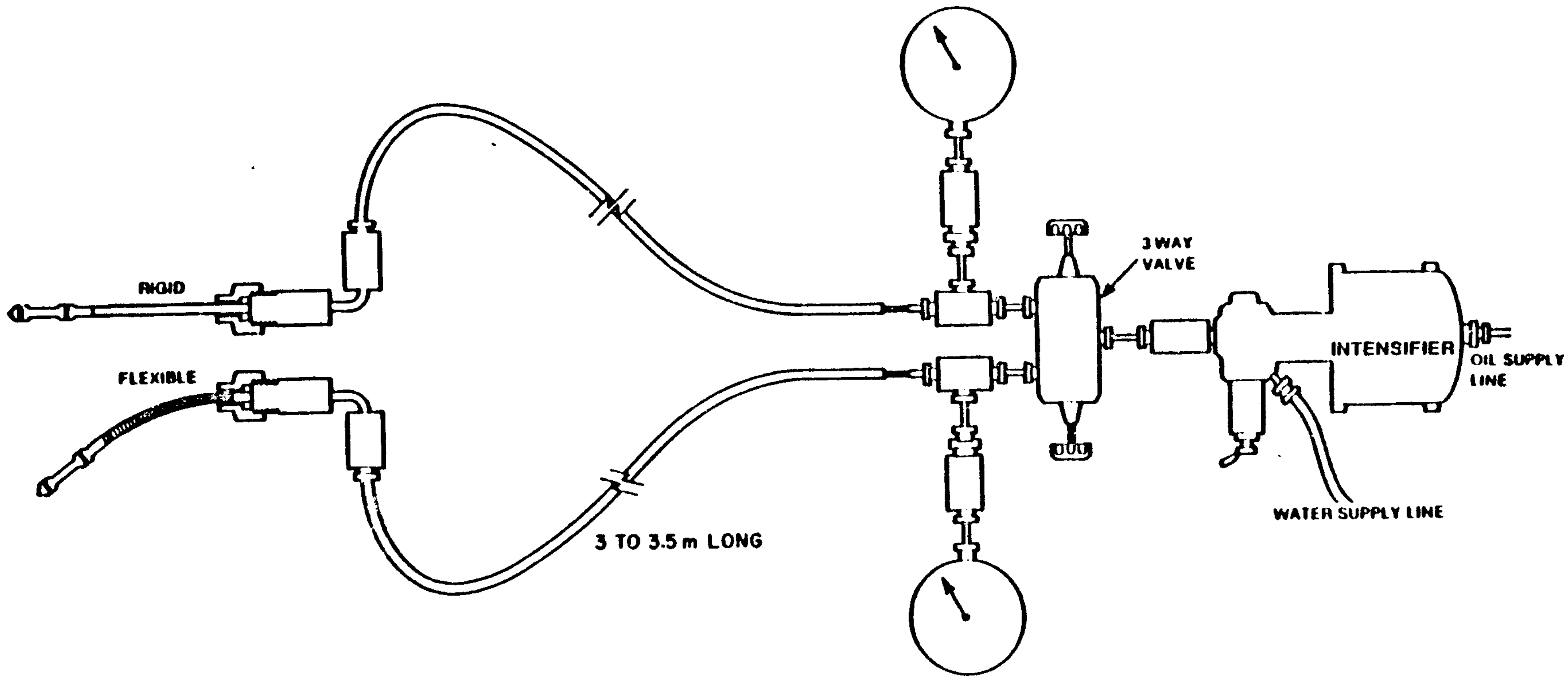


FIGURE 9 SCHEMATIC DIAGRAM OF HYDRAULIC EXPANSION EQUIPMENT

TEAM PERFORMANCE: HATCH -2

RECIRCULATION PIPE REPLACEMENT PROJECT

BY

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TEAM PERFORMANCE: HATCH - 2
RECIRCULATION PIPE REPLACEMENT PROJECT

By Alan W. Harrelson
Project Manager
Engineering & Construction Services
Georgia Power Company

Introduction

Edwin I. Hatch Nuclear Plant is a two-unit boiling water reactor (BWR) electric generating facility located in southeast Georgia. Each unit is a General Electric, BWR-4 design producing just over 800 megawatts of electrical energy. Commercial operating dates are 1975 and 1979 for units 1 and 2, respectively. In July, 1983, Georgia Power Company initiated actions to replace the reactor recirculation system piping in Unit 2 based on the results of an inservice inspection (ISI) conducted during the Spring, 1983, maintenance/refueling outage. Of the 118 welds inspected in the recirculation and residual heat removal (RHR) systems, 39 welds were observed to have reportable ultrasonic test indications. These test results were indicative of intergranular stress corrosion cracking (IGSCC) which was being observed in certain austenitic stainless steel components of BWR plants.

From July 1983 until January 1984; planning, training, procurement, and organizational efforts were underway in anticipation of the replacement work that was, subsequently, accomplished during the unit outage that began on January 13, 1984 and ended with the unit being returned to service on September 4, 1984. The actual pipe replacement work was accomplished by July 1984, within a year of the decision to replace. In order to meet the challenges, this endeavor presented, a special project organization was established and detailed action plans were developed which resulted in the smooth and satisfactory accomplishment of the work. The purpose of this paper is to identify some of the more important factors which contributed to this accomplishment.

Scope

The recirculation pipe replacement project (RPRP) scope of work included the replacement of all the reactor recirculation piping, the stainless steel portions of the RHR drywell piping and the reactor water cleanup (RWCU) piping up to the first isolation valve. Additional indications found during the replacement outage expanded the scope to include all the drywell portions of the RWCU piping. Due to the logistics of the management of major activities within the drywell, the project's scope was also expanded during the outage to include RHR checkvalve replacement (two valves), feedwater checkvalve replacement (one valve), recirculation pump repair, and the addition of scram discharge volume decontamination ports. However, this additional work was necessitated for various reasons other than IGSCC.

Removal and reinstallation of the piping and associated components, of which welding was a major activity, was not the primary focus of effort for the PPRP. The fact that the existing pipe and the working environment (drywell) were radiologically contaminated and very congested added significant complications to the job. The focus of attention in the planning stages and allocation of resources during implementation were essentially equally shared among the technical aspects of: 1. the pipe removal and replacement activities, 2. the radiological health physics activities, and 3. the interference removal and replacement activities. The magnitude, coordination efforts, and grueling pace of these major activities warn those who may be considering a project of this nature that "it is not just another piping job".

Project Organization

For several years, Georgia Power Company has used the project organization concept for selected, complex jobs; particularly, major construction jobs. Senior Georgia Power Company Management decided early in the planning phase to assemble a project team for the pipe replacement work. Important factors in this decision included the need to supplement normal plant resources to ensure continued orderly operation of the unaffected unit, to allow plant resources to perform normal outage preventive and corrective maintenance, and to establish a pipe replacement management team onsite that could focus exclusively on the project tasks. The project team was made up of proven performers from various disciplines with particular emphasis on selecting several key managers with construction experience. The project organization was directed by Georgia Power Company managers and included divisions of Engineering, Quality Control, Construction, Radiological Controls, and Procurement. (See Figure 1.) From the outset, a team concept was strongly promoted between plant and project personnel and among all project disciplines. This concept was enhanced by locating all project disciplines in one building. Open and easily accessible lines of communication within the project created a healthy environment for solving small problems early and essentially eliminating "surprises" so often caused by poor communications.

The complexities of a project of this magnitude involved the support of six principle organizations. These organizations and their "team" functions were as follows:

Georgia Power Company	-	Overall project coordination, QA/QC surveillance, program review, procedure approval, catalyst for action, glue that held the team together.

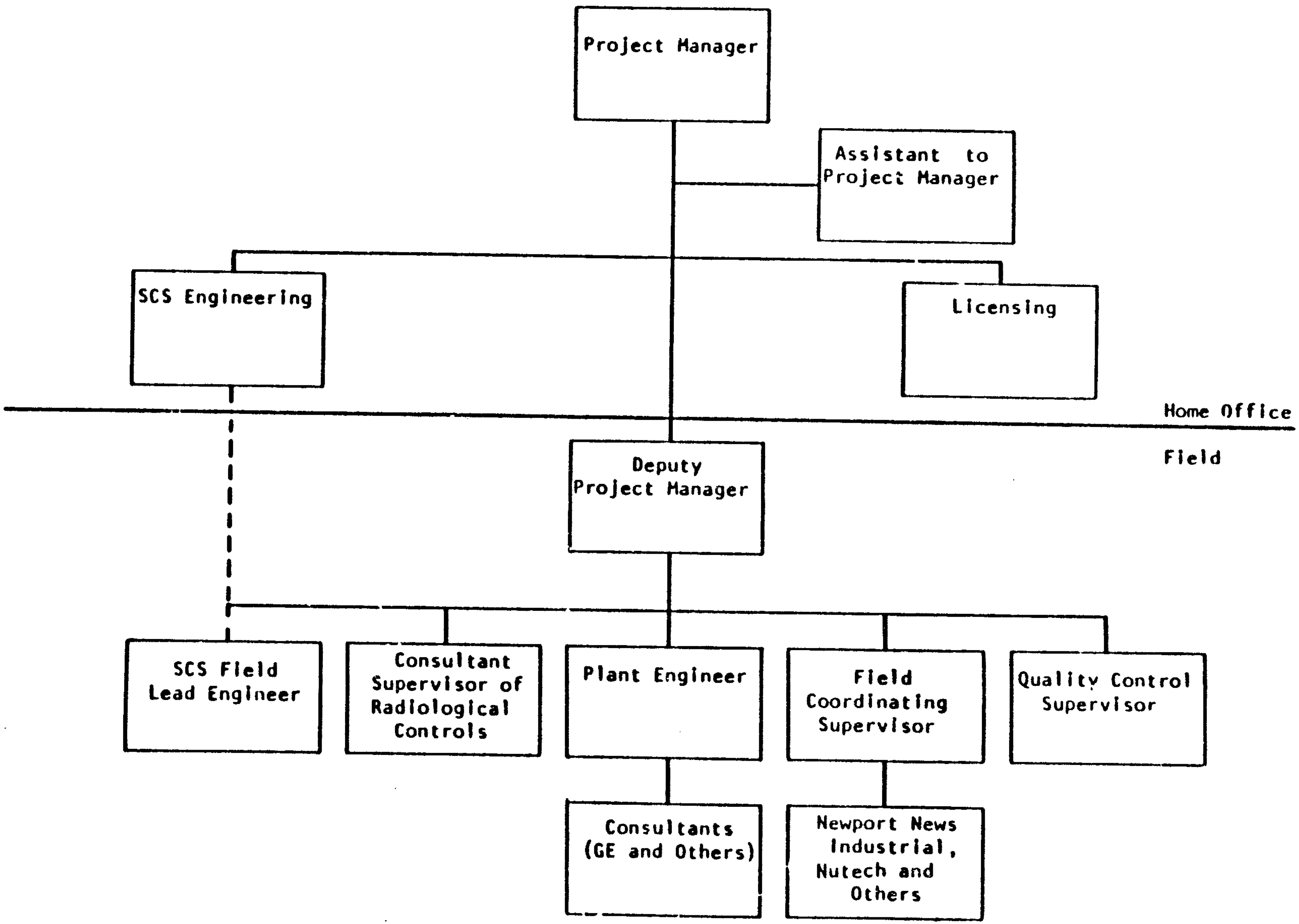

FIGURE 1

PROJECT MANAGEMENT ORGANIZATION CHART

Southern Company Services	-	Lead engineering agency with particular emphasis on interference removal and replacement procedures and control documentation.
Newport News Industrial	-	Primary replacement contractor, developed and implemented primary QA/QC program and detailed work instructions.
Hydro Nuclear Services	-	Developed and implemented project health physics program and program to maintain personal radiation exposures as low as reasonably achievable (ALARA).
Nutech	-	Inductive heat stress improvement (IHSI) contractor.
General Electric Company	-	Supplied new pipe design and new pipe. Lead advisor in vessel layup and reassembly, and unit start-up program.

Program and Procedures Development

The driving force during the planning process, August through December, was a desire to try to eliminate unforeseen circumstances that could delay the project or compromise quality. The first significant step in this direction was a working/planning session attended by key personnel from the major project entities to jointly review, in detail, the work plans of each group. These plans had been developed somewhat independently due to different organizational structure and responsibility assignments. This session was the first serious attempt to put the various planning "pieces" together to form a total project plan. The details and energies consumed in this session were enormous--meeting for 12 to 14 hours each day for four consecutive days. However, we emerged with three very important tools with which to tackle the task ahead.

1. A project plan that was supported by individual sub-plans that did not contradict each other or cause inefficient use of time or manpower.

2. Project personnel that more clearly understood the overall plan and the part each group <u>had</u> to play to support the whole.

3. Since the project plan was a product of a team effort, a cooperative working relationship had begun to develop. Later, during the outage, when unforeseen obstacles would be encountered, the fact that this working relationship and cooperative spirit was already on stable ground sometimes proved to be our only weapon or resemblance of preparedness, but that also proved to be enough to conquer the obstacles.

The potential for problems in the interface between plant administrative, maintenance, and operating procedures and superimposed project or contractor procedures are well-known. As a result, early concentrated efforts were applied to minimize interface weaknesses. Specifically, in-depth reviews of contractor, subcontractor, and project procedures were conducted by project engineering, quality control, and management personnel to ensure compatibility as well as technical accuracy. In addition, the plant Quality Assurance Department participated in an up-front prevention-oriented role in procedure review, enhancement and approval rather than relying on the audit function to catch weaknesses.

During initial planning and goal setting discussions, the need was identified for several special policies and procedures to ensure a smooth interface and clear understanding of project tasks. The following were prepared and implemented prior to the outage:

- A Function, Assignments, and Responsibilities Manual was prepared to identify the overall project organization, specific lines of authority, responsibility, and reporting relationships, and other important interface relationships.

- A detailed Radiological Protection Plan was developed to formalize overall radiological policies and goals including those associated with control of internal and external exposure, control of radioactive contamination and other ALARA factors.

- Comprehensive project control procedures were developed and issued covering the identification, removal, handling, storage, and reinstallation of materials and equipment which had to be removed to eliminate interferences with pipe replacement work.

- Based on a probing review by project and senior Georgia Power Company management, detailed plans were defined for the procurement of new piping materials including project management hold points, prime contractor and project surveillance coverage of fabrication processes and quality assurance audits.

- Technical details covering welding techniques and equipment, decontamination processes, shielding, mockup testing and ALARA were defined then hardware and software were specifically engineered for the project. Safety, cleanliness, and training also received detailed and specific attention.

- Quality Assurance and Quality Control personnel were directly involved in all project activities from procedure development to vendor surveillance. Early on, this part of the project team had a positive, prevention-oriented, questioning attitude.

- Radiological protection plans were developed not only to protect personnel and reduce exposure, but to minimize the total number of workers. Radiological personnel were directly involved with the pipe installer in work planning and process control activities.

- Health physics personnel planned their work with the attitude that if the field work was delayed because of health physics, then, they did not do their job properly.

A key factor in the development, review and approval of project policies and procedures was the direct and detailed involvement of project management personnel. Such participation not only visibly demonstrated their commitment but also essentially eliminated the "we/them" syndrome which can occur among organizational lines of responsibility.

Technical Details

Too often, policies, plans, and goals are established without the commensurate engineering and management follow-up to ensure their effective implementation. Understanding this principle, Georgia Power Company set an early course to define and engineer the specific techniques necessary for a successful project. Some of the more important actions involved the following:

- Management personnel visited another utility which had performed similar work to evaluate their results and apply lessons learned to the Hatch project.

- Special assessments were made of previous pipe cutting and removal techniques which resulted in choosing mechanical cutting rather than plasma-arc cutting to prevent unsatisfactory radiological conditions which could ultimately disrupt work in progress and vastly expand cleanup work.

- The piping layout design was modified to reduce the number of welds. Consequently, one piece components; e.g., riser piping, were used.

- GE type 316 Nuclear Grade (NG) material not exceeding 0.02 weight percent carbon was used for replacement piping and NG CF3 material was used for the replacement flow elements.

- Gas-Tungsten-Arc automatic welding was accomplished using Type 308L or 316L filler metal and consumable inserts for the 316NG to 316NG welds. Filler metal and insert material were 0.020 percent maximum carbon content by weight.

- Weld preparation configuration was a 22 1/2° $\pm$ 2 1/2° straight bevel to facilitate rapid weld-out and minimize susceptibility of insert remelt during subsequent weld passes. Heat input was controlled to a maximum of 50,000 joules per inch.

- A number of decontamination alternatives were reviewed considering all important factors such as the potential radiation exposure reductions, the low initial radiation levels of the unit, the possibility of material compatibility problems and the potential for process control problems. A water hydrolaze process was engineered for Hatch and executed successfully requiring only 1 1/2 days of critical path schedule time.

- Significant efforts were put forth to reduce radiation exposures in order to reduce the total number of workers necessary, concentrate craft and supervisory expertise and minimize the potential for exposure incidents. In addition to decontamination, many engineered factors were employed including innovative temporary shielding actions, equipment checkout, mockup testing and extensive training.

- In order to enhance quality, productivity, communications and the control of work, significant efforts were made to reduce the need for respiratory protection to a bare minimum. To achieve this goal, airborne radioactive contamination was minimized through the use of both large and local ventilation exhaust systems in the drywell, local contamination containments, the use of mechanical cutting techniques, conservative air monitoring practices, coupled with prompt actions at the first signs of developing problems.

- The entire process of worker ingress and egress from the work areas was examined and engineered to minimize non-productive time while at the same time maintaining all necessary controls. Special features employed included a computer controlled dosimetry system with multi-terminals at the pipe replacement access point resulting in individual worker processing time that took less than one minute. In addition, special state-of-the-art art personnel whole body monitoring devices were demonstrated to be technically satisfactory which reduced personnel monitoring times by a factor of 10. The benefits of these actions became clear considering that more than 65,000 entries were made into the drywell during the pipe replacement work.

- The project Quality Control Division was staffed with specialists not only in the areas of piping activities and nondestructive testing but also in areas of the electrical and civil disciplines to provide the necessary expertise in the interference removal and reinstallation work.

- Early involvement by project Quality Control personnel in the oversight and surveillance of pipe fabrication significantly reduced problems and questions when materials arrived on site.

Management Involvement

It is difficult to overstate the importance of management involvement in tackling a complex job such as replacement of recirculation piping. However, there is a danger that "management involvement" becomes a platitude if implementation directions only take the form of "brief me weekly", "prepare monthly reports" and "call me if something goes wrong". In contrast, the direct and detailed involvement of Georgia Power Company project, plant and corporate managers in many aspects of planning, material procurement, training, quality, radiological controls and work execution appears to have had a significant, positive effect on the results achieved at Hatch. Most of the actions previously discussed included direct input and actions by management. Several additional areas which should be mentioned include:

- An ALAPA Oversight Committee was established during the early planning phase which consisted of senior management from the plant, project and corporate headquarters. This committee was not proforma; rather the committee took a probing and hard-hitting approach towards establishing challenging goals, examining specific plans and following up on the results.

- Project management demonstrated a strong and continuing commitment towards quality which took the form of daily follow-up on deficiency reports and audit findings until they were properly resolved, prompt actions to factor lessons learned from problems into the work process and a continued vigilance to ensure that the various players and organizations worked as a team.

- Key project management personnel were continually involved in monitoring and controlling the "pulse" of the project. The manifestation of this commitment took the form of project management participation with craft supervision in shift turnover meetings twice daily at 5:30 AM and 5:30 PM, daily staff meetings with participation from all project organizations, and a project status/information board that was updated by management at the beginning of each shift to keep all project personnel, especially the craft personnel, posted on the progress of the preceding shift, the work scheduled for the present shift and other activities that affected the project.

Key Performance Indicators

In order to properly manage any project, it is necessary to have a means of measuring performance and then judging its merit. It is, therefore, mandatory to establish, from the outset, standards of good performance in all the pertinent aspects of the project. By periodically comparing actual performance against these established goals, more effective mid-course adjustments can be made. The key performance indicators and the associated results from the Hatch Unit 2 FPRP are listed as follows:

1. **Schedule** — **Actual performance within 3% of schedule.**

   The FPRP activities were outage critical path from the lowering of the water level in the recirculation loop piping until the last weld. The total time allotted for this continuous path of activities on the original Project Schedule was 111 days. The actual time consumed for these activities was 114. The actual schedule compares very favorably with the original schedule, especially when considering that the RHR checkvalve replacement, feedwater checkvalve replacement, scram discharge volume modifications, pump flow splitter inspection and removal, isolation valve operator replacement, and additional RWCU line replacement were added to the Project Scope during this critical path time frame.

CHRONOLOGY OF EVENTS

July 19, 1983	-	Project Manager assigned
July 29, 1983	-	Pipe material verbally awarded to GE
August 9, 1983	-	Installation contract awarded to NNI
August 16, 1983	-	NNI on site
January 13, 1984	-	Outage Began
January 29, 1984	-	Pipe delivery complete
February 10, 1984	-	Vessel layup, pipe decontamination, and first pipe cut complete.
March 4, 1984	-	Grain boundary microfissuring discovered in new induction bent 12" risers. GE began fabricating new risers immediately.
June 1, 1984	-	Last pipe weld
July 15, 1984	-	IHSI complete
September 4, 1984	-	Unit returned to service (on-line)

2.	Budget:	Actual within 5% of original budget.
3.	Weld quality:	Reject rate for completed pipe welds was less than 7%; the goal was to be less than 10%. NRC inspector praised GPC for the radiograph quality.
4.	Radiological controls:	Respirators used less than 10% of the time; goal was no more than 10%.
		Less than 2% of the total manhours were lost due to unforeseen radiological problems; goal was 0%.
		Radioactive contamination levels in drywell were lower at the completion of work than at the start which was the project goal.
		Original personnel exposure "budget" was 1700 man-rem; actual was 865 man-rem.
5.	IHSI performance:	The goal was six joints per day average. The actual performance average was better than six joints per day. The previous industry single-day record of seven joints was also bettered with eight and eleven joints per day performances.
6.	Safety:	The lost time incident rate for project personnel, including craft, was 0.50. The industry average rate for specialty work of this nature was 6.6.

Recommendations and Conclusions

For those who may be considering a recirculation pipe replacement, the following "lessons learned" are offered:

- Seriously consider modification of the piping layout design to reduce the number of welds, thereby, resulting in lower, future ISI and IHSI costs and accompanying personnel radiation radiation exposure;

- Where possible, use seamless pipe to minimize ISI requirements on seam welds, thus, further lowering future ISI costs and personnel radiation exposure;

- Carefully follow material from melt through final forming product to verify that the desired chemical and mechanical properties are provided;

- Although nuclear grade replacement material is utilized, consider the application of a residual stress improvement measure such as IHSI. Added margin to preclude IGSCC and the possibility of Nuclear Regulatory Commission (NRC) imposed augmented inspections are the benefits achieved;

- Replacement of the recirculation piping affords the "rare" opportunity to inspect components not normally accessible (i.e., recirculation pump internals, valves). Utilize this opportunity to inspect these components;

- Use the outage opportunity to replace susceptible piping even if found to be free of IGSCC to reduce any future augmented ISI requirements and possible replacement at a later date if cracks were to be found;

- Carefully review the replacement program early for any possible safety questions and submit license amendment(s) promptly for NRC review and approval. (NRC Generic Letter 84-07 should be consulted to determine the need for licensing amendment(s) in support of the piping replacement);

- Consider implementing a whip restraint or snubber reduction program, if applicable, during the pipe replacement job. This can significantly simplify reinstallation.

- To minimize replacement piping radiation build-up, consideration should be given to passivation, mechanical polishing, or electropolishing.

- The magnitude of undertaking a recirculation pipe replacement project and potential for future problems warrants that this project be given special consideration and support by management in organizing and implementing the step-by-step tasks required for successful completion. <u>It is not just another piping job.</u>

Few will be surprised at the concepts and techniques involved in the Hatch recirculation pipe replacement work. There are no tricks or easy answers that make things turn out right. Finding the time and means to engineer, implement, and follow-up on detailed action plans that are based on meaningful, specific and challenging goals is urgent. A clear understanding that "the devil is in the details" is a prerequisite in eliminating unforeseen circumstances and getting the job done.

OVERVIEW OF WELD OVERLAY FOR REPAIR OF CORROSION CRACKING IN STAINLESS STEEL PIPING WELDS

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Introduction

Weld overlay has been widely used since 1982 as a means of repairing stress corrosion cracking in welded stainless steel piping systems of boiling water reactors. Well over 900 incidents of intergranular stress corrosion cracking (IGSCC) have been reported in association with welds of the recirculation system and other BWR piping systems containing oxygenated water [1]. Many of these cracks were evaluated and found to be small enough to require no repair prior to the next scheduled in-service inspection. Others were repaired by application of weld overlay, and still others were repaired by replacing lines or portions of lines with crack-resistant materials.

The weld overlays first applied to IGSCC repair were intended to be expedient temporary repair measures, and design emphasis was placed primarily on restoration of structural integrity in the cracked joint, so that the plant could be returned to service while permanent repairs were planned and implemented. Experience with weld overlay design and application led to refinements, which were intended to increase the expected service life of the overlay or to reduce its cost in terms of installation time and occupational radiation exposure. Tests were conducted, and are still in progress, to confirm the expected long-term service performance of the overlay repair.

This overview paper discusses the current status of weld overlay repair technology, with emphasis on issues affecting suitability of weld overlays for long-term service as an alternative to replacement of lines. Because the cost of replacement is high, less expensive alternatives such as the overlay repair must be evaluated by plant owners. At this writing, owners of eight plants have elected to replace piping systems. Owners of nine plants have utilized a total of over 150 weld overlay repairs and have announced no plans to replace these lines.

Description of Weld Overlay Repair

Cracks requiring repair are located in heat-affected zones of circumferential welds in stainless steel pipe. The overlay provides needed structural reinforcement where part-through circumferential cracks are present, and it prevents leakage from short, through-wall axial cracks that are less frequently encountered. Circumferential weld beads are applied in several layers, providing a buildup typically 1/4- to 1/2-inch thick and 2 to 8 inches in axial length. Figure 1 shows the essential features of a weld overlay.

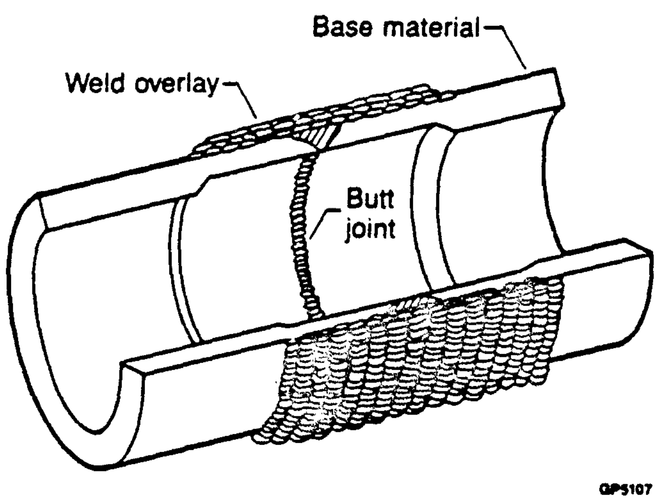

Figure 1. Weld overlay repair technique for stress corrosion cracking.

Figure 2 shows a laboratory application, using field-qualified equipment, of weld overlay to a 24-inch pipe. All weld overlay repairs still in service were made with an automatic, pulsed, gas tungsten arc welding (GTAW-P-AU) process because this equipment is suitable for remote operations. Early applications used off-the-shelf equipment. Later, equipment modification permitted wide overlays (up to 10") and automatic indexing. Addition of optical systems supported completely remote Code-quality field welding operations. Fixtures have also been designed for automatic overlay of sweepolet saddle welds [2].

A low-carbon weld deposit with sufficient ferrite content is highly resistant to corrosion cracking and presents a barrier to further IGSCC. In current practice, Type 308L stainless steel filler with a ferrite content of 11-17 FN is used to achieve ferrite levels of 8 FN or above in as-deposited material [2]. The first weld layer may be applied with low heat to minimize dilution, but higher heat input (typically 25kJ/in or more) is used for interim layers to increase the deposition rate. Internal water cooling has been used to control interpass temperatures. ID cooling also aids in producing a residual stress distribution which tends to retard further IGSCC.

Structural Design

A few weld overlay repairs were designed to reinforce the pipe wall with weld buildup of equivalent strength or thickness. Unnecessarily thick overlays are costly and appeared to produce excessive shrinkage and distortion. Later it was shown that thinner overlays produced a more favorable residual stress distribution. The structural design objective of the more recent overlay repairs is that the weld metal, or the combination of the pipe and the weld metal, will accommodate applied loads with design margins at least as great as applicable ASME Code design margins for the piping system. Since the pipe wall is thicker than necessary to sustain actual or expected loads, the weld metal thickness need be only a fraction (typically less than

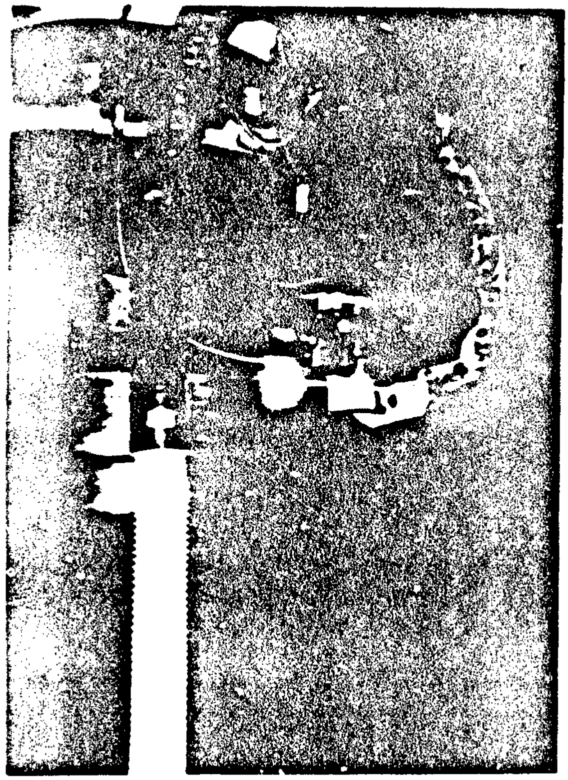

Figure 2. Overlay repair application using automatic welding equipment.

half) of the pipe wall thickness to meet this design objective [2].

The remaining strength of the cracked pipe has been utilized in some overlay designs but not in others. The so-called Type I design does not depend on the remaining strength of the pipe.

The Type II design utilizes the strength of the pipe only in portions of the circumference where no crack has been detected. Type I and Type II designs have been referred to as "full structural overlays."

The Type III design utilizes the remaining strength of the pipe as determined by crack length and depth measurements. Allowances have been made for measurement uncertainty and for crack growth during the operating interval. Obviously, the overlay reinforcement thickness will be greatest for the Type I overlay, and least for the Type III overlay. In practice, Type III overlays are used to prevent leakage from short axial cracks and to improve the residual stress distribution in the pipe where the extent of cracking does not require added reinforcement.

Reinforcement requirements for weld overlays are usually determined by applying rules of ASME Section XI, Par. IWB 3640 for flaw assessment to the reinforced weldment. Those requirements are intended to assure that the minimum structural design margins of construction codes are maintained in a pipe containing a flaw.

Current IWB 3640 requirements are applicable to austenitic stainless steels having very high fracture toughness, so that ductile collapse can be shown to preclude fracture. Some piping welds were produced by a high-heat subarc process for which the weld metal fracture toughness may not be high enough to be nonlimiting in every case. An alternative method for application of IWB 3640 rules has been proposed to cover these cases [3].

Corrosion Cracking Resistance of Weld Metal

Full structural overlays are designed to present a crack-resistant barrier to corrosion cracks which reach the fusion line beneath the overlay. Control of ferrite levels as discussed previously, and the use of .02% max carbon filler material, produce a weld clad which tests show to be highly resistant to cracking [4]. This has been confirmed by a test of 4-inch Type-304 stainless steel pipe, in which a through-wall IGSCC crack was overlaid and returned to 1000 hours of exposure in an accelerated service simulation test. Subsequent metallography (Figures 3 and 4) showed no penetration of the crack tip into the overlay [1]. Other results from this test series are discussed below.

Service experience has shown that crack-resistant materials may, nevertheless, crack when exposed to an occluded crevice environment and a high stress. A repaired crack may behave like a crevice in accumulating impurities that promote cracking. However, long-term laboratory tests indicate that high ferrite weld metal is more resistant than a variety of other materials to this exposure [5]. A fatigue precrack in weld metal did not extend under conditions of high stress and long-term exposure, which did cause cracking in low-carbon and niobium-stabilized grades of wrought stainless steel. In view of this, periodic in-service inspection of weld overlay repairs appears to be an adequate precaution against cracking of the weld metal.

Weld Shrinkage and Residual Stress

Shrinkage and distortion caused by weld overlay may have both beneficial and undesirable effects. Some compromises are required to optimize the overlay design in this respect. The following general observations are supported by experimental investigations and analytical studies:

Figure 3. Photomicrograph of a through-wall IGSCC crack, showing arrest at weld overlay.

Figure 4. Macrophoto of the crack in Figure 3, showing butt weld and weld overlay.

a. The first layer or two of weld deposit produces diametral and axial shrinkage in the pipe wall, generating compressive residual stresses near the inner wall which tend to retard further cracking [6,7].

b. Additional layers of weld deposit cause further diametral shrinkage and local bending which tends to reverse the initially favorable axial compressive stress buildup [6,7,8]. Circumferential compression is retained.

c. Axial shrinkage at the weld repair induces stress elsewhere in the piping system [7,10]. Many overlay repairs in one line could produce undesirable stresses in the line, similar to poor fit-up stresses.

d. Increasing the axial length of overlays may reduce the local bending (item b above) and improve the residual stress distribution. However, the longer overlays may also produce more shrinkage and system discontinuity stresses.

e. Other variables which affect these trends are the pipe size, the counterbore design, and the weld heat input.

IGSCC usually occurs in association with yield-level tensile stress, and any substantial reduction in residual and applied stress is expected to greatly reduce the rate of continued cracking. An objective of IGSCC stress remedies, including weld overlay, is to introduce compressive residual stress near the weld root so that very high tensile stress is avoided at crack locations even when tensile service stresses are applied. Weld overlay designs have been developed which provide both the necessary reinforcement thickness and the desired residual stress distribution [7]. Experimental data have confirmed that the expected compressive stresses are achieved by several specific weld overlay applications that are representative of in-plant repairs [11,12]. However, another test program showed that low-level axial tension (8 ksi) remained after application of 8 layers (.875 in) of weld overlay to a 28-inch pipe [14]. Still another test showed that a local region of high tensile stress remained following a weld overlay application which produced compression elsewhere near the root of the weld in a 16-inch Schedule 80 pipe [12].

Although a compressive residual stress at the inside surface is not necessarily assured for every weld overlay repair application, particularly in large thick-walled pipe, continued uncontrolled IGSCC is not to be expected. Large pipe weldments normally have a characteristic residual stress distribution that becomes highly compressive a short distance from the inside surface, and IGSCC is slowed or arrested as a result even without weld overlay repair [16].

Evaluation of axial shrinkage and system effects (items c and d above) requires a design-specific analysis. Data are available [10] which indicate that shrinkage is typically about 0.1 to 0.2 inch for a 12-inch Schedule 80 pipe, and less than 0.1 inch for a 28-inch diameter pipe. It is useful to note that the shrinkage caused by full-penetration assembly welds is greater than that due to the overlay [4] but a large number of overlays could, nevertheless, produce significant stresses in the piping system.

Long-Term Testing of Overlay Repairs

Accelerated IGSCC tests are in progress to evaluate the long-term service performance of weld overlay repairs. Overlays representing a range of dimensional and weld process parameters have been applied to precracked 4-inch pipe, and these are being exposed to 288C water at stress levels and dissolved oxygen levels that are more severe than BWR service conditions [1]. The performance of repaired welds is being compared with that of precracked and unrepaired welds under the same environmental exposure.

Figure 5 shows a finished overlay on one of the precracked 4-inch pipes. The weld bead size was scaled down to produce a residual stress distribution similar to that in a 12-inch pipe repair. After accounting for test acceleration and for pipe size, a minimum plant service life is currently projected at 4.6 fuel cycles (about 7 years) for overlays on 12-inch pipes [13]. The projected service life is increasing as the tests progress without failures of the overlays.

A weld overlay has also been applied to a precracked 24-inch pipe and exposed to stress and environmental conditions similar to the 4-inch tests. After 3000 hours, preliminary inspection results indicate no detectable extension of initial 25% through-wall circumferential IGSCC. During this same period, surface crack extension and some new cracking did occur in an unrepaired reference weld, which had also been precracked to about 25% of wall thickness. Figure 6 shows the repaired weldment, the crack location, and the dimensions of the overlay, which is similar to that shown in Figure 1.

<u>In-Service Inspection of Weld Overlays</u>

Applicable ASME Code requirements for preservice and in-service inspection of weld overlays are discussed in Reference 10. Nondestructive examination (NDE) must show that the overlay material is free of unacceptable flaws. NDE may be required, depending on the overlay design basis, to detect or size the IGSCC defects in the base material.

A substantial effort has been applied to the development and qualification of specific ultrasonic testing (UT) techniques for detection and sizing of IGSCC in wrought stainless steel pipe. However, different UT methods may be preferred for inspection of weld overlay because the duplex, large-grain structure of the

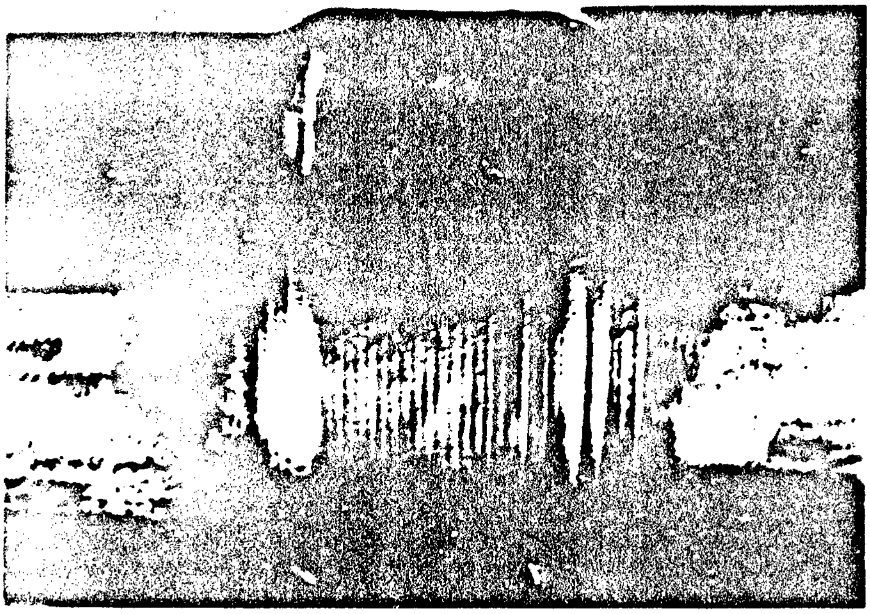

Figure 5. A finished weld overlay repair for laboratory testing.

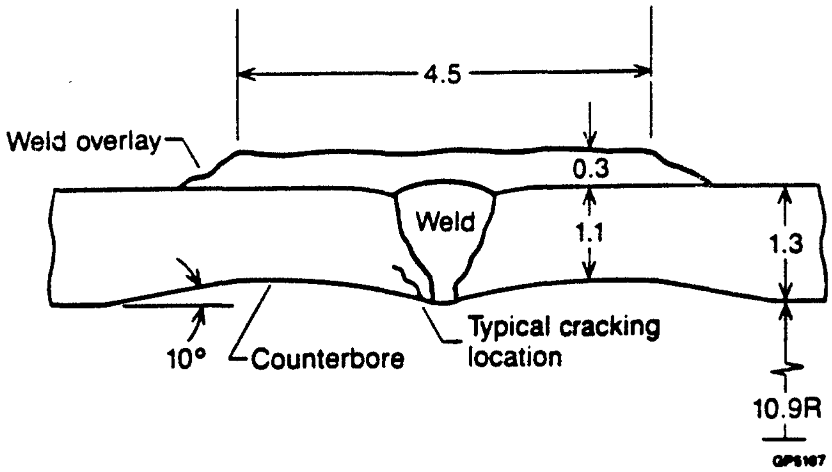

Figure 6. Weld overlay repair of a precracked 24-inch pipe.

Type 308 weld metal attenuates and scatters the ultrasound beam to a greater degree than does that wrought material [12,15]. Ultrasonic inspection of stainless steel clad PWR pressure vessels from the clad surface is a well-developed application that is similar in some respects to weld overlay inspection. In detecting underclad cracking, best results have been obtained with longitudinal-wave transducers, because longitudinal waves attenuate less rapidly in the weld metal than do the shear waves used in piping inspections [12]. Application of L-wave techniques to weld overlay inspection has shown that cracks beneath the overlay, which have reached the outer 25% of the original pipe wall, can be detected and sized [10].

The suitability of UT inspection for weld overlay repairs depends in part on the acceptance criteria that are established. If the overlay design basis does not depend on the depth and extent of cracking in the pipe (the Type I design), it is sufficient to establish by UT examination that the crack has not extended into the weld overlay. This is well within the capability of current

UT techniques. If the overlay design basis depends on the circumferential crack length (the Type II design), UT detection of the ends of the crack is required. When the adequacy of the repair depends on a limited crack depth as well as length, UT examination must confirm these crack dimensions. However, in-service inspection of many weld overlay repairs will not require crack depth sizing through the overlay.

Another approach to periodic UT examination of weld overlay repairs is change detection. In principle, a baseline signature can be established using a scanner, and subsequent crack growth can be identified by changes in the signature. Interpretation of signals in terms of crack size is not required if significant crack growth does not occur during the operating interval.

The weld overlay NDE development and the earlier RPV clad work showed that clad surface roughness controls the quality of the inspection. Smoothing and flattening of the weld overlay surface has been recommended to achieve good contact with the transducer and to increase the signal-to-noise ratio to acceptable levels [10,12].

Radiographic testing (RT) is another useful inspection method for which limited experience in this particular application exists. Application of RT to weld overlay inspection requires no fundamentally new techniques, but criteria are needed for interpretation of the results. RT is sensitive to the product of crack opening width and radial depth in this application [17]. A crack which is neither short nor tightly closed will be detectable by RT. A crack of moderate depth, as determined prior to the repair, may be tightly closed by an overlay process which produces compressive stresses in the inner wall of the pipe. This crack, representing the optimum condition for overlay repair, should not be detectable by RT. If the crack is open, due either to excessive depth or to the absence of a highly compressive residual stress distribution, RT examination will reveal it [18]. Thus, there is a rationale for the construction of meaningful acceptance criteria for RT inspection of weld overlays. Detectable cracks may be acceptable but would require further evaluation. The circumferential crack length as established by RT or UT would be an important factor in such an evaluation.

Summary and Conclusions

The relatively low cost of weld overlay repair has led to its widespread application as an alternative to replacement of piping systems in which significant IGSCC cracks are found. In addition to adding needed structural reinforcement, a well-designed overlay repair improves the residual stress distribution in the pipe and reduces the likelihood of further cracking. Moreover, correct selection of filler metal and control of the welding process produce an overlay that is highly resistant to cracking even if the crack should reach the overlay fusion line.

Periodic inspection of weld overlay repairs is prudent as a means of confirming that the expected crack mitigation benefits were achieved. Although inspection is difficult, radiography and UT techniques are available which can provide enough information to confirm the integrity of the repair. Because many overlays are conservatively designed, appropriate inspection criteria which are within the capabilities of available NDE techniques may be established on a case-specific basis.

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WELD OVERLAY REPAIRS

AT

EDWIN I. HATCH NUCLEAR PLANT

BY

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1.0 INTRODUCTION AND OBJECTIVE

Intergranular Stress Corrosion Cracking (IGSCC) incidents in the stainless steel primary coolant loop piping systems of operating boiling water reactor (BWR) plants have resulted in a major concern for the nuclear power industry. The problem of IGSCC has raised and is still raising questions about the safety of our nuclear power facilities by the public, the regulatory bodies, and even the utilities themselves. This problem has caused utilities to increase the amount of normally scheduled inservice inspection (ISI) to, in many cases, 100% of a system's welds in a fuel cycle. The increase in inspections of stainless steel piping and any subsequent repairs or replacements due to IGSCC has resulted in a decrease of plant availability and an increase in radiation exposure to plant and examination personnel.

The problem of IGSCC has caused a new thrust in the development of inspection hardware and techniques that have magnified the cost of ISI. Further, the remedies and materials that have been developed to reduce or eliminate the cracking problems have caused utilities to strain already tight budgets and this in turn has caused considerable consumer outcry over subsequent requests for rate increases.

The objectives of this paper are to relate how one utility, Georgia Power Company, has used a repair approach (i.e., weld overlays) to remedy this problem and how this repair approach has evolved over the last 3 years at Plant Edwin I. Hatch. It will show the rationale behind the use of overlays and attempt to identify problems and subsequent resolutions to those problems relative to the use of weld overlays. Also, this paper will provide suggestions to make this current interim repair method an acceptable long-term fix to the IGSCC problem.

2.0 BACKGROUND

2.1 Hatch Unit 1

During the 1982 maintenance/refueling outage, stainless steel piping welds in the Reactor Recirculation (Recirc), Residual Heat Removal (RHR), and Reactor Water Cleanup (RWCU) systems were scheduled for ultrasonic examination pursuant to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code (Reference 7). The examination scope was augmented to include known IGSCC-susceptible welds based on occurrences at other BWR plants. As a result of cracking observed earlier that year at Niagara Mohawk Power Corporation's Nine Mile Point-1 Plant (NMP-1), the U. S. Nuclear Regulatory Commission (NRC) issued I&E Bulletin 82-03 (Reference 1) which affected those BWRs scheduled to shutdown in 1982 for refueling and maintenance. While the subject I&E bulletin did not specify the quantity or type of welds to examine, it did require the utilities to qualify their ultrasonic nondestructive examination procedure(s) and inspection personnel on known cracked piping samples from NMP-1. Through the efforts of the Electric Power Research Institute (EPRI) and others, samples of NMP-1 piping were made available at Battelle-Columbus Laboratories for inspection by the various inspection teams. The inspection personnel and procedures used during the examination of stainless steel piping at Hatch Unit 1 during the 1982 maintenance/refueling outage were qualified in September 1982 prior to the start of the subject outage. During the course of the outage, reportable, crack-like indications were observed in the Recirc and RHR systems. As a result, the scope of examinations was expanded for the affected systems pursuant to the requirements of ASME Section XI. Approximately 50% of all the stainless steel piping welds were ultimately examined and included welds in the Recirc, RHR, and RWCU systems. Please refer to Section 5.0 for resolution of the crack-like indications observed during the 1982 maintenance/refueling outage. The unit was returned to power operation with NRC approval in February 1983 (Reference 8). Conditions were imposed by NRC such that they required augmented reactor coolant leakage surveillance and submittal of inspection and/or replacement plans for the stainless steel piping ninety days prior to the next scheduled outage for their review and approval.

