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REVISION HISTORY

Issue	Description	Date
00	First issue for INSA information	14-01-2008
01	Integration of technical and co-applicant comments	29-04-2008
02	PCSR June 2009 update including - Clarification of text - Addition of references - Technical updates notably regarding RCP flywheel (section 1 and manufacturing process of the primary pump casing (new section 1.5), pressuriser surge line (section 3), pipework cobalt content (section 3) and auxiliary spray line diameter (section 4)	29-06-2009
03	Consolidated Step 4 PCSR update: - PROTECT COMMERCIAL markings removed (no CCI in document) - Minor editorial changes - References added and/or updated (English translations) - Addition of specific section (§1.1.4) to describe oil collecting device for Reactor Coolant Pump - Addition of specific sections to address fast fracture risk for High Integrity Components : Reactor Coolant Pump (§1.6), Steam Generator (§2.10), Main Coolant Lines (§3.9), Pressuriser (§4.7) - Introduction of 20MND5 steel grade for Steam Generator and Pressuriser (§2.5.1) - Secondary side water chemistry treatment information moved from §5.4.2 to Sub-chapter 5.5	31-03-2011

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REVISION HISTORY (Cont'd)

Issue	Description	Date
04	Consolidated PCSR update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Minor editorial changes - Scope of fracture mechanics analysis clarified in the introduction to §1.6 (RCP flywheel), §2.10 (SG), §3.9 (MCL) and §4.7 (pressuriser) - Critical defect size added for RCP casing in §1.6 and Life Fatigue Crack Growth added for RCP flywheel in §1.6 - Clarification on the scope of qualified NDT applies to RCP casing large repair welds in §1.6 - Update of text on capability of manufacturing inspection techniques for the RCP flywheel (§1.6), the steam generator dissimilar metal welds (§2.10) and the MCL homogeneous welds (§3.9). - Introduction of the MCL design modifications CMF031 and CMF032 in §3.9 - Addition and update of references	30-08-2012
05	Consolidated PCSR update: - Update of text to supplement the safety case with fast fracture methodology applied to any repair welds and requirement to define the threshold for application of qualified volumetric inspections (§1.6) - Modifications to clarify application of "break preclusion" and HIC (§9)	31-10-2012

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SUB-CHAPTER 5.4 – COMPONENTS AND SYSTEMS SIZING

Sub-chapter 5.4 provides a description of the main reactor coolant systems and components, including as appropriate: the relevant operating conditions and interfaces; the design criteria to be applied; materials and material properties; design details and calculations; safety evaluation and assessment of mechanical integrity in accident conditions; manufacturing and inspection details. The systems and components covered include: the reactor coolant pumps, the steam generators, the reactor coolant pipework, the pressuriser and pressuriser relief line, valves associated with the reactor coolant pressure boundary, pressuriser pressure safety relief valves and severe accident depressurisation valves, and the primary component supports.

1. REACTOR COOLANT PUMPS

The supplier for the EPR reactor coolant pumps is JSPM, a subsidiary company of AREVA NP. This company is the supplier for the previous EPR project, and has already manufactured reactor coolant pumps similar in design to that proposed for the EPR.

1.1. DESCRIPTION OF REACTOR COOLANT PUMPS

The reactor coolant pumps are categorised as RCC-M level 1 (see Sub-chapter 3.8). The detailed list of level 1 reactor coolant pump assembly components is shown in Section 5.4.1 - Table 3.

The design of EPR reactor coolant pumps is based on the N4 reactor coolant pumps.

The general assembly is given in Section 5.4.1 - Figure 1.

Each reactor coolant pump consists of a mixed flow single-stage vertical pump, designed to pump large volumes of reactor coolant at high pressure and temperature.

The three major elements are the pump, the shaft seals and the motor.

- The hydraulic unit is made up of the casing, an impeller, a diffuser and a suction adapter. The other elements of the pump are the shaft, the main bearing and its auxiliary bearing, the thermal barrier flange and heat exchanger, the thermal barrier cover fitted with thermal shields, the main flange, the coupling, the spool piece and the motor support.

- The pump shaft sealing system is made up of three seals arranged in series and a standstill seal system. The first shaft seal is a controlled leak-off seal, which is a film riding face seal. The second and third seals are rubbing face seals. The standstill seal system provides a seal with metal-to-metal contact ensuring the shaft is leak tight once the pump is shut down and all the leak-off lines are closed. The standstill seal system is activated for some accidental conditions (see section 1.1.2 of this sub-chapter).

- The motor is a squirrel cage induction motor, protected from water spray, with a solid shaft, a double thrust bearing of the Kingsbury oil-lubricated type, upper and lower oil film radial guide bearings and a flywheel.

All parts in the reactor coolant pump can be replaced, with the exception of the casing, which is welded to the reactor coolant loop.

The reactor coolant pump is supported vertically by three support columns fixed to the three casing support lugs, and restrained horizontally by two snubbers fixed to the upper flange of the motor support. A restraint is also fitted in front of the casing support foot opposite the cold leg.

1.1.1. Pump

Reactor coolant enters the suction nozzle of the casing, is directed towards the impeller by the suction adapter, passes through the impeller and exits through the diffuser and the discharge nozzle.

All parts of the pump in contact with reactor coolant are manufactured from stainless steel, except for the seals, bearings and special parts. The material grades are detailed in Section 5.4.1 - Table 2.

The casing is an integral casting made from austeno-ferritic stainless steel. Casting has previously been successful in producing the complex shape of the pump casing. This method of producing casings has been selected because it guarantees good in-service behaviour, manufacturing quality and reliability conditions (see section 1.5 of this sub-chapter). The impeller is fixed to the pump shaft by a Hirth-type tooth gearing and studs (around the perimeter and in the centre). The diffuser is bolted to the base of the thermal barrier flange.

An injection flow to the seals is supplied by the chemical and volume control system (RCV [CVCS]) at a pressure slightly above that of the reactor coolant. It enters the pump through a pipe on the thermal barrier flange, and is directed into the cavity located between the thermal barrier and the shaft seals. This flow divides into two, one part flowing up the shaft through the seals, the other flowing down the shaft through the thermal barrier heat exchanger and the auxiliary bearing, where it mixes with the reactor coolant.

The Component Cooling Water System (RRI [CCWS]) supplies cooling water to the thermal barrier heat exchanger. During normal operation the thermal barrier heat exchanger limits the heat transfer between the hot reactor coolant and the auxiliary bearing and the seals. If a loss of injection flow to the seals occurs, the thermal barrier heat exchanger cools the reactor coolant flowing up the shaft to an acceptable level before it comes into contact with the seals.

The reactor coolant pump main radial bearing, located on the diffuser, is a hydrostatic bearing and is lubricated by existing fluid in the casing. An auxiliary hydrodynamic bearing, located on the thermal barrier heat exchanger, takes over from the main radial bearing if pressure and temperature transients in the primary circuit impair its operation.

The reactor coolant pump internal components can be easily removed from the casing.

The spool piece, located between the pump shaft and the motor shaft, enables maintenance work on the shaft seals system without removing the motor. Seal maintenance work can therefore be carried out quickly. (An oil injection method is used to assemble the coupling on the pump shaft).

The studs and nuts for the casing and seal N° 1 housing are designed to be tightened by hydraulic tensioning or by a heating technique.

1.1.2. Shaft sealing system

The shaft sealing system, shown in Section 5.4.1 - Figure 2, reduces the pressure from reactor coolant pressure to ambient pressure.

Seal N° 1 provides the majority of the pressure drop, with a controlled leak-off to the RCV [CVCS].

Seals N° 2 and 3 provide the remaining pressure drop, with a small leak to the vents and drains system (RPE [NVDS]).

In the event of a failure of seal N° 1, seal N° 2 takes over for a limited period enabling the reactor coolant pump to be stopped and the reactor to be shut down or the standstill seal system to be activated.

Seal N° 3 makes a minor contribution to the pressure drop with a negligible leakage. A flushing flow downstream of this seal is provided by the demineralised water system. The small leak from seal N° 3 is purged with a flushing flow to the nuclear vents and drains system RPE [NVDS].

The standstill seal system ensures leak tightness along the shaft when the reactor coolant pump is stopped, in the following events:

- simultaneous loss of water supply from the RCV [CVCS] and the RRI [CCWS] used to cool the shaft sealing system (for example, during a total loss of the electrical power supply),
- a cascading failure of all stages of the shaft sealing system.

The design of the sealing system ensures that seals N° 2 and 3 and the standstill seal system can be installed or removed at the same time (cartridge design).

1.1.3. Motor

The motor is an air-cooled squirrel cage induction motor, with class F thermo-elastic epoxy insulation. A flywheel and reverse anti-rotation device are fitted in the upper section of the motor.

The bearings are of standard design. The radial bearings are pad type bearings, and the thrust bearing comprises a Kingsbury-type double thrust bearing. All bearings are oil lubricated. Water from the RRI [CCWS] feeds the external oil cooler for the upper guide bearing and the integrated oil cooler for the lower guide bearing.

A high-pressure injection system provides an oil film in the thrust bearing surfaces before and during start up and shutdown. It also sprays oil into the upper guide bearing.

The motors’ internal components are air-cooled. Integrated fans at each end of the rotor draw air in through inlets in the frame of the motor. This air circulates within the motor and in particular to the stator end windings. It is then discharged through the outer water/air heat exchangers cooled by the RRI [CCWS].

Each motor is fitted with two heat exchangers, one each side of the motor frame. Air circulates from the motor through the heat exchangers, and is then discharged to the pump cell. The heat exchangers are designed to maintain the discharged air at an optimal temperature. Any heat not removed by the coolers is discharged into the containment area, and is extracted by the containment ventilation system, EVR [CCVS].

The flywheel can be inspected by removing its cover.

1.1.4. Oil collecting devices

The total oil inventory contained in the bearings of the reactor coolant pump motor is distributed in the following locations:

- upper bearing
- oil cooler
- lower bearing

The sectional drawing of the reactor coolant pump motor in Section 5.4.1 - Figure 3 illustrates these locations together with corresponding oil volumes.

The reactor coolant pumps are conservatively designed to reduce the likelihood of an oil leakage. In the event of a leak occurring, measures are provided to reduce the volume of oil released onto the floor, reducing the likelihood for the outbreak of fire. These are:

- Detection systems in the reactor coolant pumps which monitor parameters that will provide an early indication of a leak. The level of oil in the bearings of the reactor coolant pumps is monitored constantly. A decrease in the level of oil will trigger a ‘low oil level’ alarm in the main control room. The affected pump can then be manually switched off and the cause of the alarm investigated.
- Oil collection devices which are attached to the coolant pump body at all potential leak sources. Under normal operating conditions no oil is present in these oil collecting devices.

1.1.5. Instrumentation and control

Each reactor coolant pump is fitted with the following instrumentation for monitoring the main operational parameters:

- Two temperature sensors at the inlet of the shaft seal N° 1.
- Two position sensors for the standstill seal system.
- Two temperature sensors for the motor’s lower radial bearing pads.
- Two temperature sensors for the motor’s upper radial bearing pads.
- Four temperature sensors for each phase of the motor’s stator.
- Two temperature sensors for the upper pads of the motor axial double thrust bearing.

- Two temperature sensors for the lower pads of the motor axial double thrust bearing.
- One pressure gauge and one pressure switch on the motor's double axial thrust bearing oil injection system.
- Two shaft displacement sensors at the level of the motor coupling.
- Two vibration sensors on the motor lower flange.
- Two rotary speed displacement detectors (tachometer) at the level of the motors coupling.
- One low speed sensor.
- Fitted to each oil chamber are:
    - one high and one low level switch
    - one analogue oil level gauge,
    - one visual level gauge,
    - one temperature gauge.

1.2. DESIGN BASES [REF-1] [REF-2]

The reactor coolant pump is considered to be a "non-breakable" component, and therefore satisfies all design requirements described in Sub-chapter 5.2, section 6.

- The main design parameters for the reactor coolant pumps are given in Section 5.4.1 - Table 1 (manometric head at nominal flow rate).
- The pump's hydraulic casing design is extrapolated from the N4 design. It is able, to withstand additional variations of pressure in the Reactor Coolant System (RCP [RCS]) so that the developed head, at nominal flow rate, is within the design limits indicated in Section 5.4.1 - Table 1.
- Design of the EPR reactor coolant pump hydraulic casing is defined by an advanced flow computation code and confirmed by many scale tests on a hydraulic model on the reactor coolant pump supplier's test loop. During the plant commissioning phase, initial tests of the RCP [RCS] will be carried out in order to verify the hydraulic performance of the reactor coolant pump.
- A tolerance range for the reactor coolant pump characteristics, combined with uncertainty about the hydraulic pressure drop of the RCP [RCS] loop, results in a defined range of the reactor coolant pump performance encompassing the nominal flow rate. The minimum (thermo-hydraulic) and maximum (mechanical) design flow rates for the RCP [RCS] are defined to envelope this range (see Section 5.4.1 - Table 1).

- The reactor coolant pump rotor equipped with its flywheel provides sufficient inertia:
  - o to ensure that the house load operation threshold signal is not activated for at least one second in the event of loss of motor power supply. In addition, the period between reaching this house load operation signal and the automatic reactor shutdown signal (on reactor coolant pump low speed) must be greater than 0.3 seconds,
  - o to ensure the appropriate flow rate, and therefore sufficient Departure from Nucleate Boiling (DNB) margins before the automatic shutdown of the reactor (triggered by the reactor coolant pump low speed) in the event of a reactor coolant pump coastdown transient condition.

Inertia defined in Section 5.4.1 - Table 1 is provided mainly by a flywheel located at the top of the motor shaft. The flywheel design integrates several precautions at different levels (material procurement, integrity analysis, in factory tests, in-service inspections) in order to prevent any risk of failure which can lead to loose part generation.

- The pump casing is designed for a 60-year life. The intended lifespan of the other components is also 60 years, with the exception of certain wear parts which will be replaced periodically (bearings, shaft seals, 0-rings) due to wear.
- The reactor coolant pumps are designed to operate in the following conditions:
  - o when the reactor is operating with three reactor coolant pumps in service, and for the start-up of the shutdown reactor coolant pump when the other three are in service,
  - o under cold and hot shutdown conditions with one to four reactor coolant pumps in service, and for the start-up of one reactor coolant pump when the other three are in service during the cooling of the reactor between hot shutdown and the transition to the residual heat removal mode,
  - o in the event of an increase in the reactor coolant system pressure up to the design RCP [RCS] pressure (corresponding to cold leg pressure of approximately 160 bar abs).
- The coastdown capacity of the reactor coolant pump is ensured in the conservative event of a power loss coinciding with a design earthquake.
- In the event of LOCA, a reactor coolant pump automatic shutdown signal is initiated. The shutdown signal is “Low ΔP across the reactor coolant pump and a SI signal”. The combination with the SI signal is intended to avoid a spurious reactor coolant pump trip. The ΔP corresponds to the pressure difference between the reactor coolant pump inlet (crossover leg pressure) and the reactor coolant pump outlet (cold leg pressure).
- The reactor coolant pumps are designed to withstand the following incidents without damage:
  - o loss of water injection from the RCV [CVCS] at seal N° 1, with the pump in continuous operation or at shutdown,
  - o loss of water from the RRI [CCWS] cooling the thermal barrier heat exchanger with the reactor coolant pump operating or at shutdown,

- o simultaneous loss of injection at seal N° 1 and the thermal barrier cooling, if one of these two functions is re-established in less than two minutes.

1.3. SEAL SYSTEM SAFETY EVALUATION

The standstill seal system is classified as a F1B system and hence the single failure criterion can be considered at the level of its function (see Sub-chapter 3.2 - Table 2). The corresponding functional redundancy is provided by the thermal barrier cooling RRI [CCWS], also classified F1B.

The shaft sealing system is comprised of three seals arranged in series and a standstill seal system.

The design of seals N° 1 and N° 2 is identical to that used on the N4 and CP 1300 plant reactor coolant pump assemblies, which have had good operational experience. The design of seal N° 3 is very similar to that used on 900 MW plants' reactor coolant pumps.

Some improvements have however been adopted to comply with the EPR specification:

- Pump operation at low pressure when the RIS/RRA [SIS/RHRS] is connected to the RCP [RCS] and is operating in residual heat removal mode.
- Absence of a back-up system for rapid injection at the shaft seals in the event of total loss of electrical power.
- Standstill seal system which can be activated when the pump is shutdown.

Consequently, the shaft sealing system and the standstill seal system are fitted with 'O' rings manufactured with a grade of material qualified for high pressures and temperatures. The seal N° 1 faces are made from silicon nitride.

Seal N° 1 makes the major contribution to the pressure drop with a controlled leak-off (controlled leak-off seal, water film type), discharged to the RCV [CVCS]. Under normal operation, this seal is fed from the RCV [CVCS], via the injection line at seal N° 1. In the event of loss of water injection from the RCV [CVCS], it is fed by reactor coolant cooled by the thermal barrier heat exchanger.

Seals N° 2 and 3 (rubbing face seals) provide the remaining pressure drop, with a negligible leak-off discharged to the RPE [NVDS].

Seal N° 2 acts as a back up to seal N° 1 in the event that the latter fails, and is designed to provide seal function for at least thirty minutes with the pump rotating and for 24 hours when the pump is shut down. A failure in seal N° 1 is detected by the flow meter located on the leak-off line. In this event the leak-off line is automatically isolated, and the reactor power is reduced to an acceptable level, allowing the faulty reactor coolant pump to be shut down. Once the reactor coolant pump is shutdown, the plant is then shutdown, the standstill seal system is activated and all other leak-off lines are closed.

The standstill seal system is located on the upper section of the shaft sealing system, above seal N° 3.

Once the pump is shut down, a piston ring, actuated by a low-pressure nitrogen supply, closes the air gap between the shaft and this ring, and creates a leak-tight metal to metal surface contact. This ensures that the shaft is leak-tight once the pump is shut down and all the leak-off lines closed (these lines are closed off in the following order: seal N° 3, seal N° 2 and lastly seal N° 1).

The standstill seal system is designed to be leak-tight in the event of:

- Simultaneous loss of water supply from the RCV [CVCS] and the RRI [CCWS] used to cool the shaft sealing system during a Station Black Out (SBO).
- A cascade failure of all stages in the shaft sealing system.

In the event of a SBO, the standstill seal system and the leak off line isolating valve for the three shaft seals are automatically closed once the reactor coolant pump is shutdown (see Section 5.4.1 - Figure 2).

The standstill seal system is designed to:

- Close, once activated, in the event of the pressure and temperature resulting from SBO.
- Isolate the significant leak which would result from a cascade failure of the shaft seals.
- Prevent damage to the shaft sealing system in the event of inadvertent closure of the standstill seal when the pump is running.
- Prevent auto-closure in the event of a cascade failure of the shaft seal sealing system.
- Remain leak-tight once activated until a very low reactor coolant pressure is reached, even if nitrogen supply pressure is lost.

1.4. MECHANICAL INTEGRITY IN ACCIDENT CONDITIONS

Rotor over-speed

The three main transient conditions that may lead to over-speed are:

- Grid over-frequency.
- Disconnection of the plant from the grid.
- RCP [RCS] break combined with a loss of power supply. The most disadvantageous LOCA for the reactor coolant pump is a break in the surge line where it is connected to the RCP [RCS] loop. With a break of this size, the over-speed of the reactor coolant pumps remains below the design value.

The reactor coolant pump is designed for an over-speed of 25% relative to normal speed.

The motor in the reactor coolant pump (including the flywheel) is tested at this 25% over-speed.

The flywheel is made up of two thick plates bolted together; this conception is made in order to limit the size of the potential loose parts released in the event of flywheel failure. It is lightly shrunk onto the shaft, and three special keys provide transmission to the coupling.

The material chosen for the flywheel (20NCD14-07, technical procurement specification M2321 of RCC-M) is a high toughness material (K_{1C}= 200 MPa $\sqrt{m}$). This material can be subjected to a high stress intensity factor. Stringent criteria are used for the volumetric and surface examination (both before and after final machining) during the manufacturing of the flywheel disks.

The flywheel design is justified in a design report which analyses the following failure modes:

- ductile failure
- brittle fracture
- Crack depth propagation (performed for the most adverse location, in the keyway).

Six holes in the flywheel allow periodic ultrasonic in-service inspection of the most highly stressed areas, located in the keyway corners, without the need to remove the flywheel.

These provisions reduce the risk of a missile within the containment area caused by a flywheel failure.

Analysis of the mechanical performance of the casing

The reactor coolant pump casing is very similar to the already proven N4 design:

- The inner diameter is slightly increased, which results in a slight increase in the wall thickness.
- The suction and discharge nozzle diameters have been adapted to the diameter of the RCP [RCS] loop, principally resulting in a slight alteration to the dimensions of the discharge cone.
- The radial thickness of the upper flange has been increased, as has the diameter of the casing studs.

An elastic analysis of the casing has been carried out in accordance with B3230 RCC-M rules, using a finite element method applied to a three-dimensional model of the casing, in order to verify the design conditions below:

- design conditions (level 0 criterion).
- normal operating conditions (fatigue analysis).
- fault conditions (level D criterion).
- hydrotest conditions.

UK EPR specific fast fracture calculations have been defined in the frame of the safety demonstration of High Integrity Components, as described in section 1.6.

1.5. MANUFACTURING PROCESS OF THE PRIMARY PUMP CASING

Explanation of benefits and disadvantages of forging versus casting approach

The primary pump casing is the only cast component in the EPR primary loops, all other components being forgings. In 1995, a feasibility study was conducted by AFCEN (the French Association for the Design, Construction and Operating Supervision of equipment for ElectroNuclear boilers), in order to assess the possibility of manufacturing forged stainless steel casings that complied with AREVA's requirements in terms of hydraulics, mechanics, metallurgy, geometry and controllability; many benefits were anticipated in adopting the forging solution, including the avoidance of casting defects, the avoidance of weld repairs and the easier control of the welds with the forged primary piping. However, almost all the suppliers that were consulted declined the offer because of the weight of the ingot needed and the complex manufacturing sequence. In addition, the only favourable answer showed that the forging option had many metallurgical disadvantages:

- The 3 support feet would have to be welded to the casing, hence creating difficulties for welding and on-site inspection.
- The outlet nozzle would necessarily be shorter, resulting in a longer cold leg and a weld closer to the casing which would be harder to control.
- The forging sequence would lead to a low forging ratio.
- The volumetric control of the part would have to be performed through ultrasonic examination, which would be very difficult considering the thickness of the part.

Consequently, casting appears as the most reasonably practicable option for the manufacturing of the pump casing; it must also be emphasised that the casting solution leads to a very satisfactory product with good mechanical properties and without many major defects – a result that is achieved by applying the requirements of RCC-M M140/M160 qualification.

Progress in casting quality compared with earlier designs and experience feedback from manufacturing of EPR pump casings

In practice, the supplier must develop an appropriate casting method, in order to control the solidification of the steel and to remove potential heterogeneities from the casing. The main defects that can be encountered on cast products are inclusions, porosities and shrinkages. Many improvements (concerning the manufacturing process, the foundry method, heat treatment and quench facilities…) have been made in recent decades to reduce the size and number of such defects. This has led to an improvement of the quality of castings in austenitic-ferritic stainless steel compared to that achieved previously.

The results obtained so far on the castings manufactured by two different suppliers for the OL3 and FA3 EPR reactors (under construction respectively in Olkiluoto, in Finland and in Flamanville, in France) confirm the good quality of these parts. As shown in the table below, the first supplier (A), who has a long experience of such manufacturing processes, was able to provide a casing without any major defects. The second supplier (B) was qualified more recently to produce EPR pump casings; after slight modifications of the casting method following results obtained on a prototype part, this supplier succeeded in manufacturing castings with less than four major defects deeper than 35 mm (and only one major defect detectable through radiographic examination on four casings). In all the EPR casings, the mass percentage ratio was lower than 1% of the weight of the casing. In addition, tests showed that the parts had satisfactory and homogeneous mechanical properties.

Contract	Supplier	Pump casing reference	Number of major excavations with depth ≥ 35 mm	
			From dye penetrant testing	From Radiographic Testing
OL3	Supplier B	1	0	12
		2	0	0
		3	1	0
	Supplier A	4	0	0
FA3	Supplier B	1	0	1
		2	4	0
		3	1	(RT not performed so far)
		4	2	(RT not performed so far)

In line with this technological progress, several criteria are applied to limit the extent of defects (and associated repairs) and ensure the quality of the final product:

- A supplier whose casting method leads to an unsatisfactory integrity of the casing will not be qualified under RCC-M M140/M160 (the qualification process is applied and validated under the supervision of an independent safety authority).
- For the FA3 EPR reactor, the percentage ratio of repairs by welding was limited to 1.5% of the weight of the casing (a limitation that all earlier manufacturers would not have been able to fulfil)
- All excavations are repaired according to valid qualified welding procedures, which ensure that the tensile properties of the deposited metal are at least as good as those of the base metal.

Details of the NDE techniques used

The volume of the casing is controlled by radiographic examination. Approximately 600 individual exposures with X-Rays are used to cover the entire volume of the casing. The analysis of the radiographic films is carried out by comparison with ASTM reference boards: cast type indications are tolerated within severity levels 1 or 2 whereas cracks are unacceptable. Only two very small areas located near the support feet cannot be controlled using radiographic examination; these areas are controlled using ultrasonic examination using longitudinal waves in accordance with RCC-M.

All inner and outer surfaces are controlled by liquid penetrant examination. The notification level is 2 mm; all linear indications are unacceptable and all rounded indications (such as porosities) with any dimension greater than 5 mm are also unacceptable.

For casing repair welds both radiographic and ultrasonic examinations will be performed as described in section 1.6. Extensive tests performed on deep repair welds of a cast Reactor Coolant Pump casing mock-up demonstrated that both these techniques can be used to detect and reject defects of structural concern.

Specific requirements used in addition to RCC-M

The manufacturing of the pump casing must meet the requirements of the RCC-M as well as the additional requirements applied for the FA3 EPR casings that are listed below:

- Cobalt content < 0.1% (target 0.05%).
- Ferrite content calculated from the ladle analysis between 12 and 20%.
- Required KV at room temperature (average value) ≥ 160 J.
- Rm at room temperature ≤ 800 MPa.
- Mass percentage ratio of metallurgic repair by welding ≤ 1.5%

1.6. FAST FRACTURE ANALYSIS

As stated in Sub-chapter 3.1, the Reactor Coolant Pump is a High Integrity Component for which the specific measures described in section 0.3.6 of Sub-chapter 3.4 concerning prevention, surveillance and mitigation contribute to its high integrity demonstration.

With regards to the fast fracture risk, the three-legged approach presented in Sub-chapter 3.4 section 1.6 has been applied to the Reactor Coolant Pump casing and the Reactor Coolant Pump flywheel as follows:

- use of fracture mechanics to determine the end of life limiting defect size and demonstration that the Defect Size Margin between this critical defect size and the detectable defect increased by fatigue crack propagation over the lifetime is larger than 2 as far as practicable;
- use of suitable redundant and diverse inspections during manufacturing, supplemented by the use of qualified inspection(s) at the end of manufacturing;
- verification of the lower bound fracture toughness values used to determine the critical defect size by measurements.

This methodology applies to the whole Reactor Coolant Pump casing, including welds and base metal. However, considering the higher probability of finding crack-like defects in repair welds than in castings and considering that the toughness of weld metal is lower, the demonstration on repair welds covers the demonstration on base metal, assuming that repairs can occur anywhere in the casing. Consequently, the following sub-sections focus on repair welds.

The methodology also applies to the Reactor Coolant Pump flywheel; however, as explained in sub-section 1.6.2 of Sub-chapter 5.4, the manufacturing inspections of the flywheel are not qualified.

1.6.1. Reactor Coolant Pump casing

For the Reactor Coolant Pump casing (see Section 5.4.1 - Figure 4), due to the higher sensitivity to fast fracture of deep repairs compared to shallow repairs (residual stress, toughness), the assessment of repairs during GDA has focussed on deep repairs (i.e. thickness greater than 35 mm according to the RCC-M code). The following sections present the application of the HIC methodology to this first category of deep repair welds, which are bounding cases regarding critical defect size and controllability. The application of this approach to all repair welds is then presented.

Fracture Mechanics Analysis

A Fracture Mechanics Analysis (FMA) has been performed for deep repairs of the Reactor Coolant Pump casing [Ref-1]. In addition to residual stress, two sets of loadings have been considered: mechanical loadings corresponding to the quadratic combination of connected line break and Design Basis Earthquake, and thermal loadings corresponding to the most severe transients of categories A and B firstly, and C and D secondly.

These loadings have been applied to the most loaded areas of the pump casing, determined on the basis of elastic stress calculations. The residual stresses have been established on the basis of measurements performed on a mock-up with the Deep Hole Drilling (DHD) technique.

