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Heavy Component Replacement in Nuclear Power Plants: Experience and Guidelines

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HEAVY COMPONENT REPLACEMENT IN NUCLEAR POWER PLANTS: EXPERIENCE AND GUIDELINES

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FOREWORD

The world's fleet of nuclear power plants is, on average, more than 20 years old. Even though the design life of an nuclear power plant is typically 30–40 years, it is quite feasible that many nuclear power plants will be able to operate in excess of their design lives, provided that they operate safely.

Different material degradation mechanisms have been identified on components resulting from the ageing phenomenon. Time dependent changes in the mechanical and physical properties of system, structure and components (SSCs) are referred to as ageing. The effects of ageing lead to changes, with time, in the SSC materials, which are caused and driven by the effects of erosion, corrosion, varying loads, flow conditions, temperature and neutron irradiation.

Component replacement is often the most feasible solution to solve the effects of ageing associated with primary water stress corrosion cracking of alloy 600. Even if mitigation and/or repair were a local solution, replacement offers many advantages when addressing the assortment of potentially susceptible parts contained in a major component.

This publication is dedicated on heavy components replacement considered strategic for nuclear power plants life management, but not included in current maintenance activities carried out by utilities. The major and heavy components to be considered are:

- Steam generators of PWRs;
- Reactor vessel head of PWRs;
- Reactor vessel internals of PWRs;
- Pressurizer of PWRs;
- Reactor internal components of BWRs;
- Reactor coolant piping/recirculation piping of PWRs and BWRs;
- Steam generators and fuel channel and feeder pipes of PHWRs.

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CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Nuclear power plants worldwide are showing continuous improvements in their performance and availability. This applies to overall plant safety as well. However, problems associated with materials have been observed. These material problems may, however, not be of a purely technical nature but may also be due to factors present in the management system of the nuclear power plant concerned.

The replacement of heavy components is the result of widespread stress corrosion of alloy 600 in the primary system. Component replacement is often the feasible solution to the problems associated with primary water stress corrosion cracking (PWSCC) of alloy 600. Following the corrosion of steam generator (SG) tubes, which led to the first steam generator replacement (SGR) projects, work has begun on reactor vessel head replacements (RVHR) and pressurizer replacements.

Even if mitigation and/or repair were a local solution, replacement offers many advantages when addressing the assortment of potentially susceptible parts contained in a major component. Utilities are looking for ways to extend plant lifetime and must, therefore, prevent stress corrosion in primary components while combating other phenomena, such as thermal fatigue or certain metallurgical weaknesses.

Since the early 2000s, the driving factors behind main component replacements have become more complex and interconnected. The increase in the number of replacements of heavy components in the reactor building on specific reactor geometries has called for major technical innovations in the replacement of heavy components. Utilities have developed economic models that help them make their decisions on main components replacements and on fixing the optimum dates. The leading factors in the economic models are:

- Inspection costs (specifically those issued related to development and implementation costs for improved techniques to detect PWSCC);
- Statistical repair costs (provisional technical models are based on the material sensitivity at each location on the primary loop);
- Extra outage time and consequent loss of capacity due to extended inspection and repair;
- Replacement component supply and transportation cost;
- Replacement component installation cost, including consequent outage duration extension, if any;
- Loss of capacity due to extended period of shutdown needed for replacement;
- Benefits from improved replacement component, not only in terms of in-service inspection (ISI), outage maintenance and refuelling, including the reduction of PWSCC inspection and repair constraints, but also regarding design improvement to facilitate asset management;
- Benefits from plant life extension;
- Benefits from power uprate through increasing SG tube heat transfer surface.

These models also calculate the economics in terms of investment returns, on-going maintenance and variable costs [1, 2].

1.2. OBJECTIVES AND SCOPE

This report focuses on heavy component replacement with respect to plant life management and does not include the current normal maintenance practice carried out by utilities. The major and heavy components to be considered are:

- SGs of PWRs;
- Reactor vessel head of PWRs;
- Reactor vessel internals (RVI) of PWRs;
- Pressurizer of PWRs;

- Reactor internal components of BWRs;
- Reactor coolant piping/recirculation piping of PWRs and BWRs;
- SGs and fuel channel and feeder pipes of PHWRs.

1.3. USERS

This publication is intended for use by the senior managers and engineers of organizations involved in the replacement of heavy components including:

- Utilities;
- Project management team organizations;
- Supplier organizations for replacement and commissioning services;
- Technical support organizations;
- Vendors and equipment suppliers.

The publication also includes information which may be useful to decision makers and advisors for the replacement of heavy components.

1.4. STRUCTURE

Section 1 provides the background and Section 2 identifies the different material degradation mechanisms of heavy components resulting from the ageing phenomenon not identified at the origin of the components' construction. These material degradation mechanisms are specific for each component resulting in corrosion due to environmental aspects and the consequences of the ageing mechanism. Section 3 describes the organizational requirements necessary for the large and complex projects, such as heavy component replacements. The large and complex projects require strong project organization coupled with highly skilled personnel to resolve difficult technical issues and involve the organization of hundreds of permanent and temporary workers.

The replacement approaches, strategies, test programme and sequences for each heavy component are described separately in Sections 4–10 for SGs, reactor vessel head, reactor internal components, RVI, pressurizer, reactor coolant piping/recirculation piping, and fuel channel and feeder pipes of PHWRs. In Section 11, decommissioning approaches for free release and recycling of materials are introduced as the cost effective alternative of applying direct treatment of retired components for material recycling after free release. In the Appendices, RVHR history, SGR history, and safety, licensing and regulatory issues in Japan are introduced to give more detailed technical information.

2. MATERIAL DEGRADATION MECHANISM OF HEAVY COMPONENTS

Different material degradation mechanisms have been identified on components resulting from the ageing phenomenon not identified at the origin of the components' construction. These material degradation mechanisms are specific for each component exhibiting corrosion due to environmental aspects and are the consequences of the ageing mechanism.

2.1. PWRs

2.1.1. SG

In the original construction of PWRs, the tubes of SGs were manufactured from alloy 600. During the 1970s, in several countries, degradation of alloy 600 material tubes due to PWSCC (and alloy 82/182 for weld) in the environment of the primary system was identified. The consequence of this degradation was through wall cracking of the tubes leading to primary system to secondary system leakage.

The first strategy engaged by several utilities was to plug or sleeve the degraded tubes. However, following several cycles of operation using this strategy, the thermal power of the SGs was reduced, outage durations and maintenance cost continued to escalate. The studies to maintain the SGs in operation showed that it would be more economical to replace the SGs than continue a repair and maintenance programme.

The first SGRs were carried out in the United States of America (Surry 2, 1979; Surry 1, 1981), Germany (Obrigheim, 1983), Sweden (Ringhals 2, 1989) and France (Dampierre 1, 1990). Other SGRs were carried out in Japan (Takahama 2, 1994), Belgium (Doel 3, 1993) and Switzerland (Beznau 1, 1993).

At the same period in several countries (USA, France, Japan, etc.) some large programmes of research and development were performed and several studies were initiated to understand the degradation mechanism on alloy 600 and which alternative material could be proposed for SG tubes. Alloy 690 or alloy 800 as tube materials and alloy 52/152 for welds were proposed as alternative solutions because the behaviour of these materials under PWSCC was better than alloy 600 [3].

2.1.2. Reactor vessel head

In 1991, during the first 10-year outage inspection of the Bugey 3 in France, a leak on T54 penetration was discovered during the hydraulic testing of the reactor coolant primary circuit (Fig.1).

The leak was revealed during a hydro pressure test of the primary circuit (pressure: 207 bar) with parallel acoustic emission testing (estimated leak up to 1 L/h). The root cause identified after analysis from Bugey 3 T54 vessel head penetration investigated in a hot cells laboratory was PWSCC from inside to outside the penetration. The penetration material was alloy 600 and the weld was alloy 82/182. The crack was located in the vicinity of the welded joint between the bushing and head. The crack was due to the effect of the stress levels and

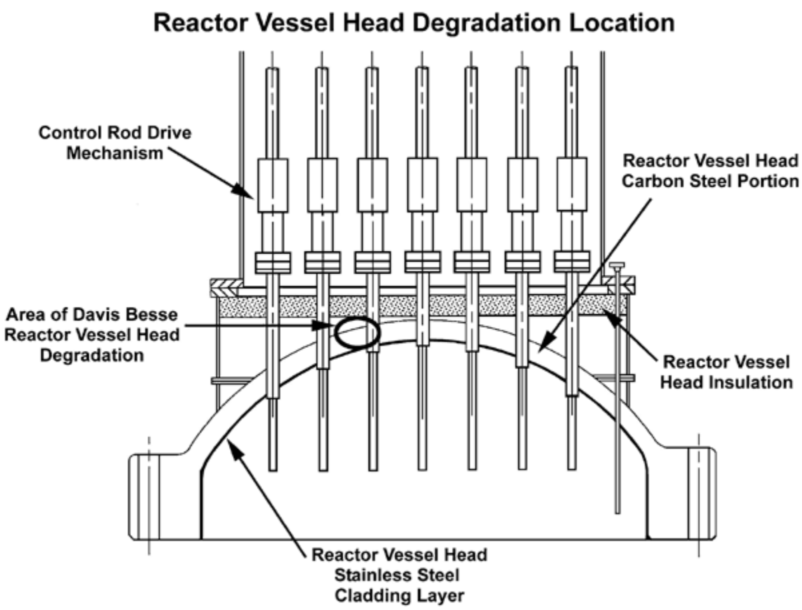


FIG. 1. PWR vessel head penetration cracking of alloy 600 allowed leakage of the borated coolant to occur.

was longitudinal, having a preferential azimuthally location between 0° and 180°, but no fatigue damage was identified.

Industry experience and comparison with SG alloy 600 tube degradation permitted identification as stress corrosion of alloy 600. Several plants have been affected by the stress corrosion of alloy 600. Between 1992 and 1994, a programme of non-destructive examination (NDE) showed that this degradation was generic in several plants (Fig. 2).

Repair solutions and appropriate NDE techniques were developed, but a strategic decision, with regard to plant life management, was taken in 1994 in France to replace all 900 and 1300 MW vessel heads, 34 vessel heads for 3-loop plants and 20 vessel heads for 4-loop plants.

In Japan, similar action was taken and a programme to replace the vessel heads of PWRs was initiated in 1996. Recently, in the USA, a significant degradation of the vessel head was discovered at the Davis Besse plant with PWSCC of alloy 600 material degradation identified as the root cause, similar to the root cause for Bugey. Davis-Besse had accumulated 15.8 effective full power years of operation when the plant shut down for its thirteenth refuelling outage on 16 February 2002. During that refuelling outage, while performing RPV closure head inspections required by the United States Nuclear Regulatory Commission, workers discovered a large cavity in the 15.24 cm thick low alloy carbon steel RPV head material. The cavity was about 16.76 cm long and 10.16–12.70 cm at the widest point extending down to the 0.635 cm thick type 308 stainless steel cladding.

2.1.3. RVI

The toughness characteristic of RVI has been reduced due to irradiation embrittlement and PWSCC. Initially, the material degradation identified was focused on the baffle bolts, and in some countries a programme of baffle bolt replacement was executed. In parallel, some utilities chose to replace the RVI as a preventive action to irraditiated assisted stress corrosion cracking. PWR RVI in Japan were replaced first. Another driving force for the replacement of RVI in Japan was that it enabled the use of advanced fuel design.

More effective maintenance and replacement of RVI has been developed and already applied to PWR plants in Japan since 2005. RVI replacement can solve several degradations at the same time, making substantial improvements against ageing degradations compared with individual maintenance approaches, because new RVI for the replacement can incorporate more effective or drastic countermeasures and be expected to reduce risks on reliability and safety.

Evaluation of ageing mechanisms is based on PWR service experience, pertinent laboratory data and relevant experience from other industries. The following ageing mechanisms were reviewed and assessed for relevance to RVI [4]:

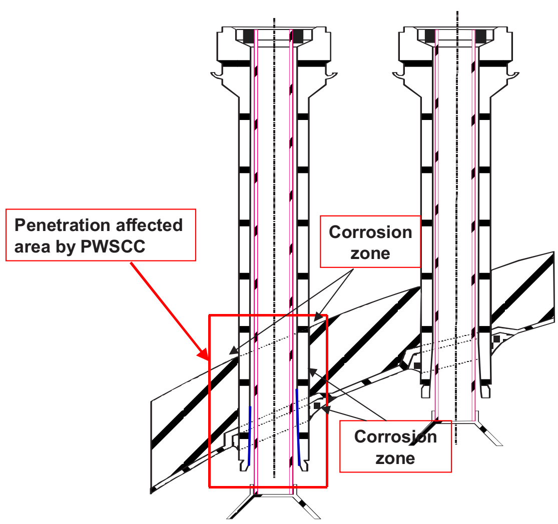


FIG. 2. Reactor vessel head penetration – PWSCC corrosion area.

- Embrittlement;
- Fatigue;
- Corrosion;
- Radiation induced creep, relaxation and swelling;
- Mechanical wear.

Significance of embrittlement and irradiation assisted stress corrosion cracking has been re-evaluated as follows.

Embrittlement

There are two types of embrittlement which could affect PWR vessel internal components. These are irradiation embrittlement, which may affect core region internals, and thermal ageing embrittlement, which may affect the cast stainless steel parts and parts manufactured from martensitic stainless steel.

Fatigue

Fatigue is defined as the structural deterioration that occurs as a result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most strongly affected locations. Subsequent continued cyclic loading can lead to the growth of the initiated crack.

Corrosion

Corrosion is the reaction of a substance with its environment that causes a detectable change which can lead to deterioration in the function of the component or structure. In the present context, the material is steel and the reaction is usually an electrochemical one.

The appearance of corrosion is governed by the so-called corrosion system consisting of the metal and the corrosive medium (the environment) with all the participating elements that can influence the electrochemical behaviour and the corrosion parameters. The variety of possible chemical and physical variables leads to a large number of types of corrosion.

Radiation induced creep, relaxation and swelling

Neutron irradiation creates a large number of interstitials and vacancies that can annihilate on sinks such as dislocations, grain boundaries, surfaces, etc., by diffusion controlled processes. The kinetics of annihilation are different for interstitials and vacancies and depend on stress, temperature, material microstructure, etc. If interstitials are eliminated rapidly, the excess vacancies coalesce into voids or bubbles inside the metal, leading to swelling of the structure.

Mechanical wear

This degradation type is broadly characterized as a mechanically induced or aided degradation mechanism. Degradation from small amplitude, oscillatory motion, between continuously rubbing surfaces, is generally termed fretting. Vibration of relatively large amplitude, resulting in intermittent sliding contact between two parts, is termed sliding wear, or wear. Wear generally results from the concurrent effects of vibration and corrosion.

2.1.4. Pressurizer

In the pressurizer, several locations are constructed utilizing alloy 600 and welds in alloy 82/182, mainly on the nozzles and heater sleeve areas (Fig. 3). The most typical locations found in the pressurizer containing alloy

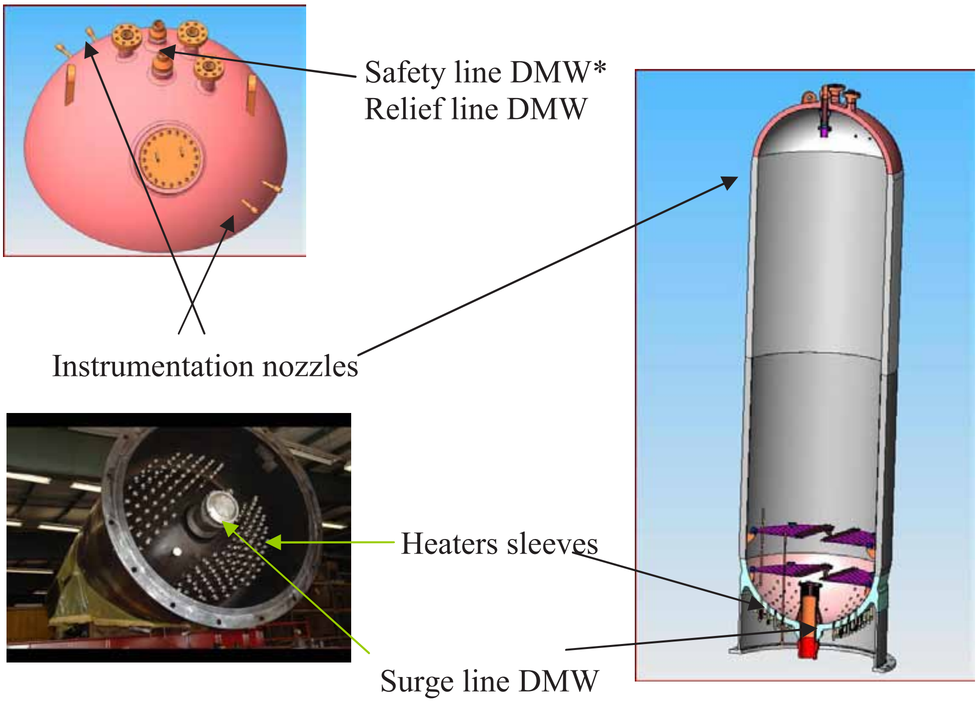


FIG. 3. Pressurizer nozzles and heater sleeves in alloy 600 and alloy 82/182 (DMW: dissimilar metal weld).

600 and welds in 82/182 have experienced flaw initiation by PWSCC. Although some indications have been found as circumferential in nature, most have been axial with no safety significance.

The first appearance of these PWSCC indications were found in small bore instrumentation nozzles of the water and steam space. Over time, these indications have been seen in virtually all alloy 600 locations on the pressurizer, with a large number of flaws identified in the j-welded heater sleeves on Combustion Engineering designed pressurizers. Most recently, flaws have been identified in the large bore dissimilar metal welds of the pressurizer surge line to nozzle connection.

Initially, pressurizer alloy 600 flaws were resolved by performing weld repair. Typically, these repairs have been effected by complete nozzle replacements as well as by half nozzle repair techniques for small bore nozzles including relocation or replacement of the j-weld pressure boundary weld. Additionally, many plants have applied mechanical nozzle seal assemblies to address both leakage and structural repair. Repair options do vary from country to country based upon regulation.

For most PWR pressurizers, proactive mitigation has become the standard approach to prevent unplanned flaw initiation including half nozzle repair for small bore and weld overlay or mechanical stress improvement for large bore dissimilar metal welds. However, for Combustion Engineering designed PWRs with large numbers of heater sleeves, most plants have selected pressurizer replacement based upon economic modelling.

2.1.5. Reactor coolant piping/recirculation piping

In 1998, a leak on the reactor heat removal circuit at Civaux 1 (1450 MW(e)) in France was found and the root cause was characterized as thermal fatigue resulting in leakage from a mixing zone area due to high differential temperature (high ΔT).

The utility initiated engineering studies and investigations to identify other areas on the reactor coolant line/loop (RCL) circuit that could be under the same conditions. An area on the cold leg at the chemical volume control system line connection (Fig. 4) was identified as a potential zone for thermal fatigue. In 2002, at Fessneheim 1 in France, an RCL cold leg part of pipe was replaced to remove the real potential for thermal fatigue of this zone [5].

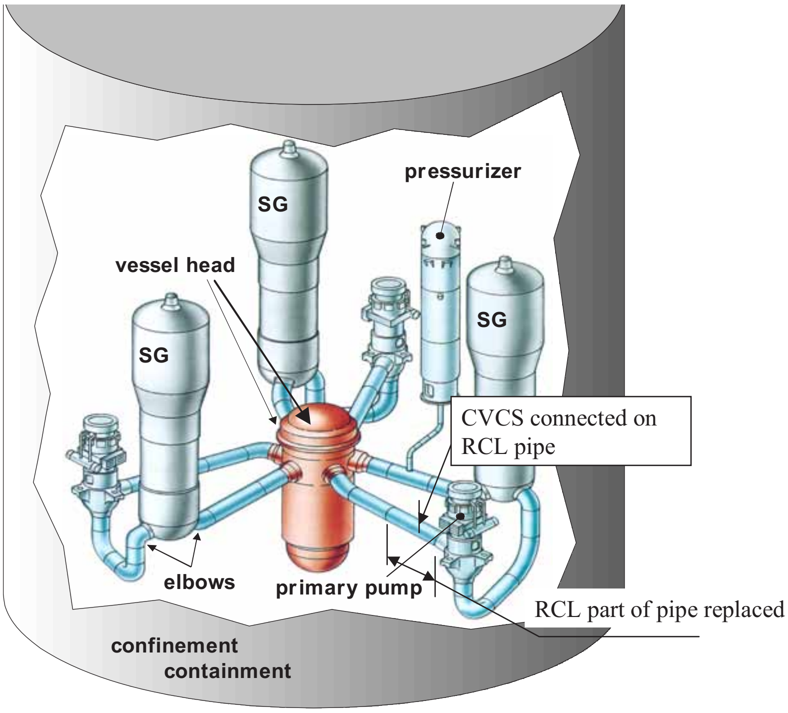


FIG. 4. Reactor coolant piping replacement.

2.2. BWRs

2.2.1. Reactor internal component

Stress corrosion cracking (SCC) occurred in BWR plants in core internals fabricated from type 304 SS near the heat affected zone of the welds which were sensitized. Utilities and suppliers have jointly developed type 316L SS which is less susceptible to sensitization by controlling the carbon content to less than 0.02%.

Recently, SCC was found in the core shroud support ring (H7a) made from type 316L SS. The investigation of boat samples taken from a cracked core shroud was performed as shown in Fig. 5. It showed that initiation of cracking occurred in a hardened layer of the surface exceeding HV300 (see Fig. 6) as transgranular stress corrosion cracking (TGSCC), with TGSCC subsequently changing into intergranular stress corrosion cracking (IGSCC). This hardened layer was formed on a machined surface such as a shroud ring during the manufacturing process, the thickness of this layer being about 100 μm.

Figure 7 shows the results of the cross-sectional observation of the shroud lower ring (H6a) in another plant. Scanning electron micrographs are shown in Fig. 6. TGSCC was observed in the surface layer, followed by IGSCC. It was observed in the laboratory that TGSCC susceptibility of cold worked type 316L appeared in BWR simulated water when hardness exceeded HV300 (see Figs 8 and 9)[6, 7].

2.3. PHWRs

2.3.1. Fuel channel

Pressure tube rolled joint residual stresses

In reactors constructed during the early 1970s, the end fittings to pressure tube rolled joints were made with fairly large diametric clearances (up to 0.5 mm) and were also rolled beyond the end fitting support. On Pickering 3 and 4, many of these joints developed cracks in the high residual stress zones of the pressure tubes.

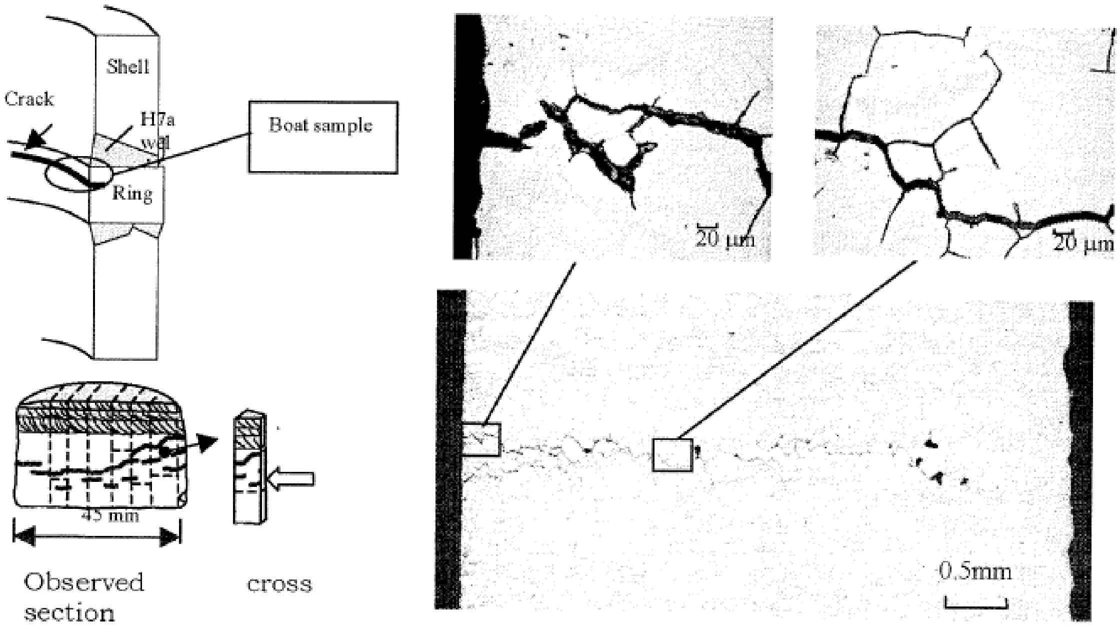


FIG. 5. Cross-sectional observation of SCC crack in shroud support ring.

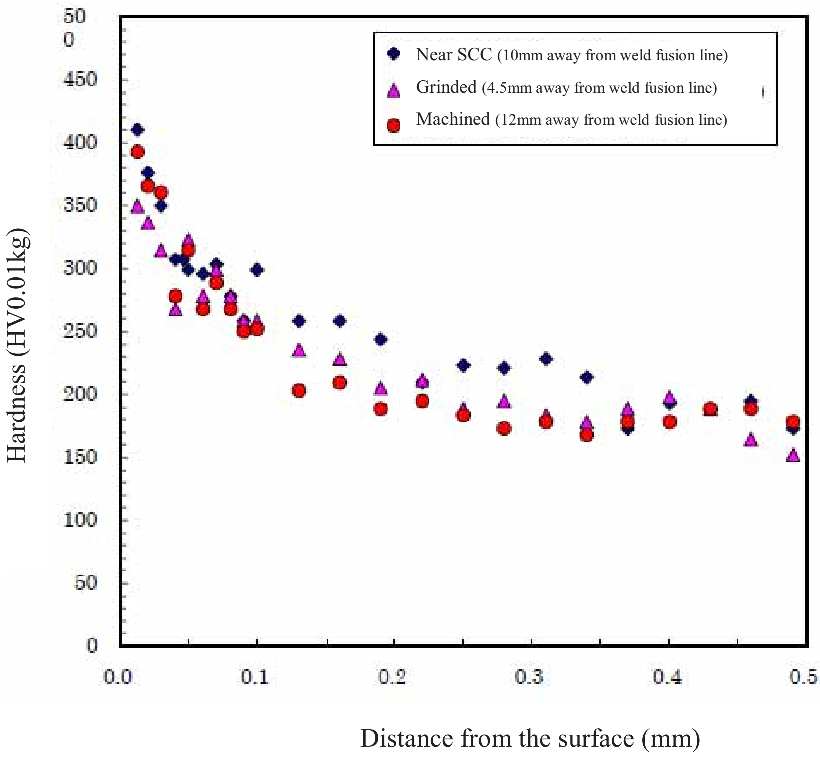


FIG. 6. Distribution of hardness in shroud support ring.

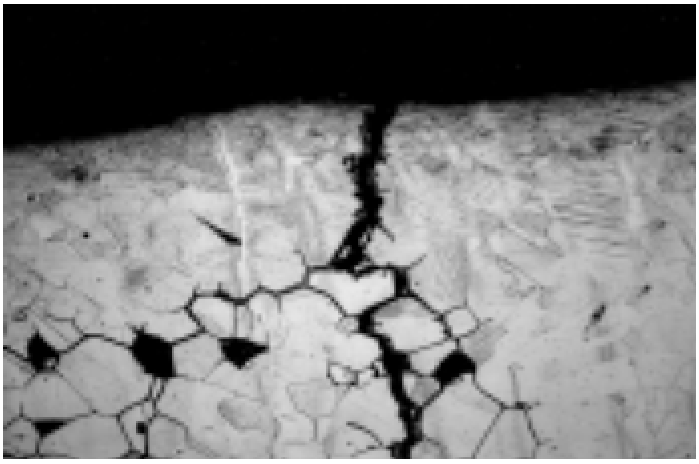


FIG. 7. Cross-sectional observation of SCC crack in shroud lower ring.

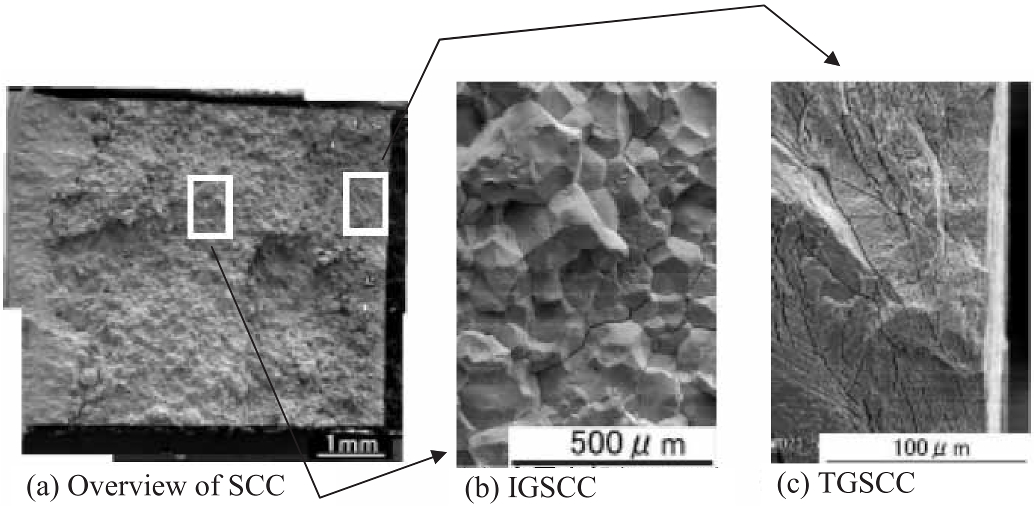


FIG. 8. Scanning electron micrographs of SCC crack in shroud lower ring.

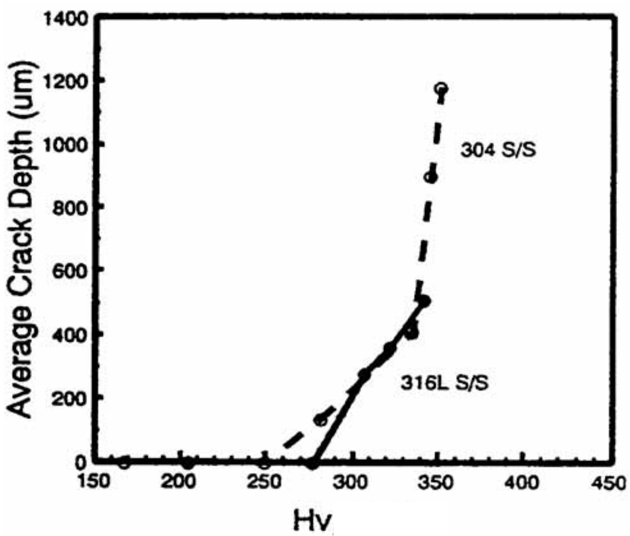


FIG. 9. Relation between average crack depth and hardness.

The cracking was determined to be delayed hydride cracking, a mechanism unknown at the time of construction.

To achieve a major reduction in the rolled joint residual stress, zero clearance rolled joints were developed and used for all new reactor installations. This resulted in a substantial reduction of residual stress in the rolled joint area of the pressure tube.

However, the reduced clearance (mostly interference) necessitated heating of the end fitting prior to inserting and rolling the pressure tube. This was impractical for single fuel channel replacement operations and for large scale fuel channel replacement (LSFCR) projects, as this would increase the time on the reactor face and the dose uptake by workers. To alleviate these concerns with respect to worker dose and still fabricate a quality rolled joint, a 'low' clearance rolled joint was developed. These low clearance rolled joints have become a standard for all fuel channel rolled joints made at the reactor face.

Pressure tube–calandria tube contact

In earlier reactors, the fuel channel annulus spacers, a toroidal spring design, were made of Zr-Nb-Cu and were loose fitting around the pressure tubes. The spring was maintained in a circular shape using a closed loop welded girdle wire. Only two spacers were installed along the length of the pressure tube. A schematic of spacers located between the pressure tube and calandria tube is shown in Fig. 10.