The next scheduled maintenance/refueling outage for Hatch Unit 1 was conducted in 1984. Inspection plans were submitted to NRC in May 1984 (Reference 12) for their review and approval. The scope of examinations included stainless steel piping welds in the Recirc, RHR, and RWCU systems and met the intent of the weld examination selection criteria specified subsequent to I&E Bulletin 82-03 in documents such as the NRC-issued safety evaluation report on the repairs and analyses performed during the Hatch Unit 1 1982 outage; NRC I&E Bulletin 83-02 (Reference 2), the follow-on bulletin to I&E Bulletin 82-03; NRC internal letter SECY 83-267C (Reference 3); and NRC Generic Letter 84-11 (Reference 4). As a result of observing reportable, crack-like indications in the original scope of examinations in the Recirc and RHR systems, the scope was expanded pursuant to the requirements of ASME Section XI. Ultimately, 100% of the welds in those two systems were examined. While crack-like indications were not observed in the RWCU system, the scope of examination for that system was expanded to include 100% of the welds. Please refer to Section 5.0 for resolution of the crack-like indications observed during the 1984 maintenance/refueling outage. Those welds found to have crack-like indications during the previous outage and repaired, as appropriate, were re-examined by ultrasonic and surface examination nondestructive examination techniques and were found to be acceptable for continued service. The unit was returned to power operation with NRC approval in January 1985. The repairs approval letter and subsequent safety evaluation report (References 10,11) issued by NRC following the 1984 outage requires that Georgia Power submit its inspection plans for NRC review and comment thirty days prior to the next scheduled outage. The next outage at Hatch Unit 1 is scheduled for Fall 1985.

2.2 Hatch Unit 2

During the 1983 maintenance/refueling outage, stainless steel piping welds in the Recirc, RHR, and RWCU systems were examined using ultrasonic examination techniques. While the scope of examinations included those welds normally scheduled to be examined pursuant to the requirements of ASME Section XI, the scope was augmented pursuant to the requirements of NRC I&E Bulletin 83-02. This bulletin was a follow-on to NRC I&E Bulletin 82-03 and pertained to those BWR plants not covered by the previously issued bulletin concerning pipe cracking. NRC I&E Bulletin 83-02 incorporated the "lessons learned" from the previously issued bulletin and factored in the cracking experience observed at the plants examined to I&E Bulletin 82-03. The two bulletins were similar in that they both required inspection personnel and ultrasonic nondestructive examination procedures to be qualified for use. It was permissable to "grandfather" the qualification performed under I&E Bulletin 82-03 provided that inspection personnel, ultrasonic examination procedures, and equipment used were the same. The significant differences between the two bulletins were that I&E Bulletin 83-02 specified which welds were to be examined, quantity thereof, and requirements for examination scope increase in the event reportable, crack-like indications were observed. During the course of inspections conducted at Hatch Unit 2 during the 1983 outage, numerous reportable, crack-like indications were observed and resulted in an expanded scope of examinations ultimately including 100% of the welds in the Recirc and RHR systems. Since reportable, crack-like indications were not observed in the RWCU piping, the scope of examinations was not augmented for that system and only 26% of the welds were examined. A total of thirty-nine welds in the Recirc and RHR systems were found to have reportable, crack-like indications and required resolution. Twenty-six welds were repaired by means of weld overlay and twelve welds were left unrepaired based on flawed piping analysis (Reference 15) performed in accordance with IWB-3640 of ASME Section XI. While one remaining weld could have been repaired by means of weld overlay, it was replaced for investigative purposes. Upon completion of the necessary repairs and analyses, the unit was returned to power operation with NRC approval in July 1983 (Reference 9). Subsequent to the return to power operation, the decision was made to replace the stainless steel piping in the Recirc, RHR, and RWCU sytems as a result of numerous reportable, crack-like indications observed during the 1983 outage. The subject piping was replaced during an outage starting in January 1984. The unit was returned to power operation in September 1984 following completion of the piping replacement, maintenance, and refueling activities. The information contained herein concerning Hatch Unit 2 is provided for historical purposes. All subsequent information in this paper will pertain only to Hatch Unit 1.

3.0 CIRCUMSTANCES LEADING TO WELD OVERLAY REPAIR

3.1 Hatch Unit 1 1982 Maintenance/Refueling Outage

While the potential for cracking of certain welds was acknowledged as a result of cracking observed at the time at other BWR plants, cracking was not expected due in part to the age of the plant. The unit had been operating commercially for slightly less than seven years. In addition, it was felt by many within the industry that the cracking in large diameter piping such as that experienced at NMP-1 was an isolated case or possibly generic to the BWR 2 plants. Should cracking be observed during the 1982 outage at Hatch Unit 1, the existence of the weld overlay as a repair technique was known to GPC as a result of participation in an EPRI workshop on reactor repairs conducted in Palo Alto, California in June 1982. It was learned then that repairs using weld overlays had been successfully applied to the RWCU System at Commonwealth Edison Company's Quad Cities Unit 2 and was approved by NRC as an interim repair. Weld overlay repairs were also being conducted at Northern States Power's Monticello plant at the same time as the Hatch Unit 1 outage.

Since cracking was observed using ultrasonic techniques in several Recirc and RHR piping welds during the 1982 maintenance/refueling outage, GPC was faced with the question of whether to repair the weld areas or to replace the affected piping sections. A number of factors were involved in the decision making process. The weld areas in question were non-isolable. Although the reactor's core was off-loaded to facilitate torus modifications because of the Mark I Containment Long-Term Program commitments to NRC, it was not desirable to replace piping if an alternate solution was available. In addition, the availability of large diameter "L" grade or "Nuclear Grade" piping was questionable. Radiation exposure received as a result of piping replacement was also a factor to be taken into consideration. The effect on outage length was considered as some unacceptable ultrasonic indications were not confirmed until approximately mid-way through the outage.

As noted above, it was known that repairs by means of weld overlay had been successfully performed at the Quad Cities plant and was in progress at the Monticello plant. Further, the process had been accepted by NRC for use. Although only considered a temporary repair by NRC for piping with IGSCC, GPC acknowledged that EPRI and the BWR Owners Group for IGSCC Research were attempting to qualify the weld overlay process as a permanent repair. Further, the weld overlays could be done remotely using automatic gas tungsten arc welding (GTAW) equipment and done without having to drain the reactor vessel to make the necessary repairs thus reducing personnel radiation exposures. The weld overlays would not only prevent reactor coolant leakage from the cracked weld if the crack grew through-wall but would also restore the structural margin of the weld area. It was also theorized that the weld overlays would induce favorable residual stress patterns that would eliminate crack propagation and growth. All of the above made the weld overlay an attractive repair solution. As a result, it was recommended to GPC management that repairs by means of weld overlay be performed in lieu of replacement of the affected piping sections. The architect/engineer that was involved with the Quad Cities and Monticello repairs and licensing thereof was contracted to perform the analysis work and any necessary repairs at Hatch Unit 1 during the 1982 maintenance/refueling outage.

3.2 Hatch Unit 1 1984 Maintenance/Refueling Outage

Immediately prior to Hatch Unit 1 shutting down for maintenance and refueling in the Fall of 1984, Hatch Unit 2 had just completed an outage approximately 7 1/2 months in length during which the piping in the Recirc, RHR, and RWCU systems was replaced. As a result, it was the intent of GPC to limit the length of the Hatch Unit 1 outage to approximately 8-10 weeks. Should reportable, crack-like indications be observed during the course of the inservice inspection of stainless steel piping conducted during that outage, analyses and repairs, as appropriate, would be performed in lieu of piping replacement. While replacement piping had been ordered for the unit and was similar to that installed at Hatch Unit 2, no commitment had been made to replace the piping.

While the desire to limit the length of the 1984 maintenance/ refueling outage was a factor, there were other factors involved also. The extent of IGSCC during the previous Hatch Unit 1 outage was not as great as that observed at Hatch Unit 2. GPC's own projection of IGSCC during the 1984 outage and the results of a damage index study (Reference 12) performed by an outside consultant indicated that while some new cracking would be observed, it would still be less than that observed at Hatch Unit 2 during the 1983 maintenance/ refueling outage. Justification for continued use of the existing weld overlays applied during the 1982 outage was submitted prior to the 1984 outage to NRC for their review and approval (Reference 12). The justification, with input from an outside consultant, included information relative to testing of weld overlays sponsored by GPC and others to confirm the analytical predictions that weld overlays produce favorable residual stress patterns thereby justifying an extended service life.

In addition to the above, industry efforts through EPRI and others have lead to more reliable nondestructive examination techniques in the detection and sizing of IGSCC. Nondestructive examination techniques were available for use during the outages to inspect welds repaired by means of weld overlay to assure that if cracking should continue, the integrity of the weld overlay would not be violated. Research efforts continue to better develop nondestructive examination techniques to adequately inspect through the weld overlays.

4.0 PIPING REPAIR WITH OVERLAY

4.1 Hatch Unit 1 1982 Maintenance/Refueling Outage

As a result of observing reportable, crack-like indications during the Hatch Unit 1 1982 maintenance/refueling outage, six welds in the Recirc and RHR systems were repaired by establishing additional "cast-in place" pipe wall thickness from weld metal deposited 360 degrees around and to either side of the existing welds as shown in Figures 4.1, 4.2, and 4.3 (Illustration Source: Reference 14). Those welds found to have reportable, crack-like indications are identified in Section 5.0, Table 5.1. The weld-deposited band over the cracks will provide wall thickness equal to that required to provide original design safety margins. In addition, the weld metal deposited produced a favorable compressive residual stress pattern, thereby precluding additional crack propagation or growth. The weld metal used was Type 308L which is resistant to the propagation of IGSCC. The weld overlay was applied using remote, automatic GTAW equipment to limit personnel radiation exposure.

An architect/engineer provided design and analyses of all six weld overlay repairs (Reference 14). In addition, they performed an analysis to show a seventh weld having reportable, crack-like indications to be acceptable without repair. The adequacy of the weld overlay repairs was demonstrated by meeting the equivalent margins of safety for strength and fatigue considerations as provided in the ASME Section III Design Rules (Reference 5). Crack growth due to both cyclic stress and steady state stress was calculated. The allowable crack depth was established based on net section limit load for each cracked and repaired weld per ASME Section XI, IWB-3600 (Note: At that time, IWB-3600 was only a proposed addition to ASME Section XI). The design life of each repair was established as the minimum of either the predicted time for the observed crack to grow to the allowable crack depth or five years.

Although NRC Generic Letter 84-11 did not exist at the time of these repairs, the weld overlays met the intent of the subject NRC letter and were of the full structural type. Current analyses of these repairs show that the weld overlays applied during the 1982 maintenance/refueling outage meet or exceed present criteria established by the NRC.

4.2 Hatch Unit 1 1984 Maintenance/Refueling Outage

During the 1984 maintenance/refueling outage at Hatch Unit 1, twenty-one welds in the Recirc and RHR systems were observed to have reportable, crack-like indications. The same architect/engineer used during the 1982 weld overlay repairs provided design, analyses, and evaluations for the seventeen welds which were repaired by means of weld overlay and four welds that were acceptable without repair (Reference 16). Representative overlay designs are shown in Figures 4.4 thru 4.7 (Illustration Sources: References 16, 21).

The 1984 weld overlay repairs meet the earlier requirements of strength and fatigue of ASME Section III and the crack growth/depth limits imposed by ASME Section XI, Paragraph IWB-3600. In addition, the repairs also meet the regulatory requirements imposed by NRC Generic Letter 84-11.

NRC Generic Letter 84-11 imposed the following additional requirements for design and application of weld overlay repairs:

(a) If cracked welds are repaired by weld overlay, the thickness of the overlay must be sufficient to provide full IWB-3600 margin during the proposed operating period. Effective overlay thickness is defined as the thickness of overlay deposited after the first weld layer that clears dye penetrant testing (PT) inspection.

(b) The minimum effective overlay thickness permitted is two weld layers after the first layer to clear PT inspection.

(c) Full structural strength weld overlays must be provided for long cracks with total circumferential extent approaching the length that would cause limit load failure if they were actually through-wall.

(d) Multiple short circumferential cracks are to be treated as one crack wih a length equal to the sum of the circumferential lengths.

NRC Generic Letter 84-11 was the first attempt by the NRC to standardize weld overlay repair and analysis.

The design, analysis, and application of weld overlay repairs at Plant Hatch has evolved from a repair method used by utilities and approved by the regulatory authorities on an individual basis to a more generic industry standard that added conservatisms due to the uncertainties associated with crack sizing.

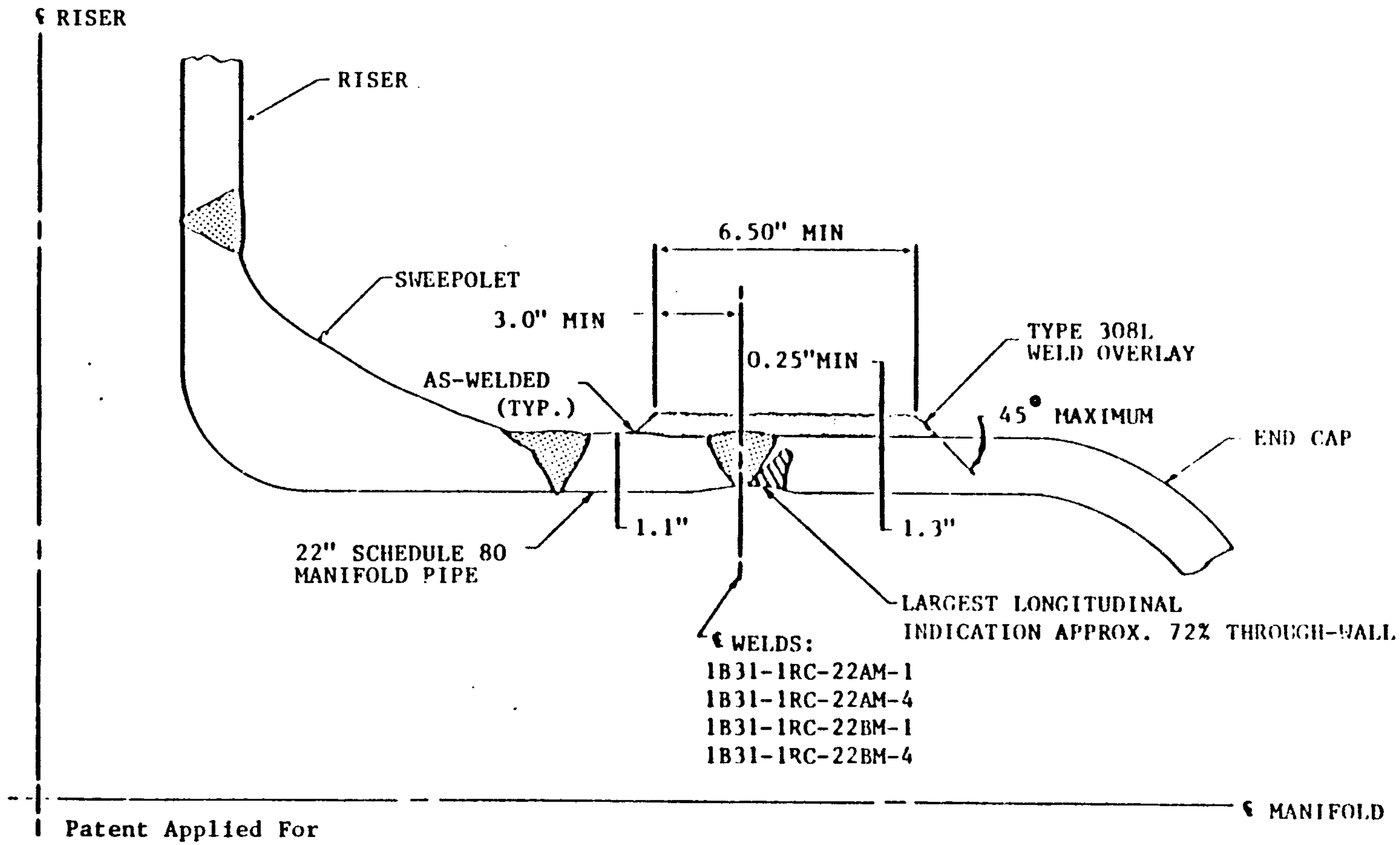


Figure 4.1

SCHEMATIC OF END CAP WELD OVERLAY

Source: NUTECH

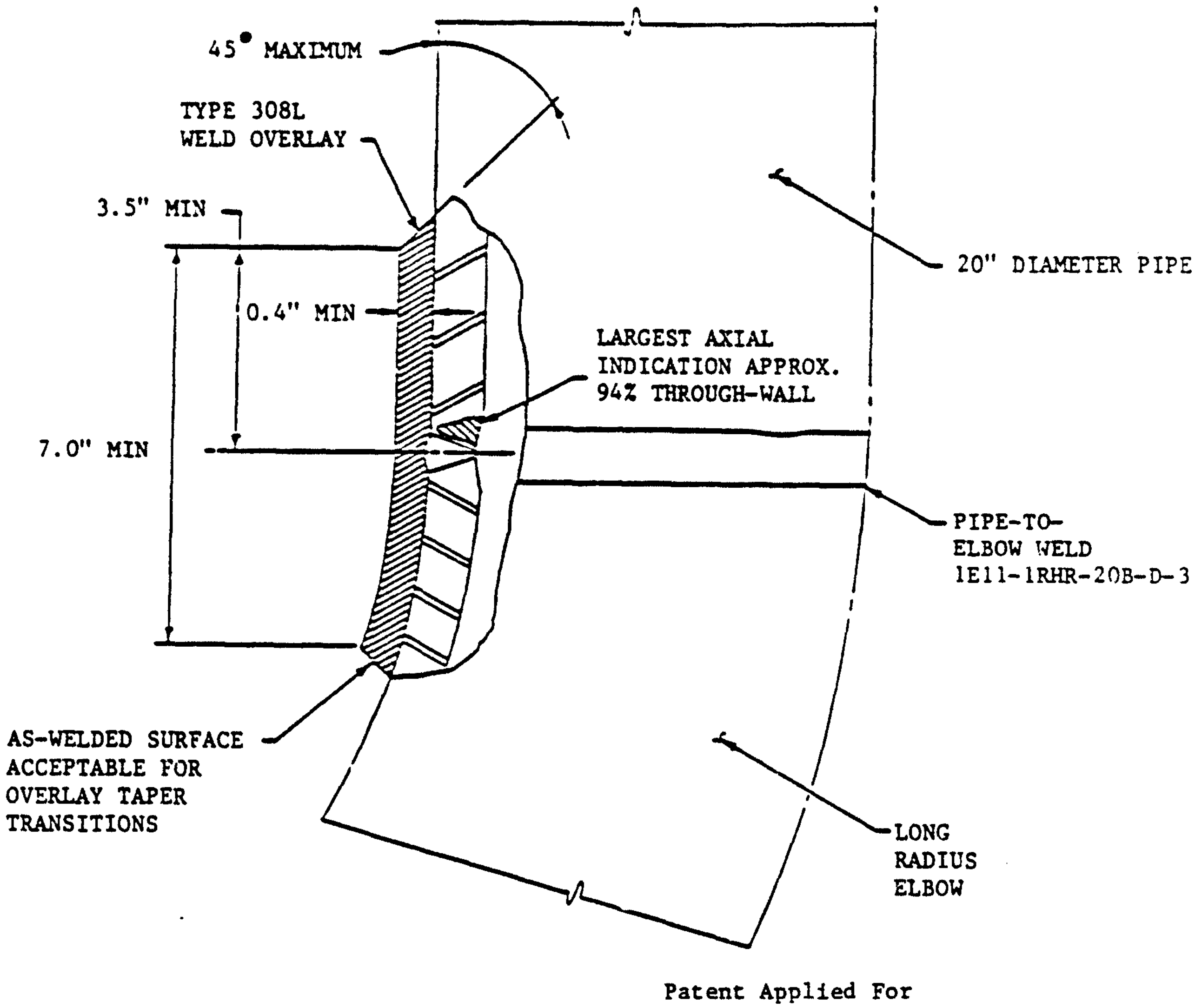


Figure 4.2
SCHEMATIC OF ELBOW-TO-PIPE WELD OVERLAY Source: NUTECH

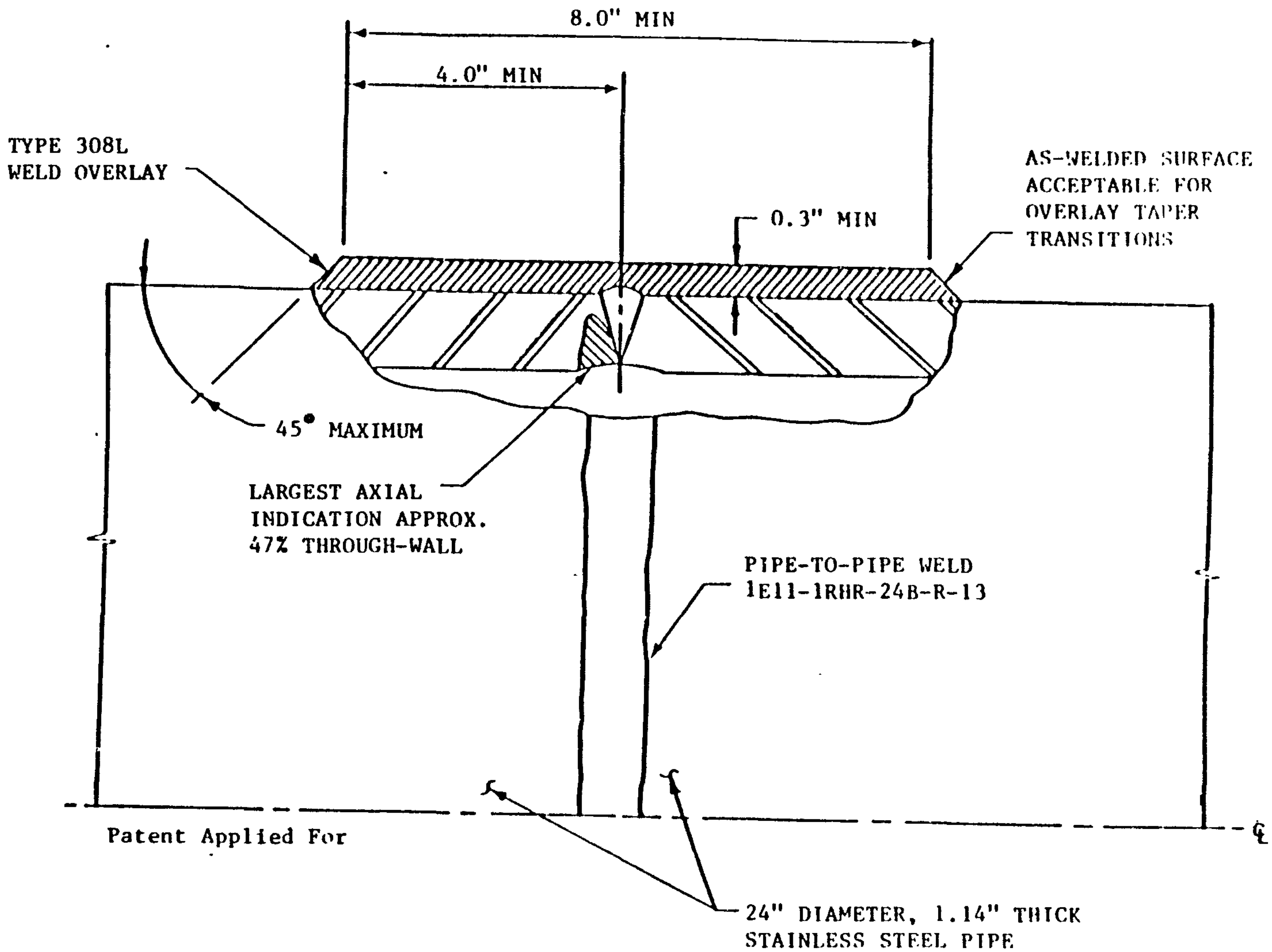
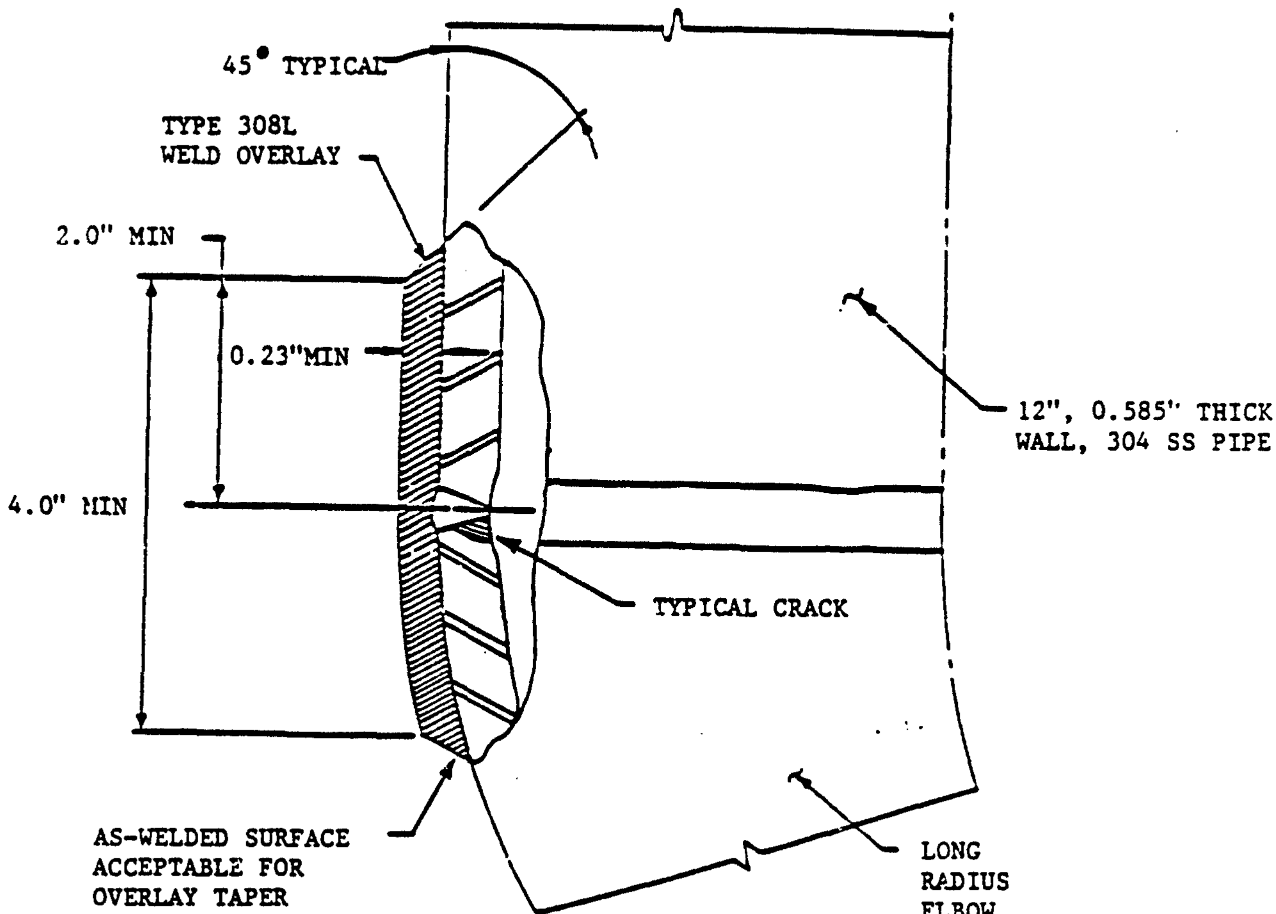


Figure 4.3

SCHEMATIC OF PIPE-TO-PIPE WELD OVERLAY

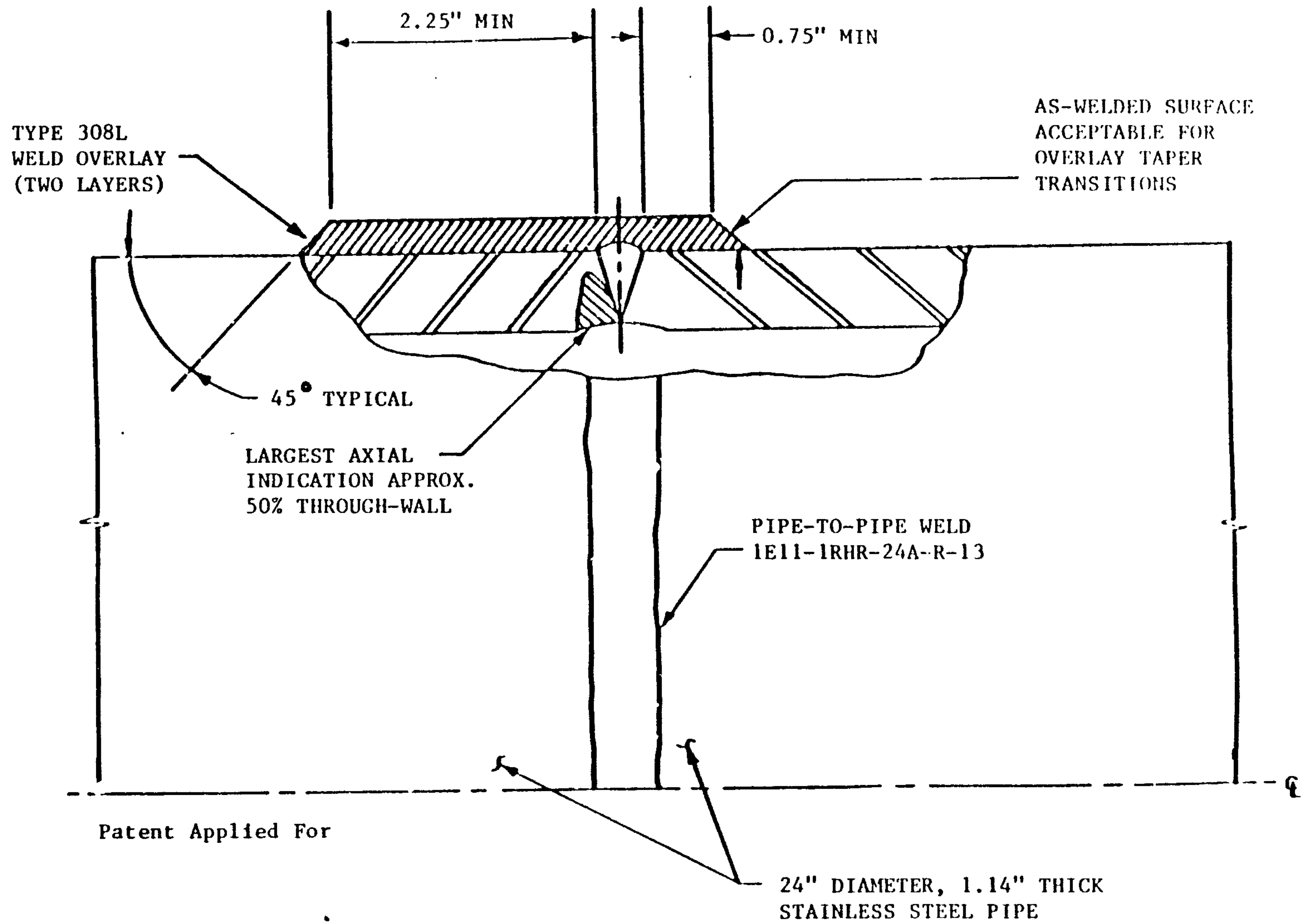
Source: NUTECH



Patent Applied For

Figure 4.4
TYPICAL CONFIGURATION OF 12"
ELBOW-TO-PIPE WELD OVERLAY

Source: NUTECH



Patent Applied For

Figure 4.5

TYPICAL CONFIGURATION OF 24" PIPE-TO-PIPE

WELD OVERLAY LEAK BARRIER

Source: NUTECH

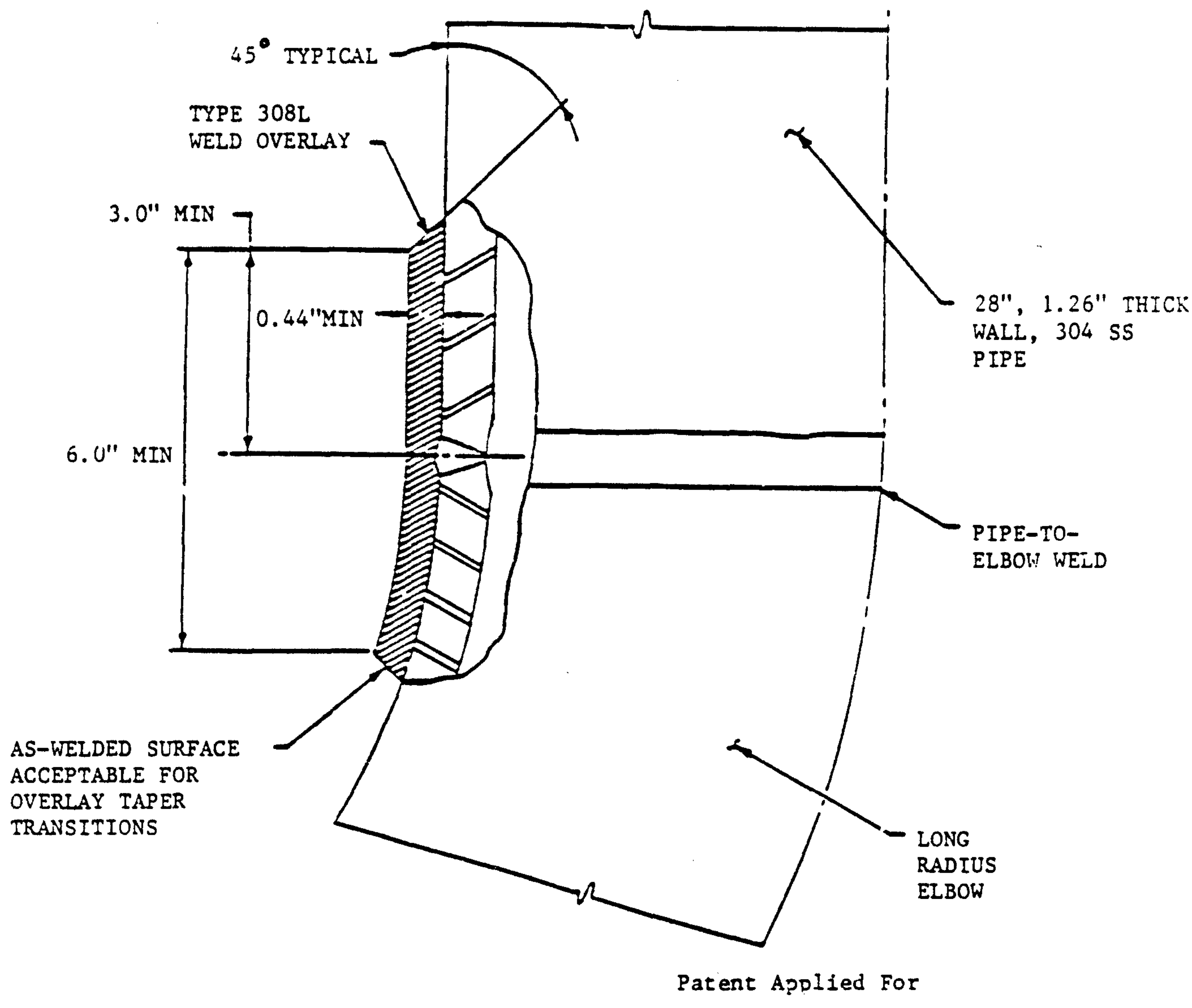
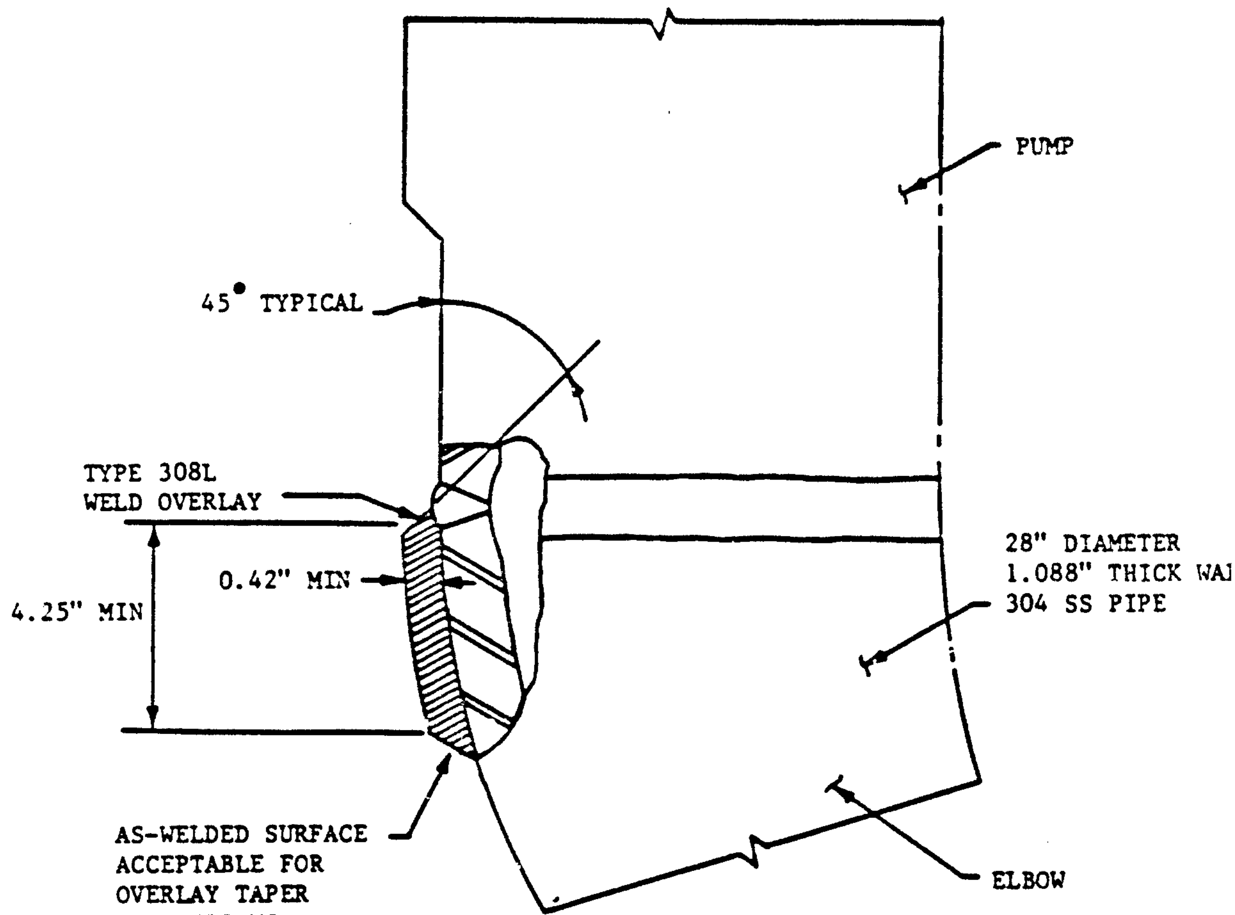


Figure 4.6

TYPICAL CONFIGURATION OF 28" PIPE-TO-ELBOW WELD CVERLAY

Source: NUTECH



Patent Applied For

Figure 4.7

TYPICAL CONFIGURATION OF 28"

ELBOW-TO-PUMP WELD OVERLAY

Source: NUTECH

5.0 SCOPE OF REPAIRS

During the 1982 maintenance/refueling outage at Hatch Unit 1, seven welds in the Recirc and RHR systems were found by means of ultrasonic examination to have indications indicative of IGSCC. These welds and their disposition are identified in Table 5.1.

Figure 5.1 is a conceptual drawing depicting the locations of the affected welds in the Recirc and RHR systems. Six of the subject welds were repaired by means of weld overlay, as depicted by Section 4.0, Figures 4.1, 4.2, and 4.3. The seventh weld observed to have reportable, crack-like indications was shown by analysis to continue to have the original design safety margins and as a result was left unrepaired. An acoustic emissions device was voluntarily installed by GPC on that particular weld to monitor for any leakage from the weld area. During the 1984 maintenance/refueling outage, the acoustic emissions device on the unrepaired weld was removed based on the results of the weld's re-examination, the leak-before-break concept, and the augmented reactor coolant leakage surveillance currently in place.

Ultrasonic examinations conducted during the 1984 maintenance/refueling outage revealed twenty one welds (including the one unrepaired weld from the previous outage) to have reportable, crack-like indications. The Recirc and RHR systems were the only systems found to be affected by IGSCC during the 1984 maintenance/refueling outage. Reportable, crack-like indications were also observed in these same systems during the 1982 maintenance/refueling outage. Table 5.2 identifies the welds observed during the 1984 outage to have reportable, crack-like indications and their resolution. Figure 5.2 provides a conceptual drawing of the affected systems depicting the location of the affected welds. Seventeen welds in the aforementioned systems were repaired by means of weld overlay in a similar fashion as that done during the 1982 outage. Section 4.0, Figures 4.4 through 4.7 are representative of the overlays applied during the 1984 maintenance/refueling outage. The four remaining affected welds were shown by analysis to continue to have adequate design safety margins and were left unrepaired. Those welds having overlays applied previously were re-examined and yielded acceptable results and did not require any corrective actions. The existing overlays and those applied during the 1984 maintenance/refueling outage were ultrasonically examined to verify the integrity of both the weld metal and its bond to the pipe base material, in a manner consistent with ASME Code, Section V, Paragraph T550 (Reference 6). In addition, a liquid penetrant examination was conducted on the weld overlay and 1" of base material on either side of the weld overlays. The next examination of the weld overlay-repaired welds and the unrepaired welds is scheduled for the 1985 maintenance/refueling outage.