The Stress Intensity Factors with plastic correction have been compared to fracture toughness values specific to repairs performed using manual metal arc welding (SMAW) with stainless steel weld metal: a lower bound fracture toughness value has been derived from tests performed using Compact Tensile specimens (CTJ) on a Reactor Coolant Pump casing mock-up after 3000 hours ageing at 400°C [Ref-2].

Finally, an End of Life Limiting Defect Size greater than 20 mm has been determined in deep repair welds for any loading case, using an analytical formula for mechanical loadings and elastic-plastic Finite Element calculations for thermal loadings. This result obtained using ductile tearing criteria for the C and D category bounding transient has been confirmed with the A and B category bounding transient with initiation criteria.

Considering:

- the lower level of residual stress in shallow repairs compared to deep repairs,
- the smaller area of reduced toughness in Reactor Coolant Pump casing containing shallow repairs compared to Reactor Coolant Pump casing containing deep repairs (the toughness of repairs being lower than the toughness of austenitic-ferritic casting),

shallow repairs are less sensitive to fast fracture compared to deep repairs. As a consequence, the End of Life Limiting Defect Size greater than 20 mm determined for deep repairs is applicable to shallow repair welds (minor and surface repairs according to the RCC-M code).

Non destructive tests (NDT)

For the Reactor Coolant Pump casing, the NDT selected to be qualified during GDA for inspection of the deep repair welds of the Reactor Coolant Pump casing are Radiographic Testing (RT) supplemented for the outer 25 mm depth by pulse-echo Ultrasonic Testing (UT) with inclined longitudinal waves and sizing of the length of the defect [Ref-3].

Measurements have been made on a Reactor Coolant Pump cast casing mock-up; the tests performed using UT and RT enable all embedded defects to be detected and thus demonstrate that these techniques are capable of detecting and rejecting defects of 10 mm through-wall extent or greater in deep repair welds.

Finally, a Defect Size Margin of 2 has been attained with fracture mechanics calculations while considering that the fatigue crack growth for this worst case defect with regards to fast fracture is negligible; this assumption will be verified during detailed design.

For these deep repair welds, the qualification of the RT will be performed for the whole thickness of the repair whereas the qualification of the UT will be limited to 25 mm depth.

For shallow repair welds, the capability of the inspection techniques is the same as that for deep repairs when applying the same qualified NDT. However, considering that shallow repairs are less sensitive to fast fracture, the minimum depth to which repairs need to be inspected by qualified volumetric inspections will be defined in the nuclear site licensing phase. Fracture mechanics calculations will be used to establish the limiting defect size, and the minimum depth of repairs which need to be inspected will be no greater than half the limiting defect size in order to respect the target defect size margin of 2.

Fracture toughness

The lower bound fracture toughness values used to determine the critical defect size based on a 3000 hours aged repair mock-up [Ref-2] will be confirmed by measurements on a repair mock-up after 10,000 hours ageing at 400°C.

1.6.2. Reactor Coolant Pump flywheel

For the Reactor Coolant Pump flywheel, the geometry to be considered is shown in Section 5.4.1 - Figure 5: the flywheel is made up of two alloy steel discs linked by a bolting assembly (no weld).

An ELLDS of 450 mm has been calculated from the inner radius of the flywheel for a complete through thickness radial defect (corresponding to the worst orientation) subject to an Reactor Coolant Pump overspeed of 1.25 times normal operating speed [Ref-1]. Moreover, this Life Fatigue Crack Growth (LFCG) calculated starting from an initial defect of 20 mm is about 1.5 mm, which is not significant. This high ELLDS value for a component manufactured using a laminated plate process with the application of UT controls required by RCC-M ensures that the flywheel is manufactured without defects of structural concern and that highly unlikely defects will also be captured. No qualified NDT has been selected for the Reactor Coolant Pump flywheel [Ref-2]. Furthermore, French experience associated with the Reactor Coolant Pump Flywheel in-situ ultrasonic examinations and hot laboratory examinations have recorded no defect and have produced satisfactory results for the 58 operating plants [Ref-3].

The verification of the lower bound fracture toughness values used to determine the critical defect size will be performed by measurements of the fracture toughness on a mock-up using Compact Tensile specimens. The mock-up will be representative of the EPR material [Ref-4].

SECTION 5.4.1 - TABLE 1

Design Parameters for Reactor Coolant Pumps [Ref-1]

REACTOR COOLANT PUMP ASSEMBLY	
– Design pressure (bar)	176
– Design temperature (°C)	351
– Total height of the reactor coolant pump (m)	9.342
– Seal 1 water injection (m^3/h)	1.8
– Seal 1 leak-off water (m^3/h)	0.680
– Thermal barrier cooling water (m^3/h)	9
– Maximum inlet temperature for the cooling water (°C)	38
PUMP	
– Best estimate flow rate (m^3/h)	28,320
– Developed head (m)	102.1
– Nominal flow rate (m^3/h)	28,192
– Nominal developed head (m)	98.5±5%
– Thermo-hydraulic flow rate (m^3/h)	27,185
– Mechanical design flow rate (m^3/h)	30,585
– Suction temperature (°C)	295.9
– Inlet diameter of the suction nozzle (m)	0.78
– Inlet diameter of the discharge nozzle (m)	0.78
– Rotation speed (rpm)	1485
– Mass without water (including motor support) (kg)	55,200
MOTOR	
– Type	Air cooled Squirrel cage induction motor
– Power rating (kW)	8,680
– Design input power, RCP [RCS] under normal conditions (kW)	8,000
– Design input power, RCP [RCS] under cold conditions (kW)	10,850
– Voltage (volts)	10,000
– Phase	3
– Frequency (Hz)	50
– Insulation class	Class F thermo-elastic epoxy insulation
– Mass (without water or oil) (kg)	60,900
– Total inertia (pump and motor) of the rotor (kg.m²)	5,210

SECTION 5.4.1 - TABLE 2

Design Parameters for Reactor Coolant Pump Materials [Ref-1]

ELEMENTS	MATERIAL	COMMENT
CASING	Z3 CN 20.09 M	CAST
THERMAL BARRIER FLANGE	Z2 CN 19.10 + N2	FORGED
SEALS AND SSS HOUSING	Z2 CN 19.10 + N2	FORGED
MAIN FLANGE	16 MN D5	FORGED
CASING NUTS AND STUDS	40 NCD 7.03 Cl.B	FORGED BAR
PUMP SHAFT	Z6 CN Nb 18.11	FORGED BAR
FLYWHEEL	20 NCD 14.7	ROLLED PLATE

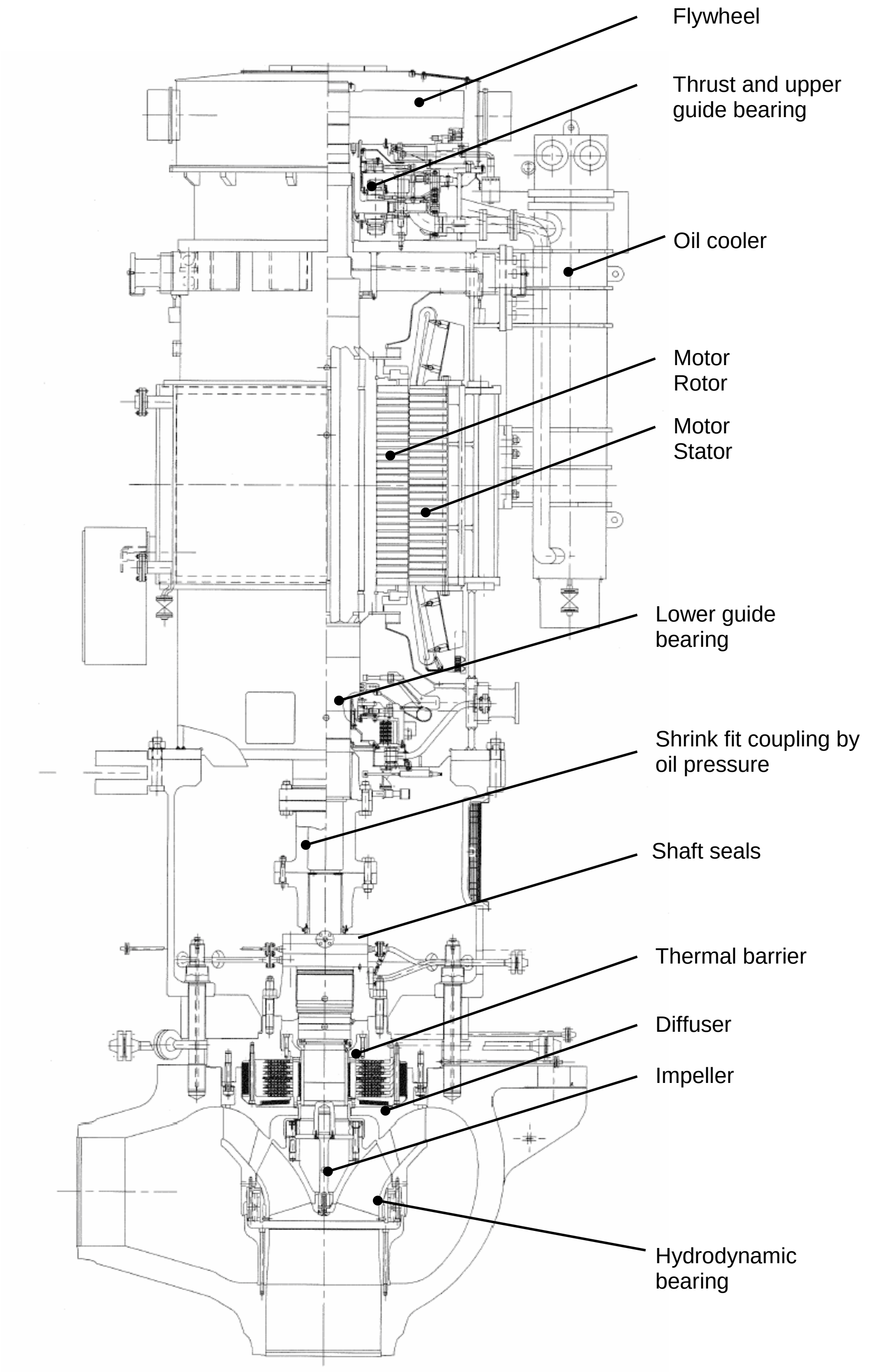
SECTION 5.4.1 – TABLE 3

List of Class 1 Components for Reactor Coolant Pumps [Ref-1]

- Casing,
- Casing studs and nuts,
- Thermal barrier flange,
- Thermal barrier heat exchanger,
- Plugs welded to the thermal barrier flange that are exposed to RRI [CCWS] system water,
- No.1 seal injection line up to the supply boundary, including the flange, counter-flange and bolting,
- No.1 seal housing,
- No.1 seal housing thermowell,
- Flange (with flow restrictor) for the No. 1 leakoff line (seal housing side) and its bolting,
- No.1 seal leakoff lines (beyond the No.1 seal housing side) up to the supply boundary,
- No.2 seal housing,
- Studs and nuts for the No.1 seal housing,
- No.1 seal ring holder,
- No.1 seal ring housing insert and its mounting bolting ,
- No.2 seal ring housing insert and its mounting bolting ,
- Shared mounting bolting for the No.2 seal housing, No.3 seal housing and SS housing,
- Main flange.
- RRI [CCWS] inlet and outlet piping on the thermal barrier including the flanges,
- No.2 seal leakoff connections including flanges.

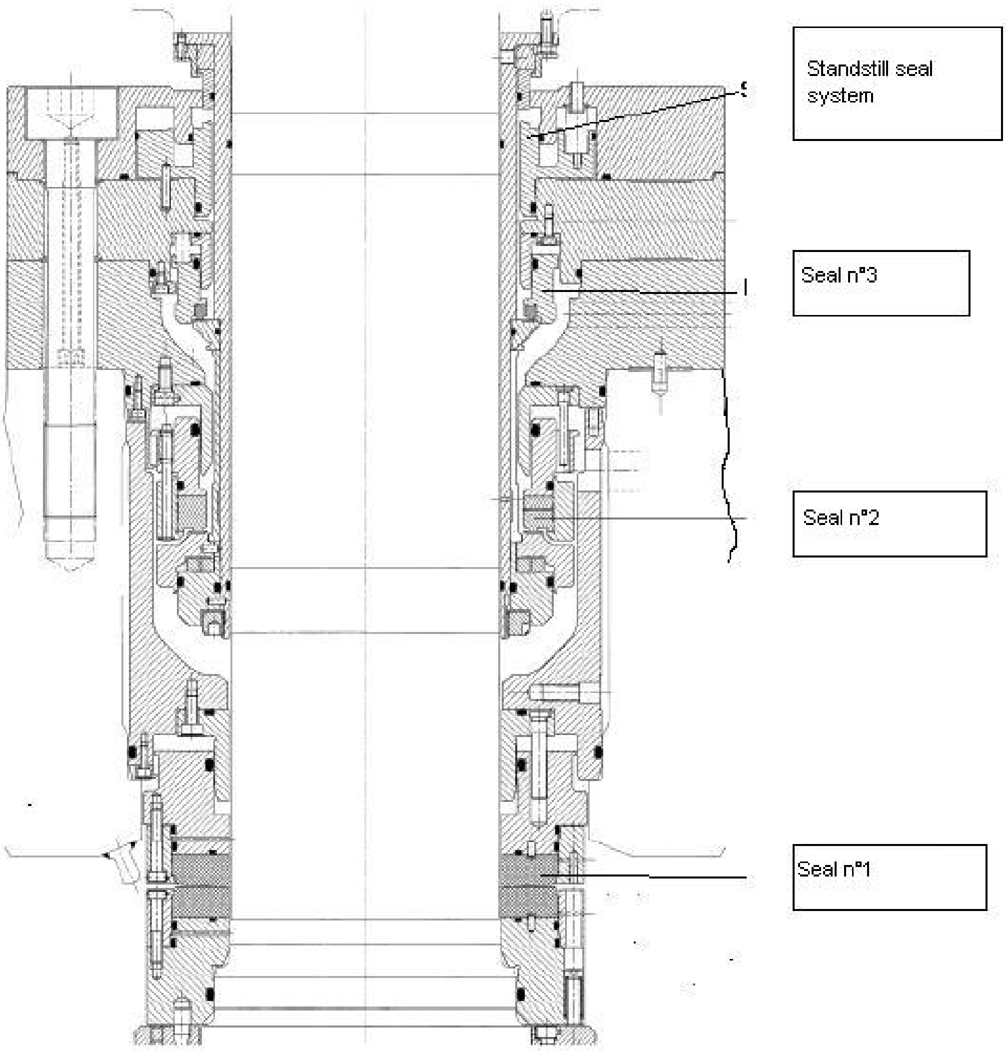
SECTION 5.4.1 - FIGURE 1

Reactor Coolant Pumps – General Assembly [Ref-1]



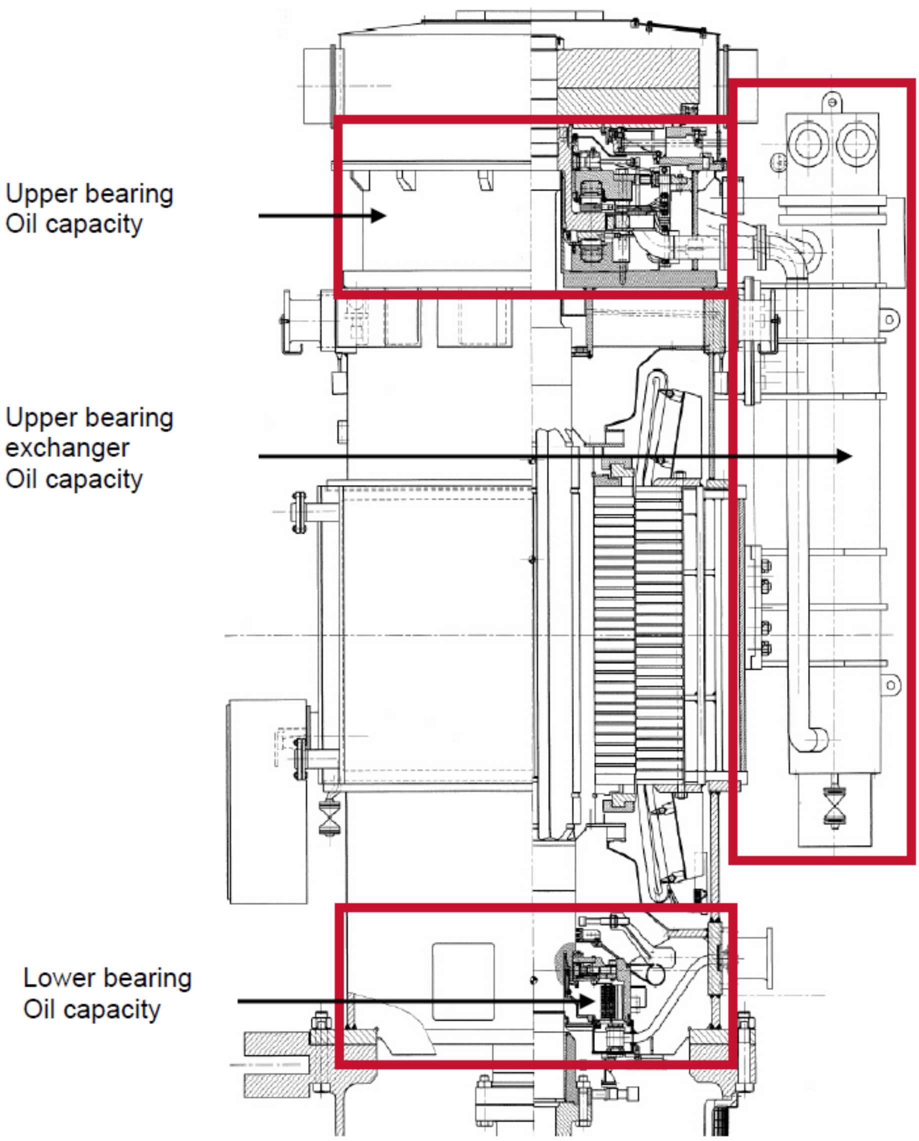
SECTION 5.4.1 - FIGURE 2

Shaft Sealing System [Ref-2]



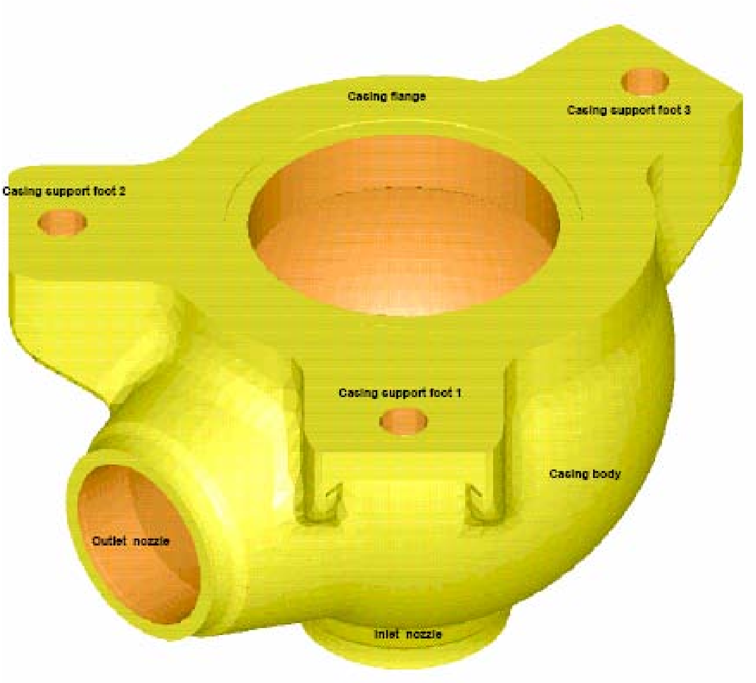
SECTION 5.4.1 - FIGURE 3

Locations of Oil Volumes within the Reactor Coolant Pump



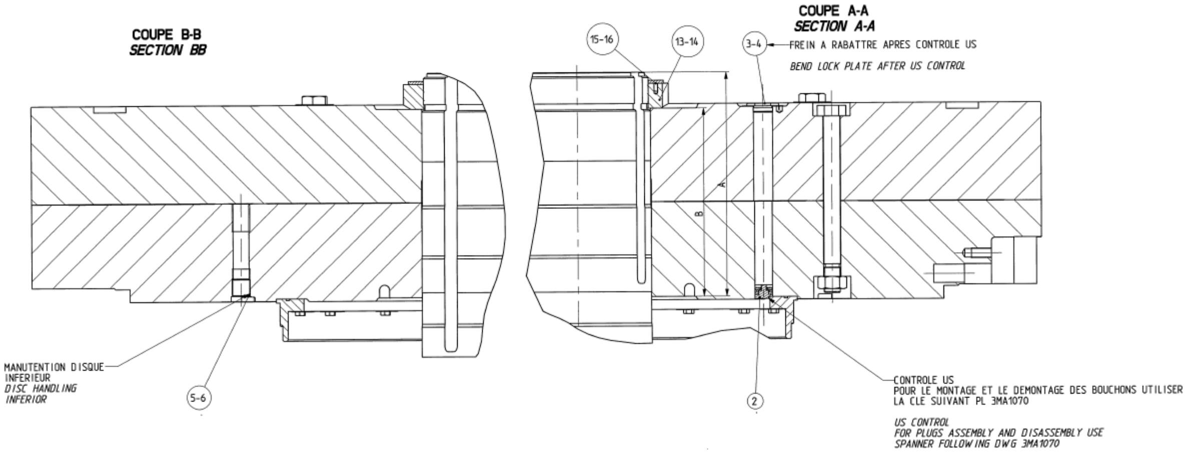
SECTION 5.4.1 - FIGURE 4

Reactor Coolant Pumps – Casing



SECTION 5.4.1 - FIGURE 5

Reactor Coolant Pumps – Flywheel



2. STEAM GENERATORS

2.1. DESCRIPTION [REF-1]

2.1.1. General characteristics

The design of the EPR steam generator is based on that of the N4 steam generator (type 73/19 TE [Ref-1] to [Ref-3]). It is a natural circulation U-tube heat exchanger fitted with an axial economiser (see Section 5.4.2 - Figure 1).

The steam generator is composed of two sub-assemblies. One ensures vaporisation of the feedwater, the other the mechanical drying of the water/steam mixture produced. It is arranged vertically, the water/steam mixture flowing upwards by natural circulation. The feedwater enters the steam generator through the main feedwater nozzle into a conical shell in order to reduce thermal stratification. It then passes through the distribution half-ring (cold leg side) equipped with a deflector sheet installed at a level above the upper section of the tube bundle.

The operating principle of the axial economiser is to direct all the feedwater to the cold leg of the tube bundle and about 90% of the recirculated water to the hot leg. This is ensured in practice by incorporating the following features to the design of a standard natural circulation U-tube steam generator:

- A double wrapper in the downcomer on the cold side to guide the feedwater to the cold leg side of the tube bundle.
- A secondary side divider plate (from the tube sheet up to the sixth tube support plate) to separate the cold leg and hot leg parts of the tube bundle. In addition and in conjunction with the previous design features, the steam generator feedwater distribution system (ring with oblong-shaped holes and deflecting sheet) extends over a 180° angular portion of the wrapper on the cold side.

This design enhances the heat exchange efficiency between the primary and secondary sides and increases the pressure of the steam produced by 3 bar, compared with a standard steam generator having the same heat exchange surface area.

Unlike other economiser designs, this design has two further benefits:

- There is no direct cross flow affecting the tubes and thus no risk of vibrations.
- There is no reduction in accessibility of the tube bundle for inspection and maintenance.

The steam generator is fully shop-built and is transported to the site and installed in its reactor building cubicle in one piece.

It is supported vertically by four support legs and laterally guided at two levels (see section 9 of this sub-chapter).

2.1.2. Lower sub-assembly

The lower sub-assembly is comprised of the following parts:

- The channel head formed by a hemispherical bottom head and a tube sheet [Ref-1] [Ref-2]. A cylindrical section has been added to the upper section of the lower head in order to improve access to the peripheral tubes for inspection. A primary partition plate divides the channel head into two leak tight compartments. One is connected to the reactor vessel outlet (hot leg) and the other to the reactor vessel inlet via the reactor coolant pump (cold leg).

- Each compartment includes a nozzle for connection to the reactor coolant system and a manway provides access for in-service inspection and maintenance operations. Specific provisions are made to allow plugging of the nozzles, and to carry out inspection and maintenance operations inside each compartment during refuelling outages.

- The lower sub-assembly of the steam generator's secondary shell is comprised of three cylindrical shell sections [Ref-3] [Ref-4]. The lower section is fitted with 8 hand holes in for in-service inspection and maintenance operations on the lower parts of the tube bundle. The intermediate section is fitted with two diametrically opposed eyeholes at the level of the sixth tube support plate on the axis of the central tube lane and a set of instrument taps to measure the steam generator water level. The upper section is provided with the upper lateral steam generator support brackets welded to the outer surface. Blowdown of the steam generator is provided by means of three blowdown taps located in the tube sheet.

- The tube bundle, made up of inverted U-tubes, providing heat exchange between the primary coolant circulating inside the tubes and the secondary cooling system [Ref-5]. It also constitutes a radiological barrier between the primary and secondary sides of the nuclear steam supply system (NSSS).

- The tube bundle is arranged in a triangular pitch.

  The ends of the inverted U-tubes are welded to the protective cladding of the primary face of the tube sheet. These welds are helium leak tested and the ends of the U-tubes are then full-depth expanded into the tube sheet in order to eliminate any crevices.

  It has been demonstrated that the equipment and procedure used for tube expansion minimises the residual stresses in the transition zone between the expanded and non-expanded parts of the tube.

  Measures are taken to ensure that the expanded length of the tubes stops just below the secondary face of the tube sheet.

- The role of the lower internal support structures is to support the tube bundle while ensuring circulation of the secondary cooling fluid [Ref-6].

They comprise:

- The bundle wrapper which isolates the recirculation water flow path from that of the water/steam mixture. It creates an annular zone between the tube bundle and the steam generator shell called the downcomer. An annular opening in the lower section allows the distribution of the water to the tube bundle.

- A semi-annular downcomer on the cold leg side comprises a double wrapper concentric with the bundle wrapper, the outlet of which is close to the top face of the tube sheet on the periphery of the tube bundle on the cold leg side.

- A secondary divider plate, welded to the bundle wrapper, separates the cold leg side from the hot leg side up to the sixth tube support plate and constitutes the economiser area. The joint between the divider plate and the tube sheet is leak-tight.

- Nine tube support plates are spaced over the height of the tube bundle to support the tubes and thus avoid any vibration/wear effects. Broached Trefoil holes ("high permeability" broaching) with flat contacts (to remove the risk of dryout) allow the secondary water/steam mixture to circulate freely.

  These tube support plates are centred by means of double slope wedges fixed on anti-seismic blocks regularly spaced around their periphery, centring the tube-bundle wrapper in the steam generator shell.

  The tube support plates are vertically interconnected and spaced by a network of tie-rods bolted into the secondary side face of the tube sheet.

  The tie-rods, the anti-seismic blocks and the tube support plates are designed to withstand in-plane and out-of-plane loads even under the faulted conditions (earthquakes, pipe ruptures), specified in section 2.6.3 of this sub-chapter.

  Over the full height of the economiser divider plate, each of the tube support plates is divided into two half-plates.

- A flow distribution baffle designed to obtain proper distribution of the fluid entering the tube bundle and to minimise the low flow velocity zone above the tube sheet.

- Three perforated tubular manifolds designed to allow rapid draining of the steam generator and continuous blowdown.

2.1.3. Upper sub-assembly

The upper sub-assembly (steam drum) is made of two cylindrical shells, a conical shell and an elliptical upper end [Ref-1] [Ref-2]. The cylindrical section is provided with:

- Two manways providing access to the steam/moisture separation equipment. They also give access to the feedwater devices, the top of the tube bundle via the hatch through the bundle wrapper and the camera holes installed on the roof of the bundle wrapper.

- A set of water level measurement taps.

The centre of the upper elliptical head contains an integral steam outlet nozzle fitted with a welded steam flow restrictor. This limits the rate of depressurisation and the forces applied to the steam generator tube bundle and internals in the event of a steam line break.