With a better understanding of the pressure tube sag behaviour in the reactor, the number of garter springs per channel was increased from two to four to prevent pressure tube–calandria tube contact during the design life of the fuel channel. However, the loose fitting garter springs were observed to have moved from their installed locations. It is postulated that this movement occurred during reactor construction and commissioning activities while the spacers were unloaded.

This garter spring movement and the resulting prolonged operation of a pressure tube in contact with a calandria tube resulted in a pressure tube[1] rupture in Pickering unit 2 channel G16. The crack initiated at a row of hydride blisters on the outside surface of the pressure tube where it had contacted the calandria tube. Blisters formed due to both the contact and the elevated hydrogen isotope concentrations in the pressure tube material. A joint effort between the vendor and CANDU operators developed a technology to locate and reposition the

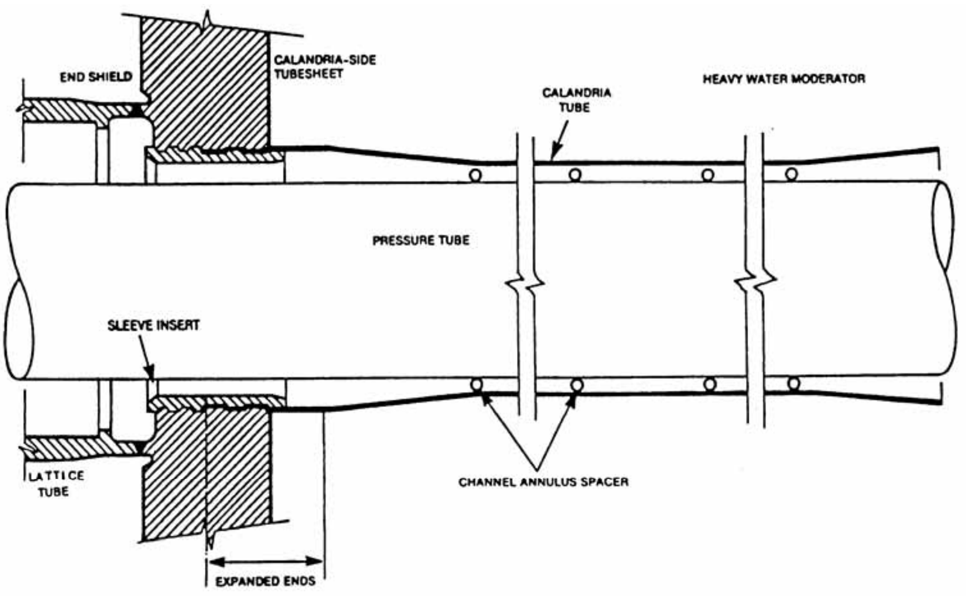


FIG. 10. Schematic of spacers between pressure tube and calandria tube.

[1] The pressure tube was made of Zircaloy-2 and had undergone significant corrosion and hydriding during operation. Zirconium–niobium alloy tubes exhibit much less corrosion and hydriding in operation.

displaced garter springs. For new reactors, the garter springs were designed to be tight fitting around the pressure tube to prevent their movement from their installed locations.

Pressure tube axial elongation

Earlier CANDU reactors were designed with the expectation that pressure tube elongation would not exceed 25 mm, which could be accommodated at one end of the fuel channel. End fittings on one reactor face were fixed to the calandria end shield while those on the other face were ‘free’, unrestrained axially.

Positioning bellows assembly

The fuel channels in some early CANDU units were fitted with flexible bellows at one end and the other end was fitted with rigid stop collars. Once the axial bearing travel was consumed on the bellows side, the stop collar weld had to be cut and the fuel channel shifted towards the stop collar side. Then the stop collar had to be rewelded. This operation, called the ‘west shift’ was necessary after less than ten years of operation and was very time consuming and dose intensive and still only achieved after another 8 years of operation.

Inboard and outboard journal rings

The bearing assemblies support the end fittings in the calandria end shield. Each bearing assembly consists of a bearing sleeve and a journal ring. In earlier reactors, the total channel axial elongation for the design life was significantly underestimated. The bearing sleeve/journal ring assemblies in these reactors could only accommodate half (or less) of channel elongation, before running out of travel [8].

3. ORGANIZATION FOR HEAVY COMPONENT REPLACEMENT

3.1. PROJECT TEAM FORMULATION

Large and complex projects, such as heavy component replacement, are high in cost (including equipment supply, installation and possible plant outage duration impact), must resolve difficult technical issues and involve the organization of hundreds of permanent and temporary workers. Such projects require a strong project organization with highly skilled personnel.

This highlights the need for:

- A strong project team structure;
- A good work organization;
- On-time labour resource mobilization;
- Trained and qualified personnel;
- Qualified processes.

The whole project will be managed by a project team, the size of which will change during the project, starting with the decision to replace one or several heavy components in the plant, until the restart of the plant fitted with its new components, after testing and delivering the ‘as built’ documentation, including the final report and project experience feedback, as well as licensing issues. The project overall duration may be significant, directly linked to the new components’ delivery (5–6 years is the actual worldwide delivery time for components made from large size forgings).

The basis of the project team organization is the work organization itself. Most utilities and engineering and services companies make their projects rely on a ‘work breakdown structure’ (WBS), which allows the project team to clearly identify all ‘work packages’, including for each of them, the responsible person as well as the duration, the necessary ‘entry’ data or supplies, and the ‘result’ of the work package, in terms of documentation and/or physical product. The WBS has a structured hierarchy, which appears through the numbering of the work packages. Specific attention should be paid to work package interfaces to allow quick and reliable interface checks. The work packages are also referenced in the project overall schedule and the numbering allows checking the validity of the schedule links.

The project team will mainly be involved in four phases of the project:

- Launching of the project;
- Design phase, with a priority for the long delivery items;
- Intermediate phase, including the exhaustive oversight of component manufacturing supply and the preparation for site installation;
- Installation, including all required tests and commissioning.

During the project launch phase the project manager will put together a reduced team. The objective is to select rapidly, on the basis of a request for quotation (RFQ) supported by specifications and corresponding offers, the experienced suppliers able to supply the new components, prepare the licensing documentation, implement the replacement with the corresponding documentation and perform the restart tests.

The project team will then grow to support the design phase, including all interfaces with the plant, the regulatory authorities and the suppliers. Within the design phase are commonly included the N-2 and N-1 walk downs on-site (see Section 3.2.3), necessary to certify, complete or modify the as built documentation.

The maximum size of the team is foreseen at the site preparation time, and during the installation. The site preparation step (which may begin more than 12 months before the implementation) includes the mobilization of skilled personnel, as well as the regulatory qualification for specific processes. The mobilized person’s first job is to support this qualification and extend training to all other concerned technicians and operators.

For the installation, it is generally recognized that the project team is deeply integrated with the installation supplier’s project team, in order to optimize the interfaces and reaction time, within the plant outage organization.

A typical project organization is given in Fig. 11. In this case, the main supplier scope includes the component supply and its implementation on-site, and as for all significant replacement projects, it was decided to create a project review board associating the utility and its main suppliers.

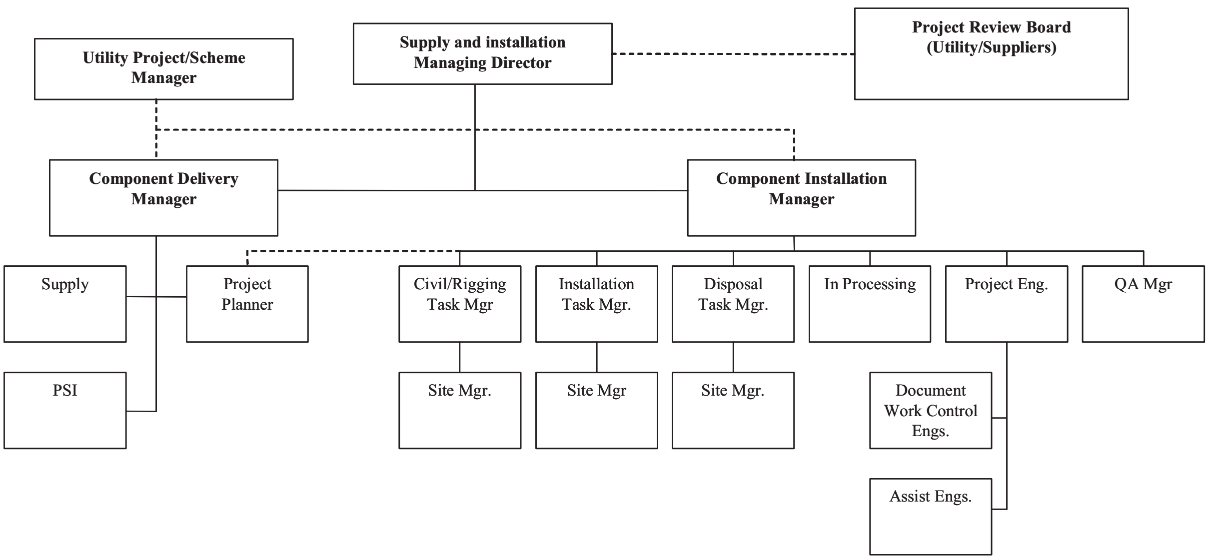


FIG. 11. Typical project organization.

3.2. PROJECT MANAGEMENT APPROACHES

3.2.1. Replacement strategy

In addition to typical tasks of a maintenance project inside a nuclear power plant, the heavy component replacement projects require a specific approach to optimize the replacement strategy in terms of safety, collective radiation dose, cost and schedule. This approach will direct all task sequences and site organization, in the reactor building, as well as on the plant site. The strategy is built in close association with the component manufacturer and installers.

The possible strategies for each foreseen case of component replacement are highly dependent on the type of component replaced. Consequently they are described in the applicable section (see Sections 4–10)[9].

The strategy will mainly affect:

- The component supply scope (e.g. one piece SGs or two piece, bare RVHR or complete with welded control rod drive mechanisms (CRDM), reuse of old CRDMs or new ones, RCL spool piece length, core shroud complete or partial).
- The shipment and the ground transportation on-site (choice of transport path, ground reinforcement, component orientations).
- The need for specific temporary facilities on-site (when SG safe end machining is done on-site, for welding of new or old CRDMs on the RVHR, for RCL nozzle machining).
- The eventual need for temporary containment opening/closure (one piece SGs may not be able to pass through the equipment hatch or due to unsuitable location of equipment hatch or too small inner diameter of hatch), the design of the containment vessel (prestressed or non-prestressed reinforced concrete?) and the choice of opening place (roof or wall).
- The concept of the reactor building wall itself may limit the capabilities for opening/closure (with or without a steel liner integrated into the concrete wall?). Outside tanks, buildings and/or underground channels may also limit the possible locations for a temporary containment opening. The following cutting technique must also be selected:
  - Core drilling;
  - Diamond saw blade;
  - Hydraulic explosive;
  - Thermal or jet stream.
- The hoisting of the component inside the containment building (the overall handling sequence inside the containment building must be defined with interface examination, e.g. crane beam clearance, crane capability).
- The civil works inside the containment building (need to modify cubicles and concrete floors at various levels).

3.2.2. Material procurement and implementation works

The process used to select vendors is conventional. The RFQ will contain the following elements allowing the potential suppliers to propose part or all of the utility requests:

- All as built data concerning the component to be replaced and its interfaces with the environment (inside and outside the containment building);
- Specific requests for any modifications (interfaces, power upgrading or material improvement);
- Conditions of implementation (outage constraints);
- Possible replacement strategies.

The potential vendors are those who can offer recognized capabilities and experience feedback, including qualified processes, for similar work in engineering/licensing, manufacturing and installation of main components of the primary circuit of nuclear power plants.

Depending on its own capabilities in engineering large projects, the utility may be in the position to place separate orders for the above listed works, splitting the installation part into several contracts and then dealing with the interfaces between these contracts. It should be noted that based on experience feedback, it might not be the best practice for the utility to manage all contractor interfaces.

The final orders will be placed only after a ‘global’ evaluation involving the different possible strategies. The RFQ will give the potential vendors opportunities to propose several options to help the utility to optimize its choice (scope of each vendor/global replacement strategy).

When separate contracts are placed for component fabrication and installation, special care should be given to physical and scope interfaces. Examples include, but are not limited to:

- Responsibility for evaluation of differences in component replacement on interfacing plant systems, licence basis loadings and any differences in results on affected structures, systems and components;
- Delivery place and conditions;
- Equipment protection for intermediary storage;
- Piping and nozzle interfaces (end position and state bevelled or not, responsibility on data from metrology, welding qualification, including thermal treatment requirements for site final welds, etc.);
- Supporting (saddles/legs, skirts) interfaces;
- Hydro test;
- Manufacturing NDE report;
- Test and spare parts (blind flanges, bolting).

To facilitate the interface management, it is highly recommended that the orders for supply, engineering and installation be placed at the same time, even if placed to different suppliers. Although material procurement is the leading factor for the overall delivery date and the implementation preparation duration is shorter, it is not worth delaying the corresponding work order.

Specific attention is to be given by the utility and main installation supplier (the ‘installer’) to sensitive implementation works. Those should be addressed by the installer itself, or recognized experienced subcontractors. The considered works are:

- Cutting, installing, welding and NDE of primary, secondary and auxiliary piping;
- Handling and hoisting;
- Containment building opening.

3.2.3. Project planning and scheduling

Project planning and scheduling is necessary for project success. For large component replacement projects there are usually four separate schedules that have to be integrated to cover the total project scope. These schedules are:

- Component fabrication;
- Component installation;
- Utility;
- Site outage.

All of these schedules need to be developed using the same scheduling tool and revision level. This is usually specified by the utility as part of the request for bid package. Each of these schedules will be developed and maintained by separate organizations but will all have links to each other. It is the responsibility of the owner of each schedule to identify links to the other schedules.

Component fabrication: This schedule is developed by the component fabricator and should include:

- Component design;
- Component licensing;
- Component fabrication;
- Component delivery.

This schedule should include utility activities for review and approval of documentation based on the utility's directions.

Component installation: The component installation schedule is managed by the installer, and includes its own design and preparation phases. For the installation phase, the interfaces are very important with the regular outage maintenance work. It is thus recommended that an integrated scheduling team be created with installer and utility plant outage personnel. The implementation schedule will also make reference to the reactor building's floor layout (lay down space/storage/accesses/scaffolding/shielding) and crane availability at each step. Typical areas covered by the installer's schedule are:

- Pre-outage activities:
  - N-2 walk downs;
  - Installation design development;
  - Installation work package development;
  - N-1 work activities;
  - Site preparation work activities;
  - Component preparation activities;
  - Pre-outage work activities.
- Outage activities:
  - Outside rigging;
  - Containment opening;
  - Inside rigging;
  - Cutting and welding;
  - Removal and replacement;
  - Site 'demob'.

This schedule should include utility activities for review and approval of documentation based on the utility's directions.

Utility: This schedule is developed by the utility's project team and consists of those activities the utility project team is responsible for. These activities should include:

- Project approval;
- Project funding;
- Project staffing;
- Contract issuance;
- Site preparation;
- Vendor document approval;
- Radiation work packages.

Site outage: This schedule is developed by the utility staff outage organization and consists of the routine outage maintenance activities. It is extremely important that this schedule is integrated into the installer's outage activities to ensure efficient utilization resources. This schedule should include:

- Shutdown;
- Reactor coolant system (RCS) draining;
- Defuelling;
- Refuelling;
- RCS filling;
- Startup;
- Routine outage maintenance.

3.2.4. Control of project progress

The project management team is in charge of continuous project progress control. The aim is to verify the conformity of deliveries in terms of cost and schedule. The WBS structure helps in getting indicators for the verification of deliveries on their due date. Standard software will help for follow-up of costs by the utility.

Delivery availability is to be determined in close collaboration with the main suppliers. Regular progress control meetings have to be planned at the very beginning of the project. The frequency of the meetings are adapted to the tasks and the periods considered.

For component supply, meetings will be regularly organized during the design phase, with indicators on documentation delivery. Material procurement requires a specific follow-up, especially for forgings and tubing.

The fabrication phase requires permanently resident QA inspectors in the manufacturing shop for continuous monitoring. Their reports will dictate the frequency of progress meetings. For implementation, meetings will be regularly organized during the design phase, with indicators on documentation delivery. Joint attendance of the manufacturer and the installer is necessary during the design progress meetings.

During the implementation preparation phase, progress meetings will be organized to check:

- The processes and tooling preparation and qualification;
- The personnel training and qualification.

During the implementation on-site, a daily progress report is necessary. It will rest on a daily revision of the integrated schedule, and a daily meeting of the utility's and the installer's site management teams. Progress of the project is to be regularly presented to the project review board.

3.2.5. Management of information

The project management team is in charge of organizing the management of the information for the utility's management through the project review board, within the project team and from and to the suppliers and the utility site organization.

Confidentiality will be required from personnel and organizations external to the utility, the safety authorities and the suppliers. All project external communication will be managed through the communication division of the utility.

3.2.6. Selection of codes and standards

The RFQs and the consequent contracts defining the heavy component supply, licensing and installation will mention the applicable codes and standards. The selected codes will be:

- Applicable national regulations, including safety authorities' codes and regulations;
- International recognized codes (ISO, ASME, RCCM, RSEM) for nuclear work;
- The utility's specific regulations and procedures.

The selected codes and standards will apply to:

- Design of component;
- Manufacturing rules;
- Licensing conditions;
- Installation conditions;
- Testing and NDE;
- Radiological protection;
- Safety of personnel;
- National labour conditions;
- Environment preservation (including waste treatment).

3.3. DESIGN APPROACH TO FACILITATE IMPLEMENTATION

Two elements facilitate and improve the design of the new component and its implementation:

- As built documentation and material specifications;
- Design tools (3-D graphics).

3.3.1. As built documentation and material specifications

Experience feedback has demonstrated that original documentation and specifications from operating experience are not a 100% true representation of the operating power plant. In fact, enhancements and modifications are implemented throughout plant life. It is then mandatory that, at the very beginning of the project, the utility project team should be able to gather the updated documentation. This documentation will be part of the RFQ, and later on be part of the contracts to the suppliers.

Nevertheless, the suppliers' installation design based on these documents will not be considered as final until the two foreseen walk downs have been executed and have provided the opportunity to check the important interface data and modify the implementation accordingly in case of discrepancy.

3.3.2. Design tools (3-D graphics)

The 3-D design tools provide the opportunity to:

- Visualize the design of the new component;
- Verify all interfaces in the reactor building, not only for the component itself, but all pipes, valves and supports;
- Visualize and check for interferences during the handling and hoisting movements.

In order to derive the maximum benefit from these software tools:

- Interfacing the component calculation code with the metrology system measurements should be carried out;
- The same tools to be used by both the utility and the suppliers;
- Preference should be given to standardized industrial tools as opposed to 'homemade' tools.

3.4. IMPROVED IMPLEMENTATION TECHNIQUES/METHODS

The high level of experience feedback on heavy component replacement has highlighted several aspects (human and technical) for which improved techniques and methods are recommended.

3.4.1. Radiation and conventional safety for workers

In the manufacturer's factory, the national regulations will apply. Special attention will be given for component implementation on-site. Compared to other outage maintenance operations, the heavy component replacement concentrates a large source of risks for personnel. In addition to the fact that heavy loads are handled (up to about 400 t), the following risk factors are present:

- Large labour concentrations (can number more than one hundred persons) present in a restricted area;
- Work at multiple levels in same area resulting in risk from any dropped tool or other object;
- Large range of different skills (machining, welding, mechanics, handling, NDT, chemistry);
- Temporary need for radiographic inspection and gamma sources;
- High temperature areas (weld preheating and stress relieving);
- Work in confined areas (pipe grinding, inspection for RCL and steam line, SG 2 pieces final girth weld);
- Work in a radiological environment.

In addition to, or within, the national regulations applicable for the implementation site, the project team should perform a specific security study to identify and address all the potential risks linked to the implementation.

3.4.2. Hoisting and handling

Major experience feedback on heavy component hoisting and handling has generated a list of recommendations:

- Overall hoisting and handling to be contracted to a specialized company directly or preferably through the installation contract. Throughout the world, a reduced number of such companies have corresponding experience.
- The positioning of the main component on its support and the piping end adjustment for welding require very precise handling (accuracy better than 1 mm).
- Exhaustive preplanning decreases the number of main component transfers to the minimum required. This is for limiting the risks and assuming a final optimal orientation while entering the containment building, where orientation changes are not easy. The shipment saddles should be used to allow transfers, and thus have the adapted design.
- Most of the power plants have not kept the erecting load capabilities of the polar crane due to maintenance reasons. Crane beams and an additional trolley should be adapted and qualified.
- The polar crane clearance below the hook is generally too low to allow for vertical lifting of the main components (SG and pressurizer). Several options should be scrutinized, such as lifting through the crane beams, using a temporarily installed additional jack lift, or lowering the cubicle walls by concrete removal, or a combined lifting with two trolleys for a component 'tilting in the air', or lifting by an externally placed ultra heavy crane through temporary openings above each component location.
- The polar crane utilization should be scheduled for the entire outage duration, including other users.
- The replacement of PWR internals in one set requires specific handling with the transportation/storage cask.

It is worth having additional auxiliary lifting means introduced in the reactor building to reserve the polar crane utilization for heavy handling.

3.4.3. Containment building opening

Containment openings are required when it is not feasible to move the main components through the plant's equipment hatch. As previously discussed, in some cases, SGs are handled in two parts: lower assembly, with the primary channel head, tube sheet, lower secondary shell and tube bundle; and the upper assembly, with the remainder of the secondary shell and internal moisture separation parts, main steam nozzle and main feedwater nozzle. The two piece process reduces the length of the component parts to be handled and allows replacement through the equipment hatch for some plants where the equipment hatch diameter is adequate and there is enough space inside the equipment hatch to handle the shorter components, but not a complete component.

In the case where the equipment hatch diameter is not sufficient or the space inside the containment is not sufficient for the heavy component handling without excessive cost for removal and reinstallation of multiple items, a temporary opening through the containment building may be the preferred.

Nevertheless, it should be noted that the containment opening may not be feasible, depending of the containment structure design (e.g. post-tensioned concrete with cables directly in contact with concrete).

Containment openings have been located in the vertical cylindrical wall and in the dome shaped roof. The choice depends on access inside and outside the containment, interferences and available space and cost of a crane capable of handling the heavy load and having the long reach needed to lift a complete component over the top of the containment.

The containment opening is a significant addition to the scope of work for the project. The type of containment has the largest effect on the amount of work involved and the cost of the opening. Post-tensioned concrete greatly increases both the design and implementation effort for temporary openings.

For all cases, engineering must consider loads during the project, including all conditions related to component movement on the structure with the opening in place and the capability of the restored structure to meet design basis criteria for the remainder of the plant's life. A number of processes have been successfully used for removal of the concrete. Key considerations are:

- Speed of removal;
- Capability to expose reinforcing steel (rebar) around the periphery of the opening for splicing without damage to the rebar;
- Removal of concrete from the liner plate and attached embedment without damage to the liner plate;
- Cost and environmental factors (dust, debris and water control).

A high pressure water jet has been used to remove concrete without damaging rebar or the liner plate. While this method is more expensive, the benefits of eliminating damage to the liner plate and rebar and lower risk of delays in removing concrete for rebar splicing have made this the preferred choice for a number of cases. Diamond wire sawing and robotic hammers are also effective tools that may be the preferred choice for specific situations.

Post-tensioned structures require additional engineering attention due to the non-linear effects of partial de-tensioning of the structure and re-tensioning after the opening is restored. The partially de-tensioned structure assumes a slightly distorted shape that results in the restored opening assuming the same distorted shape and slightly different loads internal to the structure when it is re-tensioned. In addition, the properties of the concrete in the opening are more critical due to the effects of shrinkage and creep on long term tendon forces. The post-tensioning tendons removed from the opening area are often replaced and this material has a long lead time. The sequence of replacing the liner plate, splicing rebar, installing tendon ducts, installing forms for the concrete, placing and curing the concrete and tensioning the tendons is, typically, the critical path for the project and plant outage.

3.4.4. Metrology and topometry

Use of precision measurements is strongly recommended for the replacement operation. In the first step they are used to confirm and update the as built documentation, with regard to component sizing, positioning and the piping and support interfaces. A 3-D as built modelling is obtained during the walk downs. In some cases, the collected data will be used for new component design purposes and then plant walk down will be implemented accordingly.

Specific attention is needed regarding the orientation of the components and interfaces with supporting structures. Typically, the components are installed, the supports fitted to them and the remaining surrounding structures completed. For installation of replacement components, adjustment of the supports may not be feasible. The metrology should provide the information needed to design adequate clearance for replacement components and shimming or other adjustment as applicable for the final interface with the supports.

Metrological tools are also used to perform the sizing of the new component in shop, and determine the cut sections of the main and secondary piping as well as the pipe and nozzle edges preparatory to welding.

A good accuracy (0.1 mm) of these tools is needed to obtain the required pipe end positioning (about 1 mm). Among the acceptable processes are the optical measurement (using theodolites), the digital photogrammetry, the laser tracking and the laser scanning.

Preference is to be given to tooling requiring the shortest stay of operators in the areas of severe radiological environment and having minimum impact on the critical path of the implementation.

Potential suppliers for main component replacements have experienced various measurement systems, and generally use a mix of existing measurement tools associated with specifically developed software.

3.4.5. Machining and welding

Machining and welding are part of the base technologies for main component replacements. As they directly affect the integrity of the primary circuit, the corresponding processes, tooling and operators have to be qualified in accordance with the selected codes and regulations.

Machining (cutting and bevelling) processes and corresponding tooling have to address the following criteria:

- Waste reduction;
- Prevention of foreign materials entering the piping system;
- Precise execution.

The welding process and tooling should be performed at high speed and result in a high quality uniform deposit with guaranteed chemical and mechanical characteristics.

Secondary and auxiliary piping is generally welded manually. A large part of the auxiliary piping may be prefabricated outside the containment. Primary piping and the core shroud vessel are remote control welded. Tooling and process have been specifically developed and qualified by the several potential suppliers for main components replacement implementation. Weld design will be adapted for pre-service inspection and future NDE.

Particular geometries or a high dose rate environment may require the use of robots. This has already been undertaken in RCL, CRDM welding and core shroud replacements.

3.4.6. Shielding and decontamination

In order to fulfil the as low as reasonably achievable (ALARA) programme, a shielding layout is performed, in addition to possible studies on water level optimization in the circuit and reactor pool (see for example the case for BWR core shroud replacement). The utility should also take precautions during the shutdown phase to minimize the dose rate.

The layout and corresponding volumes of temporary shielding are to be established by the utility (in a shielding plan) with the participation of the installer having experience in this aspect.

Flushing the circuit before the outage (and flushing the cut pipes with clean water) is not fully efficient with respect to the dose rate, as it is necessary to eliminate a part of the oxide skin on the inner surface of the pipes. Several processes for decontamination of pipe ends or pipe spool pieces are available. The two main types are mechanical (sand blasting) and chemical decontamination. In both cases, specific care is taken to plug the circuit against chemical products or mechanical residues (foreign material exclusion programme implementation).

Decontamination operations are complex and managed by specialized subcontractors. They require a specific qualification in accordance with the selected codes and regulations, including the innocuousness demonstration process.

3.4.7. Component transportation on-site

The component transportation on-site has to be managed by the project team in conjunction with the component supplier, generally in charge of shipment and delivery to the site, the installer, and the site management. This work can be subcontracted to the handling/hoisting supplier with restrictive conditions on the allowed areas of manner of transfer and component orientation. An engineering study will take account of the optimal mode of transportation considering any interference (e.g. high voltage cables, underground obstacles), including necessary civil work modifications and/or ground reinforcement.

3.5. SITE INFRASTRUCTURE PREPARATION AND MODIFICATION

The heavy component replacement operation has a major impact on the site infrastructure, from its necessary preparation before the selected outage up to the restart of the unit after the replacement. At the early beginning of the implementation project, the project team will define, in accordance with the site management and the installer, the 'best' plant conditions under which to start the operation. Depending on the replaced component, these conditions should affect the RCS level, as well as the primary chemistry to reach the outage configuration.

The utility should notify the suppliers (through the utility's specific regulations and procedures) regarding all material whose introduction is forbidden inside the plant and the reactor building.

The outage to fulfil the replacement will be selected when establishing the utility's strategy. It is generally favourable to select an outage including already significant works, which can be the case for other plant life management operations, or decennial maintenance, or other component replacements. Outage management should nevertheless avoid additional activities in the area of major component replacement.

In preparation for the outage, the project team will elaborate, in collaboration with the installer, documentation, including the layout of each floor of the reactor building. This documentation will show, at each step of the outage and in accordance with the implementation schedule, the position of the new and old components, all permanent and temporary storage areas for tooling, insulation and containers, in relation to the cranes' positions and reaches. This layout will also include the step by step scaffoldings and shielding.

In addition to the work inside the reactor building, preparations will be made for all the on-site work. Main areas of investigation for the project team will be:

- The storage building for replaced (old) components. This building will be designed in accordance with the site radiological classification and the national regulation. Several options are available and will be studied early. The replaced components may be stored in one piece or cut into smaller units. In all cases, a specific study will be performed in a timely manner for optimizing the radiological conditions (blind flanges, painting) for transfer and storage. The building size determination will also depend on the utility's long term strategy.
- The facilities and temporary utilities (hot and cold areas). The installer will express its need for work areas, and the energy, water, air supplies to store and prepare the new components before entering the reactor building. It may be the cold area (new SGs, RVH, internals or auxiliary piping prefabrication) for machining/welding, or the hot area (re-installation of old CRDMs). There should also be a need for a hot area to decontaminate and refurbish reused piping or SG supports.
- The disposal of radioactive waste. The utility should be in charge of disposal of these wastes, including equipment, but also reactor building personnel clothes and wastes. The project team is in charge of providing an estimate of these wastes during the preparation phase.
- The personnel facilities. As the replacement implementation needs many more people inside the controlled area than usual, the requirements in terms of locker room, specific clothes, contamination monitoring systems and site access capacities need to be addressed accordingly.

3.6. PRE-SERVICE ACTIVITIES

All pre-service inspection will follow completion of the replacement operation, according to the selected regulations. A final set of documentation has to be set up. It will include:

- The component manufacturing documentation;
- The component in-shop pre-service inspection reports;
- The installation documentation;
- The installation inspection report;
- The pre-service inspection report;
- The hydro test report;
- The restart test report.

3.7. ALARA PROGRAMME

The project team includes an ALARA team. This team is in charge of optimizing (reducing) the overall exposure for the whole duration of the implementation.

In the preparation phase, it is in charge of issuing specific documentation including:

- The applied regulation and 'rules' brought to the attention of the suppliers. One of these rules should be an incentive on the price, linked to the 'contractual cumulated dose'.
- Radiomapping of all areas dealing with the replacement implementation, inside and outside the reactor building. This mapping is based upon data given by the plant. They are to be checked/modified/confirmed according to data collected during the N-2 and N-1 walk downs.
- Calculation of the cumulated provisional dose for the implementation, for each of the implementation strategies to be evaluated. This calculation is made using the estimated time for each individual operation and the local dose rate, including the optimized requested shielding and water level management. All the necessary data are collected from the suppliers. The suppliers will prepare their own partial ALARA programme on the same basis. An optimization should be made for each supplier, on the basis of an estimated cost of the man·mSv, to select a volume of shielding and/or the use of robots. The suppliers' offers and the resulting contracts will then include a 'contractual cumulated dose' corresponding to the dose rate mapping, the final replacement strategy, the shielding layout and the selection between human/robotic tasks.
- A provisional daily implementation curve, based on the implementation schedule and the above calculation of the cumulated dose.

During the implementation, the ALARA team is in charge of:

- Checking the real radiomapping at the beginning of the outage. If this mapping deviates from the base for the dose rate calculation, action has to be taken to update the contractual cumulated dose for each supplier, including a possible specific modification of the shielding, and even of the strategy, if the real controlled radiological dose rate is quite different from the hypothesis.
- Checking that the shielding plan is properly implemented and making suggestions for improvement, if required.
- Having a daily follow-up of the individual and cumulated dose, in order to make immediate corrections in case of deviation from the provisional curve and to protect individuals. To obtain a significant follow-up, and possible corrective actions, it focuses on the need to implement a registration of the individual dose per type of work. This furthers the registration of the individual doses through a codification linked to the WBS numbering.
- Having inspectors making a survey on-site to assist the workers, ensure that all 'protective rules' are applied, and check that local dose rates are in agreement with mapping.