TABLE 5.1
PLANT E. I. HATCH UNIT 1

FLAW DISPOSITION

1982 MAINTENANCE/REFUELING OUTAGE

SYSTEM	WELD NO.	WELD DESCRIPTION	EXAMINATION RESULTS	DISPOSITION
Recirc	1B31-1RC-22AM-1	Pipe-to-End Cap	Axial indications, 3/8" long, max. est. depth 63%	Weld Overlay
	1B31-1RC-22AM-4	Pipe-to-End Cap	Axial indications, 1/2" long, max. est. depth 72%	Weld Overlay
	1B31-1RC-22BM-1	Pipe-to-End Cap	Axial indications, 1/4" long, max. est. depth 64%	Weld Overlay
	1B31-1RC-22BM-4	Pipe-to-End Cap	Axial indications, 1/2" long, max. est. depth 67%	Weld Overlay
	1B31-1RC-22AM-1BC-1	Sweepolet-to-Manifold	Axial indications, 1/2" long, max. est. depth 12%	Analysis, left unrepaired
RHR	1E11-1RHR-20B-D-3	Elbow-to-Pipe	Axial indications, 3/8" long, max. est. depth 94%; and, circumferential indications, 1 1/2" long, max. est. depth 33%	Weld Overlay
	1E11-1RHR-24B-R-13	Pipe-to-Pipe	Axial indications, 3/8" long, max. est. depth 47%	Weld Overlay

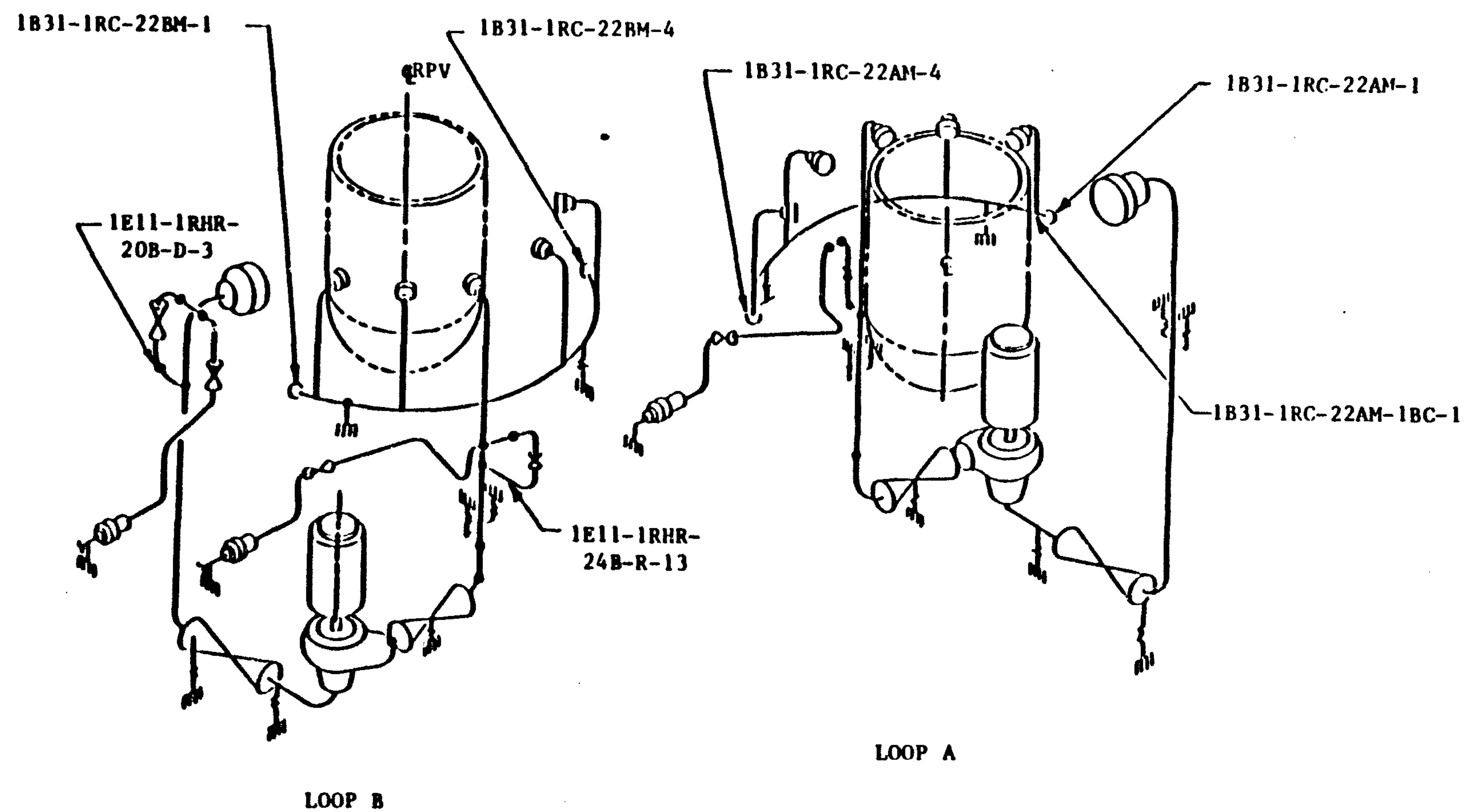


Figure 5.1
CONCEPTUAL DRAWING OF RECIRCULATION AND RHR SYSYEMS
(Reportable Indications - 1982 Outage)

TABLE 5.2
PLANT E. I. HATCH UNIT 1

FLAW DISPOSITION

1984 MAINTENANCE/REFUELING OUTAGE

SYSTEM	WELD NO.	WELD DESCRIPTION	EXAMINATION RESULTS	DISPOSITION
Recirc	1B31-1RC-12AR-F-2	Pipe-to-Elbow	Circumferential indications 360° intermittent, 20-30% max. detected depth	Weld Overlay
	1B31-1RC-12AR-F-3	Elbow-to-Pipe	Circumferential indications 360° intermittent, 20-30% max. detected depth	Weld Overlay
	1B31-1RC-12AR-H-2	Pipe-to-Elbow	Circumferential indications 360° intermittent, 20-30% max. detected depth	Weld Overlay
	1B31-1RC-12AR-H-3	Elbow-to-Pipe	Circumferential indications 360° intermittent, 20-30% max. detected depth	Weld Overlay
	1B31-1RC-12AR-J-3	Elbow-to-Pipe	Circumferential indications 360° intermittent, 20-30% max. detected depth	Weld Overlay
	1B31-1RC-12AR-K-2	Pipe-to-Elbow	Circumferential indications 360° intermittent, 20-30% max. detected depth	Weld Overlay
	1B31-1RC-12AR-K-3	Elbow-to-Pipe	Circumferential indications 360° intermittent, 20-30% max. detected depth	Weld Overlay

TABLE 5.2 (CONTINUED)
PLANT E. I. HATCH UNIT 1

FLAW DISPOSITION

1984 MAINTENANCE/REFUELING OUTAGE

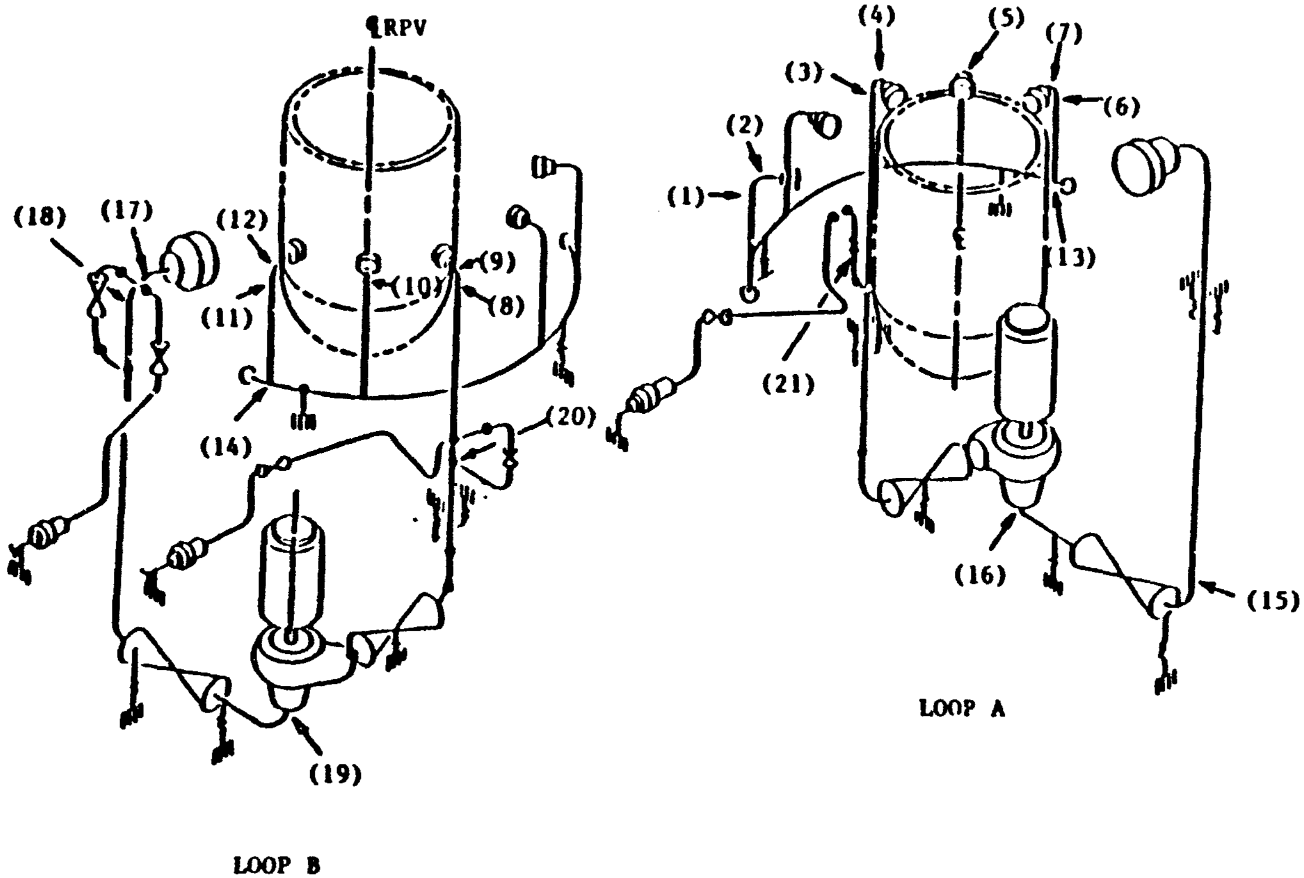
SYSTEM	WELD NO.	WELD DESCRIPTION	EXAMINATION RESULTS	DISPOSITION
Recirc (Continued)	1B31-1RC-12BR-C-2	Pipe-to-Elbow	Circumferential indications 360° intermittent, 49% max. detected depth	Weld Overlay
	1B31-1RC-12BR-C-3	Elbow-to-Pipe	Circumferential indications 360° intermittent, 66% max. detected depth	Weld Overlay
	1B31-1RC-12BR-D-3	Elbow-to-Pipe	Circumferential indications 360° intermittent, 20% max. detected depth	Weld Overlay
	1B31-1RC-12BR-E-2	Pipe-to-Elbow	Circumferential indications 360° intermittent, 25% max. detected depth	Weld Overlay
	1B31-1RC-12BR-E-3	Elbow-to-Pipe	Circumferential indications 360° intermittent, 30% max. detected depth	Weld Overlay
	1B31-1RC-22AM-1BC-1	Sweepolet-to-Manifold	Spot indication, 18% max. depth; and, circumferential indications, 11% max. depth	Analysis, left unrepaired
	1B31-1RC-22BM-1BC-1	Sweepolet-to-Manifold	Circumferential indication, 29% max. depth	Analysis, left unrepaired

TABLE 5.2 (CONTINUED)
PLANT E. I. HATCH UNIT 1

FLAW DISPOSITION

1984 MAINTENANCE/REFUELING OUTAGE

SYSTEM	WELD NO.	WELD DESCRIPTION	EXAMINATION RESULTS	DISPOSITION
Recirc (Continued)	1B31-1RC-28A-6	Pipe-to-Elbow	Axial indications, 16% max. depth	Analysis, left unrepaired
	1B31-1RC-28A-10	Elbow-to-Pump	Circumferential indications 360° intermittent, 50% max. detected depth	Weld Overlay
	1B31-1RC-28B-3	Pipe-to-Elbow	Circumferential indications 360° intermittent, 32% max. detected depth	Weld Overlay
	1B31-1RC-28B-4	Elbow-to-Pipe	Circumferential indications 360° intermittent, 31% max. detected depth	Weld Overlay
	1B31-1RC-28B-11	Elbow-to-Pump	Circumferential indications 360° intermittent, 49% max. detected depth	Weld Overlay
	1B31-1RC-28B-16	Pipe-to-Tee	Axial indications, 17% max. depth	Analysis, left unrepaired
RHR	1E11-1RHR-28A-R-13	Pipe-to-Pipe	Axial indications, 50% max. depth	Weld Overlay



Legend: Welds with Reportable Indications

(1) 1B31-1RC-12AR-F-2
(2) 1B31-1RC-12AR-F-3
(3) 1B31-1RC-12AR-H-2
(4) 1B31-1RC-12AR-H-3
(5) 1B31-1RC-12AR-J-3
(6) 1B31-1RC-12AR-K-2
(7) 1B31-1RC-12AR-K-3
(8) 1B31-1RC-12BR-C-2
(9) 1B31-1RC-12BR-C-3
(10) 1B31-1RC-12BR-D-3
(11) 1B31-1RC-12BR-E-2
(12) 1B31-1RC-12BR-E-3
(13) 1B31-1RC-22AM-1BC-1 *
(14) 1B31-1RC-22BM-1BC-1
(15) 1B31-1RC-28A-6
(16) 1B31-1RC-28A-10
(17) 1B31-1RC-28B-3
(18) 1B31-1RC-28B-4
(19) 1B31-1RC-28B-11
(20) 1B31-1RC-28B-16
(21) 1E11-1RHR-24A-R-13

* Also observed to have reportable indications during 1982 outage

Figure 5.2

CONCEPTUAL DRAWING OF RECIRCULATION AND RHR SYSTEMS
(Reportable Indications - 1984 Outage)

6.0 APPROACH AND RATIONALE FOR CONTINUED USE OF OVERLAYS

During the 1982 maintenance/refueling outage, weld overlays were a needed repair because replacement was not a feasible alternative. Replacement piping was not available, software and staffing for such a large project had not been started, and no budget was available for such an undertaking.

Weld overlays provided several advantages to the utility. First and foremost was that they were effective. They provided both a means to stop cracking and a means of restoring structural margin to the affected weld joint. Weld overlays could be applied remotely, therefore, limiting personnel exposure and, finally, the repairs could be made within current plant budgets and within the outage time frame.

The advantages however were balanced by concerns of future inspectability of the welds. Also, only a limited amount of empirical data was available on the effectiveness of weld overlays. These disadvantages led to the stance by the NRC of weld overlays being a one cycle fix.

GPC felt that the potential existed that weld overlays could be an interim fix as a minimum and had the potential for a long-term repair method that was both a safe and economical alternative to piping replacement.

Over the last 3 years, GPC has invested in this belief by sponsoring or co-sponsoring several research projects to verify analytical predictions of overlay effectiveness. In 1983, GPC co-sponsored with an architect/engineer and the NRC, a research project to evaluate the through-wall residual stress in 12" diameter piping. After overlay application at Hatch Unit 2, samples were made up at Plant Hatch of a full structural overlay, a mini-overlay, and a last pass heat sink weld and sent to Argonne National Laboratory for destructive analysis. The results of this analysis showed that the favorable residual stress pattern predicted by analysis did exist and was equal to or better than those predictions in all three cases (References 17, 18, 19).

In 1984, GPC sponsored and helped create a test for large diameter (28") piping overlays. This test again demonstrated that overlays provided the favorable through-wall residual stresses needed to stop crack growth (Reference 20). Along with these directly sponsored or co-sponsored research projects, GPC has been actively involved with both EPRI and other groups involved in this repair method. As a result of GPC efforts, either with other research and development groups or independently, the efforts have provided substantial results in extending overlay life industry-wide.

Today, GPC, along with the rest of the utility industry, must direct its efforts and expertise toward what we feel is the final obstacle to long-term acceptability of weld overlays. That obstacle is reliable inspectability of weld overlays. Large gains have been made in this area by EPRI and other groups over the last 3 years, however, additional work is needed in the area of weld overlay inspectability. GPC feels that these efforts are close to providing the results that will overcome this remaining obstacle and will provide utilities, shareholders, and, ultimately, rate-payers with a safe acceptable alternative to costly piping replacements.

7.0 LESSONS LEARNED

The following were "lessons learned" during the course of inspections and resulting repairs using weld overlays at Plant Hatch during the 1982 and 1984 maintenance/refueling outages:

1. Mechanized ultrasonic examination equipment qualified to both detect and size IGSCC indications is needed to help reduce examination personnel radiation exposure.

2. Heat input for first and second layers of weld overlays must be tightly controlled to avoid unnecessary blow-throughs as a result of deep axial IGSCC in small diameter piping.

3. Procedural and wire control are two areas closely looked at by NRC.

4. Assure that vendor documentation is correct when welding adjacent to valves with factory-installed pup (spool) pieces.

5. Radiation exposure can be kept to a minimum by use of remote, automatic welding equipment to perform repairs: similarly, use of master/slave units can be used during nondestructive examinations to limit inspection personnel radiation exposure.

6. Weld overlay finish may be inadequate for ultrasonic examination; grinding of overlay surface may be necessary to provide a surface suitable for examination.

7. Justification can be provided for additional service life of the weld overlays on a cycle-by-cycle basis contingent upon NRC review and approval.

8. Inspectability of the weld overlays is still of concern to NRC; research efforts are underway by EPRI and others to provide for better inspectability of the weld overlays.

9. Testing performed by GPC and others indicates that the analytical prediction of inducing favorable residual stress patterns by means of weld overlay is valid.

10. The potential exists that weld overlays are an acceptable repair method for the long term and may provide a safe, economical alternative to costly piping replacement.

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MECHANICAL SLEEVE SEALING OF STUB TUBE CRACKS IN LIEU OF WELDING METHODS

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ABSTRACT

Leaks from cracks in control rod drive stub tubes were discovered at the Santa Maria de Garona BWR plant in Spain. The cracks are believed to be due to intergranular stress corrosion cracking. The repair planning employed extensive ultrasonic testing of 55 stub tubes and investigated repair options that included welding, rolling, and a mechanical type of repair. Because the plant has been in operation for over a decade, severe constraints imposed high development requirements on weld repair methods. In the interest of a timely and cost effective response, a mechanical sealing sleeve installed over the stub tube assembly was used effectively to stop the leaks. The mechanical sealing sleeve design, qualification testing, and installation procedure are described.

INTRODUCTION

During the operation of some Boiling Water Reactors (BWRs), cracks were found in the area of the control rod stub tube nozzles (Figure 1) which had led to significant leakage of reactor coolant. This cracking has been attributed to Intergranular Stress Corrosion Cracking (IGSCC) phenomenon similar to that involved in BWR primary recirculation pipe cracking. While the activities directed at resolution of the piping problem have been extensively discussed, the stub tube problem is not widely known. It is the purpose of this paper to describe this cracking incident, the steps taken to address the problem, and the repair method that was successfully used at one particular plant.

RECENT CRACKING INCIDENTS

In the spring of 1981, Nuclenor, a Spanish utility, discovered cracks in the stub tubes of their Santa Maria de Garona plant. These stubs tubes are short thick walled pipe sections that were shop welded to the pressure vessel and then field-welded to the Control Rod Drive (CRD). The configuration of the CRD stub tube vessel attachment is shown in Figure 2.

The stub tube cracks typically occur in the heat affected zone approximately one inch below the stub tube-CRD housing J-weld. In 1982, visual and ultrasonic (UT) inspections indicated cracks on at least 22 tubes with at least one through-wall crack. At the location of the though-wall crack, a leakage of up to 30 liters/hr (8 gallons/hr) through the annular clearance between the stub tube and outer surface of the CRD housing was observed. This was repaired successfully with installation of a mechanical sealing sleeve. A year and half later during a scheduled refueling outage, a second stub tube was found leaking. This was similarly successfully sealed.

At Nine Mile Point 1, similar stub tube cracking was found during the refueling outage of 1984. Earlier cracking events occurred at Oyster Creek, Tarapur (India), and Big Rock Point. On the one hand, these earlier events happened before or soon after the plants started operation; on the other hand, the incidents at Santa Maria de Garona and at Nine Mile Point 1 occurred after a decade of operation. The stub tubes at Oyster Creek and Tarapur were repaired by grinding out the cracks and filling out the excavated region with Type 308L or INCO 182T weld metal and applying a weld overlay of the same material over the whole stub tube. At Big Rock Point, the stub tube leak was sealed by roll swaging the stub tube against the vessel.

The cause of the cracking in the two most recent incidents has been attributed to but not verified as IGSCC. The three essential conditions that lend susceptibility to IGSCC are present in these cases. The material of the stub tubes is conventional stainless steel AISI 304. Following the vessel to stub tube welding, the reactor vessel assembly was heat treated in the furnace for many hours leading to heavily sensitized stub tubes. The other parameters are the high dissolved oxygen content of the BWR water environment and high residual stresses close to the material's yield point.

In the case of the Santa Maria de Garona plant, there was considerable weld induced shrinkage (2.5 mm/0.100 in.) at the O.D. surface) of the stub tube in the area of the CRD housing J-weld which resulted in high residual stresses. Also, since the vessel attachment weld was thicker on the uphill side, the stub tubes became inclined from the vertical centerline. The field operation included a straightening procedure consisting of the addition of weld beads on the downhill side of the CRD J-weld. As a result, high residual tensile stresses were produced in the downhill sides of the stub tubes in the vicinity of the CRD weld where it coincided with a thinner stub tube wall. Another straightening procedure that has been reported in other cases involved heating the weld and pulling the tube, which also could have resulted in high residual stresses.

PROCEDURES

A. Definition of the Problem

The primary problem in this situation is the release of radioactive fluid to the environment which must be eliminated or controlled to within the Technical Specifications limit of the plant. Even in the case of a full 360° circumferential through-wall crack, the postulated consequence has no safety significance. This is so because gross leakage is precluded by the fact that the CRD housing is prevented from being ejected from the vessel by the stub tube piece integral with the CRD housing. Furthermore, a mechanical restraint system is provided underneath in the event of failure of the housing attachment weld.

Therefore, the systematic approach to the problem was to adopt the most cost effective way of keeping the plant in operation while the affected stub tubes were examined, the cracking and its progress were characterized, alternative solutions were investigated, and finally, a long term solution selected and implemented.

B. Inspections

The stub tubes were examined with a device containing four ultrasonic transducers that scanned in the radial (straight and at 45° inclined up wards and downwards) and circumferential directions. The automatic scanning device was inserted in the CRD housing and indexed circumferentially and vertically. This device was able to detect the cracks but could not accurately measure the depth of the deepest cracks because of limited access to the stub tube via the J-weld from the CRD housing. A similar scanning device was used on the outside surface of four baseline tubes that had major cracks. The depths and lengths of these cracks were accurately measured by this second device. Crack lengths of 190 and 130 degrees were observed on the leaking tubes. Borescope inspections, eddy current testing and radiographic methods confirmed the location of the cracks predicted by the UT inspections.

In 1982, an initial batch of 27 stub tubes were examined as required by the inspection plan. Of these, 22 had reportable indications at the same approximate location. These cracks occurred mostly on the outermost row of tubes where the profile of the vessel attachment weld is very steep. The cracks also predominated at the downhill side. A second batch of 28 tubes were examined in 1983. The results were less extreme than the initial group. A crack growth rate was determined from the consecutive inspections of the baseline tubes. A very low crack growth rate of 1.0 mm/yr (0.04 inch/yr) was found. The results of the examination also showed a high correlation between cracking and the stub tube material with the higher carbon content (0.038% C vs. 0.016%) of the two heats used.

C. Evalaution of Alternative Repair Solutions

As a result of the inspections, the areas of the CRD penetration with potential for cracking and which must be considered in a repair solution were defined as follows:

1. Region of the stub tube adjacent to the CRD J-weld. All of the observed cracking occurred in this area.

2. Region of the stub tube adjacent to the vessel attachment weld, which is highly sensitized but post weld heat treated. Weld porosity, lack of fusion and inclusions have been found in other plants. There is a potential for future cracking although a lower one than for Region 1.

3. CRD housing region adjacent to the J-weld. There is a much lower probability of future cracking in this region due to unsensitized material and less weld penetration. However, an existing stub tube crack could propagate into this region.

The following repair methods were tried or evaluated for their availability, probability of success, impact on reactor operation, risk of failure, and cost:

1. Roll Swaging

When a hydrostatic test performed in 1981 showed one stub tube was leaking, a decision was made to roll swage the tube against the vessel wall to effect a seal following the successful rolling procedure at Big Rock Point. This was a standard procedure and the equipment could be obtained on short notice. However, after a short period of operation, the leak started again. The distinguishing feature of the Big Rock Point CRD installation was an interference fit between the stub tube and the vessel which apparently insured the success of rolling. At Santa Maria de Garona however, there was a large diametral clearance gap of 0.15 mm (0.006-in.) and the plant was probably operating with the leak for sometime. Further rolling was not possible because the increased strains would have exceeded the criteria designed to prevent stress corrosion cracking of the CRD housing, a more critical problem.

2. Stub Tube Replacement and Rewelding

Alternative solutions were considered, some of the first involving welding methods. It is considered that replacing the stub tube would constitute a daunting engineering challenge requiring remote machining equipment to machine out the stub tube and sophisticated remote welding equipment to operate in a very constrained work area. Welding under water free conditions while maintaining the vessel full of water for shielding purposes is also not an easy task. But the most serious objection to rewelding is the high residual stresses that would ensue, leaving the same condition that produced the IGSCC in the first place. Further, heat treatment of a weld repaired vessel in the field is an untried method. Even if these difficulties are overcome, there are other problems such as providing an inspection program to insure good welds and repairing a bad weld. The cost would have been tremendously high and the plant operation would have been severely impacted.

3. Mechanical Sealing Sleeve

To return the plant to service after the roll swaging had not worked, a mechanical type of sealing sleeve that fit over the stub tube to seal the crack was developed and installed. The mechanical sleeve seal was an attractive approach since it presented the fewest problems overall in development and its availability was timely. It also satisfied the current need of the plant which was to provide a solution for the existing cracks in Region 1 of the stub tube. Moreover, its installation did not require any significant removal of reactor vessel internal components nor was it a factor in the plant outage time. The primary question with the use of the seal was its ability to satisfy a lifetime requirement. Should the seal fail in the future however, it could be easily retrieved and the packings replaced. For this reason, the mechanical sleeve seal was considered as an interim repair while a permanent remedy was being developed.

D. Mechanical Sealing Sleeve Installation

Description

The mechanical sealing sleeve consisted of two nested cylinders which held upper and lower packing glands and a preloading mechanism. In the installed position (Figure 3), the upper and lower packings seated on the control rod housing (above the crack) and stub tube (below the crack) O.D. surfaces, respectively, thereby effectively isolating the crack. The packing glands were made of a graphite ribbon material which stands up under reactor pressure, temperature, water chemistry, and radiation environment. With this seal packing design, a carefully prepared seating surface was not required, an important advantage in a remote application.

Design Guidelines

The primary duty of the sleeve is to eliminate or minimize the leak through the stub tube crack under all normal operating conditions. The sleeve however should not affect the functioning of the CRD drive nor its normal thermal growth parameters. Further, the mechanical loads on the stub tube by the sleeve should not affect the structural integrity of the CRD penetration nor cause the crack to propagate. In this regard, a finite element analysis showed that with installation of the sleeve, the stresses and deformations induced in the stub tube and CRD housing remain within acceptable limits.

Qualification Tests

Extensive pressurization tests were performed to qualify the mechanical sealing sleeve. These included tests to determine the effect of: seal parameters, preloading, stub tube eccentricities of up to 2 mm (0.080 in.), wall imperfections of up to 1.27 mm (0.50 in.) deep, and seal displacement. Pull tests of up to 2085 kg. (4600 lbs.) indicated that the sleeve is tightly secured to the stub tube and will not become a loose part inside the reactor. Of special interest was the thermal-hydraulic testing of sleeve/tube mockups to demonstrate sealing performance under the following conditions:

1. Cyclic pressure tests from ambient conditions to reactor pressure and temperature conditions of 8.85 MPa (1275 psig) and 577°K (580°F) in an autoclave.

2. A 139°C (250°F) thermal gradient across the CRD housing, stub tube and sleeve (to simulate CRD cooling) for several days.

3. Accumulated vibration cycles of 10^7 cycles at 4.5g and 130 Hz.

4. Extended life tests of over a year.

Successful results from these tests provided the confidence to install the sleeve. The mechanical sleeve seal samples from the life tests, which were performed in parallel with actual in-plant operation of the installed sleeve, were sectioned and these showed no deterioration of the seal packings.

Installation Method

An important consideration in the choice of the mechanical sleeve was the ability to install the seal without significant perturbation in the plant operation. The installation was performed with the CRD mechanism removed and the reactor vessel filled with water up to the reactor flange. For single stub tube installation, only a few fuel bundles directly above the subject stub tube needed to be unloaded.

The remote installation tool (Figure 4) consisted of an 18.3m (60 ft.) long mast at the end of which the seal was attached. Built into the tool was a hydraulic ram to compress the packing and a hydraulic drive motor to rotate and torque down the nut. Various parameters were monitored to obtain confidence that a good installation was effected prior to startup of the plant.

Other installation tools included a stub tube measuring device, stub tube cutter and honing tools. The measuring device consisted of four fingers fitted with strain gauges that measured the stub tube O.D., shrinkage profile, its eccentricity from the CRD housing and any wall imperfections. The stub tube cutter was designed to machine the tube O.D. back to concentricity with the CRD housing. However except for minor weld splatter, the as-built conditions (2.54 mm (0.100 in.) eccentricity and 2.54 mm (0.100 in.) O.D. shrinkage) of the stub tubes that were sealed were all within the sealing capability of the sleeve such that these tools were not required.

RESULTS AND FUTURE PLANS

The first installed mechanical sealing sleeve has been in operation for 3 years and the second for a one and a half years with no loss of seal integrity. Results of UT inspections on the stub tube with the first sleeve showed the crack growth to be arrested. Whether this was due to relaxation of the residual stresses, by sufficient advance of the crack, is not known, but it may be considered that the sleeve removed the aggressive water environment from the crack area. A third stub tube is being evaluated for sleeve installation during the next outage should the existing crack grow to exceed the wall thinning criteria (less than 5 mm (0.020 in.) remaining wall thickness).

The operating experience with the mechanical sleeve suggests that it may have an indefinite life or at least that it will perform satisfactorily for several operating cycles. Possible improvements are being studied to increase its operating life and thus serve as a long term solution. These improvements are better materials, improved preloading, and closer fit between stub tube and sleeve through stub tube preparation.

SUMMARY

In summary, a mechanical sealing sleeve repair method was quickly developed and qualified to provide a successful and timely response to the stub tube cracking problem at the Santa Maria de Garona BWR plant with little impact on reactor availability. The operating experience of the mechanical sleeve indicates that it is structurally adequate and it effectively eliminates the reactor vessel leak.

A systematic and measured approach to the problem that addressed the specific needs of the plant resulted in a cost effective solution. In the case of a plant that has entered operation, this method represented a less extreme solution than welding which requires further extensive development for the remote repair conditions of this applications.

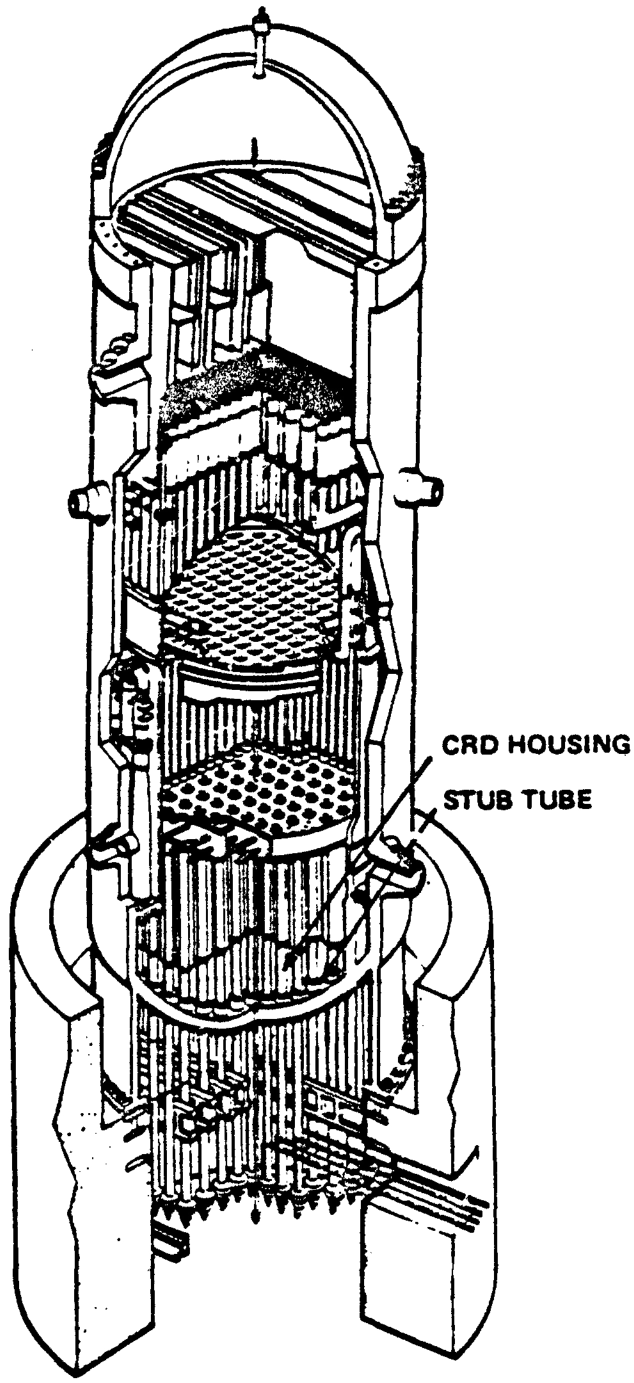

Figure 1 Boiling water reactor cutaway

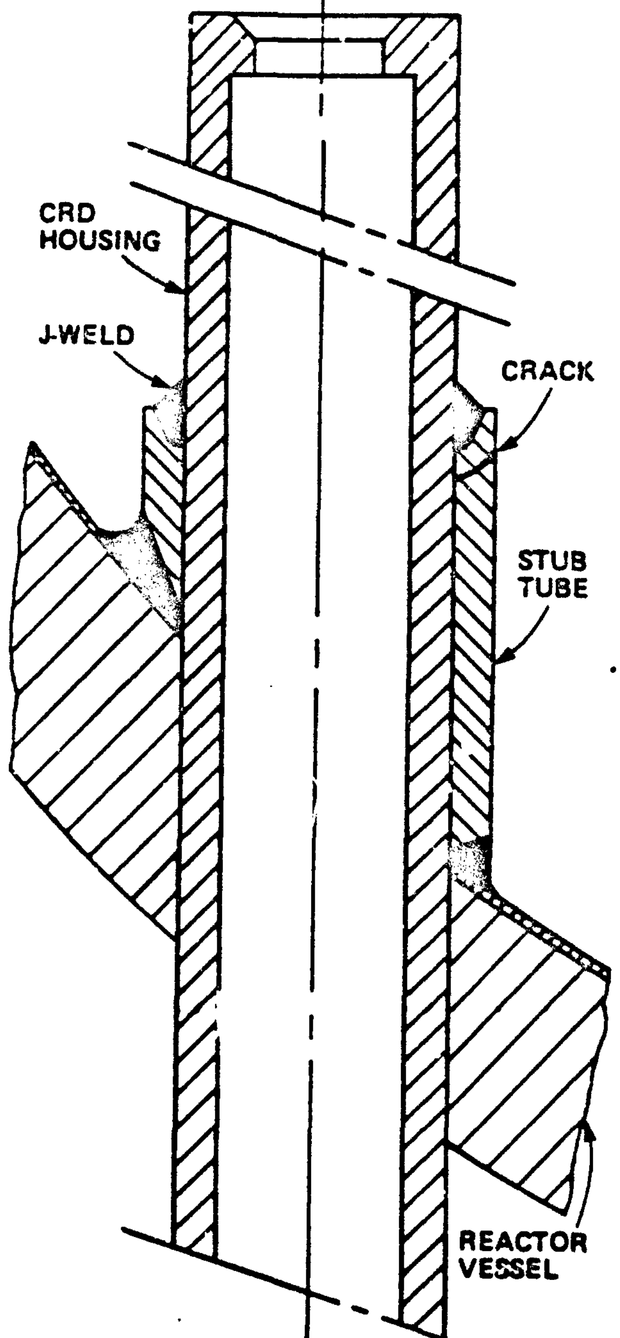

Figure 2 Sta. Maria de Garona
Stub tube/CRD housing

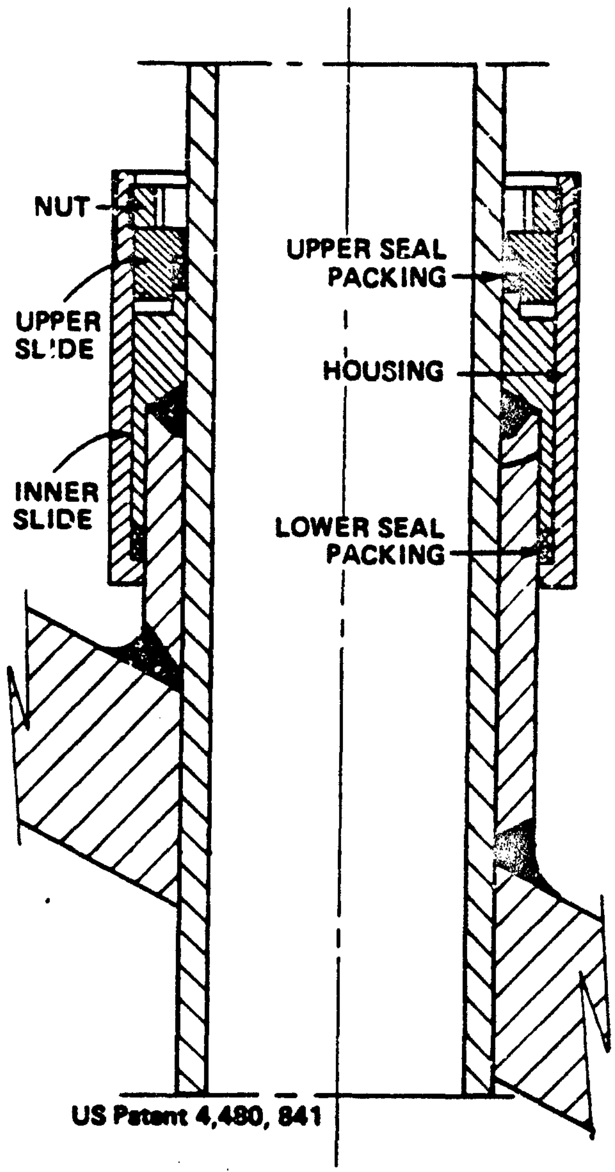

Figure 3 Mechanical sealing sleeve

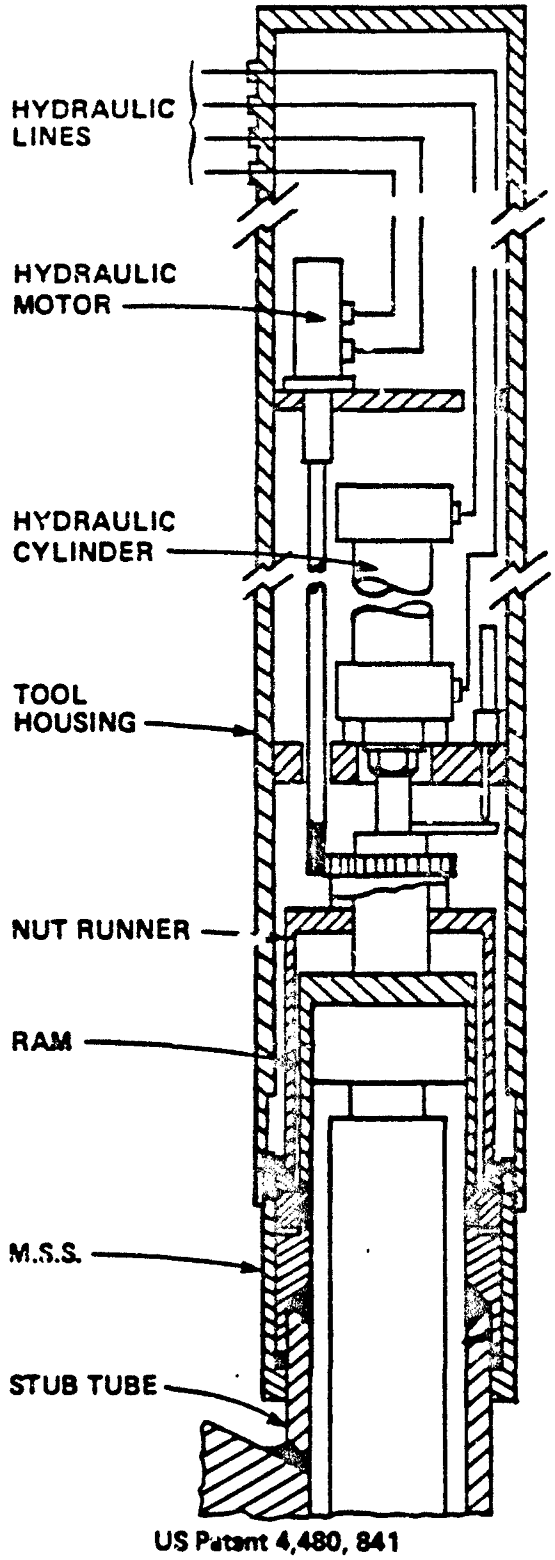

Figure 4 M.S.S. installation tool

MAIN STEAM ISOLATION VALVE SEAT REPAIR

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Electric Power Research Institute

1. Abstract

Main steam isolation valves (MSIV), installed in boiling water reactor (BWR) nuclear power plants, are periodically subjected to a seat leakage test to satisfy Technical Specification requirements imposed on the power plant. To achieve the leak rate criteria of this test it is often necessary to lap the seating surfaces of the valve. Each lapping operation performed removes some of the hardfacing material which has been applied by welding to the seat of the valve body to resist wear, corrosion, and similar phenomena. Eventually, the base material of the valve body may become exposed and therefore require the re-application of the hardfacing material to restore the valve to its original condition. Performing this work while the valve remains installed in the piping system is considered cost effective as compared with the removal and reinstallation of these large valves to achieve the repair.

EPRI Project RP 2186 was initiated with the overall objectives of: a) identifying and evaluating methods for restoring the hardfacing of installed MSIV seats; and b) demonstrating promising techniques and equipment to interested utilities. In the course of the work it was determined that equipment developed and currently in use in Japan performs all necessary functions to restore the MSIV seats to their original configuration.

Additionally, welding systems commercially available in the U.S. were used to demonstrate satisfactory deposition of the hard facing material but these systems lack the supportive equipment needed to achieve the total repair. The availability of the fully developed systems from Japan, or systems requiring further development for use in US power plants is dependent on a commitment by the utility industry to fund such activity based on the perceived need to perform the repair at some future date.

2. Introduction

A paper entitled "Application of Hardfacing Materials in Installed Main Steam Isolation Valves" was presented at the Conference for Maintenance Welding in Nuclear Power Plants/II conducted in Chicago in August of 1982. The paper defined the factors which must be considered in resolving the problems of the in situ application of hardfacing materials in large valves and additionally described the methodology to be applied by the Electric Power Research Institute (EPRI) in pursuing a research project in this area of welding technology.

This research effort is now complete and the results have been published in the final report, EPRI NP-3926, "In situ Application of Hardfacing Materials in Main Steam Isolation Valves" (March 1985). This paper presents a summary of the work performed and the results obtained in the several phases of this research effort.

3. PROCEDURES

Three activities were initiated to pursue the overall objectives of the project. These consisted of:

a) An investigation to determine and evaluate existing equipment and processes for depositing hardfacing materials on installed MSIV seats by remote control;

b) The demonstration and evaluation of other welding systems which could be utilized in the repair process; and

c) The technology transfer of information and results through a seminar and demonstration of materials, equipment and methods.

The initial activity was carried out to determine what programs have been conducted or are presently underway in the U.S. and other countries concerning the application of Stellite 21, which is the predominant hardfacing material used in these valves. A literature search was conducted and contacts made with MSIV manufacturers, welding equipment manufacturers, nuclear reactor service firms, and electric utility companies who were potentially involved in the development of such processes and equipment. The investigation disclosed two firms in Japan that had developed and utilized the necessary processes and equipment. On the basis of this information, arrangements were made to visit these firms and to observe the application of hardfacing in actual practice.

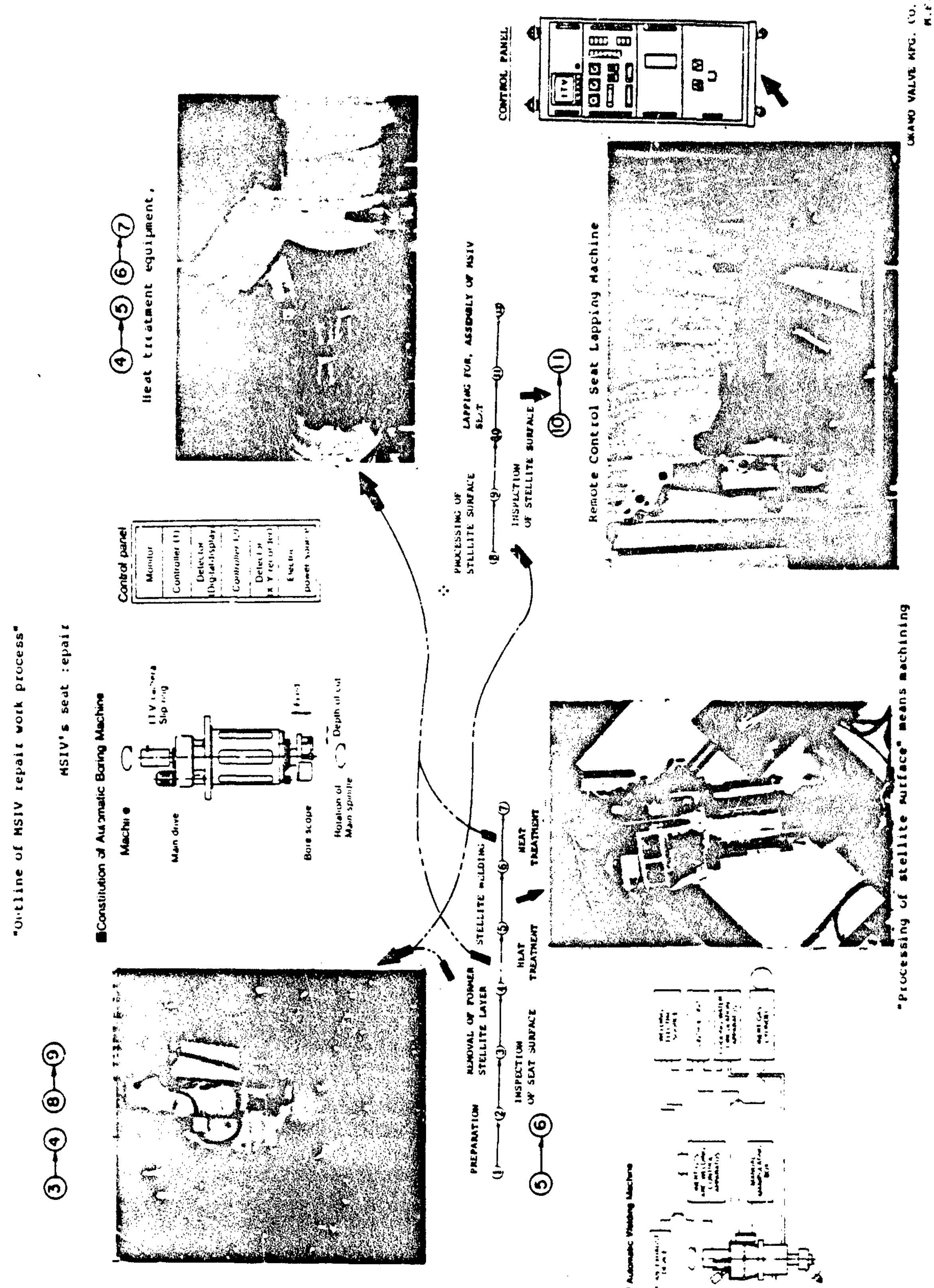
The next activity involved the development of procedures for, and the demonstration of, a controlled heat input gas tungsten arc welding process for the deposition of the hardfacing material. The process, known as the Dabber TM welding method, was anticipated to provide the advantages of minimal distortion of the seating surface and minimal dilution of the hardfacing material by the base material. Thus, higher hardness of the deposit, due to the low heat input to the base material, would result. Demonstration of the process was performed under conditions of welder position which simulate the field positioning of the MSIV.

The final activity also involved the development of welding parameters and the demonstration of manual and automatic processes for the deposition of the hardfacing material using commercially available welding equipment. The gas tungsten arc welding (GTAW) process was employed exclusively in the demonstrations. A seminar/workshop was also conducted as an avenue of technology transfer of information obtained in the several activites.

4. RESULTS

A. The organizations contacted in Japan as a result of the state-of-the-art search conducted by the EPRI contractor are: a) the Eagle Valve Co. of Okayama, whose representation in the U.S. by Atwood and Morrill Co. of Salem, Mass. is being negotiated; and b) the Okano Valve Co. of Kitakyushu City, whose representation in the U.S. by the General Electric Co. (Nuclear Energy Div.) of San Jose, CA is also being negotiated.

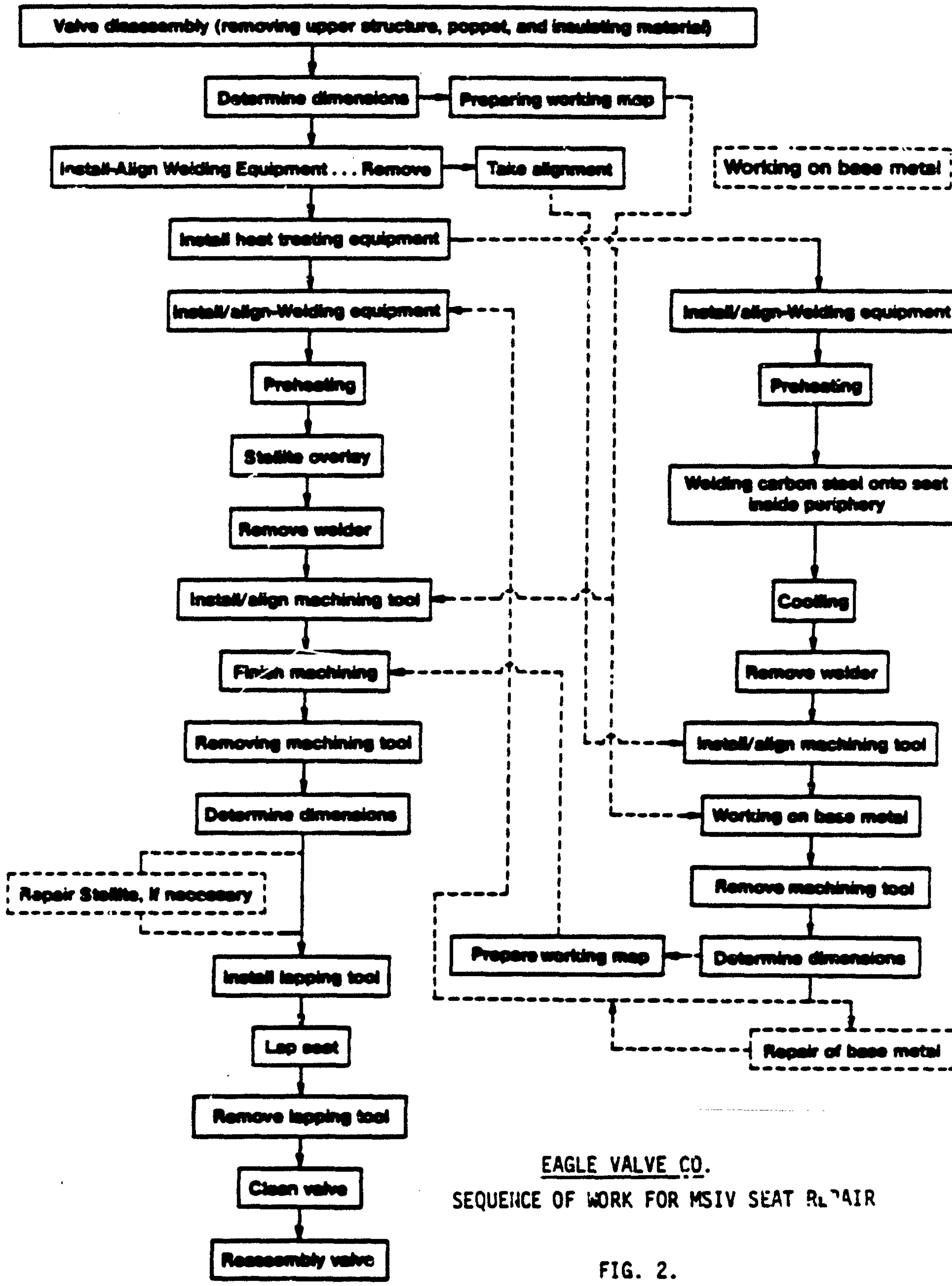
Both organizations have developed and used, in field applications, total systems for the in situ re-deposition of hard facing material in main steam isolation valve (MSIV) seats. The systems are very similar and each system uses separate remote controlled and monitored machines/equipment for: 1) machining of the base material in preparation for deposition of hardfacing; 2)heat treatment; 3) deposition of hardfacing; 4) finish machining of the valve seat; and 5) final lapping of the seat. Fig. 1 shows the four basic components of the Okano system. Figure 2 and 3 show the sequence of work activities and the welding equipment of the Eagle system. Both systems utilize the GTAW process for deposition of the hardfacing material in multiple pass application. Differences in the specification of filler wire are stated to be proprietary by these organizations. No comparable total system exists in the U.S.