The steam drum is fitted with:

- A main feedwater system comprising:
  - The main feedwater nozzle situated in the conical shell.
  - The thermal sleeve (leak tight connection between the main feedwater nozzle and the feedwater distribution ring).
  - The main feedwater half-ring fitted with a deflecting sheet to ensure a uniform distribution of all the water from the main feedwater system (ARE [MFWS]) or the start-up and shutdown system (AAD [SSS]) feedwater into the cold leg side downcomer.
  - The feedwater half-ring supports.
- An emergency feedwater supply system comprising:
  - Its own inlet nozzle.
  - The thermal sleeve (leak tight connection between the emergency feedwater system, ASG [EFWS], nozzle and the emergency feedwater ring).
  - The emergency feedwater distribution ring (equipped on its higher generatrix with I-shaped tubes orientated towards the interior of the SG to avoid any risk of cold water impact on the pressure boundary components) to distribute the emergency feedwater into both the hot leg side and cold leg side downcomer.
  - The support system.

The design and physical separation of the main and emergency feedwater systems removes the risk of water hammer and minimises the risk of thermal stratification.

- Water/steam separation equipment comprising:
  - A first stage of cyclone type separators of 335 mm diameter [Ref-1] (connected to the bundle wrapper roof). Each separator is designed to achieve a very low steam carry-under.
  - A single stage “chevron”-type dryer unit (hung on the upper head of the steam drum) which achieves low water carry-over to the steam generator outlet.

2.1.4. Support structure

The steam generator support structure is designed to allow thermal expansion of the reactor coolant system loop and displacement caused by pressure, but limits such displacement during accidents. The support structure is comprised of vertical and lateral supports.

2.2. OPERATING CONDITIONS AND INTERFACES

The operating conditions and interfaces are provided in Section 5.4.2 - Table 1 (Steam Generator Data Sheet) for 4500 MWth power.

2.3. DESIGN PRINCIPLES AND OBJECTIVES

This component is non-breakable which involves satisfying all the design requirements described in section 6 of Sub-chapter 5.2.

2.3.1. Functional requirements

The steam generator is designed to fulfil the following functions:

- Produce steam with no more than 0.25% moisture carry-over at the steam generator outlet using the reactor coolant system as the heat source [Ref-1].

- Provide capability for continuous hot blowdown of the secondary side of the steam generators. The steam generator blowdown rate permits a transition from cold lay-up water chemistry to hot standby water chemistry within an eight-hour period.

- Provide secondary side water level indications and automatic control of water level at any power level from hot no load to full power.

- Provide a leak tight boundary between the reactor coolant and the steam generator secondary side.

- Serve as the first mean for removal of decay heat from reactor coolant during plant shutdown using main feedwater as well as start-up and shutdown feedwater in normal operation or emergency feedwater in accident condition. This allows the primary coolant temperature to be reduced to a value reasonably below the saturation temperature corresponding to the operating pressure of the residual heat removal system.

- Provide for full wet lay-up of the steam generator under deoxygenated, pH-controlled conditions.

2.3.2. Main properties selected

- Full power steam pressure and capacity

The steam generator is designed to generate, under given primary coolant system pressure and flow conditions, the pressure and mass flow of steam specified at 100% of the nominal full power.

- Operational capability

The steam generator is able to fulfil its required functions in the event of RRC-A and PCC-1 to PCC-4, except for the period when reactor core cooling is carried out by the residual heat removal system (RRA [RHRS]) or by its back-up systems.

- Steam generator water inventory (heat sink reserve)

The steam generator supplies the reactor with a minimal heat sink reserve of cold water (steam generator water inventory) to ensure the mitigation of all RRC-A and PCC-1 to PCC-4 events with respect to their associated criteria and taking credit for the operation of dedicated systems – such as main feedwater system (ARE [MFWS]), start-up and shutdown system (AAD [SSS]), emergency feedwater system (ASG [EFWS]), safety injection systems (RIS [SIS]) and pressuriser safety valves.

In particular, the steam generator is designed to hold a sufficient volume of water such that should all feedwater systems (ARE [MRWS], AAD [SSS] and ASG [EFWS]) be lost, the steam generator secondary side will not dry out in less than 30 minutes.

- Total volume of the steam generator (overfilling requirement)

The steam generator serves as an expansion tank for secondary feedwater systems, and consequently protects the steam equipment against water inflow (including the turbine) after isolation of the source of the overfill.

In the event of sudden steam generator tube rupture (SGTR) (double-ended guillotine break or a break of equivalent surface area) of one or two steam generator tubes, the medium head safety injection (MHSI) pump discharge head is below the pressure setpoint of the non-isolatable main steam safety valves, which prevents these valves opening. In addition, the increased volume of the secondary side steam generator (free volume above the normal level increased from 61.5 m^3 on N4 to 82.3 m^3 [Ref-1]) allows the flow of fluid from the primary to secondary side to be accommodated without the risk of overfilling the steam generator, at the same time taking account of the capacity of the primary and secondary side isolation devices and the primary circuit monitoring systems.

It should also be noted that in association with the steam generator high level setpoint, the chemical and volume control system (RCV [CVCS]) charging line is isolated in order to avoid overfilling the steam generator.

- Moisture separation equipment

The design of the moisture separation equipment provides steam with a moisture carry-over not exceeding 0.25% under normal operating conditions with the turbine in operation.

Testing on models and thermo-hydraulic analysis has resulted in a new separator model design 1.5 times larger than that of the steam generator for the N4.

- Steam generator elevation compared with the reactor vessel

The steam generators are installed at a height relative to the reactor vessel which ensures that primary side drainage of the steam generator plenum permits inspection and/or tube plugging without reducing the water level in the reactor vessel below the reactor core or adversely affecting reactor coolant circulation in shutdown state.

- Tube bundle inspection activities (NDT)

On the shutdown critical path they will be carried out in parallel on the four steam generators, which significantly reduces the overall outage time. The ability to undertake primary side steam generator tube inspections from the plenums using protection plugs/dams in the nozzles to the reactor coolant system loops is at the design stage. It is anticipated that the inspection would be performed on EPR plant units only when the core is unloaded. During these inspections, it is planned to ensure double isolation between the spent fuel pool and the vessel pool using transfer tube isolation valves or various other sluices and gates.

The use of steam generator plugs with fuelled core is not authorised on the EPR at this stage in the analysis.

Over a 60-year plant life, for operational requirements, the use of steam generator plugs/dams with a fuelled core is not ruled out. A further safety assessment relative to the risk of pool drainage will be provided prior to any proposed use of plugs (see section 3 of Sub-chapter 9.1).

- Steam generator blowdown

Drains are provided for the steam generator primary and secondary sides. The drain system is designed to drain the steam generator at any temperature up to and including hot standby. Blowdown system equipment is utilised to provide this hot drain capability.

- Provisions for secondary side cleaning

Precautions are taken to minimise the formation of sludge in the steam generator (see sections 2.4.4 and 2.9 of this sub-chapter).

However, the lower section of the secondary side of the steam generator is designed to allow the removal, if necessary, of any sludge that may have accumulated on tube sheets in low flow velocity areas by sludge lancing.

2.3.3. Requirements for inspection, repair and replacement

The steam generator pressure boundary is designed to minimise the number of welds and optimise their location in order to facilitate in-service inspections.

- The upper end and the steam outflow nozzles of the steam drum are manufactured from a single forging.

- The conical shell is forged and has flanged ends of sufficient length to facilitate inspection of connecting welds.

In addition, thermal insulation can be removed locally. Both permanent and temporary measures are provided to ensure convenient access to welds.

Measures are also taken to ensure adequate access for inspection and maintenance of steam generator internals.

On the primary side, the support columns allow easy access to the manways with which each compartment of the channel head is fitted. Special equipment allows inspection of the internal surface of the plenum in contact with primary coolant, the tube welds on the tube sheet cladding and the steam generator tubes.

On the secondary side, special attention has been paid to access to the lower section of the tube bundle and the tube sheet as follows:

- Eight hand-holes are distributed around the secondary shell.
- The mechanical and thermo-hydraulic design of the lower steam generator support structure has been optimised (geometry of the waterway flow blocks, design of the blowdown system) to facilitate sludge lancing operations.

Access to the steam generator upper internals inside the steam drum is provided by two large-diameter manways. From this position, it is also possible to access the tube bundle anti-vibration bars (AVB). A hatch has been installed in the bundle wrapper for this purpose.

In addition, although maintenance and repair operations are not anticipated during the steam generator (SG) design life, the dryer vanes may be replaced through the manways in the upper part of the steam generator. They are held in place by screw jacks inside the U-shaped frames. By loosening the screws, dismantling is possible through openings in each.

A monitoring programme will be drawn up during the detailed design phase. It will describe the arrangements for periodical inspections, as well as the objectives, nature and frequency of non-destructive inspections implemented to detect defects harmful to the integrity of the steam generators.

2.4. THERMO-HYDRAULIC DESIGN

2.4.1. Operating parameters

Section 5.4.2 - Table 2 gives other operating parameters not included in the steam generator data sheet [Ref-1].

2.4.2. Thermodynamic criteria

The steam generator is designed so that level fluctuations at the steam-water interface as well as structural vibrations do not occur during operation.

The circulation ratio is defined as the ratio of total flow rate across the tube bundle to steam flow rate output of the steam generator. It reaches 3.54 [Ref-1] when the steam generator is operating at full load for 4500 MWth power. This value offers a satisfactory compromise between efforts to obtain steam as dry as possible and stable operation of the steam generator. This circulation ratio also improves the steam generator transient behaviour by minimising water level shrinkage in the event of large transients (turbine or reactor trip) and reduces low velocity area above the top of the tube sheet.

2.4.3. Thermal design

The heat exchange surface and tube bundle configuration adopted are similar to those used in the N4 steam generator and therefore this steam generator belongs to a wide range of proven designs. The selected tube diameter (outer diameter 19.05 mm) [Ref-1] is one of the current international standardised diameters and is a good compromise between compactness, vibration behaviour and ease of manufacture. The triangular pitch of 27.43 mm comes from the same desire for compactness, while providing the capability to carry out high-quality cleaning operations with the usual technique of sludge lancing [Ref-2] to [Ref-4]. This has been clearly demonstrated by the first lancing operations performed on the first N4 plants at Chooz and Civaux.

Proof that the required steam mass flow rate and pressure can be achieved at full power for the specified reactor coolant flow and temperature conditions has been obtained using physical models and correlations qualified on the basis of the intensive test program performed on the 25 MW MEGEVE test loop at Cadarache C.E.A. Research Centre, and verified in Chooz and Civaux.

Absence of water level fluctuations at the steam-water interface was demonstrated using the physical models qualified from the MEGEVE test results. This is mainly achieved by an adequate ratio between single-phase pressure drop and total pressure drop in the recirculation loop.

2.4.4. Hydraulic design

Hydraulic design has been carried out to obtain:

- An acceptable flow distribution in the steam generator secondary side, particularly regarding the risk of sludge deposits, the effects of erosion/corrosion and tube bundle vibrations.
- Secondary water inventory in accordance with cooling water requirements.

To limit the possible accumulation of secondary side sludge and the risk of stress corrosion cracking of the tube bundle, particular attention is paid to reducing low flow velocity areas as much as possible, particularly above the tube sheet.

This has been achieved by suitable design of the flow blockers as well as the distribution baffle. The primary result of this is that sludge deposits in operation may affect only a very low number of tubes, and these are close to the centre of the tube lane in the same area as the blowdown system intakes. It should be noted that these measures are in addition to the general measures described in section 2.9 of this sub-chapter, intended to reduce sludge generation in the secondary circuit.

No specific problems are expected regarding erosion/corrosion due to the choice of materials selected for the feedwater distribution systems, the tube support plates and the separation equipment.

Tube bundle vibrations are addressed in section 2.4.5 of this sub-chapter.

The secondary water mass of 77.2 tonnes [Ref-1] in normal operation is in accordance with the requirement to have a minimum duration of 30 minutes before dryout of the steam generator following the loss of all feedwater systems.

2.4.5. Tube bundle vibrations [Ref-1]

In the design of steam generators, the possibility of degradation of tubes due to either mechanical or flow-induced excitation is thoroughly evaluated. This evaluation includes detailed analysis of the tube support system as well as an extensive research program of tube vibration model tests.

In evaluating failure due to vibration, consideration is given to such sources of excitation as those generated by the primary fluid flowing within the tubes, mechanically induced vibration, and secondary fluid flow on the outside of the tubes. During normal operation, the effects of primary fluid flow within the tubes and mechanically induced vibration are considered to be negligible and should cause little concern. Thus, the main source of tube vibrations is the hydrodynamic excitation by the secondary fluid on the outside of the tubes. In general, three vibration mechanisms were identified:

- Vortex shedding:

  Vortex shedding does not produce detectable tube bundle vibration. There are several reasons why this happens:

  - Flow turbulence in the downcomer and tube bundle inlet region inhibits the formation of Von Karman's vortex train.
  - The spatial variations of cross flow velocities along the tube preclude vortex shedding at a single frequency.
  - Both axial and cross flow velocity components exist at the tubes. The axial flow component disrupts the Von Karman vortices.

- Fluid-elastic excitation:

  Concerning fluid-elastic excitation, the tube bundle supporting system was designed to obtain a margin with respect to tube instability according to quasi-steady Blevins-Connors model based on numerous experimental data.

- Turbulence:

  Levels of turbulent responses, determined by using umbrella turbulent force spectrums obtained from mock-up results for various fluids, are small and the contribution of the resulting stresses to fatigue is negligible.

2.5. MATERIALS AND MATERIAL PROPERTIES [REF-1] [REF-2]

2.5.1. Pressure retaining parts

All materials for pressurised retaining parts used for steam generators are selected and manufactured in accordance with the requirements of section II of the RCC-M (see Sub-chapter 3.8).

- Steam Generator tube materials:

  - Alloy 690 which is currently used worldwide as a steam generator tube material has been selected for the tube bundle tubes.

- o The excellent corrosion resistance of alloy 690 has been demonstrated both by experience at plant units and numerous laboratory tests under conditions representative of primary and secondary side conditions. The average cobalt content in the SG tubes is no more than 0.015% per SG tube bundle.

Alloy 690 has been laboratory tested for ten years by several respected bodies and selected by a significant number of operators. Alloy 690 has been used on plant units since 1988 in the USA, 1989 at Ringhals and 1990 at Dampierre.

- Pressurised retaining parts
  - o Paragraph B2000 covering level 1 components of the RCC-M code applies (see Sub-chapter 3.8).
  - o Low-alloy ferritic steel (18 MND 5 or 20 MND 5) is selected for the shells, heads, nozzles and tubesheet. A comparison of the chemical composition and mechanical characteristics shows that these two materials are similar [Ref-1] to [Ref-3].
  - o Those surfaces of the material in contact with primary fluid are clad with austenitic stainless steel (steam generator channel head) or Ni-Cr-Fe alloy (Alloy 690 type) with a cobalt content of less than 0.06% (tube sheet).
  - o The toughness of the steam generator’s ferritic material complies with the requirements of the RCC-M code (see Sub-chapter 3.8), paragraph B2000 and section II. The initial RT_{NDT} for the primary head and nozzles, the tubesheet and the lowest secondary shell is less than -20°C. For the other secondary shells and nozzles and for the elliptical head, it is less than -12°C.
- Partition plate.
- The channel head partition plate is made of Alloy 690.

2.5.2. Other main materials selected

- Tube support plate

The tube support plates (TSPs) are made of corrosion-resistant 13% Cr martensitic stainless steel and incorporate a three-lobe-shaped tube hole design that provides greater flow area adjacent to the tube outer surface compared with drilled TSPs. The resulting increased flow provides higher sweeping velocities at the tube/tube support plate intersections.

Peripheral block supports and tie-rods also provide stability to the plates so that tube fretting or wear due to flow induced plate vibrations at the tube support contact regions is eliminated.

- Anti-Vibration Bars

The sets of anti-vibration bars (AVB) are made of stainless steel 13% Cr alloy selected for its excellent wear coefficient with the tube material.

2.6. MECHANICAL DESIGN [REF-1]

In general terms, the mechanical design of the steam generator takes place in two steps:

- Sizing, to define the general arrangement of the component and its main dimensions.

- Design of the main sub-assemblies and justification of their capacity to withstand, under all circumstances, the loads and combinations of transient conditions provided for over the component's life including reactor coolant side and secondary side hydrostatic tests in the conditions defined in Section 5.4.2 - Table 1.

The following degradation mechanisms are studied [Ref-2]:

- Excessive deformation and plastic instability.

- Fatigue.

- Progressive deformation.

- Brittle fracture.

- Flow induced vibration.

- Tube wearing.

The most sensitive primary and secondary weld zones are identified and specifically studied with respect to brittle fracture concerns.

UK EPR specific fast fracture calculations have been defined in the frame of the safety demonstration of High Integrity Components, as described in section 2.10.

Design features and material selection ensure that other degradation mechanisms such as corrosion mechanisms (flow assisted corrosion, stress corrosion cracking, generalised corrosion) are addressed.

The mechanical design of the steam generators primary and secondary coolant pressure boundaries is made in accordance of the RCC-M code (level 1) (see Sub-chapter 3.8).

The combined transient conditions and load conditions applicable to mechanical components and systems (in particular to the steam generator) are dealt with in Sub-chapter 3.4 (topics specific to mechanical components, including the list of events).

2.6.1. Sizing calculations

Sizing calculations are performed for design conditions, according to chapter B3000 (class 1 components) of the RCC-M code, to determine the minimum acceptable thickness of the pressure shells and nozzles and to demonstrate the adequacy of the tube dimension.

For the secondary compartment, a corrosion allowance is taken into account.

2.6.2. Design of sub-assemblies

The following properties have been adopted, based on experience feedback from French, German, and other international plants:

- Design of the tube-to-tube sheet junction

  The tubes are fully expanded inside the tube sheet and welded on the tube sheet primary side.

  The tube expansion is made using a single step high pressure hydraulic process.

- Feedwater distribution system:

  - The main and emergency feedwater distribution systems are separate (two nozzles [Ref-1] [Ref-2] and two rings).
  - The water distribution systems are connected to the feedwater nozzles by leak tight connections.
  - Provisions are made to limit the thermal stratification in the main feedwater nozzle regions (location of the feedwater nozzle in the conical shell and under the feedwater distribution ring).

- Design of upper and lower internal components

  The lower internal components (tube bundle, tube support system, bundle wrapper and associated supports, equipment installed above the tube sheet) and upper internal components (cyclone separators, main and emergency feedwater systems, moisture separators) are designed in such a way as to enable free thermal expansion between them as well as between them and the bundle wrapper and between them and the SG pressure containment under PCC-1 to PCC-3 scenarios.

2.6.3. Design relative to pipeline ruptures and earthquakes

The steam generator and its internal components are designed to withstand deadloads as well as dynamic loads which may result from an earthquake or a guillotine break of a pipe connected to the reactor coolant system or depressurisation caused by a steam pipework break or a break in the feedwater pipework. The integrity of the steam generator tubes and the tube support structures has been demonstrated for all these events.

2.7. SAFETY EVALUATION

Safety analyses of the steam generator design are provided in the accident analyses chapters (see Chapters 14 and 16).

2.8. PROCUREMENT, MANUFACTURE AND QUALITY ASSURANCE

2.8.1. Procurement [Ref-1]

Materials procurement complies with the general specifications of the RCC-M code (see Sub-chapter 3.8) (section II), the qualification datasheet of filler material and, where appropriate, with additional requirements (see Section 5.4.2 - Table 1).

2.8.2. Manufacture

The following procedures are applied to the manufacture of main components [Ref-1]:

- Lower hemispherical head by forging.
- Tube sheet by forging.
- Cylindrical shells by forging.
- Conical shell by forging.
- Upper elliptical head by forging with integral steam outlet nozzle.
- Heat transfer seamless tubes by cold drawing and annealing.
- A final heat treatment is carried out on all straight tubes in the tube bundle to relieve stress and improve corrosion resistance. A similar stress-relief heat treatment is carried out on the smaller U-bends.
- The corrosion-resistant cladding in contact with the primary cooling system is deposited by a welding technique.
- The tube-to-tube sheet welding process is automatic TIG welding.
- The tubes are expanded along the full thickness of the tube sheet.

During manufacture, the steam generator primary and secondary surfaces are cleaned.

2.8.3. Testing and inspections

The steam generators are inspected and tested in accordance with sections III "Inspection methods" and IV "Welding" of the RCC-M code (see Sub-chapter 3.8).

During hydrostatic tests, the requirements relating to water cleanliness and quality are defined in chapter B 5000 of the RCC-M code.

UK EPR specific inspections have been defined in the frame of the safety demonstration of High Integrity Components, as described in section 2.10.

2.9. SECONDARY SIDE WATER CHEMISTRY TREATMENT – MATERIALS AND OPERATING MODES

The SG secondary side water chemistry (including corresponding data table) is detailed in Sub-chapter 5.5.

2.10. FAST FRACTURE ANALYSIS

As stated in Sub-chapter 3.1, the Steam Generator (pressure boundary parts) is a High Integrity Component (HIC) for which the specific measures described in section 0.3.6 of Sub-chapter 3.4 concerning prevention, surveillance and mitigation contribute to its high integrity demonstration.

With regards to the fast fracture risk the three-legged approach presented in Sub-chapter 3.4 section 1.6 has been applied to the Steam Generator as follows:

- use of fracture mechanics to determine the end of life limiting defect size and demonstration that the Defect Size Margin between this critical defect size and the detectable defect increased by fatigue crack propagation over the lifetime is larger than 2 as far as practicable;
- use of suitable redundant and diverse inspections during manufacturing, supplemented by the use of qualified inspection(s) at the end of manufacturing;
- verification of the lower bound fracture toughness values used to determine the critical defect size by measurements.

This methodology applies to the whole component, including welds and base metal. However, considering the higher probability of finding crack-like defects in welds than in forgings, and considering that the toughness of weld metal is lower, the majority of the locations assessed are likely to be the welds. Consequently, the following sub-sections focus on welds, and the full demonstration including base metal (e.g. nozzle corners) will be provided in the detailed design stage.

2.10.1. Fracture Mechanics Analysis

For the Steam Generator, the High Integrity Components (HIC) methodology applies to the eight pressure boundary circumferential welds, to the primary nozzle Dissimilar Metal Welds (DMWs) and to three main nozzles welds (main feedwater, emergency feedwater, and secondary manhole) [Ref-1]. Three bounding cases[1] have been identified and fracture mechanics calculations have been performed on the most severe transients:

- Tubesheet to primary head weld,
- Tubesheet to secondary shell weld,
- High shell to conical shell weld

---

[1] the primary nozzle Steam Generator DMWs are covered in terms of Fracture Mechanics Analysis by the Reactor Pressure Vessel DMWs which critical defect size calculation is presented in Sub-chapter 5.3 section 6.1

The smallest End of Life Limiting Defect Size (ELLDS) is obtained for the Tubesheet to secondary shell weld based on a mixed methodology using a Finite Element (FE) calculation for primary and secondary stresses, and an analytical calculation for the Stress Intensity Factor J. This bounding case corresponds to an inner skin defect submitted to a Total Loss Of Feedwater transient which gives an ELLDS slightly greater than **20** mm in the ductile range. It has to be noted that this result corresponds to an initiation criteria without taking benefit from ductile tearing.

The limiting defect size calculated ensures that the size of a critical defect that would lead to failure is above 20 mm. This defect size will be used for any of the SG welds, DMW included, to define the requirements of inspections which will be applied at the end of manufacturing.

Fast fracture mechanics analyses will be performed in sensitive regions of the main forgings in the detailed design stage, to check whether these are limiting.

2.10.2. Non Destructive Testing (NDT)

Ferritic welds

The NDT selected to be qualified for Steam Generator homogeneous welds are [Ref-1]:

- the pulse echo UT technique required by RCC-M code supplemented with additional beam angles, and
- a UT specular reflection technique.

  The specular reflection Tandem technique can be applied to all the main circular welds and is used to detect certain type of plausible defects such as near vertical defects which may not be detected otherwise with conventional UT.

The RT control required by the RCC-M code will continue to be applied extensively but as a non-qualified control before heat treatment on all of the main component welds.

The geometry and reflection angle of nozzle and manhole welds do not enable inspection with the UT tandem technique. For these welds inspection will be made by qualified pulse echo UT with longitudinal waves 0° from inside the nozzle supplemented by qualified RT for some cases where longitudinal waves 0° cannot be fully performed. The radiographic examination is a highly flexible technique which can be applied to any of the component welds.

The sensitivity of the qualified pulse echo UT selected for ferritic component welds has been determined in the frame of ISI and enables detection of a flat bottom hole perpendicular to the beam axis with a diameter above 3.1 mm. The equivalent performance is the detection of a defect with through wall extent of 3 mm and length 18 mm.

For circular welds the qualified UT tandem technique increases the detection capability for smooth vertical defect through the whole thickness. Its sensitivity is based on a flat bottom hole 6 mm and it has been demonstrated that this technique can detect and reject defects with 5 mm and 10 mm through wall extent.

This UT tandem technique is supplemented by a 0° Longitudinal Waves from inside the nozzle which enables a specular reflection with good coverage of the weld.

Dissimilar Metal Weld

The NDT selected to be qualified for the inspection of DMW is the manual pulse echo UT technique using longitudinal waves up to 70° refraction angle, deployed from inside the pipe for the detection of defects located on the inner half thickness of the weld and from outside the pipe for the detection of defects located on the outer half thickness of the weld, supplemented by a 0^{o} LW reflexion technique from the pipe end [Ref-2]. These techniques can cover the volume of all welds and detect all defects of structural concern.

The demonstration of the performance of the UT selected for the inspection of dissimilar metal welds, and in particular their capability to detect the hypothetical narrow near-vertical planar weld defects, relies on the following evidence [Ref-2]:

- tests performed in the Sizewell qualification frame showing that defects embedded in a mock-up, including lack of fusion, could be detected using longitudinal waves up to 70° even when scanning only from the austenitic side of the weld – from ferritic side evidence presented in the previous section “ferritic HIC welds” is still valid.

- tests performed in the 1990’s and in 2011 on an AREVA DMW mock-up concluding that detection of the defects could be successfully achieved with the pulse echo technique, and that the detection of lack of fusion at the two fusion faces could be achieved when scanning with LW 0° from the end of the safe end; this inspection from the safe end provides a robust form of specular detection, and has been recognised as useful [Ref-2].

An ALARP analysis [Ref-3] summarised in section 3 of Sub-chapter 17.5 has been performed to support the adequacy of the UT proposal. The capability of this UT proposal to detect and reject defects whose Qualified Examination Defect Size (QEDS) derived from the fracture Mechanics Analyses is 10 mm or greater can be achieved with a high level of reliability; smaller defects can also be detected with reasonable capability.

Conclusion

Finally, a Defect Size Margin of 2 has been attained with Fracture Mechanics Calculations while considering that the fatigue crack growth for these worst case defects with regards to fast fracture is negligible, either with ferritic homogeneous weld or Dissimilar Metal Welds; this assumption will be verified during detailed design.

The UT techniques (and in some cases RT technique) to be applied at the end of manufacturing will be qualified for any UK EPR project.

2.10.3. Fracture toughness

Base Metal - Ferritic welds

The verification of the lower bound fracture toughness values used to determine the critical defect size will be performed [Ref-1]:

- For the base metal by measurements of the fracture toughness on the forgings of any UK EPR project, in the ductile range,

- For the weld material on a mock-up using Compact Tensile specimens (CTJ) in the ductile range. The mock-up will be representative of the EPR material, either base metal or wire/flux.

Dissimilar Metal Welds

The verification of the lower bound fracture toughness values used to determine the critical defect size will be performed by measurements of the fracture toughness on a mock-up using Compact Tensile specimens (CTJ) in the ductile range. The mock-up will be representative of the EPR materials, and the results will be used to evaluate the conservatism of the tests performed on an EPR DMW mock-up on Olkiluoto project.

SECTION 5.4.2 - TABLE 1

Steam Generator Data Sheet [Ref-1] [Ref-2]

STEAM GENERATOR	UNITS	
OPERATIONAL CHARACTERISTICS AT 100% OF NOMINAL POWER.		
- Number of SGs		4
- Type		Natural circulation with axial economiser
- Primary operating parameters		
. Mass flow		
* Thermo-hydraulic condition (TH)	kg/s	5558
* Best estimate condition (BE)	kg/s	5785
* Mechanical condition (ME)	kg/s	6237
. Exchanged power (TH, BE or ME capacity)	MWth	1131
. Inlet temperature (with no tube plugging or fouling):		
* TH	°C	329.8
* BE	°C	328.9
* ME	°C	327.2
. Outlet temperature (with no tube plugging or fouling):		
* TH	°C	295.4
* BE	°C	295.7
* ME	°C	296.4
. Operating pressure (inside tubes)	MPa abs	15.5
. Total pressure losses (with no tube plugging or fouling):		
* TH	MPa	0.274
* BE	MPa	0.294
* ME	MPa	0.338
. Margins relating to the heat exchange surface for plugging and fouling:		
* Total margin	%	10
* Plugging	%	5
* Fouling	%	5
- Secondary operating parameters (with no tube plugging or fouling):		
. Saturation pressure (TH, BE or ME capacity)	MPa abs	7.8
. Static steam pressure downstream the flow restrictor (TH, BE or ME capacity)	MPa abs	7.71
. Output steam flow (blowdown flow 1%; TH, BE or ME capacity)	kg/s	638.1
. Moisture content at SG outlet	%	≤ 0.25
. Feedwater temperature	°C	230

SECTION 5.4.2 - TABLE 1 (CONT.)

