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SUB-CHAPTER 10.6 – MAIN FEEDWATER SYSTEM

0. SAFETY REQUIREMENTS

0.1. SAFETY FUNCTIONS

0.1.1. Reactivity control

In normal operation, the feedwater flow regulation system (ARE [MFWS]) must contribute to limiting primary circuit over-cooling.

In accident situations (PCC, RRC-A, specific studies), the ARE [MFWS] system must contribute to limiting primary circuit over-cooling.

0.1.2. Decay heat removal

When operating at power the ARE [MFWS] system must contribute, with the main steam system circuit, to removing the heat produced by the reactor core.

0.1.3. Containment of radioactive substances

In case of break on RCP [RCS] inside the containment, ARE [MFWS] parts inside the reactor building must be considered as an extension of the third containment barrier.

In the case of accidental contamination of the steam generator secondary system:

- The ARE [MFWS] pipework forms part of the containment of the steam generator,
- The ARE [MFWS] system must prevent overfeeding of the steam generators.

In the case of a secondary break inside the Reactor Building, the ARE [MFWS] system must limit pressurisation of the containment.

0.2. FUNCTIONAL REQUIREMENTS

0.2.1. Reactivity control

In normal operation, the ARE [MFWS] must maintain the required water level in the steam generators.

In post-accident conditions, the main feedwater system must rapidly isolate feed to prevent primary system over-cooling.

0.2.2. Removal of decay heat

The ARE [MFWS] must maintain the level of water in the steam generators at the required value and within limits compatible with the NSSS protection systems during steady state and normal operating transients.

0.2.3. Containment of radioactivity

The feedwater system inside the reactor building must be designed to remain intact and to form part of the containment boundary in the event of a reactor coolant pipe break, ranging from the smallest PCC-2 breaks to the breaks dealt with in RCC-A conditions, and in some other specific cases.

The only cases of accidental contamination of steam generators are accidents involving the rupture of one or more steam generator tubes (in PCC-3, PCC-4 events and some specific situations). In all of these cases, the following features must combine to ensure that the associated discharge limits are met:

- the check valves inside the Reactor Building and the isolation valves outside the Reactor Building must be able to contain the primary coolant which has leaked into the secondary system,

- isolation of the ARE [MFWS] supply to the affected SG is required for radioactive release mitigation.

In the case of a secondary break inside the Reactor Building, the ARE [MFWS] lines must be isolated to prevent the design pressure of the containment from being exceeded.

Note: For the main reactor coolant pipes, pipework rupture is precluded in accordance by application of the Break Preclusion principle.

0.3. DESIGN REQUIREMENTS

0.3.1. Safety classification requirements [Ref]

0.3.1.1. Safety classification

As a system contributing to the containment of radioactivity (in case of SGTR) and fulfilling safeguard functions, the ARE [MFWS] is safety classified according to the classification principles presented in Sub-chapter 3.2 for classification of structures.

0.3.1.2. Single failure criterion (active and passive)

For components ensuring F1 functions, the single failure criterion must be applied to ensure a satisfactory level of redundancy (see Sub-chapter 3.2).

0.3.1.3. Backed-up power supply

The power supply of components with an F1 function must be backed-up to ensure operation following a loss of off-site power.

0.3.1.4. Qualification for operating conditions

The parts of the ARE [MFWS] which fulfil safety functions must be qualified to function during the associated normal and post-accident conditions (see Sub-chapter 3.6).

0.3.1.5. The classification of mechanical, electrical and Instrumentation and Control equipment

The classification of mechanical, electrical and Instrumentation and Control equipment must follow the rules laid down in Sub-chapter 3.2.

0.3.1.6. Seismic classification

The parts of the ARE [MFWS] contributing to containment functions and receiving control signals must be seismically classified according to the rules presented in Sub-chapter 3.2.

0.3.2. Other regulatory requirements

0.3.2.1. Essential safety rules

See Sub-chapter 3.1

0.3.2.2. Technical Guidelines

The Technical Guidelines for the design and construction of the next generation of nuclear pressurised water reactors are described in Sub-chapter 3.1.