OKANO VALVE CO.
EQUIPMENT FOR MSIV SEAT REPAIR

FIG. 1

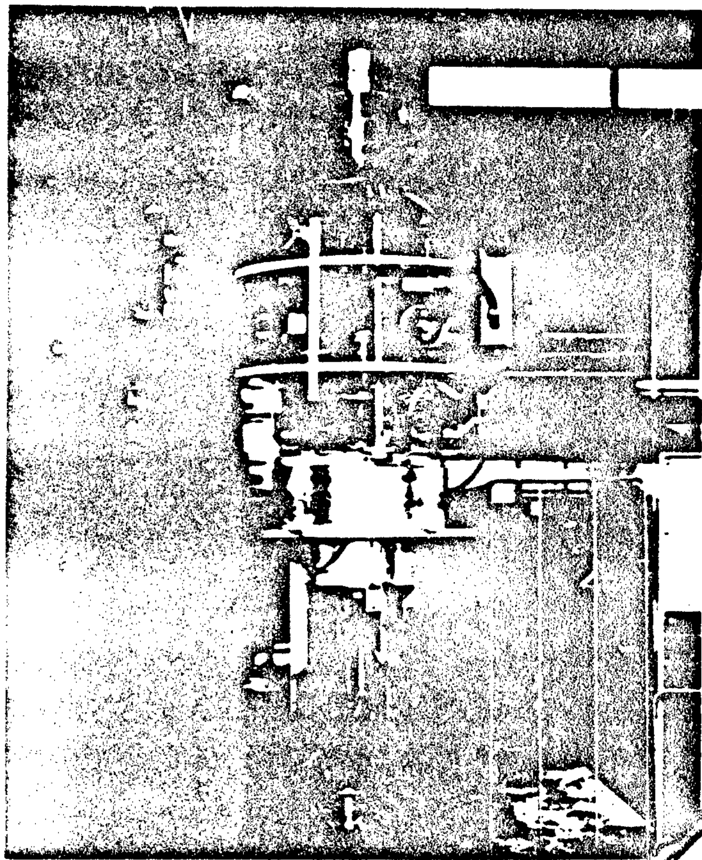
Procedure for Field Welding (Typically)



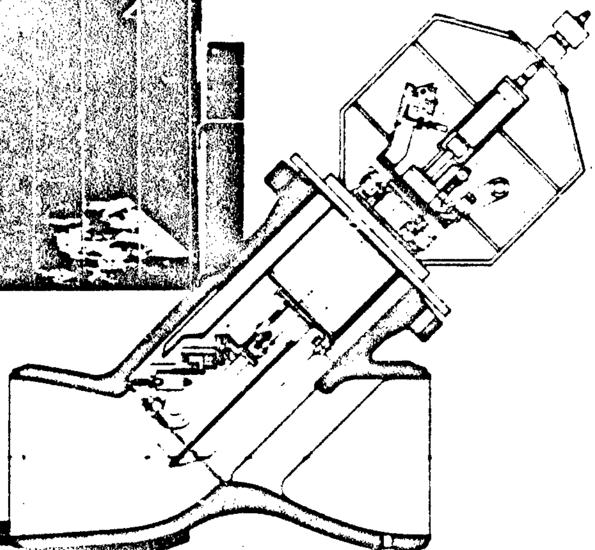
EAGLE VALVE CO.

SEQUENCE OF WORK FOR MSIV SEAT REPAIR

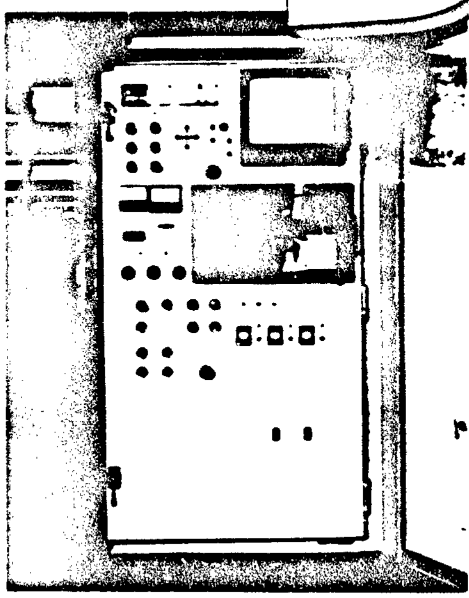
FIG. 2.



View of Welding Equipment



Welding Equipment in Valve Body



Control Panel

EAGLE VALVE CO.
WELDING EQUIPMENT
FOR
MSIV SEAT REPAIR
FIG. 3.

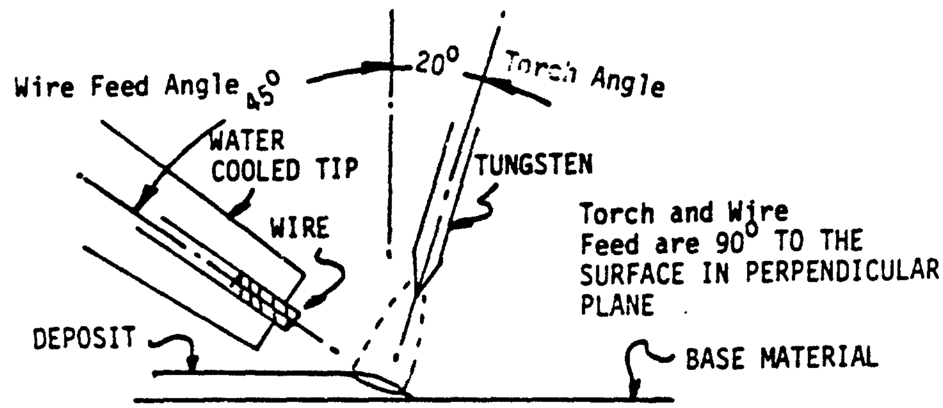


FIG. 4. Dabber synchronized current pulsation in the backhand mode. With current in low pulsation, the weld puddle and heat are directed at the deposited weld metal.

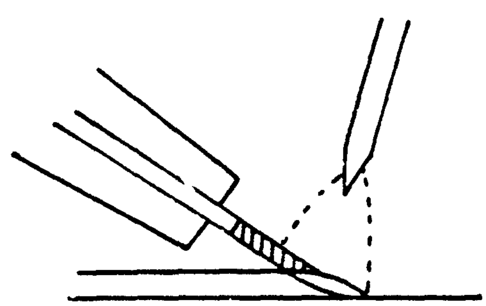


FIG. 5. Dabber synchronized current pulsation in the backhand mode. With current in peak pulsation, the heat is directed at the deposited metal, and the weld puddle flows onto the base material.

FIG. 4 & 5 HOBART BROS. CO. "DABBER" PROCESS

B. Hobart Brothers Co. of Troy, Ohio utilized a standard Dabber TM welding system (typical of those used for weld repair of jet engine knife edge seals) to develop weld parameters for the deposition of Stellite 21 on to mild steel in welding positions which simulated field conditions of the MSIV. In the Dabber process the wire is fed through a water cooled copper tip and is retracted from and inserted into the weld puddle in synchronization with low (~40A) and high (~200A) levels of current pulsation at the welding torch. Figures 4 and 5 illustrate the configuration. Deposition in the backhand mode was found to decrease the dilution of the hardfacing with the base material, resulting in higher hardness of the deposit as compared with travel in the other direction. A total of twelve beads, as shown in Fig. 6, was required to produce sufficient deposit of material for the valve seat using .062 inch dia. wire.

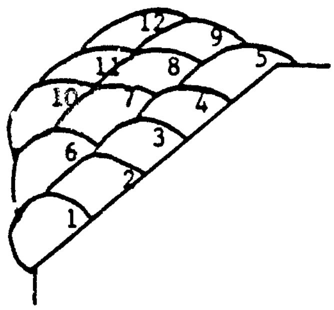
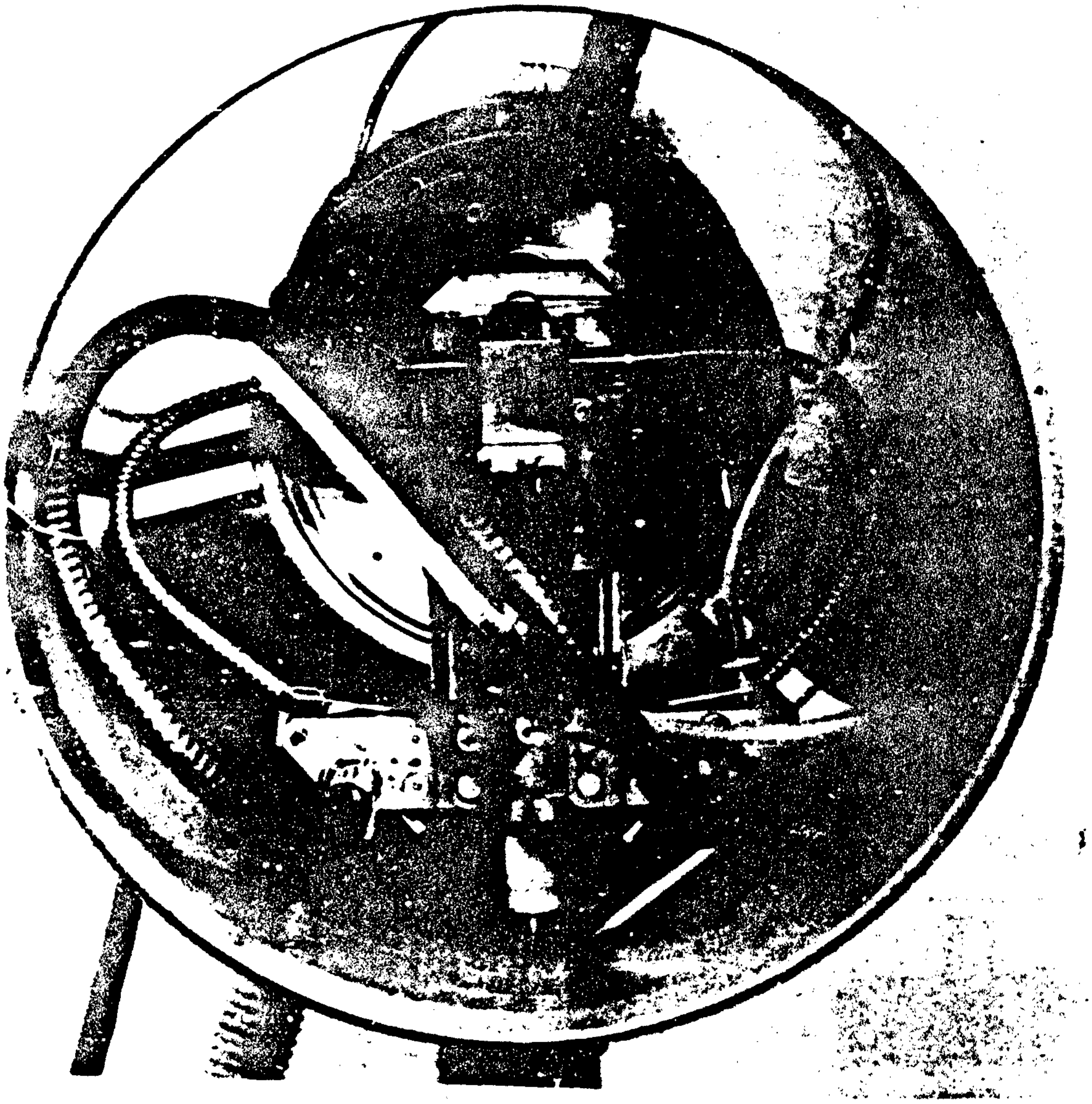


Figure 6. Bead sequence of the weld deposit for Dabber Process.

No preheat or post heat was applied and heat-build-up of the base material due to the welding process was negligible. Fluorescent dye penetrant testing of the as deposited and the finish machined material was performed. No indication of cracking was found but some minor (<1/32") indications of inclusions or porosity were disclosed. Specimen sections were prepared and hardness measurements taken at the mid-depth of the depositon. The average hardness of the specimens ranged from Rc 30 to Rc 34. This level of hardness is desirable and is higher than the suppliers published value of Rc 27. Achieving this hardness is attributed to the very limited dilution of the hardfacing material with the base material.

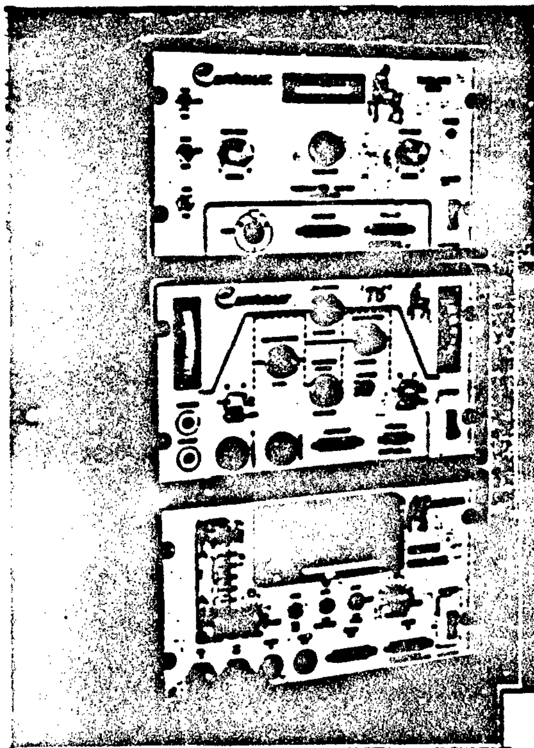


ARC MACHINE MODEL 43 I.D. HEAD POSITIONED IN PIPE

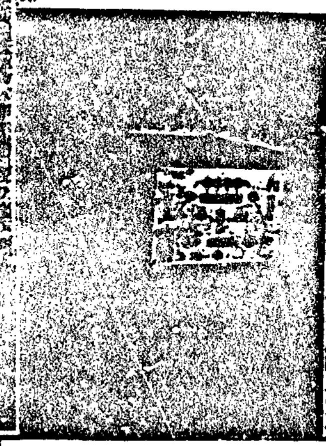
FIG. 7

Centaur systems are modular welding control units which inter-connect to create custom tailored automatic gas tungsten arc welding systems. Our power supply produces a most accurate and narrow pulsed arc with beam-like results. Our fully transistorized closed loop system maintains weld current to 1% of set value.

Control modules include:
- Digital weld sequence timing and weld gas control.
- Automatic arc length control.
- Wire feed and fixture drives.

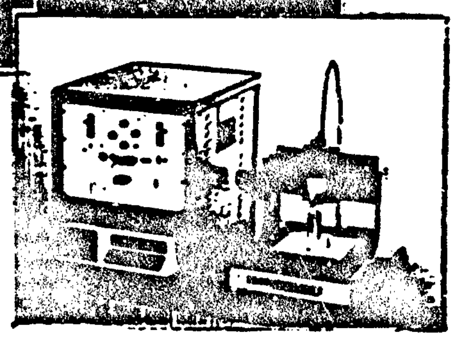


CENTAUR



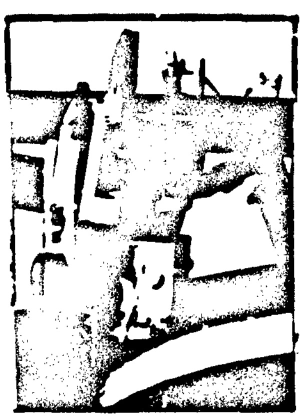
TUBE WELDER

Tube Welder is an automatic welding system for producing high speed tube welds. Unit allows adjustment of all weld functions to suit a wide variety of tube sizes and wall thicknesses. Adjustable sequenced currents provide perfect automatic heat control, 360 degrees around the tube.



The Computapath is a computer controlled automatic gas tungsten arc system for making high speed, low heat welds in a variety of work piece shapes. Powerful stepping motors produce weld speeds of up to 100" a minute around any corner radii with no overshoot or inaccuracies.

COMPUTAPATH



FULL-FUNCTION IN-PLACE WELD HEAD

I.D. WELD HEAD

DIMETRICS, INC. WELDING EQIP.

FIG. 8.

C. Additional weld parameter development, which was conducted at EPRI's Maintenance Equipment Application Center (MEAC) in Charlotte, NC, demonstrated the feasibility of using commercially available sytems for both manual and automatic deposition of the hardfacing material. Both pre-heat (300 F) and non pre-heat test plates were utilized in preparing the manually applied specimens using both standard and pulsating GTAW equipment and methods. The pre-heated test plate yielded a more uniform weld contour than did the non-preheated plate, and the pulsating GTAW process produced an average hardness of deposited materials of Rc29 as compared with an average hardness of Rc18 produced by the standard equipment.

The automatic welding equipment utilized in this activity is normally used for pipe I.D. remote welding (Figures 7 & 8). The hardfacing material, normally furnished in a minimum diameter of .0625 inch, was drawn to an .045 inch diameter to be compatible with these "off-the-shelf" welding heads. Mechanical equipment to align and position the welding head in the main steam isolation valve is needed to adapt this equipment for the intended repair service together with equipment for preliminary and finish machining and lapping of the repaired surfaces.

A seminar was conducted at the MEAC to provide interested utilities with information which had been developed in the various activites. Presentations were made by the following organizations: Polymet Corp of Cincinnati, Ohio and Cabot Corp of Kokomo, Ind. producers of the hardfacing alloy wire; ARC Machines of Pacoima, CA, Dimetrics Inc. of Diamond Springs, CA., and Hobart Brothers of Troy, Ohio, producers of welding equipment; and the Eagle Valve Co of Okayama, Japan and the Okano Valve Co. of Kitakyushu Japan, producers of total systems for the in situ repair of MSIV seats.

5. CONCLUSIONS

The total systems offered by the Eagle Valve Co. (together with Atwood and Morrill Co.) and the Okano Valve Co. (together with General Electric) for the preparation, deposition, and final finishing of hardfacing materials for MSIV seats are considered to be fully developed, commercially available systems for performing this repair. Welding equipment available in the U.S. provides a viable alternate to the use of these systems for at least one phase of the repair and may provide improved quality and hardness of the deposited metal together with reduced application time if developed to a comparable level. Such development effort however, requires that a market be defined by the electric utility industry for the use of such equipment.

6.ACKNOWLEDGEMENTS

The author wishes to acknowledge the contribution of the following persons who contributed to the work presented in EPRI Report NP-3926, "In situ Application of Hard Facing Materials in Main Steam Isolation Valves". This report was used as a reference in the preparation of this paper.

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Mr. Bill Westfall, Hobart Brothers Co.
Mr. K. Brittain, J.A. Jones Applied Research Co.
Mr. R. Dahlke, "
Mr. L. Harrell, "
Mr. T. French, "

WP:B-04

APPLICATION OF REMOTE WELDING FOR REACTOR MAINTENANCE

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Ontario Hydro, Canada

INTRODUCTION

The remote welding and inspection system was devoloped for the reactor rehabilitation work at Ontario Hydro's Bruce Nuclear Generating Station "A". All units operated by the Utility are CANDU (Canadian Deuterium Uranium) reactors which use pressurized heavy water as the primary coolant. Each unit at Bruce "A" is rated at 750 MW(e). The core contains 480 fuel channel assemblies (Figure 1) holding fuel bundles within their pressure tubes. Each fuel channel assembly is welded to the reactor structure at the west end, whereas at the east end, it is supported on bearings which permit axial travel. This compensates for its axial elongation. The magnitude of the axial elongation of the zirconium - 2-1/2% niobium pressure tube would cause the channel to exceed the bearing travel limit if corrective maintenance is not undertaken.

The corrective action involves severing the weld which holds the fuel channel in place, shifting the channel axially to compensate for the elongation and reattaching with a 5/32 inch (4 mm) leg, fillet weld. This weld is primarily a structural rather than a pressure retaining weld in that it has to withstand loads imposed by the fuelling machine during fuelling and defuelling of the fuel channel. This cut, shift and reweld operation needs to be carried out on all 480 fuel channel assemblies and is estimated to require 85 days per reactor.

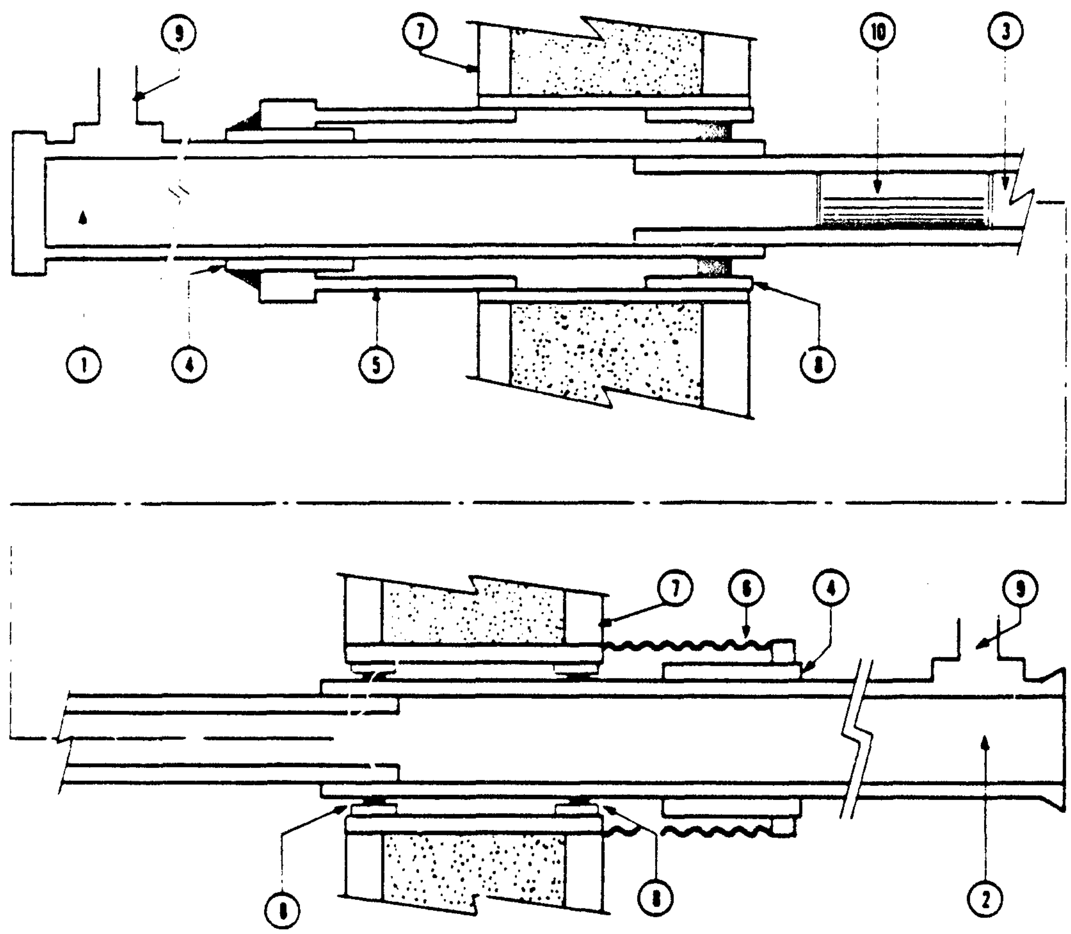
Sever constraints are imposed on the tooling as they are required to function in a restrictive environment. These can be broadly classified into two categories: environmental and spatial.

The work is carried out inside the reactor vault where the radiological conditions after shutdown present a restrictive environment (Table 1). The spatial constraints are imposed by the reactor geometry, the operating envelope (Figure 2) available for the tooling is bounded by the reactor tubesheet, innermost feeder pipe and adjacent stop collars. In addition, the arrangement of feeder pipes allows limited access space through which all tooling is inserted.

GAMMA FIELDS (MR/HR)	
Average Field at Reactor Face	260
Maximum Field at Operating Area	310
General Field on Vault Floor	80
AIRBORNE CONTAMINATION (MPCa)	
Average Tritium Level	40
Average Particulate	100

Table 1 Vault Environment After Shutdown

Radiation exposure during the outage is reduced by controlling and inspecting the weld remotely. However, some personnel must be present at the reactor face for tool insertion and retrieval. They are housed in a shielding cabinet which attenuates radiation levels by a factor of ten. They are in voice communication with those at the remote control centre at all times. The shielding cabinets can travel on the existing fuelling machine bridge to access all fuel channels. Breathing quality air pumped into the shielding cabinet maintains a slight positive pressure preventing ingress of airborne contamination.



1 West End Fitting
2 East End Fitting
3 Pressure Tube
4 End Fitting Sleeve
5 Stop Collar
6 Bellows
7 Reactor Face Tube Sheet
8 Bearing and Journal
9 Feeder Pipe
10 Fuel Bundle

Figure 1 Fuel Channel Assembly Details

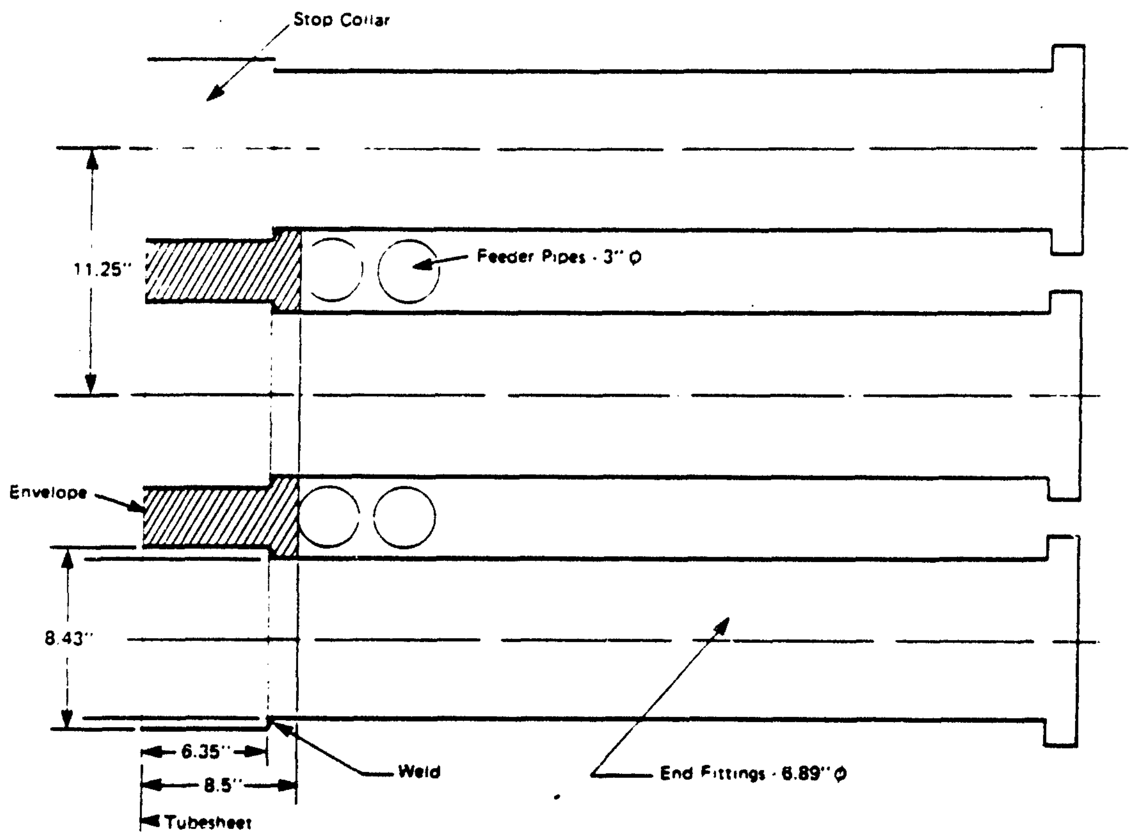


Figure 2 Operating Envelope for the Tooling

The tooling required to prepare and execute a weld can be grouped into four sub-groups:

(a) An orbiting platform or tool carrier to carry the cut, clean and weld tools heads around the stop collar.

(b) Cutting, cleaning and welding tool heads and associated umbilicals.

(c) A fibrescope - video system to provide a remote view of the work area.

(d) A welding power system and associated controls.

The operations involved in carrying out the welding process are:

(a) Installation of the weld head onto the tool carrier and connection of the service umbilicals.

(b) Verification of initial head positioning prior to striking of the arc.

(c) Full 360 degrees pre-weld inspection.

(d) Welding.

(e) Full 360 degrees post-weld inspection.

(f) Disconnection of services and removal of weld heads from the tool carrier.

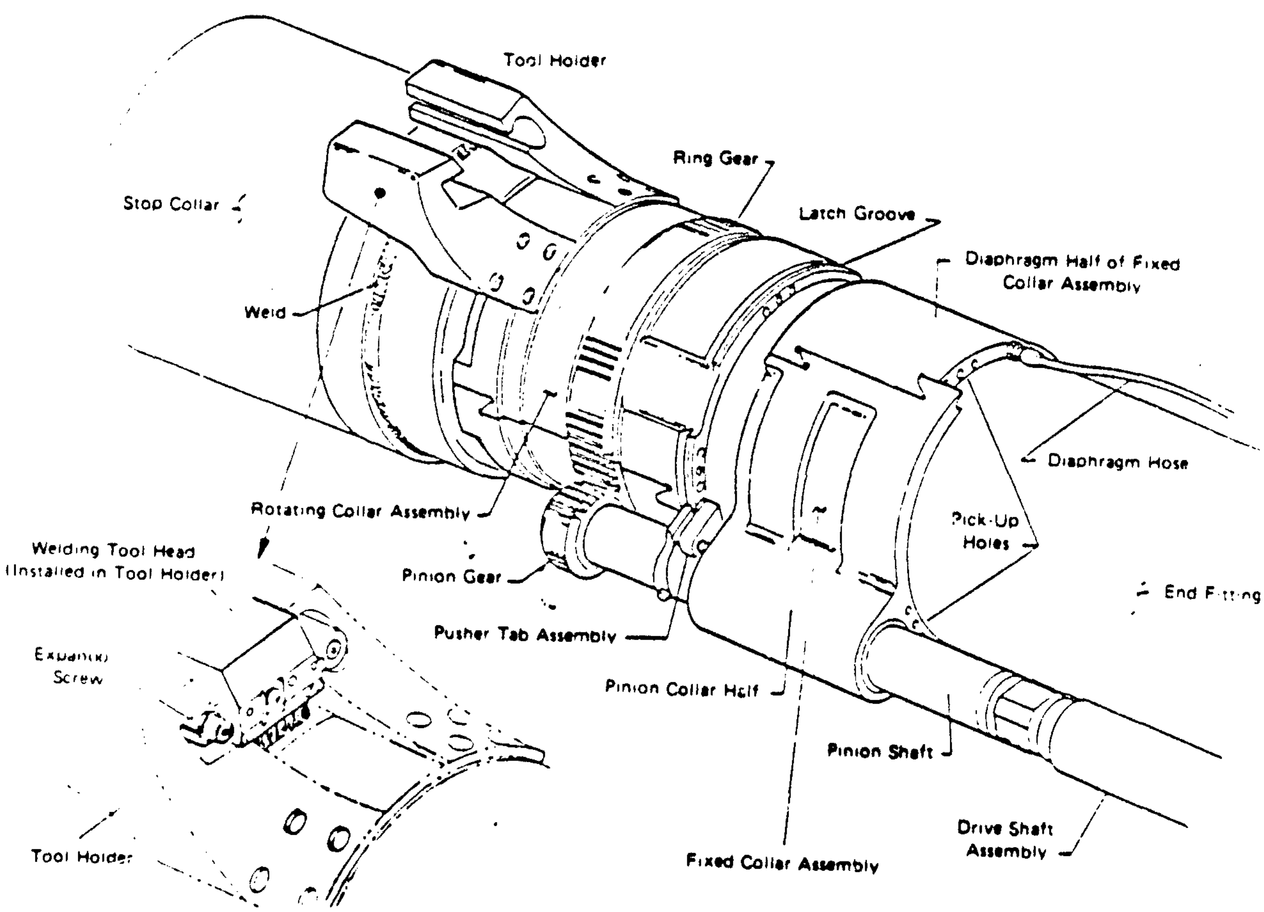


Figure 3 High Torque Tool Carrier

To accomplish the control and qualification of tho weld from the remote location, the following requirements were imposed on the tooling systems:

The integrated fibrescope-video system must be capable of assisting in confirming the weld head position prior to striking the arc, specifically the arc gap setting and the filler wire delivery; be capable of providing a picture of the complete weld area so that the operator can view the process and if necessary take corrective action; be capable of generating a picture with adequate resolution for identifying unacceptable weld flaws (0.030 inch (0.76 mm) and greater) and heat zones.

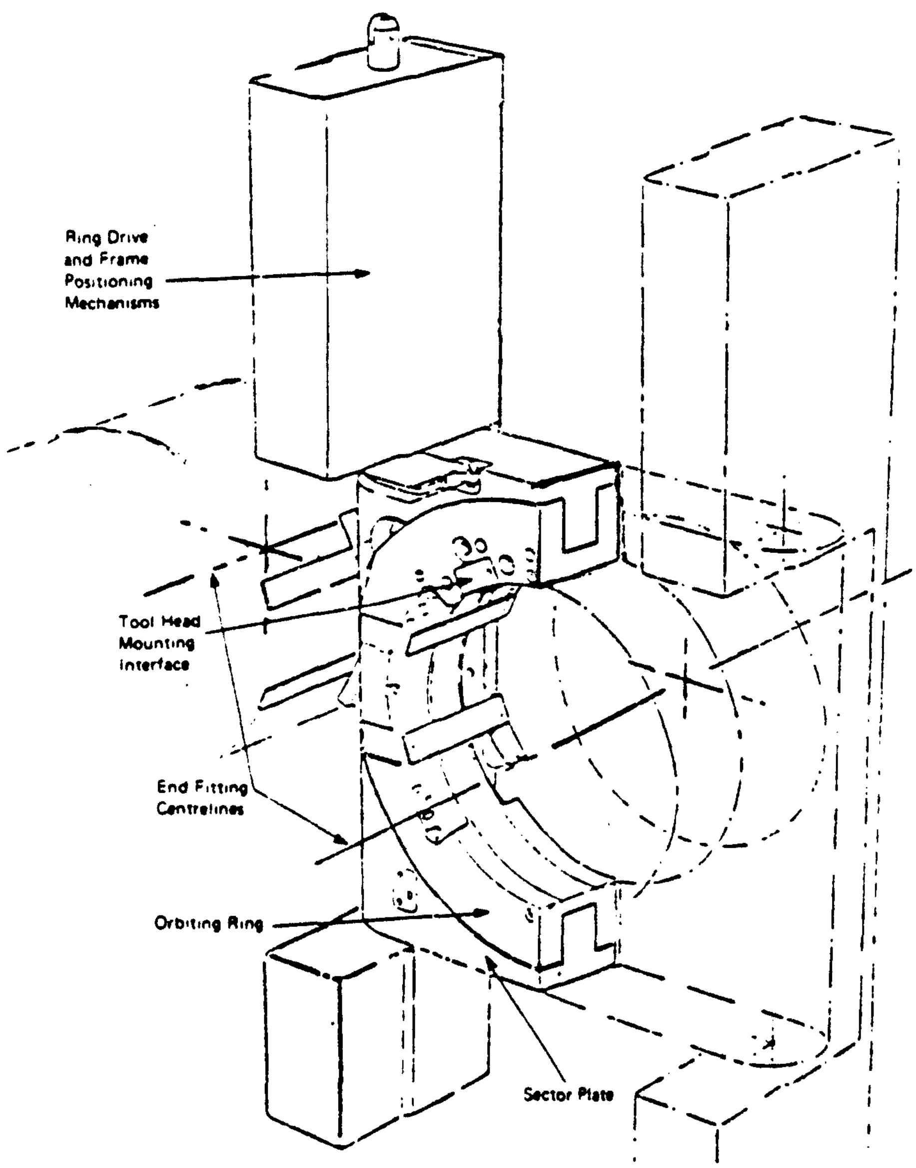


Figure 4 Low Torque Tool Carrier

Two major groups of tooling have been designed, the High Torque tooling and the Low Torque tooling. The designations refer to the relative amounts of torque required to orbit the tools during the cutting operation for detaching the fuel channel from its stop collar. The High Torque tool carrier system (Figure 3) consists of a rotating collar with an "ear" to accept the tool heads. The motive power is supplied from within the shielding cabinet. The Low Torque tool carrier system (Figure 4) consists of a computer controlled tool orbiter operated from the remote control centre.

Two types of weld heads were designed, a water cooled head as part of the High Torque system and a passively air cooled head as part of the Low Torque system. Both systems use the same welding machine and control panel.

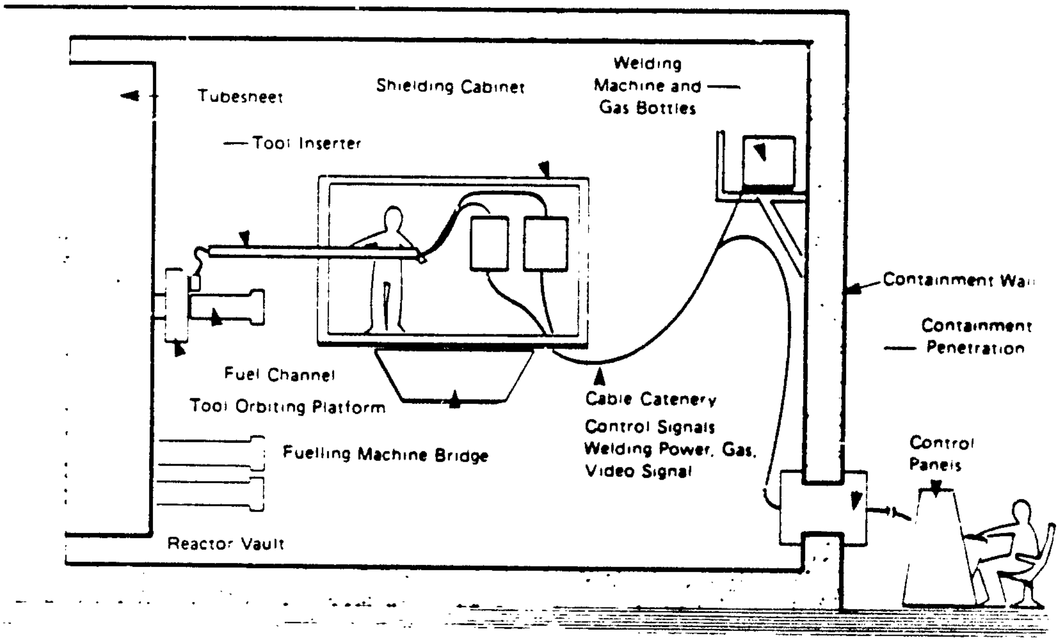


Figure 5 System Block Diagram

SYSTEM CONFIGURATION

Figure 5 is a block diagram showing the various components of the overall system. The weld head in each system is interfaced with the welding system and the video system inside the shielding cabinet. The interconnecting umbilicals run inside the inserters and provide routing for the cover gas, welding current, filler wire, the coolant (for the water cooled head) and the fibrescope. Separate inserter-probes are used to view the weld tool positioning prior to initiating the welding process.

AIR COOLED WELD HEAD

The head assembly (Figure 6) consists of two parts; a rigid subassembly which includes the pivot spindle, guide pins and the clamping screws for mounting the head and a floating subassembly which includes the torch assembly, the follower system to maintain correct torch positioning, the wire feed nozzle and bracketing to mount the fibrescope.

The torch houses the tungsten electrode and the gas cup directing the Argon cover gas. To maintain flow conditions with adequate cover, a stainless steel wire mesh screen (60 x 60 x 0.008 inch dia wire, 27.2% open) is used inside the gas cup. The torch subassembly itself is held in place by a bracket and clamp arrangement that allows torch position adjustments during tool refurbishment. The electrode is oriented 45 degrees to the fillet with 0 degree lead angle.

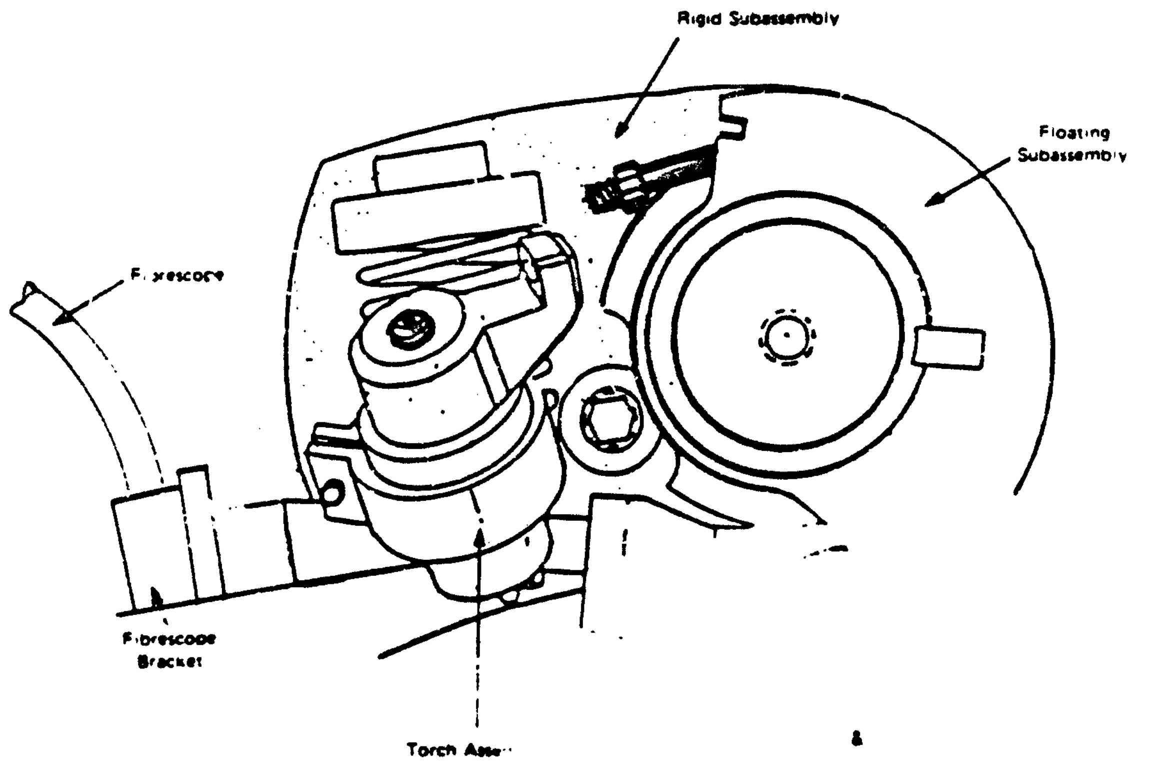


Figure 6 Air Cooled Weld Head

The tracking and positioning is achieved by the radial and axial guides. The spring loaded floating subassembly positions these hardened guides, maintaining the proper gap setting with the reference to the work piece over the range of tool carrier positioning tolerances.

The fibrescope clamp and bracket assembly, mounted on the floating subassembly, insulates the fibrescope and allows its positioning to provide the optimum view, trailing the weld puddle by 1-1/2 inches (38 mm).

The flexible cable assembly (umbilical) consists of a covergas/power line, nylon/teflon wire feed guide tube and a nylon fibrescope guide tube. The umbilical is 25 feet (7.6 m) long and is covered in a teflon spiral wrap for protection from heat and mechanical damage. The umbilical passes through an aluminum cover tube inserter and is paid out manually during the welding process.

WATER COOLED WELD HEAD

The weld head assembly (Figure 7) consists of two major subassemblies; a hollow metal body consisting of an Expando-bolt attachment system and the electrode holder which supports and is part of the torch.

The radial and the axial gauge feet with hardened tips are mounted on the body and facilitate positioning of the head itself on the work piece. The filler wire guide tube, mounted on the body, directs wire to the weld puddle.

The fibrescope bracket mounted on the body provides support and protection to the fibrescope. The arrangement allows for its positioning to provide the optimum view, trailing the weld puddle by 2-1/2 inches (63.5 mm).

The copper electrode holder routes the water and supports the invar collet holder. The assembly is coated with "Hysol" for high frequency insulation and is mounted into the cavity of the body using "Duralco" epoxy resin. The tungsten electrode is installed into the electrode holder with an invar collet. The electrode orientation and the gas cup wire mesh are similar to those used in the air cooled head.

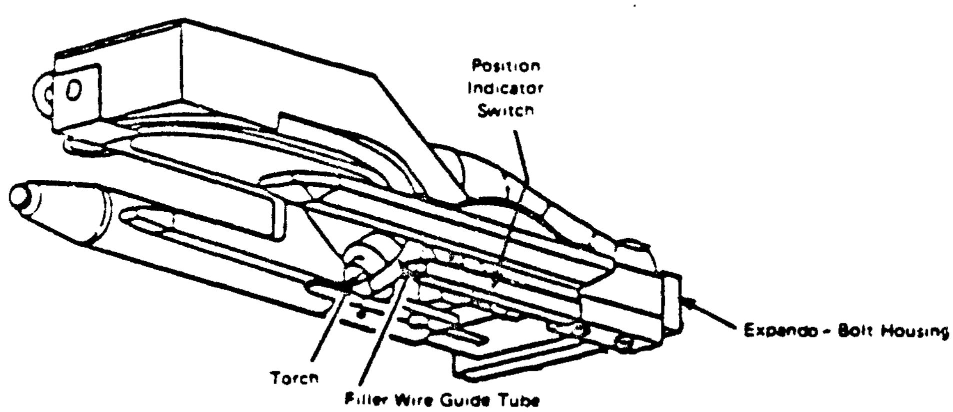


Figure 7 Water Cooled Weld Head

The flexible cable assembly (umbilical) consists of a covergas line, coolant water/power lines (including water return line), nylon/teflon wire feed guide tube and a teflon fibrescope guide tube. The assembly is 28 feet (8.5 m) long and is wrapped in teflon tape to protect the assembly from heat and mechanical damage. This umbilical passes through a steel cover tube inserter and is paid out manually during the welding process.

The tracking is provided by the spring inside the bullet subassembly and the flat spring on the guide tongue, allowing the complete head assembly to float within the collar ear.

WELDING POWER SYSTEM

The system design follows the norm for gas tungsten arc welding, supplying the services and controlling the welding parameters. However, the significant departure is that it is operated remotely. The system provides welding current, cover gas, filler wire to both High Torque and Low Torque tools, as well as coolant water specific to the High Torque tool.

The welding machine is a CYBER-TIG-CT-300-DC-S model manufactured by Hobart Bros., modified for remote control of gas flow, weld start, weld down slope and weld stop. Power input is 600V, 3 phase, 26 Amps at 60 Hz. The rated output is 300 Amps DC at 32 Arc Volts.

The remote 1.5 Hz high frequency unit used for arc initiation is a modified model #1611, manufactured by Hobart Bros.

The wire feed magazine unit is a custom fabricated assembly, with a large proportion of "off the shelf" items manufactured by Magnatech. The assembly consists of a reel storing 65 feet (20 m) of filler wire and a pinch roller delivery system feeding the torch through the umbilical.

The filler wire is AWS E-70-S3 drawn down to 0.018 inch (0.45 mm) diameter and coiled to result in an outside diameter of 0.084 inch (2.13 mm).

The water cooler/circulator is a model 2010 manufactured by Bernard Coolers. The flow is set to 1.3 Imp gal/min at 50 psi.

To minimize line losses, the high frequency unit, wire feed magazine, and water cooler/circulator are located within the shielding cabinet. The welding power supply itself is located inside the reactor vault providing the services to the hardware in the shielding cabinet through cable catenaries.

FIBRESCOPE-VIDEO SYSTEM

The system consists of a fibrescope mounted onto a weld tool head, a light source, a video camera, a video cassette recorder, monochrome and colour video monitors connected through coaxial cables. The colour monitor and the recorder are located in the remote control centre, whereas the rest of the equipment is located in the shielding cabinet. The coaxial cable connecting the two locations runs through the reactor vault penetration. The spatial constraints impose restrictions on the diameter and the bend radius of the fibrescope. The distal tip of the fibrescope is clamped at the weld head, at a standoff distance of 3/4 - 1 inch (19 - 25 mm), providing a view of the weld area to include portions of the stop collar and sleeve for reference. The positioning and the desired view are shown in Figures 8 and 9.