Steam Generator Data Sheet [Ref-1] [Ref-2]

STEAM GENERATOR		
	UNITS	
REFERENCE CONDITIONS		
- Primary side:		
. Design pressure	MPa abs	17.6
. Design temperature	°C	351
- Secondary side:		
. Design pressure	MPa abs	10
. Design temperature	°C	311
TEST CONDITIONS		
- Primary hydrostatic test:		
. Primary side pressure	MPa abs	25.13
. Secondary side pressure	MPa abs	0.1
- Secondary hydrostatic test:		
. Primary side pressure	MPa abs	0.1
. Secondary side pressure	MPa abs	14.95
- Test temperature		As per RCC-M code

SECTION 5.4.2 - TABLE 1 (CONT.)

Steam Generator Data Sheet [Ref-1] [Ref-2]

STEAM GENERATOR		
DIMENSIONAL CHARACTERISTICS AND WEIGHTS	UNITS	
- Steam drum maximum outer diameter (nozzles excluded)	m	5.168
- Overall height	m	24.621
- Tube outer diameter	mm	19.05
- Tube wall thickness	mm	1.09
- Number of tubes		5980
- Overall height of the tube bundle above the tube sheet	m	11.720
- Tube support plate thickness	mm	30
- Heat transfer area	m²	7960
- Total mass-empty	t	500 ± 5 %
- Tube sheet thickness (before cladding)	mm	620
- Channel head manway inside diameter (ID) (after cladding)	mm	516
- Secondary shell handholes (ID)	mm	190
- Upper assembly manways (ID)	mm	600
- Feedwater nozzle		
. Elevation from SG lower support plane	mm	13551.3
. Diameter (inner/outer)	mm/mm	442.9/508
- Emergency feedwater nozzle		
. Elevation from SG lower support plane	mm	15515
. Diameter (inner/outer)	mm/mm	98.3/114.3
- Steam outlet nozzle		
. Elevation from SG lower support plane	mm	23530
. Diameter (inner/outer)	mm/mm	738/813
- Instrumentation taps (level measurement and sampling)		
. Quantity		14
. Diameter (inner/outer)	mm	20/57

SECTION 5.4.2 - TABLE 1 (CONT.)

Steam Generator Data Sheet [Ref-1] [Ref-2]

STEAM GENERATOR		
	UNITS	
DIMENSIONAL CHARACTERISTICS AND WEIGHTS (cont'd)		
- Blowdown taps		
. Quantity		3
. Diameter (inner/outer)		42/89
- Lower support		
. Type		Forged integral pads onto the Channel head
- Upper support		
. Type		Lateral support brackets
. Elevation from SG lower support lane	mm	11545
MATERIALS		
- Tubes		Alloy 690 TT
- Shells, heads, nozzles and tubesheet		18 MND 5 or 20 MND 5
- Cladding (channel head)		Stainless steel
- Cladding (tube sheet)		Ni Cr Fe alloy
- Tube support plates		13% Cr improved Stainless steel

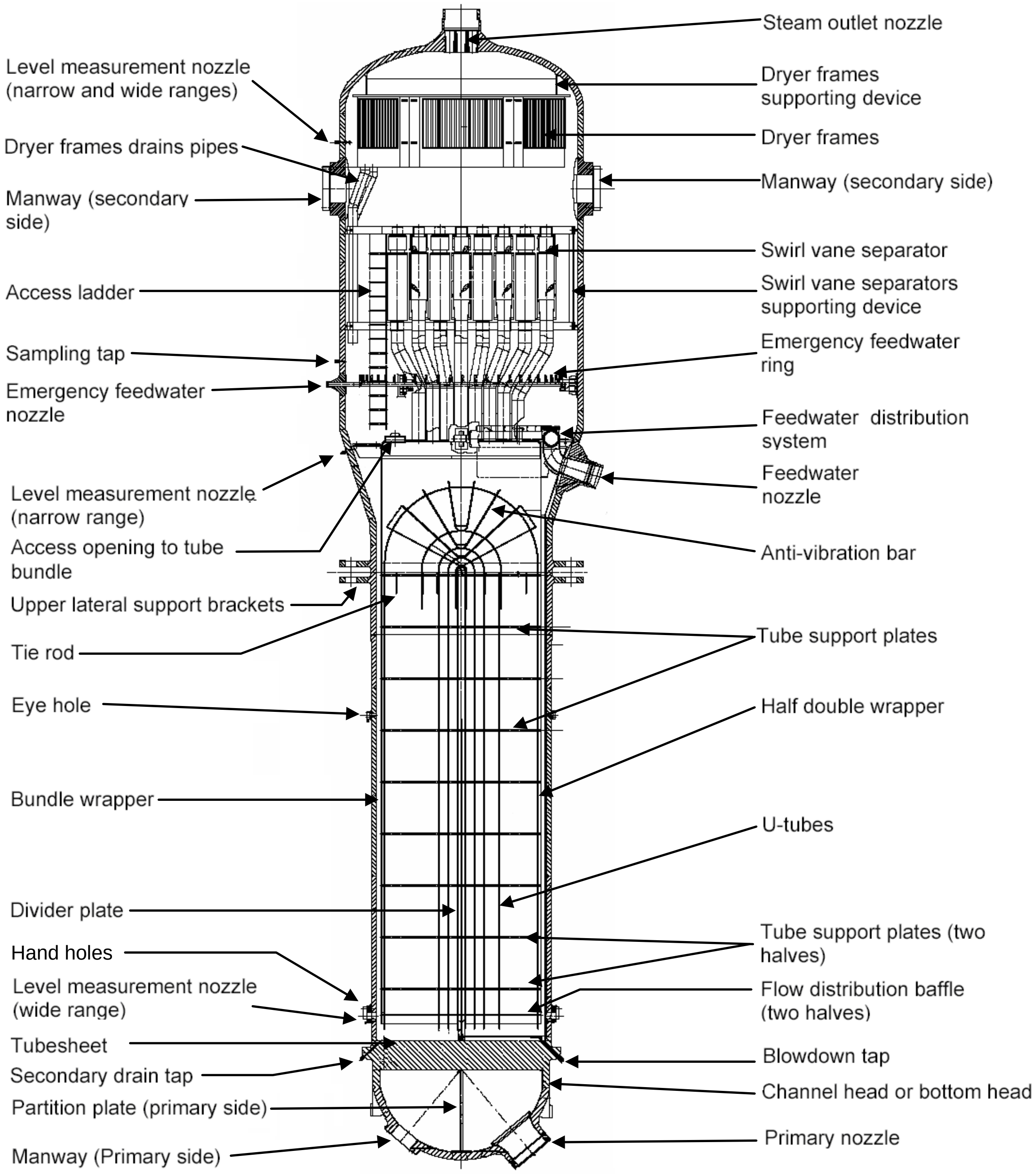
SECTION 5.4.2 - TABLE 2

Additional Operating Parameters [Ref-1]

SG OPERATING PARAMETERS (full load, TH primary capacity)		
	UNITS	
- Water level above tube sheet	m	15.69
- Hot leg circulation ratio		2.37
- Cold leg circulation ratio		1.17
- Global circulation ratio		3.54
- Water mass in SG	t	77.2
- Steam mass in SG	t	5.5

SECTION 5.4.2 - FIGURE 1

Steam Generator Section [Ref-1]



3. REACTOR COOLANT PIPEWORK

This section describes the Main Coolant Lines (MCL), including the Pressuriser Surge Line of the EPR. This pipework is forged in austenitic stainless steel[1].

3.1. DESCRIPTION [REF-1]

The pipes of the four reactor coolant loops and the surge line are located in the reactor building and form part of the reactor coolant pressure boundary CPP [RCPB] belonging to the nuclear steam supply system (NSSS).

The primary pipework carries reactor coolant from the reactor pressure vessel (RPV) to the steam generators (SG) and then to the Reactor Coolant Pumps (RCP). The fluid is then returned to the RPV.

The surge line connects the hot leg of loop 3 to the pressuriser. It is designed to prevent thermal stratification under steady-state operation.

There are four reactor coolant loops, each comprising:

- a hot leg (HL),
- a crossover leg (UL),
- a cold leg (CL).

Large and small branch connections are provided on each leg respectively for auxiliary lines and measurement systems.

<u>Hot leg</u> (HL) (see Section 5.4.3 - Figure 1)

Each hot leg connects the RPV to a SG. It is made of two forged pieces and comprises three straight sections and two elbows [Ref-2].

<u>Crossover leg</u> (UL) (see Section 5.4.3 - Figure 1)

Each crossover leg connects a SG to a Reactor Coolant Pump. It is made of three forged pieces and comprises straight sections and three elbows. The design of the UK EPR crossover leg [Ref-2] has been modified [Ref-3] to improve the inspectabillity of the welds as described in section 3.9; the information in Section 5.4.3 - Figure 1 will be completed after finalisation of the detailed design of the crossover leg.

<u>Cold leg</u> (CL) (see Section 5.4.3 - Figure 1)

Each cold leg connects a Reactor Coolant Pump to the RPV. It is made of one forged piece and comprises two straight sections and one elbow [Ref-2].

[1] - Elbows are inductive bends or machined from bored forged parts
- Large nozzles are built-in on the main pipe

Dimensions

The inside diameter of the crossover leg, hot leg and cold leg was primarily sized to minimise pressure drop in the main coolant lines and to reduce flow velocity.

The nominal dimensions of the main coolant lines are [Ref-4]:

- inner diameter: 780 mm,
- thickness range: 76 – 97 mm.

Surge line (see Section 5.4.3 - Figure 2)

The surge line is comprised of seven weldless sections constructed of forged austenitic stainless steel [Ref-5]. Each section contains one elbow.

The nominal dimensions of the surge line are:

- inside diameter: 325.5 mm,
- thickness: 40.5 mm.

3.2. SUB-ASSEMBLIES DESIGN (NOZZLES AND SLEEVES)

General design (see Section 5.4.3 - Figure 1)

Each reactor coolant loop contains nine homogeneous welds made with austenitic materials.

The connecting welds between auxiliary lines and the primary pipework are also homogeneous welds.

Dissimilar metal welds between heavy components (in ferritic steel) and primary pipework (in austenitic steel) are shop welded (see section 7 of Sub-chapter 5.3). Each dissimilar metal weld is combined with a homogeneous weld to produce a safe end (for RPV and SG). This homogeneous weld is carried out on site.

Branch connection design

Large branch connection (nominal diameter > 150mm)

Large branch connections are integrated with the Main Coolant Lines. They are machined out of forgings. From one to three large branch connections are integrated in the same loop.

Small branch connection (nominal diameter < 150mm)

Small branch connections and bosses are attached to the MCL by welding. An exception to this is for the chemical and volume control system RCV [CVCS] charging nozzles, which are integrated with the Main Coolant Lines (machined out of a forging), to take into account the experience feedback regarding the thermal mixing zone.

Thermal sleeves

If the fatigue analysis of a branch connection leads to a cumulative usage factor exceeding the RCC-M criteria (see Sub-chapter 3.8), then it is protected by the installation of a thermal sleeve. The sleeve is machined from an intermediate forging and is shop welded to the nozzle.

Based on past experience, only the RCV [CVCS] charging nozzles require a penetrating thermal sleeve (PTS). This PTS was designed by AREVA for the French EPR, to avoid any risk of thermal mixing and fatigue damage in the RCV [CVCS] nozzle corner. The fatigue analysis will be performed in the Detailed Design Phase.

3.3. DESIGN CALCULATIONS

A sizing calculation is performed on the primary pipework, the surge line, and the integrated and welded branch connections, in accordance with the RCC-M Code (see Sub-chapter 3.8).

The reinforcement of branch connections is not on the run side, but on the branch side.

The thicknesses of the straight sections and the outside bends are determined in accordance with the RCC-M Code section B3600. The thickness of the inside bends are increased to ensure a suitable stress level.

Furthermore, straight (90°) or inclined (45°) integrated branch connections are checked by calculation according to the rules for the reinforcement of openings of the RCC-M Code section B3600.

Break preclusion

The technology for the Main Coolant Lines complies with the requirements of the Break Preclusion Concept as defined in section 3 of Sub-chapter 5.2.

UK EPR specific fast fracture calculations have been defined in the frame of the safety demonstration of High Integrity Components, as described in section 3.9.

3.4. METHODS AND TOOLS FOR STRESS ANALYSIS

The primary pipework is analysed with loads corresponding to normal, upset, emergency and faulted conditions. The stress analysis performed depends on the operational mode under consideration.

3.4.1. Load conditions

The loads used for the stress analysis are listed in section 1 of Sub-chapter 3.6.

3.4.2. Loads used for stress analysis

The loads used for stress analysis depend on the operating conditions under consideration and on the criteria level.

In accordance with the RCC-M requirements (see Sub-chapter 3.8), loads are classified into four categories:

- design conditions (first category),
- normal operation and upset conditions (second category),
- emergency conditions (third category),
- faulted conditions (fourth category).

3.4.3. Stress analysis tools

Calculations are carried out in accordance with the RCC-M Code sections B3600 and B3200 (see Sub-chapter 3.8).

The RCC-M requirements concern the range of primary plus secondary stresses and total stresses, which require the calculation of temperature transients through the thickness of the component wall.

The transients, as defined in section 1 of Sub-chapter 3.6, are used to compute these thermal variations. Either ROCOCO a one-dimensional finite difference heat transfer computer program, or SYSTAR a three-dimensional finite element computer program, is used to solve the thermal stress calculations.

The outer surface of the pipework is assumed to be adiabatic, while the inner surface boundary experiences the temperature of the coolant fluid. Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness, which varies with time. This temperature distribution is sub-divided into four parts; following RCC-M requirements:

- Average temperature (T_A) is the average through-wall temperature of the pipe which contributes to general expansion loads,
- Radial linear thermal gradient which contributes to through-wall bending moment (ΔT_1),
- Radial non-linear thermal gradient (ΔT_2) which contributes to peak stress associated with shearing of the surface,
- Discontinuity temperature (T_A - T_B) representing the difference in average temperature at cross sections on each side of a discontinuity.

Each transient is described by at least two load sets representing the maximum and minimum stress state during each transient. The construction of the load sets is accomplished by combining the following factors, to calculate the maximum (minimum) stress state during each transient.

- ΔT_1
- ΔT_2
- $\alpha_A T_A - \alpha_B T_B$
- Moment loads due to T_A
- Pressure loads.

The ranges of primary plus secondary stresses and total stresses are computed following the equations of RCC-M section B3653.

For all relevant load set combinations, the primary-plus-secondary and peak stress intensities, elasto-plastic stress correction factors (Ke) and cumulative usage factors (CUF), are calculated in accordance with RCC-M section B3650.

The incremental usage factor is calculated for the combination of load sets yielding the highest alternating stress intensity range. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all fatigue relevant load combinations have been covered. The cumulative usage factor is the summation of the incremental usage factors.

3.5. STRESS CALCULATIONS

A simplified stress analysis of the primary pipework has been performed for pipes, elbows and nozzles.

3.5.1. Primary stresses

In accordance with the equation (9) of the RCC-M Code, calculations demonstrate that, with Basic Design loadings, the criteria defined for primary stresses are fulfilled in each area of the primary pipework.

3.5.2. Fatigue analysis

In accordance with the equation (12) of the RCC-M Code, calculations demonstrate that, with Basic Design loadings (thermal expansion moments), the criteria defined for secondary stresses are fulfilled in each area of the primary pipework.

Analysis of the geometry, thermal loads and number of cycles, shows that all circumferential welds in the reactor coolant system exhibit a usage factor less than 0.1. Specific zones (mixing zones) may have a higher usage factor value whilst remaining within acceptable limit for fatigue damage.

Based on past experience, only the RCV [CVCS] charging nozzles require a penetrating thermal sleeve. The fatigue analysis will be performed in the Detailed Design Phase.

3.6. MATERIAL SELECTION

3.6.1. Base metal

The selection of base metal is the result of significant feedback and manufacturing improvements [Ref-1], taking the following requirements into account, in accordance with the technical rules drawn up for the fabrication of the reactor coolant system:

- excellent mechanical properties with a significant margin (according to the design requirements),
- high toughness values,

- corrosion resistance,
- easy manufacturing and good weldability,
- manufacturing and in-service inspections,
- significant reduction in the number of welds,
- easy replacement after long operating periods (if necessary before the end of the 60-year operating time).

All these requirements are satisfied when forged legs are used, with forged nozzles and elbows, in austenitic stainless steel with very low carbon content.

These materials present the following benefits:

- uniformity of material for all pipework (including connected lines)
- high toughness values in all situations (even in the event of cold water injection)
- ease of on-site welding, no heat treatment needed
- ease of on-site connection of numerous auxiliary lines and the surge line
- improved manufacturing and associated non-destructive tests
- good in-service inspectability.

Grade

The materials grades, used to manufacture the MCL including the surge line, are selected in such as way as to minimise erosion-corrosion and ensure compatibility with the environment in which they are used.

The specified materials are austenitic stainless steels (see Section 5.4.3 - Tables 5 and 6).

For the hot, cold and crossover legs, a single grade is selected: X2 CrNi 19.10 with controlled nitrogen content.

For the surge line, another grade is used: X2 CrNiMo 18.12 with controlled nitrogen content.

For the welded branch connections, the grade is X2 CrNi 19.10 with controlled nitrogen content.

These grades have very low carbon content and therefore require no stabilisation. Parts produced are obtained by forging, with a forging ratio about 3, which provides fine grains and adequately consistent mechanical properties. The parts are delivered already heat-treated in order to remove any risk of intergranular corrosion in-service.

The parts, as delivered, can be welded with no risk of sensitisation to intergranular corrosion and with no loss of mechanical properties even in the heat-affected zone.

3.6.2. Filler metal [Ref-1]

For MCL circumferential welds, a narrow-groove orbital TIG (TOCE) automatic welding procedure with an ER 316L wire (X3 CrNiMo 19.13.03 grade) is used.

For welded nozzles, a manual TIG welding with E 316L coated electrodes is used.

The filler metals used for welding (in-shop and on-site) are assessed to meet qualification procedure requirements, including the following requirements for the chemical analysis [Ref-1]:

- Cr ≥ 18%
- C ≤ 0.030%
- S ≤ 0.020%
- P ≤ 0.025%
- Co ≤ 0.06%
- Ferrite delta = 5% to 15%.

3.6.3. Mechanical properties

The values specified for base forged metals in the MCL and surge line are given in the RCC-M Code (STR M3321) (see Sub-chapter 3.8), with any further requirements detailed in the equipment specifications [Ref-1].

The filler metals used for welding (in-shop and on-site) are assessed to meet qualification procedure requirements, including an assessment to establish whether the welds exhibit the same properties as the base metals.

3.7. MANUFACTURING PROCESS FOR THE MAIN COOLANT LINES AND THE SURGE LINE [REF-1]

Procurement

The loop design and applicable manufacturing technology require the following forged parts to be supplied:

- a hot leg, comprising two separate forged sections: one has the integrated large nozzles and a 6° elbow and the other has a 50° elbow at the SG inlet,
- a cold leg, comprising a single-piece forged section: this is a straight section (with large integrated nozzles (nominal diameter ≥ 150 mm) and RCV [CVCS] charging nozzle) and a 27°30’ elbow at RPV inlet,
- a crossover leg, comprising three forged sections: each consisting of a straight section and an elbow,
- a surge line, using inductive bends when required, made of seven straight sections.

The forging operations are optimised to obtain the required mechanical properties and to satisfy a good ultrasonic permeability in order to assure in-service inspectability.

Previous French manufacturing experience and qualification results demonstrate that the properties will be achieved if the forging ratio is about 3 and the grain size about ASTM 1.

Hot and cold legs

Seamless sections are produced by forging a solid ingot. Heating is carried out several times to permit the successive operations such as blooming, drawing, nozzles forging and final shaping.

A significant amount of machining is needed to bore the inside of the pipes and produce the required shape for the nozzles.

Final machining is carried out before acceptance testing and associated non-destructive tests, which are as follows:

- liquid-penetrant testing of inside and outside finished machined surfaces,
- ultrasonic testing on the entire volume by longitudinal and transverse beams.

Crossover leg

The crossover leg is produced from a hollow ingot. Forging sequences are significantly reduced because of the simple design (no nozzles). Only a few heating cycles are necessary to perform the sequences of hot drawing. Three parts are obtained by sawing the rough product.

Each part is machined before induction bending of the elbow.

After the final solution heat treatment, the exact geometry is achieved either by machining or grinding, before performing the acceptance tests and associated non-destructive examinations. The same examinations are used for the cold and hot legs.

Workshop manufacturing

Shop manufacturing is limited to welding the small nozzles and welding the seamless parts.

Nevertheless, the provision of welded hydrotest blanks is necessary to perform tests at the required pressure according to RCC-M Code section B5000.

After cutting off the hydrotest blanks by mechanical means, final cleaning shall be performed before packaging and shipment to the plant for field erection.

In the mixing zones which experience significant temperature variations, the inside surface treatment of CVCS/CL, SIS/CL and surge line/HL branch connections must have specific roughness values lower than the usual practice to minimise the risk of local thermal hydraulic phenomena.

Field erection

Assembly of primary loop pipework requires only a small number of welds between similar metal. Such welds need no preheating or post-weld heat treatment and are carried out by narrow-groove orbital TIG automatic welding. For one primary loop (see Section 5.4.3 - Figure 3), there are nine homogeneous welds, including six on-site. The intermediate welds of the hot leg (one weld) and the crossover leg (two welds) are carried out in shop.

Subsequent inspections of the welds are:

- liquid-penetrant testing,
- radiographic testing.

All welds on the main coolant lines and surge line that are accessible must be flush with the inside and outside of the pipes.

3.8. INSPECTABILITY [REF-1]

Inspectability is the capacity of the design to allow checks or inspections (installation and materials).

3.8.1. Inspections during manufacturing

Materials used for the main coolant lines must intrinsically be able to be inspected during manufacturing.

Volumetric checks, radiographic or ultrasonic inspections must be carried out on the base metal as well as on the deposited weld metal.

The austenitic stainless steel base metal in the Main Coolant Lines including the Surge Line must be subject to a volumetric inspection. The sensitivity of the inspection has been proved on test specimens.

The choice of inspection method (radiographic or ultrasonic) depends on the shape and location of the most likely defect to arise for a given manufacturing method.

Consequently, ultrasonic inspection is most frequently used for the forged products and radiographic inspection is preferred for welds. As a result, inspection requirements can be summarised as follows:

Main Coolant Lines:

- surface inspection = 100% liquid-penetrant testing of inner and outer surfaces,
- volumetric inspection = 100% ultrasonic testing from the outer surface (inspection from the inner surface may also be carried out for some zones, in particular for single-piece nozzles).

Surge line:

- surface inspection = 100% liquid-penetrant testing of outer surfaces,
- volumetric inspection = 100% ultrasonic testing from the outer surface.

Connection welds:

- surface inspection = 100% liquid-penetrant testing of inner (where accessible) and outer surfaces.

Note: Liquid-penetrant testing must be carried out after the welds have been ground flush.

- volume inspection = 100% radiographic testing of the weld and an area of 10mm each side of the weld, supplemented at the end of manufacturing by specific inspections defined in the frame of the safety demonstration of High Integrity Components, as described in section 3.9.

3.8.2. In-service inspections

The typical areas for in-service inspections are:

- welds,
- base metal areas with a high usage factor (UF > 0.5).

All welds, whose location is indicated in Section 5.4.3 - Figures 1 and 2, can be inspected in-service, in particular by ultrasonic testing (see Section 5.4.3 - Tables 7 and 8).

The procurement requirements applying to base metals (forging ratio and grain size) associated with the welding process used for the main welds (narrow-groove orbital TIG) ensures that, if needed, the resulting ultrasonic permeability achieves the necessary sensitivity for in-service ultrasonic inspection.

3.9. FAST FRACTURE ANALYSIS

As stated in Sub-chapter 3.1, the Main Coolant Lines are a High Integrity Component for which the specific measures described in section 0.3.6 of Sub-chapter 3.4 concerning prevention, surveillance and mitigation contribute to its high integrity demonstration.

With regards to the fast fracture risk, the three-legged approach presented in Sub-chapter 3.4 section 1.6 has been applied to the Main Coolant Lines as follows:

- use of fracture mechanics to determine the end of life limiting defect size and demonstration that the Defect Size Margin between this critical defect size and the detectable defect increased by fatigue crack propagation over the lifetime is larger than 2 as far as practicable;
- use of suitable redundant and diverse inspections during manufacturing, supplemented by the use of qualified inspection(s) at the end of manufacturing;
- verification of the lower bound fracture toughness values used to determine the critical defect size by measurements.

This methodology applies to the whole pipework, including welds and base metal. However, considering the higher probability of finding crack-like defects in welds than in forgings, and considering that the toughness is very high in the austenitic base metal, the majority of the locations assessed are likely to be the welds. Consequently, the following sub-sections focus on welds, and the full demonstration including base metal will be provided in the detailed design stage.

3.9.1. Fracture Mechanics Analysis

For the Main Coolant Lines, the HIC methodology applies to the nine girth welds [Ref-1]. Two bounding cases have been identified and fracture mechanics calculations have been performed on the most severe mechanical loadings and transients:

- RPV Outlet safe-end to Hot Leg weld,
- Reactor Coolant Pump Outlet nozzle to Cold Leg weld,

The smallest End of Life Limiting Defect Size is obtained for the RPV Outlet safe-end to Hot Leg weld. The bounding case corresponds to an outer skin defect submitted to a mechanical loading set including connected line break and earthquake which gives an ELLDS of **19.5 mm** in the ductile range. It has to be noted that this result corresponds to an initiation criteria without taking benefit from ductile tearing.

The limiting defect size calculated ensures that the size of a critical defect that would lead to failure is near 20 mm. This defect size will be used for any of the MCL homogeneous welds to define the requirements for inspections which will be carried out at the end of manufacturing.

3.9.2. Non Destructive Testing

The NDT selected to be qualified for the inspection of MCL welds is the UT technique used in addition to the non-qualified RT technique [Ref-1]. This UT technique is made up of automatic phased array UT using longitudinal and transverse waves up to 70° refraction angle, deployed from outside the pipe, supplemented by a self-tandem UT technique also deployed from outside [Ref-2]. These techniques can cover the volume of all welds and detect all defects of structural concern, including pipe to Reactor Coolant Pump casing inlet and outlet nozzle welds, which are only inspectable from the pipe side.

The demonstration of the performance of the UT selected for the inspection of homogeneous welds, and in particular their capability to detect the hypothetical narrow near-vertical planar weld defects, relies on the following evidence [Ref-2]:

- tests performed in the Sizewell qualification frame showing that defects embedded in a mock-up, including lack of fusion, could be detected with longitudinal waves up to 70° even when scanning only from one side of the weld,
- tests performed in Olkiluoto Pre-Service / In-Service Inspection qualification frame concluding that defects with a size of 5 x 50mm located in the inner third thickness of the weld are detectable using automated phased-array UT when scanning from outside the pipe,
- tests performed in the 1990’s and in 2011 on an AREVA DMW mock-up concluding that all defects can be detected either by phased-array or self-tandem UT. In particular, the use of tandem UT technique provides a robust form of specular detection, and has been recognised as useful [Ref-2].

In order to enable a 70° incidence angle to be achieved at all weld locations a modification of the crossover leg geometry has been implemented and its feasibility verified [Ref-3]: as presented in Section 5.4.3 - Figure 5, the straight pipe sections adjacent to the welds connecting the crossover leg to primary components have been extended by approximately 250 mm by lowering the crossover leg. This modification enables also to overcome the problem of surface profile undulation arising from the fabrication bending process by ensuring that such effects would be outside the UT scanning region.

Moreover, in order to limit the interference between the beam reflecting from the probe and the inner surface when using the self-tandem technique, a modification of the weld preparation has been implemented and its feasibility verified [Ref-4]: the length of counterbores of the MCL pipework have been increased from 25 mm to approximately 100 mm from the weld centreline. This modification also enables the interpretation of UT results to be made more easily when scanning from the outside with the phased-array UT technique.

