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REVISION HISTORY

Issue	Description	Date
00	First issue for INSA information.	11-12-2007
01	Integration of technical and co-applicant review comments.	27-04-2008
02	June 2009 update including: - Text clarifications - Inclusion of references - Plant characteristics evolutions to account for December 2008 design freeze (notably partial cooldown rate, pressuriser safety valves and normal spray classification and thermal hydraulic parameters)	28-06-2009
03	Consolidated Step 4 PCSR update: - Minor editorial changes - Addition of the automatic RBS [EBS] actuation upon “SG pressure < MIN4” (section 5.1.2 and Table 9) - Table 9 extended to include F1 signals for steam line activity (KRT-VVP) and secondary side sampling system activity (KRT-RES); Figure 4 updated	28-03-2011
04	Consolidated PCSR update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Deletion of French reference NFPSR DC 1033 - Minor editorial changes Continued on next page	27-09-2012

REVISION HISTORY (Cont'd)

Issue	Description	Date
04 cont'd	Consolidated PCSR update: - Additional I&C action for anti-dilution protection; loss of RCV [CVCS] with detected boron dilution results in closure of the REA [RBWMS] and automatic actuation of the RBS [EBS] by the RCSL (§5.2) - Steady-state error increased for average coolant temperature (footnote added) and pressuriser level (Table 2)	

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SUB-CHAPTER 14.1 - PLANT CHARACTERISTICS TAKEN INTO ACCOUNT IN THE ACCIDENT ANALYSES

This sub-chapter describes the plant characteristics that apply to the accident analyses. The characteristics which are specific to a particular accident analysis are specified within the section describing that accident analysis.

The plant characteristics assumed in the accident analyses cover:

- The plant geometrical data,
- The plant initial conditions,
- The reactivity coefficients,
- The residual heat,
- The I&C signals, related to reactor trip and safety systems operation,
- The safety systems characteristics.

Where appropriate, the PCC accident analyses described in Sub-chapters 14.3 to 14.5 use suitably conservative values of these characteristics (either maxima or minima), in accordance with the PCC analyses rules of Sub-chapter 14.0, and these values are given in this sub-chapter. When used in the RRC-A accident analyses described in Sub-chapter 16.1, the realistic values are also given so that all of the generic data is grouped within this sub-chapter.

1. PLANT GEOMETRICAL DATA

Sub-chapter 14.1 - Table 1 lists the main geometrical data related to the reactor coolant system and the steam generator secondary side.

Sub-chapter 14.1 - Figure 1 and Figure 2 show the SG and pressuriser geometries, with indication of the corresponding level measurements.

2. PLANT INITIAL CONDITIONS

In accident analyses, conservative initial conditions are obtained by adding or subtracting, as appropriate, the maximum steady state errors to the nominal values.

The steady state errors include the measurement error, the steady state fluctuations and, where appropriate, the control dead band.

The nominal values and the associated maximum errors are defined in Sub-chapter 14.1 - Table 2 [Ref-1], for all the relevant parameters:

- core power,

- pressuriser pressure,
- RCP [RCS] average temperature,
- pressuriser level,
- SG level,
- SG water mass.

The nominal core power used in the accident analysis is 4500 MWth. The assumed operating point corresponds to a SG heat transfer regime without plugged tubes or SG tube fouling.

Note: Extended fuel cycle operation is not covered by the safety demonstration presented in this Pre-Construction Safety Report.

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimises adverse power distributions by adequate positioning of control rods and adherence to operating instructions.

The most conservative power distributions which can occur during normal operation are assumed for the starting point of the transient studies. They correspond to the surveillance function thresholds on the maximum linear power density and minimum DNBR. This is discussed in Sub-chapter 4.4 for those transients that lead to a departure from nucleate boiling ratio (DNBR) below the limit and in Sub-chapter 4.3 for the transients that may be limited by high power level protection.

Sub-chapter 14.1 - Table 3 provides additional information on other pertinent parameters.

The reactor pressure vessel (RPV) flow mixing assumed for the EPR_{4500} is as follows:

- When low RPV flow mixing is conservative for the accident analysis, low mixing between RCP [RCS] loop flows is taken into account inside the RPV, based on the following two assumptions (preliminary values):

    - o 86% of the flow entering through the inlet RPV-nozzle I_i remains in the associated core quarter Q_i at the core inlet,
    - o 95% of the core outlet flow issued from core quarter Q_i goes through the outlet RPV-nozzle O_i (where i is related to loop i).