At the end of the replacement operation, the ALARA team prepares its final report, including the cumulated dose and the general feedback including the deviations and lessons learned. This report is included in the implementation final report.

3.8. QUALITY MANAGEMENT

All supplies, documentation and operations will be done under a quality assurance programme as applicable to the activity and defined in the contract. The project team includes a quality programme team referring directly to the project team manager.

All subcontractors will refer to the quality management manual of their company, according to contract requirements. Typical quality assurance standards include:

- ISO 9001 last revision;

- IAEA Safety Standards Series No. GS-R-3, the Management System for Facilities and Activities Safety Requirements;
- US-10 CFR 50 appendix B.

The quality system of all subcontractors will have been evaluated by a recognized authority and certified. The quality programme team is in charge of:

- Preparing the quality programme requirements in the RFQs and suppliers' contracts;
- Making the quality survey during the design and manufacture of the component;
- Permanently controlling documentation and surveying all processes and tooling qualifications;
- Making the quality survey for the implementation and licensing documentation;
- Checking all documentation availability at the beginning of the outage;
- Inspecting and assisting during installation, pre-service inspection and prestart tests;
- Checking the availability and pertinence of the final implementation report.

3.9. PROJECT STRUCTURE FOR SGR

As the first step towards efficient processing, the entire project is broken down into job packages, the most important of which are listed below. The structure of the projects is of fundamental importance to subsequent phases of the project. The project team is formed on the basis of this structure (with the appointment of the specialists required), and project schedules (engineering schedules) are prepared which embrace the entire preparatory phase up to the commencement of work at the plant.

3.9.1. Putting the project team together

All specialists required for the individual job packages from an organizational unit are under the coordination of a project manager who becomes their direct superior for the duration of the project and coordinates all activities with the main contractor. Processing projects in this manner has proven extremely effective for the following reasons:

- Communication and information pathways are short and the number of interfaces is minimized, allowing the team to respond rapidly to special customer requests.
- Efficiency is high because there is a direct link between work activities and the decision making process within the team.
- Motivation is high and team members identify with their assigned task because they are only involved in a single project.

3.9.2. Project schedules

Project schedules are prepared for all job packages. They have a standardized arrangement and contain the following sections:

- Project preparations;
- Design, engineering, licensing;
- Order processing and manufacturing;
- Qualification of personnel and equipment;
- Site planning;
- Preparation for work at the power plant.

These schedules are an important basis for high quality project processing. They are used to coordinate the activities of all participants, including subcontractors, and are presented along with a monthly status report to the customer so that they are always fully informed and integrated into the project. Each schedule contains approximately 100 activities. These activities are included in condensed form in a higher level schedule (the main engineering time schedule):

- Structuring into work packages;
- Project team assignment;
- Generation of project schedules;
- Engineering, design, design calculations;
- Licensing;
- Purchasing, manufacturing, documentation;
- Installation procedures;
- Qualification of processes, equipment, personnel;
- Site mobilization;
- On-site performance, commissioning, documentation;
- SG ground transportation;
- SG rigging and SG support;
- SG vessel work;
- Cutting/machining nozzles;
- Optical measurement;
- Decontamination elbows;
- RCL piping clamping/fit up;
- Welding of RCL piping;
- Secondary, auxiliary piping;
- Instrumentation and control insulation;
- Mock-up;
- Multipurpose building;
- Startup, testing;
- Radiation protection, safety;
- Other facilities outside
- Temporary utilities;
- Scaffolding, shielding;
- Cleaning;
- Other services.

3.9.3. Documentation for work

One of the most important planning steps is a complete visual inspection of the SG during one of the refuelling outages prior to replacement. This inspection also includes the secondary piping, the SG support structure and the RCL. All potential interferences must be recorded, the rigging paths measured and openings checked by using mock-ups. In preparation for this visual inspection, a plant walk down list is drawn up showing all activities which must be performed during the plant outage prior to replacement of the SGs.

The ‘general work sequence plan’ is the main document for all activities on-site. It lists in a logical sequence all work that is to be performed. It includes approximately 15 000 activities indicating the steps involved in each, and makes reference to all applicable documents such as drawings, work procedures, system descriptions, etc.

Another important document is the inspection plan which is the official quality management document. It contains all necessary inspection steps and is prepared in accordance with project specific requirements and country specific standards which regulate all activities concerned with quality management, NDEs and other tests and also specifies who will witness each (consortium, supplier, customer, licensing authority).

A site schedule is prepared for the replacement work. Independent of their association with the job packages, the required activities are planned chronologically and incorporated by the customer into the overall outage planning. The site schedule comprises about 6000 separate activities, with one hour being the shortest unit of time assigned.

The site schedule and the general work sequence plan are also used in resources planning, i.e. in deciding which personnel, equipment, and machinery to use, and in estimating radiation dose.

3.9.4. Personnel training

Experience with such replacement projects has shown that personnel training contributes enormously to optimum performance in terms of work quality, meeting deadlines, and minimizing the radiation exposure of the personnel. Hence, comprehensive theoretical and practical training programmes have been developed for the most important work steps. Consortium and subcontractor employees are given specialized training for the tasks and conditions of the SGR at hand (see Figs 12 and 13).

All personnel training programmes are performed using a full-scale mock-up of the lower section of the SG (primary channel head with a section of the RCL and the SG nozzles) including the interferences in this area.

Training and qualification are performed for the following main activities:

- Welding of the RCL;
- Cutting and machining of the RCL;
- Decontamination of the remaining ends of the RCL;
- Mounting of special shielding devices;
- Video inspections inside the RCL and the primary channel head.

FIG. 12. Personnel training using mock-ups.

FIG. 13. Narrow gap welding equipment installed at the mock-up.

4. SGR IN PWRs

As a result of PWR SG heat transfer tubing degradation and subsequent required tube plugging, many PWR SGs have had to be replaced in order for the plant to be able to produce rated power levels. Similarly, many WWER SGs have experienced degradation of ligaments in the perforated cylinders of the cold side primary coolant collectors, which also required tube plugging repairs and eventual replacement. The cause of the PWR SG tube degradation was PWSCC of alloy 600 mill annealed tube material. The cause of the WWER cold side primary water collector ligament cracking is believed to be a 'slow strain rate' assisted SCC of the low alloy steel material.

Replacement PWR SG tube bundles are made from thermally treated alloy 690 tube material, which has been shown to be resistant to PWSSC as well as OD chemical attack. Primary water collectors of SGs in WWER are made from stainless steel, which has also been proven to be resistant to SCC attack. The replacement collectors have also been made with increased thickness and ligament width to lower the stress level in the material [10].

4.1. STRATEGIES FOR SGR

4.1.1. Design, design calculation, licensing

In SGRs, the tasks of special importance are design and design calculation. These activities apply for temporary equipment (such as SG rigging, devices for piping, etc.) and for permanent plant equipment including all related modifications (such as re-routing of piping, thermal insulation, steel liner of containment opening). Piping design covered re-routing of feedwater and auxiliary feedwater, adaptation of instrumentation piping, main steam, blowdown, reactor temperature detection, drain, sampling. Design calculation basically covered analyses of structural, seismic and fluid dynamic data.

In parallel with basic engineering, the safety of all activities leading to modification of plant equipment or activities introducing a specific risk, such as handling, rigging, transportation, waste handling, is evaluated for review by the licensing authorities. These evaluations are in accordance with 10CFR50.59.

4.1.2. Scope and sequence of SGR

An SGR project includes all of the following activities [11–13]:

Fabrication of replacement SG components:

- Development of SGR technical specification;
- Consideration of potential power uprate or change in design concept, such as feeder ring versus preheater design, increased steam pressure or increased tube plugging margin;
- Consideration of design enhancements, such as electropolishing channel head, pre-service preservation of new tubing or capability for shell-side fluid circulation during wet lay-up and chemical cleaning;
- Consideration of instrumentation location changes, such as change in level tap height due to taller tube bundle;
- Design report documenting design analysis of replacement components in accordance with ASME code and/or other national jurisdictional authority;
- Design data report documenting physical capacities of replacement components (primary fluid mass, secondary fluid mass at various power levels, metal mass, weight, centre of gravity, etc.), which are used to assess impact on plant licensing basis;
- Transportation of SGR components to plant site, including ocean vessel, river barge, railroad or overland carrier.

Installation of SGR

- Metrology (usually performed with a laser templating system) of all required pipe cut locations, support interfaces and control point targets installed as 3-D reference locations, performed two cycles before replacement outage in order to impact replacement component design.
- Construction of permanent on-site storage building (‘mausoleum’) for original components (building might also function as temporary storage location for replacement components upon arrival).
- As built metrology of replacement components, construction of 3-D model, determination of critical interface coordinates, machine nozzle weld preparations on replacement components.
- Civil engineering ranging in scope from temporary movement of obstacles in component installation path to broaching concrete and steel containment building and major interior subcompartment walls.
- Creation of required site infrastructure, such as temporary cranes, lay-down platforms in containment, processing facilities for exceptionally large site labour force, etc.
- Cutting reactor coolant piping, main steam piping, feedwater piping and blowdown piping.
- Rigging and removal of original SG components and relocation to permanent on-site storage building (mausoleum).
- Decontamination of reactor coolant piping near cut ends in support of ALARA.
- Metrology survey of piping interface locations after component removal and machine piping weld preparations.
- Rigging and installation of SGR components, fit-up of nozzle to pipe joints, automated narrow groove welding of all pipe joints, reconnection of component supports, NDE of all welded pipe joints.
- Replacement of SGR component thermal insulation, reinstallation and reconnection of level measurement and other instrumentation.
- Civil engineering including closing steel and concrete containment building, post-tensioning/rebinding reinforcement material, restoring interior subcompartment walls and piping to original configuration, NDE of all rewelded pipe joints.

- System leak test or hydro test of both primary and secondary systems to include all welded joints in accordance with the requirements of jurisdictional authority.
- Multiple replacement component installation during a single outage (e.g. SGRs, reactor pressurizer, reactor vessel head) is efficient but requires a high level of coordination and sharing of physical staging areas inside containment during replacement outage.

Licensing points of SGR components

- Scope of licensing effort is dependent on the extent of differences between the original SG and the SGR.
- Weight and centre of gravity – significant differences in weight and/or centre of gravity of the SGR might affect the seismic analysis of the component or the RCS loop analysis, depending on the design margin in the original licensing calculations.
- If the tube bundle size has increased, weight and centre of gravity increases might be partially offset by constructing the pressure shell and tube sheet from higher strength and thinner material.
- Fluid inventories – significant differences in either primary or secondary fluid inventories of the SGR might have an impact on the response of the SGR to design basis transients or the SGR might have an impact on the transient response of the RCS.
- Higher fluid inventories increase the mass and energy release in the event of a hypothetical pipe break, event in containment, while lower fluid inventories reduce the rate of core cooling during other design basis events.
- Primary pressure drop – significant changes in the primary pressure drop might impact the core cooling margin or core lift-off margin.
- If the SGR tube bundle is taller, adding additional tube circuits could compensate and help to maintain the original pressure drop.
- Secondary pressure drop – significant chances in the secondary pressure drop impact the SG circulation ratio, which affects the secondary fluid inventory.
- If the tube bundle is taller, increasing the width of the downcomer annulus could compensate and help maintain an adequate circulation ratio and avoid concentration of ‘foreign’ ionic species that might attack the tubes.
- In order to minimize the amount of licensing effort, the SGR needs to be as physically close to the original SG as possible.
- Uprating the power level of the plant in combination with the SGR requires additional licensing effort.

Reactor coolant piping works

- As built measurement of primary piping and unmachined new SG
- Evaluation of primary piping and new SG measurements
- Clamping of the primary piping
- Cutting of the primary piping

- Removal of the old SG decontamination and setting of in-pipe shielding plug

- Optical measurements of primary piping
- Evaluation of optical measurements of primary piping
- Vertical and horizontal displacement of primary piping for shrinkage compensation
- Optical measurements of primary piping after displacement

- Marking of weld edge location W1 and W2
- Machining of primary piping weld edge geometry in W1 and W2

- Displacement of cross over leg for installation clearance
- Lowering of new SG
- Fit-up of SG with hot leg
- Release of cross over leg displacement for installation clearance
- Fit-up of cross over leg with SG

- Fixing of components by hydraulic jacks
- (SG, hot leg, cross over leg)
- Welding of W1 and W2 simultaneously
- Release of displacement of hot leg and cross over leg during welding according to weld shrinkage process

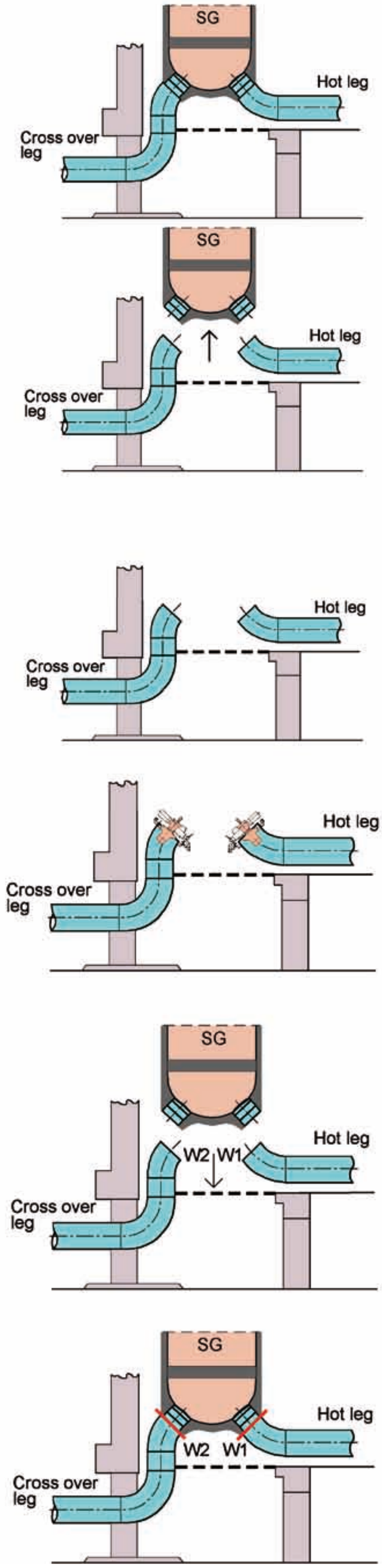


FIG. 14. Sequence for reactor coolant piping works.

4.2. OUTLINE OF SGR

4.2.1. Optical survey

The optical survey is performed by applying a combination of photogrammetry and industrial optical techniques using electronic theodolites. The process was previously proven in terms of accuracy, reproducibility, radiation exposure and speed of use. This technique was also used during previous outages to determine the geometry of the existing reactor coolant pipes which allowed machining the new SG nozzle during the manufacturing phase (see Fig. 15.).

Pre-outage operations:

Operating module 1: Preliminary as built measurement:

- Specify or verify new SG dimensions.

Operating module 2: Measurement of new SG prior to final machining of nozzles:

- Confirm number of cuts per loop and define final machining for new SG;
- Final machining of new SG nozzles.

Operating module 3: Measurement of new SG after final machining of nozzles.

SGR outage operations:

Operating module 4: Measurement of cubicle when old SG is removed:

- Move pipe ends to fit-up position;
- Calculation of operating modules 3 and 4 — specify pipe fit-up position;
- Installation of reference rings, machine pipe ends, installation of new SG, weld pipes to nozzles.

Operating module 5: Measurement when primary pipe ends are adjusted at their fit-up:

- Calculation of operating modules 3 and 5 — specify pipe bevel to be machined.

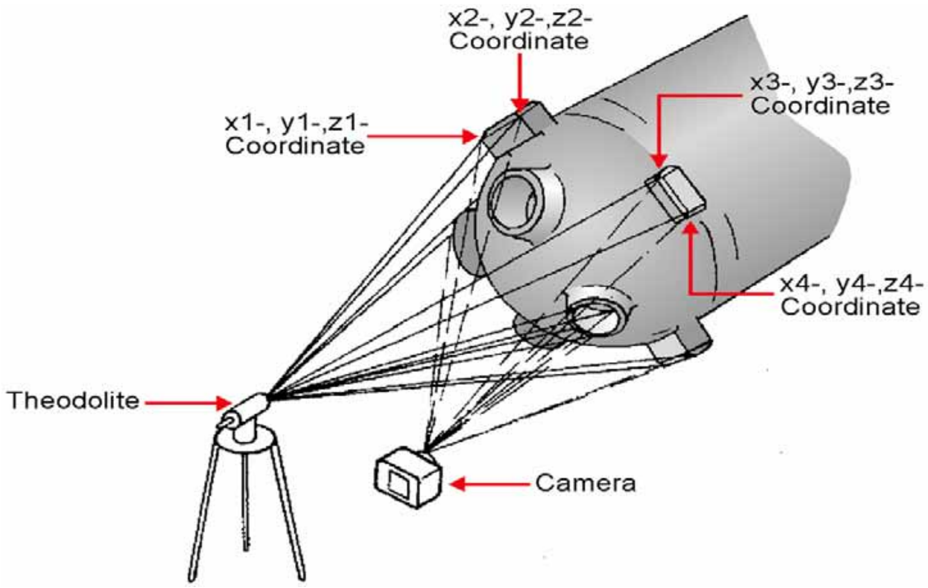


FIG. 15. As built measurement by optical survey for the new SGs with theodolite system and photogrammetry.

4.2.2. Cutting/machining

Specifically developed portable machines are used:

- For cutting old SG nozzles: This machine was designed with a cutting wheel to keep chips from entering the pipe.
- For machining of pipe elbows: At a position previously determined by optical measurement to obtain a fit-up tolerance of less than 1 mm (see Fig. 16).

4.2.3. Welding

The main target of the activities performed on the reactor coolant pipes is to achieve a proper fit-up of the new SGs to the reactor coolant pipes and to minimize the residual stresses after welding. This is achieved by using a special installation sequence and the GTA narrow gap welding technique.

The special configuration of the welding edge in connection with the equipment provides the following advantages:

- Smaller weld volume which resulted in reduced welding time.
- Remote controlled, TV monitored process contributed to a reduction of the accumulated radiation dose.
- Owing to the sequence of weld fabrication, lower residual stresses were produced in the welds (length allowance for shrinkage), thus residual stresses in the piping system were minimized.

4.2.4. Decontamination

The purpose of this process is to reduce the radiation dose in the area of reactor coolant pipe ends, and to achieve local cleanliness of pipe interiors there. Several processes are available on the market using chemical or mechanical approaches, for example, the mechanical approach is described as:

- Blasting by electrocorundum to remove the oxide layer, followed by blasting with glass beads to improve the superficial stress conditions and to smooth the surface. Use of a closed circuit system with subatmospheric pressure prevented abrasive particles and dust from escaping into the atmosphere (aerosol build-up is avoided) keeping radioactive waste build-up to a minimum.

FIG. 16. Pre-cutting of the reactor coolant piping.

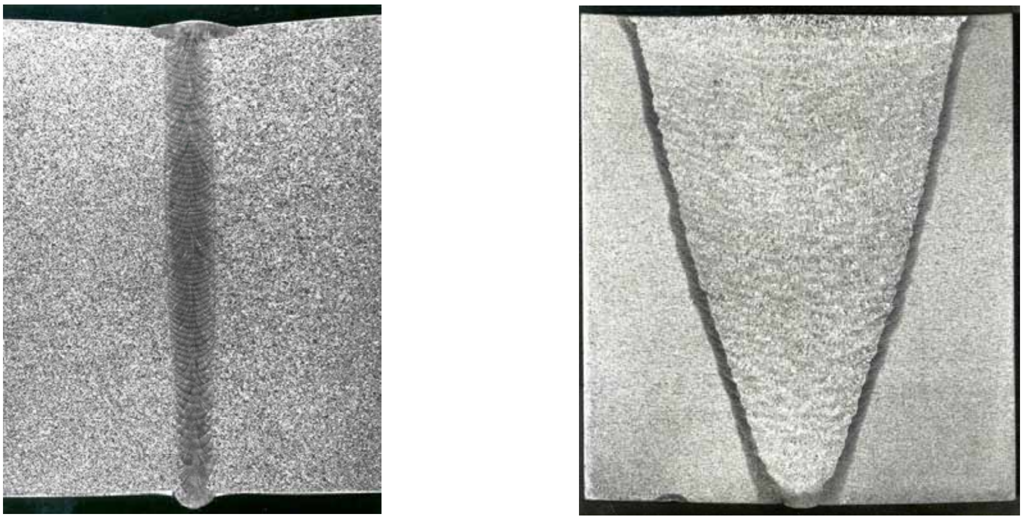


FIG. 17. Narrow gap weld versus conventional weld.

The main activities with regard to radiation during SGR are:

- Construction of scaffolding next to the primary loops;
- Removal of thermal insulation from primary loops and SG;
- Installation of shielding;
- Clamping of primary piping to hold it in place;
- Cutting of primary piping;
- Decontamination of remaining pipe ends;
- Machining the new weld lips on the pipe ends;
- Welding of new SG to pipe ends;
- General pipe works for auxiliary systems in the loop rooms;
- Cleaning activities;
- Health physics.

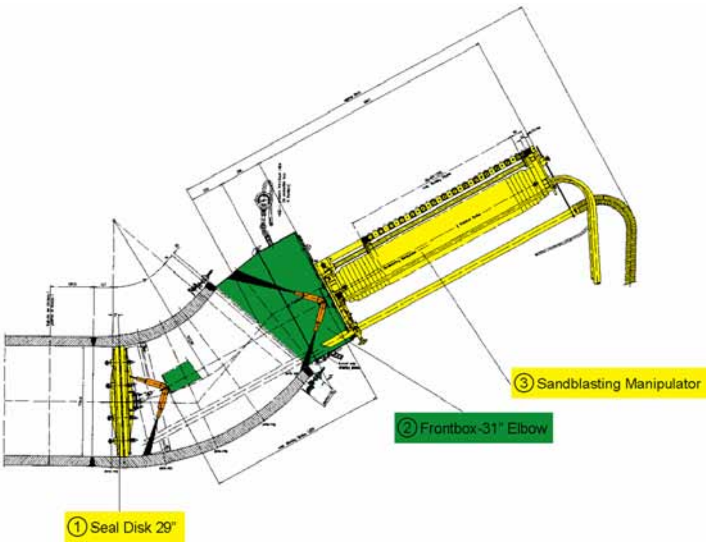


FIG. 18. Mechanical decontamination process. Assembly of decontamination equipment at the hot leg.

All in all, over a thousand single work activities had to be coordinated and the doses to more than 400 persons controlled daily. This was a difficult task which required close collaboration.

4.2.5. SG transportation and rigging

SG transportation can be challenging, depending on the locations of the component manufacturing facility and the utility's site. Delivery to some utility sites will require multiple modes of transportation, while others may be limited to just one or two. The types of transportation that may be required are:

- Ground transportation using self-powered motor transports;
- Ground transportation using rail;
- Ground transportation using large trucks;
- River transportation using barges;
- Ocean transportation using ships.

Each change in the mode of transportation will require development of specific rigging and handling plans designed to ensure protection of the replacement component. In addition, rigging and handling equipment designed specifically for these transfers will need to be provided. For overseas shipments, marine cargo inspectors will be required to inspect the shipping vessel and component rigging. Insurance providers may also require specific inspections prior to each portion of the shipment.

Shipping plans involving overseas or waterway transportation should also take into consideration:

- Droughts, flooding or freezing of waterways that could restrict shipping activities;
- Potential for hurricanes or typhoons;
- Impact on any endangered or protected species.

In many cases, utility sites no longer have facilities for receiving the components and either must reconstruct the facilities used during original site construction, build entirely new facilities, or arrange to have the components delivered to an alternative location. As part of the component supplier selection process, utility owners should consider the types of transportation required and the number of transfers that will be required.

For very large replacement components such as SGs, which require overland transport, specialized transporters or rail (Schnabel) cars would be needed. For example, the Palo Verde SGRs, shown in Fig. 19, were shipped from Genoa, Italy, by ocean vessel, offloaded in Mexico (Fig. 20) and shipped overland across the Mexican desert to the plant site near Phoenix, Arizona.

FIG. 19. Palo Verde Unit 2 SGR during transport.

Fig. 20. Palo Verde SGRs offloaded in Mexico.

Another option is transportation via rail, for those sites where a rail network is available, using a Schnabel car. The Duke Power Cherokee Unit 1 major components were delivered to the plant site in central North Carolina using a 36 axle Schnabel car equipped with the capability to shift the vessel both horizontally and vertically to negotiate curves and obstructions. While the Cherokee plant never became operational, the shipment of the reactor vessel and SGs by Schnabel car was successful and the 36 axle Schnabel car is still in use for transportation of heavy equipment.

Rigging and handling of the original and SGRs require specialized equipment and planning. These activities are usually divided into outside and inside containment rigging and handling. For outside containment, all transportation load paths involving movement of both the original and replacement components need to be identified and verified or modified to accept the required loads. This usually involves analysis of the components passing over some safety related system piping (Figs 21 and 22).

Depending on the type and location of the containment opening, specialized lifting equipment may be required to move the components in and out of containment. Issues that need to be considered in the design and installation of specialized lifting equipment include:

- Seismic qualification of equipment if built in proximity to safety related equipment:
  — Supports erection of equipment at N-1 outage or during power operations prior to the replacement outage.
- Interference of equipment with other site outage activities:
  — Outage support trailers;
  — Tool staging areas;
  — Equipment lay down areas;
  — Containment ingress/egress paths.

Examples of SGRs rigged into a containment opening on top of the building are TVA's Sequoyah and Watts Bar power plants. At these plants, a tower crane was used to raise and lower the SGRs through the opening and into the SG subcompartments. Figure 23 illustrates the tower crane outside lift system. Figure 24 shows an SGR being lowered through the containment opening in the metal liner.

Depending on the specific containment design, rigging inside containment can be handled in several ways, usually depending on the ability to utilize the girders of the containment building's polar crane. If use of the polar crane girders is allowed, then a heavy lifting device is usually placed on top of the girders in order to achieve the lifting height requirements to clear and bioshield walls. If use of polar crane girders is not allowed then a heavy lift tower crane will most likely have to be installed in the containment.

FIG. 21. SG rigging outside containment.

FIG. 22. SG rigging inside containment.

Issues to consider, especially if the containment polar crane is being utilized to lift the SGs, are:

- Loss of polar crane to perform other lifting activities.
- Need for additional cranes in containment to facilitate:
  - Efficient removal of all interferences;
  - Staging of tools and equipment;
  - Routine outage activities.

In order to utilize efficiently all containment cranes, a crane management plan is needed and the integrated project schedule needs to schedule crane resources to prevent any conflicts in use.

4.2.6. Secondary line connections

This section describes the recommended practices regarding the secondary lines when interfering with the SGR. Two types of lines are considered:

- The large diameter line, i.e. steam line and feedwater line;
- The auxiliary piping (mainly purge, water level, temperature monitoring).

FIG. 23. Outside lift system.

Fig. 24. Rigging SGR into containment building.

Steam and feedwater lines

These lines are made of high yield low carbon steel. The steam line is an 800–900 mm diameter pipe, and the feedwater line is a 400 mm (about) diameter pipe. Cutting the lines from the old SGs and reconnecting them to the new ones require highly skilled personnel and appropriate tooling. This work needs to be included in the scope of the experienced company in charge of the SGR.

The steam line is connected to the upper head of the SG, or laterally, and generally the connecting part is an elbow. To facilitate handling the old and new SGs, a long piece of pipe should be cut from the steam line, sometimes referred to as the 'closer'. Two options are then available:

- Reuse the old closer.
- Take the opportunity to install a new closer. This option is generally taken when it is wished to upgrade the plant, and/or apply new legal requirements (this may be the case when the old closer is made with rolled and bent welded plates).

New fabrications now give preference to forged pieces bent via induction. In the case where the use of new closers is elected, specific attention should be paid to the guaranteed yield, after bending, on all parts of the closers, to ensure that they are in accordance with the specifications. The new closers have to be hydraulically tested before erection (authorities may approve this test to be made after welding, in case the secondary circuit will be integrally tested before restarting the unit).

For both lines the proper scenario is as follows:

- Selection of the cutting plans.
- Installation of scaffolding allowing personnel to gain easy access to the working areas (especially for cutting, machining, welding and NDE tasks).
- Removal of the insulation and interfering supports.

- Fix-up of the line before cutting (note that for the feedwater line, fixing on its own supports may be sufficient, but the steam line requires the use of a specific additional fixing tool).
- Measurement of the line connection (topometry). This measurement requires the same type of tooling that is used for the SG installation and primary piping reconnection.
- Cut the line (in two places, in order to allow clearance for the SG handling).

Mechanical or thermal cutting process

A mechanical or thermal cutting process may be adopted. Thermal cutting is quicker, but should preferentially be considered only for the waste parts of the pipe, with special precautions taken to prevent burnt material falling into the existing piping. Nevertheless, in the case of piping reuse after thermal cutting, the heat affected zone has to be eliminated. It is necessary to:

- Adapt the design of the edge profiles (including bevels) to the SG nozzle thicknesses. This profile will also be adapted for post-replacement NDE and ISI.
- Machine the nozzles and pipe ends according to the designed profile and the recorded topometry.

The machining requires specific tooling. The main suppliers of SGR have developed this type of tooling. The procedure involves:

- Movement of the pipe ends to the selected position using the fix-up device.
- Installation the pipe spool pieces (closer).
- Release of the fix-up device.
- Preheating of the weld edges according to codes, in regard to the steel normalization, as per the welding procedure, which has been prepared accordingly and should be approved by the utility engineering department, as well as by the safety authorities and QA/QC representatives.
- Undertaking the welding. As much as the experience feedback is known, the welding is performed manually by qualified welders.

It should be noted that the thickness graduation from the SG nozzle to the pipe edge may require a large number of ‘finishing’ layers in order to allow a smooth external design and allow NDE.

In spite of the confidence of the welding companies regarding their process and welders’ capabilities, it may be beneficial to perform intermediary NDE (20, 50 and 75% of the weld thickness) within the welding procedure to avoid heavy repairs afterwards. The procedure involves:

- Performance of external and internal grinding to allow NDE (according to code requirements, this examination may require X ray, ultrasonic testing (UT) and/or PT);
- Performance of NDE of the welds;
- Repair defects exceeding the codes’ acceptability criteria (if relevant);
- Stress relief of the welds, with respect to the temperature/timing chart according to steel normalization, code requirements, and weld procedure;
- Reinstallation and fix-up of the supporting devices.

Hydraulic test

The steam line and the feedwater line need to be hydraulically tested to the test pressure defined in the applied code. The extent of the tested circuit and the test schedule needs to be agreed with the plant. It may include piping only, or in the case this test was not performed at the fabrication shop, the secondary part of the new SGs and the closers. This is specifically the case for the SGs that have been introduced in two parts in the reactor building, and had the final secondary envelope welded seam made in the reactor building. In that event, special care needs to be given to the primary tube bundle, so that it will not have to support an unacceptable external pressure. Sometimes a balancing internal water pressure may be necessary in the primary bundle.

Reinstallation of insulation

Depending on the requirements of the utility, and dose evaluation, insulation may be the old one or a new one. In this later case a new design may be adapted, and needs then to have been defined in the early phase of the project.

Auxiliary lines

This paragraph mainly applies to 25 mm or 50 mm carbon steel lines, dedicated to purge, water level control or temperature control. All lines connected to the SGs will be disconnected before retrieval and reinstallation of the SGs. The need is also to give sufficient clearance around the SGs to facilitate handling and installation of the replacement units. The recommendation is to proceed for each auxiliary line with a 'two cuts' process. The first cut is generally located at the SG corresponding nozzle, the second one at the nearest process equipment, which may be a valve, a sensor, a flow meter, etc. When the line is not directly connected to the SG (interfering lines), the cuts are located at process equipment limits.