An ALARP analysis [Ref-5] summarised in section 3 of Sub-chapter 17.5 has been performed to support the adequacy of these two modifications. The capability of this UT proposal to detect and reject defects whose Qualified Examination Defect Size (QEDS) derived from the fracture Mechanics Analyses is 10 mm or greater can be achieved with a high level of reliability; smaller defects can also be detected with reasonable capability.

In conclusion, a Defect Size Margin of 2 has been attained with Fracture Mechanics Calculations while considering that the fatigue crack growth for these worst case defects with regards to fast fracture is negligible; this assumption will be verified during detailed design.

The UT techniques to be applied at the end of manufacturing will be qualified for any UK EPR project.

3.9.3. Fracture toughness

The verification of the lower bound fracture toughness values used to determine the critical defect size will be performed by measurements of the fracture toughness on a mock-up using Compact Tensile specimens (CTJ) in the ductile range. The mock-up will be representative of the EPR materials [Ref-1].

SECTION 5.4.3 - TABLE 1

Nominal Dimensions of Hot Legs [Ref-1]

Inside diameter of straight parts between ends	780 mm
Thickness of straight parts between ends	76 mm
Inside diameter at each end	780 mm
Thickness on steam generator side	97 mm
Thickness on reactor vessel side	76 mm
Inside diameter of taper end (all ends)	785.5 mm
Inside diameter of bent parts	780 mm
Thickness of wall on outer side of bent parts	74 mm
Thickness of wall on inner side of bent parts	90 mm

SECTION 5.4.3 - TABLE 2

Nominal Dimensions of Crossover Legs [Ref-1]

Inside diameter of straight parts between ends	780 mm
Thickness of straight parts between ends	76 mm
Inside diameter at each end	780 mm
Thickness on steam generator side	97 mm
Thickness on pump side	90 mm
Thickness at other ends	76 mm
Inside diameter of taper end (all ends)	785.5 mm
Inside diameter of bent parts	780 mm
Thickness of wall on outer side of bent parts	74 mm
Thickness of wall on inner side of bent parts	90 mm

SECTION 5.4.3 - TABLE 3

Nominal Dimensions of Cold Legs [Ref-1]

Inside diameter of straight parts between ends	780 mm
Thickness of straight parts between ends	76 mm
Inside diameter of each end	780 mm
Thickness at each end	76 mm
Inside diameter of taper end (all ends)	785.5 mm
Inside diameter of bent parts	780 mm
Thickness of wall on outer side of bent parts	74 mm
Thickness of wall on inner side of bent parts	90 mm

SECTION 5.4.3 - TABLE 4

Nominal Dimensions of Surge Line [Ref-1]

Outside diameter	406.4 mm
Inside diameter	325.5 mm
Thickness	40.5 mm
Inside diameter of taper end	328.0 mm

SECTION 5.4.3 - TABLE 5

Chemical Analyses – Base Metal [Ref-1]

ELEMENT	GRADE SPECIFICATIONS	
	X2 CrNi 19.10 N_2	**X2 CrNiMo 18.12 N_2**
C	≤ 0.035 %	≤ 0.035 %
S	≤ 0.015 %	≤ 0.015 %
P	≤ 0.030 %	≤ 0.030 %
Si	≤ 1.00 %	≤ 1.00 %
Mn	≤ 2.00 %	≤ 2.00 %
Ni	9.00 – 10.00 %	11.50 - 12.50 %
Cr	18.50 – 20.00 %	17.00 - 18.20 %
Mo	-	2.25 - 2.75 %
Cu	≤ 1.00 %	≤ 1.00 %
Co	≤ 0.06 %	≤ 0.06 %
B	≤ 0.0018 %	≤ 0.0018 %
N_2	≤ 0.080 %	≤ 0.080 %

SECTION 5.4.3 - TABLE 6

Mechanical Properties of Base Metal for Forged Pipework [Ref-1]

Test	Temperature	Properties	X2 CrNi 19.10 N_2	X2 CrNiMo 18.12 N_2
Tension	Room	$R_{p\ 0.2\%}$	≥ 210 MPa	≥ 210 MPa
		R_m	≥ 510 MPa	≥ 510 MPa
		A% (5d)	≥ 35	≥ 35
	350°C	$R^t_{p\ 0.2\%}$	≥ 125 MPa	≥ 130 MPa
		R_m	≥ 368 MPa	≥ 407 MPa
Charpy V-Notch	Room	Energy (average value)	≥ 100 J	≥ 100 J

SECTION 5.4.3 - TABLE 7

Inspectability of the Primary Loop [Ref-1]

DESCRIPTION	ITEM	INSPECTION
RPV outlet nozzle / Hot Leg	H1	RT-UT-LP-V
Hot Leg / Elbow 50°	H2	RT-UT-LP-V
Elbow 50° / SG inlet nozzle	H3	RT-UT-LP-V
SG outlet nozzle / UL section A	U1	RT-UT-LP-V
UL section A / UL section B (1)	U2	UT-LP-V
UL section B / UL section C (1)	U3	UT-LP-V
UL section C / RCP inlet nozzle (1)	U4	UT-LP-V
RCP outlet nozzle / Cold Leg (1)	C1	UT-LP-V
Cold Leg / RPV inlet nozzle	C2	RT-UT-LP-V
Set-in Branch Connections welds	-	UT partial-LP-V
Branch Connections Auxiliary welds	-	UT-LP-V

Abbreviations: UT (ultrasonic inspection)

RT (radiographic inspection)

LP (liquid-penetrant testing)

V (visual inspection)

1: Radiographic inspection not used owing to access difficulties, other than for U4 and C1 welds if the reactor coolant pump hydraulics are removed.

SECTION 5.4.3 - TABLE 8

Inspectability of the Surge Line [Ref-1]

DESCRIPTION	INSPECTIONS (2)
Hot Leg nozzle / Section A	UT-LP-V
Section A / Section B	UT-LP-V
Section B / Section C	UT-LP-V
Section C / Section D	UT-LP-V
Section D / Section E	UT-LP-V
Section E / Section F	UT-LP-V
Section F / Section G	UT-LP-V
Section G / Pressuriser surge nozzle	UT-LP-V

Abbreviations: UT (ultrasonic inspection)

RT (radiographic inspection)

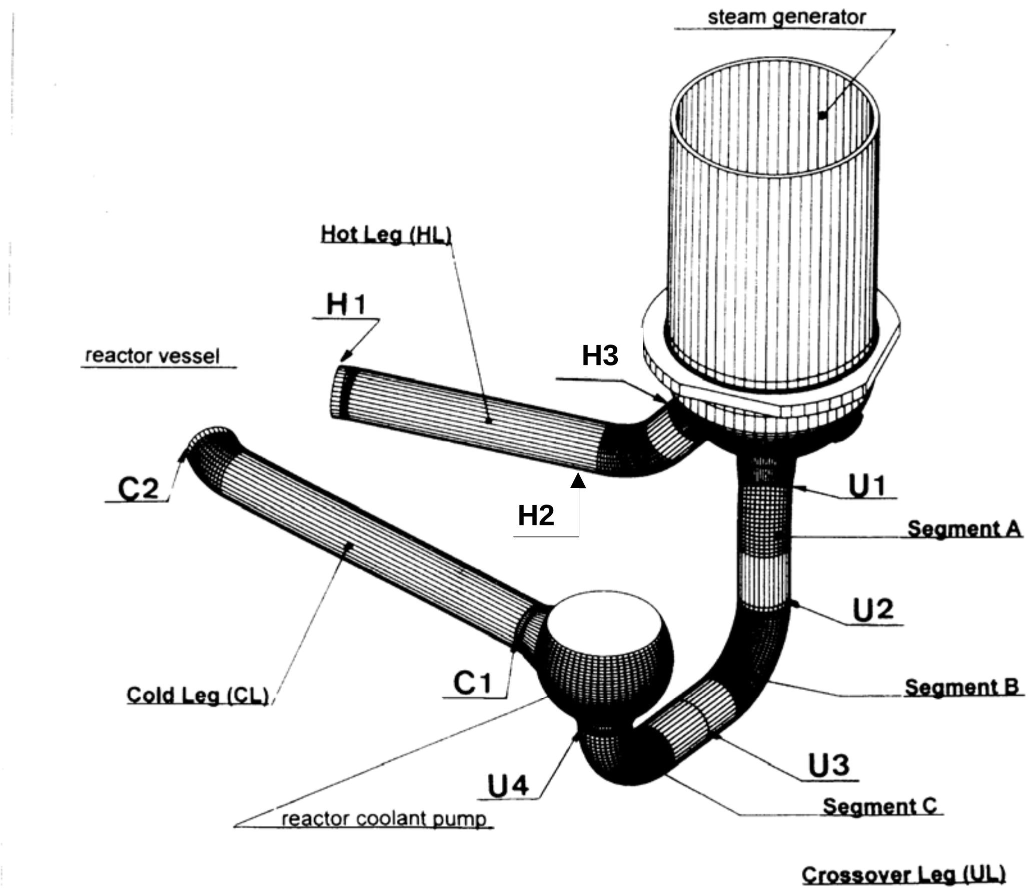
LP (liquid-penetrant testing)

V (visual inspection)

2 Radiographic inspection not used owing to access issues

SECTION 5.4.3 - FIGURE 1

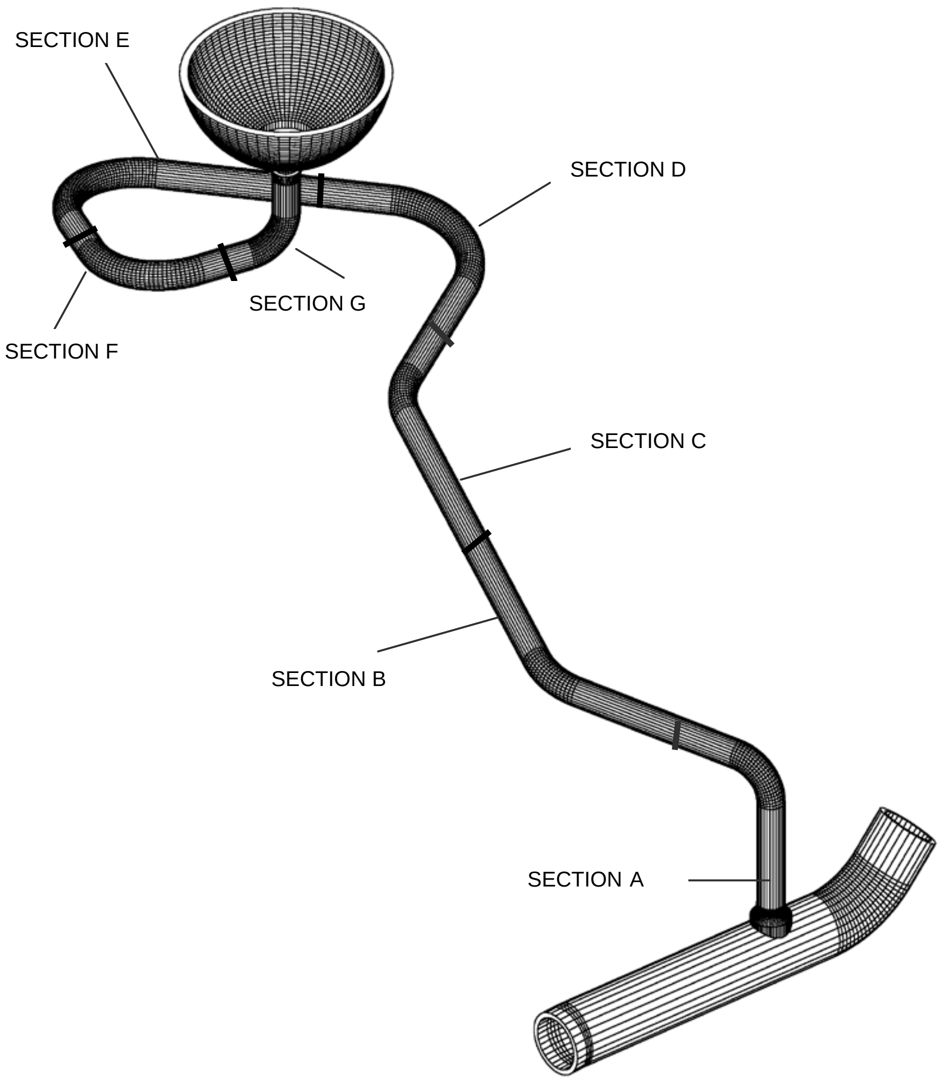
Primary Loop with Weld Location (Hi, Ui, Ci) [Ref-1] [Ref-2]



	HOT LEG	COLD LEG	CROSSOVER LEG
DEVELOP LENGTH	6669 mm	7667 mm	9129 mm
INSIDE SURFACE	16.3 m^3	18.8 m^3	22.4 m^3
METAL WEIGHT	11790 Kg	13055 Kg	*To be defined*
METAL VOLUME	1.47 m^3	1.65 m^3	*To be defined*
FLUID VOLUME	3.18 m^3	3.66 m^3	4.36 m^3

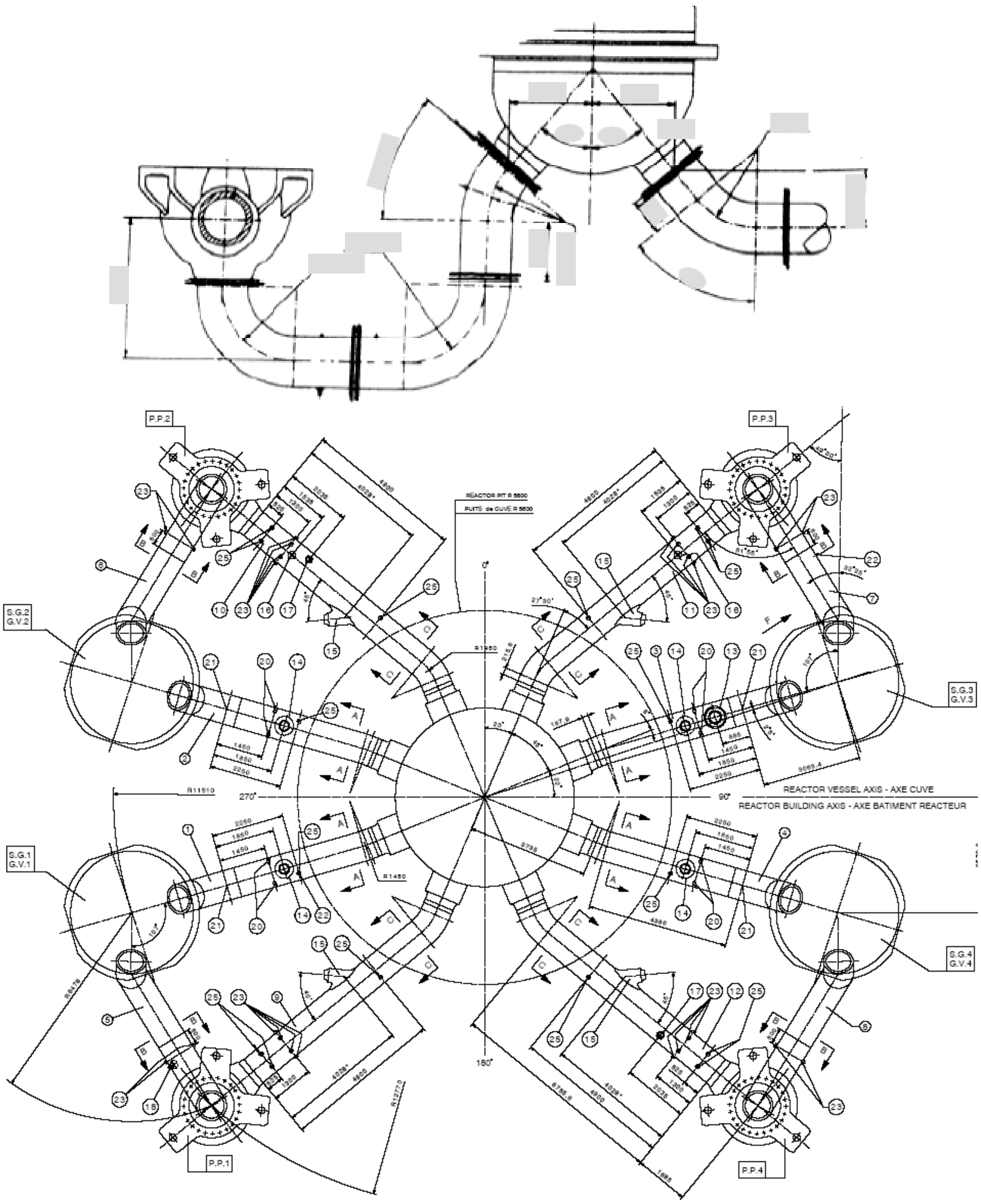
SECTION 5.4.3 - FIGURE 2

Surge Line with Weld Location [Ref-3]



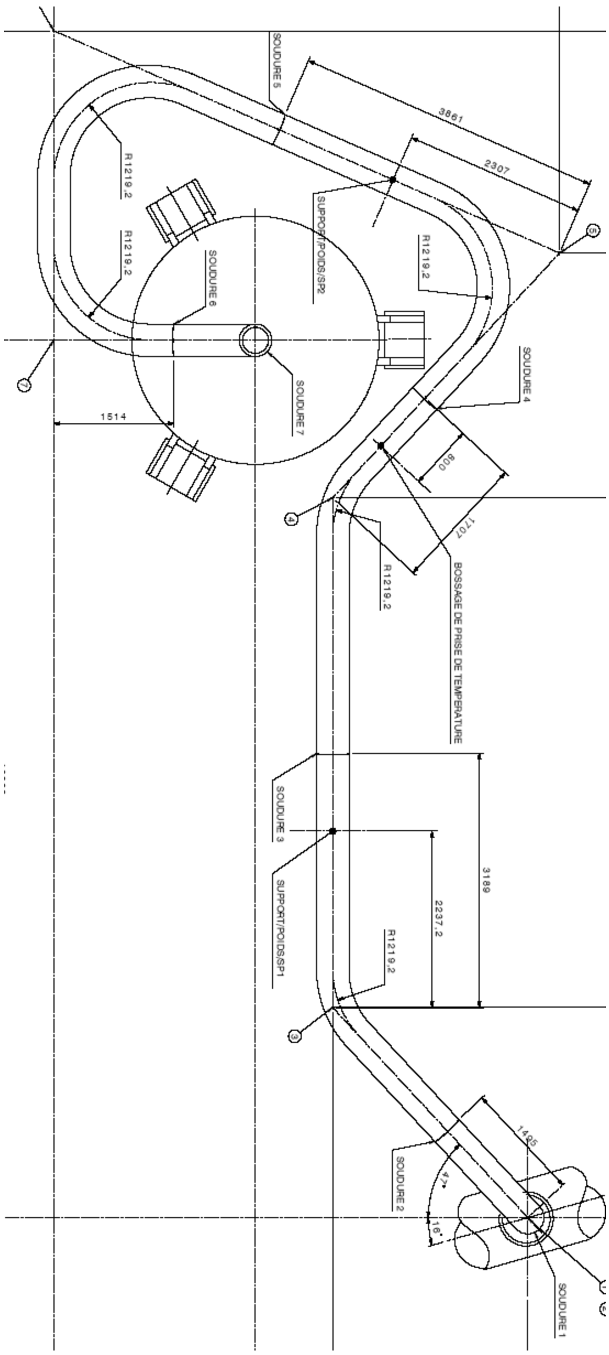
SECTION 5.4.3 - FIGURE 3

Reactor Coolant System [Ref-1]



SECTION 5.4.3 - FIGURE 4

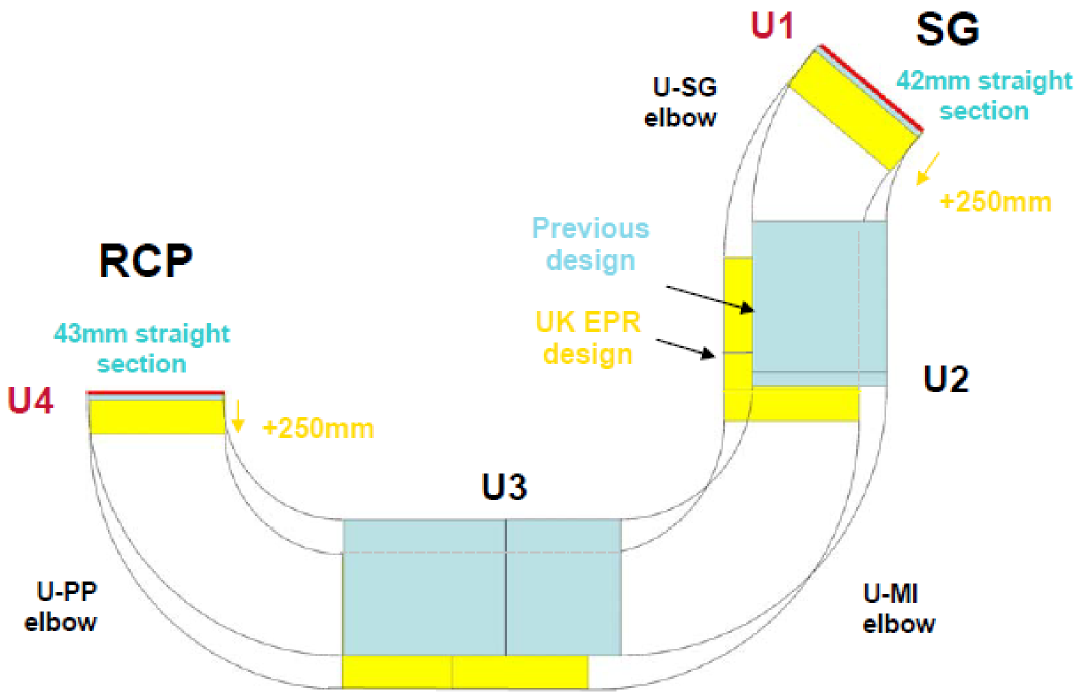
Pressuriser Surge Line [Ref-1]



DIMENSIONS	
I.D.	325,5 mm
THK.	40,5 mm
SLOPE	~ 5°
DEVELOPED	27,4 m
WEIGHT	~10000 kg
DESIGN PRESSURE	17,6 MPa
DESIGN TEMPERATURE	362° C
MATERIAL	X2CrNiMo 18-12

SECTION 5.4.3 - FIGURE 5

Crossover leg modification principle [Ref-4]



4. PRESSURISER

The pressuriser is a “non-breakable” component, which satisfies all the design requirements described in section 6 of Sub-chapter 5.2.

4.1. DESCRIPTION

The pressuriser (PZR) is a pressure vessel forming part of the reactor coolant pressure boundary (CPP [RCPB]). It consists of a vertical cylindrical shell, closed at both ends by hemispherical heads [Ref-1]. It is made of ferritic steel, with austenitic stainless steel cladding on all internal surfaces in contact with the reactor coolant.

- The upper hemispherical head is a hot forged single-piece unit. It is equipped with:
    - three nozzles connected to the pressure relief valves
    - one nozzle connected to the dedicated bleed valve line.

    The first three nozzles are each equipped with a scoop inside the pressuriser in order to maintain a water seal below each valve seat.

    - A manhole providing access inside the pressuriser.
    - A vent nozzle
- The forged cylindrical shell consists of three sections. It is equipped with:
    - upper (steam phase) instrument nozzles
    - lateral bracket supports
    - The lateral spray system consists of three separate nozzles welded laterally near the top of the upper cylindrical shell:
        - Two nozzles for the main spray lines (connected to two cold legs)
        - One nozzle for the auxiliary spray line (connected to the RCV [CVCS])

    The three spray nozzles have integral welded thermal sleeves. Each thermal sleeve is extended by a lance. The end of each lance holds a spray box with screwed spray heads which inject spray flow into the pressuriser steam space.
- The lower hemispherical head is a hot-formed single-piece unit. It is equipped with:
    - the axial surge line nozzle
    - lower (water phase) instrument nozzles.
    - heater sleeves equipped with connecting flanges

- A screen installed at the surge line nozzle in the bottom head which prevents the passage of loose parts from the pressuriser to the reactor coolant pipework.

The pressuriser is equipped with 116 heater rods, including 8 spare heaters, arranged vertically, inserted into the heater sleeves. There are no spare sleeves without heaters. The heaters are mounted using flanged connections for easy replacement. They are similar in manufacture to spare heaters currently produced for existing plants.

Section 5.4.4 - Table 1 presents the main characteristics of the heaters.

The heater flanged connections are comprised of the following parts:

- heater sleeves, welded to the inner cladding of the bottom head after the final post weld heat treatment of the pressuriser
- open flanges of austenitic stainless steel with threaded holes (replaceable)
- slip-on flanges (upper flange) of austenitic stainless steel, installed prior to welding the heater sleeves
- heater flange attachments, welded on the heater sheath, containing grooves for O-ring seal
- metal O-ring seals
- slip-on flanges (lower flange) of austenitic stainless steel
- studs and nuts

Two areas are free of heater penetrations, these being the central area around the surge line nozzle and the area located above the surge line routing. This allows access to the heaters for maintenance and replacement.

The pressuriser has thermal insulation on the outside surface.

4.2. OPERATING CONDITIONS AND INTERFACES

The operating functions of the pressuriser and its associated equipment are:

- the RCP [RCS] pressure boundary function as a part of the reactor coolant system and the second barrier
- the RCP [RCS] volume control (coolant expansion vessel of the RCP [RCS])
- the RCP [RCS] pressure control, overpressure protection and depressurisation functions

These functions are provided by:

- the presence of water and steam phases in the pressuriser vessel
- the normal and auxiliary spray systems

- the heaters
- the pressuriser pressure relief valves.

The interfaces providing these functions are:

- Interface with the RCP [RCS] hot leg: The pressuriser is connected to hot leg N° 3 of the RCP [RCS] through the surge line. This connection allows continuous adjustment of the volume and pressure between the reactor coolant system and the pressuriser
- Interface with the reactor coolant system cold legs: Two main spray nozzles are connected to two spray lines from two cold legs of the reactor coolant system. One of the cold legs belongs to the same reactor coolant system loop as the one connected to the surge line. The spray water is injected into the steam volume as fine droplets, creating an instantaneous condensing surface
- Interface with the RCV [CVCS]: an auxiliary spray pipeline is connected to the RCV [CVCS]. The auxiliary spray water has a much lower temperature than the normal spray water
- Interface with the pressuriser relief tank: three nozzles are connected to the pressuriser relief valves which discharge into the pressuriser relief tank and into the reactor building if the pressuriser relief tank rupture disc fails. An additional nozzle is connected to the dedicated bleed line in the event of a severe accident

4.3. DESIGN PRINCIPLES AND OBJECTIVES

The design objectives and main characteristics of the pressuriser are as follows:

- reliable operation and suitability for all operating conditions and loading, by choosing an appropriate structural design which minimises as far as possible the stress levels and the stress distribution
- reduced fatigue in all loaded points for the EPR requirement of 60 years design life [Ref-1].
- selection of acceptable and proven materials
- use of good manufacturing practice, following industrial techniques used by traditional manufacturers and complying with manufacturing and in-service inspection requirements
- a design which allow easy access for maintenance and in-service inspections
- a design which reduces personnel radiation exposure.

4.3.1. Main characteristics

Surge line nozzle

The surge line is connected to the pressuriser via the surge line nozzle which is located vertically in the centre of the bottom head.

This design limits the effects of excessive thermal loads on the nozzle under normal operations and excursions. The surge nozzle is equipped with a thermal sleeve opened at the lower end to avoid the accumulation of radioactive particles.

Loads resulting from thermal expansion of the pressuriser vessel are minimised by the short distance between the nozzle and the lateral supports, the vertical position and the surge line route.

The axial location of the nozzle at the lowest point of the pressuriser helps continuous sweeping of the pressuriser bottom area to avoid stagnant areas and the deposition of radioactive particles.

The surge line nozzle is made of ferritic forged steel, and provided with an austenitic safe end. The welding metal used for the bimetallic weld is Inconel 52. The nozzle is clad with austenitic stainless steel on all surfaces in contact with the primary coolant

Spray system

The spray system is located at the top of the pressuriser upper shell and comprises three spray nozzles [Ref-1]:

- Two lines are connected to two of the reactor coolant system cold legs (one of these two loops being the surge line loop) and provide the normal spray function in the pressuriser

- One line is connected to the RCV [CVCS] system

In order to protect the spray pipelines against excessive thermal loads and to reduce fatigue damage as much as possible, the spray nozzles are fitted with thermal sleeves.

Each spray lance is extended with a water box equipped with screwed spray nozzles, which provide a fine droplet spray.

The distance between the spray nozzles and the area where spray fluid hits the pressuriser wall is relatively large. The size of droplets is relatively small due to the choice of small spray nozzles and ensures good heat transfer to the droplets before they reach the pressuriser wall. This design limits the risk of thermal fatigue in the spray/pressuriser wall contact area.