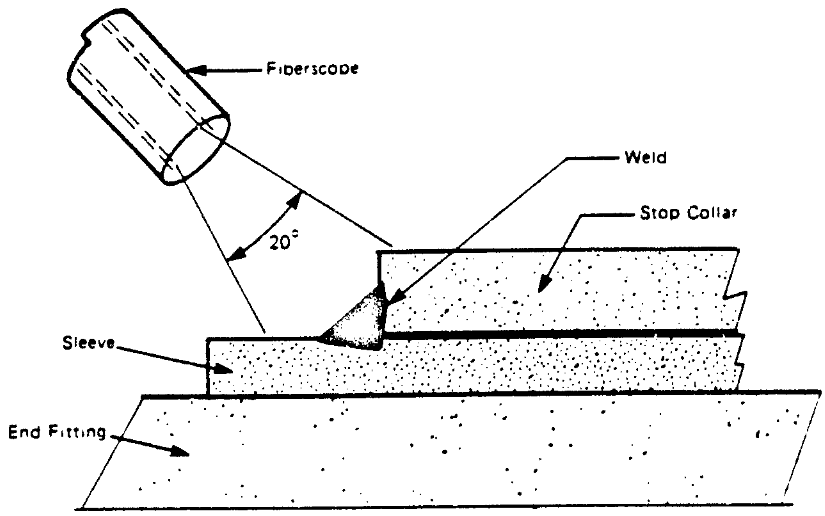


Figure 8 Fibrescope Viewing Position

The fibrescope is a custom design Diaguide model S-006-MT, manufactured by Dianichi Nippon . The fibrescope is made up of a quartz fibre bundle with 6000 image carrying fibres or pixels. The fibrescope is 25 feet (7.6 m) long with an outside diameter (including the image bundle, the light bundle and the protective covering) of 0.236 in (6 mm). The fibrescope is capable of operating in a gamma radiation field of 1 R/hr for an integrated dose of 26,000 Rads. It can withstand 5000 bending cycles to a bend radius of 1.18 inch (30 mm). The required lighting during pre and post-weld inspections is achieved by using a metal halide cold light source. An Ar-49 lens mount adapter is used to connect the fibrescope to the camera inside the shielding cabinet.

The camera used is a SONY BVP-110K, a broadcast quality single tube colour camera with a 0.43 inch (11 mm) to 4.3 inch (110 mm) zoom lens. It has a 0.67 inch (17 mm) TRINICON picture tube and provides 400 horizontal lines of resolution. The gain settings available are 0dB, +6dB and +12dB. It requires a minimum illumination of 80 lux. The video output is 1 volt peak to peak, sync negative at 75 ohms unbalanced.

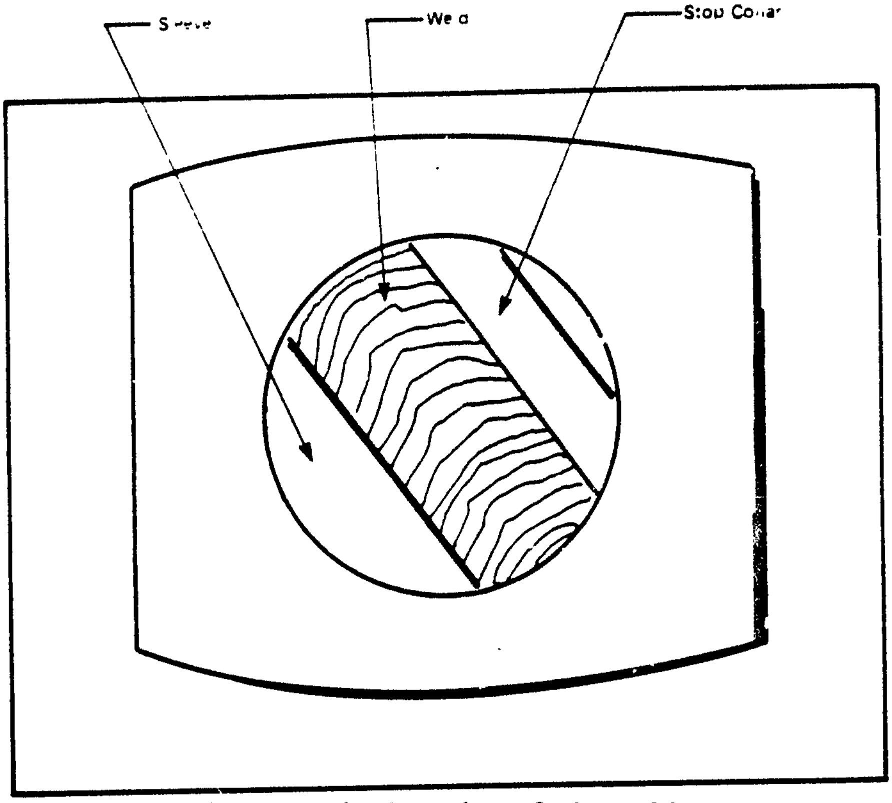


Figure 9 Display View of the Weld Area

The colour monitor located at the remote control centre is a 12 inch (305 mm) SONY model CVM-1270, with 500 TV lines of resolution. A custom fabricated mounting bracket holds the camera in place inside the shielding cabinet and provides alignment at the camera-fibrescope interface. The monochrome monitor inside the shielding cabinet is a 12 inch (305 mm) RCA model TC1112 and is used by the operator within the shielding cabinet to focus the fibrescope and adjust the camera to achieve optimum picture quality.

The VCR used for recording is a JVC model CR-6060U, which uses a professional 3/4 inch (19 mm) tape, providing 240 lines of resolution, operating at a tape speed of 3-3/4 inch (95.25 mm) per second. Though the welds are normally examined directly, the video recorder will be used as the need arises to document certain welds.

CONTROL SYSTEM

The operations are controlled from a remote control centre located approximately 300 feet (91 m) away, outside of reactor containment. The interconnecting cables pass through a specially built 3-foot (0.9 m) diameter containment penetration. The penetration assembly temporarily replaces an existing airlock bulkhead. To reduce worker radiation exposure during installation, prefabricated cables with quick disconnects are used. Further, to avoid electromagnetic interference from the welding machine and other power devices, only control and video signals are passed through this penetration.

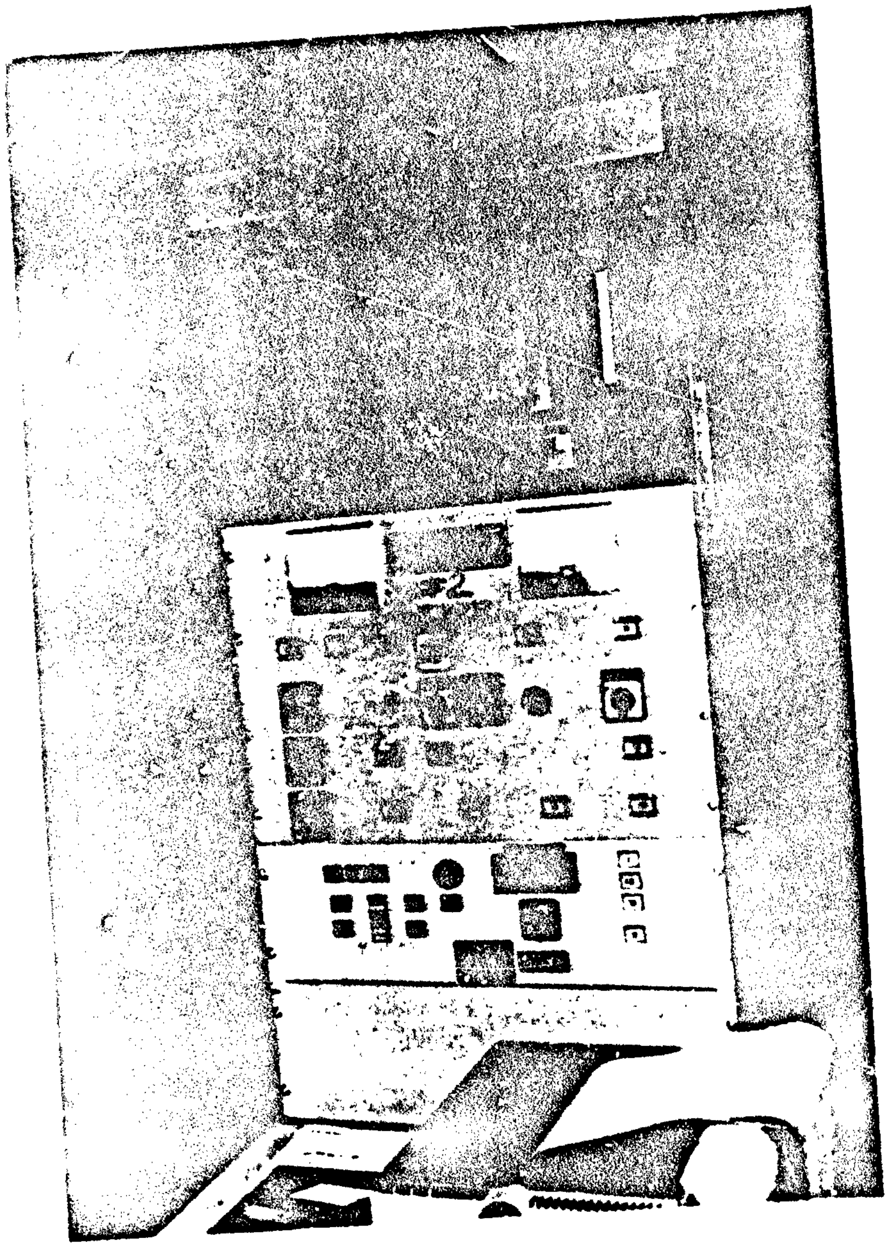


Figure 10 Layout of the Welding Control Panel

The control panels are free standing portable Hammond units that are lined up to form a control console. The console is manned by operators and qualified welders who are in constant communication with those inside the shielding cabinet. Auxiliary equipment, such as relay racks etc, that require no man/machine interface are arranged in a second row, behind the control consoles.

The layout of the welding control panel is shown in Figure 10. The top section accommodates the video monitor, the mid section contains the welding controls and the lower portion contains the controls for Low Torque cutting and cleaning tools. The welding panel display consists of analog meters for welding DC voltage and current, light alarms for loss of coolant, gas flow, filler wire delivery and overcurrent. Indicator lights are also provided for the gas flow and high frequency unit operation.

Hand switches for the operating mode (High Torque/Low Torque), coolant flow (auto/man/off), tool carrier direction (cw/ccw/off) for High Torque mode only, wire feed (rev/for/off), gas flow (auto/man/off) and power (on/off) are provided.

Push buttons for tool jog (High Torque mode only), wire feed jog, weld start, weld downslope and weld stop are also provided.

Interlocks are provided to automatically stop welding if welding speed, welding current, gas flow, or coolant flow deviate from predefined values.

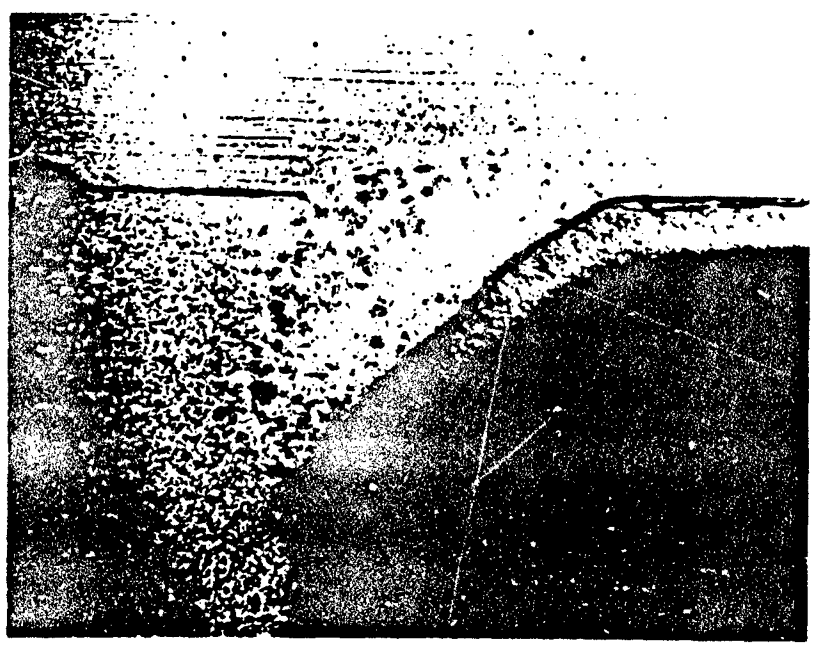


Figure 11 Typical Weld Cross Section

OPERATING EXPERIENCE

The tooling will be used during the scheduled maintenance outages, the first of which will occur in the spring of 1986. However, to confirm that the tooling met all requirements, an Integration and Test Facility was established. All prototype and production tooling was sent to this facility where tools underwent testing and full system operation. The focus of the facility was full-scale mock-ups built from reactor design documents and from as built measurements, to simulate the reactor face. To duplicate the actual heat sink and material conditions, components were fabricated to the original specifications. Numerous tests were performed to achieve acceptable results consistently.

Fine tuning of arc gap settings, torch angle settings, filler wire delivery rate and welding current levels was necessary to overcome undesirable characteristics, namely, unequal fillet leg lengths, undersized cross section, porosity and insufficient heat input. The resulting weld parameter values are shown in Table 2.

Parameter	Setting
Initial Current	40 Amp
Preflow Time	30 sec
Upslope Time	15 sec
Downslope Time	13.5 sec
High Pulse Time	0.1 sec
Low Pulse Time	0.3 sec
Weld Current (High)	198 Amp
Weld Current (Low)	62 Amp
Polarity	Straight DC
Post Heat Time	30 sec
Post Flow Time	30 sec
Weld Time	571.5 sec
Electrode Material	2% Thoriated Tungsten
Cover Gas	Argon
Gas Flow	25 CFH
Filler Wire Feed Rate	13 in per min
Welding Speed	2.8 in per min
Arc Gap Setting	0.135 x 1.135 inch

Table 2 Welding Parameters

Throughout the development phase, sectioned weld sample (See Figure 11) were sent to Ontario Hydro Metallurgical Laboratories to determine weld penetration, profile and grain structure. Only after sectioned welds were found to be acceptable was the full welding system qualified for use.

Using the full system equipment, approximately 200 welds have been performed at a success rate of 95 percent.

Tooling has now been delivered to site where personnel are training with the hardware in preparation for the outage.

Acknowledgement

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ALTERNATIVES TO 1/2 BEAD REPAIR TECHNIQUE (GTAW)

by

P.J. Alberry (CEGB) and J.G. Feldstein (Babcock and Wilcox)

ABSTRACT

A general six layer GTAW repair procedure, which does not require post weld heat treatment, is described as an alternative to the ASME XI half bead repair procedure for thick section Light Water Reactor components. The structure and properties of repair weldments produced using both techniques are described. The alternative six layer GTAW repair procedure is amenable to automation and may offer some advantages in radioactive environments.

1. INTRODUCTION

A repair procedure for thick section Light Water Reactor (LWR) components which does not require post weld heat treatment is described in Section XI Article IWB-4000 of the ASME boiler code for repair of pressure boundary components. This procedure is known as the half bead technique and involves manual grinding to remove half the first layer of weld metal deposited. In addition, a Winter 1984 addendum to Section XI has been issued and subsection IWE-4000 provides details of an additional manual repair technique, for use without post weld heat treatment, which does not involve grinding. This technique is known as the butter bead-temper bead technique for repair of containment materials. However, these two techniques are heavily dependent on manual activities and an alternative mechanised repair capable of automation has been developed by Babcock and Wilcox (1984) Table 1, which may offer some advantages for repair of pressure boundary components in radioactive environments. The manual and mechanised techniques are based on the same principle. The layer thicknesses of successive layers of weld metal are carefully controlled to promote high levels of refinement and tempering in the underlying heat affected zone (HAZ). The half bead technique achieves this control by careful grinding. The butter bead-temper bead technique and the alternative six layer technique achieve the same results by careful control of the welding parameters of successive layers.

TABLE 1 SIX LAYER MECHANISED GTAW PROCEDURE DEVELOPED BY BABCOCK AND WILCOX (1984)

WELDING PARAMETER	LAYER NO		
	1	2	3-6
Current, A	180	200	220
Voltage, V	11	11	11
Travel speed, $mm.s^{-1}$	3.60	2.96	2.54
Wire diameter, mm	0.89	0.89	0.89
Wire feed speed, $mm.s^{-1}$	16.5	25.0	27.5
Weld bead overlap, %	50	50	50
Preheat, °C	150	150	150
Maximum Interpass, °C	260	260	260
Gross heat input, $KJ.mm^{-1}$	550	743	953

Shielding gas: AR 18 CFH with long gas cup. Electrode: 2% Thoria Tungsten; 4 mm diameter 65 mm total stick out, 22.5° included angle at the tip.

The alternative mechanised six layer repair procedure involves the deposition of six layers of weld metal using a mechanised tungsten inert gas process (GTAW). The welding parameters used require specific values to be selected as shown in Table 1. Note that layers 1 and 2 require different welding conditions from each other and from layers 3-6. The six layer procedure is suitable for all-positional welding.

The following sections provide a summary of the structure and properties produced by the ASME XI half bead technique and the specific six layer weld procedure. The data provided demonstrate that repair weldments can be produced, which do not require post weld heat treatment, with material properties adequate for service in nuclear pressure boundary components. However, it is also possible to generalise the six layer weld procedure to allow the selection of a wide range of possible welding parameter combinations which retain all the beneficial features of the specific six layer procedure. This generalisation takes the form of a simplified methodology which provides guidance for the selection of appropriate combinations of welding parameters which may have some practical advantage in any given repair situation.

2. STRUCTURE AND PROPERTIES

Previous studies (Babcock and Wilcox, 1984) have shown that the manual ASME XI and the alternative specific mechanised six layer weld procedure, Table 1, produce repair weldments with material properties adequate for service in nuclear pressure boundary components. The main structural features produced by the manual and mechanised weld procedures are summarised in Table 2. This shows that the main structural features are comparable and that the mechanised six layer procedure is highly tolerant to weld parameter variations (Alberry, 1985a). This implies that a wide range of welding conditions can be specified for each of the six layers while retaining the refinement and tempering action resulting from the deposition of successive layers of weld metal.

Some of the measured mechanical property data taken from work carried out by Babcock and Wilcox (1984) is summarised in Table 3 and Figures 1-5, to illustrate the expected tensile, fracture toughness and fatigue test data from the HAZ and weld metal of repair weldments. Note that the heat treatment caused by the refinement and tempering action of six successive layers is equivalent to the effects of a post weld heat treatment in terms of the HAZ hardness, HAZ toughness and weld metal toughness produced. However, there is little effect on the as-welded residual stress level which is expected to be tensile and of yield point magnitude in the repair weld metal. The tensile stresses are expected to decay rapidly becoming compressive just beyond the HAZ of the repair cavity both on the surface and through the thickness (Hepworth, 1985).

The generalised six layer procedure, described in the following section, is designed to exploit the tolerance of the specific six layer procedure so as to produce equivalent HAZ structures and properties to those measured by Babcock and Wilcox.

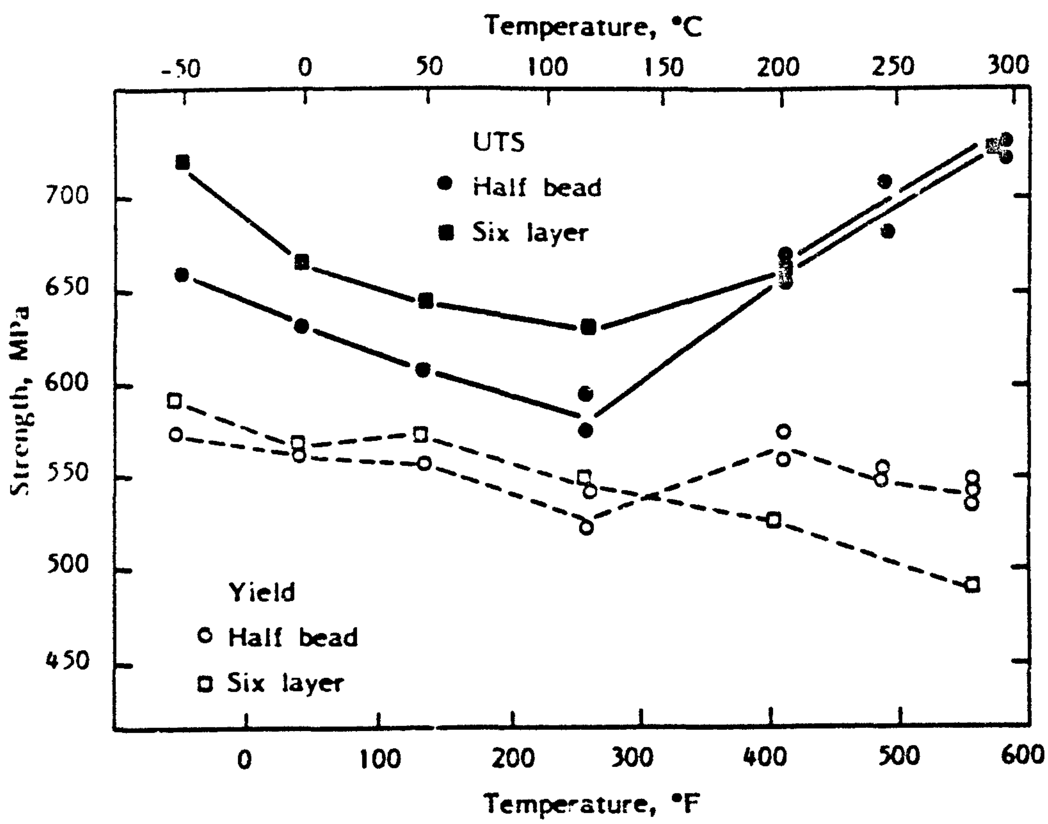

FIGURE 1 - TENSILE TEST DATA FOR WELD METAL SPECIMENS BABCOCK AND WILCOX (1984)

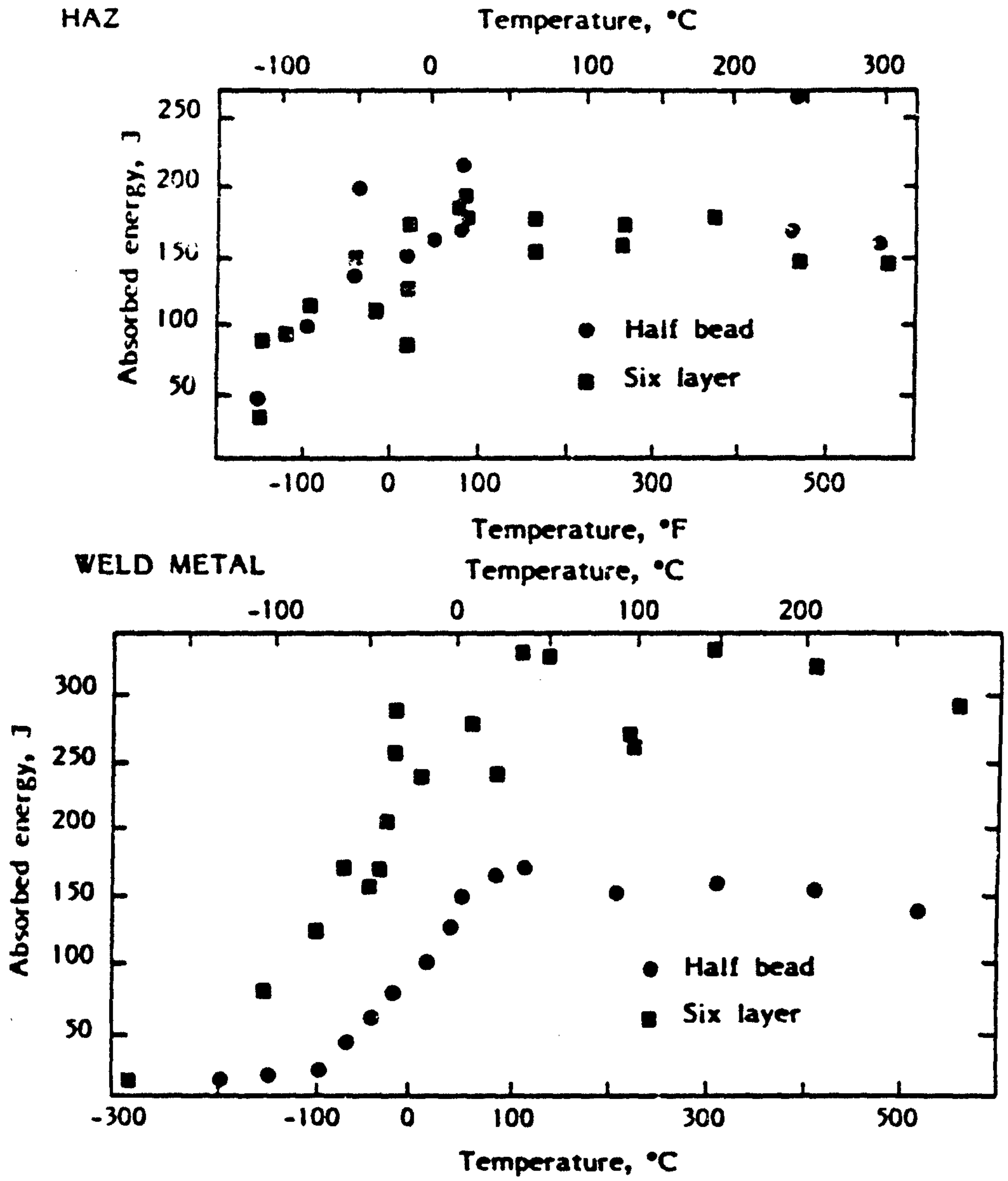

FIGURE 2 - CHARPY V NOTCH DATA FOR HAZ AND WELD METAL SPECIMENS, BABCOCK AND WILCOX (1984)

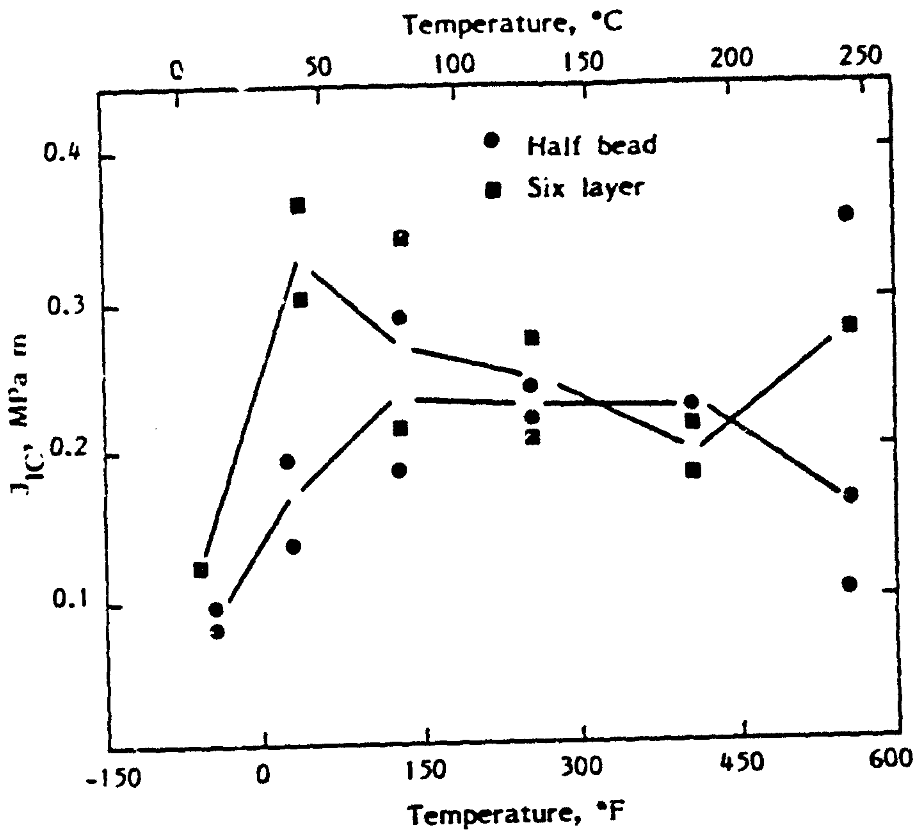

FIGURE 3 - FRACTURE TOUGHNESS DATA FOR HAZ SPECIMENS BABCOCK AND WILCOX (1984)

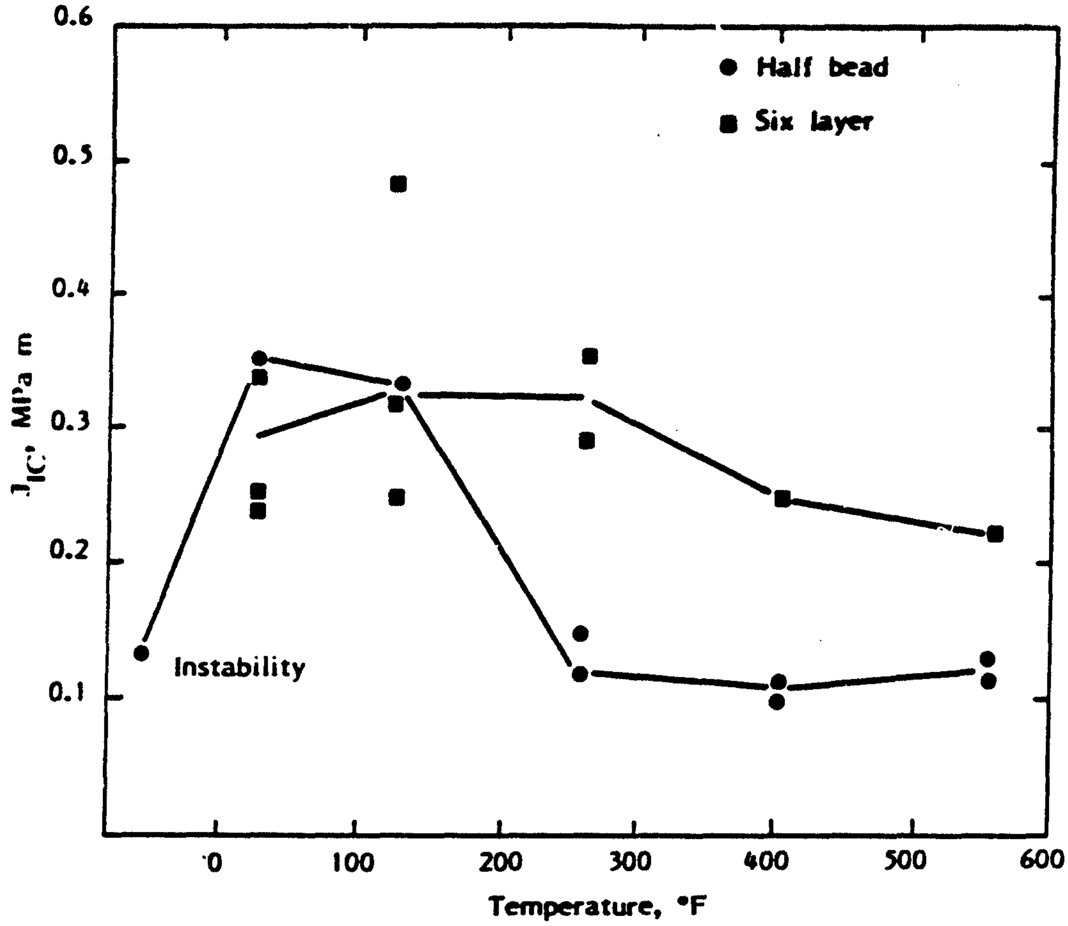

FIGURE 4 - FRACTURE TOUGHNESS DATA FOR WELD METAL SPECIMENS BABCOCK AND WILCOX (1984)

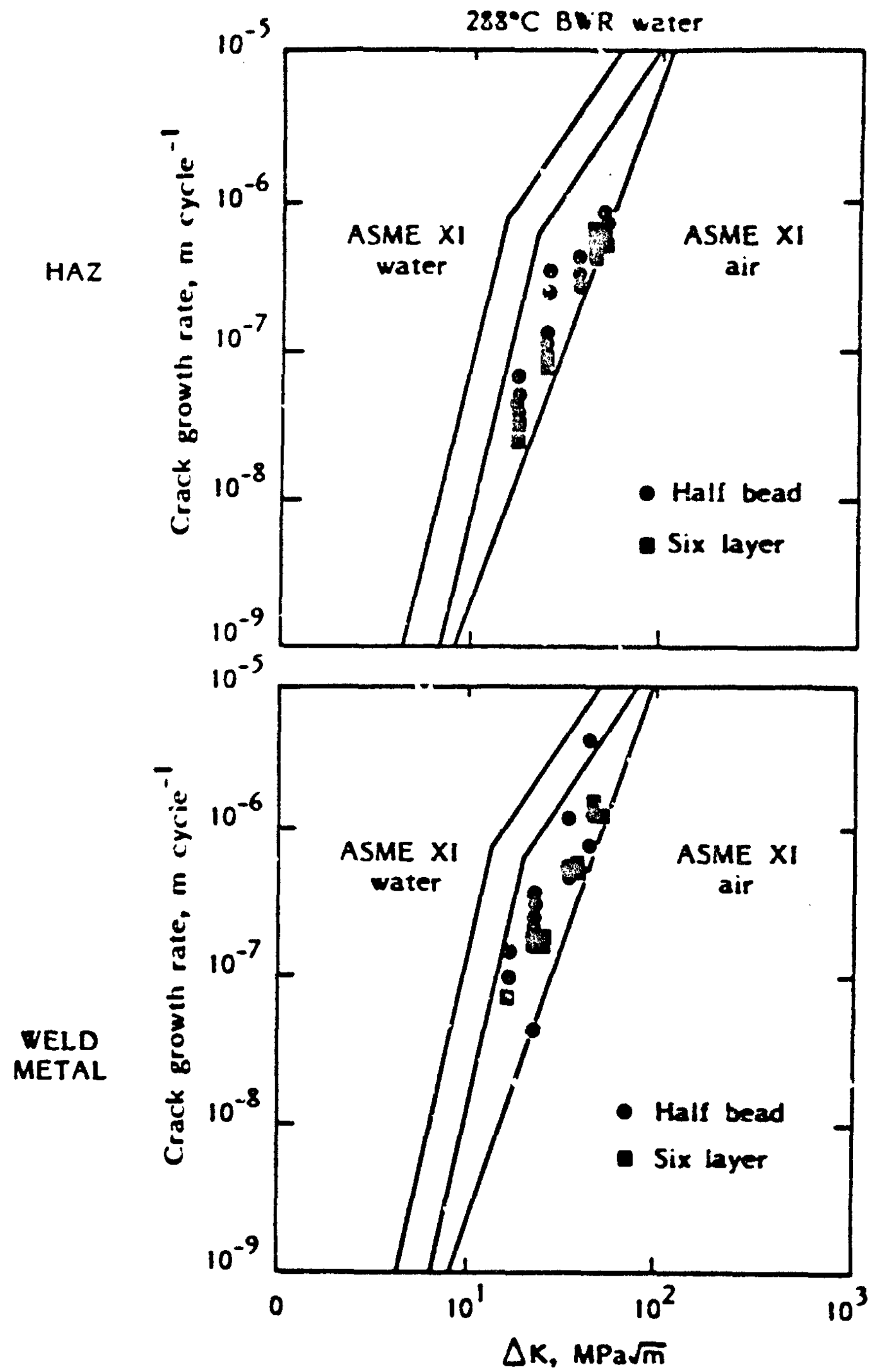

FIGURE 5 - <u>FATIGUE TEST DATA FOR HAZ AND WELD METAL SPECIMENS, BABCOCK AND WILCOX (1984)</u>

TABLE 2 COMPARISON OF HAZ STRUCTURE AND HARDNESS PRODUCED BY ASME XI HALF BEAD TECHNIQUE AND MECHANISED SIX LAYER ALTERNATIVE PROCEDURE FOR SA508 CLASS 2 BASE MATERIAL

	ASME XI HALF BEAD PROCEDURE	ALTERNATIVE 6 LAYER GTAW PROCEDURE
HAZ Structure	Mainly martensite and lower bainite	Same
Grain size	Highly refined, 5% chance of isolated pockets up to 100 μm	Highly refined up to 15 μm
Refinement sensitivity	Grinding tolerance - 0.6 mm + 1.1 mm	Highly tolerant to extreme variations in any welding parameters
Mean hardness *	325	317
Standard deviation	25	13
Hardness sensitivity	Peak hardness sensitive to layer thicknesses Layer 1main controlling variable is ground layer height Layer 2main controlling variables are Weld bead overlap and travel speed	Highly tolerant to variations in welding parameters

* Calculated values corrected for segregation effects

TABLE 3 DROP WEIGHT NDTT VALUES, BABCOCK AND WILCOX (1984)

HAZ	Half Bead	- 51°C	- 60°F
	Six Layer	- 23°C *	- 10°F
Weld Metal	Half Bead	- 51°C	- 60°F
	Six Layer	- 68°C	- 90°F

* Would not fail in HAZ

3. GENERALISED SIX LAYER PROCEDURE

The mechanised six layer GTAW procedure developed by Babcock and Wilcox, Table 1, uses the principle of careful control of individual layer thicknesses of weld metal to promote high levels of refinement and tempering in the underlying HAZ. Each layer progressively shifts bands of refining and tempering isotherms up through the first layer HAZ by an amount equal to the layer thickness. The six layer procedure has three key features:

1. Refinement and tempering in the underlying HAZ is produced by the action of successive layers.

2. The tolerance of the welding procedure to variations in welding conditions increases as the number of layer by layer HAZ interactions increases, i.e. as the layer thicknesses decrease.

3. Individual layer thicknesses are controlled largely by wire feed speed for most practical situations.

The Babcock and Wilcox six layer procedure utilises each of these three key features. However, it is clear that they do not uniquely define a specific welding procedure. They can be expressed quite generally for any arbitrary number of layers (N) and for a wide range of welding conditions.

The first key feature requires that refinement and tempering in the underlying HAZ be produced by the action of successive layers. Simplistically, this requires that half the deposited layers should cause refinement and the other half should result in tempering. This implies that after the addition of half the layers, the refining isotherms are required to be clear of the underlying HAZ so that the tempering can occur as the remaining layers are deposited. For constant welding conditions for all N layers, this simple requirement can be expressed as an equivalence:

$$\frac{N}{2} \times \text{single layer thickness} \equiv \text{HAZ DEPTH}$$

This equivalence ensures that half of the layers penetrate the HAZ and cause refinement. It also ensures that when half the layers have been deposited the total layer thickness produced is equal to the original HAZ depth. Consequently, the next layer of isotherms do not penetrate the HAZ causing refinement, instead they cause a substantial tempering effect. It follows that the equivalence can be maintained using different layer thickness for the first half of the deposited layers, as in the specific Babcock and Wilcox procedure. However, the procedure has been generalised and simplified at the same time using the equivalence expressed above and keeping the welding conditions constant for each layer.

Since the layer thickness is defined by the amount of deposited weld metal per unit length together with the weld bead overlap the equivalence for constant welding conditions can be expressed as

$$\frac{N}{2} \times \frac{(\text{WFS})\pi D^2}{4v} \times \frac{100}{(100 - \Delta)} \times \frac{1}{\text{BEAD WIDTH}} \equiv \text{HAZ DEPTH}$$

where WFS is the wire feed speed
D is the wire diameter
ν is the travel speed
and Δ is the weld bead overlap

Since the number of layers required for the mechanised procedure is six and the weld bead overlap is specified as 50% of the weld bead width, Babcock and Wilcox (1984) it follows that:

$$WFS = \frac{2}{3\pi D^2} \times \begin{matrix}\text{HAZ}\\\text{DEPTH}\end{matrix} \times \begin{matrix}\text{BEAD}\\\text{WIDTH}\end{matrix} \times \nu \qquad (1)$$

Since the HAZ depth and weld bead width are defined by the welding conditions, the derived equation defines the appropriate wire feed speed required to maintain the refinement and tempering action of successive layers of weld beads. This will produce equivalent HAZ structure and properties to those obtained using the specific six layer procedure.

The advantage of the simplified, generalised procedure is that any set of welding conditions can be selected for a given practical situation, providing that the same conditions are used for each of the six layers and that the wire feed speed is determined using equation (1).

Hence low heat input welding conditions which require thin layers can be used for shallow cavities and higher heat input welding conditions which require thicker layers can be used for deeper cavities. In this way the welding conditions and layer thicknesses will always be matched, thereby retaining all the beneficial features of the specific six layer procedure in the most cost effective way.

4. COMPUTER MODEL VALIDATION OF THE GENERALISED SIX LAYER WELD PROCEDURE

A computer model for the prediction of weld HAZ structures produced by the GTAW process has been described by Alberry, Brunnstrom and Jones (1983). The basic HAZ metallurgy required as input to this model has been described by Alberry and Lambert (1982) for SA508 class 2 material. The basic weld bead shape data required by the model was obtained from weld bead data carried out on the same cast of SA508 class 2 by Babcock and Wilcox (Alberry, 1985b). The measured weld bead variability relevant to this cast of material was also input and used by the model to simulate bead to bead variations (Alberry, 1985b). The computer model predictions of HAZ structure and hardness have been validated for the ASME XI half bead procedure and a range of six layer weld procedures carried out by Babcock and Wilcox (Alberry, 1985b). Examples are given in Figures 6 and 7.

The computer model hardness predictions and the hardness measurements can also be compared statistically. The five hardest readings taken from each HAZ traverse have been used to calculate a mean and standard deviation for the Babcock and Wilcox six layer procedure measurements and the corresponding computer model calculations. The mean and standard deviations are given in Table 4 together with some tolerance limit statistics and extreme value statistics derived from these hardness data. The tolerance limit statistics, which assume a gaussian distribution of hardness values show good agreement with the extreme value statistics which are not limited to gaussian distributions. For example, the tolerance limit statistics for the Babcock and Wilcox measurements show that there is 99% confidence that 95% of the HAZ hardnesses are less than 35£ Hv. This compares favourably with the equivalent extreme value

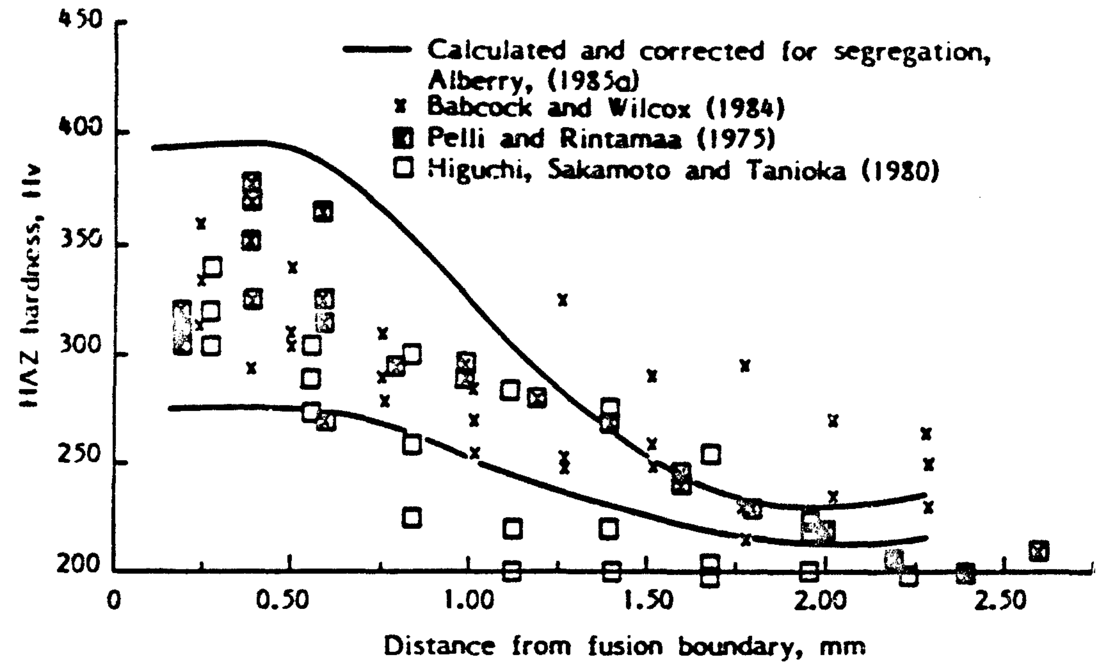

FIGURE 6 - MEASURED AND CALCULATED HAZ HARDNESS VALUES FOR THE ASME XI HALF BEAD TECHNIQUE

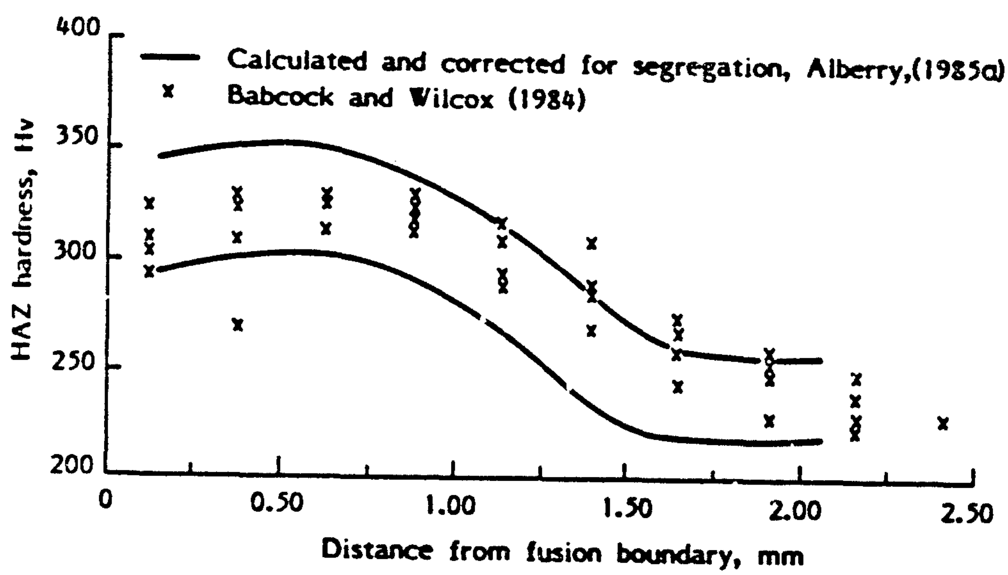

FIGURE 7 - MEASURED AND CALCULATED HARDNESS VALUES FOR THE SPECIFIC SIX LAYER MECHANISED WELD PROCEDURE (BABCOCK AND WILCOX, 1984)

statistics derived from the measured and calculated data, although some small differences exist due to the difference in sample sizes, i.e. 15 HAZ hardness measurements compared with 150 HAZ hardness calculations.

TABLE 4 COMPARISON OF HAZ HARDNESS MEASUREMENTS AND CALCULATIONS FOR THE BABCOCK AND WILCOX SIX LAYER PROCEDURE

			Tolerance Limit Statistics		Extreme Value Statistics	
	Mean	σ	99% Sure 95% Less than	99% Sure 99.9% Less than	% of Data Less than 350	Pr(Max < 350)
Measured	313	16	358	393	98	0.96
Computer M del	327	13	345	370	99.5	0.95

NOTE: Mean is calculated from the five highest hardness readings per traverse with 10 HAZ hardness indentations per traverse.

As a result of the agreement between the measured and calculated hardness values, the computer model has been used to examine six generalised test case six layer weld procedures. The six test case six layer welding procedures selected are shown in Table 5 and include possible extreme case variants involving low and high heat inputs.