The utility may also take the opportunity of the SGR to modify several lines, or the new SG may have modified nozzle locations. The cutting location will then integrate this requirement. The corresponding design and licensing will have been integrated in the design phase and proper documentation issued and approved before replacement. The early walk down dedicated to the SGR will verify the as built geometry of those lines and the local environment, in order to validate the proposed new design.

The overall scenario for each line is as follows:

- Removal of the insulation from the selected pipe length (from SG to first process equipment).
- Production of the as built 3-D line diagram for the considered pipe piece and verification that it is in agreement with the proposed new design of the line, when applicable.
- Removal of the supports from the considered pipe piece.
- Fix-up of the remaining existing pipe line with the 'neighbouring supports'.
- Cutting of the pipe piece at the selected locations, in the original socket welds.
- Adjustment of the neighbouring supports, in order to have the original line at the required position.
- Prefabrication of the new pipe pieces in a dedicated cold shop on-site, according to the above 3-D line diagram, or to the new design drawings, if applicable.
- Application of relevant codes and regulation (and have the required control and test procedures included in the welding procedure) for the new pipe piece acceptability (generally PT of the intermediary welds, if any, and hydro test).
- Preparation of the weld edges.
- Installation of the prefabricated pipe pieces between the new SG (after its installation and primary piping welding) and the original line edge.
- Release of the fix-up supports, allowing the original line to be free to move when welding.
- Welding at the two edges of the pipe piece. The required process is a manual socket welding, according to regulation, and described in the proper welding procedure, which will have already been prepared and qualified during the design phase.
- Carrying out the weld NDE, as required by applied regulation (generally PT).
- Installation of the supporting devices (original or new ones, if required by a new design of the line).
- Installation of the insulation. Except for new design, the original insulation may be reinstalled.

If required by the relevant codes and regulation, extra controls may be performed at the restart phase of the plant. This operation will be coordinated with the utility's operational personnel. Owing to multiple interfaces with the proper SGR operation, the auxiliary lines' disconnection/reconnection will be included in the SGR project and realized under its project management organization. However, the cutting/bevelling/welding/NDE operations may be handled by a specialized piping company, on a subcontract basis.

4.2.7. Scaffolding

Erection of scaffolding to support SGR activities is a critical path activity that needs to be well planned. Scaffold plans for SGR should be coordinated with other scaffold needs such as ISIs and outage maintenance activities. To the extent possible, scaffolding should be erected during the N-1 outage or at power just prior to the replacement outage. This usually will require seismic evaluation of the scaffolding, but this cost should be more than off-set by the schedule reduction for the replacement outage. In cases where scaffolding cannot be erected prior to the replacement outage, scaffolding should be stored in racks close to its intended location to reduce erection time.

4.2.8. Instrumentation location changes

Relocation of instrumentation during SGR activities can result in several issues including:

- Analysis and modifications to support systems;
- Instrument set point changes;
- Revisions of operating and maintenance procedures;
- Changes in operating characteristics.

4.2.9. Thermal insulation replacement

Replacement of thermal insulation as part of an SGR project is recommended. Thermal insulation can either be removed prior to the original SG being removed from containment or can remain on the original SG during removal from containment. There are advantages and disadvantages to each approach that need to be considered on a site specific basis:

- Depending on the level of loose surface contamination on or underneath the insulation, contamination levels in containment could be increased during removal activities.
- Leaving insulation installed during removal and replacement activities could increase the interference removal scope and the size of the containment opening.
- Condition of the original insulation could make it impractical to remove the SGs with the insulation installed.

4.2.10. Utility site preparation

Preparing the utility site for an SGR project represents a major part of the overall project (see Fig. 25). The following issues will need to be addressed as part of this process:

- Receiving and unloading facilities: Many utility sites no longer have facilities to handle the receipt and unloading of the SGRs. For these sites the original site construction unloading facility can be reconstructed, a new unloading facility can be constructed or a remote unloading facility can be utilized.
- Component preparation facilities: Following arrival to the site the SGRs need to be placed in a facility that supports the needs of installation preparations. These preparations typically include:
  - Nozzle end preparations;
  - Pre-service eddy current inspections;
  - Electropolishing;
  - Removal of protective coatings.

For sites that plan on storing the original SGs in mausoleums, this building is sometimes used as the preparation facility. Arrangements for providing electricity, water and air usually have to be made since the mausoleum does not usually contain these services.

Other sites typically erect a temporary facility in the form of a tent structure or a sheet metal building. For such temporary facilities soil loading and the need for electricity, water and air services should be considered.

FIG. 25. Typical SG preparation facilities.

Regardless of the facility type, provisions for foreign material and dust control should be in place prior to opening any covers.

Warehouse facilities:

- The large amount of materials and consumables that will be required to support installation of the SGRs can easily overburden the utility sites' normal receiving and warehouse facilities. For this reason, consideration should be given to construction of an alternative receiving and warehouse facility specifically in support of the SGR project.

Craft assembly facilities:

- At the peak of the SGR project, 500–800 craft personnel can be on-site. Facilities to support briefing areas, break rooms, radiation protection dosimetry and containment dress out and undress areas will need to be provided. Such facilities should be designed to eliminate chock points for the flow of craft personnel to and from their work locations.

Management and engineering offices:

- At the peak of the SGR project, 200 management and engineering personnel can be on-site. Offices and work locations to support this number of personnel should be provided. These facilities should be arranged in functional areas such as:
  — Management;
  — Engineering;
  — Document control;
  — Scheduling;
  — Quality control;

— NDE;
— Radiation protection;
— Deconners.

Cafeteria and restroom facilities:

- At the peak of the SGR project, 800–1000 additional personnel can be on the utility's site. Such a temporary increase in personnel will require additional cafeteria and restroom facilities. Depending on the site's infrastructure, these additional facilities could require significant modifications to the site's potable water and sewage treatment systems.

Mausoleum for contaminated components:

- For utilities that elect to store the original SGs on-site, construction of a mausoleum will be required. This mausoleum should be constructed prior to commencement of the SGR outage in order to minimize the site's craft requirements and allow the mausoleum to function as an SGR preparation facility.

Parking and traffic control:

- At the peak of the replacement outage an additional 1000 personnel could be on-site. The capability of the site to handle traffic control and parking for such a large number of personnel is critical to efficient use of resources. It is common for sites to work with local law enforcement agencies to help control the flow of traffic, especially at shift changes. It is also common for the site to construct additional parking facilities and, depending on the location of such facilities, arrange for shuttle services between the parking facility and the site entrance facility.

Site lay down planning:

- Depending on site specific needs it is not uncommon for more than 100 semi-truck loads of equipment to be delivered to the site. Such a large amount of equipment will require lay down space and planning. The lay down areas should have adequate soil conditions to permit access to the equipment in inclement weather.

4.2.11. Fit-up and alignment of the reactor coolant lines

Prior to moving the SGRs into position for welding, the locations of the cut RCS pipes will be measured using laser templating equipment. Based on these measurements, the pipe ends will be machined to fit-up to the nozzle safe ends on each SGR. The as built replacement SG primary nozzle safe end positions are measured after fabrication or alternatively after cutting off the nozzle caps, which were welded on for hydrostatic testing and shipping. The process for measuring the position of the primary piping nozzle safe ends on the SGRs was discussed previously in Section 4.2.1.

It is common practice to manufacture the nozzle safe ends with extra stock for attaching the hydrostatic test nozzle caps and for fit-up adjustment. Once the position of the cut ends of the RCS pipes is determined, a 'best fit' cut line can be established and final machining of both the RCS piping and the primary nozzle safe ends can be performed. If there is significant cold spring in any of the primary coolant pipes after cutting, the cut end of the pipe might need to be temporarily jacked into position before the welding fit-up can be accomplished. If primary coolant piping cold spring is anticipated, the pipe could be blocked against movement before the pipe cut is made.

Each SGR can then be rigged into final position, aligned with the RCS pipes and fit-up for welding. Before welding, SGR alignment with the vessel support system should be verified. It should be noted that alignment of vessel supports and primary coolant pipe connections could be simplified by the planned use of support shims, which could be added or removed as necessary in order to facilitate primary coolant pipe fit-up. Typically, several shim thicknesses are figured into the vessel design, so that required adjustments during fit-up are simplified. After final fit-up is achieved, the welding operation can commence. After initial primary coolant pipe welding is performed, reinstallation of the main steam and feedwater piping can be performed in parallel with the completion of the primary coolant pipe welding.

4.3. CIVIL WORKS

The main tasks of civil works for SGR (see Figs 26–29) are:

- Specific opening/closure of the containment;
- Construction of the site storage of the old components;
- Site transportation roads capabilities verification and/or adaptation;

FIGs 26–29. Containment opening and handling/rigging.

- Cubicle civil work;
- Dock facilities construction.

4.3.1. Opening/closure of the reactor building containment

The opening and closure should be adapted according to the building containment structure, which may be a self-standing steel shell with a separate reinforced concrete shield building or pre-stressed concrete with steel liner. The opening is required and developed when:

- SGs cannot go through the equipment hatch either in one or two pieces;
- Pre-stressing cables are not directly embedded in concrete.

Good practices based on the several operations have already been performed in Belgium, Sweden and the USA as follows:

- Qualified engineering division has to justify the containment opening and give operational design and instructions to:
  — Size and shape the opening in accordance with the rigging requirements;
  — Cut steel reinforcement bars;
  — Retrieve the tension cables;
  — Cut the tendon tubes;
  — Eliminate the concrete concerned;
  — Cut the reactor building liner plate for possible reinstallation later;
  — Retrieve, handle and store the cut liner plate;
  — Protect and secure the hatch;
  — Reinstall and weld the liner plate;
  — Reinstall and weld reinforcement bars and tendon tubes;
  — Redeploy the tension cables;
  — Fill up the cavity with concrete (composition to be defined and controlled) after concrete hardening;
  — Place adequate tension on the cables and fix up the extremities.
- Careful selection of a company for tendon work.
- Careful selection of a company for the concrete elimination:
  — Several processes may be applied, but mechanical and hydro pressure are the preferred ones.
  — Special care should be taken not to damage the building liner plate.
- Integrate the opening and closing works with the outage schedule, as some of the included tasks are on the critical path of the outage (concrete hardening, etc.).

4.3.2. Multipurpose building and storage facility for old components

Conventional civil work companies manage these tasks to build a multipurpose building and storage facility for old components under their responsibility. The construction works should be completed before the SGR outage (see Section 11.1).

4.3.3. Site transportation and dock facilities

For transportation of old and new SGs, the following points should be considered:

- Actual site road capacities, taking into consideration underground piping;
- Clearances of the pass ways;
- Modification of existing buildings/stores as well as road reinforcements.

The schedule should be clearly established and controlled, in order that no additional civil work should be done during the outage (Figs 30–34).

FIGs 30–33. Dock facilities at site for moving heavy components.

4.3.4. Cubicle civil work

Although it is preferred not to have any civil work dealing with the SG cubicle, the worldwide experience feedback noted that SG handling may sometimes lead to cutting part of the cubicle concrete and rebuilding it after the new SG installation.

Generally, the upper part of the cubicle is the obstacle to the SG's free pass out from the cubicle. Two countermeasures are available:

- Increase the free lift of the polar crane. This is currently managed to a specific trolley on the polar crane beams and hydraulic jacks, allowing an extra lift of the SG between the polar beams.
- Evacuate and introduce the SG in two parts (final weld being made in the cubicles). This solution was applied for the Fessenheim and Bugey SGRs in France.

If no alternative solution is available, the 3-D handling simulation provides the data for the cubicle's cutting. Major recommendations for this work are to:

- Develop a specific ALARA approach to manage (reduce) the dose;
- Avoid any destructive process which may spread the concrete dust;
- Provide a dust evacuation system (vacuum);
- Optimize the reconstruction duration (prefabricated elements);
- Carefully integrate the cubicle civil work tasks in the outage schedule.

4.4. SGR FOR ALARA ASPECTS

This section is going to consider radiologically related parameters important for SGR implementation. These parameters are identified as ALARA related parameters, owner–contractor relationship, planning, health physics with logistic services, and time required for the replacement. ALARA related parameters such as source or initial dose rate and plant system configuration define the initial conditions for the planning.

Generally, plant decisions on maintenance or repair procedures under radiation conditions take into account ALARA considerations. However, in the main it is difficult to judge the results of an ALARA study, usually in the form of a collective dose estimate, because a comparison standard is missing. That is, very often the planned work is of a one-off nature so comparisons are not possible or the scopes are not the same. In such a case, the collective doses for other types of work are looked at and a qualitative evaluation is made. In the case of an SGR this is not the case. Over years of SGRs conducted worldwide, a standard has gradually been developed.

4.4.1. Assessment of management procedures

There are three important factors that control the collective dose but which have little to do with direct ALARA management procedures. These are:

- Primary circuit dose rate;
- Number of SGs;
- Amount of shielding.

The last item can be misunderstood. This is indeed a very important ALARA measure. However, ALARA management procedures are required only to install and to remove it. In other words, a great deal of lead reduces the need for management. Once it is installed its very material mass influences the collective dose but this has nothing to do with management procedures.

In order to make an ALARA assessment of managerial procedures, these three factors have to be filtered out of the collective dose to enable proper ALARA management comparison. To define a quantity called figure of merit (FOM), a method related to Monte Carlo calculations is used. The FOM also becomes an artificial factor which includes the three previously named factors independent of ALARA managerial work procedures. The resultant dose is then weighted with the FOM to produce a new weighted dose, a dose FOM or D_{FOM}, which reflects more the managerial ALARA aspects rather than physical conditions over which no control has been exerted. To make this more clear:

$$D_{FOM} = \frac{\text{Total collective dose} \quad \text{Amount of lead}}{\text{Number of SG} \quad \text{Primary dose rate}} = \text{Total collective dose} \times \text{FOM}$$

4.4.2. SGR ALARA review

Unlike other SGR projects, the one for Krosk nuclear power plant had an extremely short replacement time. This was 29 days to operational delivery. Normally SGR projects, depending on scope, have replacement times of the order of 40 days. This short SGR time had negative effects on ALARA towards the project end. This indicates, too, that further efforts to reduce replacement times could lead to higher doses. An example of this is if more activities have to be performed in parallel towards the end of the SGR. At this stage shielding is being removed prior to handover. Thus more dose is incurred than would be case with shielding still in place. One boundary condition for SGR was the primary circuit dose rate, which was very high. In the unshielded state the contact dose rate was 3.4 mSv/h and 8.6 mSv/h when drained.

The collective dose estimate itself requires large amounts of input for personnel and dose rates. These data are weighted by occupancy factors, which are based on experience, and decide the actual time spent in radiation fields. SGR doses were analysed according to job codes used in performing the jobs. Because exact planning of work is not always possible and also because corrective actions cannot be predicted, then an estimate accuracy in a band of around ±10% can be regarded as exact and around ±20% as acceptable.

4.4.3. Some examples of good ALARA practice

Experienced personnel should be engaged for the shielding. Scaffolding work should be speedily carried out to avoid incurring a dose. This will involve smaller personnel needs than that expected. A trained team with good advance planning can reduce the required scaffolds.

To prevent cross-contamination of large items, special clean areas are needed in which no shoe covers are required. As radioactive items are removed, this practice will be extended by issuing instructions in the form of ALARA cleaning regulations. Thus these clean areas are extended to the operation deck in the containment and then to the auxiliary building.

Dose monitoring is needed on a daily basis in order to compare actual doses with the predictions. This will enable timely intervention to modify procedures if it is shown that conditions will change or produce unexpected doses if the work is continued unmodified.

5. RVHR IN PWRs

5.1. STRATEGIES FOR RVHR

5.1.1. RVHR history in PWRs

Head replacement history

In 1991, a leakage in one CRDM head penetration (alloy 600) was discovered during the hydro test at the Bugey power plant in France. This was the first instance of a leakage caused by PWSCC in the world. The longitudinal cracks were propagated from inside the penetration and leakages have also been found by visual inspection in the reactor vessel heads in reactors in Japan and the USA. Some leakages were produced from the crack in the J-weld which propagated to the outer surface of the head penetration. The cracks were caused by PWSCC with high residual stress by the J-weld. In 1994, the decision was made in France to replace all vessel heads of 900 MW(e) and 1300 MW(e) units. Subsequently, replacement of vessel heads started in Japan and the USA (see Appendix 2 for a list of RVHRs) (see Fig. 34).

5.1.2. Scope of RVHR

To prevent cracks in the J-weld and head penetration, reactor vessel head and head penetrations should be replaced as a minimum scope. The following scopes for RVHR will be decided with regard to reuse or replacement:

- CRDM including pressure housings and latch mechanisms;
- Coil stacks;
- Control rod position indicator;
- Upgrading packages.

The CRDM may be with or without canopy seals. In the case of reusing the CRDM, the weld between the CRDM housing and the head penetration flange will be performed at site. To reuse the existing CRDM for the reactor vessel head, special care should be taken regarding storage, cleaning and re-machining (canopy).

If a new CRDM is applied for replacement of the reactor vessel head, the following advantages are expected:

- No extra time for machining and cleaning the old one;
- Better knowledge of the chemistry of the canopy seal;

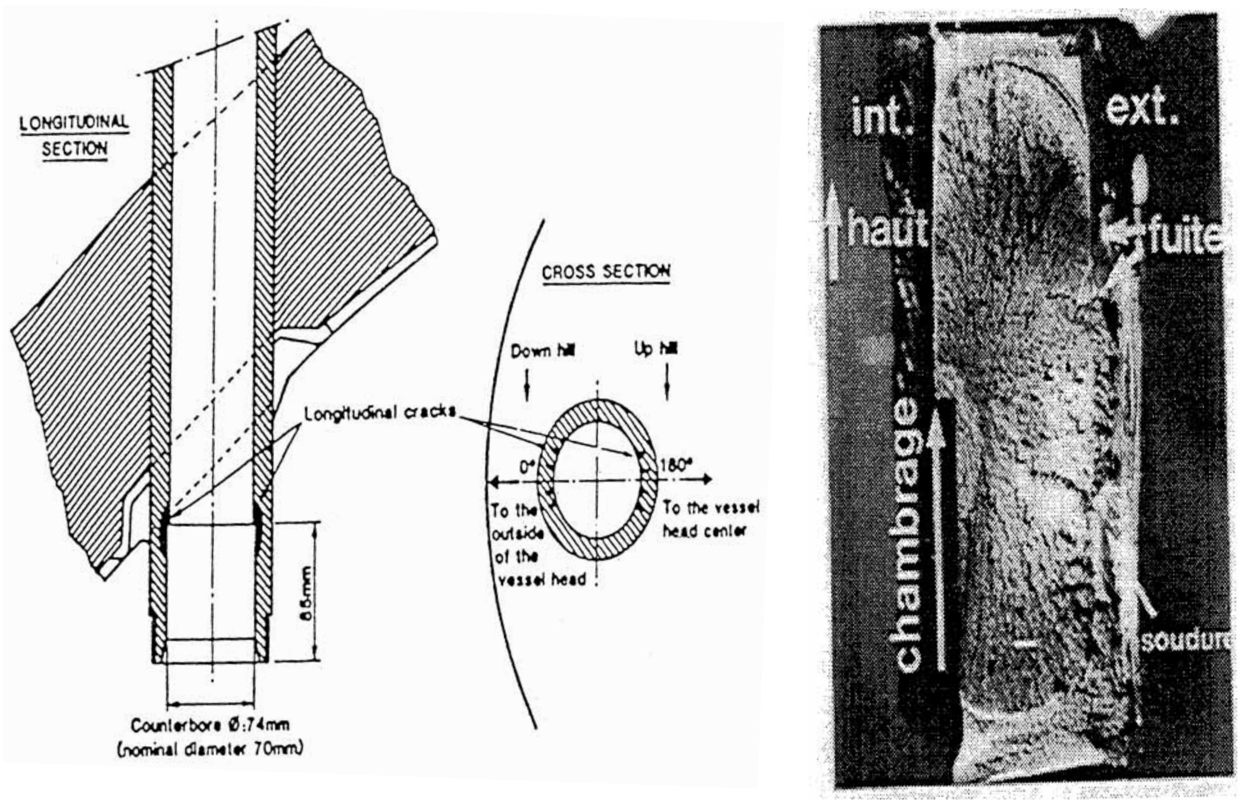


FIG. 34. Longitudinal cracks propagated from inside the penetration.

- Less handling and operating time for the old CRDM resulting in better radiological aspects.

Most related scope of disassembly/reassembly work to the head is as follows:

- Remove/prepare/reinstall CRDM;
- Remove/replace service structure;
- Remove/reinstall/test coil stacks and rod position indicators;
- Scaffolding (planning is key);
- Encapsulate/package for shipment.

5.2. APPLICABILITY OF CANOPY-LESS CRDM HOUSING (BUTT WELD TYPE HOUSING)

The use of new CRDM designs (Butt weld type housing) can eliminate the canopy seal weld for more reliability, and therefore eliminate the risk of leakage. At the same time, the new design has the advantage of an integrated head assembly.

5.3. REDUCTION OF OUTAGE PERIOD WITH PACKAGE IN/PACKAGE OUT

An integrated head assembly with a package in/package out concept was designed and applied to reduce the outage period. The package in/package out means the whole replacement of the reactor vessel head and CRDM with insulation and without dismantling. Regardless of whole replacement or not, the following items will be considered (see Fig. 35):

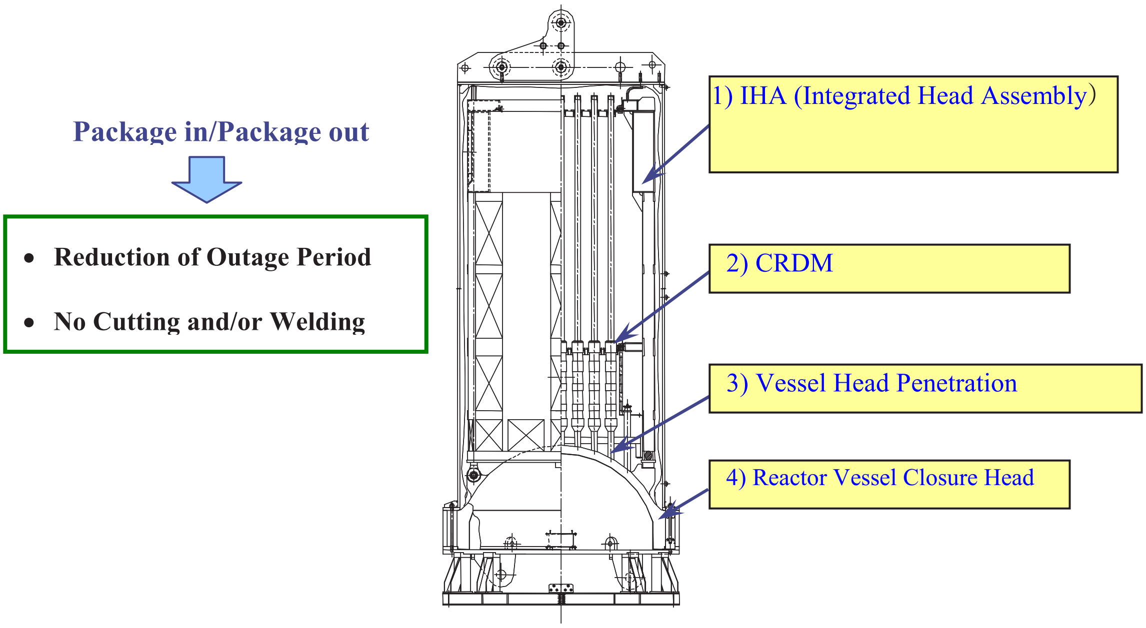


FIG. 35. Integrated head package.

- Limitation for size of assembly with consideration of the transportation route;
- Crane capacity and accessibility;
- Reuse of items with due consideration of cost;
- Supplier’s scope.

5.4. SEQUENCE OF RVHR

The major stages of RVHR comprise the following:

- Material procurement;
- Fabrication;
- Installation (including removal and storage of existing head).

Before the fabrication stage, the as built dimension should be considered. For a good fitting between the new head and the existing vessel flange, the field measurement of the interface dimension at flange may be taken before machining of the new head, in addition to confirmation of the as built drawing.

Design, calculation and licensing activities (without thermal sleeve)

The configuration of the new reactor vessel head is to be designed with the same concept as that used for the existing reactor vessel head except for the following:

- Material change to alloy 690 for head penetration and J-weld material;
- Possible change of vent line routing;
- Possible elimination of CRDM canopy seal in the case of replacing the CRDM.

5.5. OUTLINE OF RVHR

For the whole replacement of the reactor vessel head with new a CRDM, excluding integrated head assembly, the typical sequence is as follows:

- Preparing auxiliary handling system.
- Removing existing vessel head with head structure from reactor pressure vessel.
- Placing existing vessel head on the cover plate or stand.
- Removing head structure from existing head to disassembly area.
- Bringing new head with new CRDM into containment vessel through equipment hatch. The new head with CRDM is covered by transportation cask.
- Upending the new head with CRDM covered by transportation cask.
- Lifting and removing the cask from the new head with CRDM.
- Putting the transportation cask on to existing head.
- Laying down the existing head with CRDM covered by transportation cask.
- Carrying out the existing head with CRDM from containment vessel through equipment hatch.
- Assembling the head structure on to the new head including welding of vent line and thermocouple connecting, assembling coil stacks and/or RPI's if they are reused.
- Installing the new head on to reactor vessel after outage activities inside of reactor vessel.
- Hydro/leak test according to code.

5.6. LESSONS LEARNED AND CHALLENGES

Before starting the project, the lessons learned from other projects should be reviewed in order to prevent non-conforming items and activities. The following items are typical examples of lessons learned:

- Strong ALARA and QA members on team;
- Strong welding engineer on team;
- Station personnel involvement on team;
- Establish procedure and other engineering document approval process;
- Keep very close track of engineering progress (especially of outside organization);
- Single point of responsibility for all areas;
- Decide what activities will be supported by the station and arrange for support;
- Project manual to describe responsibilities, interfaces and general project processes;
- Project personnel must manage project, not station or corporate, or contractors.

6. RVI REPLACEMENT IN PWRs

6.1. STRATEGIES FOR RVI REPLACEMENT IN PWRs

6.1.1. RVI replacement history

Until 2004, there was no report on replacement of whole RVI, an upper internal and a lower internal together, although an upper internal had been replaced at Prairie Island in 1986. Since 2005, replacements of whole RVI have been completed at several PWRs in Japan (see Table 1).

Objectives of those replacements for the two-loop plants designed and manufactured during early generation of PWR plant construction in the 1970s are as follows:

TABLE 1. RVI REPLACEMENTS IN PWRs IN JAPAN

Unit	Supplier	Replacement
IKATA 1	Mitsubishi Heavy Industry	2005
GENKAI 1	Mitsubishi Heavy Industry	2005
IKATA 2	Mitsubishi Heavy Industry	2006

- Applying proactive and preventive countermeasures against the potential ageing degradation of irradiation assisted SCC on baffle former bolts.
- Adding four more guide tubes (drive lines) to the upper internal to keep enough shutdown margins in preparation for applying high burnup fuels.

6.1.2. Scope of RVI replacement

Getting feedback from operational experiences

Through plant operations in the world since the 1970s, the RVI has suffered from several ageing degradations, such as irradiated assisted SCC on baffle former bolts, PWSCC on support pins and flexure pins of guide tubes, wear on locking devices of support pins, wear on thimble tubes, and so on. In general, design modifications, including alternative material selection, are required as countermeasures against those experienced issues.

In the case of individual maintenance approaches that have been usually taken up to now, it is usual to study what design modifications can be made directly on the failed parts in developing countermeasures. However, there is a limitation on flexibility of applicable design modifications because of the necessity for keeping compatibilities between the failed parts and other related existing parts. In contrast, there is wide flexibility to apply much better design modifications in the case of whole RVI replacement, and all design modifications are easily incorporated into the new RVI.

Improving plant performance

Whole RVI replacement has the potential to improve plant performance. For example, a three-loop plant, converting fuel assembly types from 15 × 15 fuel rods to 17 × 17 fuel rods results in power uprating, and it is necessary to equip 17 type guide tubes to the upper internal. Also, in a two-loop plant, applying high burnup fuels results in improving fuel cycle economics, and it is necessary to add four more drive lines to the upper internal in the case of Japanese PWRs. Although such large scale structural modifications would be required with those aims, it would be easily applicable to the new RVI in the case of whole RVI replacement.

Keeping compatibilities with existing components

The RVI has many interfaces with the reactor vessel; irradiation specimens; fuels; non-fuel-bearing components such as rod cluster controls, plugging devices, etc.; and thimble tubes. The new RVI has to be designed with full compatibility with those existing components.

Removal of the original RVI

An original RVI, in particular the lower internal, is strongly radioactive compared with other heavy components in PWRs such as SGs and reactor vessel heads. For example, a two-loop plant operated for more than 20 years in Japan has a radiation dose rate at the outer surface onto RVI estimated to be about 2×10^5 mSv/h. So, it is critically important how to deal with not only removing the original RVI but also reducing radiation exposure to workers during removal operations.

Even though decontamination is the conventional method of reducing radiation exposure during such heavy component removal operations, it would be less effective in the case of a PWR's RVI replacement because activated products such as ^{60}Co are generated inside materials of the RVI and those products cannot be removed by decontamination. The effective measure is to shield workers from the original RVI by using a proper cask.

In the case of RVI replacements in Japan, the original RVI was first stored in the special cask with enough shield capacity against radiation exposure from the inside. After that, the cask containing the original RVI was handled during removal operations without extreme radiation exposure of workers outside of the cask.

Installation of the replacement RVI

Installation procedures for the replacement RVI are basically common with those at plant construction. However, the RVI has close interfaces to the reactor vessel, for example, in the two-loop plants in Japan, the designed gap between the RVI and the reactor vessel is 1.4 mm at the outlet nozzle and 0.4 mm at the crevice insert, respectively.

Therefore, the new RVI has to be installed and adjusted with high precision to the existing reactor vessel. Even though it was feasible to install and adjust the RVI to the reactor vessel based on the manually measured gap at the interface parts during plant construction, it is not feasible to apply the same procedures in the case of the RVI replacement, because of difficulty in accessing the existing reactor vessel due to high radiation exposure. For installing the new RVI, it is necessary to measure the gaps at the outlet nozzles and crevice inserts underwater by using remotely operated measurement systems.

6.2. OUTLINE OF RVI REPLACEMENT

Procedures for RVI replacement are mainly categorized into two stages:

- First stage is to remove the original RVI from the existing reactor vessel;
- Second stage is to install the new RVI to the existing reactor vessel.

6.2.1. Removal of the original RVI

Typical procedures for removing the original RVI are as follows (see Fig. 36):

- Design the special cask with enough shield capacity, which can contain the original RVI, reduce radiation exposure to workers outside the cask and fit for handling procedures during all removal operations.
- Install the cask and temporary crane into a containment vessel.
- Remove the original RVI from the existing reactor vessel and store them in the cask.
- Hoist up the cask containing the original RVI and seal it.
- Bring out the cask through an equipment hatch.
- Transport the cask and store it in a storage building.

6.2.2. Installation of the new RVI

Typical procedures for installing the new RVI are as follows:

- Bring the new RVI into the containment vessel through the equipment hatch as a reverse operation for removing the original RVI.
- Finalize the assemblies of both the upper internal and the lower internal inside the containment vessel.
- Install and adjust the new RVI to the existing reactor vessel with gap measurements made underwater at interface parts between the RVI and the reactor vessel.