The spray system and its component parts are easily accessible for inspection and maintenance. The entire replacement of a spray lance with its spray nozzles can be carried out.

Relief valve and dedicated bleed valve connections

Three nozzles are connected to the pressure relief valves. An additional nozzle is connected to the dedicated line used in the event of a severe accident. They are located on the upper hemispherical head of the pressuriser on which are also provided three taps for the safety valve pilots. The relief valves are shielded from spray pipeline radiation by a concrete floor.

Connection to heaters

The pressuriser heaters are installed vertically in the bottom of the pressuriser. They are connected by bolted flanges to the pressuriser heater sleeves.

The flanged connection is a design proven on the Konvoi pressuriser [Ref-2].

The flanged connection allows easy and reliable replacement of defective heaters.

A heater support plate is installed in the pressuriser bottom head. It is designed with a wide central opening which allows access to the surge line nozzle retaining screen for inspection purposes.

Supports

The pressuriser is supported by three brackets welded to the lower cylindrical shell. These brackets rest on the supporting floor by means of an intermediate supporting structure which allows free radial thermal expansion.

These supports accommodate all operating loads from normal to accident conditions. For accident loads, eight radial stops fixed on the civil works at the level of the pressuriser centre of gravity provide stability to the pressuriser.

These upper lateral supports allow free radial and vertical thermal expansion of the vessel.

See also the section dedicated to primary equipment support in section 9 of this sub-chapter.

Instrument lines

The instrument taps are small diameter nozzles welded to the pressure retaining wall cladding. The pressuriser is equipped with:

- Eight taps used for pressuriser level measurement

Four of the eight taps are all installed at the same level, near to the top of the cylindrical shell (steam volume). The other four taps are all installed at the same level in the bottom end (water volume).

The distance between the two levels represents the level measuring range (scale).

The taps are designed and configured to provide reliable level-measurements.

- one sample line tap provides samples for water chemistry analysis
- two temperature measurement taps, one located in the steam volume and the other in the water volume, are provided at the same levels as the pressure/level measurement taps.

Manway

The manway flange is located on the upper hemispherical head, on the pressuriser centreline.

Rigid construction combined with a rigid blank cover limits deformation due to pressure in the sealing area and ensures proper leak tightness of the closure.

The sealing elements are an "expanded graphite" type gasket or some other proven seal.

The manway opening provides access to the interior of the pressuriser for inspection and maintenance.

A small degassing tap is provided in the manway nozzle as a complement to the venting nozzle to allow the complete removal of non-condensable gases.

4.3.2. Main dimensions

The main dimensions, characteristics, and nozzles connected to the pressuriser are listed in Section 5.4.4 - Table 2.

4.3.3. Functional requirements

The functional requirements for the pressuriser are [Ref-1] [Ref-2]:

a) design pressure: 17.6 MPa

b) design temperature: 362°C

c) RCP [RCS] volume control : The pressuriser volume is sufficient to meet the following requirements:

- the volumes of water and steam combined are sufficient to meet the desired pressure response caused by changes in RCP [RCS] system volume
- the water volume is large enough to prevent the heaters being uncovered in PCC-2, PCC-3, and PCC-4 conditions and at the same time large enough to accommodate coolant expansion between 0% and 100% of the power level under PCC-1 conditions
- the steam volume is large enough to accommodate overpressure protection requirements in respect of the RCP [RCS] overpressure criteria in PCC-2 to PCC-4 conditions
- fluctuation in steam pressure during normal operation should avoid frequent actuation of the pressure regulation devices
- the pressuriser will not empty following reactor trip or turbine trip
- the safety injection signal will not actuate during reactor trip or turbine trip

d) Pressuriser pressure regulation:

Three spray nozzles are located in the upper section (two separate nozzles for normal spray, and one for auxiliary spray).

The heaters are located in the lower section of the pressuriser (water volume).

Three nozzles for the relief valves connection and one for the dedicated bleed valves connection are located in the upper part of the pressuriser head (steam volume).

e) Surge line requirements:

The surge line connects the pressuriser to a reactor coolant system hot leg. The surge line is connected vertically to the nozzle at the bottom of the pressuriser.

The surge line differential pressure (ΔP) during overpressure transient conditions (rising flow) is within the maximum allowable pressure loss.

$\Delta P < 2$ bar for an insurge of up to 2500 m³/h (from reactor coolant pump to PZR water volume),

and

$\Delta P < 5$ bar for an insurge of up to 5000 m³/h (from reactor coolant pump to PZR water volume).

4.3.4. Requirements for inspection, repair and replacement

Inspection

The outer surface of the pressuriser around the butt welds can be fully inspected.

The shape and slope of welded parts, including safe-end-to-nozzle welds, allow both radiographic and ultrasonic examination.

The thermal insulation can be removed from all areas subject to in-service inspection such as:

- circumferential welds on the body of the pressurised vessel (there are no longitudinal welds)
- nozzle welds on hemispherical heads
- lateral support bracket welds

The nozzle-to-head welds are sufficiently remote from other welds to allow performance of required ultrasonic examination.

Inspection of heater sleeve welds can be carried out through the sleeve after heater removal.

The inner cladding inspection may be performed from the outside by a remote-controlled camera. Access is via the manway opening.

The pressuriser design does not require the presence of personnel inside the pressuriser to carry out in-service inspections. All inspections are possible from the outside and some may be accomplished using automatic inspection tools.

Repair

The repair of the pressuriser is made possible for the following components or areas where fatigue, corrosion-erosion, seizing or ageing may occur:

- Manway threads (mechanical damage, threading tearing): repair by threaded inserts

- Manway sealing surfaces (scratches, tearing or corrosion): the flat surface design of the sealing surface and cladding extra thickness allow easy repair by machining

Replacement

The following components or parts can be replaced if necessary:

- manway studs and nuts
- spray lances
- spray heads
- heater rods (flanged connection)
- heater rods studs, nuts and threaded open flange

4.4. MATERIAL PROPERTIES [REF-1]

All materials used in the pressuriser pressure vessel construction comply with RCC-M requirements.

Basic materials

Low-alloy ferritic steel (18 MND 5 or 20 MND 5) is selected for the shells, hemispherical heads, main nozzles and the lateral bracket supports. A comparison of the chemical composition and mechanical characteristics shows that these two materials are similar [Ref-2] to [Ref-4].

The determination of the initial RT_{NDT} is based on both Pellini and Charpy V notch tests.

The initial RT_{NDT} for the pressuriser shells and nozzles is less than -20°C.

Studs and nuts

The pressuriser studs are small diameter studs (D < 60 mm) made of high-strength bolting steel.

Pressuriser safe ends, heater wells and instrument nozzles

The safe ends are welded to the pressuriser nozzles during manufacture in the factory. The safe ends are manufactured from austenitic stainless steel forged bars.

The welding of safe ends onto the nozzles is carried out with a Ni-Fe-based alloy without previous buttering.

Cladding material

The internal cladding of the pressuriser is applied in two successive layers using austenitic stainless steel welding strips.

Heater wells and instrument nozzles welding

The heater wells and instrument nozzles are welded to the internal cladding. The cladding thickness is locally increased in the weld area.

4.5. MECHANICAL DESIGN [REF-1]

This section presents the main results of sizing calculations for the main parts and sub-assemblies, primarily the pressurised vessel and closure parts.

For mechanical design, the pressuriser is a class 1 RCC-M component.

The design life is 60 years.

4.5.1. Sizing calculations [Ref-1]

a) The thicknesses of the pressure retaining walls are determined on the basis of the design pressure and design temperature.

b) The manway closure assembly is sized taking into account the design conditions (pressure and temperature) and the mechanical characteristics of the gaskets, as provided by the gasket manufacturer: a graphite expanded type gasket or other proven design is used for the manway assembly (relatively frequent openings).

4.5.2. Design of sub-assemblies

Analysis of the surge nozzle behaviour

A fatigue evaluation of the surge nozzle was performed in order to verify the acceptability of usage factors for the 60-year design life. The fatigue calculation was based on the most onerous dimension changing transients (during heat-up and shutdown of the plant unit). The results demonstrate that the usage factor is acceptable at each point in the nozzle.

Pressure relief valve nozzles

The relief valves nozzle loads were calculated considering the discharge forces. Safe end calculations have been carried out; this area being considered to be the most stressed due to the geometry and materials properties.

The stresses in nozzle safe ends have been calculated for pressure, temperature and external moments (using the set of loads given for second category conditions and for accident conditions) resulting from the pipework calculations. The calculated stresses are acceptable with very large margins in the weakest points of the structure.

Lateral fastening support welds

The stresses in brackets support welds on the pressuriser shell were calculated based on the loads given by the loop analysis results. Stresses induced in the welds are acceptable for all operating conditions and accident situations.

4.5.3. General layout drawing

The general drawing of the pressuriser is shown in Section 5.4.4 - Figure 1 and Section 5.4.4 - Figure 2.

UK EPR specific fast fracture calculations have been defined in the frame of the safety demonstration of High Integrity Components, as described in section 4.7.

4.6. MANUFACTURING AND PROCUREMENT

The pressuriser vessel is manufactured from the following parts:

- Forged cylindrical shells
- Hot-formed hemispherical heads
- Forged nozzles
- Forged plates for covers
- Forged safe ends
- Forged bars for small diameter branch pipes
- Plates for lateral supports
- Plates for heater support

Section 5.4.4 - Table 5 lists procurement conditions for the main components and parts.

Cladding of parts in the pressurised vessel utilises stainless steel strips which are deposited using automatic welding with manual finishing of the circumferential welds.

The safe ends are welded to the ferritic forged parts using a narrow groove welding process using a Ni-Fe-based alloy with 30% Cr welding material.

Pre- and post-welding heat treatment and final heat treatment of welds must comply with RCC-M requirements.

During the final stage of the manufacturing process, the weld surfaces and transitions must be prepared in order to allow a surface inspection to be carried out (liquid penetrant testing, magnetic particle inspection) and volume inspections (radiographic, ultrasonic).

UK EPR specific inspections have been defined in the frame of the safety demonstration of High Integrity Components, as described in section 4.7.

The final surface finish of the cladding must be suitable to allow inspection by liquid penetrant and ultrasonic tests.

Pressuriser shell and ends have no longitudinal welds

4.7. FAST FRACTURE ANALYSIS

As stated in Sub-chapter 3.1, the Pressuriser (pressure boundary parts) is a High Integrity Component for which the specific measures described in section 0.3.6 of Sub-chapter 3.4 concerning prevention, surveillance and mitigation contribute to its high integrity demonstration.

With regards to the fast fracture risk, the three-legged approach presented in Sub-chapter 3.4 section 1.6 has been applied to the Pressuriser as follows:

- use of fracture mechanics to determine the end of life limiting defect size and demonstration that the Defect Size Margin between this critical defect size and the detectable defect increased by fatigue crack propagation over the lifetime is larger than 2 as far as practicable;

- use of suitable redundant and diverse inspections during manufacturing, supplemented by the use of qualified inspection(s) at the end of manufacturing;

- verification of the lower bound fracture toughness values used to determine the critical defect size by measurements. .

This methodology applies to the whole component, including welds and base metal. However, considering the higher probability of finding crack-like defects in welds than in forgings, and considering that the toughness is lower in the weld metal, the majority of the locations assessed are likely to be the welds. Consequently, the following sub-sections focus on welds, and the full demonstration including base metal (e.g. nozzle corners) will be provided in the detailed design stage.

4.7.1. Fracture Mechanics Analysis

For the Pressuriser, the HIC methodology applies to the four circumferential welds and to the six main nozzles welds [Ref-1]. Two bounding cases have been identified and fracture mechanics calculations have been performed on the most severe transients:

- Spray line nozzle set-in weld,

- Medium shell circumferential weld,

The smallest End of Life Limiting Defect Size is obtained for the spray line nozzle weld based on an analytical calculation and corresponds to outer skin defect submitted to a cold overpressure transient which gives an ELLDS of **27.5 mm** in the fragile range.

The limiting defect size calculated ensures that the size of a critical defect that would lead to failure is above 20 mm. This defect size will be used for any of the Pressuriser welds to define the requirements for inspections which will be carried out at the end of manufacturing.

Fast fracture mechanics analyses will be performed in sensitive regions of the main forgings in the detailed design stage, to check whether these are limiting.

4.7.2. Non Destructive Testing

The NDT selected to be qualified for Pressuriser welds are [Ref-1]:

- the pulse echo UT technique required by the RCC-M code supplemented by additional beam angles, and

- a UT specular reflection technique.

  The specular reflection Tandem technique can be applied to all the main circular welds and is used to detect certain type of plausible defects such as near vertical defects which may not be detected otherwise using conventional UT.

The RT control required by the RCC-M code will continue to be applied extensively but as a non-qualified control before heat treatment on all of the main component welds.

The geometry and reflection angle of nozzle and manhole welds do not enable inspection with the UT tandem technique. For these welds inspection will be made using qualified pulse echo UT with longitudinal waves 0° from inside the nozzle supplemented by qualified RT for some cases where longitudinal waves 0° cannot be fully performed. The radiographic examination is a highly flexible technique which can be applied to any of the component welds.

The sensitivity of the qualified pulse echo UT selected for ferritic component welds has been determined in the frame of ISI and enables detection of a flat bottom hole perpendicular to the beam axis with a diameter greater than 3.1 mm. The equivalent performance is the detection of a defect with through wall extent of 3 mm and length 18 mm.

For circular welds, the qualified UT tandem technique increases the detection capability for smooth vertical defect through the whole thickness. Its sensitivity is based on a flat bottom hole of 6 mm and it has been demonstrated that this technique can detect and reject defects with 5 mm and 10 mm through wall extent.

This UT tandem technique is supplemented by 0° longitudinal waves from inside the nozzle which enables a specular reflexion with good coverage of the weld.

Finally, a Defect Size Margin of 2 has been attained with Fracture Mechanics Calculations while considering that the fatigue crack growth for these worst case defects with regards to fast fracture is negligible; this assumption will be verified during detailed design.

The UT techniques (and in some cases RT technique) to be applied at the end of manufacturing will be qualified for any UK EPR project.

4.7.3. Fracture Toughness

The verification of the lower bound fracture toughness values used to determine the critical defect size will be performed [Ref-1]:

- for the base metal by measurements of the fracture toughness on the forgings of any UK EPR project, in the ductile range,

- for the weld material on a mock-up using Compact Tensile specimens (CTJ) in the ductile range. The mock-up will be representative of the EPR material, either base metal or wire/flux.

SECTION 5.4.4 - TABLE 1

Heaters Characteristics [Ref-1]

CHARACTERISTICS	VALUE	UNIT
Individual power rating	24	kW
Design voltage	400	V
Outside diameter of the rods	22	mm
Heating length inside the pressuriser	1200	mm
Maximum surface wattage	32	W / cm²
Number of heaters	108	-
Total nominal installed power	2592	kW
Installed power / PZR volume	34.56	kW / m³

SECTION 5.4.4 - TABLE 2

Main Dimensions and Characteristics of the Pressuriser [Ref-2]

FEATURE		**VALUE**	**UNIT**
Internal volume at 20 °C		75	m^3
Internal diameter measured at the ferritic wall		2820	mm
Spherical heads inside radius at the ferritic wall		1430	mm
Cylindrical shell thickness		140	mm
Upper head thickness		120	mm
Lower head thickness		120	mm
Cladding thickness		5	mm
Heaters sleeves thickness		6 (current part) 4.2 (locally)	mm
Cylindrical shell length		10740	mm
Pressure retaining body (internal) length		13103	mm
Total (overall) pressuriser length		≈ 14400	mm
NOZZLES	QTY	DIAMETER	
Surge line nozzle internal diameter	1	325	mm
Safety valve nozzle internal diameter	3	132	mm
Normal spray line nominal diameter	2	DN 100	mm
Auxiliary spray line nominal diameter	1	DN 100	mm
Dedicated bleed valve nozzle	1	132	mm
Venting nozzle	1	66.9	mm
Manway diameter	1	533	mm
Number of heater sleeves	116	23	mm
Total weight: tare, as delivered		150000	kg
Total weight filled with water (hydrostatic test)		225000	kg

SECTION 5.4.4 - TABLE 3

Chemical Composition required for the Base Material (Heat) [Ref-1] [Ref 3]

CODES	**RCC-M**	
content / element	**18 MND 5**	**20 MND 5**
C (%)	0.20 (max)	0.22 (max)
Mn (%)	1.15 – 1.60	1.20 – 1.50
P (%)	0.008 (max)	0.008 (max)
S (%)	0.008 (max)	0.008 (max)
Si (%)	0.10 – 0.30	0.15 – 0.30
Ni (%)	0.50 – 0.80	0.40 – 1.00
Cr (%)	0.25 (max)	0.25 (max)
Mo (%)	0.45 – 0.55	0.45 – 0.60

SECTION 5.4.4 - TABLE 4

Thickness Evaluation for Pressurised Parts [Ref-1]

Designation	Inner radius (mm)	Sm* (MPa)	Thickness choice (mm)
Cylindrical shells	1410	200	140
Spherical heads	1430	193	120 (Reinforcement)
Instrument nozzles	10	114	10
Heaters sleeves	11.4	114	3,6
Surge nozzle	168.5	200	189.7 (Reinforcement)
Surge nozzle safe end	162.5	114	40
Safety nozzle	73	200	67 (Reinforcement)
Safety nozzle safe end	66	114	39 (External loads)
Spray nozzle	105	200	145 (Reinforcement)
Spray nozzle safe end	100	114	25
Venting safe end	34.45	114	11
Manway forging	266.5	200	170
Manway cover		193	175

Nota * : Sm values based on 18MND5

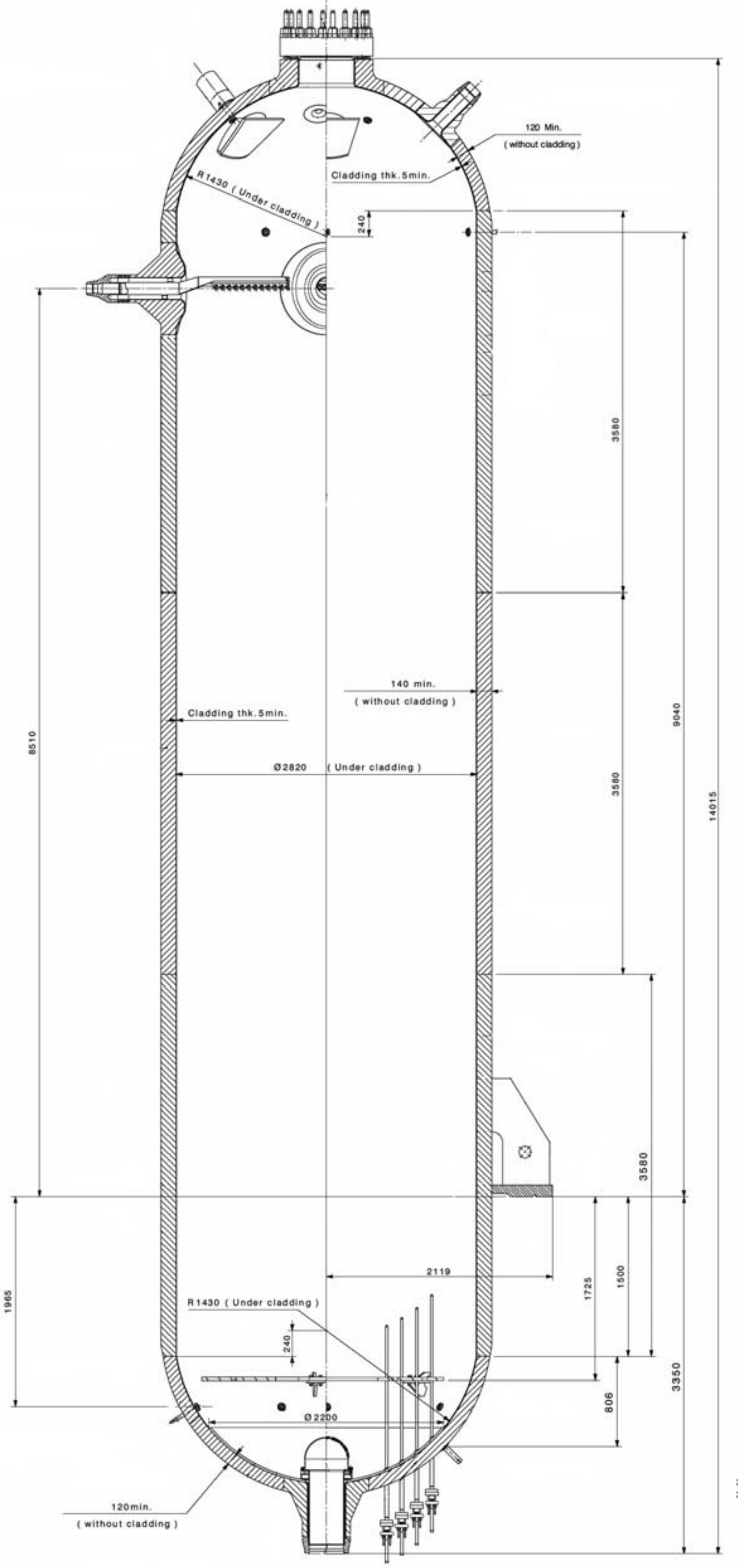
SECTION 5.4.4 - TABLE 5

Main Parts [Ref-1]

PARTS	MATERIAL GRADE	PRODUCT
Cylindrical shells	18 MND 5 or 20 MND 5	Forged
Spherical heads		Hot formed plates or forgings
Main nozzles		Forged
Safe ends	Z2 CND 18.12 + N2	Forged
Manhole cover Lateral supports	18 MND 5 or 20 MND 5	Plate Plates
Heater support	Z2 CND 18.12 + N2	Plate
Small nozzles and heater sleeves	Z2 CND 18.12 + N2	Forged bars
Bolts	42 CDV 4	Forged bars
Heater flanges	Z2 CND 18.12 + N2	Rolled or forged bars

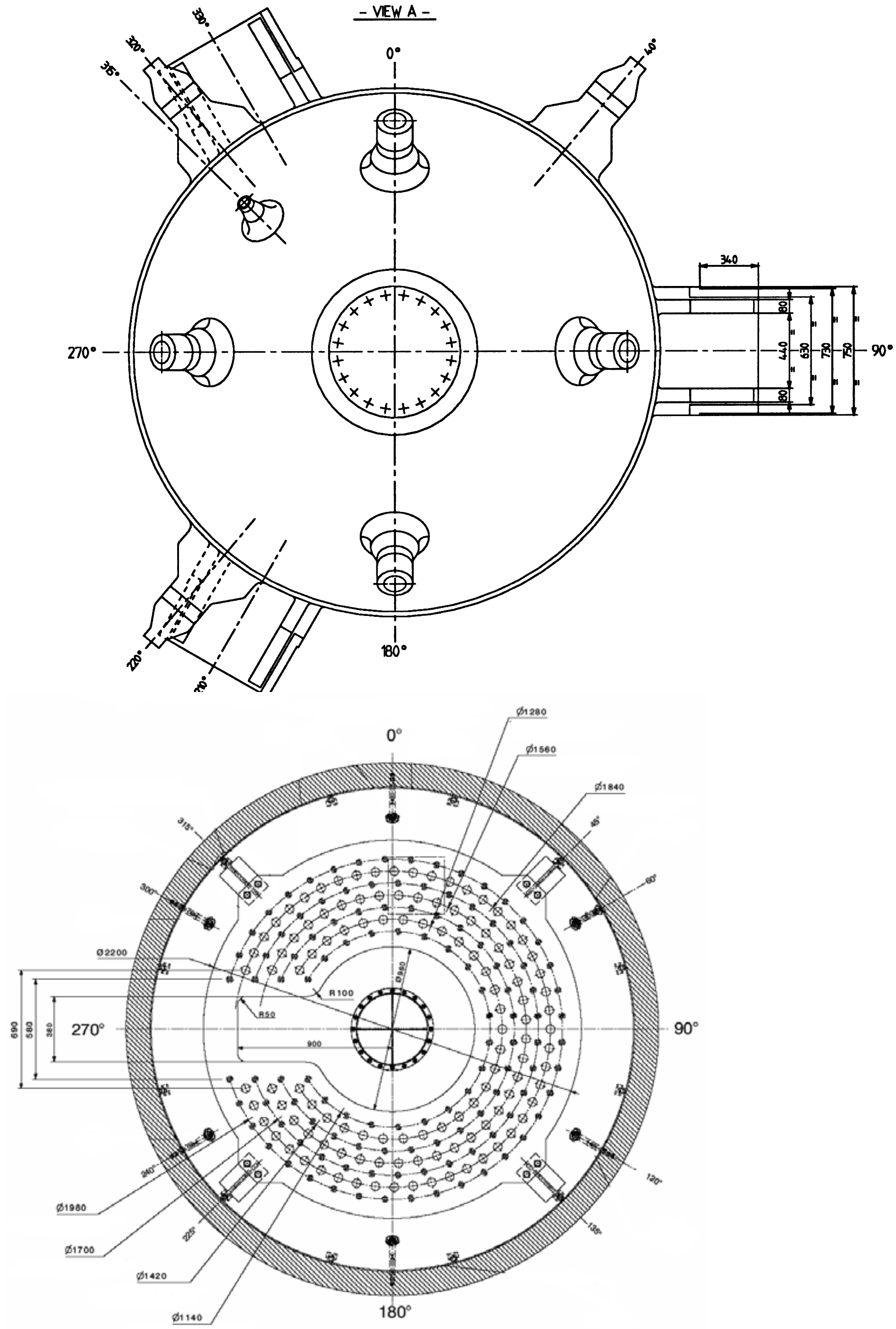
SECTION 5.4.4 - FIGURE 1

Longitudinal View of the Pressuriser [Ref-1]



SECTION 5.4.4 - FIGURE 2

Axial View of the Pressuriser [Ref-1]



5. PRESSURISER RELIEF LINE

5.1. SAFETY FUNCTIONS AND FUNCTIONAL ROLE

5.1.1. Safety role

The pressuriser relief line collects and transports steam and water discharged by the Pressuriser Safety Relief Valves (PSRVs) to the Pressuriser Relief Tank (PRT). If rupture disks burst, steam is discharged into the reactor building at the level of the reactor coolant pump bunkers. Steam thus discharged into the reactor building dilutes hydrogen produced in the event of a severe accident (inerting concept). In the event of a severe accident, the pressuriser relief line also collects fluid released by the valves dedicated to depressurisation also connected to the pressuriser.

The pressuriser relief line contributes to the following reactor coolant system safety roles:

- Protection against category 3 and 4 overpressure transients via the pressuriser PSRVs.

- Capability to provide the feed and bleed function by opening the severe accident dedicated valves, in conjunction with safety injection actuation, to remove heat from the reactor core during RRC-A events.

- Capability to provide the reactor coolant system RCP [RCS] depressurisation by opening the pressuriser PSRVs to reach the long-term safe shutdown in PCC events.

- Depressurisation capacity in the event of a severe accident (RRC-B) by using the dedicated valves, to prevent core melt under pressure.

- Overpressure protection at cold shutdown, by opening the pressuriser PSRVs, the Residual Heat Removal System RRA [RHRS] being isolated.

The primary discharge system (pressuriser safety relief valves) is F1A classified because it is required to reach a controlled state under PCC conditions.

5.1.2. Functional role

In order to prevent reactor coolant discharge into the reactor building during pressure safety relief valve testing, the PRT collects, condenses and cools the entire volume of steam discharged during such tests.

The PRT is also used to collect potential leaks from the pressure relief valves during reactor normal operation.

The pressuriser relief line supports are designed to sustain transient loads induced by flow of sub-cooled water from the water seal and/or steam following the opening of the PSRVs.

5.2. APPLICABLE CRITERIA, HYPOTHESES AND CHARACTERISTICS

5.2.1. Applicable criteria

The pressuriser relief line design criteria are detailed in Sub-chapter 3.2.

5.2.2. Hypotheses and characteristics for the pressuriser relief line

5.2.2.1. Pressuriser relief valve lines

The design basis for reactor coolant system overpressure protection sizing is given in Sub-chapter 5.2.

Each of the pressuriser PSRV lines has a flow capacity of 300 t/hour of saturated steam at 176 bar abs [Ref-1].

5.2.2.2. Severe accident dedicated lines

The total capacity of the dedicated line used to depressurise the reactor coolant system during RRC-B sequences is 900 t/hour of saturated steam at 176 bar [Ref-1].

5.2.2.3. Pressuriser relief tank

The design of the PRT is based on the requirement to condense and cool the volume of steam discharged during pressuriser relief valves tests without exceeding a final given operating temperature/pressure in the PRT.

The pressure and temperature reached during testing are below the PRT design conditions.