0.3.3. Internal and external hazards

0.3.3.1. Internal hazards

The ARE [MFWS] must be protected against internal hazards as defined in Sub-chapter 13.2.

0.3.3.2. External hazards

The ARE [MFWS] must be protected against external hazards as defined in Sub-chapter 13.1.

0.4. TESTING

0.4.1. Commissioning tests

Commissioning tests are required to demonstrate the satisfactory design and performance of the ARE [MFWS].

0.4.2. Periodic tests and in-service inspection

This system must be designed to allow periodic visual inspection of the main components.

The ARE [MFWS] must be designed to allow the performance of periodic tests in compliance with the operating requirements.

1. FUNCTIONAL ROLE OF THE SYSTEM

1.1. STEAM GENERATOR WATER SUPPLY

The main function of the ARE [MFWS] is to maintain the level of water in the steam generators at the required value and within limits compatible with the NSSS protection systems, both in steady state operation and during normal operating transients. At power, the ARE [MFWS] ensures the regulation of feedwater supplied by the conventional island feed pumps. The APA [MFWPS] (Motor-driven FeedWater Pump System) supplies water to the steam generators at power and during all normal steam generator shutdown scenarios [Ref]. The AAD [SSS] system is used during the start-up and shutdown phases [Ref]. Nevertheless, each set of APA [MFWPS] pumps can supply water to the steam generators via the AAD [SSS] system if the AAD1 sub-function fails. Feedwater flow rate is controlled by the ARE [MFWS].

1.2. STEAM GENERATOR ISOLATION

The ARE [MFWS] fulfils the following functions:

- it prevents an excessive mass flow to ensure that there is no overfeeding of the SG and no unacceptable rise in containment pressure, in the event of MFWLB or MSLB in the containment.
- it prevents containment bypass in the case of LOCA,
- it prevents the depressurisation of unaffected SGs in the event of an non isolable rupture of an ARE [MFWS] pipe inside the containment,
- it avoids the depressurisation of SGs in the event of an isolable rupture of an ARE [MFWS] pipe inside the containment,
- it limits radioactive discharges in the event of SGTR,
- it limits over-cooling in the event of rupture of main steam pipework.

2. DESIGN [REF]

2.1. STEAM GENERATOR FEED SUPPLY

Depending on the SG level, each SG is supplied with water from the conventional island at flow rates up to 645 kg/s at a maximum temperature of 230ºC (at 100% of nominal power). The flow supplied depends on the power and operating state of the reactor and hence the steam pressure and flow rate from the SG (plus the blowdown flow).

However the system is designed on the basis of 105% of nominal power (NP) to provide a margin which leads to a total design feedwater flow (Qmax x 105%).

The flow rate is regulated by pneumatic control valves.

The low flow control valve is designed so that the flow rate never exceeds 20% of the water supply needed at 100% NP.

The high flow control valve is designed for Qmax (the low flow control valve remaining open when the flow rate exceeds 20%).

2.2. STEAM GENERATOR ISOLATION

To avoid over-filling of the SG, the ARE [MFWS] high flow control valves close:

- on a reactor trip signal,
- on a high SG water level signal.

The low flow ARE [MFWS] control valves close:

- on a high SG water level signal (failure of the feedwater, SGTR),
- on signals indicating SG depressurisation (MSLB, FWLB).

The design basis for the closure time of the ARE [MFWS] valves is the rupture of a feedwater pipe or the rupture of a steam pipe. The closure time is consistent with the accident analysis assumptions (see Chapter 14).

Isolation of the SG complies with the single failure criterion by using redundant isolation valves.

3. SYSTEM DESCRIPTION AND DESIGN PARAMETERS

3.1. GENERAL DESCRIPTION OF THE SYSTEM

<u>Outside the reactor building containment</u>

Four main ARE [MFWS] lines go from the feedwater manifold to the SGs and are each fitted with a high-flow control valve.

Each high-flow line is fitted with a bypass line with a low flow control valve.