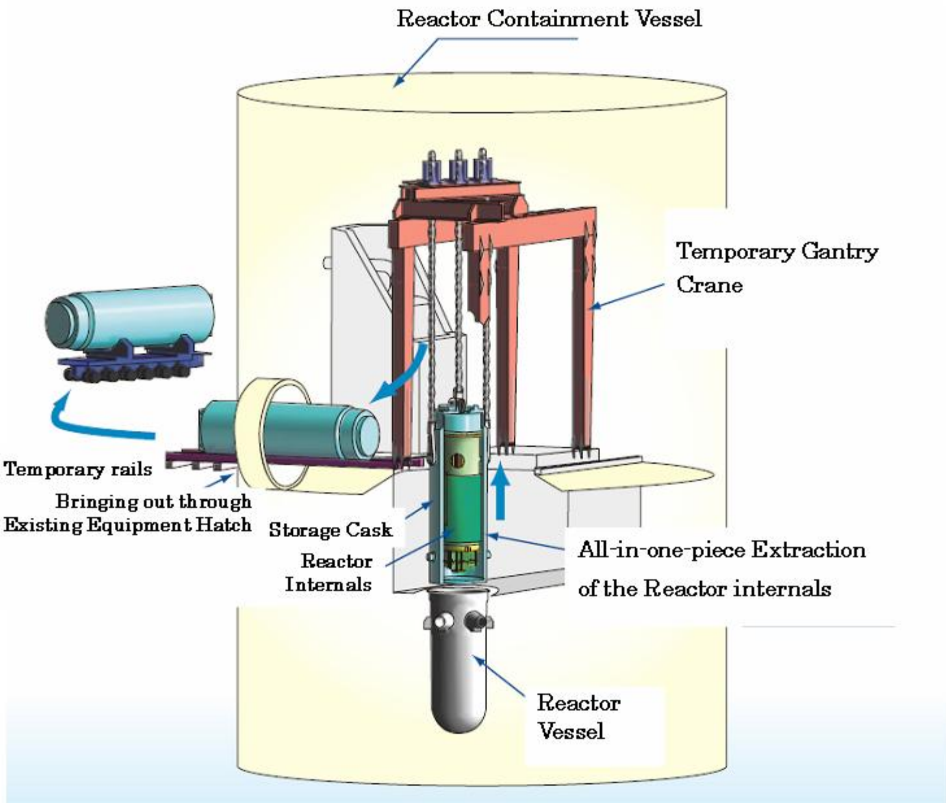


FIG. 36. Outline for removing the original RVI.

7. PRESSURIZER REPLACEMENT IN PWRs

Some pressurizer designs have required a high level of maintenance due to PWSCC of the alloy 600 which is used for heater sleeves as well as for instrumentation nozzles. Alloy 600 material is especially susceptible to PWSCC at the temperatures in the pressurizer, which run hotter than the reactor vessel or the SG. Upper piping dissimilar metal welds made of A182 for safety lines and relief lines are also very sensitive to PWSCC. It is also applicable to the surge line connection of the pressurizer. High residual stresses resulting from weld shrinkage of the small diameter heater sleeves also contribute to PWSCC susceptibility. Replacement pressurizers employ either austenitic stainless steel or alloy 690/52/152 material for heater sleeves and weld process controls are implemented to reduce the amount of weld shrinkage during fabrication. New dissimilar metal welds are made of A52/152 material more resistant to PWSCC.

7.1. PRESSURIZER ASSET MANAGEMENT STRATEGY

As discussed above, the pressurizer component is subject to PWSCC. As such, several repair options exist for remedy or mitigation of the potential as well as known degraded locations. Additionally, pressurizer replacement, similar to SGR, exists as an option for resolution of this issue. Decisions for repair/replacement generally evolve around the economics and technical aspects of repair versus replacement. Four Combustion Engineering design pressurizers have been replaced to date within the USA. In Europe, they are planned with the procurement phase.

In US plant experience, plants with less than 40 heater sleeves have consistently been selected to repair or mitigate by either welded pad half nozzle repair or inner diameter half nozzle weld repair. In either approach the pressure boundary is relocated and re-established with A52/152 material. The remaining small bore instrument nozzles are also repaired using half nozzle repairs. Large bore dissimilar metal welds are normally mitigated using mechanical stress improvement or by structural weld overlay.

For those US plants with pressurizers containing more than 40 heater sleeves, replacement has been the chosen approach. This is based upon cost savings for reduced outage duration compared to repair/mitigation of many sleeves. This comparison also assumes that a dedicated containment opening does not have to be constructed for the replacement. Often, replacements are timed to be concurrent with other major component replacements such as reactor vessel heads and SGs. This provides a substantial reduction to overall outage impact and optimizes the typical speciality contract resources necessary for rigging, radiological controls, special welding processes, etc.

7.2. PROJECT ORGANIZATION AND COMPLEXITY

Organizational and financial considerations for pressurizer replacement are consistent with those discussed in Section 3 for organization of heavy component replacement. Although the scale of a pressurizer replacement project is substantially less in financial terms for design and fabrication, the team structure is very consistent for all phases of the project.

Replacement pressurizer project complexity is increased with effects such as change in component volume or design, plant power uprate, latent design concern resolution as well as installation issues such as surge line residual force management, pressurizer containment structure issues, reactor building crane limitations or necessity to create reactor building openings to support replacement. Those stations that have managed their original components such that pressurizer replacements can be planned in conjunction with those of SG and reactor head are best able to optimize cost and organizational structure as well as leverage limited fabrication slots based upon volume. Pressurizer project team personnel utilization can be optimized in speciality areas such as safety analysis, welding, rigging, design, licensing, etc., when timed to occur with other major component replacements.

7.3. APPLICATION OF LESSONS LEARNED

US plants have been able to capitalize on substantial lessons learned in major component replacement due to significant practical experience combined with INPO/WANO world operating experience and a maturing model for self-assessment and sharing of experience. The following sections summarize key opportunities and lessons learned for pressurizer replacement.

7.3.1. Pressurizer design and fabrication

- Design and fabrication organizational structure and human resource compliment are consistent with SGs and reactor vessel head.
- 3-D modelling of original and SGR is essential to validate like for like replacement physical characteristics and the capability to interface flanges, piping supports, skirt limiting retrofit design scope and emergent issues.
- Oversight of vendor fabrication by engineering and QA throughout the process are critical to preclude rework at the end of the fabrication process, having direct impact on equipment shipping (see Figs 37, 38).

7.3.2. Insulation replacement

- Metal reflective insulation design and installation requires extensive experience. Shortcomings in this area have direct impact on dose and schedule.

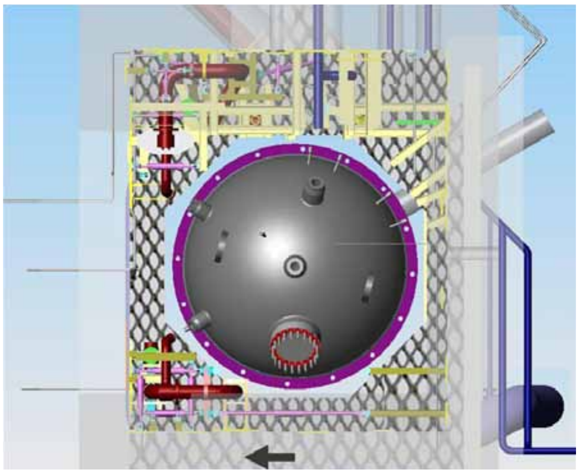


FIG. 37. 3-D modelling of top of pressurizer.

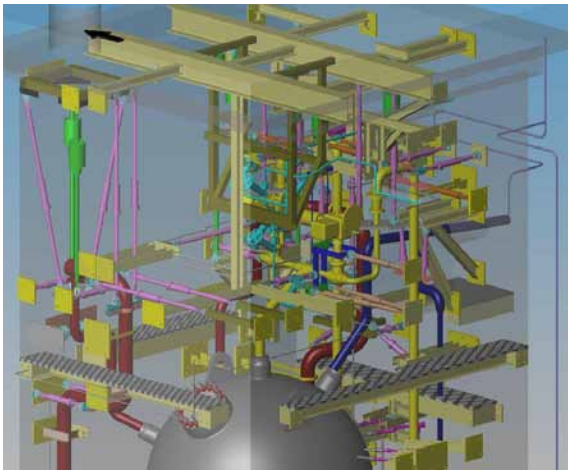


FIG. 38. 3-D modelling of piping and structure of pressurizer.

- Assuming economic viability, new insulation should be procured and verified to fit the pre-outage schedule to reduce outage dose and schedule impact for insulation modifications.
- New insulation is designed in smaller panels for ease of installation and removal.
- Insulation removal has been sequenced to maximize efficiency.
- If old pressurizer insulation is reused for new pressurizers, evaluate the capability to reduce outage time and dose and reduce in-process insulation modifications. As a minimum, ensure old insulation fits new vessel during pre-outage phase.

7.3.3. Piping

- Work off the top pressurizer flanged connections when possible for critical measurements (3-D) and replace nozzle/piping inclusive of the flanges to eliminate field welds.
- Perform integrated design and field engineering walk downs to identify pipe cut locations such that designs are accurate, work package planning is correct and field changes that cause design/analytical rework are minimized.
- Identify and procure replacement elbows if the original installation used spliced and mitred elbows to obtain fit-up. These fit-ups are high risk as regards rework, with replacement being a good option owing to alignment difficulty, oval shaped pipe, etc.
- Include the surge line nozzle dissimilar metal butt weld in the fabricator scope with a random extension for the surge line such that the final weld can be finalized utilizing 3-D input and support a non-dissimilar metal field weld.
- Ensure radiologically ‘hot’ piping locations are identified in the N-1 outage where cut locations can be pre-identified and prefabricated pipe sections can be built to replace these high dose locations.
- Utilize automated machine welding when adaptable to reduce dose and rework potential.
- Pre-identify mitred and short pipe joints at elbows for purpose of managing machining operations. Machining marks can result and if not cleaned up prior to fit-up and welding post-weld NDE acceptance may be impacted and rework may be required on acceptable welds just to resolve indications within the area of concern.
- For the surge line an engineered approach is critical to stabilize the piping for residual forces during cutting operations to ensure post-replacement alignment and fit-up.
- Use seal welded or bolted blind flanges on the old pressurizer openings for closure and contamination control.
- Ensure the pipe section removed on surge line for this purpose is accounted for in manufacturing the new pressurizer.

7.3.4. Snubbers/support/electric

- An experienced snubber team should be utilized to perform the work.
- Replacing connectors should occur after component and piping removal with shielding in place. New heater connectors should be installed installed pre-outage.
- Experienced crews, including instrumentation and control, should be utilized to perform electrical work.

7.3.5. Pressurizer heaters

- Place significant engineering oversight on selection, design and fabrication of replacement heaters to ensure performance and critical parameters are met such as fit relative to sleeve inside diameter, placement of the hot heater section in the water space and not in the sleeve, and water space coverage.
- Install new heaters into vessel prior to moving into reactor building.
- Leave old heaters in old vessel because of dose, contamination, storage and disposal concerns.
- Reuse of old heaters in the new vessel can be done but adds risk due to damage in transfer, dose, contamination control.
- Install new connectors on heaters pre-outage.
- Ensure electrical continuity, Megger readings, etc., prior to plant startup to prevent rework on critical path.
- Take advantage of existing routine electrical surveillance as much as possible during plant startup to prevent issues with new startup test plans on critical path.
- Utilize experienced crews for electrical scope and startup test performance.

7.3.6. Rigging

- Temporary lifting device may be considered to minimize polar crane impact but cost may be prohibitive unless used during SGR. Rigging scheme preferably utilizes polar crane.
- Polar crane main hook will likely be replaced with an alternative lifting device. Ensure a load cell or dynamometer is factored into the plan.
- Consider use of jacks at skirt to ensure pressurizer is free from restraint prior to picking up with crane to preclude break away force effect on polar crane/cabling.
- Verify the totality or ‘stack up’ of rigging equipment (drum, slings, strand jacks, crane components, etc.) is supportive of picking up the vessel with adequate clearance of obstructions.

7.3.7. Other good practices

- A developed plan for pressurizer skirt grout and concrete surface preparation is essential, including contingency equipment and material for base resurfacing and shim packs of varying dimension.
- Both on-site intermediate range storage and permanent disposal are common approaches to old pressurizer disposition. Storage of the old pressurizer vessel is often much more feasible than for SGs. Factors that influence this decision include legal requirements, transportation options, owner liability for transport risks, economics for a temporary storage facility versus disposal, economy of scale factors associated with concurrent SG and reactor vessel head disposal and the availability of a disposal processing centre or site.
- A documented foreign material management plan geared to prevention, as opposed to recovery, is necessary for direction and oversight in the fabrication and installation phase of replacement.
- Any areas not accessible for 3-D modelling and as built verification should be evaluated for risk and contingency plans should be developed for design and installation to prevent schedule and dose impacts.
- Experience notes that successful replacement project teams use a documented project phase based on risk identification process coupled with mitigation planning to support effective project execution.

8. RCL REPLACEMENT IN PWRs

8.1. STRATEGIES FOR RCL REPLACEMENT

8.1.1. RCL replacement history

Main replacement operations concerning parts of the primary piping have already been carried out in France, Switzerland and the USA. In France, there were elbow replacements at:

- Dampierre 3 in 1995 (3 elbows C type + 2 elbows D type);
- Gravelines 4 in 2000 (2 elbows C type + 1 elbow D type + 1 elbow B type, resulting in the replacement of half of the cross over leg);
- Tricastin 4 in 2004 (2 elbows C type).

A large piece of the cold leg pipe was also replaced at the Fessenheim plant in 2002. In Switzerland, there were replacements on both units of the Beznau plant — a hot leg elbow and more than half of the cross over leg:

- Beznau 1 1993 (2 elbows A type + 2 90° B type elbows + 2 D type elbows and the related straight pipe pieces);
- Beznau 2 1999 (2 elbows A type + 2 90° B type elbows + 2 D type elbows and the related straight pipe pieces).

In the USA, the surge line connection to the pressurizer, including the first elbow, was replaced in 2005 at the Saint Luice 1 plant and a hot leg spool piece, including the RPV nozzle safe end, was replaced at the VC Summer plant in 2000–2001 [14].

Scope of RCL

The replaced parts are elbows or part of the cold or hot leg or surge line. The referenced elbows are shown in Fig. 39. The replacement of elbows and/or parts of the primary piping is due to the following degradation phenomena:

- Thermal fatigue due to a large number of cycles (which is the case for mixing areas), the consequence of which can be surface cracking;
- Thermal ageing of cast iron/steel, with a resultant loss of its mechanical features.

In the first case, replacement is an alternative solution to repair, particularly when expecting an extension of the plant life. It also offers the opportunity to bring some design improvements to prevent a recurrence of this phenomenon. In the second case, replacement is the only actual foreseen solution. The major application refers to the RCL elbows replacement, for which new elbows are made from forged steel, showing no sensitivity to thermal ageing.

Sequence of RCL

The strategy to be applied for the replacement is directly linked to the part of the RCL to be replaced, but preferably at the same time as a major component (SG or pressurizer) replacement. This is particularly applicable for elbows replacement. All the above listed operations for elbows replacement in France and Switzerland were planned and executed during SGR outages.

In the case of France, the new elbow is welded on the SG in the SG manufacturer's shop in order to minimize the on-site work (see Fig. 40). Consequently, all rules and processes applied for a classical SGR are

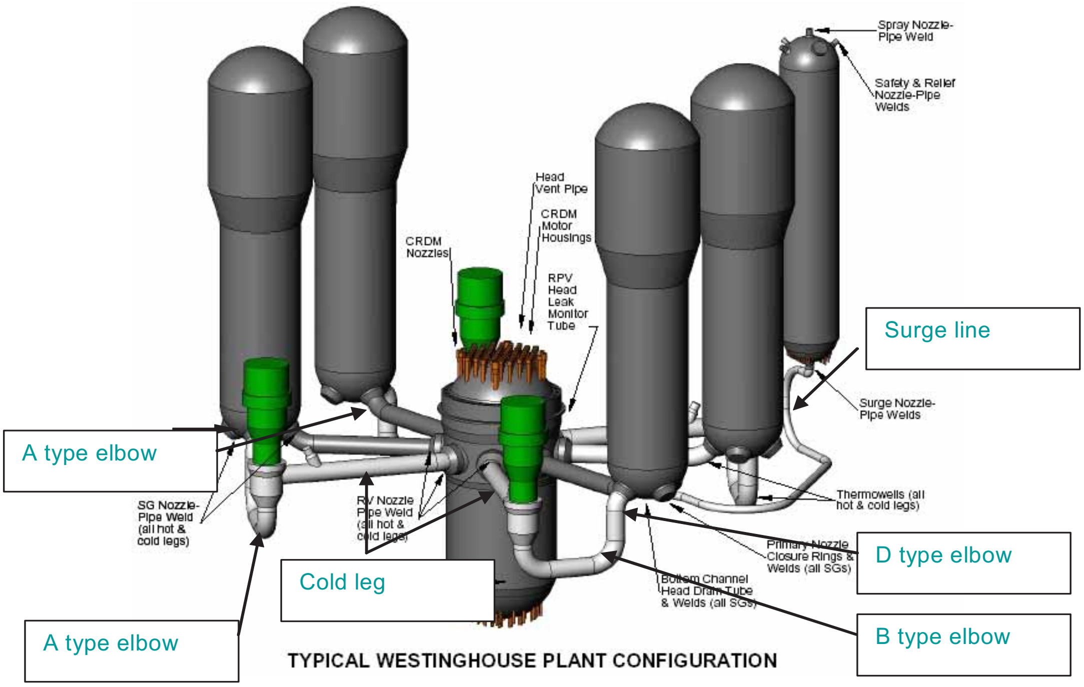


FIG. 39. Location of replaced RCL parts and elbows.

valid; the only difference with this operation is that the cutting section of the RCL is located on the path of the elbow.

However, at the both Swiss plants, all elbow and straight pipe welding was performed on-site by the replacement team offside of the SG manufacturer's shop (see Fig. 40).

When several elbows from the same leg or major pieces of primary piping are replaced, the replacement has to be considered as a heavy component replacement project on its own, even if included in a global replacement project such as an SGR.

An example of a typical scenario is the one applied at Fessenheim 1 in 2002 for the cold leg pipe replacement. The replaced pipe is 3.5 m long and 698.5 mm ID. The length was selected in order to keep the original number of welds on the RCL so that no additional licensing work is required. It should be noted that in

FIG. 40. SGR with welded hot leg elbow in manufacturer's shop.

addition to the RCL welds, the replaced pipe was connected to other lines, including the chemical volume control system.

Owing to the size of the replaced part, to the connection to other piping and to the reactor building environment, the implementation was sequenced as for an SGR, with adaptation to the pipe diameter:

- Fixing up of the reactor coolant pipe casing and of all pipes connected to the cold leg;
- Cutting (2 cuts for the RCL+ 8 cuts for connecting pipes);
- Old pipe and auxiliary piping handling and removal;
- RCL end and reactor coolant pipe casing decontamination;
- Bevelling of the RCL and casing pipe ends;
- 3-D topometry of the cold leg bevels;
- Bevelling of the new pipe spool piece in the workshop;
- Handling and positioning of the spool piece;
- Narrow gap orbital TIG welding of the RCL (2 welds);
- 3-D topometry of the auxiliary pipes' bevel ends;
- Prefabrication of the auxiliary pipes spool pieces;
- Handling and positioning of the auxiliary prefabricated pipes;
- Welding of the auxiliary piping.

It should be noted that, differing from a classical SGR scenario, the choice was made to adapt the new RCL spool piece machining to the measured cut pipe ends, in order to minimize the technical risks (reactor coolant pump casing machining) and the cumulated radiological dose (see Figs 41 and 42).

In addition, this scenario also required the removal of the reactor coolant pump motor and hydraulics, in order to give access inside the RCL, after welding, for root pass grinding and inspecting (penetration test) the welds. This work was carried out by a mobile robot. In the case where this operation is to be done without any other component being simultaneously replaced, a preferred scenario would be to introduce the robot through the RPV nozzle, avoiding the reactor coolant pump components.

When this type of replacement is carried out within an SGR implementation, experience shows that if specific attention is dedicated to all interfaces, the classical SGR duration is not increased.

Another typical scenario is based on the simultaneous replacement made at Bugey of one SG and 2 elbows of the corresponding cross over leg. The optimization of an RCL replacement scenario generally focuses on:

FIG. 41. New RCL pipe spool, welding already completed on-site, during 3-D topometry inspection ready for transportation to its welding position.

FIG. 42. Replacement RCL pipe.

- Limiting the residual stresses of the RCL after replacement;
- Limiting the works in a confined and radiological environment;
- Using the standard SGR tooling and personnel;
- Using the standard SGR methodologies.

The resulting strategy in Bugey's case was to supply the new SG fitted with the D elbow, welded in the manufacturer's shop and to install the B elbow on the RCL pipe after the removal of the old SG and before the installation of the new one.

Design, calculation and licensing activities

The 'prescriptions' applied are the same as those for an SGR. It is important not to change the design (geometry, welds) of the RCL to avoid additional licensing work or impact on the surrounding equipment (e.g. supports), except for significant improvements (e.g. use of extruded forged RCL pipes to allow butt welding of auxiliary piping).

8.1.2. Outline of RCL

The more sensitive technologies for RCL parts replacement are very close to those used for an SGR.

Optical survey of RCL

Special care has to be taken for an optical survey in the following two cases:

- Supply of new SG with elbows welded in the manufacturer's shop. The final positioning for the RCL pipe welds requires precise measurement in the manufacturer's shop and on-site after cutting.
- Solitary RCL spool piece replacement. In this case, adaptation of the new piece on-site on the basis of a 3-D measurement, after bevelling the existing ends, is recommended.

Clamping, cutting and machining are the same as for an SGR (see Section 4.2).

Decontamination of ends of RCL

The same processes and technologies as those for an SGR may be used. When replacing an RCL spool piece, special attention should be paid to retrieval of the RCL plugging system preventing chemical products from invading the RCL. Owing to the long distance from the plugs' location to the access (reactor vessel or reactor coolant pump), it is recommended that a mobile robot be used to perform this operation.

RCL rigging

For rigging of SGs fitted with elbows, it is recommended that complete dynamic 3-D modelling be carried out to optimize the paths and clearances. The orientation of the SG is also a constraint that needs follow-up since the installation of the new SG on its shipment saddle.

Handling of RCL spool pieces remains a job for specialists, even more so when long pieces are concerned, but it is not as complicated as for an SGR.

Fit-up and alignment of RCL

The same accuracy is required as for an SGR because the same welding processes are used. Same processes are also used for 3-D metrology and RCL fit-up activity.

Connection line/supporters

Specific design and processes may be applied to adjust, machine and weld connection lines. For significant diameters (150 mm and above), butt welds are preferred and thus require specific designs for the forged RCL spool pieces (extruded forgings).

Connecting lines should be cut in two places in order to allow the prefabrication of new connecting spool pieces adapted to the final configuration of the RCL and the existing end of the auxiliary pipe.

A specific study will take care of the pipe supports, in order to select which ones are to be removed and replaced/reinstalled, depending on the position of the cut sections.

Welding of RCL

The same processes as for an SGR must be used. In addition, the stainless steel pipe welding process (manual or automatic) needs to be qualified.

In-pipe robot system for back groove machining and inspection

The need for such robots or manipulators was highlighted in Sections 8.1.3 and 8.2.3. Various types are available within the potential main suppliers' tooling. Nevertheless, for each application, a complete qualification procedure and personnel training have to be undertaken.

8.1.3. Civil engineering

For most RCL spool piece replacements, there is no need for specific civil works outside the reactor building. In addition, the spool piece path inside the reactor building can be done through the reactor building equipment hatch. The implementation, including the old spool piece removal and the fit-up of the new one may include temporary removal of structure interferences, if revealed by the 3-D dynamic modelling. The civil work engineering should then be treated as for an SGR or a pressurizer replacement.

8.1.4. Radiation protection

As a master piece of the primary circuit, the RCL is a severe radiological and confined environment. Any replacement of a part of the RCL will consequently be relevant for an ALARA programme. When this replacement is part of a major replacement (SGR or pressurizer replacement) operation, it will be treated within the overall ALARA programme.

The cumulated dose rate of the operation may be significant. As an example the replacement of a cold leg spool piece between the RCP and the reactor vessel in Fessenheim resulted in a cumulated dose of 236 man·mSv.

The ALARA programme includes an estimated, and then contractual, cumulated dose, based on the radiomapping undertaken during the walk downs and the selected replacement strategy. It includes the definition of the shielding volume and layout.

8.1.5. Test programme

The test programme will be set up and performed in accordance with the selected codes and standards specified in the RFQs and part of the suppliers contracts. Depending of these codes and standards, the test programme may include inspections (welds) and hydro tests (in shop, and/or on-site).

8.1.6. Lessons learned

The major lessons learned concerning the replacement of parts from the RCL are that:

- It is worthwhile replacing RCL parts jointly with the replacement of a major component (SG or pressurizer), in terms of cost, schedule, organization and dose rate.
- Owing to the fact that it affects the primary boundary, a stand alone RCL spool piece replacement should be considered as a significant project requiring an organization (project team), human capabilities (design, licensing), and manufacturing and implementation skills from suppliers similar to those required for an SGR or a pressurizer replacement.

9. REACTOR INTERNAL COMPONENT REPLACEMENT IN BWRs

9.1. STRATEGIES OF INTEGRATED REACTOR INTERNAL COMPONENT REPLACEMENT FOR BWRs

9.1.1. Integrated reactor internal component replacement history

Reactor internal component replacement history

SCC on the core shroud of a BWR was first found in the 1990s in several countries (Germany, Japan, Sweden, USA, etc.). Since the shroud is typically welded to the RPV and other parts of core internals, mechanical repair, brackets, or tie rod devices have been applied to reinforce the shroud with respect to cracking, under regulatory approval. A method of replacement of the core shroud together with other internal components such as jet pumps was developed jointly by Japanese BWR owners and fabricators and applied to the following six plants:

- Fukushima daiichi Unit 3 (Tokyo Electric Power Company, performed in 1998);
- Fukushima daiichi Unit 2 (Tokyo Electric Power Company, performed in 1999);
- Tsuruga Unit 1 (Japan Atomic Power Company, performed in 2000);
- Fukushima daiichi Unit 5 (Tokyo Electric Power Company, performed in 2000);
- Shimane Unit 1 (Chugoku Electric Power Company, performed in 2001);
- Fukushima daiichi Unit 1 (Tokyo Electric Power Company, performed in 2001).

In Europe, there is an experience of BWR reactor internal component replacement work accomplished in Oskarshamn unit 1 in Sweden in 1997. The content of the replacement work was different from current Japanese work, because the majority of the internal components of that plant were bolted connected structures, differing from the welded structure that is the major feature of Japanese BWR plants.

9.1.2. Basic policy of replacement work

In these replacement works for welded reactor internal components, there are several basic policies to follow to perform the work efficiently:

- The possibility to work on the reactor bottom during a certain period of the replacement work greatly assists performing complex and difficult work.
- To gain access to the bottom, chemical decontamination should be performed.
- To gain access to the bottom, effective shields should be installed during certain periods of the replacement work.
- Removal of the old reactor components should be performed remotely in order to reduce the radiation exposure.
- New core shroud weld edge is a narrow groove employed to reduce residual stress and minimize the welding time.

9.1.3. Scope of reactor internal component replacement

The main objective of integrated replacement is to replace immediately all components susceptible to SCC, such as core shroud, jet pumps and so on, thereby reducing total schedule, cost and radiation exposure. Several components were replaced at the same time, they are as follows (see Fig. 43):

- Core shroud;
- Top guide;
- Core plate;
- Core spray spargers;
- Feedwater spargers;
- Jet pumps;
- Differential pressure liquid control piping;
- In-core monitor guide tubes;
- Internal piping and nozzle safe ends connected to these components.

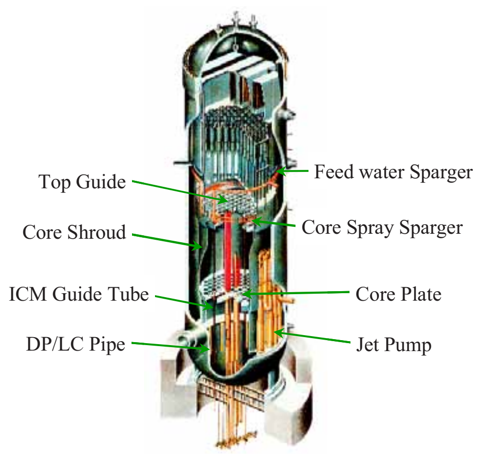


FIG. 43. Outline of integrated reactor internal component.

9.1.4. Replacement sequence

The outline sequence of reactor internal component replacement work consists of seven steps as follows [15, 16]:

(1) Chemical decontamination

- After taking out removable internal components such as a steam dryer, shroud head, fuels, control rods and guide tubes, the chemical decontamination is performed to remove the radioactive metal oxide deposits in order to reduce the radiation level in the RPV for in-vessel work.

(2) Removal of existing core shroud and jet pumps

- Remove feedwater spargers, top guide and core plate. Existing core shroud is remotely cut into two pieces using electric discharge machining cutter under water. Other internals such as feedwater spargers, top guide, core plate, jet pumps are also cut under water. The components removed from the vessel are transferred to the dryer separator pool in order to conduct the slicing prior to disposal.

(3) Installation of in-vessel shielding and installation of new jet pumps

- After removal of the internal components, radioactive metal oxide deposits in the vessel are removed by using mechanical decontamination technology. In-vessel shielding is installed to reduce radiation from the vessel's inner wall in order to enable human access. New jet pump installation by welding is done by semi-automatic technology assisted by the worker in the vessel.

(4) New core shroud fit-up

- Welding preparation for new core shroud is accomplished by remote machining tool prior to new shroud installation. Then, the new core shroud is installed in the RPV using an overhead crane and fitted up smoothly onto the shroud support by using the jacking system which was previously equipped on the control rod drive housings.

(5) New core shroud welding

- The installed new core shroud is welded onto the existing shroud support. A narrow gap welding technique, using an automatic welding machine and laser alignment system, is used. According to the individual plant's programme, the core shroud may be installed in two steps: (1) lower core shroud installation and (2) upper core shroud installation.

(6) Core plate and top guide installation

- A new core plate and a new top guide are installed.

(7) Restore fuels and removable internal components

- Fuels and removable internal components are restored.

The replacement sequence is illustrated in Fig. 44.

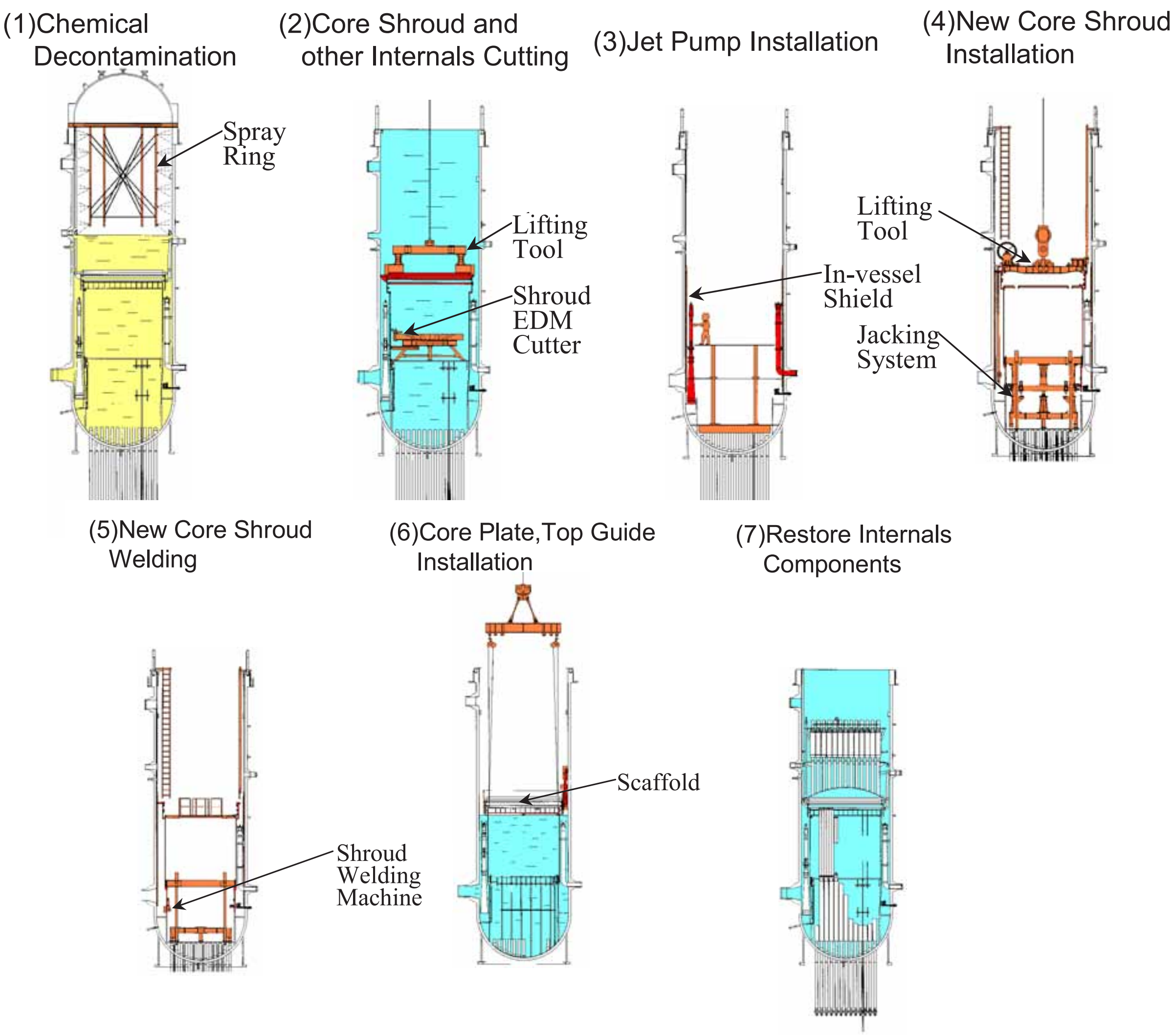


FIG. 44. Replacement sequence of reactor internal components.