    As a result, approximately 82% (95% x 86%) of inlet RPV-nozzle I_i flow goes through outlet RPV-nozzle O_i (O_i and I_i being associated with the same core quadrant Q_i). The remaining flow is equally split between the three other loops.

    The minimum mixing data are obtained from the EPR RPV thermal-hydraulic design studies [Ref-2].

- When high RPV flow mixing is conservative, or when the extent of mixing has no significant impact in the accident analysis, full mixing is assumed within the RPV. In this case the temperatures of the four core quadrants are identical and equal to the average value.

3. REACTIVITY COEFFICIENTS

The transient response of the reactor system is dependent on reactivity feedback effects, in particular on the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in Sub-chapter 4.3.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. The values used are given in Sub-chapter 14.1 - Table 4. They refer to the core point kinetic model, and do not address some specific accidents where a three-dimensional core model is used.

The RCP [RCS] boron concentrations at the initial state for nominal operating conditions, and the required boron concentration at the RIS/RRA [SIS/RHRS] connection conditions, are addressed in Sub-chapter 14.1 - Table 5 for UO_2 and 30%-MOX fuel managements.

4. FISSION POWER AND DECAY HEAT AFTER REACTOR TRIP (RT)

The power in a subcritical core consists of:

- The residual fissions due to delayed neutrons (term A),
- The decay of U239 and Np239 capture products (term B),
- The decay of fission products and actinides, with the exception of U239 and Np239 (term C).

The thermal power due to residual fissions (term A) following RT depends on the number of neutrons generated by the different neutron sources in the core, including:

- The decay of the delayed neutron precursors,
- The spontaneous fissions of actinides,
- The (α, n) reactions.

The residual thermal power due to the decay of the fission products and actinides (term B+C) depends on the quantity of these products at the time of the RT. The main parameters that influence the nuclide composition in the core are related to the fuel type and the fuel management:

- The initial fuel enrichment,
- The number of fuel batches in the core,
- The enrichment and the burnup of the different batches,
- The operating history of each batch, covering cycle length and specific irradiation power.

The code ORIGEN-S, described in Appendix 14A, is used to calculate the evolution of the nuclide compositions inventory during fuel irradiation and the decay heat power of each chemical element in the core after the RT. The ORIGEN-S databases contain a total of 1700 nuclides.

The code ORIGEN-S provides the term B+C of the residual decay heat. Different uncertainties are modelled in the ORIGEN-S results [Ref-1] [Ref-2]:

- For applications requiring realistic calculations, no uncertainties are applied to the ORIGEN-S results. This yields a best-estimate decay heat curve, BE.
- For applications requiring conservative calculations, the uncertainties applied are [Ref-2]:
    - Max decay curve with 1.645σ (MAX 1.645σ), one-sided probability distribution at a 95% confidence level
        - For UO_2: + 12.34% (t < 30 min), + 8.22% (30 min < t < 1 day)
        - For MOX: + 12.34% (t < 30 min), + 10% (30 min < t < 1 day)
    - Max decay heat with 2σ (MAX 2σ), one-sided probability distribution at a 97.5% confidence level
        - For UO_2: + 15% (t < 30 min), + 10% (30 min < t < 1 day)
        - For MOX: + 15% (t < 30 min), + 12.16% (30 min < t < 1 day)

The resulting decay heat curves are called "BE", "MAX 1.645σ" and “MAX 2σ" respectively:

- "BE" is used in the RCC-A accident analyses,
- "MAX 1.645σ" is used in the PCC accident analyses, except for LOCA-short term,
- "MAX 2σ" is used in the PCC LOCA short term accident analyses assessing peak clad temperature.

Sub-chapter 14.1 - Table 6 shows the three decay heat curves giving the term B+C, each one bounding all the EPR UO_2 and MOX fuel management assumptions.