The high flow and low flow control valves can be isolated by motorised valves:

- the high flow line is fitted with two isolation valves, one upstream and one downstream,
- the low flow line is fitted with two isolation valves, one upstream and one downstream,

- the common line downstream of these two lines is fitted with a main isolation valve.

A very low flow line, used while operating with the AAD [SSS], is fitted with a control valve, a flow meter and a downstream isolation valve (common with the low flow line).

A venturi is fitted to each low flow supply line. Another venturi is fitted to each common line. An orifice plate (KD) is fitted upstream of the venturi on each common line.

Inside the reactor building

A check valve is installed on each supply line, as close as possible to the SG.

A simplified flow diagram [Ref] for one train (the four trains are identical) of ARE [MFWS] is given in Sub-chapter 10.6 - Figure 1.

3.2. LOCATION OF EQUIPMENT

Almost all of the ARE [MFWS] is located in the upper part of two of the safeguard buildings (SB), with the exception of the components beyond the containment penetrations (SG connections and the isolation check valves) which are located in the reactor building.

3.3. SYSTEM DESIGN PARAMETERS

Flow control valves

The low flow control valve is designed so that the flow rate never exceeds 20% of the water supply needed at 100% NP.

The high flow control valve is designed for 100% of the feedwater flow needed at 100% NP (see section 2.1).

Operating requirements (decoupling value from conventional island side) lead to a ΔP of around 3.5 bar across the valves for the design flow rate.

The control valves operate quickly enough to maintain the level of the SG at the setpoint value during the standard PCC-1 transients and following reactor trip.

Isolation valves

The isolation valves have a backup power supply and are actuated as follows:

- The main feedwater supply isolation valves are powered by the associated division.
- The high flow and low flow isolation valves are powered by the neighbouring division.

The isolation valve closure times are consistent with the accident analysis assumptions as discussed in section 2.2.

Check valves

The check valve close to the SG is fitted with a shock absorber to lessen the effects of water hammer when closing, in case of upstream feedwater line rupture.

Requirements of the interface with the conventional island

The requirements on the conventional island consist of supplying the SG with water at the correct flow and pressure conditions from the main feedwater pump trains so as to ensure the proper control of flow (see section 2.1).

The temperature and water chemistry for the SG is also controlled by the conventional island.

4. RANGE OF OPERATING CONDITIONS

4.1. NORMAL OPERATION

In normal operation, the ARE [MFWS] maintains the level in the steam generators at the required value by regulating the ARE [MFWS] flow rate supplied by the feedwater pumps.

In normal operation, the SG level control system actuates the high flow and low flow control valves successively.

4.2. START-UP, COOLING UNTIL LHSI CONNECTION CONDITIONS IN RRA [RHR] MODE

The isolation valves corresponding to each SG are open.

During the cooling phase, the feedwater flow rate is automatically adjusted by the ARE [MFWS] system via the SG level control system. During plant warm-up, the required flow being lower, the SG supply is manually controlled.

The feedwater is supplied by the AAD [SSS] system.

4.3. ISOLATION AND REGULATION

Principles

- Isolation function:

Isolation is performed for each SG by the reactor protection system by closing the redundant isolation valves.

- SG water level control function:

Control of SG water level is performed for each SG. The SG water level is kept at its setpoint by a flow control loop which provides two different setpoints, one for each control valve (low flow and high flow control valves). At low load (< 20% Nominal Power [NP]), the control system actuates the low flow control valves, At higher load (> 20% NP) it acts on the high flow control valves (once the low flow valves are fully open).

5. SAFETY ASSESSMENT

Break Preclusion

The Break Preclusion concept does not apply to the ARE [MFWS] lines within and outside the containment for the following reasons:

- a guillotine break on ARE [MFWS] pipework is acceptable in terms of safety (see Chapter 14);
- as their wall thickness is less than that of the VVP [MSSS] lines, the ARE [MFWS] lines are under greater stress and less tolerant to large defects. In particular they are more sensitive to corrosion/erosion damage and fatigue damage;
- rupture of an ARE [MFWS] line outside the containment does not have any knock-on effects on the steam lines or on other ARE [MFWS] lines as each ARE [MFWS] line outside the containment is situated in a heavily enclosed area.