9.2. RECIRCULATION PIPING REPLACEMENT

9.2.1. US experience

In 1962, the first recorded SCC crack occurred at GE Vallecitos nuclear power plant on weld sensitized 304 stainless steel reactor recirculation piping. This event was followed by an observation of the first SCC occurrence at the Dresden-1 plant in 1965, again on the weld sensitized 304 stainless steel reactor recirculation piping. Since then, many instances with similar characteristics have been recorded.

The next historically significant SCC event took place in 1974, when several through-wall cracks on Dresden-2 reactor's recirculation bypass line were found. This event triggered discoveries of similar observations at Quad Cities 2 and Millstone 1, and further revealed cracking at 15 locations including 5 instances of through-wall cracking at 6 plants as a result of examinations performed in 21 BWR plants.

When NUREG-0313 was published in 1977, it stated that IGSCC would not be a significant threat to safety. However, such an optimistic belief was proved to be wrong when significant cracks were found next year at nozzle-to-safe end welds of 254 mm reactor recirculation inlet nozzles at Duane Arnold [17].

In 1982, leakage during hydrostatic test at nozzle-to-safe end welds of two 711 mm reactor recirculation outlet nozzles of Nine Mile Point 1 raised a concern about the reliability of examination because UT performed during previous outage turned out to be ineffective. Lessons learned from Nine Mile Point 1 event prompted a development of NDE coordination plan at the EPRI-NDE Center, where signification efforts were made to facilitate improvement of UT technicians' skill in detecting IGSCC. Finally, Appendix VIII was added to ASME Section XI in 1989, and training, procedure/equipment/personnel qualification became part of regulatory requirements.

In parallel with continuous improvement of examination capability, piping replacement was performed by employing high IGSCC resistant materials specified in NUREG-0313 Rev.2. Piping of reactor recirculation and residual heat removal systems were partly or entirely replaced with 316NG or 316L stainless steel at 12 BWR plants through 1999.

The weld overlay technique was developed and utilized in lieu of piping replacement at 13 BWR plants through 1983. It continued to be applied at over 260 locations in over 20 plants through 1999. Effectiveness of this technique was proven when over 1000 follow-up examinations were made without signs of crack propagation.

9.2.2. German experience

Stabilized austenitic stainless steel has been used for all German LWRs with only one exception, Gundremmingen-A, which used 304 stainless steel and was permanently shut down in 1980. With respect to piping material only, titanium stabilized 321 stainless steel was chosen for all BWR plants with only one exception, the first German BWR which is not operating anymore, where niobium stabilized 347 stainless steel was used.

Unlike BWRs in the USA, all BWRs in Germany are equipped with reactor internal pumps with no reactor recirculation piping outside the pressure vessel. Type 347 stainless steel is used for reactor internal components such as the core shroud. Also unlike US BWRs, all BWR plants in Germany operate under normal water chemistry.

The first crack with 347 stainless steel piping was found in 1991. The cause of cracking was attributed to fabrication defects. The first crack with 321 stainless steel was reported in 1992, when a GRS notice was issued requiring all BWR plants with ~100 000 operating hours to be subject to an enhanced NDE programme.

In response to this instruction, over 3100 welds of 321 stainless steel piping greater than 50 mm (DN 50) were examined at 6 BWR plants and a total of 58 welds exhibited cracking through 1995. They were mostly confined within the heat affected zone. However, there were some instances where chromium depletion along grain boundary was not evident.

All cracked 321 stainless steel pipes were replaced with 347 stainless steel pipes section by section. So far, no crack initiation on replaced sections was detected.

9.2.3. Japanese experience

As observed internationally, BWR plants in Japan stemmed from the technology brought from the USA through the General Electric company began to exhibit SCC on 304 stainless steel piping from the mid-1970s.

In 1974, SCC occurred on the elbow section of the reactor recirculation system bypass line at Unit 1 of the Fukushima Daiichi plant. This was followed by another occurrence observed on core spray piping at Unit 2 in 1976. Detection of SCC was so frequent in late 1970s that new cracking was found during every scheduled outage. Every time new cracking was found, efforts were made to characterize it. If it was verified to be SCC, then the affected section of piping was replaced as a corrective action (Fig. 45).

Typical corrective actions taken in the mid-1970s included replacement of sections of piping while reducing sensitization using water cooled welding, elimination of those components such as the reactor recirculation bypass line which only provided extra sites for SCC initiations with no specific functions, and application of induction heat stress improvement on the weld joints where replacement was not needed.

These conventional corrective actions were replaced by a new approach in the early 1990s when low carbon stainless steel was introduced for new plant construction. As a permanent remedy for the then operating plants, 316NG stainless steel was selected for replacing existing piping. In conjunction with such pipe replacement, new fabrication technologies became available to produce large diameter seamless pipes, single elbow nozzle forging integrated into T-joints, and heat induction bent pipes composed of elbow and straight

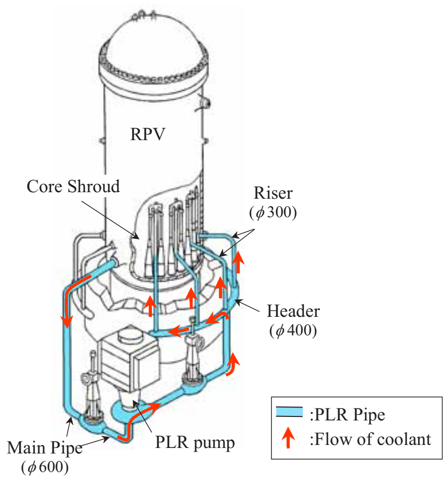


FIG. 45. BWR recirculation piping.

sections. Those new products significantly reduced the number of field weld joints and contributed to the reduction of person-hour and person-rem associated with pipe replacement work at the site.

When activities to mitigate 304 stainless steel SCC were mostly completed in the late 1990s, it was found later that low carbon stainless steel (316NG and 316L) was also susceptible to SCC by surface hardening. Some characteristics of this type of SCC are as follows:

- Cracking initiated in the vicinity of the weld, typically several millimetres from the fusion line, and propagated towards the weld metal in the mode of IGSCC.
- Majority of cracks ended their propagation before reaching weld metal, whereas some cracks propagated further to penetrate through the fusion line.
- Cracks initiated at the hardened surface layer. Machine induced microstructure possibly formed during the fabrication stage was noticed near the initiation spot.
- SCC occurrence frequency for the butt weld joints decreases as pipe diameter decreases (305 mm and 405 mm pipes exhibit lower SCC occurrence frequency than 610 mm pipe).
- Regardless of pipe size and in-service duration, the depth of most SCC cracks was less than 10 mm.

Methods of reducing the susceptibility of low carbon stainless steel piping to SCC include the following, for each of which Japan has had experience of practical application:

- Improvement of weld residual stress profile by heat sink welding (welding process applied for the piping with cooling water inside), induction heat stress improvement, solution heat treatment and narrow groove welding;
- Wall cladding to inside of the pipe heat affected zone;
- Moderate hydrogen water chemistry.

In 2003, a flaw evaluation guideline was made available for the nuclear power plants in Japan. However, the only case where it has been used so far has been for the reactor recirculation piping at Unit 3 of Kashiwazaki-Kariwa power plant. Pipe replacement of degraded sections has been performed for all other cases where cracks were detected by UT during scheduled outage.

9.3. MAJOR TECHNOLOGY INVOLVED

9.3.1. Chemical decontamination

For the human in-vessel work such as installation of automatic tools, etc., reduction of the radiation from both the irradiated vessel inner wall and radioactive metal oxide deposited on the components is required. Chemical decontamination and in-vessel shield installation are expected to reduce the radiation level in the RPV. There are some kinds of technologies for chemical decontamination and it depends on the power plant owners' selection which technology is to be applied. Here, the technology is shown from the Japanese plants' experience for example.

After taking out removable internal components, such as the steam dryer, shroud head, fuels, control rods, and so on, the chemical–oxidation reduction and decontamination with an ultraviolet light process is used to reduce the metal oxide built up inside the vessel.

Figures 46 and 47 show the outline of the decontamination loop, which consists of the RPV and reactor recirculation system loop and temporary circuit. The decontamination solution was circulated by the reactor

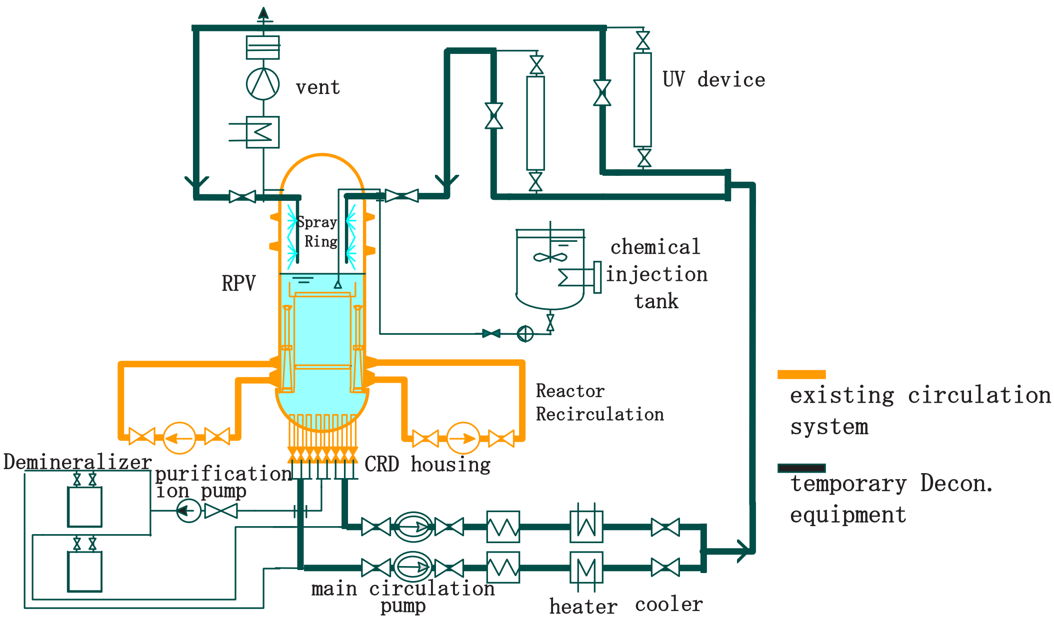


FIG. 46. Full system decontamination flow diagram.

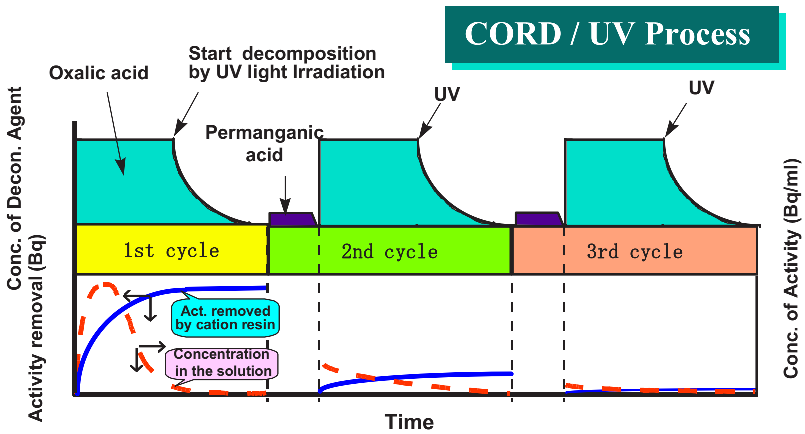


FIG. 47. Chemical oxidation–reduction/ultraviolet process.

recirculation system pumps and the temporary circulation pumps aligned between control rod drive housings/ in-core monitoring housings and a spray ring, which is installed between the RPV and the RPV head. The reactor recirculation system pumps are operated at the minimum flow rate to circulate decontamination solution in the RPV, especially at the bottom and the lower downcomer region.

In the experience of RINR work, three cycles of chemical decontamination were performed intentionally. After the three cycles of decontamination, all the isolated pipes and the small instrument piping were back flushed into the RPV by make-up water. Furthermore, the water was purified by the mixed bed resin columns, and the absence of residual chemicals was confirmed. After the chemical decontamination, the mechanical cleaning procedure was performed.

The outline sequence (1–8) of the mechanical cleaning is as follows (see Fig. 48):

(1) Suction cleaning underwater;
(2) Brush cleaning (reactor well, RPV wall and bottom);
(3) Water drain and cleaning (reactor well);
(4) Reactor water drain;
(5) Water jet cleaning in the air (RPV well, bottom and baffle plate);
(6) In-vessel shielding set up underwater;
(7) High pressure jet cleaning in the air (RPV bottom);
(8) Final rinsing (RPV bottom and baffle plate).

The RPV bottom dose rate was lowered to 0.1 mSv/h after the chemical decontamination and to 0.03 mSv/h after the mechanical cleaning under water together with shielding. After draining the reactor water, the dose rate in air was 0.2 mSv/h, low enough for human access inside the RPV (see Fig. 49).

9.3.2. RIN cutting

The existing core shroud is cut under water by the electric discharge machining (EDM) process. As shown in Figs 50 and 51, some EDM cutting carriages are mounted on the circular track assembly installed on the core plate (for the upper shroud cut) or on the control rod drive housings (for the lower shroud cut). Each EDM cutting carriage has a rotating graphite electrode, and a swarf collection hood is attached to the carriage. These carriages can run individually because of easy handling for the installation onto the work platform and shortened cutting time.

The existing shroud is cut circumferentially into two pieces in order to assure the water shielding after and during the underwater transfer to the dryer/separator pit (DSP), where they are cut into small pieces for disposal. The upper face of the remaining shroud support is machined to ensure a sufficient flatness for the new shroud welding. The shroud support edge preparation machining tool, which uses a milling head, is shown in Fig. 51.

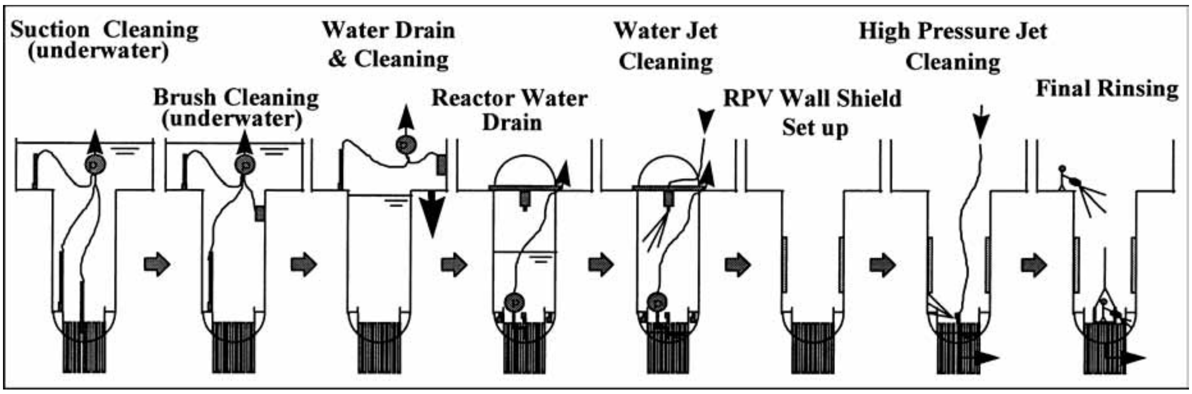


FIG. 48. Outline sequence of the mechanical cleaning.

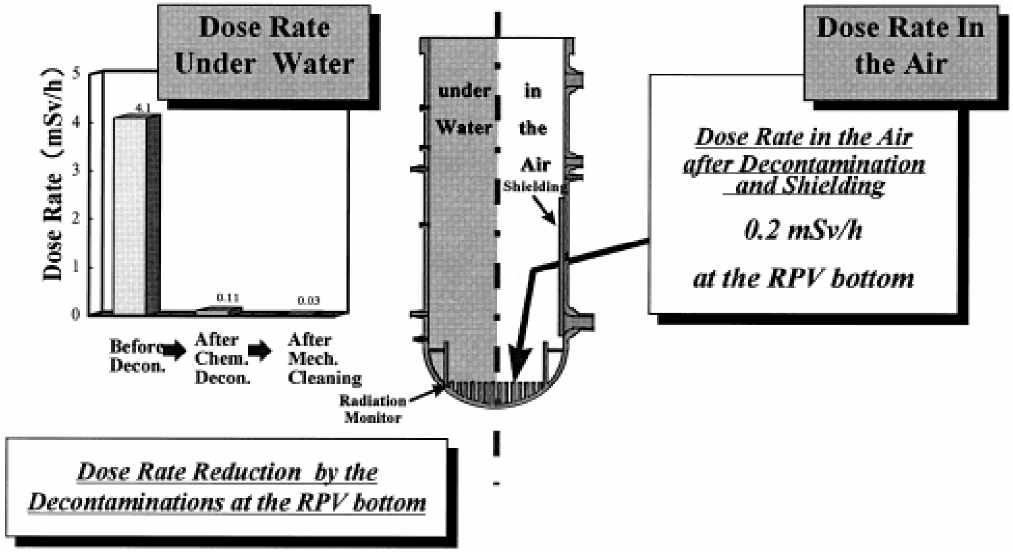
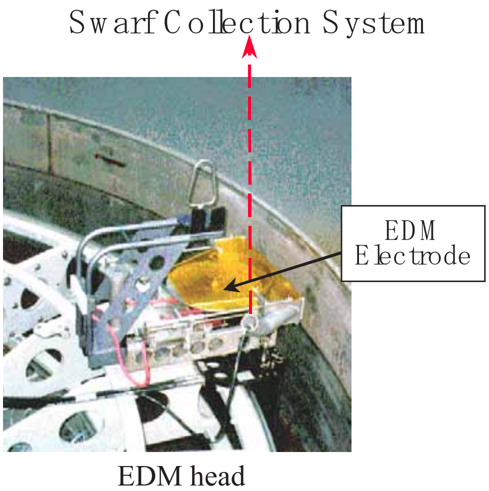


FIG. 49. Dose rate comparison in air and under water.



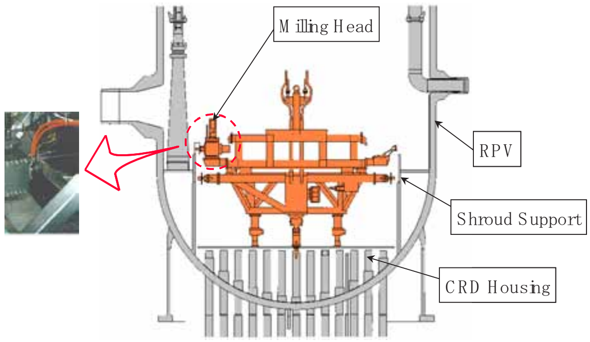


FIG. 50. The EDM process.

Fig. 51. Shroud support edge preparation machining tool.

After cutting, the upper shroud and lower shroud are moved out from the reactor to the equipment pool where slicing using ultra high pressure water cut technology is performed. Then the small pieces are packed into the cask to be transferred to the site bunker pool or dry storage facility. Other internal components to be removed such as feedwater spurger, jet pumps, etc., are cut by EDM and removed in order.

9.3.3. Installation of in-vessel shielding

Figure 52 shows the schematic view of the in-vessel shielding. Two types of shielding are used, one is the shielding panel for the RPV inside wall, the other is the shielding plate for the vessel bottom. The RPV shielding system consists of stainless steel encased lead panels, with interlocking panel design, that are hung around the vessel inside diameter to provide a circumferential radiation barrier. Shield panel design includes considerations other than personnel exposure, existing obstructions, accessibility to the work area behind the shields, interface with the hardware installed during the work, and shield panel removal after the hardware installation work. These panels are made of lead covered with stainless steel. The dose rate after chemical decontamination and

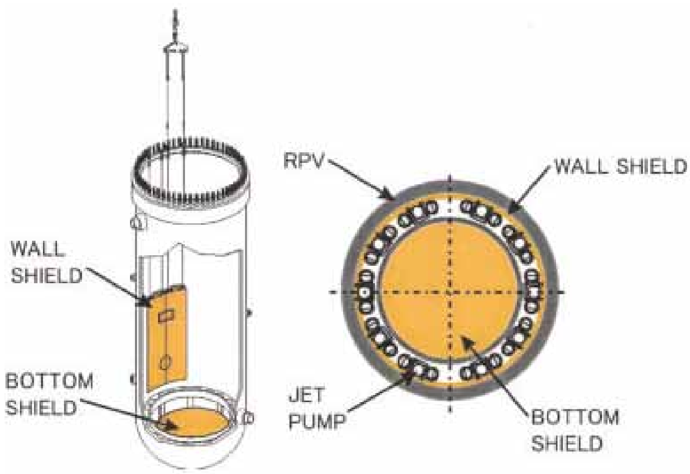


FIG. 52. Schematic view of the in-vessel shielding.

in-vessel shield installation is expected to be significantly lowered. Tungsten could be used instead of lead for the higher dosage area in order to improve the effect.

9.3.4. Installation of jet pump

The main components of the jet pump installation are the riser, diffuser and riser brace. The jet pump installation procedure is as follows:

Shroud support plate preparation:

- The machining tool is installed and the old material is removed. Then the baffle plate is machined to re-overlay configuration and the overlay welded by automatic remote welding machine. As the final step, the weld preparation for the final diffuser weld is machined.

Riser assemblies and safe end welds:

- The inside diameters of the thermal sleeve at the safe end and riser elbow were measured and the thermal sleeve was machined. Then the riser assembly was installed into the RPV and welded to the nozzle safe end.
- Before commencing the riser brace installation, the templating tool is centred around the riser pipe and the positions of the four riser brace leaves were measured. Based on these values, riser braces were machined and installed on location. The automatic remote weld robot welded the riser brace to the RPV and riser yoke weld.
- Using a jacking clamp, the diffuser was aligned in its proper position and tack welded. Then the diffusers were welded to the shroud support plate while monitoring the diffuser alignment using by target/camera assembly installed in the top of the diffuser.

9.3.5. Installation of core shroud

The new core shroud should be designed to improve the SCC resistance from the point of view of material and welding. For example, a new core shroud designed in Japan has:

- A reduced number of welds by applying the three forged cylinders (schematic explanation eliminating the vertical and horizontal welds is shown in Fig. 53);

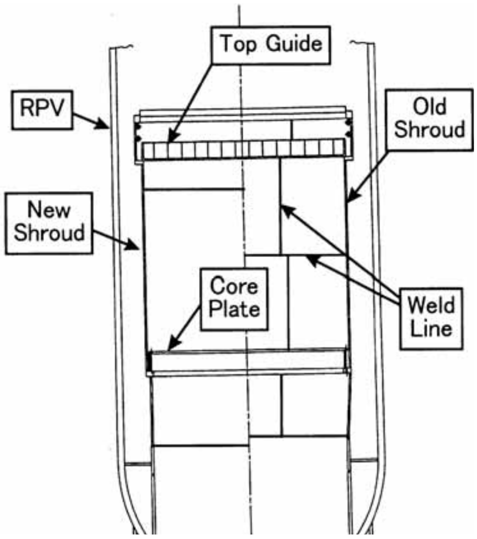


FIG. 53. New core shroud.

- A new core shroud made of type 316L stainless steel that has high SCC resistance.

The new shroud has to be transported in two halves, divided into upper and lower parts. First, the lower part of the new shroud is moved into the reactor building from the large equipment hatch and is raised to the refuelling floor using the overhead crane.

Next, the upper part of the new shroud is moved into the reactor building and set onto the top of the lower part of the shroud that was moved in previously by overhead crane. The lower and the upper parts of the new shroud are welded into one on the refuelling floor. Alternatively, if possible, a specific containment opening may be considered (see Fig. 53.).

Handling new core shroud in the reactor building

Depending on the reactor building configuration, the new core shroud may need to be handled in separate parts. Considering the plant's situation, if the height of the shroud is greater than the restriction set by the refuelling floor's height, a short ring method for core shroud installation could be applied and no civil reconstruction would be needed.

The new shroud is lowered into the RPV by overhead crane and aligned to the surface of the shroud support weld preparation edge. To avoid the unexpected deformation of the shroud support cylinder by the landing of a heavy new shroud, a jacking system was installed prior to the shroud fit-up. The jacking system is shown in Figs 54 and 55.

The new core shroud is welded to the shroud support with full penetration and narrow gap welding. Welding is performed automatically by the welding robot (see Fig. 56).

The core shroud installation and welding sequence is summarized below:

- Lower the new core shroud into the RPV and set onto the jacking system those that are prepared beforehand;
- Core shroud fit-up;
- Install the weld equipment;
- Perform weld;
- Inspection of welding (liquid penetration test);
- Removal of the welding equipment.

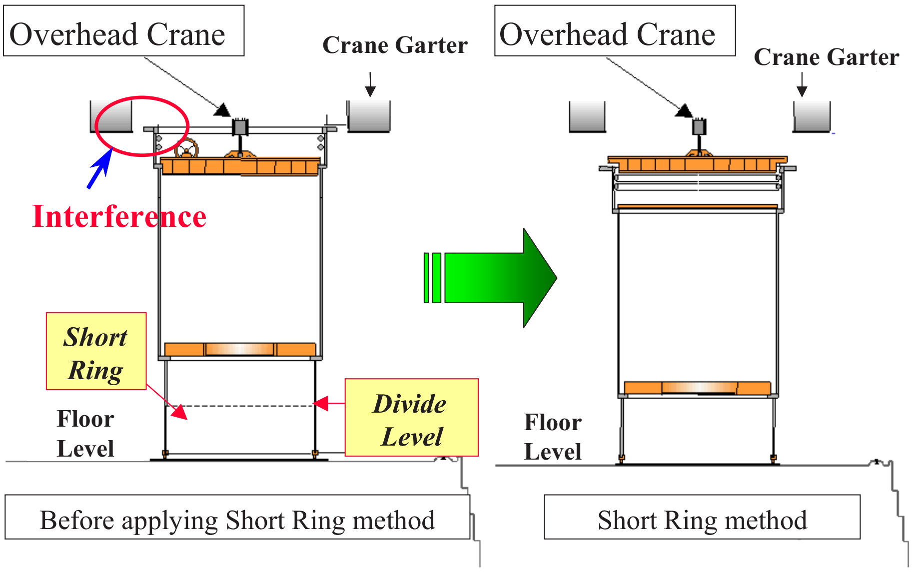


FIG. 54. Handling new core shroud in the reactor building.

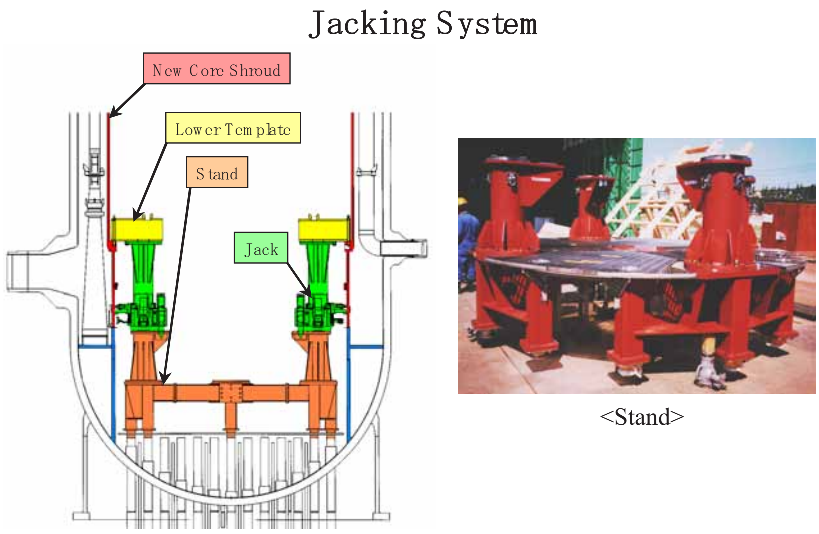


FIG. 55. Jacking system to install new core shroud.

9.3.6. Installation of top guide and core plate

After the new core shroud installation, the platform supporting the core plate and the top guide are installed. Before the platform supporting core plate is installed, the in-core guide tubes and other lower internals have to be restored [18].

The platform supporting the core plate is integrated into the operation platform and lowered into the vessel. A laser alignment tool is used to ensure the platform supporting core plate is attached with high precision. After the core plate installation, the top guide is installed using the integrated platform and specialized tools. The top guide should also be installed with high precision while adjusting its alignment pin.

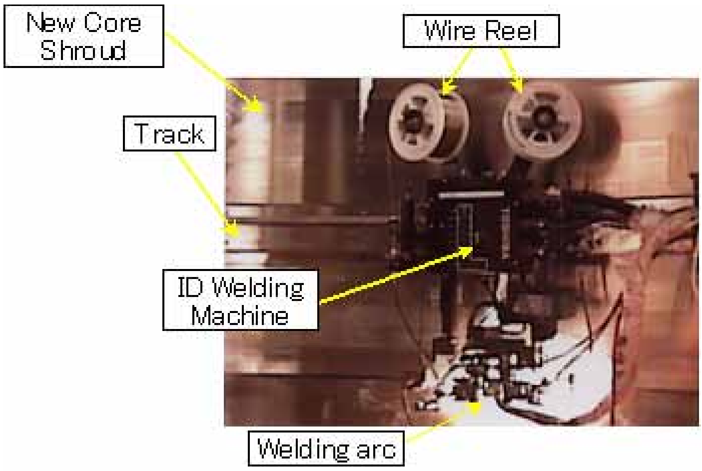


FIG. 56. Inside diameter welding machine.

9.3.7. Restore internal component

Final replacement work involves the restoration of the piping within the vessel. Lower the core spray lines into the vessel and carry out the settings. Then, the feedwater sparger is to be set at its designated location.

9.3.8. Waste management

Waste management is to be programmed by consideration of the individual plant's situation. The old core shroud and other old internals replaced must be cut into small pieces for the transfer from the reactor building. These cutting works are performed in the DSP. Owing to the limited space of the DSP, the waste must be removed from the DSP before the next component is transferred to it. In the experienced works, cutting is performed by means of an underwater plasma arc cutting method because of its fast cutting speed. In the case of Japanese plant's works, the wastes are contained in radioactive waste casks and transferred from the reactor building to the site bunker pool or dry storage facility at the power plant. The site bunker pool stores relatively high level radioactive wastes and the dry storage facility stores lower level radioactive wastes.

9.4. TEST PROGRAMME

The test programme is planned through the replacement work. Each component replaced requires proper testing during the replacement work. For example, the material of newly replaced components is confirmed, and as for the welding test, there are groove dimensional and liquid penetration tests to be performed at the proper time. As for the post-installation test, there are visual, dimensional and hydraulic tests. Each test is to be witnessed by governmental ministry or agency, or by utility.