TABLE 5 EXTREME CASE WELD PROCEDURES USED FOR COMPUTER MODEL VALIDATION OF THE GENERALISED SIX LAYER WELD PROCEDURE

Test Case Weld	Current amps	Voltage volts	Travel Speed $mm.s^{-1}$	Gross Heat Input $J.mm^{-1}$	Required Wire Feed Speed $mm.s^{-1}$
1	180	11	3.6	550	15.7
2	200	11	2.96	743	17.4
3	220	11	2.54	953	19.8
4	200	11	2	1100	18.5
5	250	11	2	1375	24.8
6	300	11	2	1650	32.0

NOTE: The same welding conditions are used for all six layers of each test case weld.

In addition, the weld bead variability input to the model was increased to allow for cast to cast variations, as shown in Figure 8.

Computer model calculations of HAZ hardnesses were then carried out for each of the test case six layer welding procedures. Mean HAZ hardness values and their standard deviations calculated from the five

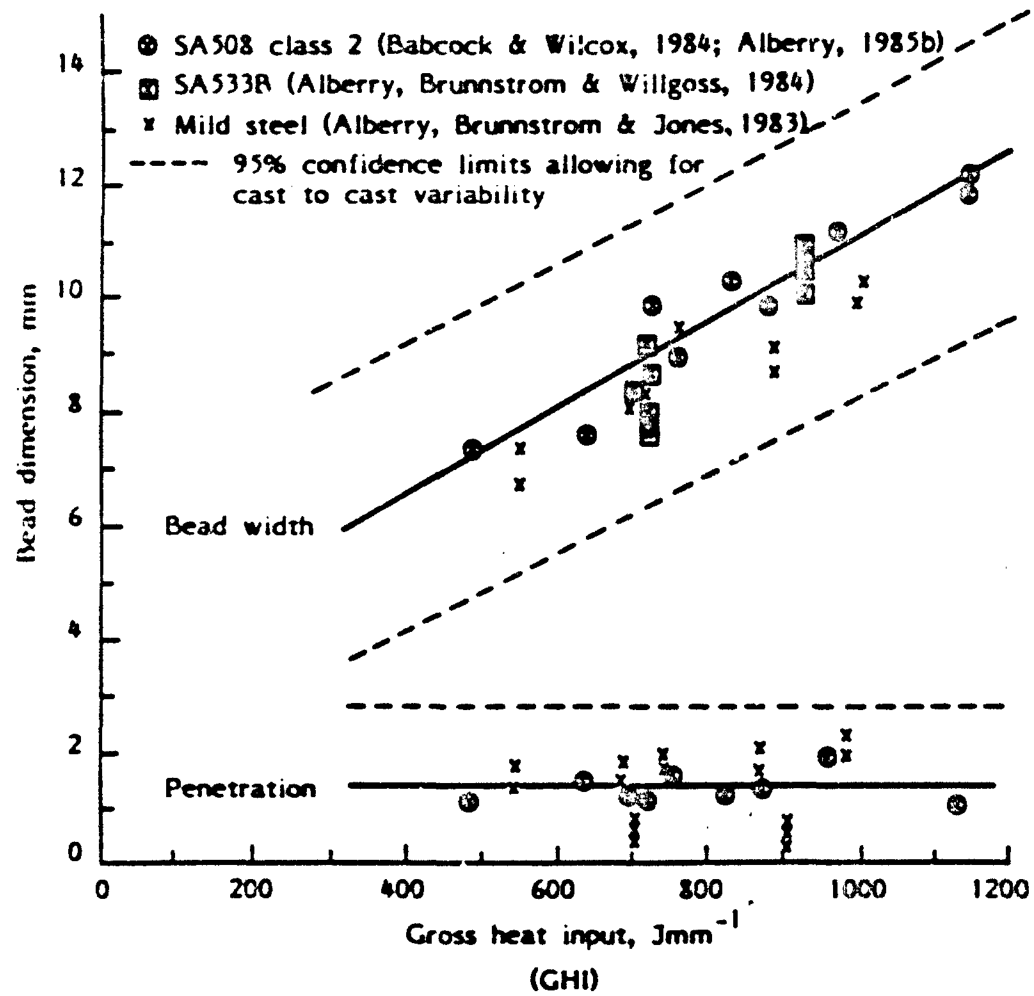

FIGURE 8 - WELD BEAD DIMENSION VARIABILITY BETWEEN DIFFERENT CASTS OF MATERIAL

highest HAZ hardness readings per traverse are shown in Table 6 together with the tolerance limit and extreme value statistics derived from the data.

The data confirm that each of the test case weld procedures derived using the generalised six layer methodology is equivalent to the Babcock and Wilcox specific six layer procedure in terms of mean HAZ hardness and likely variability.

TABLE 6 CALCULATED HAZ HARDNESS VALUES FOR SIX TEST CASE SIX LAYER WELDS TAKING ACCOUNT OF THE LIKELY WELD BEAD VARIABILITY DUE TO CAST TO CAST VARIATIONS

Test Case Weld	Mean Hv	σ	Tolerance Limit Statistics: 99% Sure 95% Less Than	Tolerance Limit Statistics: 99% Sure 99.9% Less Than	Extreme Value Statistics: % of Data Less Than	Extreme Value Statistics: Pr(Max < 350)
1	324	14	356	381	97	0.65
2	317	12	344	366	99.5	0.89
3	313	12	340	362	99.9	0.90
4	310	13	340	363	99.8	0.94
5	306	12	333	355	99.9	0.97
6	303	11	328	348	> 99.9	> 0.995
B&W Meas	313	16	358	393	99.5	0.96

Since the successful operation of the generalised six layer methodology is based on the maintenance of a given wire feed speed in relation to an appropriate set of welding conditions, it is important to obtain some estimate of the precision required in practice. This has also been obtained using the computer model. The six test case welds have been examined for a range of wire feed speed values deviating up to ± 60% from the required target value defined by the generalised methodology. For this purpose the weld bead variability was set to zero to reduce the noise in the model and reduce the computational effort required. However, a large number of confirmatory calculations which included the appropriate weld bead variability have been carried out for test case weld 1, which is the most sensitive of the test case welds to the effects of weld bead variability.

The results shows that there is a gradual deterioration in mean hardness level and hardness variability beginning at about ± 45% in wire feed speed. Therefore, it is recommended that wire feed speed values be controlled to ± 30% of the target value required by the six layer methodology. This should ensure that the weld quality is maintained whilst allowing a considerable flexibility of choice to cope with any practical difficulties which might be encountered.

5. QUALITY CONTROL REQUIREMENTS FOR THE GENERALISED SIX LAYER PROCEDURE

The complexity of HAZ structures and hardness variations which occur in multipass welds imply that quality control requirements must be based on an adequate statistical sample from the weld HAZ. The known HAZ hardness variability between traverses makes it unlikely that a single HAZ hardness traverse will be adequate. Consequently, the computer model was used to determine the minimum number of hardness traverses required to sample a single weld. The welding conditions for test case 1 were studied using the computer model containing the appropriate cast to cast weld bead variability. A total of 42 separate hardness traverses were computed for a single weld spaced at 1 mm intervals. The highest five readings from each traverse were used to compute a mean and standard deviation. Random groups of HAZ hardness traversas extracted from this population showed that a minimum of three separate traverses with ten hardness readings per traverse were required to ensure that the values obtained were not significantly different from the true mean values estimated from all the readings. The mean and standard deviation derived from the highest five readings from each of the three separate HAZ hardness traverses can also be used as a go/no go quality control criterion.

6. QUALITY CONTROL CRITERION FOR THE GENERALISED SIX LAYER PROCEDURE

A simple quality control criterion is required to decide whether or not a given weld procedure produces the intended level of refinement and tempering of the first layer HAZ. The mean and standard deviation of the five highest readings from each of three separate hardness traverses must satisfy the condition.

Mean Value + Standard Deviation < 350 Hv

This criterion has been derived from the Babcock and Wilcox hardness measurements taking into account the full range of welding procedure variants and the possible detrimental effects of weld bead variability (Alberry, 1985c). Since the SA508 class 2 base plate composition used by Babcock and Wilcox is one standard deviation to the alloy rich side of those compositions in service, the quality control criterion formulated above is particularly stringent so as to provide a reasonable upper bound for more hardenable compositions. It is assumed that HAZ hardness and toughness are inversely correlated. This means that less hardenable compositions, which have lower mean HAZ hardnesses which readily comply with the quality control criterion, have correspondingly higher HAZ toughness values than more hardenable compositions.

Interestingly, the quality control criterion can also be used to assess the ASME XI half bead procedure. The mean and standard deviation of the five highest readings from each of three separate hardness traverses taken from Babcock and Wilcox (1984) are 298 Hv and 33 Hv, respectively. The sum of the mean value and standard deviation is 331 Hv which complies with the quality control criterion of 350 Hv and indicates that the procedure was carried out satisfactorily.

7. PROCEDURE QUALIFICATIONS FOR THE GENERALISED SIX LAYER PROCEDURE

The generalised six layer weld procedure described is equivalent to the specific six layer procedure which is highly tolerant to variations in each of the welding parameters (Alberry, 1985a). This tolerance has been designed into the generalised six layer weld procedure so that the layer thicknesses are specified to progressively refine and temper the underlying first layer HAZ. The tolerance is implied by the use of six thin layers which, in turn ensure that a large number of HAZ refinement and tempering interactions occur. Simplistically, the tolerance can be visualised from the following argument. The first layer HAZ is refined by each of the following three layers, i.e. layers 2-4. Layer 3 is designed to provide optimum refinement. However, if the layer thicknesses are too small then layer 4 will compensate and if the layer thicknesses are too large then layer 2 will compensate. Similarly, layers 4-6 are capable of providing sufficient tempering. Layer 5 is designed to provide optimum tempering and deviations either side of the mean values are compensated for by either layer 4 or layer 6. Hence layer 4 has a dual role, it either provides refinement or it provides tempering depending on the layer thicknesses produced by a given set of welding conditions.

The general equation described in Section 3 ensures that the balance of refinement and tempering is maintained for any given welding conditions by specifying the appropriate wire feed speed required to maintain the individual layer thicknesses.

However, it can be shown that the generalised six layer procedure is tolerant to wide variations in the weld bead dimensions (Alberry, 1985a). The tolerance is such that the weld bead variability can exceed the likely weld bead variability typical of cast to cast variations before the weld quality begins to deteriorate. The high level of tolerance to weld parameter and weld bead variability implies that the traditional weld procedure qualification test may be inappropriate for carefully designed mechanised weld repair procedures. It might be more meaningful to specify appropriate calibration procedures and levels of machine reproducibility. This could be incorporated into an on-line quality control monitoring system for each of the important weld parameters. This might prove to be more meaningful in statistical terms as an assurance of weld quality for individual repairs than repeated, traditional, weld procedure qualifications.

8. PRACTICAL IMPLEMENTATION

The specific six layer procedure developed by Babcock and Wilcox has sufficient tolerance to allow it to be simplified and generalised to define a wide range of potential weld parameter combinations which preserve the refinement and tempering action of successive layers.

The selection of an appropriate combination of weld parameters for any generalised six layer weld repair is as follows:

1. Select a combination of weld parameters suitable for the deposition of six layers of weld metal into the repair geometry. This will include a first guess at an appropriate wire feed speed.

2. Using the selected parameters deposit a single weld bead on to a representative base material.

3. Determine the weld bead width and HAZ depth at the weld bead centre from two suitable cross sections.

4. Set the wire feed speed as follows:

$$\text{WIRE FEED SPEED} = \frac{2}{3\pi D^2} \times \text{HAZ DEPTH} \times \text{BEAD WIDTH} \times \text{TRAVEL SPEED}$$

where D is the wire diameter.

5. Repeat stages 2-4 until the wire feed speed determined by step 4 is within 30% of the wire feed speed value used for step 2 to deposit the single weld bead.

6. Carry out a representative trial weld using the welding conditions determined by steps 2-5.

7. Take three separate hardness traverses from the representative trial weld. Each hardness traverse should consist of not less than ten indentations placed in the HAZ of the weld.

8. The five highest hardness readings from each of the three traverses should be averaged to produce a mean value and associated standard deviation.

9. The sum of the mean value and the standard deviation should not exceed 350 Hv.

9. FUTURE WORK

The overall position of repair welds in heavy section LWR components which do not require post-weld heat treatment may be summarised as follows:

- a generalised, mechanised, GTAW weld repair alternative has been developed which is at least as defensible as the ASME XI half bead technique in terms of grain refinement, tempering and mechanical properties.

- a weld metal chemistry has been identified which is not susceptible to strain ageing.

- the automated GTAW process is not subject to significant variability in terms of refinement and tempering.

- both the ASME XI half bead technique and the mechanised GTAW alternative give rise to substantial residual stresses.

The remaining issue is to demonstrate that the presence of these residual stresses is not detrimental to service performance of these as-welded repairs. A further programme of residual stress evaluation and management investigating the effects of re-cladding,hydro test, service cycles and low temperature post weld heat treatment would be valuable. In this case, a further consideration of possible environmental issues for repairs with managed residual stresses would be advisable, but, more importantly, a fracture mechanics methodology should be validated which is capable of evaluating the integrity of as-welded repairs in materials which may possess some degree of embrittlement due to their previous service history.

10. CONCLUSIONS

A generalised six layer GTAW repair methodology has been developed as an alternative to the ASME XI half bead procedure. A wide range of welding procedures can be specified which do not require post weld heat treatment and produce repair weldments with material properties adequate for service in nuclear pressure boundary components.

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ALTERNATIVES TO HALF-BEAD WELD REPAIRS USING PULSED GAS METAL ARC WELDING

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ABSTRACT

Although existing manual shielded metal arc welding (SMAW) procedures are included in the ASME boiler and pressure vessel code for use in non-post-weld heat treated repair welds, situations exist in which manual welding is undesirable due to adverse environments, such as radiation or high temperatures in the area of the repair. Under such conditions, a remote welding system would be required which incorporates a welding process capable of producing all-position welds of high integrity with controlled deposition sequence so that post-weld heat treatment can be avoided. This paper describes the development of a two-layer grain refinement welding technique using the pulsed gas metal arc welding process as an alternative to the manual SMAW half-bead method. Two layer welding trials on SA516-70 and SA533-B1 illustrate that with the correct combination of preheat, heat input and deposition sequence, a high degree of heat-affected zone (HAZ) grain refinement and significant HAZ hardness reduction can be achieved.

1.0 INTRODUCTION

The aging of nuclear plants has become a topic of great interest as nuclear generating stations reach the latter part of their design lives. In some cases weld repairs have been required to rectify defects which exceed the allowable code guidelines. When these situations arise it is desirable to have weld repair procedures available which can be used to repair the vessels without excessive downtime or danger of damage to the vessel.

One of the more time-consuming and difficult tasks associated with performing a vessel weld repair is in-situ post-weld heat treatment (PWHT). Welding procedures, such as the half-bead repair method, have been developed which eliminate the need for PWHT by the use of controlled welding and grinding sequences/1/. For carbon and low alloy steels the ASME approved methods for nuclear welding repairs allow the use of the manual shielded metal arc welding (SMAW) process only. This is seriously limiting in situations where a mechanized or automated welding system would be desirable to perform a repair, for example, in an environment unacceptable for lengthy welder exposure because of radiation or excessively high temperatures. For these reasons a repair procedure using a welding process which is adaptable to automated systems was required. The repair procedure would be used in the same way as the manual SMAW procedure, to avoid the unnecessary use of PWHT following a repair.

Recent advances in power source development have produced a new generation of gas metal arc welding equipment which use power transistors to pulse the current between background and peak levels in order to closely control metal transfer. These pulsed gas metal arc welding (GMAW-P) power sources are capable of welding in spray transfer mode in all positions and thus show great potential as an alternative approach to manual SMAW for a non-PWHT repair welding procedure.

This paper describes the development of a two-layer weld repair technique for nuclear and conventional pressure vessels which uses pulsed GMAW as an alternative to manual SMAW. The procedure requires close control of preheat, heat input and deposition sequence to produce acceptable grain-refined heat-affected zone regions. The repair procedure is demonstrated on two grades of pressure vessel steel, A516 Grade 70 and A533 Type B class 1 to produce refined HAZ regions with reduced hardness levels compared to conventional repair methods.

2.0 BACKGROUND

The half-bead welding repair method was introduced into Sections III and XI of the ASME Boiler and Pressure Vessel Code to permit welding repairs to be performed on ferritic components under circumstances where PWHT that is normally required is impractical or impossible. The half-bead procedure uses a controlled heat input and grinding sequence to prevent the formation of low toughness transformation products in the HAZ. This is achieved by using small diameter (2.4 mm, 3/32 inch) SMAW electrodes to deposit the first layer of weld metal in a controlled manner, followed by a grinding procedure which is required to remove half of the layer of deposited weld metal. A second layer of weld metal is then deposited with slightly larger (3.2 mm, 1/8 inch) electrodes in order to completely cover the first layer. The remainder of the cavity is filled using standard welding techniques and up to 4 mm (5/32 inch) diameter electrodes. The half-bead method has been used successfully in many situations; however the consistency and reproduceability of the grinding step is somewhat questionable/2/.

A variation on the half-bead technique has been developed which uses similar methods of control to produce adequate HAZ properties but eliminates the grinding step from the repair procedure. This repair method, labelled temper-bead welding, uses 2.4 mm diameter SMAW electrodes for the first layer of weld deposit employing a stringer bead welding technique with no weaving. The second layer of weld metal is deposited using 3.2 mm diameter electrodes directly onto the first layer and again no weaving is allowed. A schematic cross section of the temper-bead technique is shown in Figure 1. The use of only stringer beads and close limits on the welding current provide a method of controlling heat input during manual welding. The manual temper-bead welding technique has been used in a number of applications at Ontario Hydro for weld repairs to vessels in nuclear and fossil plants as well as for vessels in wet hydrogen sulphide service.

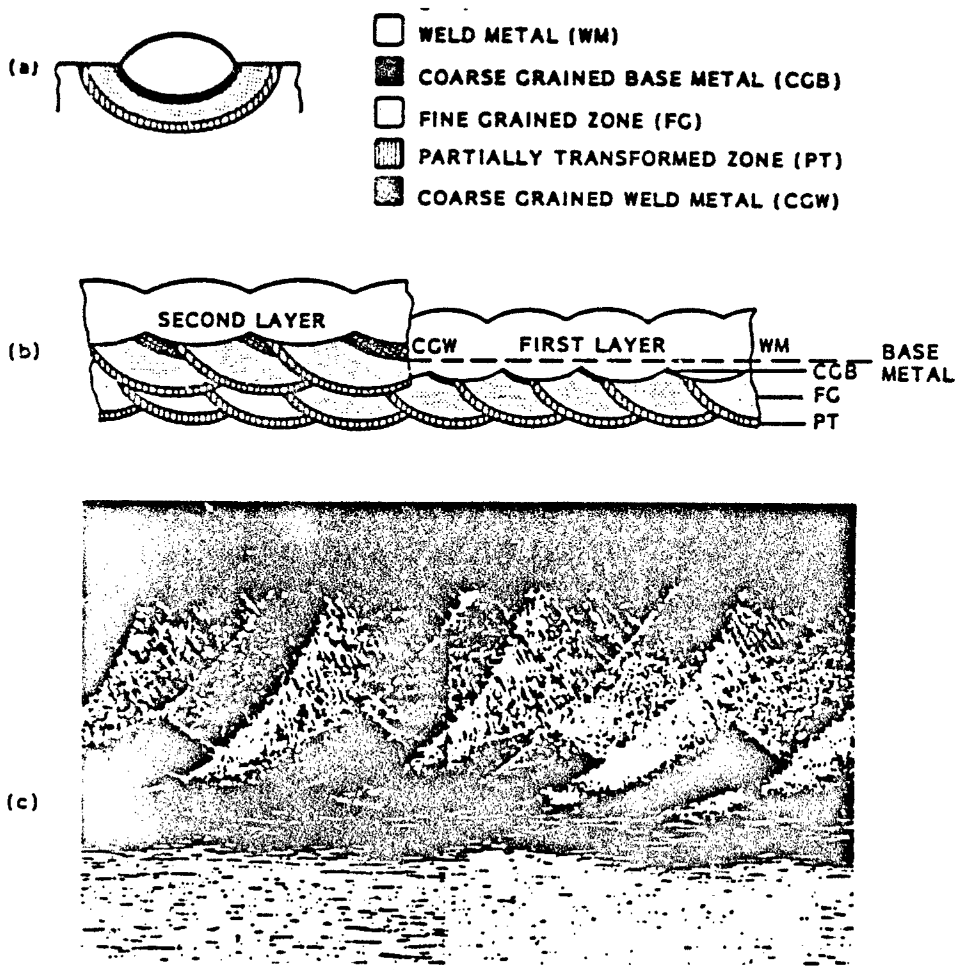

FIGURE 1
TWO-LAYER GRAIN REFINEMENT WELDING TECHNIQUE
(A) SINGLE WELD BEAD
(B) CORRECTLY APPLIED WELD BEADS ILLUSTRATING REFINEMENT OF BASE PLATE COARSE GRAINED ZONE
(C) CROSS SECTION OF PULSED GMAW TWO-LAYER TECHNIQUE

Although the manual temper-bead procedure has proven itself to be successful, there are a number of applications which would be more suited to mechanized or automated weld repairs. Repairs in radioactive environments, for example, could not be done manually in many cases due to the high radiation fields. The labour intensive nature of the temper-bead procedure, requiring many man-hours devoted to welding and cavity preparation, compounds the problem.

In order to develop a mechanized repair welding procedure a suitable welding process must be identified. The process should have an integral wire feed capability to provide filler material, it must be capable of operating remotely and continuously and it must have the capability of welding in all positions. Recent developments in power transistor control technology applied to welding equipment have produced a new generation of pulsed GMAW power sources which possess the necessary characteristics for remote automated operation. The increased control of metal transfer afforded by pulsing the current have made this form of gas metal arc welding an attractive process for repair welding applications.

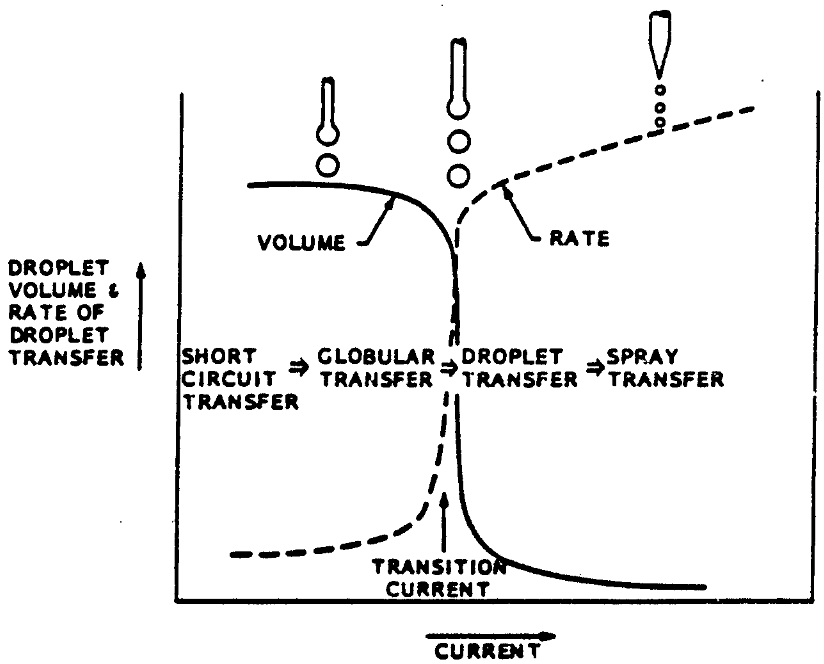

FIGURE 2

THE INFLUENCE OF INCREASED CURRENT ON METAL TRANSFER IN GAS METAL ARC WELDING

3.0 PULSED GAS METAL ARC WELDING

The GMAW process has in the past had several hindering characteristics which have prevented its widespread use in applications where high integrity deposits are required in all welding positions. These problems are related to the inherent metal transfer characteristics of mild steel wires as illustrated in Figure 2. For a given wire size, the metal transfers in a short-circuiting or globular mode at low current levels and, as current increases, there is a marked transition to a streaming spray mode of transfer at high currents. In the low current short-circuiting and globular transfer modes all-position welding is possible; however the heat input is insufficient to fully melt the underlying baseplate and lack-of-fusion defects result. At higher currents where metal transfer occurs in the spray mode, the heat input is excessive causing a relatively large molten pool to form making positional welding no longer possible.

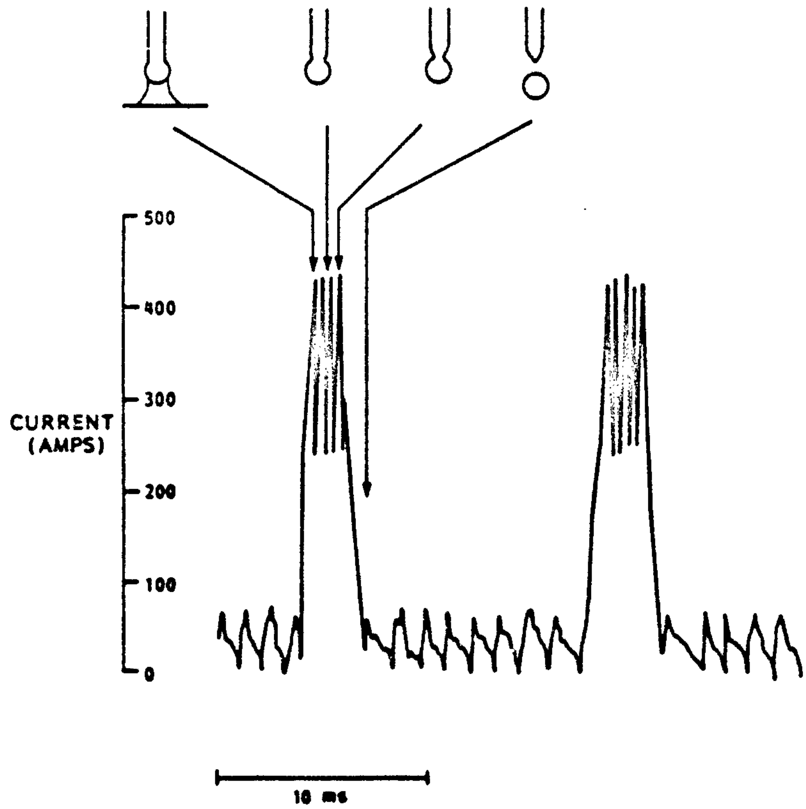

FIGURE 3

CURRENT WAVEFORM OUTPUT FOR PULSED GMAW AND ASSOCIATED METAL TRANSFER. VALUES SHOWN ARE TYPICAL FOR ARGON/CARBON DIOXIDE SHIELDING GAS AND 0.9mm (0.035 IN) DIAMETER CARBON STEEL WIRE.

Pulsed GMAW is able to overcome these difficulties by controlling the metal transfer during welding. This is accomplished through accurate control of peak and background current levels so that, ideally, one drop of molten metal is transferred for each current pulse. The parameters which must be controlled during pulsed GMAW include peak current, background current, peak duration and frequency. In order to maintain a constant spray transfer at low average currents, the peak current must be above the transition current sufficient time for a droplet to form and begin to detach. The current then drops to the background level which is sufficient to sustain the arc. Figure 3 illustrates this principle schematically with the output from the power source used in this experimental program. The power source is an Osaka Transformer Corporation (OTC) Transistarc 350 which uses frequency modulation to control metal transfer to increase the average current for greater burnoff rates. As the wire feed speed increases, the pulse peak value, the background value and the pulse current duration stay the same, only the frequency changes. A number of other control methods are available for pulsing the current to achieve stable metal transfer. These other techniques modulate peak or background current, peak current width, or some other combination of the pulse parameters.

4.0 EXPERIMENTAL WELD TRIALS

The experiments to develop the pulsed GMAW two-layer refinement technique were conducted using the OTC power source and a mechanized welding carriage. Weld trials were done in the horizontal position on flat surface-ground plates. Development work was performed on SA516-70 carbon steel, the material used in most pressure vessels in Ontario Hydro's nuclear generation systems. In addition, test welds were made on SA533-B1 low alloy pressure vessel steel. The composition of the materials used for the welding tests are listed in Table 1.

TABLE 1

COMPOSITION OF STEELS USED IN WELDING EXPERIMENTS

STEEL DESIGNATION	COMPOSITION %						
	C	Mn	P	S	Si	Ni	Mo
SA 516 GRADE 70	0.23	1.16	0.016	0.015	0.24	-	-
SA 533 CLASS B1	0.19	1.3	0.013	0.015	0.27	0.64	0.55

A preliminary test program was conducted to select a solid wire consumable based on handling characteristics and deposit properties. The results of the program indicated that, of the three mild steel wires tested, ER70S-3 should be the candidate for use in repairs on carbon steel vessels. Table 2 gives the compositions of the wires tested and Figure 4 shows the impact toughness curves for each of the consumables. A subjective welder evaluation of the consumables concluded that ER70S-3 possessed superior handling properties compared to the other electrodes evaluated. The evaluation also showed that slag formation for ER70S-3 was similar to that of ER70S-2 but less than that with ER70S-6.

TABLE 2

COMPOSITION LIMITS FOR CARBON
STEEL FILLER WIRES AND AS DEPOSITED
WELD METAL CHEMISTRY

WIRE		COMPOSITION %							
		C	Mn	P	S	Si	Al	Ti	Zr
ER 70S-2	SPECIFICATION[1]	0.07	0.9-1.40	0.025	0.035	0.40-0.70	0.05-0.15	0.05-0.15	0.02-0.12
	AS DEPOSITED	0.073	0.96	0.007	0.011	0.39	0.007	0.028	0.002
ER 70S-3	SPECIFICATION	0.06-0.15	0.90-1.40	0.025	0.035	0.45-0.70	-	-	-
	AS DEPOSITED	0.088	0.73	0.009	0.013	0.31	-	-	-
ER 70S-6	SPECIFICATION	0.07-0.15	1.40-1.85	0.025	0.035	0.80-1.15	-	-	-
	AS DEPOSITED	0.092	1.05	0.018	0.013	0.61	-	-	-

NOTE:

1 AWS SPECIFICATIONS FOR CARBON STEEL FILLER METALS FOR GAS SHIELDED ARC WELDING A5.18. COMPOSITIONS ARE MAXIMUMS OR RANGES.

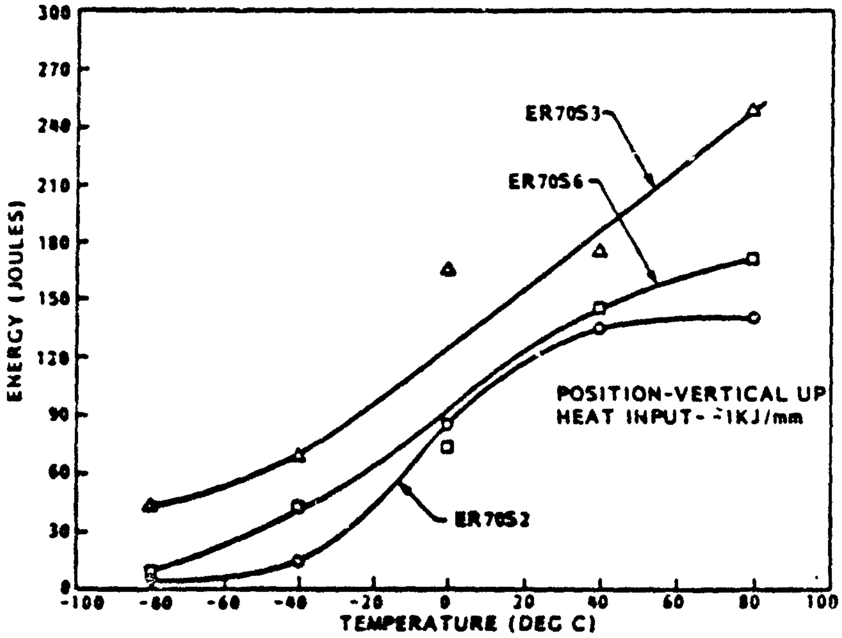

FIGURE 4

CHARPY IMPACT TRANSITION CURVES FOR VARIOUS CARBON STEEL
WELD METALS DEPOSITED WITH PULSED GMAW
(1J=0.74 FT-LB, °F = 1.8 x °C + 32)

The early development work using SMAW electrodes for temper-bead welding used the concept of constant runout length for electrodes at fixed current levels to control heat input. That is, if stringer beads only are used, a relatively consistent heat input will be produced for a fixed electrode size and current level. For the manual temper-bead procedure the heat inputs for the first and second layers respectively are approximately 580 J/mm and 1000 J/mm. This yields a heat input ratio,(HI_2/HI_1), the heat input for layer two (HI_2) over the heat input for layer one (HI_1), if the same length of weld metal is deposited for each electrode,of 1.7.

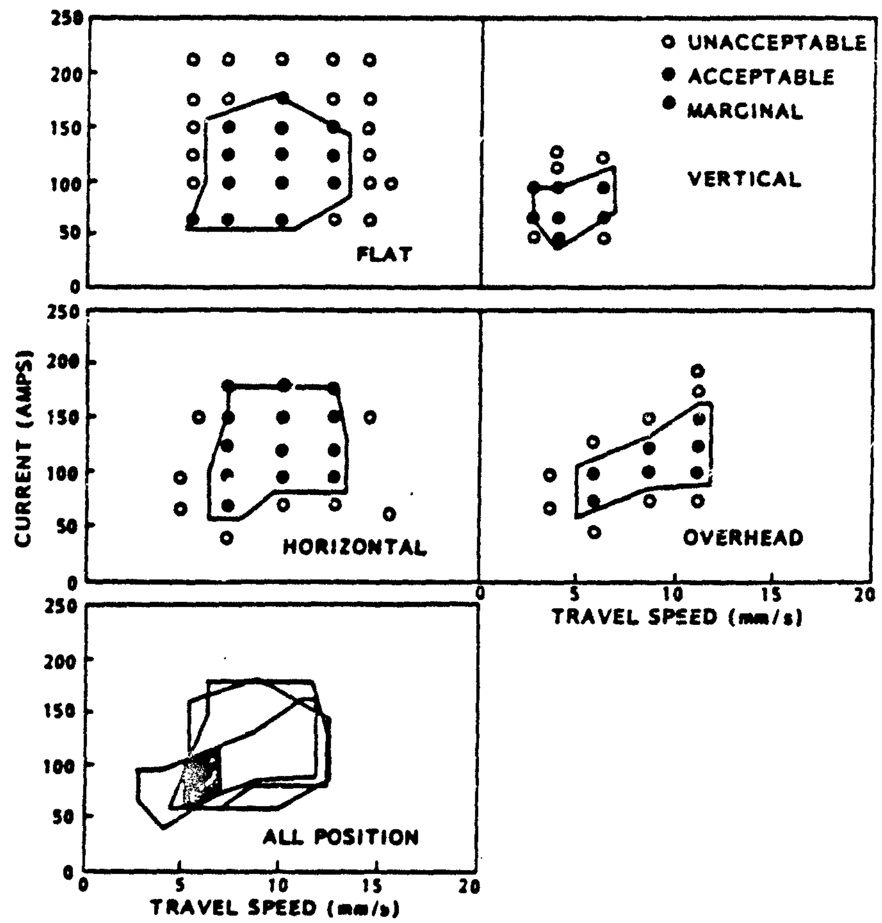

FIGURE 5

INDIVIDUAL AND ALL POSITION TOLERANCE BOXES FOR PULSED GMAW BEAD-ON-PLATE WELDS

WIRE: CARBON STEEL 0.9mm DIA (0.035 IN)

GAS: 80% Ar, 20% CO_2

(1mm/s = 2.4 IN/MIN)

A similar heat input ratio approach was employed while developing the two-layer technique with pulsed GMAW. A stringer bead welding technique was used throughout the mechanized welding trials so that, if the test program were successful, the two-layer pulsed GMAW welding method could be assessed for manual applications. It is not practical to include weaving in manual welding procedures which attempt to control heat input with reasonable accuracy. However, fully mechanized procedures should take advantage of the benefits that weaving can produce.

A series of bead-on-plate tests were conducted to determine the acceptable current and travel speed tolerance boxes for all-position welding. Figure 5 shows the construction of the acceptable operating region. Subsequent tests were conducted using parameters from within these tolerance boxes.

A number of two-layer weld trials were performed on plate sections of SA516-70 to assess the effect of various heat input ratios on HAZ grain refinement. Table 3 lists the parameters used in each of the welding trials. All welds were made in the horizontal position using a preheat of 177°C and heat input ratios in the range 1.2 to 2.0. The HAZ regions were examined metallographically and hardness surveys and traverses were done to identify unrefined regions of hard transformation product. In addition to test coupons, a simulated repair cavity was filled in a SA516-70 plate using the two-layer weld technique to butter the surface followed by a normal fill procedure. Limited two-layer trials were performed on a section of A533-B1 to demonstrate the grain refinement capability on steel of greater alloy content and higher carbon equivalent.

TABLE 3

WELDING PARAMETERS FOR PULSED GMAW
TWO-LAYER REFINEMENT TECHNIQUE WELD TESTS

TEST NUMBER	BASE MATERIAL	LAYER 1 CURRENT (AMPS)			LAYER 2 CURRENT (AMPS)			HEAT INPUT RATIO (HI_2/HI_1)
		CURRENT (AMPS)	VOLTAGE (VOLTS)	TRAVEL SPEED (mm/s)	CURRENT (AMPS)	VOLTAGE (VOLTS)	TRAVEL SPEED (mm/s)	
1	SA 516-70	125	24	6.35	175	24	6.35	1.4
2	SA 516-70	125	24	8.47	175	24	6.35	1.8
3	SA 516-70	125	24	6.35	150	24	6.35	1.2
4	SA 516-70	125	24	8.47	150	24	6.35	1.6
5	SA 516-70	125	24	10.58	150	24	6.35	2.0
6	SA 533-B1	125	24	6.35	175	24	6.35	1.4

NOTES: - TESTS WERE PERFORMED IN THE HORIZONTAL POSITION
- PREHEAT WAS CONTROLLED AT 177°C
- WIRE ER70S-3, 0.9MM DIAMETER (1MM/S = 2.4 IN/MIN)

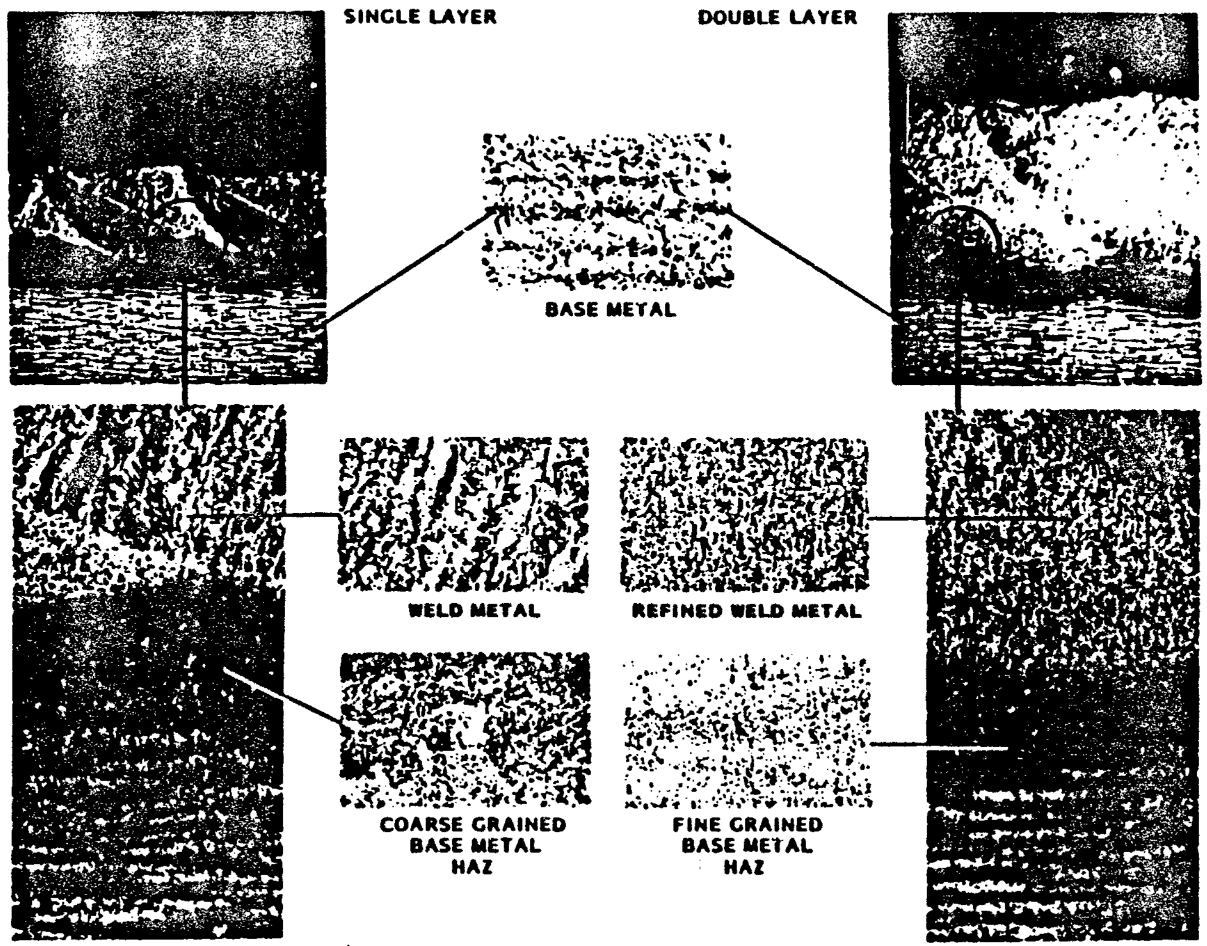

FIGURE 6 MICROSTRUCTURAL CONSTITUENTS FOR SINGLE AND DOUBLE LAYER PULSED GMAW REPAIR WELDS

5.0 TEST RESULTS

5.1 Metallography

Metallography was performed on each of the weld test coupons to assess the effectiveness of GMAW two-layer refinement. The various microstructural regions can be easily identified as illustrated in Figure 6 for single and two-layer HAZ regions. The grain-refined HAZ area was examined and the amount of grain refinement was determined over a distance of 20 mm (0.79 inches). An arbitrary value of 30 μm (0.0012 inches) was chosen to differentiate between coarse and fine grains. Figure 7 shows the influence of the heat input ratio on grain refinement for two-layer repair coupons produced at 177°C preheat. Two measurements of grain refinement were also made on single layer weld coupons which would represent worst case unrefined area. The results are plotted on Figure 7 at zero heat input ratio since only one weld layer was deposited.

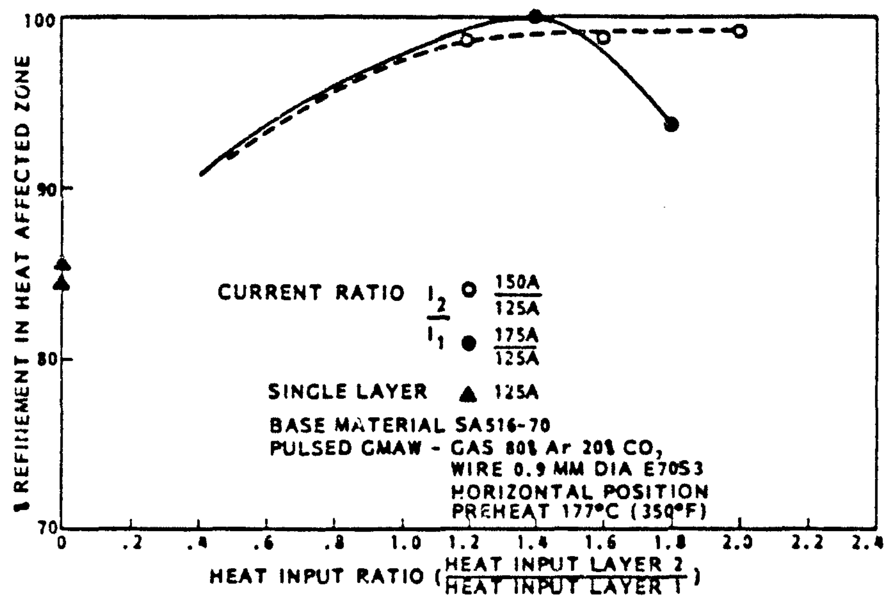

FIGURE 7

% REFINEMENT AS A FUNCTION OF HEAT INPUT RATIO FOR PULSED GMAW TWO-LAYER REFINEMENT WELD REPAIRS ON SA516-70 CARBON STEEL

5.2 Hardness Test Results

Hardness profiles were performed on single and two-layer weld coupons to assess the influence of the two-layer technique on hardness reduction. Figure 8 shows hardness values plotted against distance from the fusion line for both single and two-layer welds. Hardness profiles were taken in the region of maximum penetration and the cusp areas of the weld coupons as illustrated in Figure 8.

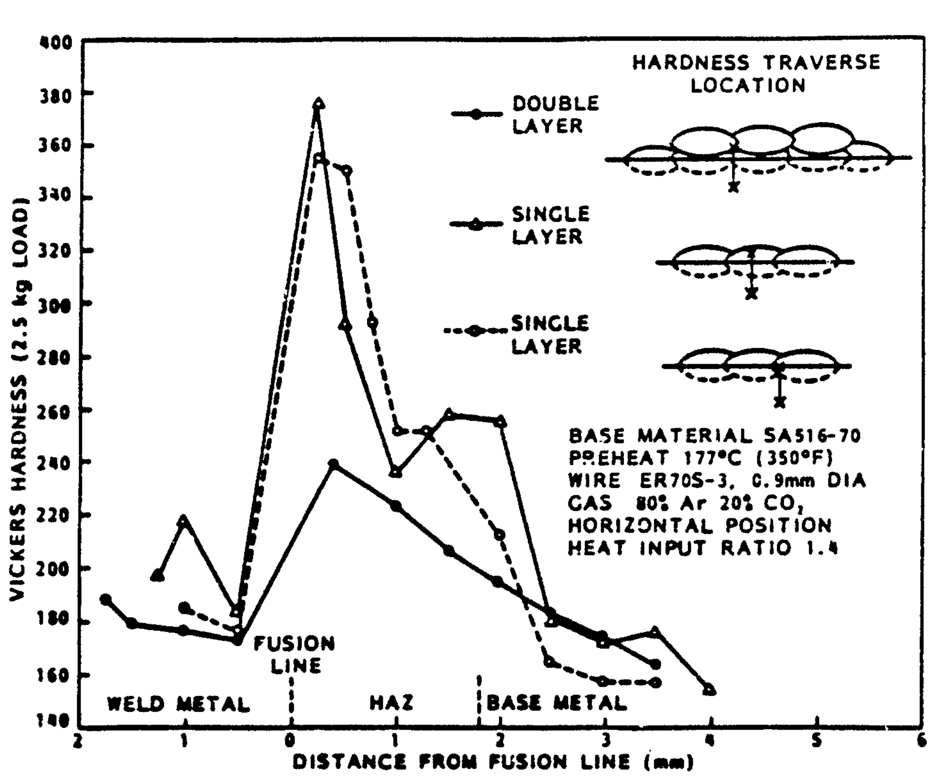

FIGURE 8

HARDNESS TRAVERSES THROUGH SINGLE LAYER AND TWO-LAYER HEAT AFFECTED ZONES IN PULSED GMAW REPAIR WELDS IN SA 516-70 CARBON STEEL (1mm = 0.04 IN)

Peak hardness values for single layer welds were typically in the 350 to 450 Hv range. The two-layer technique reduced the peak hardness to less than 300 Hv in welds with optimum heat input ratios.