The term A depends on the prompt and delayed neutron characteristics and the variation in the core effective multiplication factor K during and after RT:

- The kinetic parameters for prompt neutrons and the six groups of delayed neutrons are defined,
- The variation of the effective multiplication factor as a function of time K(t) depends on the characteristics of the RT. These include the rod reactivity worth as a function of time, based on the decoupling dropping characteristic provided in Sub-chapter 14.1 - Table 7 and on the thermal-hydraulic parameters of the core.

Calculation of the term A is performed using a neutron kinetics model, based on the decoupling shutdown rods characteristics given in Sub-chapter 14.1 - Table 8 and the reactivity coefficients given in Sub-chapter 14.1 - Table 4, and relying on the actual core thermal-hydraulic behaviour.

Term A is either calculated as part of the accident analysis, or is provided as input data conservatively generated using a decoupled RT simulation e.g. in the case of LOCA, FWLB, SGTR.

5. I&C SIGNALS

5.1. PRIMARY AND SECONDARY (P/S) I&C SIGNALS

5.1.1. List of F1-signals

The I&C signals considered in the PCC accident analyses refer either to RT actuation or to F1-classified systems actuation. Some non F1-classified signals may be considered if in accordance with the accident analyses rules.

The F1-signals related to the primary and secondary systems (P/S), which are used in the PCC accident analyses, are listed in Sub-chapter 14.1 - Table 9, together with details of their setpoints and associated uncertainties. Sub-chapter 14.1 - Table 9 also includes the F1-systems which are not actuated by the I&C, such as the pressuriser and SG safety valves.

The lists of F1-signals do not address the manual F1A and F1B actions; any such actions are discussed in the relevant sections dealing with the specific accident analysis study.

When forming conservative assumptions, the maximum time delays for actuation of a signal and completion of the resulting action are assumed. Sub-chapter 14.1 - Table 11 and Table 12 define the delays considered in the PCC accident analyses.

5.1.2. Short description of specific F1A P/S I&C functions

The F1 I&C systems are described in Sub-chapter 7.3. Some explanations about specific I&C signals are given below.

<u>RIS [SIS] signal</u>

For the accident analyses, the RIS [SIS] actuation signals considered are:

- In state A:

  "PZR pressure < MIN3"

- In state B, or state C with Reactor Coolant Pumps on:

  "hot leg ΔP_{sat} < MIN"

- In state C with Reactor Coolant Pumps off, or state D:

  "hot leg RCP [RCS] loop level < MIN"

These signals rely on the following sensors:

- For "PZR pressure":

  Four pressuriser pressure sensors with 2/4 logic,

- For "ΔPsat":

  Four hot leg temperature (one per loop) and four hot leg pressure (1 per loop) sensors, with 2/4 logic (two loops out of four).

  ΔP_{sat} is defined as:

  ΔP_{sat} = actual P_{HL} - P_{sat} (actual T_{HL}),

- For "RCP [RCS] loop level":

  Four ΔP-level (1 per loop) measurements, with sensing lines located respectively at the bottom and at the top of the hot leg, with 2/4 logic (two loops out of four).

The actuated RIS [SIS] components are:

- In state A, and state B before accumulator isolation:
  - Start of all MHSI and LHSI pumps injecting into the RCP [RCS] cold legs,
  - Confirmation of accumulator valves opening in state A, accumulator valves remain open in state B.
- In state B after accumulators isolation:
  - Start of all MHSI and LHSI pumps injecting into the RCP [RCS] cold legs,
  - The accumulator valves remain closed.
- In states C and D:
  - Start-up of all MHSI pumps injecting into the RCP [RCS] cold legs,
  - The LHSI pumps remain in the RHR operating mode or in standby,
  - The accumulator valves remain closed.
- Upon RIS/RRA [SIS/RHRS] connection, the following changes are implemented in the PSV and MHSI systems:
  - The PSV setpoint is decreased to approximately 50 bar (I&C electrical signal),
  - The minimum flow line of each MHSI pump is opened to decrease the MHSI delivery pressure into the RCP [RCS] to approximately 40 bar.

Partial cooldown

A partial cooldown is automatically actuated on receipt of the RIS [SIS] actuation signal, in state A and in state B when the SG pressure is above approximately 60 bar.

RCP [RCS] cooldown is required to ensure MHSI injection as the MHSI delivery pressure is low (in the range of 85 to 97 bar, as described in Sub-chapter 6.3). This is to avoid the risk of operation of the SG safety valves following SGTR.