9.5. LESSONS LEARNED

Through work experience there are many lessons learned that are useful in the following work. Chemical decontamination and mechanical cleaning can achieve a remarkable reduction in dose rate. Chemical decontamination is performed for 3 cycles and the decontamination effect of each cycle is clarified by the investigation in the field. Development of a remote robot for operating in high dose environments such as jet pump riser brace can be a great help in reducing human radiation exposure. Total radiation exposure through

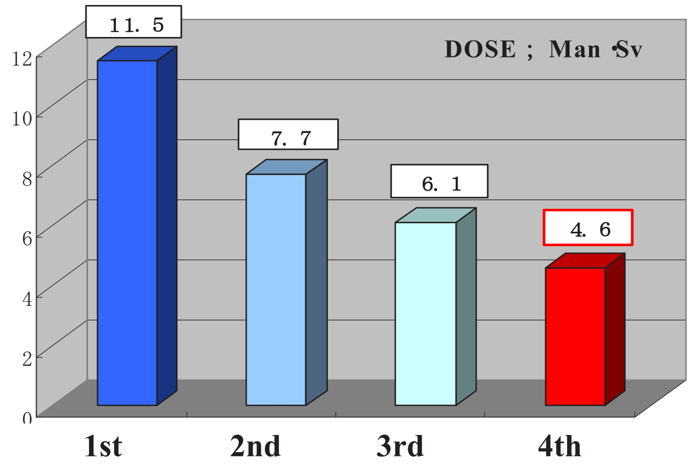


FIG. 57. Gradual reduction in radiation exposure of each work cycle.

the work has been reduced gradually through experience. Considering the individual plant's restriction, the short ring method for core shroud installation has been adopted with no civil reconstruction (see Fig. 57).

10. HEAVY COMPONENT REPLACEMENT IN PHWRs

10.1. INTRODUCTION

Reactor component replacements have taken place in CANDU reactors ranging from individual pressure tube replacement, also called single fuel channel replacement (SFCR), to entire core pressure tube replacement (LSFCR). LSFCR has been completed successfully in Pickering Units 1–4. LSFCR projects require extended outages beyond the regular maintenance outages normally planned for routine maintenance work. SFCR can be accommodated during a routine maintenance outage.

Bruce A Units 1 and 2 are currently undergoing major rehabilitation that will extend the in-service life of these units by at least 25 years. Whereas SGs have been replaced in many LWRs around the world, replacement in CANDU type reactors has just started. Bruce A Units 1 and 2 are currently undergoing SGR, as well as LSFCR and feeder pipes replacement. Unit 2 is scheduled for synchronization to the grid in 2009, followed by Unit 1, also in 2009.

Point Lepreau is the first CANDU 6 class plant, which will be refurbished to extend its service life by 30 years. The refurbishment outage for Point Lepreau is planned for 2008 and will last for 18 months. The Point Lepreau refurbishment project does not include SGR. However, it will comprise the replacement of all the fuel channels (LSFCR) including all feeder pipes. Other CANDU 6 plants, which went into commercial service in the early 1980s are expected to follow Point Lepreau's lead.

Generic SGR studies for CANDU 6 plants have been prepared for the eventuality that they may be needed in the future for life extension reasons. Pickering B is studying life extension beyond its current design life. Current projected outage for Pickering B Unit 1 refurbishment is between 2012 and 2017. It is projected that if Pickering B life extension is to go ahead, the following heavy components will need to be replaced: LSFCR, connecting feeders and SGs. The economics of such an undertaking are still under evaluation.

10.2. CANDU FUEL CHANNEL REPLACEMENT

10.2.1. Replacement history

In the late 1980s, all fuel channels in four CANDU units were replaced to address the degradation mechanisms discussed above. A total of 1560 fuel channels were replaced at that time. As part of the planned life extension of CANDU units, all fuel channels, a total of 1720 in four CANDU units, are slated for replacement over the next few years. The current plan is to remove and replace 380 fuel channels including the calandria tubes in one CANDU 6 unit in approximately 9 months.

10.2.2. Process for fuel channel removal and replacement

The basic operations involved in the removal and replacement of the fuel channel assemblies during a reactor retube are summarized in the following sections [19, 20].

Fuel channel removal

- Defuel the fuel channels;
- Drain the primary heat transport and moderator systems;
- Disconnect the feeder pipes from the fuel channels;
- Remove the positioning assemblies from both sides;
- Cut the bellows from both ends;
- Cut the pressure tubes at both ends just inboard of the end fitting;
- Remove the end fittings from both sides;
- Remove the pressure tubes;
- Shock heat and release the calandria tube from the end shields;
- Remove the calandria tubes.

Inspection

- Clean and inspect the tube sheet bores;
- Clean and inspect the lattice tube and bellows.

Fuel channel replacement

- Install calandria tube inside the calandria;
- Roll both ends of the calandria tube;
- Check for leak tightness – if required;
- Pre-assemble end fitting assemblies;
- Cut the pressure tube to the required length;
- Roll the pressure tube to one end fitting in the shop;
- Leak test the rolled joint in the shop;
- Install the subassembly inside the calandria tube;
- Install the spacers in the annulus;
- Verify their installed locations;
- Install end fitting from other side;
- Roll the other end of pressure tube into the end fitting;
- Leak test the rolled joint – if required;
- Weld bellows to the end fitting at both ends;
- Reconnect the feeders to the end fittings;
- Perform system pressure test.

10.2.3. Tooling for removal and replacement of fuel channels

Significant investment has been made in designing and building specialized tooling to remove and to replace the fuel channels very efficiently. Remote controlled tooling operated by programmable logic controllers has been designed and built to handle radioactive components in order to reduce the worker dose.

Heavy work platforms, work tables, flasks, specialized cutting tools, welding tools and inspection tools are some of the tools that have been built for this purpose. Additionally, a significant number of mock-ups were built to test and qualify these tools and to train personnel.

Special tooling of interest

To address the concerns related to radioactive waste disposal, AECL has developed a 'volume reduction system'. This tool shears both the pressure tubes and calandria tubes into small square metal pieces. This process reduces the volume of the radioactive waste by more than 50%. Also, the volume reduction system directly deposits the crushed pieces into a waste handling flask, thus reducing the material handling and the associated dose to workers. Both the tooling and the flasks are handled using vault cranes and brought inside and taken out of the reactor vault through existing equipment airlocks.

The sheared pressure tube material is dropped into a waste storage container and brought to a long term storage facility.

10.2.4. Summary

The CANDU fuel channel design has undergone improvement over the years to mitigate the known degradation mechanisms. Continuous improvement in design through operating experience, R&D findings, inspection results and maintenance needs over the past thirty years has resulted in a very robust fuel channel design that is easier to construct with predictable, safe and reliable performance, and which is easy to maintain over the life of the plant. Fuel channel removal and replacements are part of a life management strategy for CANDU reactors to extend their operating life. AECL has specialized tooling and procedures for the safe and effective removal of irradiated fuel channel assemblies and reinstallation of new fuel channel assemblies.

10.3. SGR IN CANDU REACTORS

10.3.1. Background

The most obvious design evolution for CANDU SGs has been size; with the size increasing as the manufacturing capability has increased, to the stage where the advanced CANDU reactor SGs are of a similar size to the larger PWR designs. However, there have been important evolutions in other key areas, which include tube material, support material, support design and balance of plant feedwater/steam cycle piping and equipment design and materials. These design evolutions came as a result of lessons learned from operating plants and knowledge gained from R&D programmes in Canada and other countries with nuclear programmes.

10.3.2. Process for removal and replacement of SGs

Each CANDU 6 class unit comprises four identical SGs with integral preheaters. The SGs consist of an inverted vertical U-tube bundle installed in a shell. Steam separating equipment is housed in the drum at the upper end of the shell. The lower end of each SG has a support stool, which is welded at the site to a vertical column. The column is approximately 13.4 m high and supports the SG on the fuelling machine vault floor. For radiation shielding purposes, the portion of the SG housing the U-tube bundle is located inside a concrete boiler box. The boiler box also serves to house the SG lateral supports and the redundant and normally unloaded vertical back up steel cable supports [21–23].

10.3.3. Outline of SGR

As stated earlier, all CANDU 6 containment buildings are cylindrical structures and use grouted prestressed cables, which precludes the possibility of cutting a large opening in the spherical containment dome, or following the original SG installation route, which was through a large temporary opening at grade level.

Owing to the size limit of the equipment airlock and the type of containment prestressing design used, only one replacement method is viable for CANDU 6 plants, namely the cartridge replacement method, described in this section. It requires removal and storage of the steam drum within the reactor building for reuse, since the steam drum is too large to pass through the equipment airlock.

The steam drum will be removed by machining the inner shroud and shell away from the lower part (the cartridge) of the SG, containing the heat exchanger and the primary head. The cartridge, which weighs approximately 130 t, would be jacked out of the boiler box and then manipulated through a series of steps, using a trolley carriage leading to its lowering through the boiler room floor hatchway and transported out of the equipment airlock. Figures 3 and 4 illustrate a general view of the cartridge exit path. A new cartridge is brought in through the equipment airlock, hoisted up the hatchway and installed in the boiler box.

Cartridge removal and replacement methodology

To confirm that the proposed methodology for the removal and replacement is feasible, a 3-D computer model of the reactor building and that of the SGs was used. To implement this methodology in the field, it is expected that high resolution laser scans of the areas that house the SGs will also be required to confirm the exact location of the cartridge, nozzles and clearances available.

Many activities must be completed prior to cutting and moving the steam drums and removing the old cartridges. These activities include the partial removal of the dousing system structure and equipment above the boiler boxes to provide sufficient overhead clearance for the replacement activities to take place. The work involves the construction of temporary structures necessary for the storage of the SG drums, and removal and installation of the old and new cartridges, including the material handling equipment such as the trolleys and carriages.

The erection of scaffolding will be required to enable the various trades to perform the necessary work around the SGs. Some temporary structures will also be needed to protect the reactivity mechanisms deck and other equipment, located in the boiler room floor underneath the travel path of the cartridge. The weight of each cartridge is estimated to be approximately 130 t. Because of insufficient weight carrying capability, the boiler room crane will not be used and will be parked. A temporary movable trolley structure should be installed at the same location and used for the removal and replacement activities. The existing crane tracks should be replaced with heavier rails to carry the weight of the cartridge.

As stated earlier, the SG drums will be reused. These will be separated by machined cuts in the shroud and in the secondary shell of the vessel below the drum to the shell cone section and stored in the reactor building on temporary structures located on the boiler box. The remaining bottom part of the SG (the cartridge) including the primary head, and the secondary shell complete with the tube bundle should be removed as one assembly.

Prior to severing all piping connections and supports, restraints must be installed to maintain the position and orientation of the large bore piping connected to the SGs. This includes the steam lines (attached to the steam drums), feedwater lines (attached to the secondary shell of the cartridge) and all the primary heat transport system lines (attached to the primary heads of the cartridge). All instrumentation tubing will also be removed. It is also necessary to take the weight of the cartridge on the jacking assembly over the boiler box. After the nozzles are severed, they are all capped as well as the opening at the top of the cartridge.

The cartridge should be jacked out of the boiler box using hydraulic jacks, a trolley being positioned under the lifted cartridge. The cartridge is then lowered onto the trolley and into a horizontal position. The trolley then transports the cartridge into the centre of the boiler room along temporary rails. The cartridge trolley direction is changed perpendicular to the boiler box and moved near the hatch of the 'D' side of the reactor building where it is lowered to the 117 M elevation, rotated and placed above the hatchway in a vertical manner for lowering onto a transporter on the 105 M elevation. The installation should be done as one assembly as well.

The equipment airlock at the 105 M level would be used for transporting the old or new cartridge out of or into the reactor building. No new opening would be required. The reactor building floors at the 117 M elevation and the 105 M elevation and the service building floor at the 105 M elevation would require temporary shoring. Instead of modifying the service building crane, an additional mobile crane would be used to place the cartridge on a transporter for moving it to the storage building. The replacement cartridge would be installed in the boiler box by generally reversing the steps of the removal process.

Schedule considerations

For the first generation CANDU 6 which came into operation in the early 1980s, reactor retubing is a prerequisite for life extension. SGR may be required in some plants owing to problems arising from specific plant design and operation. Whereas SGR in a CANDU 6 can be performed independently from reactor retubing, it is more economic if both can be implemented during the same extended outage.

The duration of the SGR work in the reactor building by the cartridge method is approximately seven months and the overall duration of reactor retubing is approximately thirteen months, excluding defuelling, commissioning and restart.

Some activities pertaining to the reactor retubing programme and SGR programme will require the same space allocation inside the reactor building including the equipment airlock and service building, hence the sequence of both programmes needs to be harmonized and scheduled to avoid interference and work inefficiencies.

10.3.4. Disposal of the removed cartridges

The most common industry practice has been the permanent storage of removed SGs or the cartridges in a secure building without any further processing.

11. STORAGE FOR FREE RELEASE AND RECYCLING OF MATERIAL

11.1. STORAGE OF RETIRED LARGE COMPONENTS IN MAUSOLEUM

In some countries, utilities have prepared a temporary storage for large retired components using a mausoleum at or near their sites. This implies easy and short distance transportation based on national or local transport regulations. The mausoleum can open to take samples from various parts of the components at different periods (in the future), e.g. for studying life time issues and for investigations of different types of damage such as fatigue or cracking, and it allows tracing decay of initial activities and dose rates.

The building of such mausoleums, however, increases waste project costs, owing to the need for extra building and its commissioning and service as well as later decommissioning. The retired objects have to be dealt with later, both mechanically and economically, and be stored finally in the national repository, implying additional costs due to final conditioning to satisfy acceptance conditions for this repository and transport from the mausoleum to the repository.

Storage of retired large SG components in mausoleum

In some countries, after SGR, an alternative solution is carried out for decommissioned SG storage. The used SGs are stored directly on-site in a dedicated mausoleum building. This is the case in, for example, Brazil, France, Japan, the Republic of Korea and the USA. The mausoleum built on the site for an SGR is an adapted standard building with concrete walls and roof 50–80 cm thick, for environmentally safe site storage (Fig. 58).

This mausoleum building may accommodate 2 or 3 decommissioned SGs per unit. Considering these countries' regulations and laws, the major requirement of the mausoleum is to not pollute air, water or soil, not

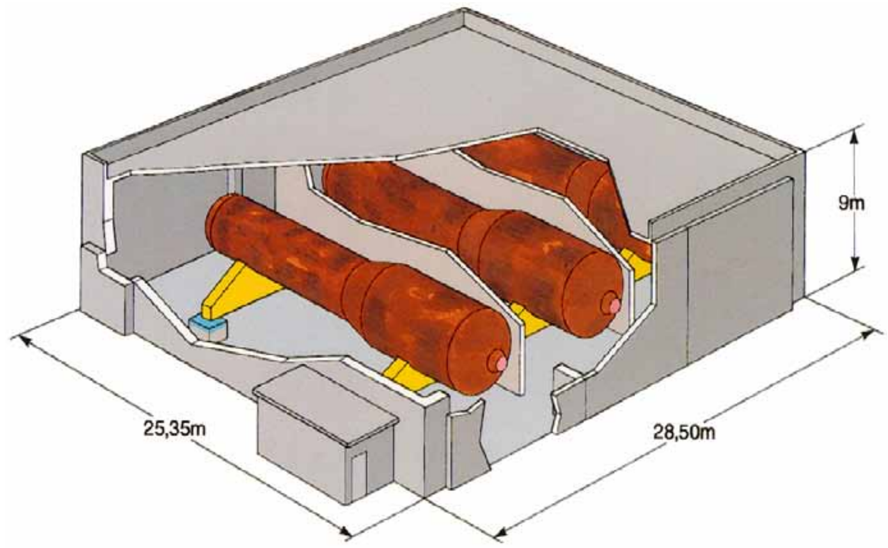


FIG. 58. Example of decommissioned/used SG storage mausoleum building.

generate any waste and to consider it a controlled zone with monitoring. The mausoleum is integrated on the power plant site and a dedicated monitoring programme is applied on this building (Figs 59 and 60).

11.2. STORAGE OF RETIRED LARGE COMPONENTS OF REACTOR VESSEL HEAD INTERIM/ FINAL STORAGE SITE

The temporary and long term storage of decommissioned reactor vessel heads are adapted in several countries considering mandatory regulations and legal status. Some countries have a particular and specific storage programme based on removal from the reactor building, packaging for transportation to interim storage and then long term storage. Before removal from the reactor building, the heads are packed in a special package complying with the regulations governing personnel and environmental protection. This consists of several casings ensuring containment, biological protection and weather protection during transportation (Fig. 61).

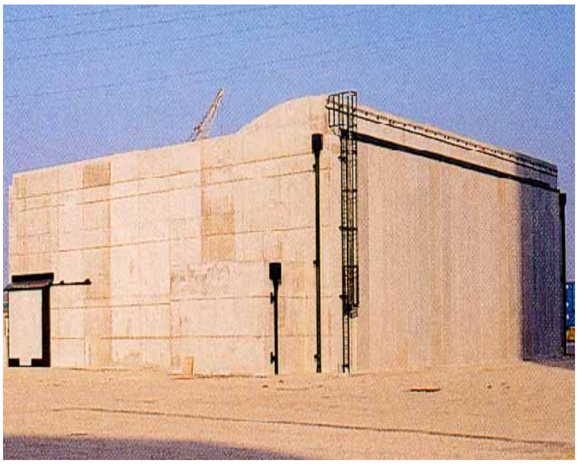


FIG. 59. Example of decommissioned SG mausoleum in France.

FIG. 60. Example of decommissioned SG mausoleum in the USA.

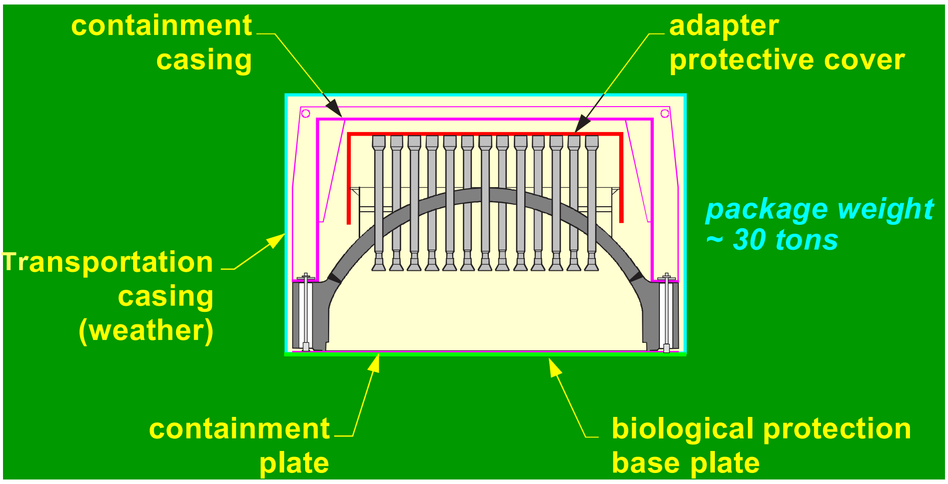


FIG. 61. Reactor vessel head transportation and protection packaging.

The packaging contains:

- Adapter protective cover;
- Containment casing and a containment plate;
- Biological protection;
- Containment casing;
- Weatherproof transportation casing.

The reactor vessel head is kept on-site in its packaging before transportation (Figs 62 and 63). Depending on the national rules, reactor vessel head transportation is subject to authorization from the competent authorities. Because of the low radioactivity and fast activity decay of reactor vessel heads, the strategy is not to decide immediately on the final storage and an irreversible solution. The utility prefers to allow some time for assessment of the level of low activity before transporting vessel heads to the final storage site or before decontamination and recycling.

The interim storage is a solution that involves grouping the vessel heads in a room sized and fitted out so as to ensure safe interim storage. Solutions contemplated and investigated by the utility for disposal are as follows:

FIG. 62. Reactor vessel head on-site in its packaging before transportation.

FIG. 63. Reactor vessel head transportation.

• Recycling;
• Melting for volume reduction;
• Complete head storage.

A complete head storage solution is a possible option worth submission with respect to the country's laws and authorization from the safety authorities. This storage consists of:

• Transportation of the heads and the packages to the storage centre;
• Immobilization by concrete directly injected into packages without breaking the containment.

11.3. COST EFFECTIVE ALTERNATIVE TO APPLYING DIRECT TREATMENT OF RETIRED COMPONENTS FOR MATERIAL RECYCLING AFTER FREE RELEASE

Swedish regulations allow treatment of metallic waste using melting for recycling. This is valid for metallic waste from small pieces in drums up to large retired components. Melting is demanded by the competent Swedish authorities since it is the only method that guarantees destroying hot spots in a controlled manner with no chance of 'missing' something as opposed to using 'free measurements' for example. Also, a small sample of some grams is representative of the whole batch of a couple of tonnes due to the homogenization effect in an induction furnace. A metallurgical analysis of these samples is also possible to provide good first hand information to the end user on the releasable metal quality (Fig. 64).

Preparations before melting, such as decontamination, segmentation and sorting, make possible the free release of such objects (and not only large components but any metallic objects in the LLW or ILW category) either directly after melting or after some period of temporary, i.e. short term decay, storage to reach free release levels. The nuclide specific levels for free release in the Swedish regulations are based on Ref. [24].

When using the methods developed in Sweden (since 1987), statistically more than 95% of the treated material can be free released and is reused by the Swedish steel industry. The steel industry receives and accepts free releasable material in the form of 600 kg ingots for steel or 400 kg for aluminium or other metals (e.g. Cu, Pb, brass). The steel industry has to guarantee to remelt these ingots, blended 1:10 with normal material, which

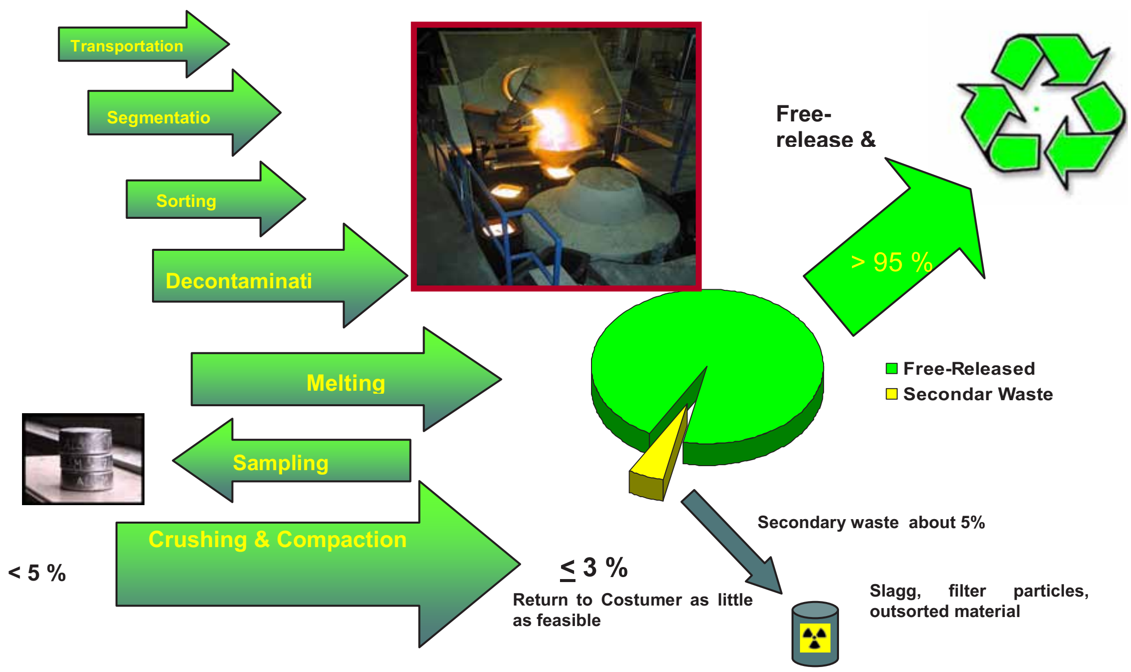


FIG. 64. Treatment of metallic waste using melting.

TABLE 2. TABLE 2. SOME EXAMPLES OF NUCLIDE SPECIFIC FREE RELEASE VALUES FROM REF. [24]

Nuclides	Mass specific Bq/g	Nuclides	Mass specific Bq/g
H 3	1000	Co 58, 60	1
C 14	100	Cs 137	1
Mn 54	1	Ra 226	1
Fe 55	10 000	U 235, 238	1
Ni 59, 63	10 000	Am 241	1
Pu 238, 239, 240	1		

implies that the free release levels at the end, when ready for recycling, are actually ten times lower than the stipulated RP-89 values.

This implies that less than 5% of secondary waste such as slag, filter dust and non-free-releasable cutting residues, will be returned to the original utility/waste owner for final storage at the customer's national repository. The costs of final storage of these objects are thus reduced correspondingly.

Two important results show:

- Total cost of waste treatment is reduced;
- The utility rids itself of its waste problem sooner owing to 'change of ownership' after free release.

11.4. METHODS OF TREATMENT FOR RECYCLING

The LLW material is normally transported within Sweden (domestic material) or to Sweden (foreign material) either in 6 m containers by truck on roads or as individual large components by boat. To date, the vessel M/S Sigyn, owned by SKB, has been used (Fig. 65). The containers or the objects are transported under internationally accepted transport regulations, e.g. EU-92/3. For objects that do not comply with any of these specific EU categories an alternative transportation form can be used, called 'special arrangements', and this has been applied successfully for international transport (e.g. between Germany and Sweden) as a legal procedure,

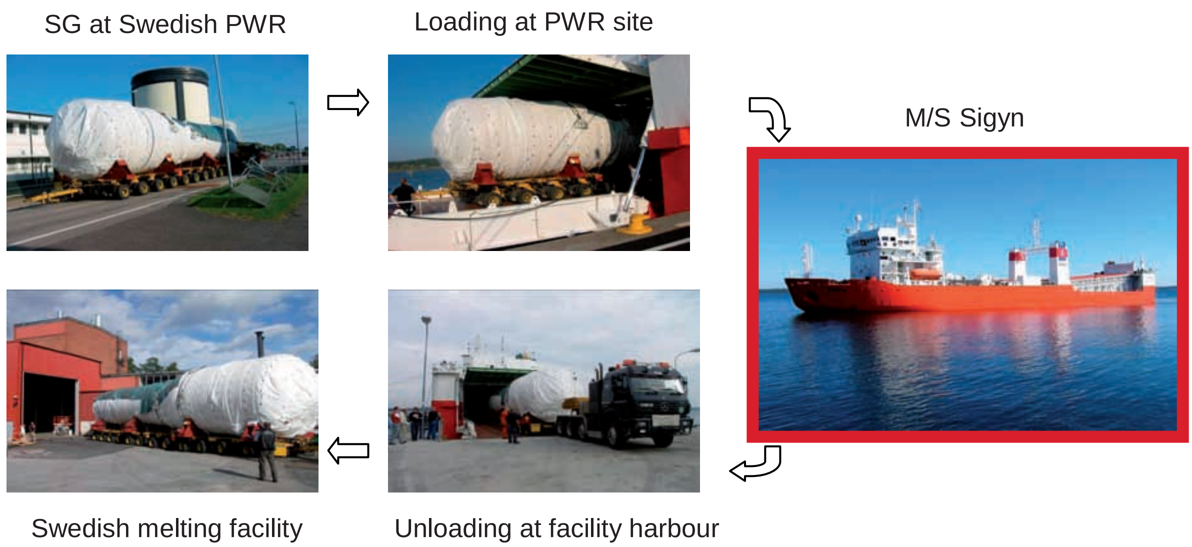


FIG. 65. Transportation of replaced SG.

according to international transport rules. It just demands a complement of special packaging and stowing treatment to guarantee otherwise required safety aspects.

The delivery is then monitored at arrival as regards dose rates for acceptance and mode of planned treatment. On the basis of the mandatory customer's documentation, the mode of treatment is decided in a customer related project. Customer specific treatment is necessary to avoid cross-contamination with nuclides from other deliveries — to the extent feasible — with nuclides which any specific customer in question has not generated in their waste and thus does not want to take back in return. Before melting, the demand for decontamination is investigated. Decontamination may be necessary for two reasons:

- To allow the workers to handle the objects with doses as low as feasible according to the ALARA principle;
- To allow free release of the radioactive material either directly after melting or within a stipulated short term period — limited by the authorities — in a temporary decay storage to reach free release levels.

Sometimes decontamination has to be conducted before segmentation (due to ALARA reasons), sometimes after segmentation in order to be able to access those areas of the object which have to be treated specifically before melting, e.g. by blasting. All material has to be cut into pieces so as to fit into the melter opening (typically 50 cm); all parts have to be controlled carefully to open 'closed compartments', in order to avoid a steam explosion when entering the melter.

In some countries (e.g. Japan), cutting of waste objects is mandatory just to satisfy acceptance criteria (all in 200 L drums) for their final repository. Under all circumstances cutting needs a 'controlled area' including radiation protection, ventilation and monitoring.

Material is sorted — in some cases in connection with segmentation — into the different types of metal to provide as 'clean' an ingot as reasonable for the end user: ingots of steel, aluminium, brass, copper or lead are requested on the 'market'. After melting and after having taken off the slag from the melt surface, the now homogenized melted material can be poured into an ingot mould. From the same melt, three samples may be taken for analysis, as required for the producer's record, for the customer and for the authorities if later control is requested by any party. The ingots are then cooled down and stored away temporarily to await free release for reuse/end use for recycling.

After free release it is possible to offer the original owner a document stating change of ownership to enable the customer to demonstrate to the authorities that his 'waste problem' is solved satisfactorily.

11.5. DECONTAMINATION EXPERIENCE

Various methods of decontamination apply to various parts of radioactive objects. In order to minimize secondary waste, however, experience shows that blasting is superior to chemical means. Although chemical methods may have the advantage of reaching surfaces with odd geometries, this method produces cubic metres of sludge and contaminated fluids in comparison to some litres of — many times recycled — blasting residues. Blasting can be conducted in blasting cabins (for large pieces), in tumble blasters (for small pieces) or in tube blasting devices (for tubing with contaminated inner surfaces) (Fig. 66).

There is, however, no need to drive decontamination too far; the goal is only to get the object 'clean enough' to allow for free release after melting. As an example, if temporary short term decay storage is allowed for 10 years, then considering ^{60}Co as the main nuclide, with a half-life of about 5 years, a residual activity after melting of less than 4 Bq/g is enough to reach the stipulated free release level of 1 Bq/g within the allowed period.

11.6. EXPERIENCE ON AMOUNT OF RECOVERED MATERIAL

Statistically more than 95% can be free released from metallic scrap; it may be in the form of small pieces (delivered in drums or baskets) or larger LLW objects such as turbines and heat exchangers. As regards SGs, 85% can be recovered for recycling — except for the tubes (Fig. 67). If the SG, however, is pre-decontaminated

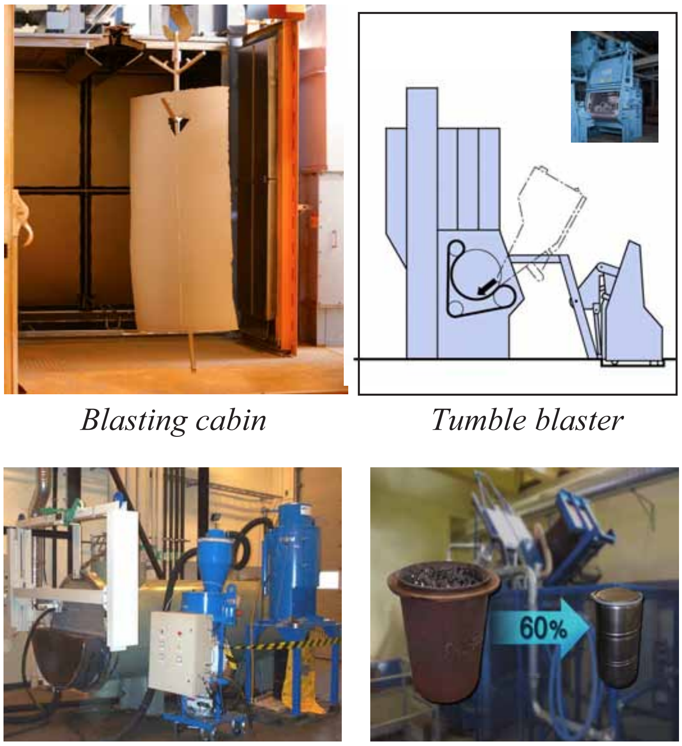


FIG. 66. Minimization of secondary waste by decontamination and crushing.

by system decontamination while the reactor is still in operation, the tubes may become clean enough for a more improved treatment, i.e. either volume reduction by melting or even ready for free release after additional forced secondary cleaning. Tubes made of Ni–Cr alloys provide valuable material which is worth retrieving if possible (ALARA). This may then also contribute to an economic refund.