5.3 Repair Cavity Simulation

Following the two-layer trials and examination of the resultant HAZ microstructures, a simulated repair weld was made in a section of SA516-70 plate using parameters which would produce near optimum HAZ refinement. The cavity was made to the dimensions shown in Figure 9. All welding was performed by a mechanized torch positioner and carried out in the horizontal position. The first two layers were applied using a stringer bead buttering technique with a heat input ratio of 1.3. The welding parameters used to fill the cavity are given in Table 4. A cross section was taken through the centre of the cavity and examined metallographically, revealing no defects as illustrated in Figure 10. Hardness traverses were made through the HAZ in each of the three wall regions of the repair. The hardness locations and values are also shown in Figure 10. It should be noted that the parameters listed in Table 4, were chosen based on the results of tests performed in the horizontal position. Although the repair cavity was oriented in the horizontal position, the top and bottom walls required the welding position to approach overhead and flat respectively. No attempt was made to optimize parameters to achieve refinement in these positions. Consequently, the overhead penetration profile was somewhat less consistent than the back and lower wall sections of the cavity. However, as the hardness traverses indicate, the thermal cycles imposed on the HAZ were adequate to reduce the hardness to a very acceptable level.

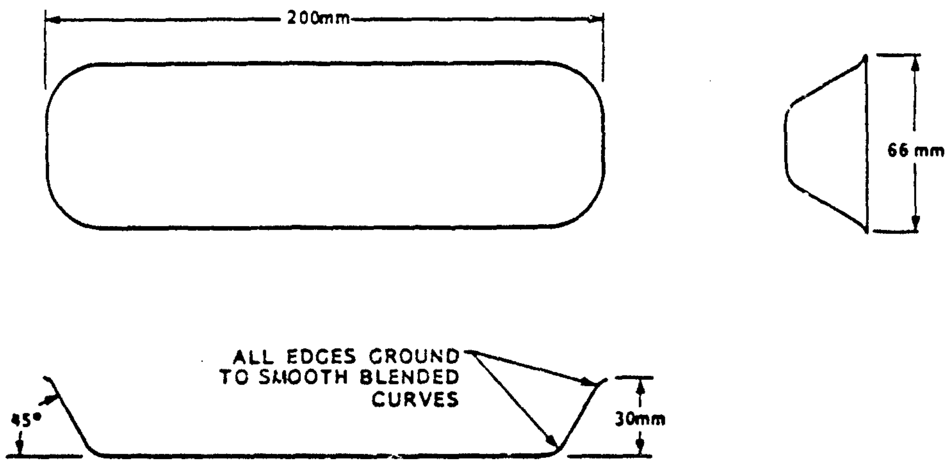

FIGURE 9

WELD DIMENSIONS OF REPAIR SIMULATION CAVITY FOR MECHANIZED PULSED GMAW TWO-LAYER REPAIR TECHNIQUE (1mm=0.04 IN)

TABLE 4

WELDING PARAMETERS FOR PULSED GMAW SIMULATED TWO-LAYER WELD REPAIR

		LAYER 1	LAYER 2	FILL PASSES
BASE MATERIAL SA 516-70	CURRENT (AMPS)	100	130	130
	VOLTAGE (VOLTS)	24	24	24
	TRAVEL SPEED(MM/S)	6.35	6.35	6.35

HORIZONTAL POSITION
TWO-LAYER HEAT INPUT RATIO 1.3
PREHEAT - 177°C (350°F)

WIRE - ER70S-3, 0.9MM DIAMETER
SHIELDING GAS - 80% Ar 20% CO_2
(1MM/S = 2.4 IN/MIN)

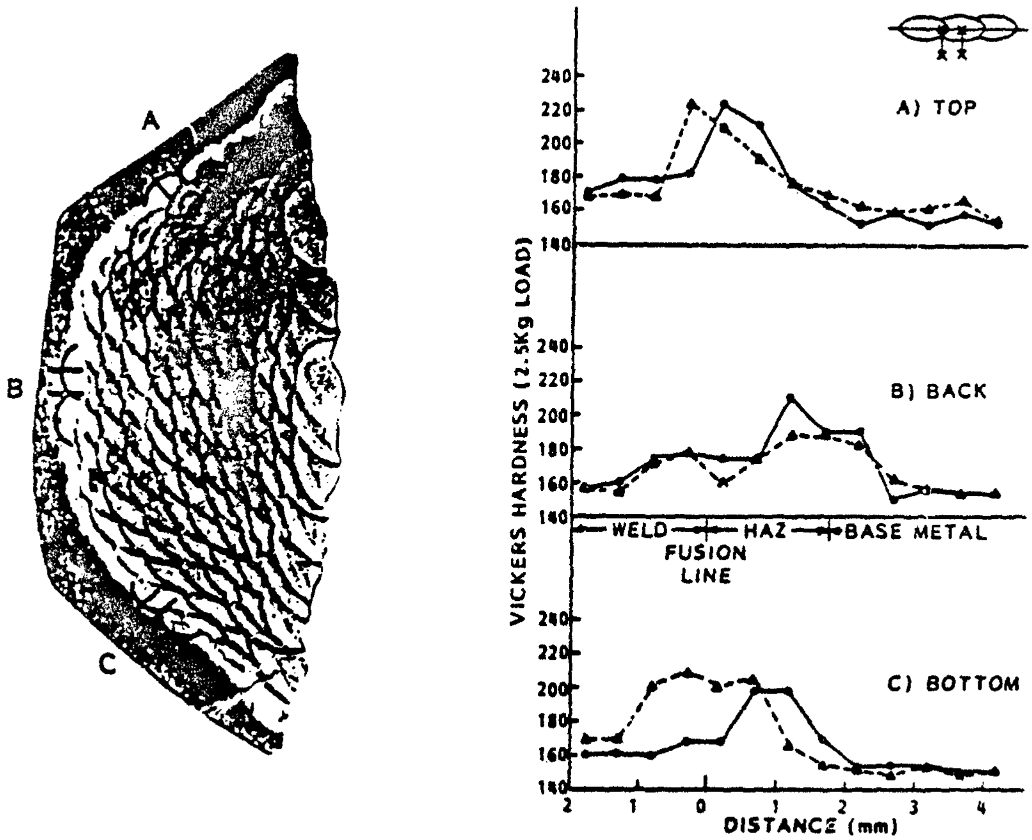

FIGURE 10

HARDNESS TRAVERSE VALUES FOR WELD REPAIR SIMULATION. SA516-70 USING PULSED GMAW AND THE TWO-LAYER REFINEMENT TECHNIQUE (1mm=0.04 IN)

5.4 Two-Layer Repairs on SA533-B1

To demonstrate the effect of the two-layer technique on a steel other than SA516-70, a section of SA533-B1 was used and a two-layer test weld was performed on the surface in the horizontal position. Preheat was controlled at 177°C (350°F) and a heat input ratio of 1.4 was used for the two first layers as indicated in Table 3. Hardness traverses were performed on a polished section throughout the weld in single and two-layer regions. As the results of the hardness tests in Figure 11 indicate, the second layer thermal cycle has reduced the HAZ peak hardness significantly.

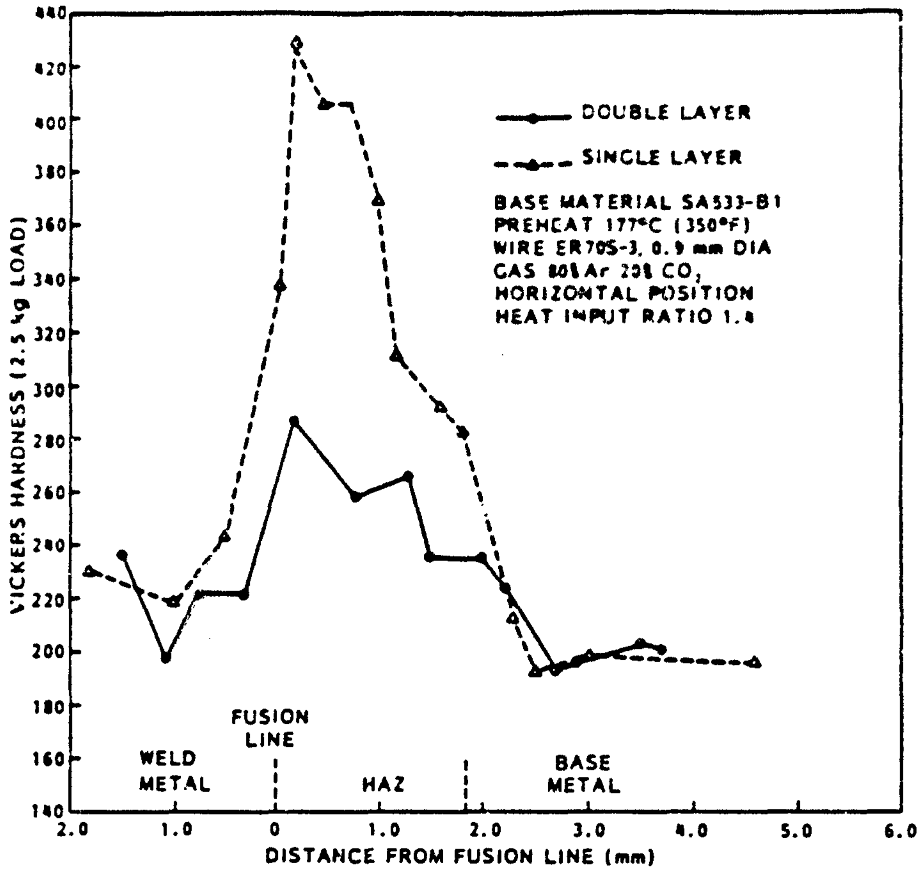

FIGURE 11

HARDNESS TRAVERSES THROUGH SINGLE LAYER AND TWO-LAYER HEAT AFFECTED ZONES IN PULSED GMAW REPAIR WELDS IN SA533-B1 STEEL (1mm = 0.04 IN)

6.0 DISCUSSION

6.1 Grain Refinement Mechanism

The success of the two-layer repair welding technique is based on subjecting the weld HAZ to a controlled sequence of thermal cycles. The aim is to achieve full HAZ grain refinement with two layers of weld metal leaving the remainder of the repair to be filled using a normal welding technique. The first layer is deposited at a fixed heat input such that a consistent deposit thickness is achieved. The bead spacing and torch orientation are chosen to produce smooth deposits of constant height. This is achieved by aiming the electrode at the toe of the previous bead, resulting in weld beads with approximately 50% overlap. More overlap would cause layer height build up, and a smaller amount of overlap would produce a ripple effect on the weld deposit surface.

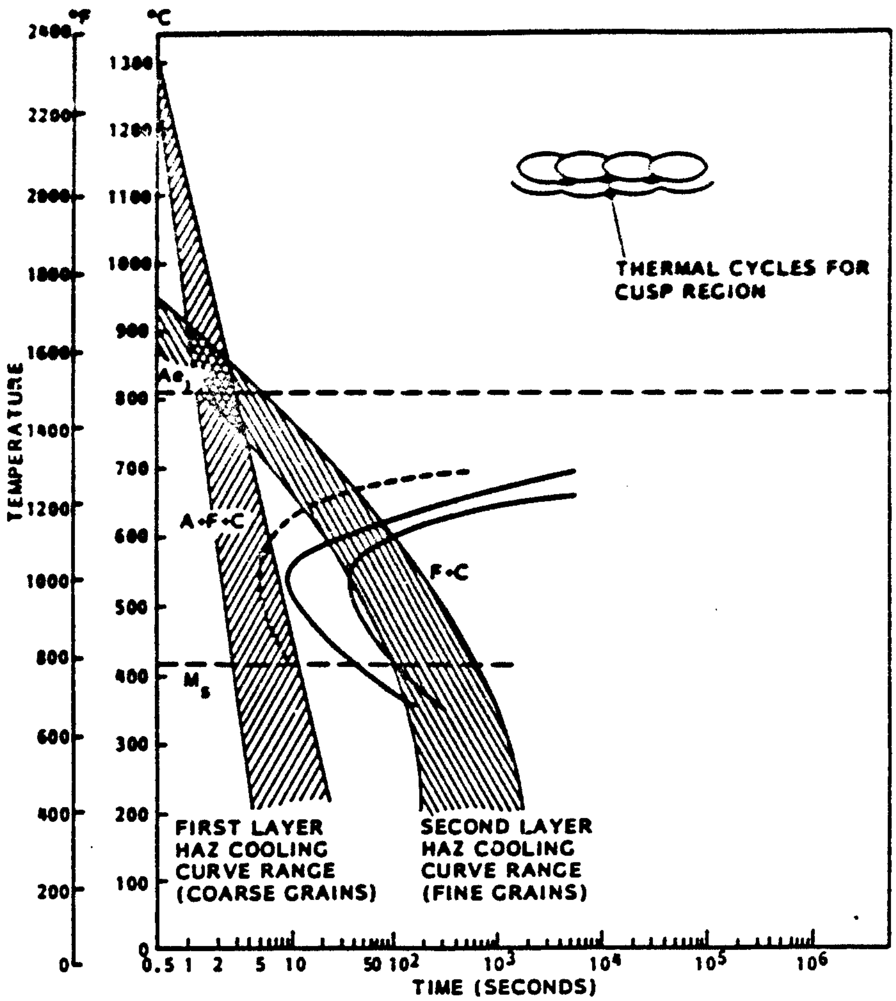

FIGURE 12

HEAT-AFFECTED ZONE CUSP REGION IDEAL THERMAL CYCLES FOR TWO-LAYER GRAIN REFINEMENT WELDS IN SA516-70 (A-AUSTENITE, F-FERRITE, C-CARBIDE, M_S-MARTENSITE START)

The HAZ in single layer welds contains small pockets of unrefined microstructure, usually of higher hardness. These are in the cusp region between weld beads where the coarse-grained HAZ is not refined by adjacent beads of the same layer. The second layer weld parameters must be chosen such that these regions are subjected to a thermal cycle with peak temperatures in the lower austenitizing region. Figure 12 schematically illustrates the thermal cycles in the HAZ for ideal heat inputs from the first and second layers of weld deposit. In two-layer welds, several locations are prone to having coarse-grained regions of higher hardness depending on the welding parameters used. Hard regions have been found in areas where the second layer heat input is excessive, causing the temperature in the base plate to reach the upper austenitizing range and upon cooling form a new coarse-grained region. Hard areas can also remain in the HAZ if the second layer heat input is insufficient to grain-refine the cusps in the first layer HAZ. A similar problem can occur if improper bead spacing is used on either the first or second layer, resulting in an inconsistent weld surface profile.

6.2 Control of Welding Parameters

The data in Figure 7 indicate that there is a narrow range of parameters which can be used to achieve full HAZ refinement and hardness reduction with the two-layer technique. This data was generated welding in the horizontal position and would not likely be optimum for vertical or overhead welding. A similar set of experiments would be required in each welding position to determine parameters which would provide full grain refinement. Ideally, one set of parameters could be used for the two-layer refinement passes, but it is likely that the welding parameters would have to be selected according to welding position. This suggests that a rather complex welding procedure would be necessary to produce acceptable welds in a cavity where access is limited and all-position welding would be required.

A number of trials have been performed on simulated repairs using both manual and mechanized systems. It is apparent from the results of these trials that a great degree of control is required for travel speed, bead placement and torch orientation. It is also evident that manual welding may not attain the close parameter control required to achieve the desired refinement levels. If one examines Figure 7 however, there is some indication that over a wide range of travel speeds, a high degree of refinement can be produced. Although this is only a preliminary observation, other parameters which can vary in manual welding would also have to be examined as to their influence on refinement. It is quite possible that the tolerance limits for each of the uncontrolled variables in manual pulsed GMAW would, combined, be unable to provide an acceptable operating range for full refinement with the two-layer technique.

6.3 Future Areas of Development

Although it can be shown that successful two-layer welds with fully refined HAZ regions can be made in order to avoid the use of PWHT, a number of problem areas remain which must be addressed. It is reasonably clear that manual pulsed GMAW welding does not possess the control necessary to produce welds consistent enough for the two-layer application. Optimum conditions of travel speed, welding current, bead spacing and torch orientation must remain relatively constant to maintain the correct heat input ratio and produce adequate grain refinement. Mechanized welding, which is limited to only regular geometries and can require a large degree of manual adjustment, is an intermediate step in the development of repair systems. For real applications involving repairs on cavities or geometries which have irregular contours, suitable control of deposition could only be achieved by a positioner or robot which possesses a high degree of motion control.

Further work is also required in the area of deposition sequencing for all positions, and electrode manipulation or weaving during welding. In addition, detailed procedures are required to deal with adequate refinement in weld toe regions and in areas of weld stops and starts.

7.0 SUMMARY

The use of pulsed GMAW has been demonstrated as a potential alternative to the ASME half-bead SMAW technique for two-layer non-PWHT repair welds for nuclear power plant piping and vessel repairs in carbon-manganese steels. HAZ grain refinement and hardness reduction can be controlled using the proper combination of heat input and deposition sequence. Although a mechanized welding procedure can produce acceptable results on geometrically regular surfaces, increased dexterity, such as that provided by robotic motion control would be required for irregularly shaped cavity repairs. It is unlikely that sufficient control of welding parameters can be obtained for this two-layer technique using manual pulsed-GMAW to achieve acceptable results. Other areas of development, such as deposition sequence optimization and grain refinement in areas of weld discontinuities (starts, stops, toe regions), require additional effort.

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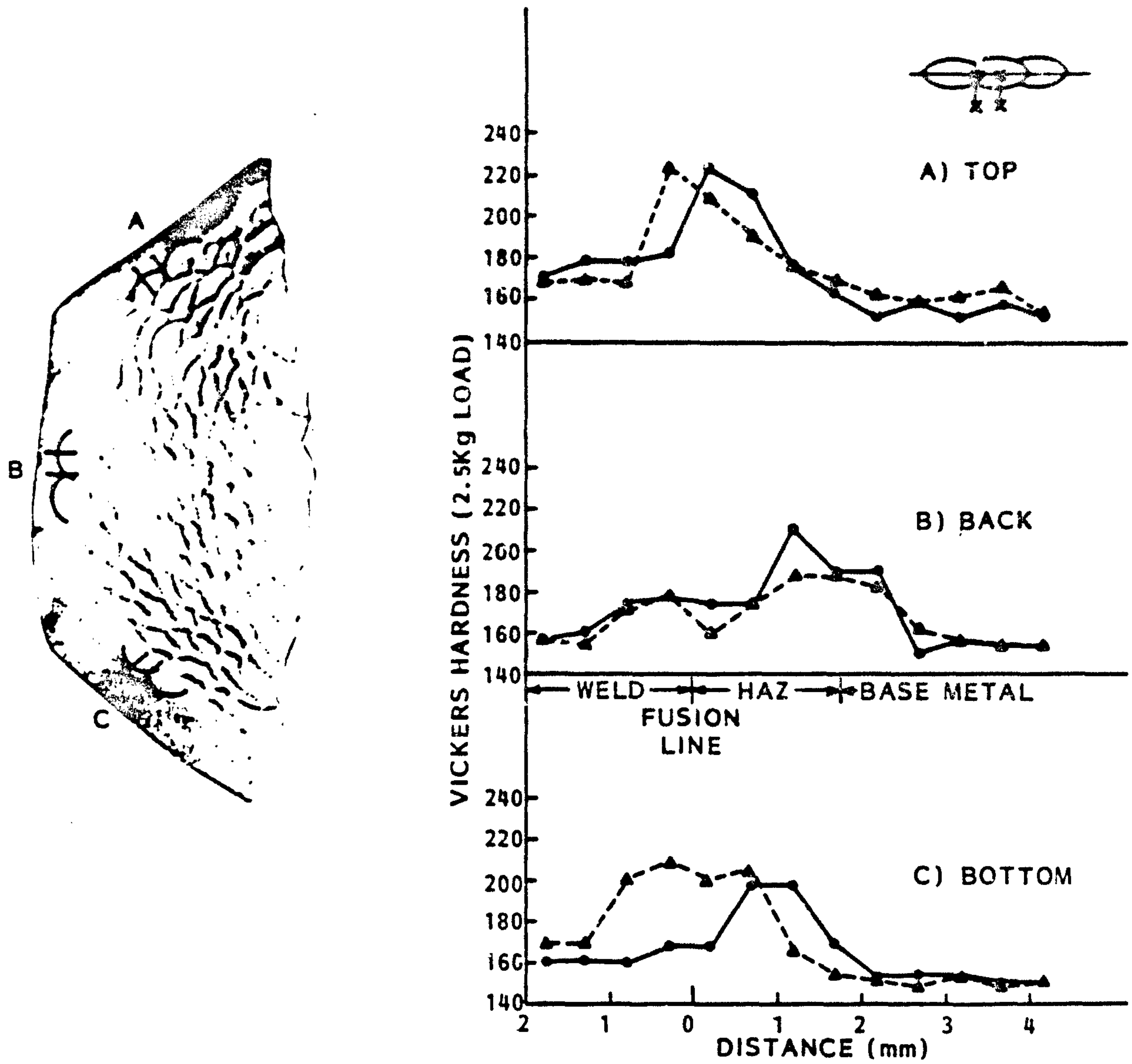

FIGURE 10

HARDNESS TRAVERSE VALUES FOR WELD REPAIR SIMULATION. SA516-70 USING PULSED GMAW AND THE TWO-LAYER REFINEMENT TECHNIQUE (1mm=0.04 IN)

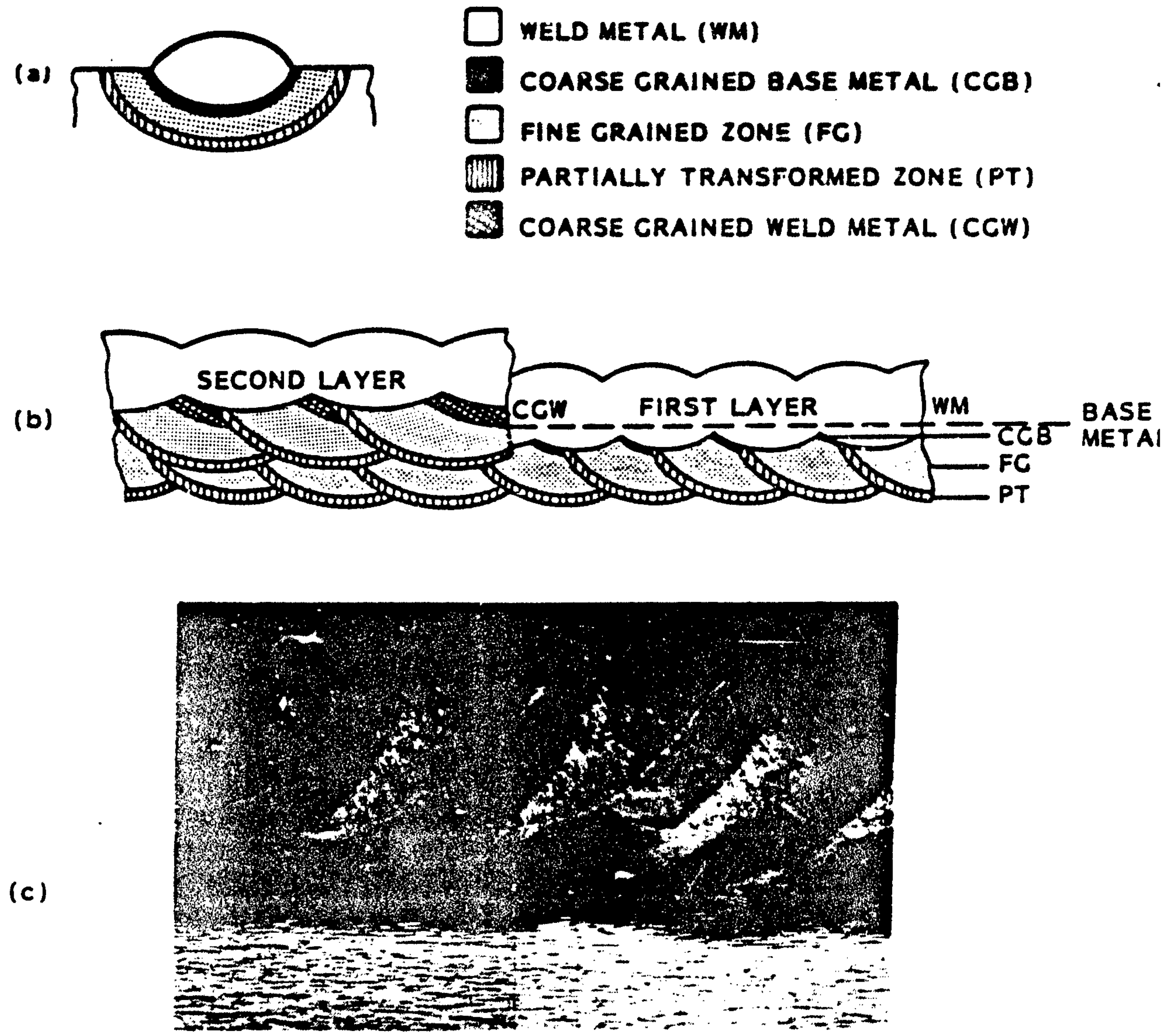

FIGURE 1

TWO-LAYER GRAIN REFINEMENT WELDING TECHNIQUE
(A) SINGLE WELD BEAD
(B) CORRECTLY APPLIED WELD BEADS ILLUSTRATING REFINEMENT OF BASE PLATE COARSE GRAINED ZONE
(C) CROSS SECTION OF PULSED GMAW TWO-LAYER TECHNIQUE

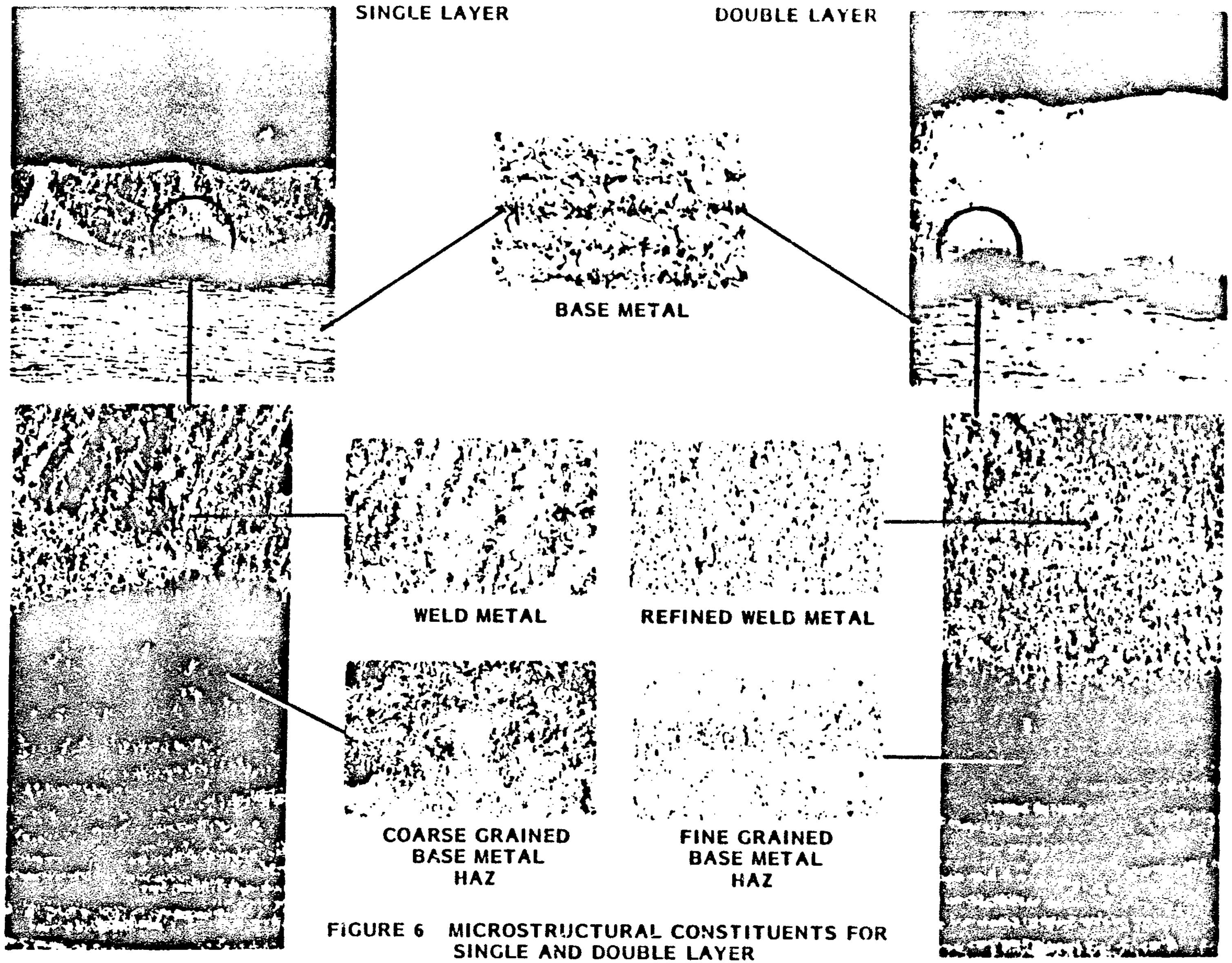

FIGURE 6 MICROSTRUCTURAL CONSTITUENTS FOR SINGLE AND DOUBLE LAYER PULSED GMAW REPAIR WELDS

APPLICATIONS OF EXPLOSIVE WELDING

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INTRODUCTION

The explosive welding process was first noted in ordnance tests where fragments from exploding projectiles would become welded to metal targets. Also, in early studies of explosive forming, it was discovered that an overcharge system would often result in portions of the formed part becoming welded to the die cavity. [1] Studies to prevent welding from occurring during other explosive forming operations led to the new field of explosive welding.

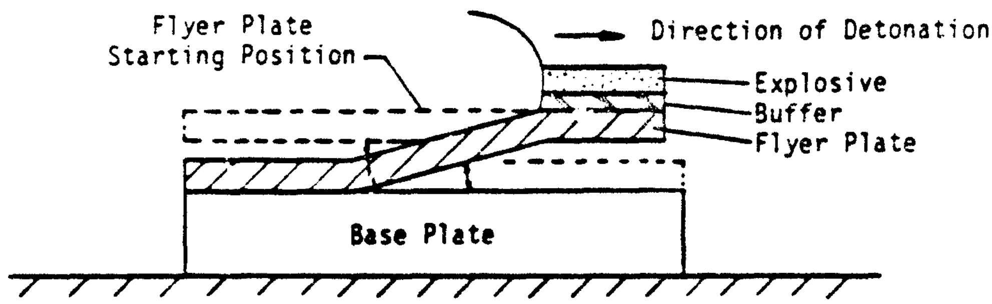
The process was first viewed mainly as a replacement for conventional welding in remote locations. Interest developed, however, when it was demonstrated that strong welds could be obtained when joining metal combinations that possessed widely different melting points, greatly different thermal expansion characteristics and large hardness differences-property variations which do not lend themselves to conventional welding techniques. By 1967 more than 260 combinations of similar and dissimilar metals had been explosively welded [2].

This paper discusses briefly the mechanism of explosive welding, some of the advantages of the process, typical applications and finally, describes in detail the development of a tube-to-tubesheet welding process aimed specifically at the fabrication and repair of nuclear power plant condensers and high integrity heat exchangers.

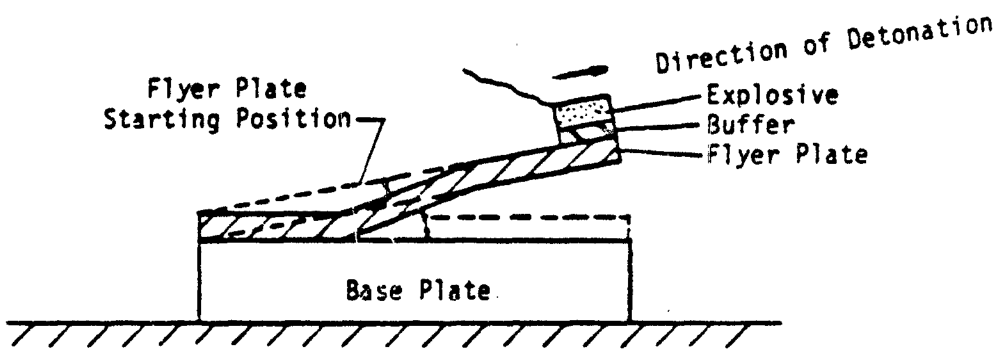
THE MECHANISM OF EXPLOSIVE WELDING

Figure 1 shows the basic set-up used for explosive welding. The parts to be joined are prearranged so that upon detonation, the driven component called the flyer plate, is caused to collide with the stationary component called the base plate, at a specific angle. In order to achieve this angle and produce a weld, the important initial parameters are the standoff gap and the amount and the detonation velocity of the explosive used. When these parameters are correct, the following events occur to produce a metallurgical bond between the two components:

(1) A very thin layer of melting occurs at the collision point. Most of this molten material is ejected by jetting from the interface. With this jet of molten material, the surface oxides and contaminants are removed.



Arrangement for Parallel Plate Welding



Arrangement for Welding with Plates
Initially at an Angle

FIGURE 1

Two Basic Arrangements of Components for Explosive Welding

(2) Immediately following the jetting of molten material is an increase in pressure at the interface. This pressure is maintained for a sufficient length of time to achieve interatomic bonding.

Figure 2 shows the microstructure of a typical explosively formed bond between a titanium tube and a titanium tubesheet.

FIGURE 2

Titanium Tube Explosively Welded to a Titanium Tubesheet
Wavy Interface Typical of Explosively Created Bonds

ADVANTAGES OF EXPLOSIVE WELDING

Explosive welding offers distinct advantages over fusion welding in many applications. Some of these advantages are:

(1) Ability to bond dissimilar metals that are either difficult or impossible to join by other methods.

(2) There is no heat affected zone, thus the resultant weld is as strong as the weakest base material, free from any detrimental effects generally associated with heat affected zones.

(3) For many applications, such as large area cladding and high volume tube-to-tubesheet weldments, the costs are lower than those for conventional fusion welding operations.

(4) The time for performance of the weld is faster than conventional welding methods. The explosive devices needed to produce the welds can be prefabricated, thus the actual time to perform the welding

is very short, making this process ideally suited for field repairs where short turn around time is essential.

(5) The process can easily be performed remotely, making it a valuable tool for welding in nuclear or other hazardous environments.

These are the more obvious advantages, others become apparent when specific applications are introduced.

SOME APPLICATIONS OF EXPLOSIVE WELDING

Described below are some of the applications of explosive welding now being extensively used:

(1) Cladding - Explosively clad plates are extensively used in the chemical and power industries to provide corrosion protection for vessels, piping and heat exchanger equipment. The process allows designers to use less expensive carbon steel alloys as the structural components of the equipment, using only a thin clad layer of more expensive corrosion or wear-resistant metals to inhibit the inservice attack of hostile environments.

(2) Tube-to-Tubesheet Joining - Shell and tube heat exchanger designs usually require secure leak-tight attachment of the tubes to the tubesheets. Explosive welding provides a method of obtaining strong leak-tight tube-to-tubesheet joints. Especially attractive is the ability to join tubes of a material dissimilar to the tubesheet. The body of this paper will discuss this application more thoroughly.

(3) Tube Plugging - Leaking tubes of heat exchangers are commonly plugged to prevent mixing of the shell side and tube side fluids. Explosive welding has been used extensively to secure plugs into these leaking tubes. The process has become popular because of the speed of installation and the ready adaptation to remote handling.

(4) Transition Elements - Often in equipment design it becomes necessary to join dissimilar metals such as aluminum bulkheads to steel decks of modern ships. Aluminum/steel transition elements are explosively bonded. These elements are then fusion welded in the yards by conventionally welding the aluminum side of the element to the bulkhead and the steel elements to the deck. This is only one example, many others are commonly in use.

EPRI/FOSTER WHEELER TUBE-TO-TUBESHEET PROCESS

Previous explosive welding of tubes to tubesheets were limited to cases where the tubesheets are relatively thick (typically 5 to 10 times the tube diameter or greater) with generous ligaments. The process was also limited to single expansions. Special development was needed for condenser applications because of the thin tubesheets and the need for mass production of the welds. Thus, the Electric Power Research Institute (EPRI) embarked upon an extensive program to develop an explosive tube-to-tubesheet welding process tailored primarily for welding tubes to nuclear condenser tubesheets. EPRI contracted the development to Lockheed Missiles and Space Corporation. Aside from removing the two major restrictions of thick tubesheets and single detonations, the program was also focused on the following additional requirements:

* To develop the ability to bond dissimilar corrosion-resistant materials generic to condenser fabrication.

* Complete bonding at the front face of the tubesheet and closure of the crevice to the back face.

* A bond length of at least six times the tube wall thickness.

* The explosive charge package was to be safe, reliable and cost effective.

In order to provide bonding at the face of the tubesheet, it was found necessary to initiate the explosive from within the tube and have the direction of bonding occur toward the tubesheet face (see Figure 3). This type of detonation was achieved using a charge package as shown in Figure 4. The main bonding charge is initiated by a small transfer charge. The bonding charge then detonates back toward the tubesheet face. The bond then occurs until it reaches the face of the tubesheet. At this point the projecting tube shears off

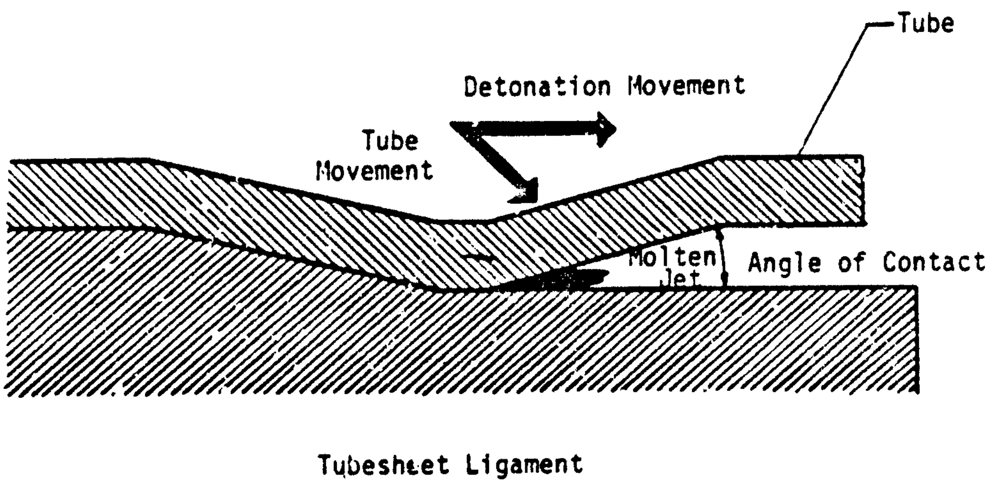


FIGURE 3
EPRI/FW Explosive Tube Welding at
Movement of Impact

leaving a crevice-free terminus of the bond. By carefully tailoring the plastic behind the charge, the unbonded portion of the tube is expanded into the tubesheet effectively eliminating any crevice between them.

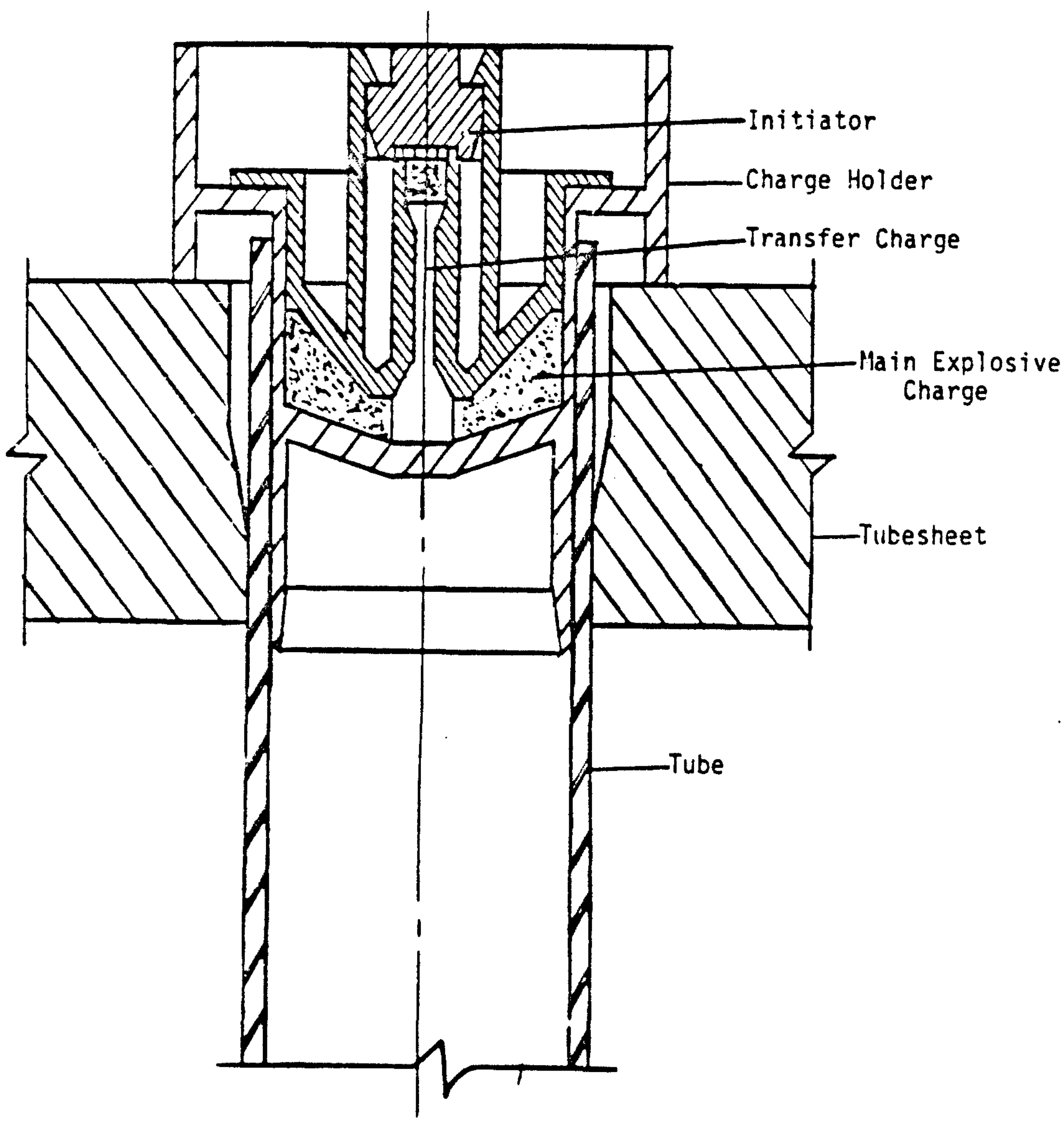


FIGURE 4

EPRI/FW Tube Welding Charge Package in Position Ready for Detonation

Charge parameters and the tube hole profiles were developed to consistently produce axial bond lengths that exceeded six times the thickness of the tube wall. This level of bond length provides the joint with an axial strength larger than the base tube and a bond thicker than the tube wall.

Because of the large numbers of tubes contained by condensers and other heat exchangers, it is advantageous, for economical reasons, to bond large numbers of tubes simultaneously. A firing system was devised which permitted large numbers of tubes to be welded by a single detonation with the detonation front being controlled using a minimum amount of transfer explosive (see Figure 5).

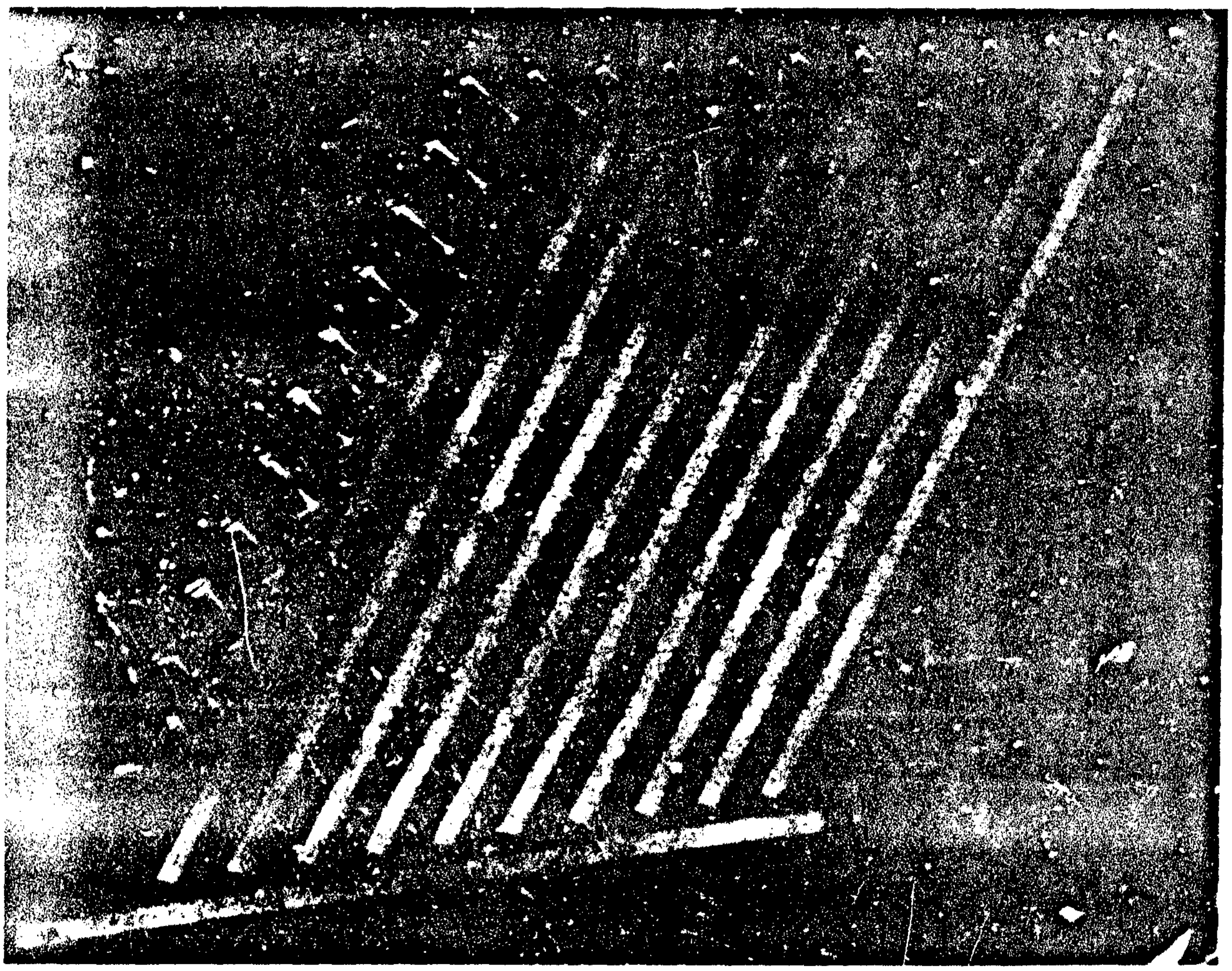


FIGURE 5

Multiple Firing System for Explosive Tube-to-Tubesheet Welding

The explosives selected for the bonding charges were chosen for performance, reliability and safety. The main or bonding charge is nitroguanadine. This explosive is procured as a fluffy powder and can be compacted into molds to form whatever shape is desirable. The detonation velocity of this explosive is a function of its density, thus it lends itself well to explosive welding applications where detonation velocity is critical. This explosive is extremely safe to handle and will merely burn when exposed to open flame.

The remainder of the charge system, the detonation train, is composed of PETN, another relatively insensitive explosive, but one which detonates with sufficient energy to initiate the nitroguanadine.

APPLICATION OF THE EPRI/FW PROCESS

The process was developed by EPRI for nuclear condenser applications with special emphasis on bonding titanium tubes to either titanium or stainless steel tubesheets.

Foster Wheeler has extended the technology for other applications. The material combinations and sizes thus far successfully bonded, are shown on Table 1. These combinations have been developed for application on geothermal, nuclear steam generators, fossil plant and OTEC (Ocean Thermal Energy Conversion) condensers, feedwater heaters and other critical service heat exchangers.