6. SPECIFIC PROVISIONS FOR TESTS

The closure of safety classified isolation valves is tested periodically. The correct functioning of the high flow control valves is confirmed during the normal operation of the unit and does not require any specific test.

Periodic tests are carried out on the orifice plate (KD), which include cross-checking with the venturi measurements and performance during in-service inspection.

The in-service inspection programme will be drawn up during the EPR detailed design phase. It will be confirmed that each inspection site is easily accessible. It will also be confirmed that the performance expected from the non-destructive tests is compatible with design and manufacturing (surface state, geometry, etc.). For any workshop weld for which an inspection is planned, non-destructive tests will be carried out in the factory and will have the status of 'initial test'.

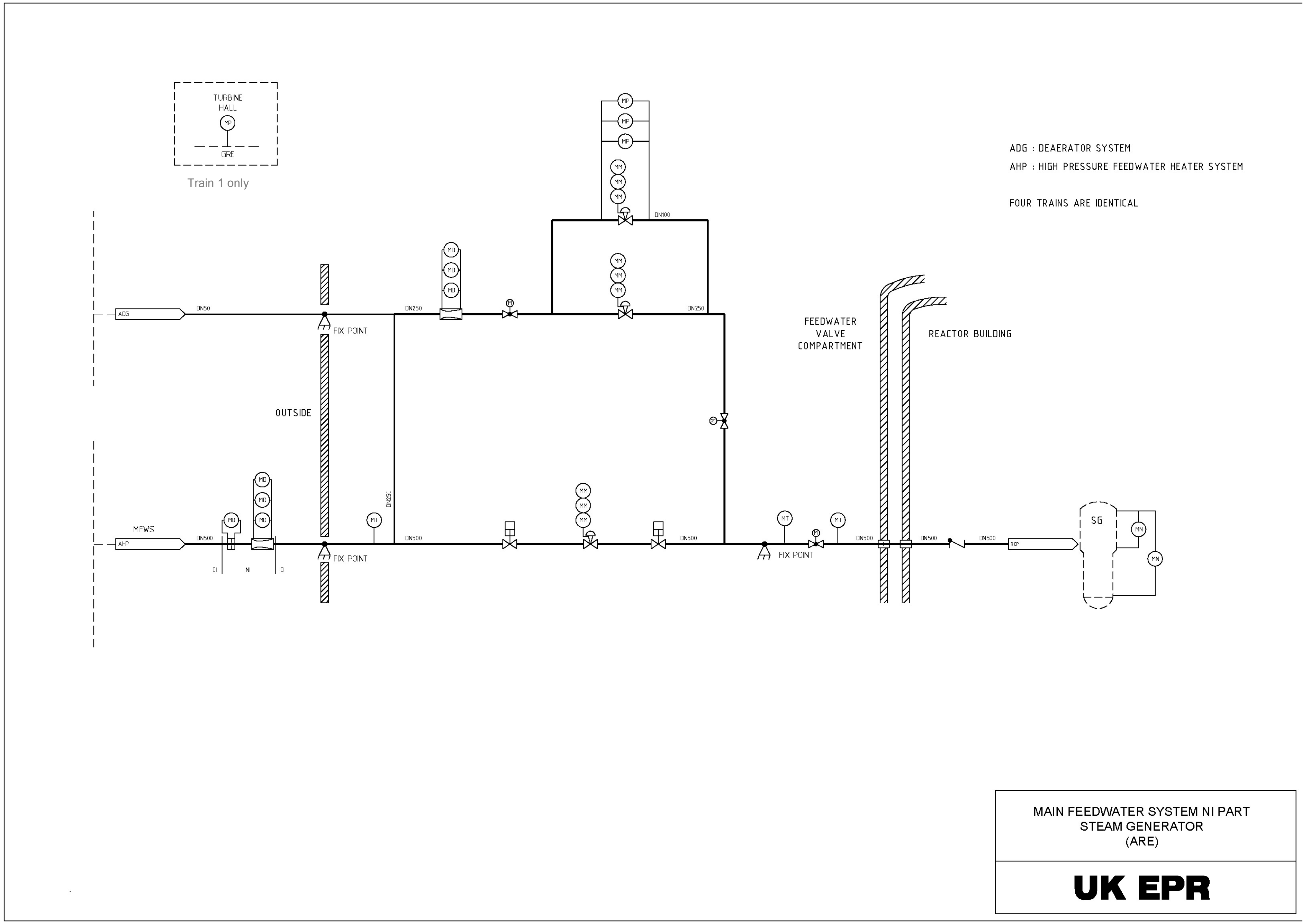
A list of non-sensitive areas will be drawn up in the inspection programme.