The tank design data are given in Section 5.4.5 - Table 1.

The minimum volume of water in the PRT is determined by the energy content of the steam to be condensed and cooled by the water between its initial and final operating temperatures.

The minimum volume of gas (nitrogen) in the PRT is determined by the variation of water volume during the transient, assuming an initial gas temperature and the expected final operating pressure.

If the expected final operating pressure/temperature values in the PRT are exceeded, fluid is discharged to the reactor building following rupture of bursting disks installed in the outlet nozzle. This discharge only occurs during category 3 or 4 over-pressure accidents and RRC events, the disks being designed for the pressure increase due to pressuriser relief valve testing. The PRT collects any fluid that may leak from the pressuriser relief valves. There is no discharge towards the PRT during category 2 overpressure events except that due to pressuriser relief valve tests.

During normal operation, the PRT collects any gas, mainly hydrogen, leaking from pressuriser relief valves. Gas collected is purged to the gaseous waste treatment system.

5.3. EXPLANATION OF FUNCTIONAL DIAGRAMS, FUNCTIONAL CONNECTIONS AND IMPORTANT EQUIPMENT CHARACTERISTICS

The functional diagram for the pressuriser relief line is given in Sub-chapter 5.1 - Figure 2.

5.3.1. Pressuriser safety relief valves

RCP [RCS] overpressure protection is ensured by three lines of protection.

Each line is connected to the pressuriser upper spherical head through a specific nozzle.

The Pressuriser Safety Relief Valves (PSRVs) are described in section 7 of this sub-chapter.

5.3.2. Depressurisation during a severe accident and in RRC-A accidents (feed and bleed)

This is provided by two lines connected to the pressuriser through the same dedicated nozzle. The isolating principle for these lines is described in section 8 of this sub-chapter.

5.3.3. Discharge pipework

All the above valves (PSRVs and dedicated severe accident valves) discharge together into a common header routing the water, steam or steam/water mixture towards the PRT.

Rupture disks are installed in outlet nozzles of the PRT. The disks isolate pressuriser valves from the reactor building during normal reactor operation.

Five connections for the three RCS [RCP] overpressure lines of protection and the two severe accident lines from the gaseous waste treatment system, just-down stream of each PSRV and of each severe accident dedicated valve, are used to inject nitrogen into the discharge pipework.

5.3.4. Pressuriser relief tank

The Pressuriser Relief Tank design data are given in Section 5.4.5 - Table 1. The tank is connected to the discharge header through a dedicated pipe, upstream of the rupture disks. The pipe terminates inside the PRT. Pipe geometry inside the tank is designed to accomplish steam condensation during pressuriser safety valve testing.

The PRT is provided with the following connections:

- Gaseous Waste Processing System connection to remove gases from PRT gaseous phase.
- Drain line to the RPE [NVDS] (vents and drains system) for cooling or draining the tank.
- Return line from the RPE [NVDS] for cooling of the tank.
- Filling line from the demineralised water distribution system to adjust water volume in the tank.
- Vacuum breaker line connecting the PRT gaseous phase to the discharge header.

The return line from the RPE [NVDS] and the filling line from the demineralised water distribution system join together in a spray header located inside the PRT.

5.4. DESCRIPTION

5.4.1. Normal reactor operation

The PSRVs and the severe accident dedicated valves are normally closed.

The rupture disks isolate the discharge header from the reactor building.

The PRT is partially filled with water.

Purging of the discharge header and the PRT gaseous phase is carried out by:

- Nitrogen injection from the gaseous waste processing system TEG [GWPS] connections downstream of the pressuriser relief valves.

- Nitrogen removal to the TEG [GWPS] connection on the PRT (via nitrogen sparging in the PRT water content).

5.4.2. PSRV periodic testing

Steam released by the PSRVs is condensed and cooled in the PRT. The PRT and the rupture disks are sized to permit testing of all valves without causing their rupture.

PSRV testing is carried out with the pressuriser at 40 bar abs, for a duration of approximately 120 seconds [Ref-1]. This takes into account the possibility that the PSRV may remain open for the duration of the test (120 seconds).

After each PSRV opening for testing, the PRT vacuum breaker line prevents the formation of a vacuum in the discharge pipe due to steam condensation.

5.4.3. Pressuriser relief tank cooling

Once pressuriser relief valve testing has been completed, the PRT is cooled by a closed loop using the RPE [NVDS] pumps and the heat exchanger.

5.4.4. Pressuriser discharge transient

The discharge fluid is first routed to the PRT until the pressure in the system reaches the rupture disks set pressure.

After disks rupture, reactor coolant is discharged inside the reactor building and the PRT has no more function, except as pressure boundary to prevent reactor coolant from discharging directly into the containment.

In the event of repeated pressuriser safety valve openings, the vacuum breaker line on the discharge header prevents a vacuum forming in the pipe following steam condensation after each safety valve closure.

5.5. BRIEF DESCRIPTION OF VALVE ACTUATIONS

5.5.1. Opening of the PSRVs (category 3 or 4 over-pressure transient conditions or PCC events)

There are three different opening modes for each PSRV:

- During category 3 or 4 RCP [RCS] overpressure transients in hot conditions, each PSRV is automatically actuated when pressure reaches the set point of its pilot valve.

- During category 3 or 4 RCP [RCS] overpressure transients in cold conditions, the PSRVs are opened by a dedicated instrumentation and control system including:

    - Set point determination according to reactor parameters.

    - PSRV opening by electrical control activation of the pilot valve if the set point is reached.

5.5.2. Opening of the Severe Accident dedicated valves in RRC-A accidents (Feed and Bleed) and in severe accidents (RRC-B)

These specific valves are opened remotely by the operator from the main control room.

5.6. COMPLIANCE WITH CRITERIA, CLASSIFICATION, FAILURE

5.6.1. Classification

Reactor coolant system RCP [RCS] safety classification, including pressuriser relief line, is described in Sub-chapter 3.2.

5.6.2. Single failure criterion

The single failure criterion applies to the F1 section of the circuit.

It includes:

- The PSRVs.

- The Severe Accident Valves in normal conditions for the RCP [RCS] for isolation function.

- The severe accident relief line in the event of a severe accident for the RCP [RCS] for opening function.

5.6.3. Protection against internal hazards

The pressuriser relief line is protected against possible damage caused by hazards within the plant by:

- Location of the system inside the reactor building.
- Equipment (pressuriser PSRVs and the severe accident valves) installation in bunkers.

5.6.4. Protection against external hazards

The pressuriser relief line is protected against external hazards by:

- Location of the system inside the reactor building.
- Seismic classification of the systems which covers protection against other external hazards such as aircraft crash or external explosions.

5.7. SPECIFIC TESTING PROVISIONS

The PRT and associated rupture disks, are designed to allow testing of the PSRVs at low RCP [RCS] pressure (approximately 40 bar abs [Ref-1]) without discharge to the reactor building.

These tests include verification of the set point and operability of the PSRVs.

SECTION 5.4.5 - TABLE 1

Preliminary Characteristics of the Pressuriser Relief Tank [Ref-1]

Design pressure, bar abs	25
Pressure, bar abs	0.8 [1]
Design temperature, °C	224
Initial water temperature, °C	55 [1]
Volume, m^3	40
Initial water volume, m^3 (minimum)	31 [1]
Initial gas volume, m^3 (maximum)	9 [1]
Set pressure defined for the rupture disk, bar g	19

[1] For initial operating conditions

6. VALVES

This section deals with the first and second isolation devices of the reactor coolant pressure boundary (CPP [RCPB]). The largest diameter valves (Nominal Diameter (ND) above 50 mm) are located in the Safety Injection System / Residual Heat Removal System (RIS/RRA [SIS/RHRS]) and Chemical and Volume Control System (RCV [CVCS]) systems.

6.1. DESIGN [REF-1]

All valves that ensure the integrity of the CPP [RCPB] are classified M1 and subject to the nuclear construction code RCC-M (see Sub-chapter 3.8).

The design requirements as well as the required operating conditions are detailed in the descriptions of the corresponding systems.

The manufacture, supply and testing of the parts forming the pressure boundary must be carried out in accordance with the provisions of the RCC-M.

These valves carry primary coolant. Their bodies are therefore made of stainless steel and their internal components, mainly those in contact with the coolant, are selected from corrosion resistant materials. The use of cobalt-based hard-facings should be limited as far as possible.

The valves are connected to the pipework by butt-welding.

Valves with ND ≤ 150 mm providing an isolation function and carrying radioactive fluid are globe valves fitted with stem bellow seals in order to provide total external leak-tightness [Ref-2].

Valves with ND ≤ 150 mm not fulfilling an isolation function nor located outside containment and carrying radioactive fluid are globe valves either fitted with bellows seals or with packings and leak-off recovery [Ref-2]. Bellows are only used for globe valves.

For valves carrying radioactive fluid and with larger nominal diameter (ND > 150 mm), gate valves with packings are used [Ref-2]. In general, leakage recovery pipes are routed to the nearest header system.

These selection rules apply to "standard" valves, as defined in valve catalogues. However, valves defined as "specific" can be used when usual design/manufacture does not provide the required function and/or when the choice of the valve does not comply with the selection rules.

The systems considered are mainly fitted with angle or straight globe valves, angle or straight check valves, or gate valves.

Angle check valves are piston check valves which can be equipped, if necessary, with a lockup device.

All other check valves are of the swing type for sizes larger than ND 50 mm. The swing check valves do not have body penetrations for the hinge. Maintenance performed through access openings in the valve bodies.

6.2. INSTALLATION

The first and second isolation devices are located within the reactor building. They are located in areas such that their temperature during normal operation is as low as possible.

6.3. OPERATING CONDITIONS

The capability of the valves to function correctly in normal and accident conditions has been demonstrated by analysis which takes account of the system requirements.

The chemical composition of the reactor coolant is given in valve equipment specifications to ensure materials compatible with the reactor coolant are selected. The chemical composition of the reactor coolant is periodically analysed to verify that it conforms to the specifications.

The design requirements and installation principles discussed above, coupled with the devices previously described to reduce leakage to a minimum, enable the valves to function at all times during plant operation.

6.4. QUALIFICATION

Analytical studies and testing are carried out to ensure that active valves will operate under required conditions.

All valves are subjected to hydrostatic tests, leak tests to verify the leak-tightness of the valves to the atmosphere, leak tests across the valve seats, operability tests and inspections as required by the equipment specifications.

7. PRESSURISER SAFETY RELIEF VALVES (PSRV)

7.1. DESIGN

This section describes how the PSRVs are connected to the pressuriser on independent, discrete lines.

The design of the PSRVs is directly linked to the requirements concerning protection of the primary circuit in the event of over-pressurisation.

7.2. ARRANGEMENT

The PSRVs are pilot-operated valves, fitted with “hot” pilots and position sensors. The operating conditions for the control devices are the same as those of the pressuriser.

A “hot” pilot is a pilot fixed rigidly on the body of the valve. The connections between the pilots and the main valve control volume are made by internal passages within the body of the valve.

Three lines of protection are installed between the dedicated nozzles on the pressuriser (one per line) and the common relief line discharging into the pressuriser relief tank. Each line comprises a PSRV, manufactured by SEMPELL, directly welded onto each nozzle on the pressuriser dome (see Section 5.4.7 - Figure 1). A scoop is welded inside the pressuriser, upstream of each inlet line, to ensure the presence of a hot water seal upstream of the valve to avoid any leak of hydrogen via the valve seat. The valves are connected downstream to the pressuriser relief lines by flanges.

The SEMPELL valve is of the hermetically sealed type. It opens by depressurisation of the control volume above the disc, under the action of the pilots. Each valve is fitted with two parallel spring-loaded pilots (with a pilot isolated in normal operating conditions) and with two solenoid pilots mounted in series (see Section 5.4.7 - Figure 2).

7.3. SAFETY ROLE

The safety functions of each line of protection are as follows:

Protection of the RCP [RCS] against overpressure in Category 3 or 4 conditions: PSRVs work in automatic mode via spring-loaded pilots.

Protection of the reactor vessel against overpressure events in shutdown conditions: PSRVs work in remote control mode via two solenoid pilots mounted in series.

Long-term depressurisation of the RCP [RCS]: PSRVs are controlled by their solenoid valves.

Temperature measurements are provided downstream of the PSRVs pilot discharge ports. An increase in the temperature of a relief line is an indication of a leaking pilot.

A temperature measurement is provided upstream of each PSRV. An increase in temperature indicates the loss of the water seal.

The position of each PSRV is indicated in the control room.

The design characteristics of the PSRVs are given in Section 5.4.7 – Table 1.

7.4. OPERATING CONDITIONS

To protect the RCP [RCS] against overpressure events at power, the opening pressures of the main safety valves on each line of protection are staggered. The first line of protection is set at the lowest pressure to limit the number of valves which will open and to minimise the discharge flow during the majority of the transients expected.

To avoid PSRV pilots or discs fluttering during opening or closing, the sensing lines of the pilots are directly connected to the pressuriser steam phase.

To protect the RCP [RCS] against overpressure events during the reactor in shutdown condition at least two safety valves are actuated (to accommodate a single failure). Their automatic opening is triggered by a specific I&C system which activates the solenoid pilots. The opening and closing times (stroke and dead time) of the safety valves in cold conditions will be compatible with process requirements.

For the long-term depressurisation of the RCP [RCS], the PSRVs are operated remotely from the control room using the solenoid pilots.

7.5. QUALIFICATION

The SEMPELL safety valve has already been tested (flow rate, response time) in accident conditions (discharge of steam, saturated and sub-cooled water) on a full flow test rig. Cycling tests (endurance) of the pilot have also been carried out.

The SEMPELL PSRV is installed on a large number of German power stations albeit with a different spring pilot than that used on the EPR PSRVs. However, the spring loaded pilot used on EPR PSRVs is installed on the Goesgen power station in Switzerland.

The correct operation of the main valve and its pilots has been demonstrated during the qualification tests with one main valve and its spring loaded pilot at full pressure and temperature and reduced flow on the AREVA-NP test facility in Erlangen.

This will be confirmed, for each PSRV and both their spring-loaded and solenoid pilots, under the same operating conditions, by the SEMPELL test facility during Final Acceptance Tests. Furthermore the correct operation of the whole system will then be checked on the plant under full flow at nominal reactor pressure (155 bar) during plant Hot Functional Startup Tests, and also during periodic tests at 40 bar after each refuelling.

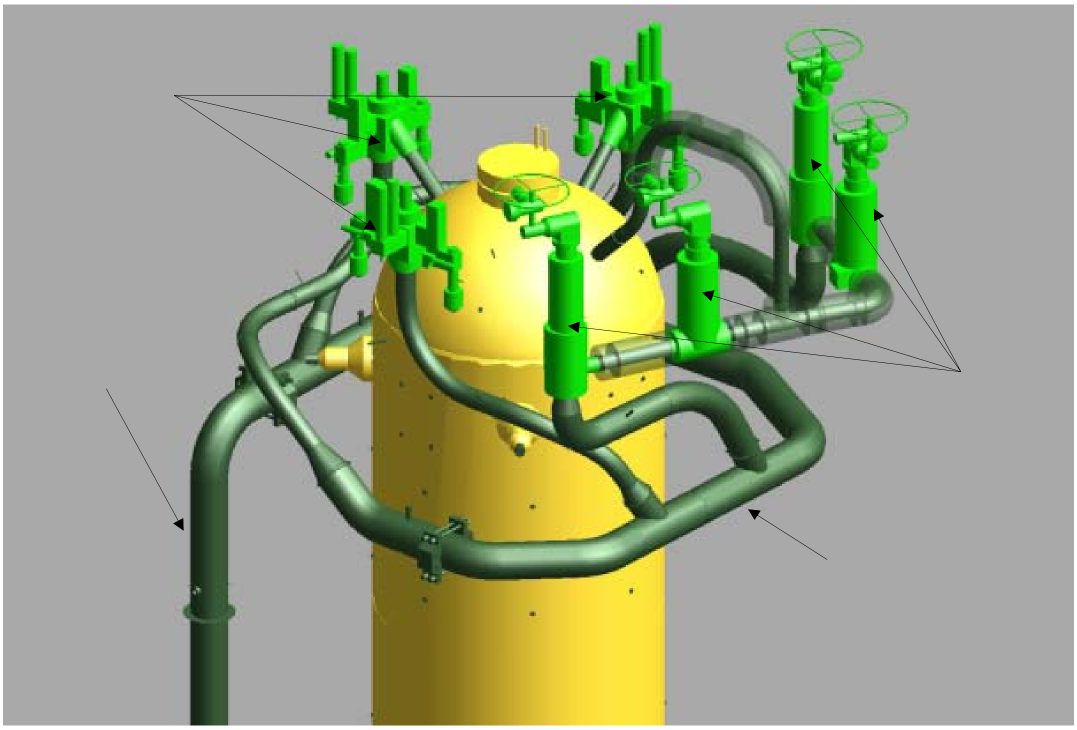
SECTION 5.4.7- TABLE 1

Design Characteristics [Ref-1]

DESIGN CRITERIA		
Number of valves		3
Design pressure, bar abs		176
Design temperature °C		362
P (required flow rate, kg/hour)		300,000 Saturated steam
Max dead time opening, s		0.5
Max Stroke time opening, s		0.1
Downstream pressure bar abs	Min	1.2
	Max, during discharge	50
Ambient conditions		As defined by the RCC-E chapter D-2200
Set pressure, bar abs	1st PSRV:	175 ± 1.5
	2nd PSRV:	178 ± 1.5
	3rd PSRV:	181 ± 1.5

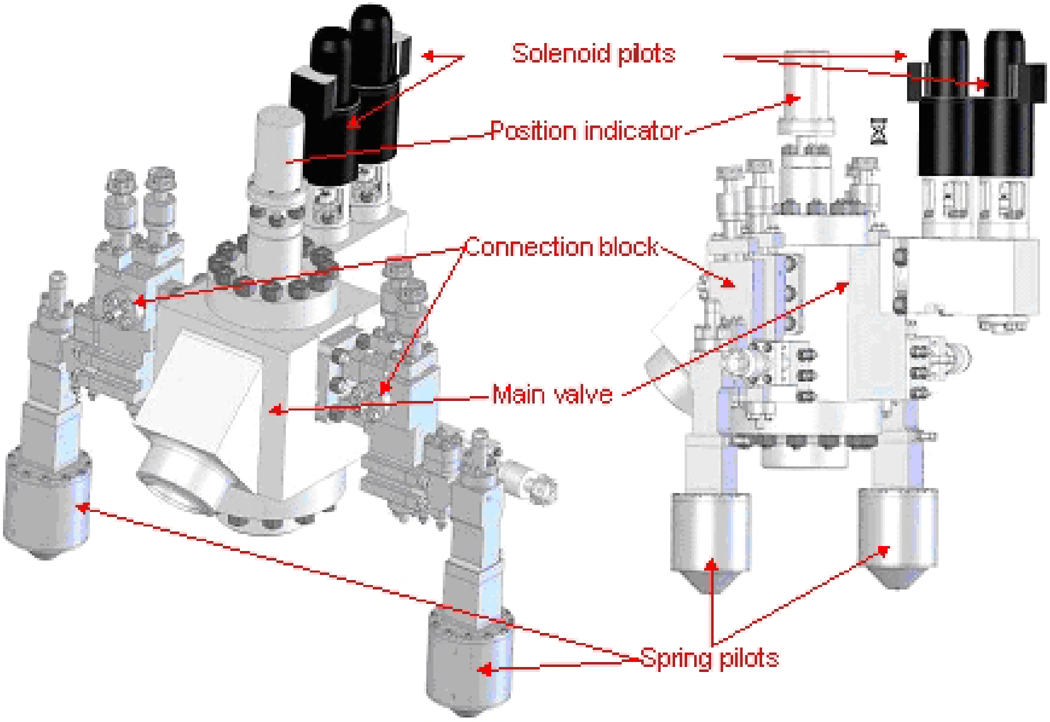
SECTION 5.4.7 - FIGURE 1

Pressuriser with Discharge Equipment [Ref-1]



SECTION 5.4.7- FIGURE 2

Components of Pressuriser Safety Relief Valve [Ref-1]



8. SEVERE ACCIDENT VALVES

8.1. DESIGN

The operation of the dedicated valves is governed by depressurisation requirements to prevent core melting under pressure in the event of severe accidents.

Dedicated valves are installed in 2 lines connected to a common specific nozzle on the dome of the pressuriser. Each line contains one gate and one globe valve in series.

8.2. INSTALLATION

The general layout is shown in Section 5.4.7 - Figure 1.

A water seal upstream of the valves prevents the valves becoming hot during normal plant operation.

Each dedicated line is connected to the main discharge line between the pressuriser safety relief valves and the pressure relief tank.

All valves are closed under normal conditions.

8.3. OPERATING CONDITIONS

The valves are designed to discharge a flow rate of 900 kg/hour of saturated steam at a pressure of 176 bar abs [Ref-1] [Ref-2].

The maximum admissible temperature at the opening of the valves is 600°C [Ref-1] [Ref-2] relating to the depressurisation of the RCP [RCS]. However, opening of the valve is expected to occur at a temperature not greater than the normal pressuriser design temperature of 362°C [Ref-1] [Ref-2].

The maximum temperature during the transient period remains lower than 1100°C [Ref-1] [Ref-2]. In spite of these conditions, the valves must remain open until there is a minimum pressure difference of zero bar.

The valves are supplied by two different electrical trains. If a complete loss of AC supply occurs electrical control and power to the valves will be provided by batteries.

8.4. QUALIFICATION

The qualification will be done by analysis and extrapolation from similar designs which have undergone qualification tests.

9. PRIMARY COMPONENT SUPPORTS

Primary component support design is based on the design principles of the N4 unit. Important changes result from the implementation of measures originating from the EPR generic basic design and related to the Break Preclusion principle applied for the FA3 EPR, and considered for the UK EPR as conservative measures (see Sub-chapter 5.2).

9.1. DESCRIPTION

The primary component supports are fabricated from steel sections.

Plate and shell type structures are used for the reactor pressure vessel support. All other supports are of the linear type (struts, support columns, snubbers).

Attachments to the supported equipment are non-integral except for steam generator upper lateral support and for pressuriser lower vertical supports. They are bolted to or bear against the components. The supports are fixed to the concrete structures by anchor bolts or embedded tie rods.

Primary component support structure designs provide for adequate adjustment to enable correct mounting, alignment and construction, allowing for manufacturing and construction tolerances. This adjustment is achieved by grouting at the support/concrete interface and by shimming at the support/equipment interface.

9.1.1. Reactor pressure vessel support

The reactor pressure vessel support is shown in Section 5.4.9 - Figure 1.

The vessel is supported through its pads on support blocks incorporated into a support ring.

The support ring is a welded structure made up of a lower plate resting on the edge of the reactor pit, a cylindrical shell and an upper plate made up of eight support blocks. The various parts are welded together.

The eight support blocks each contain a recess, machined parallel to the nozzle axes, to allow free radial movement of the vessel due to thermal expansion and pressure.

The support ring is held in place against horizontal loads (Design Earthquake, LOCA) by means of eight vertical keys incorporated in the lower plate and fitting into grooves in the concrete.

Openings are provided in the keys and stiffener plates located above them to allow the passage of the guide-tube for the measuring chamber for external core instrumentation.

In the event of a severe accident, the overall rigidity of the RCP [RCS] loops allows the vertical loads acting on the reactor vessel to be taken by the primary loops. Consequently, it is not necessary to have a specific device to restrain the support ring in the vertical direction.

The support ring is cooled by the reactor pit ventilation system.

A restraint is installed on top of the control rod drive mechanisms in order to limit their movements in the event of dynamic seismic loads.

9.1.2. Steam generator supports

The steam generator supports are shown in Section 5.4.9 - Figure 2.

Vertical supports

Four articulated columns support the steam generator vertically. Each column is made up of a tube with ball joint clevises at each extremity. The female clevises are screwed or welded to each end of the tube.

The upper part of each column is fixed by screws to the lower head of the steam generator. The columns are spaced at 90°.

The lower parts of the columns (male clevis) are fixed to the heavy-duty floor via four tie rods. After final mounting, the tie rods are pre-tensioned using hydraulic cylinders to a value that will prevent the columns lifting should accident loads be imposed on the supports.

The design of clevises is able to accommodate the thermal expansion of the loop and of the base of the steam generator. This allows free movement of the steam generator during primary loop temperature changes.

The columns are vertical in normal operating conditions. This is achieved by the position of the lower clevises.

The steam generator height can be adjusted on site by two methods:

- base adjustment is made using metal plates or by injecting grout between the base and the floor,

- accurate adjustment is made using shims located between the upper male clevises and the lower head of the steam generator.

Lower lateral supports

The lower lateral supports guide the steam generator and limit its movement in accident conditions.

Two metal stops provide lateral guidance on the lateral wall of the building. These are mounted on the bunker walls. They guide the steam generator, allowing it to move in a radial direction from the centre of the reactor vessel during thermal expansion of the reactor primary system. The design is such that it gives easy access to the welds of the tube sheet for in-service inspection. A third metal stop is located on the hot leg centre-line direction; this is designed to limit the steam generator movement and to stabilise it in the presence of the conventional “2pA static load” (2 = dynamic load factor; p = operating pressure; A = surface of the break) (see Sub-chapter 5.2).

All three stops are constructed from steel plates and use screw-jack systems. One side of the plate is fixed to the bunker wall by embedded anchor bolts. The other side has a screw-jack system to adjust the gap between the stops and the flats machined on the lower head of the steam generator.

The stops are cooled by forced air convection from the steam generator bunker ventilation system.

Upper lateral supports

The upper lateral supports are installed at the operating floor level.

Two sway struts allow the steam generator to move in a radial direction from the centre of the reactor vessel during thermal expansion of the reactor primary system.

Two snubbers, placed in the axis of the reactor vessel, limit sudden movements caused by an earthquake or a pipeline break. They allow slow movements caused by the thermal expansion of the RCP [RCS] loops. The sway struts and the snubbers have ball clevises to avoid stresses from off centre loads.

The sway struts and snubbers are anchored into the structure of the building. These four lateral supports are fixed directly on the outer surface of the steam generator at the level of its centre of gravity.

9.1.3. Reactor coolant pump supports

The reactor coolant pump supports are shown in Section 5.4.9 - Figure 3.

Vertical supports

Three vertical articulated columns support the reactor coolant pump. These columns are of the same design as those for the steam generator (apart from their length). They have an upper cylindrical part linked by a stud to the reactor coolant pump casing lug.

The three columns are located around the pump casing support. This location takes the pump outlet nozzle direction into account.

They are anchored by tie rods to the heavy-duty floor.

Lateral supports

Two hydraulic snubbers, located at the centre of gravity of the pump, give the pump stability in accident conditions.

These snubbers are attached at one end to the lower part of the motor frame and at the other end are anchored to the building structure by tie rods.

These snubbers are identical in design to those used for limiting sudden movements of the steam generators.

A stop, with a gap, is located on the axis of the cold leg. This is designed to limit the movement of the pump and to ensure its stability in the presence of the conventional “2pA static load” (2 = dynamic load factor; p = operating pressure; A = surface of the break) (see Sub-chapter 5.2).

9.1.4. Pressuriser supports

The pressuriser supports are shown in Section 5.4.9 - Figure 4.

Lower vertical supports

The pressuriser is supported by three brackets welded to its lower shell resting on structures fixed to the floor by pre-tensioned tie rods. The design allows free radial movement caused by pressure and thermal expansion. However, the central axis of the pressuriser remains fixed in relation to the bunker.

Upper lateral supports

A lateral support device is fitted to the pressuriser. This provides stability in accident conditions. The lateral support is made up of eight equally-spaced stops located adjacent to the centre of gravity of the pressuriser. This allows free axial expansion.

9.1.5. Surge line supports

The pressuriser surge line is supported by spring devices. These absorb normal operational forces due to the weight of the surge line (pipework, fluid content and thermal insulation).

In the event of an earthquake, the surge line is retained by appropriate supports.

9.2. OPERATING CONDITIONS AND DESIGN LOAD CASE

The load conditions defined for the study of the supports are as follows (see Sub-chapter 3.4):

Weight

The weight of reactor coolant system (RCP [RCS]) components is transmitted to the building structure by vertical supports and reactor pressure vessel support.

Thermal loading

Thermal loads correspond to normal operating and hot shutdown conditions.

Seismic loads

Seismic loads are evaluated by basing them on the floor response spectrum. The methodology is described in Sub-chapter 3.1.

Breaks

Guillotine breaks are postulated in the lines connected to the reactor coolant system RCP [RCS] main lines or connected to the main steam lines (part of the CSP [MSS]). HIC requirements do not apply to these connected lines.