TABLE 1

Explosively Bonded Tube-to-Tubesheet Material Combinations and Sizes Using the EPRI/FW Process

Material		Tube Diameter (inches)	Tube Wall Thickness (inches)	Tubesheet Thickness (inches)	Potential Application
Tube	Tubesheet				
Titanium	304 SS	1.000	0.022	1.000	Condenser
Titanium	Carbon Steel	1.000	0.022	2.000	Condenser
Titanium	Titanium	1.000	0.022	1.000	Condenser
Titanium	Titanium	1.125	0.028	1.125	Condenser
Titanium	Muntz	1.000	0.022	1.000	Condenser
Titanium	Titanium	0.875	0.028	1.000	Condenser
Titanium	Titanium	0.750	0.028	1.000	Condenser
Al 29-4C	304 SS	1.125	0.028	1.125	Condenser
304 SS	304 SS	0.750	0.035	4.000	Sleeving
304 SS	Muntz	1.000	0.028	1.000	Condenser
Inconel 690	Inconel 600	0.750	0.035	4.000	Sleeving
Carbon Steel	Carbon Steel	0.750	0.030	4.000	F.W. Heater
Aluminum	Aluminum	1.000	0.062	1.000	OTEC
Aluminum	Aluminum	1.000	0.072	2.000	Geothermal

As of this writing, FW has been given a contract by ALCAN International Limited to adapt this process to explosively weld 16 OTEC condenser test modules containing 19 tubes each. The tubes are 1" O.D. x .072" wall welded to 2" thick tubesheets. The tubes and tubesheets are made of four different grades of aluminum. Preliminary qualification tests showed this process to be well suited for such an application.

CLOSING REMARKS

A promising, improved process for explosively welding tubes to tubesheets has been developed. This process can be applied to thin tubesheets without significant change to traditional ligament efficiency parameters. The multiple detonation initiating device makes this process ideally suited for mass production and hence field retubing or repair projects. The process was found to be highly reliable, and safe because of the use of nitroguanadine as the energy source. Initial applications of this process confirms our laboratory evidence of these advantages.

Since acquiring the license from EPRI, Foster Wheeler has extended the process development to support other potential applications including:

a. Smaller size tubes typical of application to feedwater heaters and chemical plant heat exchangers.

b. Sleeving operations for nuclear steam generators and other high integrity heat exchangers.

c. Tube plugging in which a weld is desired.

Our laboratory results on these other applications are also very promising. We are now talking to a number of utilities to discuss possibility of demonstrating the effectiveness of this process by installing a few hundred tubes using this process during a scheduled retubing outage. Because of the significant advantages cited in the body of the paper on this process, we are convinced that this method produces a superior joint when compared to either conventional rolling or fusion welding.

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ALARA PLANNING

by

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Industry experience over the past several years has provided positive indications that conscientiously applied, aggressive planning towards the implementation of the As Low As is Reasonably Achievable (ALARA) philosophy can provide pleasant "payback" results that go beyond the saving of person-rem, the normal yardstick. Major outages at nuclear power facilities, specifically the "recirculation piping replacement" projects at several BWRs, have supplied both the opportunity and the proper test environment for ALARA concepts, approaches, practices, and personnel in order to provide a more complete picture of their viability and value. The lessons-learned are the direct result of Hydro Nuclear's collective experience during the Recirculation Piping Replacement Project at Georgia Power Company's Plant Edwin I. Hatch, Unit II. ALARA embraces more than those time-honored considerations of "time, distance, and shielding." The tools which provide ALARA programs this expanded capability each increase a nuclear utility's capacity for effective planning, reduced downtime and expenses, and enhanced profitability.

The primary purpose of a nuclear power facility is to produce low cost, reliable electricity. Radiation protection programs are an essential, integral part of the operations make-up of nuclear power facilities and can not be influenced by production-oriented considerations in performing their assigned functions. ALARA, both as a concept and as a functional area, is generally categorized with radiation protection programs. This categorization carries with it all the attendant connotations, including the myths that 1) ALARA is not to be influenced by economic considerations and 2) that ALARA cannot and should not have a positive effect on a company's balance sheet. For some, these myths have been dispelled and replaced with an expanded appreciation for the potential benefits of properly applying ALARA planning tools and techniques.

The tools of the trade for ALARA planning can be categorized into three broad categories: equipment (hardware and software), expectations and communications. Each area has a number of specific implementation choices, some of which include:

Equipment

- The availability and use of data gathering and manipulation/analysis equipment (primarily computers);
- The innovative application of doserate monitoring and telecommunications equipment;

Expectations

- The setting of goals and objectives;
- The monitoring of Key Performance Indicators;

Communications

- o The providing of lessons-learned on a routine basis;
- o The consistent and active involvement of personnel at each level of participation from those "down in the trenches" to the chief executive officer.

The approach to planning is individually determined and strongly influenced by past experience, habit, perceived organizational preference, and the perceived level of aggravation to accomplish the desired planning, and available time. The amount of planning accomplished often appears to be determined by the available time and perceived aggravation level. The quality of the planning effort is most often determined by the planner's level of confidence that the fruits of his labor will receive a reasonable reception and be acted upon by the "decision maker" in an objective manner. Experience is now showing that the benefits being derived from conscientiously planning maintenance activities far outweigh the work and potential aggravation.

The advent of the computer-assisted "information explosion" has provided planners with access to data that had been historically carried around in someone's head or written on some obscure slip of paper and stored in an already overflowing file drawer. Trending was usually accomplished by recounting the number of instances that could be remembered while in a meeting or while talking on the telephone. Telephone surveys were made to ascertain a sister utility's experiences in a particular situation, or with a particular pump, valve, or other component. Unfortunately, a sense of frustration accompanied most surveys of this type because the information was almost always preceded by caviats like "I think..." or "I believe..." The credibility of an ALARA program does not ride confidently upon this type of information, especially when personnel safety is involved. Good records and readily retrievable information do not come without a price, however, and the cost must be justifiable.

A few of the benefits derived from having readily retrievable information concerning past maintenance activities are:

- o Increased confidence in basis for decisions;
- o Improved probability of a "quality" decision;
- o Improved justifiability of resource expenditures, especially when questioned by public utility commissions;
- o Improved chances of avoiding the "reactionary syndrome" by consistently tracking symptoms to establish cause-and-effect relationships;
- o Improved probability for success of preventative maintenance activities and programs; and
- o Improved profitability by increasing megawatt output through increased equipment reliability, reducing overall resource allocation required to stay on-line, and reducing total elapsed time required to get back on-line once something has malfunctioned.

Some might question, "How can that be? ALARA is just one extra step in getting the job done." TIME and ATTITUDE are the key ingredients. During two recent outages for replacing recirculation piping, Outage "A" logged 122,000 and Outage "B" recorded 120,000 radiation work permit (RWP) manhours. The use of ALARA-instituted engineering controls caused a difference in the percentage of time workers spent wearing respiratory protection equipment from an estimated 47% to a measured 12%, respectively, when comparing the two projects. Using the generally accepted figure that working in respiratory protection gear reduces a person's productivity by about 50%, a similar ALARA action could have reduced the Outage "A" RWP manhours by approximately 23,000 manhours. Outage "A" took more than nine months to complete; Outage "B" required only three (3) days more than the originally scheduled 183 days. It is very difficult to ascertain if saving RWP manhours would result in saved calendar days, but the potential definitely exists.

The value of "purchased replacement power" for an off-the-line nuclear power facility is reluctantly considered by few, and made an integral part of daily decision-making by even fewer. This smaller number is usually the operations/outage manager and members of executive management. "It is their job to worry about things like that," most would say. True, but they are not the only ones affected. Numbers vary but most fall in the range from $400,000 - $1,000,000 per day. Conscientiously applied planning, with ALARA being an integral part, can quickly add up where it is obvious and appreciated, i.e., in the "money saved" department. TIME is money, TIME not spent in a radiation field is saved exposure. Therefore, time should become the primary consideration in both the planning and the decision-making processes of the following three groups:

- Utility management;
- ALARA personnel; and,
- All workers (both individually and collectively).

Computers are not mandatory for the acquisition and analysis of data useful to the planning process. The same end result can be accomplished by manually gathering, logging, sorting, and filing. The computer is simply faster in accomplishing these tasks, and should allow the clerical and professional manhours to be used more productively. An on-line system was installed and field-tested specifically for the Plant Hatch outage. Over 500,000,000 pieces of information were processed during this 186 day period. The number of manhours to accomplish the same feat would be prohibitive. The information that was used daily to make management decisions simply would not have been available.

Recovery from the incident at TMI 2 has provided both the need and incentive for serious exploration of increased application of remote surveillance and monitoring technology. Furthermore, it has led to the development of new equipment in this area. Reducing the amount of time spent in high radiation fields is a goal of every ALARA planning effort. Effective ways of reducing total manhours spent in a radiation field are:

- Eliminating the need for a person to make an entry;
- Providing "constant" job coverage without requiring a radiation protection technician to physically hold the detection instrument;
- Having the ability to know a worker's accumulated dose without requiring a work stoppage just to check dosimetry; and,
- Observing on-going work to identify and correct operational inefficiencies, and problems.

Remote visual surveillance of work locations and equipment operation provide the opportunity for supervisors to keep step with the pace of work and to stay that "step ahead" that often spells the difference between an outage that is ahead of schedule, o. ,chedule, or always having to play catch-up. And if this function can be accomplished without their entry into a radiation area, the total time spent receiving exposure is also reduced.

Several equipment applications available when planning the next job include:

- o Observing automatic cutting/welding machines for stoppages or other indications of trouble by using inexpensive TV cameras with pan-tilt-zoom (PTZ) capability;
- o Conducting shift turnovers from crew-to-crew on operations and equipment status by combining TV with two-way voice communications;
- o Reducing the time waiting for forgotten tools or materials by substituting two-way radio communications for "the runner";
- o Eliminating the guess work in predicting dose accumulation rates and staytimes by using portable doserate monitors with built-in telecommunications;
- o Reducing the risk of a worker either exceeding assigned dose limits or receiving a surprise dose that might jeopardize his ability to continue contributing to the job;

  NOTE: This can be especially important for personnel with critical, hard-to-replace skills and key supervisory personnel who usually excel because they do not mind getting their hands dirty.
- o Increasing the radiation protection technician's effectiveness by allowing one person at a monitoring console to provide coverage for multiple locations, thereby reducing the number of personnel required on shift by freeing personnel to perform other functions. The ongoing TMI 2 recovery effort will most likely provide the impetus for additional advances in equipment, techniques and applications. It is clear from the review of these efforts that the benefits are certainly there.

It is often asked if achievement can be measured if the desired end result is not clear. Individuals and organizations tend to rise to the perceived level of expectation set for them, whether by themselves or by others. The effectiveness of ALARA planning for maintenance outages is automatically self-limiting if goals and objectives are not established, examined, and re-established (as necessary) on an on-going basis. Many maintenance projects fall short in the "job satisfaction" department ONLY because overall goals and intermediate implementing objectives are never set. Executive management tends to remain uneasy because the Board of Directors may have to be approached with either a request for more money or an explanation of schedule slippage. The Project Manager tends to wonder if the project status being presented to the "boss," his peers or subordinates is accurate. The work supervisor tends to remain unsure how well the overall job is going and often resolves himself to take care of the business at hand, which usually means taking each individual assignment one at a time. The individual worker looks up the management chain for purpose and challenge so that he can feel good about his work. However, seeing little but uncertainty, he usually resolves to "merely put in his hours". On the otherhand, those maintenance projects that have taken the time to establish a high level of communicated expectation have usually produced visibly better results.

The Plant Hatch outage provided an excellent example of achieving the desired results by setting project goals and objectives and also taking the time to effectively communicate to all personnel the status of their project. The Project Manager expended the resources at the very beginning to have each functional area explained to all project personnel. He also explained how he was going to conduct business on a daily basis. Effective communications did not stop there, however. The Project Manager personally wrote a daily summary of the previous day's accomplishments and shortfalls on the project's status board. Each functional area was required to provide a daily update which included an assessment of any critical objectives or schedule milestones that were considered in jeopardy. Eleven man-months of planning and preparatory effort were expended on the area of radiation protection and ALARA <u>prior</u> to the first official day of the outage. Even so, there were enough surprises during the course of the work to provide plenty of opportunities for on-going ALARA planning. And the pipe-fitters, millwrights and laborers personally expressed such sentiments as "This is the first big job of which I have ever been made to feel a part."

Crafts personnel became progressively more interested in what the project's ALARA group was doing to keep their dose down so they could continue to work. This sincere degree of concern from the individuals involved and the Project Manager, provided both the opportunity and the impetus for the ALARA group to convert the crafts from sideline observers to active participants. This concern motivated participation from the crafts as a whole, and directly contributed to the project's successful completion only three days off schedule, but more importantly for 51% of the projected radiation exposure.

A recently published study prepared by Brookhaven National Laboratory for the U.S. Nuclear Regulatory Commission addresses ALARA incentives. "The relative importance of various worker and manager ALARA incentives as judged by the plant manager, the maintenance supervisor, and the radiation protection manager at each of ten nuclear sites provide the following:

Workers are motivated to reduce dose by:

- Being made aware of radiation risk,
- Receiving recognition,
- Acquiring prestige,
- Experiencing a sense of involvement,
- Receiving positive or negative feedback,
- Being aware of management concern, and
- Receiving awards.

Likewise, managers are motivated to reduce dose by humanitarian and monetary considerations associated with:

- Increasing the use of experienced plant and utility workers,
- Improving employee attitude and relations due to management's concern for worker safety,
- Receiving an annual salary adjustment for the successful accomplishment of department and station dose-reduction goals,
- Receiving monetary savings from critical path and labor savings resulting from ALARA preplanning efforts, and
- Feeling concern for the safety of fellow employees."

These are the ALARA incentives as seen through the eyes of those in various levels of management within a sampling of utilities. ALARA planning, especially for major maintenance outages, must involve those who perform those services and functions usually associated with high-dose jobs. Table 1 provides person-rem averages for several repetitive high-dose jobs at both BWRs and PWRs.

TABLE 1

COLLECTIVE DOSE AVERAGES FOR OUTAGE HIGH-DOSE JOBS
FOR BOTH PRESSURIZED AND BOILING WATER REACTORS

Job Title	Average Collective Dose (person-rem) PWR	BWR
Snubber, Hanger, and Anchor Bolt Inspection and Repair	59	290
In-Service Inspection	34	150
Insulation Removal/Replacement	11	44
Torus Repair, Inspection, and Modification	N/A	280
Turbine Overhaul and Repair	N/A	6

How should you, as members of the American Welding Society, become involved in ALARA preplanning activities? First, arm yourselves with the following types of questions:

- Who is the ALARA Coordinator for this job?
- Where is the ALARA office?
- How can we become involved in job-specific preplanning?
- How can we become involved in ALARA post-job reviews?
- How do we input ALARA suggestions?
- What is being done for contamination control?

The most significant of the lessons-learned from the Plant Hatch outage ALARA planning experience included efforts to:

- Establish specific Key Performance Indicators that can be tracked, evaluated and routinely reported to both project and executive management throughout the outage.
- Ensure both project and executive management have ample opportunity to help choose the individual Key Performance Indicators, and that all involved understand what each will mean.
- Require key management personnel from involved functional areas to meet periodically so that each may be brought up-to-date on how well ALARA goals and objectives are being accomplished, so that they become actively involved in helping resolve problems so that lines of communication do not atrophy through lack of use.
- Establish a specific goal for each doserate reduction action so the "Is this enough?" question is automatically answered.

- Minimize the administrative workload on the "planners" and "motivators" in each functional area by pooling clerical support.
- Require the ALARA organization to have available for scrutiny the considered "pluses" and "minuses" of its recommended actions.

What are the major motivators and incentives for you, the members of the American Welding Society, to become involved in ALARA planning to reduce occupational doses? There should be three primary incentives namely to:

- Reduce total TIME spent in radiation areas.
- Reduce dose to your membership.
- Reduce COST (to your company and to the utility).

The price to be paid will require more effort expended in three activities:

- Preplanning,
- Coordinating, and
- Communicating.

The payback will be saved time, saved money and saved exposure.

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Review of Chemical Decontamination Processes

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SUMMARY

Reducing the radiation exposure of nuclear plant workers is a goal of the U.S. Electric power industry. Because radiation fields outside the reactor core are the source of most of the exposure, removal of the radioactive corrosion products that cause these fields is one of the most cost-effective ways of controlling radiation doses. Chemical decontamination is particularly effective for reducing radiation exposure during special maintenance work, and the technology has become widely used for BWR recirculation-system piping repair and PWR steam generator maintenance.

This overview outlines various technical considerations in the choice of reagent, including decontamination effectiveness, waste handling, corrosion, and recontamination.

1. INTRODUCTION

The electric power industry has directed considerable resources at the problem of minimizing radiation exposure during special maintenance work. Reducing out-of-core radiation fields by decontamination represents one of the most cost-effective methods for achieving this, and it is not surprising that the use of this technology on commercial plants has grown rapidly, particularly the application of dilute chemical decontamination processes. The first use of chemicals on the primary coolant circuit of a utility-owned plant occurred as recently as 1982, but already a dozen similar applications have taken place in the past 12 months. There are now at least six commercial organizations offering chemical decontamination technology to the utility industry.

The majority of primary coolant circuit decontaminations have been carried out on the recirculation piping systems of BWRs, but several applications to the steam generator channel heads of PWRs have also been made. In the future it is anticipated that complete BWR reactor coolant systems and entire PWR steam generators will be decontaminated regularly.

Recently, the NRC has changed its position with regard to utility notification to NRC of a planned decontamination. In most instances NRC now only <u>requests</u> notification to them so that they may be aware in case questions come to their office regarding such decontaminations. The NRC position continues to be one of encouraging decontamination to reduce exposures.(1)

2. RADIOACTIVE CONTAMINATION

The removal of radioactive contamination from the surfaces of LWR systems by chemical decontamination requires an understanding of the manner in which these radioactive contaminates are deposited on the surfaces. The source of BWR corrosion products, as shown in Figure 1, is the feedwater heaters and drains. The corrosion products released from these surfaces are carried by the feedwater into the core region where the corrosion products are activated and then are available following their release to the coolant water for deposition on reactor coolant surfaces, particularly those of the recirculation piping.

The transport processes in PWRs are basically similar, but in this case the main source of corrosion products is the large surface area of Inconel-600 tubing in the steam generators (Figure 2).

At the fuel surface, particles, colloids and ions can deposit on the zircaloy oxide surfaces. Following the irradiation from the neutrons, materials are released, particularly by either ionic dissolution or particle erosion. The principal radioactive constituent in BWRs is cobalt-60. The steel surfaces which are in contact with the reactor coolant will then similarly pick up particles, colloids and ions. These will be incorporated in the growing corrosion film which exists on these surfaces. Corrosion products are released from the steel base metal, some of which are incorporated in the growing corrosion oxide film, others of which are released directly to the coolant itself.

When the fresh stainless steel surface is initially brought in contact with reactor coolant, particles from the coolant, principally Fe_2O_3, are deposited on the surface and will be incorporated in the growing film. During the early stages of corrosion, very little cobalt-60 is available for incorporation since there has been little opportunity for its generation on the fuel surfaces. As the growth of this film proceeds, cobalt-60 is now available and will be incorporated in the film. As further growth of the film takes place, separate and distinct layers form, each of which may contain cobalt-60. Consequently, the removal of significant amounts of cobalt-60 from steel surfaces requires the removal of the inner layer of corrosion film, primarily composed of magnetite. Chemicals selected for chemical decontamination must be able to penetrate through the outer hematite layer and perform their dissolution on the inner magnetite layer.

The differences between BWR and PWR decontamination are worth noting, because radioactive films in PWR plants are more difficult to dissolve than those in BWRs. In both cases spinel structures (iron-chrome-nickel oxides) are predominant, but a significant difference is the chromium content. PWR primary circuit chemistry is reducing as a result of dissolved hydrogen concentrations which are high enough to consume radiolytically generated oxygen. Steam generator corrosion films are primarily iron-nickel chromites, with up to 30% chromium. PWR fuel crud is mainly nickel ferrite. BWR chemistry is oxidizing and films are spinel structures with much less chromium, since the oxidizing coolant conditions cause it to be present as soluble chromate.

The higher chromium content of the PWR oxides makes these films more intractable to the dilute organic acid mixtures used in most decontamination processes. It is necessary to modify processes effective in BWRs for PWR use by adding an oxidizing pretreatment that dissolves the chromium predominant in the corrosion film. This makes the remaining material, which includes most of the radioactivity, more readily soluble in the dilute acid. The chromium content in BWRs is variable, and it has proved necessary on a few occasions to use a chromium-removal step along the lines noted above for PWR applications. These oxidizing processes need to be evaluated carefully as their application has been shown to require as much attention as the subsequent decontamination step. For instance, most oxidizing steps involve permanganate, which can be reduced to solid manganese dioxide if pH and exposure to air are not controlled.

3. DECONTAMINATION SOLVENTS AND PROCESSES

Two types of dilute decontamination systems are available today. One type consists of organic acids (such as citric and oxalic acids used in the PNS CITROX process)and chelating agents (such as EDTA and NTA) which are mildly reducing in nature. These reagents dissolve oxides by simple acidic dissolution, e.g.,

$$\underset{\text{acid}}{8H^+} + \underset{\text{oxide}}{Fe_3O_4} \rightarrow \underbrace{Fe^{2+} + 2Fe^{3+}}_{\text{metal ions in solution}} + 4H_2O$$

and reductive dissolution:

$$\underset{\text{acid}}{8H^+} + \underset{\text{reducing agent}}{2e^-} + \underset{\text{oxide}}{Fe_3O_4} \rightarrow \underset{\text{metal ions in solutions}}{3Fe^{2+}} + 4H_2O$$

The radioactive impurities, such as Co-60, Co-58, Fe-59, Mn-54, are released at the same time.

In order to prevent redeposition of the metal ions, a chelating agent is used to increase the solubility of the metal ions in solution by forming a chelating complex. The London Nuclear CAN-DECON, C-E/KWU, IT Corp. NS-1, and Westinghouse DCD processes fall in this category. A corrosion inhibitor may be necessary for some applications, depending on the materials in the system, the process temperature duration, and reagent strength.

The second type of decontamination reagents are based on low oxidation state metal ions (LOMI) which are more strongly reducing and do not require a corrosion inhibitor. The most widely used LOMI reagent is vanadous picolinate/formate in which vanadous ions (V^{2+}) reduce the oxidation state of the iron in the oxide, enabling it to be rapidly dissolved in picolinic acid, which is also a mild complexing agent. Bechtel National, IT Corp., London Nuclear and Pacific Nuclear have licenses for the LOMI process from EPRI.

As mentioned above, both types may require an oxidizing pretreatment for situations where high chromium oxides are present. Potassium permanaganate is often used for this purpose, dissolving chromium oxides by oxidizing them to soluble chromate, e.g.,

$$\underset{\text{chromium oxide}}{Cr_2O_3} + \underset{\text{oxidizing agent}}{30x^+} \longrightarrow \underset{\text{soluble chromate}}{Cr^{6+}} + 3e^- + \text{residuals}$$

The chemical reagents can be added in the solid form (as with CAN-DECON) or as liquids (for instance LOMI is a mixture of dilute solutions of vanadous formate and picolinic acid). Depending on the type of circuit to be decontaminated, feed-and-bleed or fill-and-drain methods may be used. In most cases, the radioactive waste is removed on ion-exchange resin.

There are three basic methods of processing decontamination solutions: evaporation, precipitation, and ion-exchange. The first of these is the least satisfactory, in general, requiring large plant, a long time and a high energy cost. Precipitation is the most attractive options for concentrated reagents, though it can also be applied to dilute processes. Ion exchange is the method generally used for current commercial processes.

4. DECONTAMINATION EXPERIENCE

A list of recent decontaminations by U.S. utilities is given in Table 1 for BWRs and Table 2 for PWRs. Although complete circuit cleaning including the fuel has been carried out routinely in Canada and Britain, U.S. experience has been mainly with part-circuit decontaminations. Three systems have been decontaminated

(a) steam generator channel heads

(b) reactor water cleanup systems

(c) recirculation piping systems.

Typically nozzle dams are used to isolate steam generators for decontamination. The Millstone-2 and Ginna experiences are described elsewhere.[2][3] The flow path used for a typical channel head decontamination, in this case with the CAN-DECON process, is shown in Figure 3.

BWR recirculation piping has been decontaminated using several different methods of circulation. The most effective of circulating the decontamination solvent is to use the recirculation pumps, with the reactor pressure vessel annulus used in the return path of the solvent, as shown in Figure 4. This method has been used at several plants.[4] Alternatively, the solvent can be kept below the vessel inlet and circulation achieved by oscillating the solvent level. This does not provide the same degree of circulation but nevertheless, excellent decontamination factors can be achieved by this method, as shown in figure 5 for the Monticello decontamination with LOMI.[5] Another approach is to cut the piping, cap the ends and then circulate the solvent through connections in the caps or the piping close to the ends. This method was used at Peach Bottom and Cooper plants. BWR decontaminations have been described at a recent conference.[6]

A recent survey[7] of 9 plants that have decontaminated reactor coolant systems has highlighted both the good and bad features of the existing technology. On the positive side, the estimated dose savings is 2169 man rems per plant. Estimating the average cost of the decontamination and waste handling to be $1 million, then the cost per man-rem saved is $460, which is highly cost effective. On the other hand, each of the nine plants surveyed reported operational problems associated with the decontamination. For instance, 7 of the 9 plants experienced actual or potential leaks due to hose, gasket, seals or valve problems, 7 of the 9 had failures with the vendor skids, 3 had process or chemistry problems and 3 had clean-up or waste handling difficulties. These problems have not usually been serious and have generally been solved on a plant by plant basis. However, they do indicate that improved quality control of equipment and connections, with increased pre-operational testing, can significantly improve the decontamination operation.

5. PROCESS SELECTION

It is not appropriate to give here a detailed evaluation of all the processes competing in the marketplace today, but it is possible to highlight a few key issues to be considered in the selection process and in the planning for the decontamination.

In-line outage time is of prime economic importance in choosing the process. If the process works as claimed by the vendor, the main parameters are time to set up the system, the duration of the process (are multiple applications required?), and the time taken to clean up. For BWRs, application times are generally shorter if the pressure vessel annulus can be included in the system being decontaminated, thereby providing a good circulation path. Considerably longer times are required if cutting and capping of piping is necessary. Process temperature has some impact, since the use of temperatures above $100^{0}C$ require pressurization, which may necessitate replacement of the pressure vessel head. The above variables may be significant but far more important are the delays that can result if something goes wrong. Serious delays have occurred due to equipment malfunctions, with leakage of radioactive solutions being one of the most worrisome aspects. Long-term field testing of the decontamination process equipment is one of the best ways of demonstrating system integrity and engineering quality.

Effectiveness of the chemical reagents does not appear to be a major issue. All the processes currently offered are likely to give adequate decontamination factors,* although some solvents have greater margin in hand for coping with the more intractable oxides. Data from some recent plant decontaminations are presented in Figure 6.

Radioactive waste handling is an important consideration, and requirements differ for the various decontamination processes available. Although waste removal and transportation off site have not proved difficult to date, packaging and disposal procedures may have to be re-evaluated and altered to meet revised 10CFR61 requirements.(8) Several methods are currently used, including blending of ion-exchange resins into the radwaste tanks, transportation in a dewatered condition in high-integrity containers, and solidification.

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*Decontamination factor = $\frac{\text{Radiation field before decon}}{\text{Radiation field after decon}}$

Corrosion and stress corrosion cracking are important issues that must be addressed early in the planning process for decontamination. The chemical reagents currently used on commercial plants are relatively nonaggressive to typical plant materials, and general corrosion and pitting attack are not perceived to be a significant problem. However, the effects of candidate processes on specific materials, particularly any that show enhanced susceptibility to stress corrosion cracking, must be evaluated on a plant-specific basis. BWR corrosion issues have been discussed by Kass[9] and PWR issues by Wolfe[10] at a recent conference.

Finally, the issue of recontamination must be addressed. Recontamination occurs when the freshly decontaminated surface with its corrosion film removed reestablishes its protective corrosion film. Initially such corrosion is rapid. When part circuit decontamination has taken place, reactor coolant will contain high concentrations of radioactive species compared to when corrosion films were initially developed on these surfaces. Consequently the new, rapidly developing corrosion film will incorporate significant concentrations of radioisotopes causing a rapidly increasing radiation field in the vicinity.

Data is available to show that steam generator channel heads that have been decontaminated using a dry grit-blasting process show increased radiation field buildup after return to power, such that fields become higher than on nondecontaminated channel heads. Similar effects have been observed in BWRs; after one fuel cycle of operation both decontaminated pipes and replacement pipes in recirculation systems show radiation fields little different from those observed in equivalent parts of the plant before the decontamination or replacement work. Clearly conditioning of the decontaminated or new piping giving a coherent oxide film is beneficial, as subsequent contamination can be significantly reduced.

In the longer term, the recontamination problem may be almost completely eliminated by decontaminating the entire circuit, including the fuel, in order to remove the source of the subsequent recontamination.

EPRI is sponsoring a field test of fuel decontamination, in collaboration with Commonwealth Edison, to evaluate the effects of three commercially-available decontamination solvents on fuel and other in-core materials. This project will start in June 1985.

6. CONCLUSIONS

Practical and proven decontamination technology has become available for utility application at a time when there is a growing need to reduce radiation doses resulting from major maintenance and repair requirements. This paper has reviewed some of the technical considerations and highlighted key issues to be evaluated by utilities planning decontamination activities.

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TABLE 1

CHEMICAL DECONTAMINATION CARRIED OUT AT U.S. BWR PLANTS IN PAST 2 YEARS

DATE	PLANT	PLANT SYSTEM	CONTRACTOR/PROCESS
1983	Vermont Yankee	Recirc System	LN/CAN-DECON
	Peach Bottom 3	RWCU	LN/CAN-DECON
	Quad Cities 2	Recirc System	LN-CAN-DECON
	Dresden 3	Recirc System	LN-CAN-DECON
1984	Pilgrim	Recirc System	IT/Dilute NS-1
	Quad Cities 1	Recirc System	LN/CAN-DECON
	Monticello	Recirc System, RWCU	Q/LOMI
	Brunswick	RWCU	PN/AP CITROX
	Millstone 1	Recirc System	LN/CAN-DECON
	Pilgrim	RWCU	IT, PN/Dilute NS-1
	Peach Bottom 2	Recirc System	LN/CAN-DECON
	Dresden 1	Reactor Coolant System	IT/NS-1
	Cooper	Recirc System	PN/AP CITROX
	Dresden 2	Recirc System	IT, PN/Dilute NS-1

Notes: This listing is not necessarily complete

LN = London Nuclear

Q = Quadrex HPS

IT = IT Nuclear Services

PN = Pacific Nuclear Systems & Services

TABLE 2

CHEMICAL DECONTAMINATION CARRIED OUT AT U.S. PWR PLANTS IN PAST 3 YEARS

DATE	PLANT	PLANT SYSTEM	CONTRACTOR/PROCESS
1982	Ex-Surry SG	Channel Head Cold Leg	LN/CAN-DECON
	Ex-Surry SG	Channel Head Hot Leg	Q/LOMI
1983	Millstone 2 PWR	Channel Heads	CE/OZOX-A
	Ginna PWR	Channel Heads	LN/CAN-DECON
1984	Palisades	Channel Heads	LN/CAN-DECON

Notes: This listing is not necessarily complete

LN = London Nuclear

Q = Quadrex HPS

CE = Combustion Engineering

Figure 1

BWR CONTAMINATION RESULTS FROM ACTIVATION OF
CORROSION PRODUCTS IN THE FEEDWATER
COBALT-60 IS THE PRIMARY RADIOACTIVE ISOTOPE

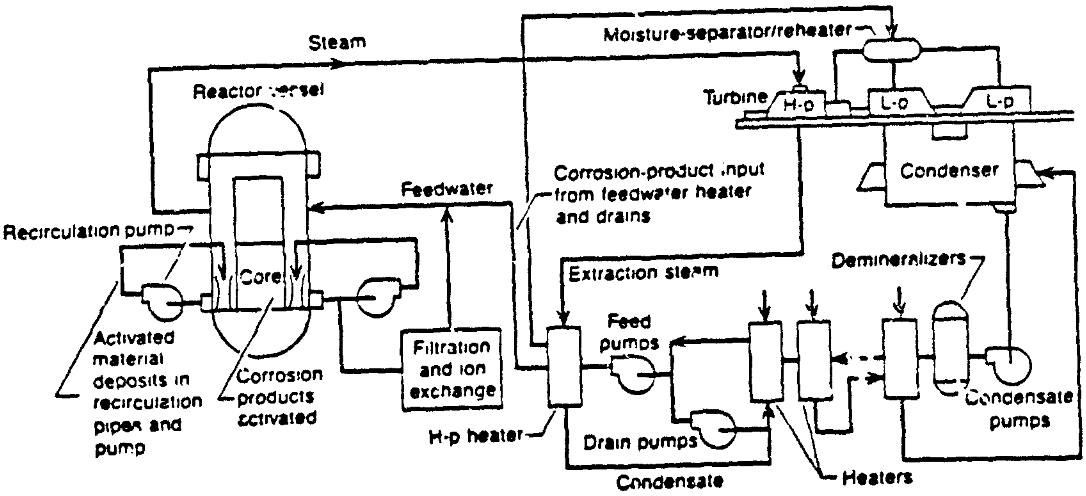
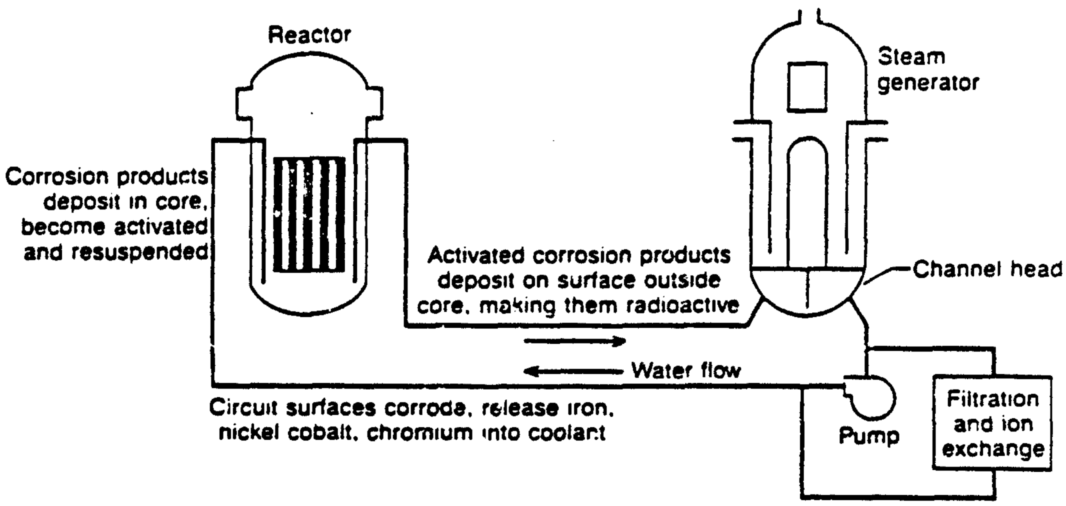


Figure 2

CORROSION PRODUCTS IN PWR PRIMARY CIRCUIT ARE ACTIVATED IN CORE & DEPOSITED ON OUT-OF-CORE SURFACES PRODUCING CONTAMINATED PUMPS, PIPING, & STEAM GENERATORS



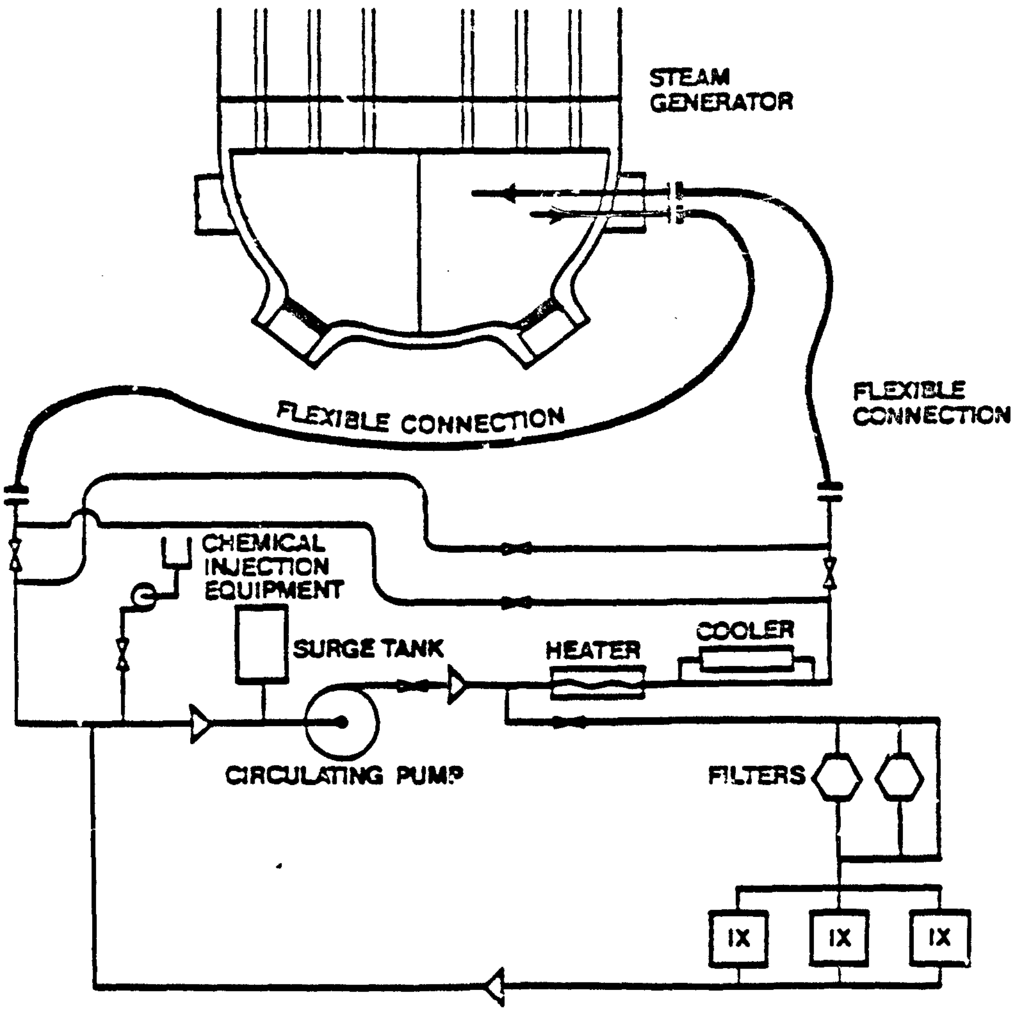


Figure 3. Flow path for a steam generator channel head decontamination using the CAN-DECON process.

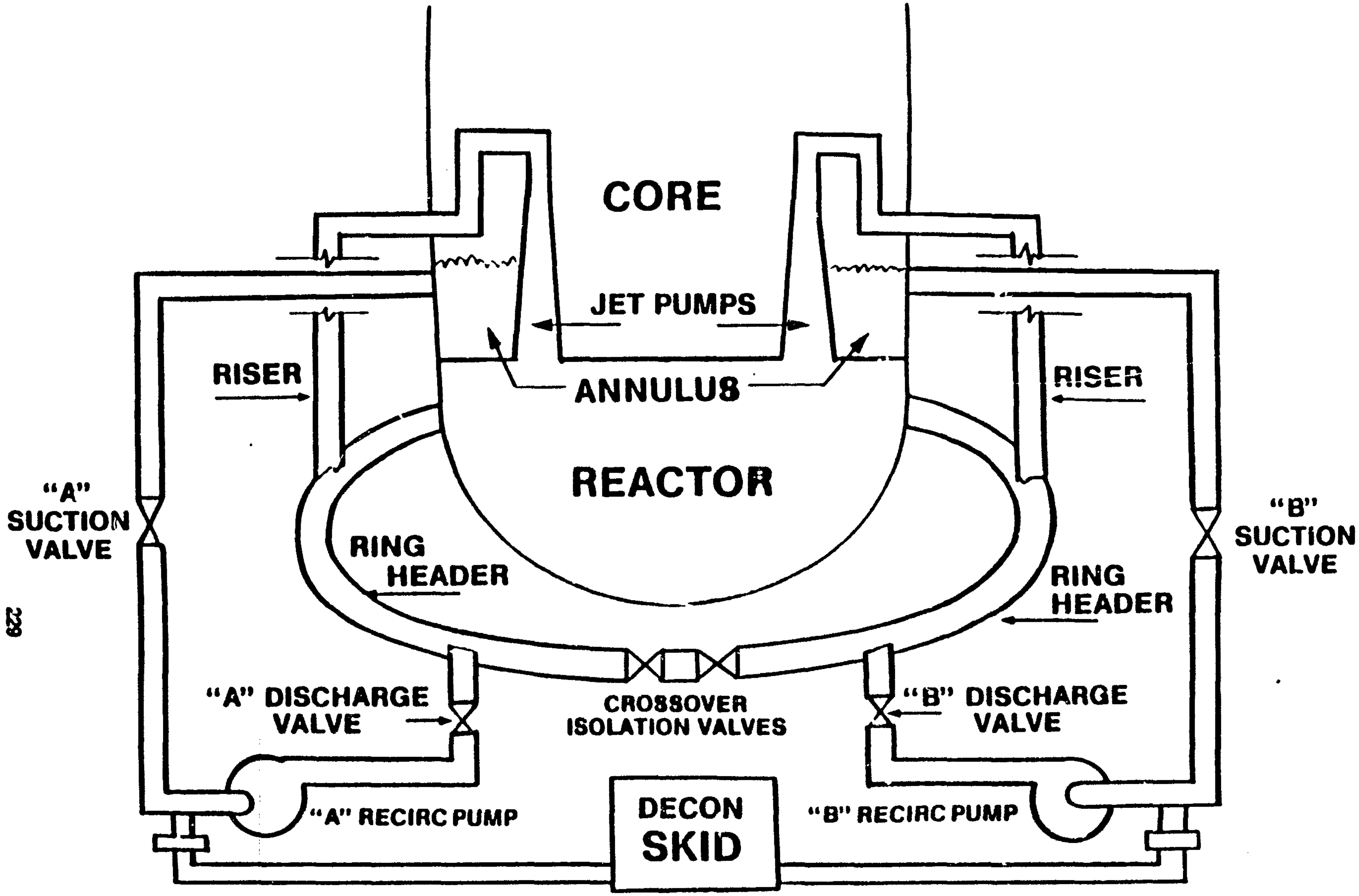
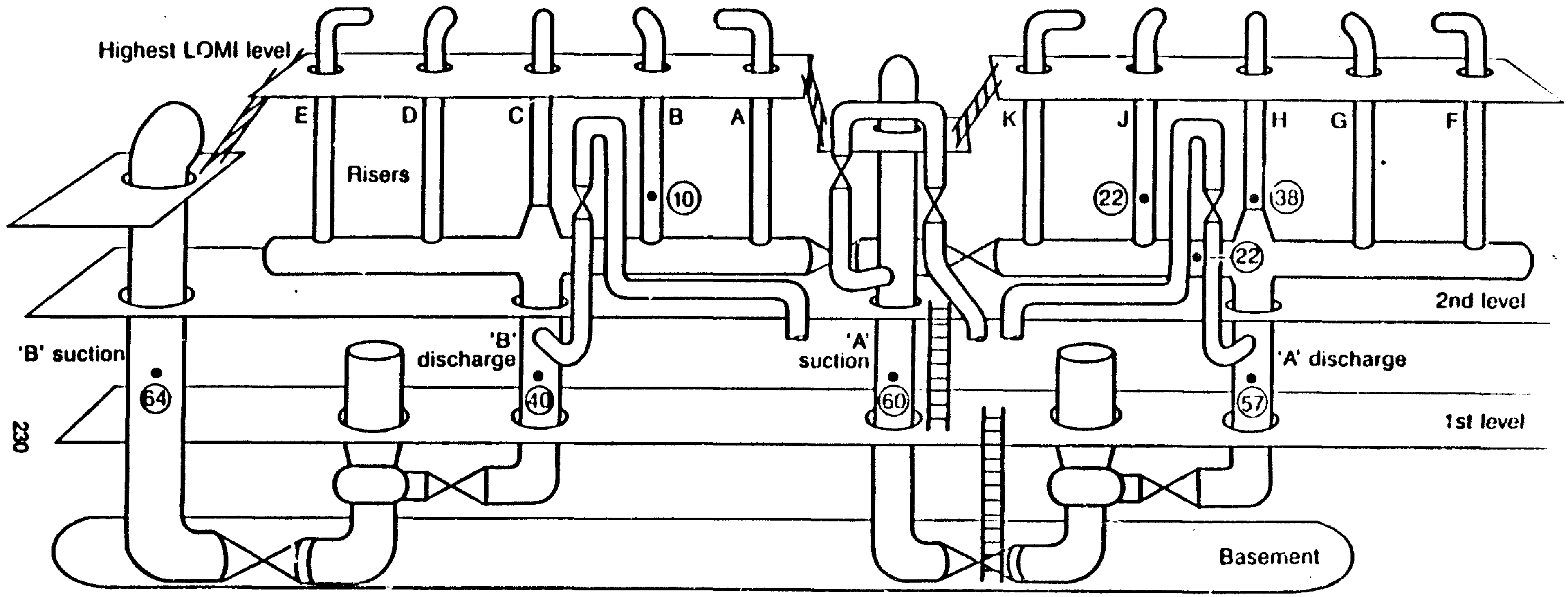


Figure 4. Flow path for a BWR recirculation piping decontamination using the reactor pressure annulus.

FIGURE 5. **MONTICELLO PIPING SYSTEM SHOWING LOMI DECONTAMINATION FACTORS**



● Survey points with decontamination factors indicated. (DF)

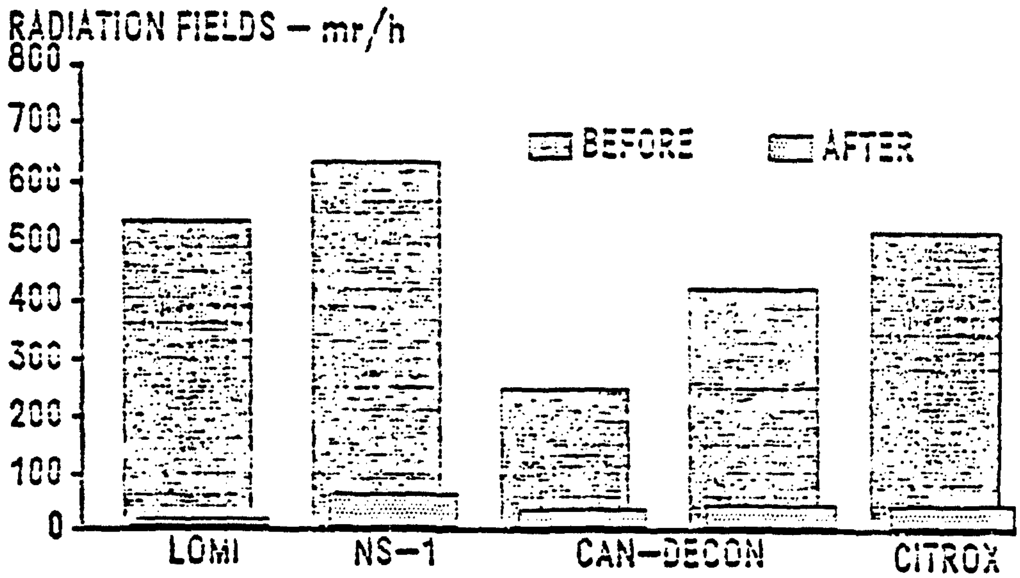


Figure 6. Radiation fields before and after chemical decontaminations of BWR recirculation piping (1984 data).