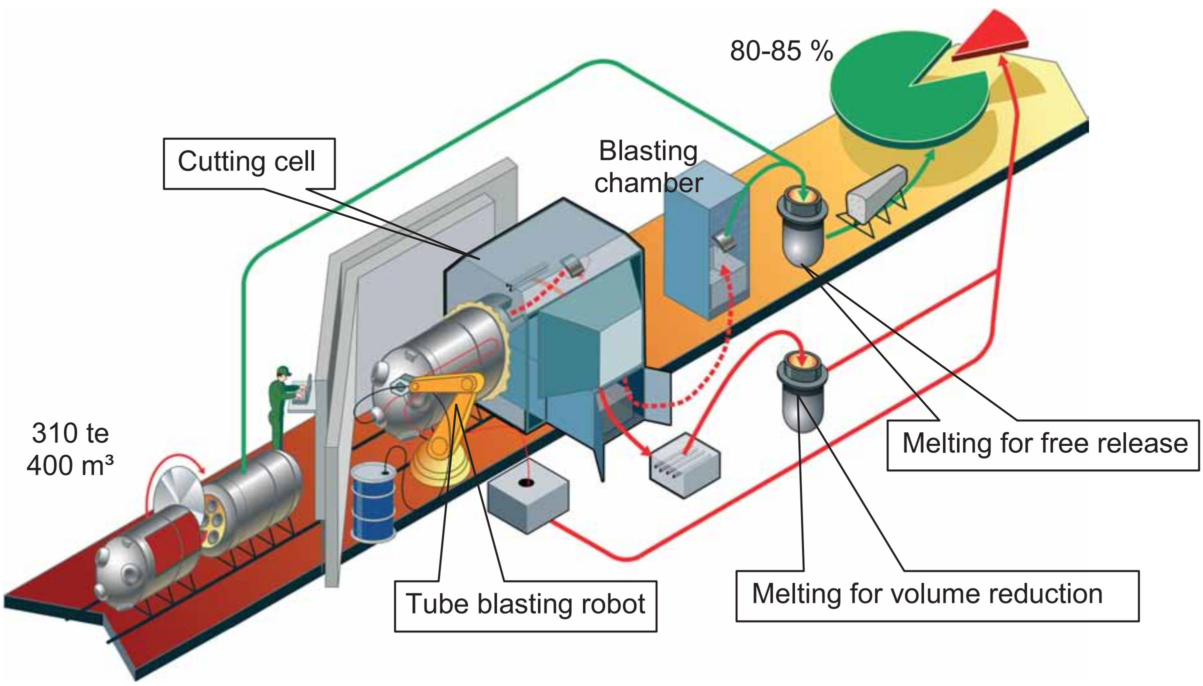


FIG. 67. SG waste treatment concept in Sweden.

An SG (typically 400 m^3, 310 t from Ringhals in Sweden) without decontamination may draw 30–60 mman·Sv in collective dose during a waste recycling treatment as demonstrated during 2006–2007. Such results can be achieved if a careful 'dose budget' is established before the treatment in order to investigate any of the dose demanding phases and feasible counter measures proposed to minimize personal dose exposition to the workers.

Overall personal dose exposure when treating radioactive material from nuclear power plants (main nuclide ^{60}Co) as well as from fuel fabrication factories (main nuclides from uranium and ^{241}Am) was demonstrated to be less than 0.8 mSv, a mean value over a year's handling, with a maximum of about 3 mSv for any person in the melting facility.

In Sweden, the SGs are treated in a specifically designed 'cutting cell' with necessary shielding and computerized and robotized tools inside. An operator stand equipped with lead glass windows allows visual control when necessary. Normally, remote supervision by cameras, internal lights and positioning control is applied successfully. Also, inside packaging of remotely segmented pieces into different boxes for further treatment — controlled by robots — allows optimized handling to provide a maximum of free releasable material for recycling.

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ABBREVIATIONS

CRDM	Control rod driving mechanism
EDM	Electric discharge machining
EPRI	Electric power research institute
IGSCC	Intergranular stress corrosion cracking
LSFCR	Large scale fuel channel replacement
NDE	Non-destructive examination
OSG	Original steam generator
PWSCC	Primary water stress corrosion cracking
RCL	Reactor coolant line/loop
RCS	Reactor coolant system
RFQ	Request for quotation
RPV	Reactor pressure vessel
RVHR	Reactor vessel head replacement
RVI	Reactor vessel internals
SCC	Stress corrosion cracking
SFCR	Single fuel channel replacement
SG	Steam generator
SGR	Steam generator replacement
TGSCC	Transgranular stress corrosion cracking
UT	Ultrasonic test
WBS	Work breakdown structure

Appendix I

SAFETY, LICENSING AND REGULATORY ISSUES IN JAPAN

I.1. LICENSING PROCESS

The procedure for obtaining the permits and licences required for the replacement of equipment is described in this appendix.

I.2. LAWS AND AGREEMENTS

The major laws and agreements to consider when replacing equipment are listed in Table 3. To ensure that the design of the replacement equipment is appropriate, changes must be made to the nuclear reactor installation permit and construction work plan licence application forms in accordance with Act on the Regulation of Source Material, Nuclear Fuel Material and Reactors and Electricity Utilities Industry Act. Excessive worker exposure to radiation must also be minimized, as well as the safety of cranes assured, in accordance with Industrial Safety and Health Law, Laws Concerning the Prevention of Radiation Hazards due to Radioisotopes and others, and other applicable laws.

One procedure in Japan is that the consent of the local communities involved, based on a safety agreement, must also be obtained. This is not stipulated by law, but to ensure the safety of these communities, it is necessary to forge an agreement with the local governing body, obtain the consent of the communities to the plans, ahead of time, in accordance with the agreement, and report to the communities on the progress of the construction work that is performed.

I.3. FLOW OF PERMITS AND LICENCES

Using the replacement of an SG as an example, Fig. 68 shows the procedural flow involving permits and licences based on the Act on the Regulation of Source Material, Nuclear Fuel Material and Reactors and Electricity Utilities Industry Act, from among the laws and agreements listed in the previous section. There are two perspectives from which to consider procedures covering the replacement of SGs: (1) the replacement of the SGs themselves and (2) the storage of the SGs after their replacement.

(1) Nuclear reactor installation change permit application forms: When SGs are to be replaced, this gives rise to changes to items in the main text (heat transfer pipe dimensions and steam generation amounts are added), and nuclear reactor installation change permit application forms become a requirement. Similar applications are also required for items stored in the SG storage facilities.

(2) Construction work plan licence application forms: When SGs are to be replaced, construction work plan licence application forms must be filed, covering changes in the materials used by the SGs themselves, changes in the heat transfer surfaces and internal structures, and changes in the strength and resistance to earthquakes calculations accompanying weight increases and other factors. In addition, applications for repairs in regard to the provisional openings of nuclear reactor containment vessels, provisional openings of external shields, and annulus seal cutting, which are all required for the work, need to be filed. Applications for construction work plan licences concerning the capacities, shields and other aspects of the SG storage facilities are also filed.

I.4. IMPACT OF REPLACEMENT

After the nuclear reactor installation change permit applications have been filed, they go through the primary and secondary screening stages, and it takes about one year altogether before the permits are issued.

TABLE 3. LIST OF PERMITS AND LICENCES

Laws/agreements	Permit/licence types	Remarks
Safety Agreement	Prior consent Notification of construction plans; progress made in construction work, completion Use of building sites	Procedures based on agreements made with local communities
Act on the Regulation of Source Material, Nuclear Fuel Material and Reactors	Nuclear reactor installation permit application form Public security regulations Control zone changes	Applicable to basic changes in equipment design Applicable to changes in safety analysis results
Electricity Utilities Industry Act	Construction work plan licence application form Revisions, changes, notification, etc. Pre-use inspections	Applicable to materials, strength, resistance to earthquakes
Natural Parks Law	Geological surveys Building site preparation, etc.	Applicable to usage of sites for storage facilities
Factory Location Act	Green spaces	Pertaining to usage of sites for storage facilities
Building Standards Law	Construction verification	Applicable to storage facility structural parameters
Laws Concerning the Prevention of Radiation Hazards due to Radioisotopes and others	Control zone changes	
Industrial Safety and Health Law	Plans Cranes	Applicable to plans concerning safety of construction work Applicable to crane installations, notifications regarding remodelling
Fire Service Law	Fire prevention equipment Hazardous materials	Applicable to fire prevention equipment in storage facilities Handling requirements of hazardous materials
Waste Management and Public Cleansing Law	Disposal of concrete waste materials by burying on premises Plans and performance reports	
Ship Safety Law	Marine transportation plans	
Foreign Exchange Control Law	Introduction of technologies	

Following the application process for the nuclear reactor installation change permits, an additional month or two is needed for the construction work plan licence application to be processed. An overview of the processes is shown in Fig. 69.

I.5. REGULATORY POSITIONS AND UPFRONT LICENSING

In Japan, the replacement of equipment is not subject to regulations. These replacements of SGs and other equipment have been implemented in accordance with the maintenance plans of the operators.

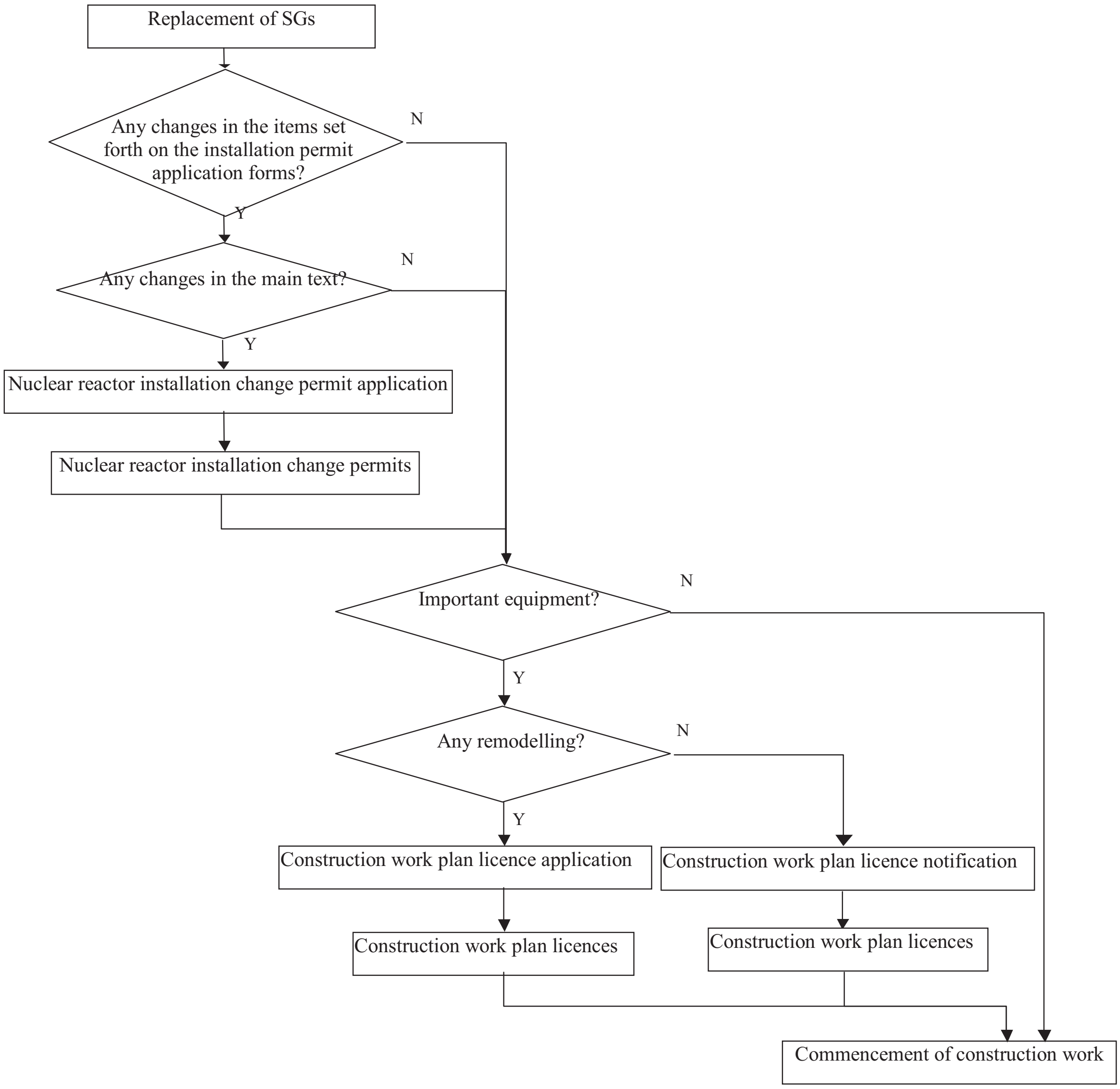

FIG. 68. Flow of permits and licences.

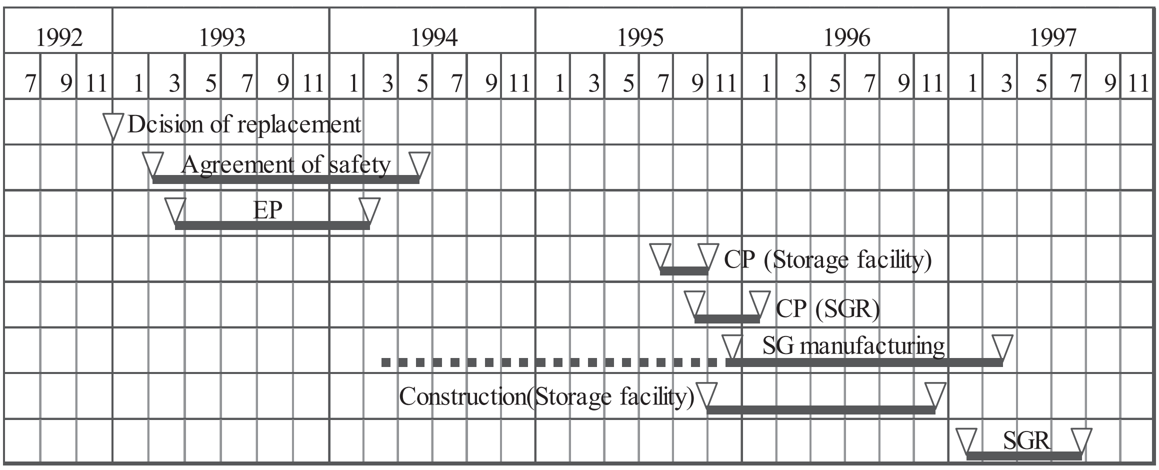

FIG. 69. Overview of permit and licence processes.

Appendix II

SG REPLACEMENT HISTORY (PWR) 1979–2005

Country	Plant	OSG designer	Reactor type	SG model	Year
France	DAMPIERRE 2	FRAMATOME	3 loop 900	47-22	2005
USA	ARKANSAS ONE 1	B&W	2 loops Model B 177 R	OTSG / OTSG	2005
USA	CALLAWAY 1	WESTINGHOUSE	Model 412	F / 73-19T	2005
USA	PALO VERDE 1	CE	2 loops - System 80	CE-80	2005
Belgium	DOEL 2	WESTINGHOUSE	2 loops W	44 / MHI	2004
France	TRICASTIN 4	FRAMATOME	3 loop 900	51M / 55-19	2004
USA	OCONEE 2	B&W	2 loops Model B 177 R	OTSG / OTSG	2004
USA	OCONEE 3	B&W	2 loops Model B 177 R	OTSG / OTSG	2004
USA	PRAIRIE ISLAND 1	WESTINGHOUSE	2 loops W	51 / 56-19	2004
France	ST LAURENT B-2	FRAMATOME	3 loop 900	51M / 55-19	2003
USA	CALVERT CLIFFS 2	CE	2 loops CE	CE-67 / x	2003
USA	OCONEE 1	B&W	2 loops Model B 177 R	OTSG / OTSG	2003
USA	PALO VERDE 2	CE	2 loops - System 80	CE-80 / x	2003
USA	SEQUOYAH 1	WESTINGHOUSE	4 loops – Ice containment	51 / D75	2003
France	FESSENHEIM 1	FRAMATOME	3 loop 900	51A / 47-22	2002
USA	CALVERT CLIFFS 1	CE	2 loops CE	CE-67 / x	2002
USA	SOUTH TEXAS 2	WESTINGHOUSE	Model 414	E / D94S	2002
Belgium	TIHANGE 2	FRAMATOME	3 loops FRA	51M / MHI	2001
France	TRICASTIN 3	FRAMATOME	3 loop 900	51M / 47-22	2001
Japan	IKATA 2	MHI	2 loops - 2nd generation	51M / MHI	2001
USA	FARLEY 2	WESTINGHOUSE	3 loops W	51 / x	2001
France	GRAVELINES B-4	FRAMATOME	3 loop 900	51M / 47-22	2000
USA	ARKANSAS ONE 2	CE	2 loops Model ***	*****	2000
Japan	GENKAI 2	MHI	2 loops - 2nd generation	51M / MHI	2000
Slovenia	KRSKO	WESTINGHOUSE	Model 212	D4,2 / FRA	2000
USA	DC COOK 1	WESTINGHOUSE	4 loops – Ice containment	51 / x	2000
USA	FARLEY 1	WESTINGHOUSE	3 loops W	51 / x	2000
USA	INDIAN POINT 2	WESTINGHOUSE	4 loops W	44 / 44F	2000
USA	SOUTH TEXAS 1	WESTINGHOUSE	Model 414	E / D94S	2000
Japan	IKATA 1	MHI	2 loops - 1st generation	51 / x	1999
Switzerland	BEZNAU 2	WESTINGHOUSE	2 loops W	33W / 33-19	1999
Belgium	TIHANGE 3	WESTINGHOUSE	Model 314	E1 / 70-19T	1998
France	TRICASTIN 1	FRAMATOME	3 loop 900	51M / 47-22	1998
Korea	KORI 1	WESTINGHOUSE	Model 212	51 / KHIC	1998
USA	BRAIDWOOD 1	WESTINGHOUSE	4 loops W	D4,1 / D4	1998
USA	SHEARON HARRIS 1	WESTINGHOUSE	Model 312	D75	1998
USA	ST LUCIE 1	CE	2 loops CE	CE-67 / 67	1998
France	TRICASTIN 2	FRAMATOME	3 loop 900	51M / 47-22	1997
Japan	OHI 2	WESTINGHOUSE	Model 412	51A / 54FA	1997
Spain	ALMARAZ 2	WESTINGHOUSE	Model 312	D3,2 / 61W-D3	1997

Country	Plant	OSG designer	Reactor type	SG model	Year
USA	BYRON 1	WESTINGHOUSE	4 loops W	D4,1 / D4	1997
USA	MCGUIRE 1	WESTINGHOUSE	4 loops – Ice containment	D2,1 / x	1997
USA	MCGUIRE 2	WESTINGHOUSE	4 loops – Ice containment	D3,1 / D3	1997
USA	SALEM 1	WESTINGHOUSE	4 loops W	51 / x	1997
Belgium	DOEL 4	WESTINGHOUSE	Model 314	E1 / 79-19T	1996
France	GRAVELINES B-2	FRAMATOME	3 loop 900	51M / 47-22	1996
Japan	MIHAMA 3	MHI	3 loops - 1st generation	51 / 54F	1996
Japan	TAKAHAMA 1	WESTINGHOUSE	Model 312	51 / 54F	1996
Spain	ALMARAZ 1	WESTINGHOUSE	Model 312	D3,1 / 61W-D3	1996
Spain	ASCO 2	WESTINGHOUSE	Model 312	D3,2 / 61W-D3	1996
USA	CATAWBA 1	WESTINGHOUSE	4 loops – Ice containment	D3,2 / CFR-80	1996
USA	R.E. GINNA	WESTINGHOUSE	2 loops W	44 / x	1996
USA	POINT BEACH 2	WESTINGHOUSE	2 loops W	44 / 44F	1996
Belgium	TIHANGE 1	FRAMATOME	3 loops FRA	51A / MHI	1995
France	DAMPIERRE 3	FRAMATOME	3 loop 900	51M / 47-22	1995
France	ST LAURENT B-1	FRAMATOME	3 loop 900	51M / 47-22	1995
Japan	MIHAMA 1	WESTINGHOUSE	2 loops W	CE-33 / 35F	1995
Spain	ASCO 1	WESTINGHOUSE	Model 312	D3,1 / 61W-D3	1995
Sweden	RINGHALS 3	WESTINGHOUSE	Model 312	D3,2 / 72W-D3	1995
USA	NORTH ANNA 2	WESTINGHOUSE	3 loops W	51D / 54F	1995
France	GRAVELINES B-1	FRAMATOME	3 loop 900	51M / 47-22	1994
Japan	GENKAI 1	MHI	2 loops - 1st generation	51 / 52F	1994
Japan	MIHAMA 2	MHI	2 loops - 1st generation	44 / 46F	1994
Japan	OHI 1	WESTINGHOUSE	Model 412	51A / 52FA	1994
Japan	TAKAHAMA 2	MHI	3 loops - 1st generation	51 / 52F	1994
USA	V.C. SUMMER 1	WESTINGHOUSE	3 loops W	D3,1 / D75	1994
Belgium	DOEL 3	FRAMATOME	3 loops FRA	51M / 61W	1993
France	BUGEY 5	FRAMATOME	3 loop 900	51A / 51B	1993
Switzerland	BEZNAU 1	WESTINGHOUSE	2 loops W	33W / 33-19	1993
USA	MILLSTONE 2	CE	2 loops CE	CE-67 / 67	1993
USA	NORTH ANNA 1	WESTINGHOUSE	3 loops W	51 / 54F	1993
USA	PALISADES	CE	2 loops CE	CE-67 / 67	1991
France	DAMPIERRE 1	FRAMATOME	3 loop 900	51M / 51B	1990
Sweden	RINGHALS 2	WESTINGHOUSE	3 loops W	C51 / 51	1989
USA	DC COOK 2	WESTINGHOUSE	4 loops – Ice containment	51D / 54F	1989
USA	INDIAN POINT 3	WESTINGHOUSE	4 loops W	44S / 44F	1989
USA	POINT BEACH 1	WESTINGHOUSE	2 loops W	44S / 44F	1984
USA	H.B. ROBINSON 2	WESTINGHOUSE	3 loops W	44S / 44F	1984
Germany	OBRIGHEIM (KWO)	KWU	2 loops KWU	KWU	1983
USA	TURKEY POINT 4	WESTINGHOUSE	3 loops W	51 / 44F	1983
USA	TURKEY POINT 3	WESTINGHOUSE	3 loops W	44S / 44F	1982
USA	SURRY 1	WESTINGHOUSE	3 loops W	51D / 51F	1981
USA	SURRY 2	WESTINGHOUSE	3 loops W	51D / 51F	1979

Appendix III

RVHR LIST FROM THE ORIGIN 1993–2005

Country	Unit	Original design/ new RVH supplier	Plant family	RVHR year
France	BUGEY 5	FRAMATOME/ FRAMATOME	type 900	1993
France	BLAYAIS 1	FRAMATOME/ FRAMATOME	type 900	1994
France	BUGEY 2	FRAMATOME/ FRAMATOME	type 900	1994
France	BUGEY 3	FRAMATOME/ FRAMATOME	type 900	1994
France	GRAVELINES B-4	FRAMATOME/ FRAMATOME	type 900	1994
France	BLAYAIS 2	FRAMATOME/ FRAMATOME	type 900	1995
France	BLAYAIS 3	FRAMATOME/ FRAMATOME	type 900	1995
France	FLAMANVILLE 1	FRAMATOME/ FRAMATOME	type 1300	1995
France	GRAVELINES B-3	FRAMATOME/ FRAMATOME	type 900	1995
France	SAINT ALBAN 1	FRAMATOME/ FRAMATOME	type 1300	1995
France	TRICASTIN 1	FRAMATOME/ FRAMATOME	type 900	1995
France	BLAYAIS 4	FRAMATOME/ FRAMATOME	type 900	1996
France	DAMPIERRE 1	FRAMATOME/ FRAMATOME	type 900	1996
France	FESSENHEIM 1	FRAMATOME/ FRAMATOME	type 900	1996
France	PALUEL 4	FRAMATOME/ FRAMATOME	type 1300	1996
France	SAINT ALBAN 2	FRAMATOME/ FRAMATOME	type 1300	1996
France	ST LAURENT B-2	FRAMATOME/ FRAMATOME	type 900	1996
France	TRICASTIN 4	FRAMATOME/ FRAMATOME	type 900	1996
Japan	TAKAHAMA 1	WESTINGHOUSE/ MHI	type 900	1996
Spain	ALMARAZ 1	WESTINGHOUSE/ WESTINGHOUSE	type 900	1996
Sweden	RINGHALS 2	WESTINGHOUSE/ MHI	3 loops W	1996

Country	Unit	Original design/ new RVH supplier	Plant family	RVHR year
France	BELLEVILLE 2	FRAMATOME/ FRAMATOME	type 1300	1997
France	BUGEY 4	FRAMATOME/ FRAMATOME	type 900	1997
France	CRUAS 4	FRAMATOME/ FRAMATOME	type 900	1997
France	DAMPIERRE 2	FRAMATOME/ FRAMATOME	type 900	1997
France	DAMPIERRE 4	FRAMATOME/ FRAMATOME	type 900	1997
France	GRAVELINES C-5	FRAMATOME/ FRAMATOME	type 900	1997
Japan	MIHAMA 3	MHI/ MHI	3 loops W	1997
Japan	TAKAHAMA 2	MHI/ MHI	3 loops W	1997
Spain	ALMARAZ 2	WESTINGHOUSE/ WESTINGHOUSE	type 900	1997
Spain	JOSE CABRERA (ZORITA)	WESTINGHOUSE/ WESTINGHOUSE	1 loop W	1997
France	CATTENOM 2	FRAMATOME/ FRAMATOME	type 1300	1998
France	CRUAS 2	FRAMATOME/ FRAMATOME	type 900	1998
France	DAMPIERRE 3	FRAMATOME/ FRAMATOME	type 900	1998
France	FESSENHEIM 2	FRAMATOME/ FRAMATOME	type 900	1998
France	FLAMANVILLE 2	FRAMATOME/ FRAMATOME	type 1300	1998
France	PALUEL 3	FRAMATOME/ FRAMATOME	type 1300	1998
Belgium	TIHANGE 1	FRAMATOME/ FRAMATOME	type 900	1999
France	CATTENOM 1	FRAMATOME/ FRAMATOME	type 1300	1999
France	CATTENOM 3	FRAMATOME/ FRAMATOME	type 1300	1999
France	GRAVELINES B-1	FRAMATOME/ FRAMATOME	type 900	1999
France	TRICASTIN 2	FRAMATOME/ FRAMATOME	type 900	1999
Japan	MIHAMA 2	MHI/ MHI	2 loops W	1999
Japan	OHI 2	WESTINGHOUSE/ MHI	4 loops W	1999
France	BELLEVILLE 1	FRAMATOME/ FRAMATOME	type 1300	2000

Country	Unit	Original design/ new RVH supplier	Plant family	RVHR year
France	CHINON B-2	FRAMATOME/ FRAMATOME	type 900	2000
France	GRAVELINES B-2	FRAMATOME/ FRAMATOME	type 900	2000
France	NOGENT 1	FRAMATOME/ FRAMATOME	type 1300	2000
Japan	OHI 1	WESTINGHOUSE/ MHI	4 loops W	2000
France	GRAVELINES C-6	FRAMATOME/ FRAMATOME	type 900	2001
France	NOGENT 2	FRAMATOME/ FRAMATOME	type 1300	2001
France	PALUEL 2	FRAMATOME/ FRAMATOME	type 1300	2001
Japan	GENKAI 1	MHI/ MHI	2 loops W	2001
Japan	GENKAI 2	MHI/ MHI	2 loops W	2001
Japan	IKATA 1	MHI/ MHI	2 loops W	2001
Japan	MIHAMA 1	WESTINGHOUSE/ WESTINGHOUSE	2 loops W	2001
Japan	IKATA 2	MHI MHI	2 loops W	2002
USA	NORTH ANNA 2	WESTINGHOUSE/ FRAMATOME	3 loops W	2002
China	GUANGDONG 2	FRAMATOME/ FRAMATOME	type 900	2003
France	CATTENOM 4	FRAMATOME/ FRAMATOME	type 1300	2003
Spain	ASCO 1	WESTINGHOUSE/ WESTINGHOUSE	type 900	2003
USA	CRYSTAL RIVER 3	B&W/ FRAMATOME	B&W	2003
USA	R.E. GINNA	WESTINGHOUSE/ B&W	2 loops W	2003
USA	NORTH ANNA 1	WESTINGHOUSE/ FRAMATOME	3 loops W	2003
USA	OCONEE 1	B&W/ B&W	B&W	2003
USA	OCONEE 3	B&W/ B&W	B&W	2003
USA	SURRY 1	WESTINGHOUSE/ FRAMATOME	3 loops W	2003
USA	SURRY 2	WESTINGHOUSE/ MHI	3 loops W	2003
USA	Three Mile Island 1	B&W	B&W	2003

Country	Unit	Original design/ new RVH supplier	Plant family	RVHR year
China	GUANGDONG 1	FRAMATOME/ FRAMATOME	type 900	2004
France	ST LAURENT B-1	FRAMATOME/ FRAMATOME	type 900	2004
France	TRICASTIN 3	FRAMATOME/ FRAMATOME	type 900	2004
Spain	ASCO 2	WESTINGHOUSE/ WESTINGHOUSE	type 900	2004
Sweden	RINGHALS 4	WESTINGHOUSE/ MHI	type 900	2004
USA	FARLEY 1	WESTINGHOUSE/ MHI	3 loops W	2004
USA	KEWAUNEE	WESTINGHOUSE/ MHI	2 loops W	2004
USA	OCONEE 2	B&W/ B&W	B&W	2004
USA	TURKEY POINT 3	WESTINGHOUSE/ FRAMATOME	3 loops W	2004
France	GOLFECH 1	FRAMATOME/ FRAMATOME	type 1300	2005
France	PENLY 1	FRAMATOME/ FRAMATOME	type 1300	2005
Sweden	RINGHALS 3	WESTINGHOUSE/ MHI	type 900	2005
USA	ARKANSAS ONE 1	B&W/ FRAMATOME	B&W	2005
USA	FARLEY 2	WESTINGHOUSE/ MHI	3 loops W	2005
USA	MILLSTONE 2	CE/ MHI	2 loops CE	2005
USA	POINT BEACH 1	WESTINGHOUSE/ MHI	2 loops W	2005
USA	POINT BEACH 2	WESTINGHOUSE/ MHI	2 loops W	2005
USA	PRAIRIE ISLAND 2	WESTINGHOUSE/ MHI	2 loops W	2005
USA	H.B. ROBINSON 2	WESTINGHOUSE/ MHI	3 loops W	2005
USA	SALEM 1	WESTINGHOUSE/ FRAMATOME	4 loops W	2005
USA	SALEM 2	WESTINGHOUSE/ FRAMAOTOME	4 loops W	2005
USA	ST LUCIE 1	CE/ FRAMATOME	2 loops CE	2005
USA	TURKEY POINT 4	WESTINGHOUSE/ FRAMATOME	3 loops W	2005
Japan	OHI 3	MHI/ MHI	4 loops W	2005

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