2pA loading

The 2A-LOCA event is discounted from the design, as the UK EPR Main Coolant Lines are HIC. However, as a conservative measure, a static load equivalent to a rupture of the RCP [RCS] main lines, or a rupture of main steam lines, known as “2pA”, is considered to ensure the stability of the large components.

Severe accident

The resulting vertical stress applied to the reactor vessel is described in Sub-chapter 16.2. The support ring of the reactor vessel is not designed specifically for this event, and reactor vessel restraint is assumed to be only provided by the reactor coolant system pipework.

9.3. ACCESS AND INSPECTION REQUIREMENTS FOR PRIMARY COMPONENTS SUPPORTS

The primary components supports will not impede access for in-service inspection of the primary components and the reactor coolant system pipework.

All functional gaps between supports and components can be checked.

Inspection and maintenance, as necessary, is carried out during scheduled refuelling shutdowns.

9.4. REPAIR AND REPLACEMENT OF PRIMARY COMPONENT SUPPORTS AND PRIMARY COMPONENTS

The nominal design lifetime of the primary circuit supports is 60 years.

Primary circuit support replacement is possible, with the exception of reactor vessel support.

Primary circuit supports do not impede work on the primary circuit components, including component modifications and possible replacements.

9.5. SEISMIC DESIGN

Seismic analysis methods are described in Sub-chapters 3.1 and 3.4.

9.6. DESIGN RELATIVE TO PIPEWORK RUPTURES

Pipework ruptures analysis methods are described in Sub-chapter 3.1.

9.7. CALCULATION OF STATIC AND DYNAMIC LOADS APPLIED ON COMPONENTS, COMPONENTS SUPPORTS AND CONCRETE

(See Sub-chapter 3.4)

9.8. DESIGN RULES FOR SUPPORTS

Primary component supports design allows virtually unrestrained thermal movements of the RCP [RCS] loops and components during plant operation. They provide restraint to the loops and components during accident conditions (earthquake, auxiliary and secondary pipe break, etc.).

The loading combinations and design stress limits are described in Sub-chapter 3.4.

Primary circuit support design conforms to the specification RCC-M Section I, Subsection H. (see Sub-chapter 3.8).

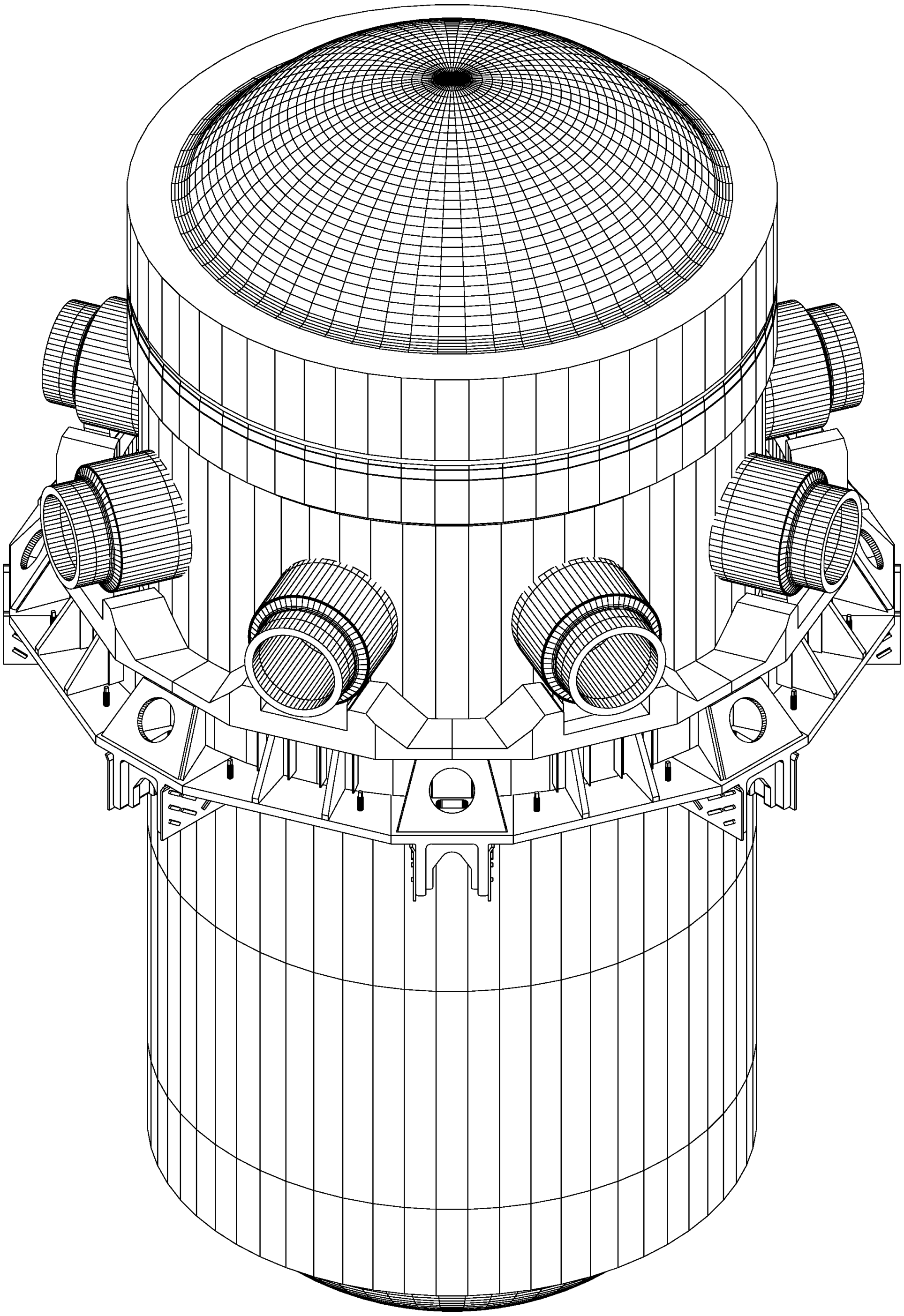
The design adequacy and structural integrity of the reactor coolant loop and of the primary component supports is ensured by detailed evaluation. This detailed evaluation is made by comparing analytical results with established criteria for acceptability. Structural analyses are performed to demonstrate design adequacy for safety and reliability of the plant in the event of seismic disturbance or loss of coolant accident conditions. Loads that the system is expected to encounter often during its lifetime (thermal, weight, pressure) are applied and stresses are compared to allowable values as described in Sub-chapter 3.4.

9.9. MATERIALS AND MANUFACTURE

Material acceptance and testing, manufacturing controls, welder qualification, welding procedures, weld inspection and standards are specified in accordance with RCC-M Section I, Subsection H.

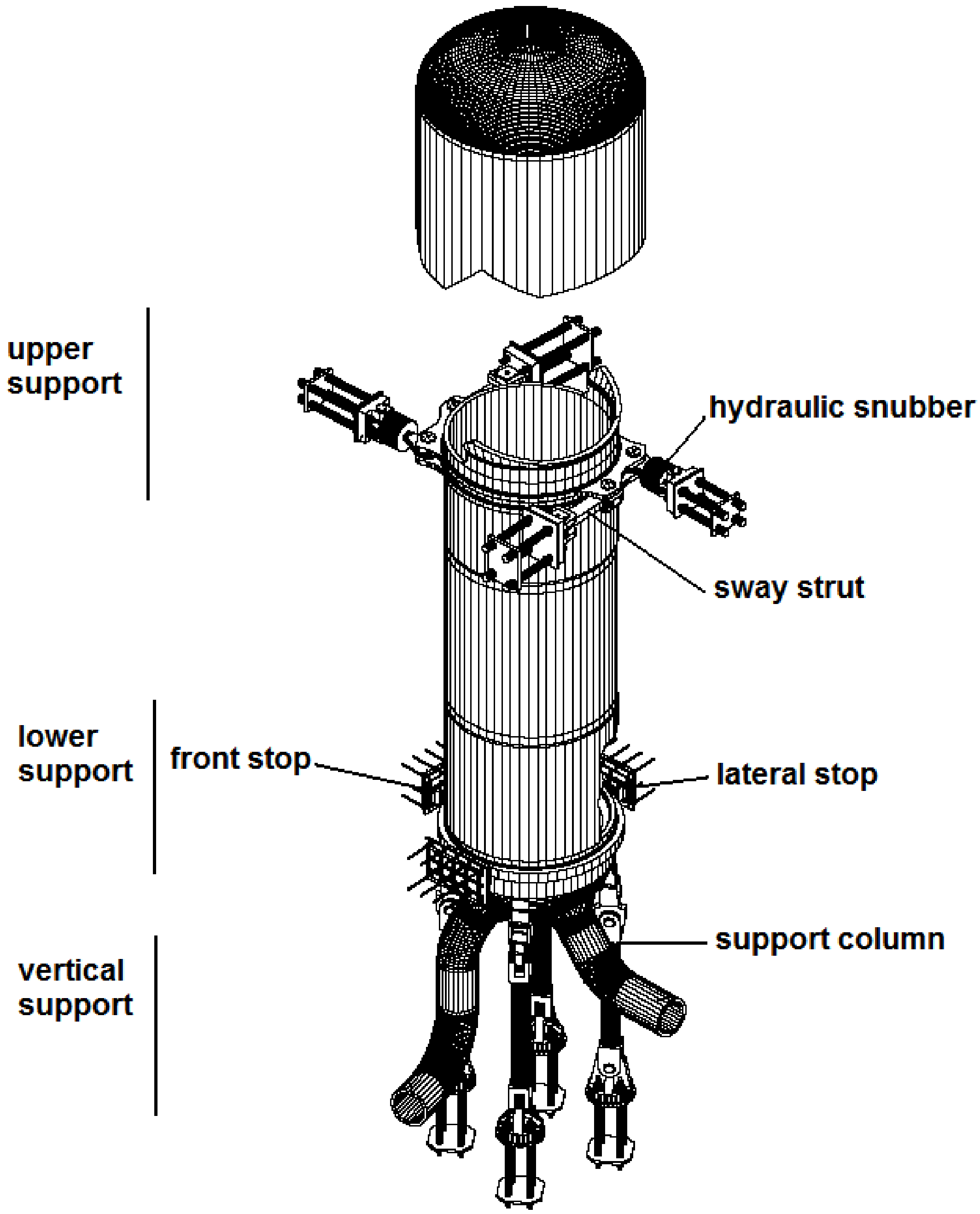
SECTION 5.4.9 - FIGURE 1

Reactor Vessel Support [Ref-1]



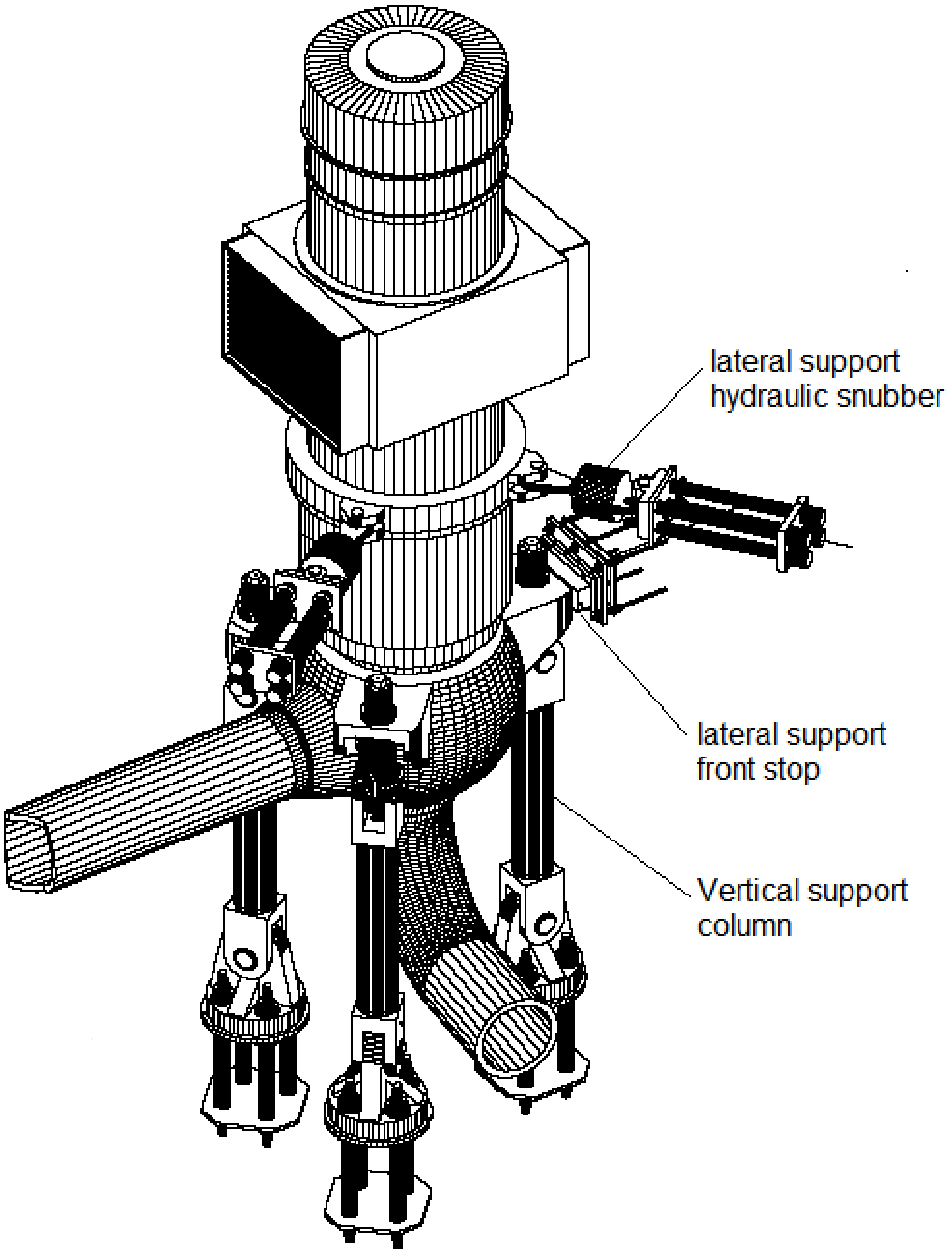
SECTION 5.4.9 - FIGURE 2

Steam Generator Supports [Ref-2]



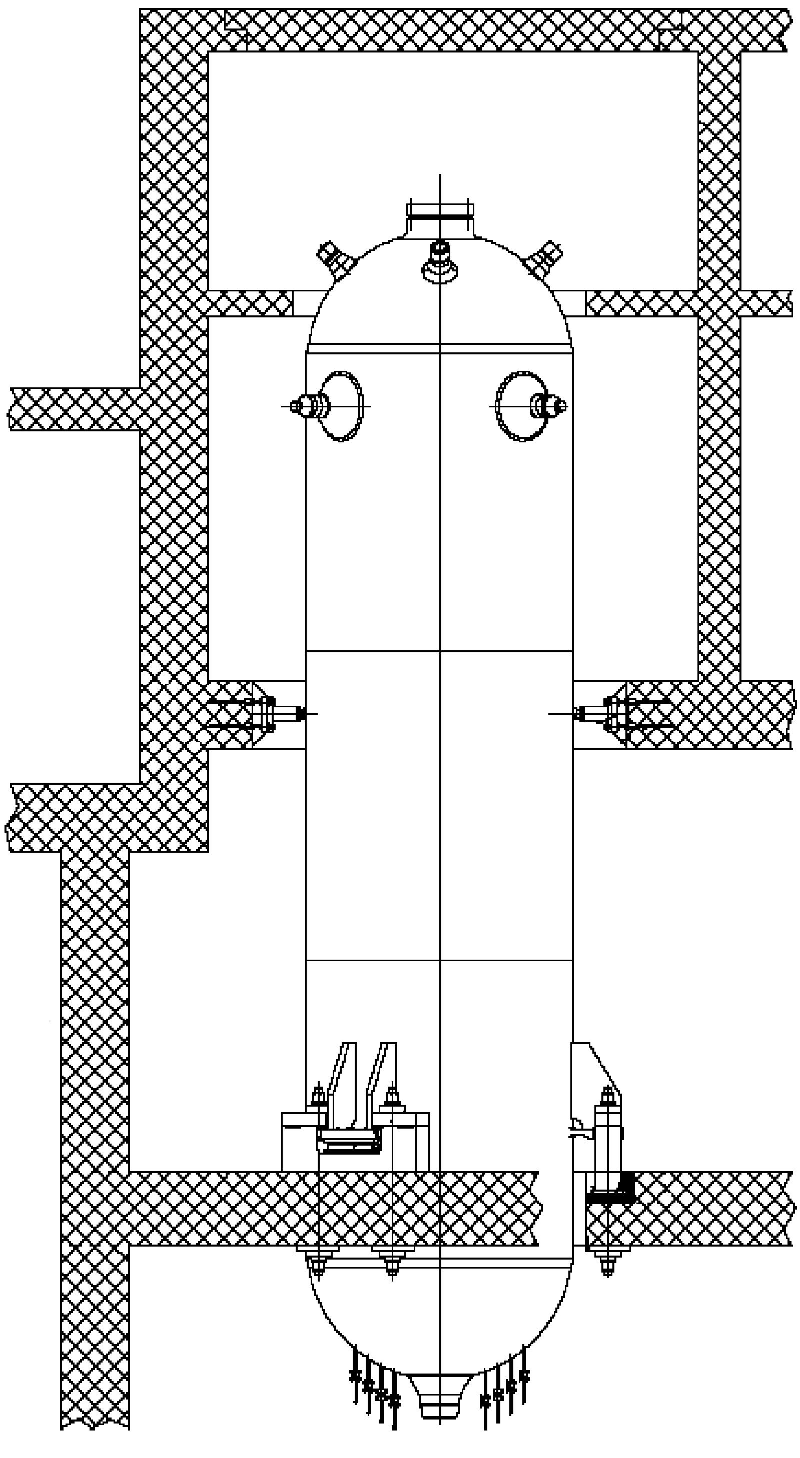
SECTION 5.4.9 - FIGURE 3

Primary Pump Supports [Ref-3]



SECTION 5.4.9 - FIGURE 4

Pressuriser Supports [Ref-4]



SUB-CHAPTER 5.4 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

1. REACTOR COOLANT PUMPS

1.2. DESIGN BASES

[Ref-1] EPR FA3 – Equipment specification for the RCP (Reactor Coolant Pumps). NEEG-F DC 4 Revision D. AREVA NP. 2008. (E)

[Ref-2] UK EPR – Reactor Coolant Pump Characteristics. PEEO-F DC 167 Revision A. AREVA. August 2012. (E)

PEEO-F DC 167 Revision A is the English translation of JSPM JSR 6GA20735 Revision B.

1.6. FAST FRACTURE ANALYSIS

1.6.1. Reactor Coolant Pump casing

[Ref-1] Demonstration of integrity of High Integrity Components against fast fracture. Fracture Mechanic Analyses – Non Destructive Testing – Fracture Toughness NEER-F 10.2070 Revision D. AREVA. August 2012. (E)

[Ref-2] Fracture toughness properties of repair welds in cast pump casing. PEEM-F 11.0567 Revision A. AREVA. March 2011. (E)

[Ref-3] UK EPR Technical proposition for the NDT examination of major repair welds of the primary pump casing. PEEM-F 10.2218 Revision B. AREVA. September 2011. (E)

1.6.2. Reactor Coolant Pump flywheel

[Ref-1] Demonstration of integrity of High Integrity Components against fast fracture. Fracture Mechanic Analyses – Non Destructive Testing – Fracture Toughness. NEER-F 10.2070 Revision D. AREVA. August 2012. (E)

[Ref-2] Non Destructive Tests performed on the Reactor Coolant Pump flywheel during manufacturing. PEEO-F 10.0715 Revision C. AREVA. December 2011. (E)

[Ref-3] Principles for the RCP Flywheel In-Service Inspection. ECEMA111847 Revision A. EDF. August 2011. (E)

[Ref-4] UK EPR Avoidance of fracture: Fracture mechanics and minimum toughness aspects – Material data. NEEM-F 10.0179 Revision E. AREVA. November 2010. (E)

SECTION 5.4.1 - TABLES 1 TO 3

[Ref-1] EPR FA3 – Equipment specification for the RCP (Reactor Coolant Pumps).
NEEG-F DC 4 Revision D. AREVA NP. 2008. (E)

SECTION 5.4.1 – FIGURES 1 AND 2

[Ref-1] Reactor Coolant Pumps EPR FA3 – Outline drawing.
JSPM JSR 1GA20230 Revision D. AREVA NP. 2008. (E)

[Ref-2] GMPP [RCP] EPR FA3 – Ensemble joint d'arbre et DEA.
[Reactor Coolant Pumps EPR FA3 - Shaft Sealing System]
JSPM JSR 1GA20341 Revision B. AREVA NP. 2008.

2. STEAM GENERATORS

2.1. DESCRIPTION

[Ref-1] C Hazelard. Générateur de vapeur 79/19TE - Plan d'ensemble - GV droit et gauche.
[Steam Generator 79/19TE – Layout – SG right and left].
NEEG-F DB 1200 Revision H. AREVA NP - FA3 project. November 2008.

2.1.1. General characteristics

[Ref-1] C Hazelard. Générateur de vapeur 79/19TE - Plan d'ensemble - GV droit et gauche.
[Steam generator 79/19TE - General view – Steam Generator right and left]
NEEG-F DB 1200 Revision H. AREVA NP - FA3 Project. November 2008.

[Ref-2] C Hazelard. Générateur de vapeur 79/19TE - Vue extérieure - GV droit et gauche.
[Steam generator 79/19TE - Exterior view – Steam Generator right and left]
NFPMG DB 1201 Revision G. AREVA NP - FA3 Project. November 2008.

[Ref-3] C Hazelard. Générateur de vapeur 79/19TE - Vue extérieure – Coupes et vues - GV droit et gauche.
[Steam generator 79/19TE - Exterior view – Steam Generator – cross-sections and layout - right and left].
NFPMG DB 1202 Revision H. AREVA NP - FA3 Project. November 2008.

2.1.2. Lower sub-assembly

[Ref-1] C Hazelard. Générateur de vapeur 79/19TE - Boîte à eau primaire – Ensemble et détails - GV droit.
[Steam generator 79/19TE – Primary water box – General and details – Steam Generator right].
NFPMG DB 1204 Revision H. AREVA NP - FA3 Project. November 2008.

[Ref-2] C Hazelard. Générateur de vapeur 79/19TE - Boîte à eau primaire – Ensemble et détails - GV gauche.
[Steam generator 79/19TE – Primary water box – General and details – Steam Generator left].
NFPMG DB 1205 Revision H. AREVA NP - FA3 Project. November 2008.

[Ref-3] C Hazelard. Générateur de vapeur 79/19TE - Enceinte secondaire – Vue générale - GV droit.
[Steam generator 79/19TE – Secondary side casing – General view – Steam Generator right].
NFPMG DB 1208 Revision G. AREVA NP - FA3 Project. November 2008.

[Ref-4] C Hazelard. Générateur de vapeur 79/19TE - Enceinte secondaire – Vue générale - GV gauche.
[Steam generator 79/19TE – Secondary side casing – General view – Steam Generator left].
NFPMG DB 1209 Revision G. AREVA NP - FA3 Project. November 2008.

[Ref-5] D Venturi. Générateur de vapeur 79/19TE – Faisceau tubulaire et nomenclature - GV droit et gauche.
[Steam generator 79/19TE – Tube bundle and nomenclature - Steam Generator right and left].
NEEG-F DB 1212 Revision D. AREVA NP - FA3 Project. June 2007.

[Ref-6] D Venturi. Générateur de vapeur [Steam generator] 79/19TE – Lower internals assembly – Right and left hand SGs.
NEEG-F DB 1213 Revision E. AREVA NP - FA3 Project. April 2008.

2.1.3. Upper sub-assembly

[Ref-1] C Hazelard. Générateur de vapeur [Steam generator] 79/19TE – Upper internals assembly – Right and left hand SGs.
NEEG-F DB 1225 Revision G. AREVA NP - FA3 Project. November 2008.

[Ref-2] C Hazelard. Générateur de vapeur [Steam generator] 79/19TE – Upper internals – Sections - Right and left hand SGs.
NEEG-F DB 1226 Revision F. AREVA NP - FA3 Project. November 2008.

2.3. DESIGN PRINCIPLES AND OBJECTIVES

2.3.1. Functional requirements

[Ref-1] T Millequant. Functional requirements on the Steam Generator – NFPSR DC 1144 Revision A. AREVA NP – FA3 Project. May 2006. (E)

2.3.2. Main properties selected

[Ref-1] T Millequant. Functional requirements on the Steam Generator – NFPSR DC 1144 Revision A. AREVA NP – FA3 Project. May 2006. (E)

2.4. THERMO-HYDRAULIC DESIGN

2.4.1. Operating parameters

[Ref-1] A Nicoli. EPR 79/19 TE – Steam Generator operating parameters.
NEEG-F DC 5 Revision F. AREVA. January 2009. (E)

2.4.2. Thermodynamic criteria

[Ref-1] A Nicoli. EPR 79/19 TE – Steam Generator operating parameters.
NEEG-F DC 5 Revision F. AREVA. January 2009. (E)

2.4.3. Thermal design

[Ref-1] S Francke. EPR – Steam Generator 79/19 TE – Primary and Secondary Sizing.
NEEG-F DC 35 Revision D. AREVA NP- FA3 project. July 2007. (E)

[Ref-2] Steam Generator Cleaning – Lancing (SNL) System Specification.
ECEMA0051005 Revision A1. EDF. December 2009. (E)

[Ref-3] Dossier de Système Elémentaire – Système de lançage des générateurs de vapeurs (SNL), P4 – Schémas mécaniques détaillés.
[System Design Manual - Steam Generator Cleaning – Lancing (SNL) System.
P4 – Detailed flow diagrams.]
EYTF/2008/FR/0042 Revision A. Sofinel. October 2008.

[Ref-4] Dossier de Système Elémentaire – Système de lançage des générateurs de vapeurs (SNL), P4.2 – Schémas mécaniques détaillés.
[System Design Manual - Steam Generator Cleaning – Lancing (SNL) System.
P4 – Detailed flow diagrams.]
EYTF/2008/FR/0043 Revision A. Sofinel. October 2008.

2.4.4. Hydraulic design

[Ref-1] A Nicoli. EPR 79/19 TE – Steam Generator operating parameters.
NEEG-F DC 5 Revision F. AREVA. January 2009. (E)

2.4.5. Tube bundle vibrations

[Ref-1] C Blanchard. EPR 79/19 TE – Tube bundle linear vibration analysis.
NEEG-F DC 232 Revision B. AREVA NP – FA3 Project. May 2008. (E)

2.5. MATERIALS AND MATERIAL PROPERTIES

[Ref-1] C Hazelard. Générateur de vapeur 79/19TE - Boîte à eau primaire – Nomenclature - GV droit et gauche.
[Steam generator 79/19TE – Primary water box – Nomenclature – Steam Generator right and left].
NFPMG DB 1207 Revision J. AREVA NP - FA3 Project. November 2008.

[Ref-2] Priol. Générateur de vapeur 79/19TE - ¨Enceinte secondaire – Nomenclature - GV droit et gauche.
[Steam generator 79/19TE – Secondary side casing – Nomenclature – Steam Generator right and left].
NFPMG DB 1211 Revision G. AREVA NP - FA3 Project. January 2009.

2.5.1. Pressure retaining parts

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[Ref-2] Ultrasonic examination of MCL homogeneous and dissimilar metal welds. PEEM-F 11.0505 Revision C. AREVA. March 2012. (E)

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[Ref-1] C Hazelard. Générateur de vapeur 79/19TE - Plan d’ensemble - GV droit et gauche. [Steam Generator 79/19TE – Layout – right and left SG]. NEEG-F DB 1200 Revision H. AREVA NP - FA3 project. November 2008.

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[Ref-3] EPR UK – MCL Crossover lowering.
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[Ref-4] EPR UK – Main Coolant Lines – Dimensioning report.
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3.6.1. Base metal

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4.5.1. Sizing calculations

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[Ref-2] C. Borrel. Pressurizer - General assembly. NEER-F DB 1221 Revision E. AREVA NP. February 2009. (E)

[Ref-3] UK EPR^{TM} - RCC-M Modification Sheet FM 1060. Introduction of 20 MND 5 steel grade. AFCEN. (E)

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5. PRESSURISER RELIEF LINE

5.2. APPLICABLE CRITERIA, HYPOTHESES AND CHARACTERISTICS

5.2.2. Hypotheses and characteristics for the pressuriser relief line

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[Ref-2] Brouttelande & Ghebbi. Steam generator supports, general assembly. NEER-F DB 1248 Revision B. AREVA NP. March 2007. (E)

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