The non-sensitive area inspection programme will be reduced provided that:

- it has been demonstrated that there is no risk of defects developing during operating conditions,
- it has been confirmed by NDT that there are no unacceptable defects.

The periodic test programme and the preventive maintenance programmes will be drawn up during the detailed design stage of the EPR unit.

SUB-CHAPTER 10.6 - FIGURE 1: ARE [MFWS] SIMPLIFIED FLOW DIAGRAM – OVERALL VIEW FOR TRAIN 1



SUB-CHAPTER 10.6 – GENERAL DESCRIPTION

External references are identified within this sub-chapter by the text **[Ref]** 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.

0. SAFETY REQUIREMENTS

0.3. DESIGN REQUIREMENTS

0.3.1 Safety classification requirements

[Ref] System Design Manual - Main Feedwater System (ARE [MFWS]) - Part 2: System Operation. NESS-F DC 576 Revision A. AREVA. September 2009. (E)

NESS -F DC 576 Revision A is the English translation of NESS-F DC 70 Revision C.

1.1. STEAM GENERATOR WATER SUPPLY

[Ref] System Design Manual – First Stage System Design Manual for the Motor-Driven Feedwater Pump System (APA [MFWPS]) ETDOFC060117 Revision A1. EDF December 2009. (E)

ETDOFC060117 Revision A1 is the English translation of ETDOFC060117 Revision A

[Ref] System Design Manual - Start-Up and Shutdown Feedwater System (AAD [SSS]) - 1st Stage. ETDOFC060129 Revision A1. EDF. January 2010. (E)

ETDOFC060129 Revision A1 is the English translation of ETDOFC060129 Revision A.

2. DESIGN

[Ref] System Design Manual - Main Feedwater System (ARE [MFWS]) - Part 2: System Operation . NESS-F DC 576 Revision A. AREVA. September 2009. (E)

NESS -F DC 576 Revision A is the English translation of NESS-F DC 70 Revision C.

[Ref] System Design Manual - Main Feedwater System (ARE [MFWS]) - Part 3: System Design. NESS-F DC 577 Revision A. AREVA. September 2009. (E)

NESS-F DC 577 Revision A is the English translation of NESS-F DC 71. Revision D.

[Ref] System Design Manual - Main Feedwater System (ARE [MFWS]) - Part 4: Flow diagrams. NESS-F DC 593 Revision A. AREVA. September 2009. (E)

NESS-F DC 593 Revision A is the English translation of NESS-F DC 72. Revision E.

[Ref] System Design Manual – Main Steam Generator Feedwater System (ARE [MFWS]) P5 - Instrumentation and Control. NESS-F DC 611. Revision A. AREVA. December 2009. (E)

NESS-F DC 611 Revision A is the English translation of NESS-F DC 161 Revision C.

3. SYSTEM DESCRIPTION AND DESIGN PARAMETERS

3.1. GENERAL DESCRIPTION OF THE SYSTEM

Inside the reactor building

[Ref] System Design Manual - Main Feedwater System (ARE [MFWS]) - Part 4: Flow diagrams. NESS-F DC 593 Revision A. AREVA. September 2009. (E)

NESS-F DC 593 Revision A is the English translation of NESS-F DC 72. Revision E.