The cooldown is performed using the secondary side and consists of decreasing the VDA [MSRT] setpoint of the four SGs from 95.5 bar to 60 bar, at a rate giving a cooldown of -250°C/h.

At the same time, the GCT [MSB] setpoint is decreased from 90 bar to 55 bar at the same rate. The partial cooldown using the GCT [MSB] is not F1-classified.

Some I&C functions use the signal "partial cooldown complete": this is generated when the pressure in the four SGs is lower than the target pressure for the cooldown (i.e. 60 bar).

Reactor coolant pump trip

In the EPR design, an automatic reactor coolant pump trip signal is actuated following a LOCA.

The purpose of this signal is to improve the LOCA accident mitigation, by preventing a delayed reactor coolant pump trip. This could occur later in the transient as the pumps are not qualified to operate under LOCA conditions. The late trip has the potential to severely increase the rate of depletion of the RCP [RCS] water inventory. This would result in more severe core heating.

The reactor coolant pump trip signal considered is "ΔP over Reactor Coolant Pumps < Min1 and RIS [SIS] signal". The combination with the RIS [SIS] signal aims to prevent spurious reactor coolant pump trip. The ΔP refers to the pressure difference between the pump inlet (crossover leg pressure) and the pump outlet (cold leg pressure). For LOCA mitigation from the power state, one differential pressure sensor per reactor coolant pump is sufficient, with a two out of four logic involving the four pumps (two loops out of four).

In the accident analyses, the most conservative assumption between the earliest reactor coolant pump trip time and the latest pump trip time is selected, based on the relevant decoupling criteria. The "ΔP over reactor coolant pump < MIN1 and SI signal" is assumed when a late trip is limiting, and the "Reactor coolant pump trip at LOOP occurrence" is assumed when an early trip is limiting.

SG pressure drop signals

Two SG pressure drop signals are used for protection against secondary side breaks.

Sub-chapter 14.1 - Figure 3 illustrates the principle that is the basis for these signals: each function is in fact a "SG pressure < MIN setpoint" one, where the MIN setpoint is variable and equal to the actual SG pressure minus 7 bar for the “SG pressure drop > MAX1” function, and minus 17 bar for the “SG pressure drop > MAX2” function, with a limitation of -5 bar/minute on the rate of decrease of the setpoint.

The maximum values of these setpoints is limited to 75 bar and 65 bar, respectively, in order to prevent VIV [MSIV] closure during the SG pressure reduction that follows an overpressure transient (e.g. reactor trip and VDA [MSRT] demand on “SG pressure > MAX1”).

<u>ARE [MFWS] isolation</u>

For each SG, the ARE [MFWS] main line is divided into two parallel paths in the ARE [MFWS]-valves area: a high-load line and a low-load line, as described in Sub-chapter 10.6:

- During full power operation, both the high-load and low-load lines are open. The flow is controlled using the high-load line control valve.
- During shutdown operation, only the low-load line is open. The flow is controlled using the low-load line control valve.

Isolation of the ARE [MFWS] main line is performed in two successive steps:

- Firstly, the high-load line is isolated by closing the two F1A-classified valves located in series on this line. The ARE [MFWS] main line is thus partly isolated, the ARE [MFWS] low-load line remaining open.
- Secondly, the low-load line is isolated by closing the two F1A-classified valves located in series in either the low-load line or the main line. The ARE [MFWS] main line is then fully isolated.

Following any reactor trip signal, the ARE [MFWS] high-load lines are isolated on all SGs. The ARE [MFWS] low-load lines are isolated for individual SGs only as required.

The following illustrates the ARE [MFWS] isolation sequence in case of an uncontrolled SG cooldown (e.g. SLB or FWLB) [Ref-1]:

As a first step, upon generation of signals “SG pressure drop > MAX1” or “SG pressure < Min1” in one SG, the ARE [MFWS] high-load lines are closed in all SGs: the purpose of this function is to limit the feed to the affected SG in case of a secondary side break. In addition, the four VIV [MSIV] are closed in order to isolate the unaffected SG from the affected one. The ARE [MFWS] low-load lines remain open in all SGs, in order to take benefit from the ARE [MFWS] (if the pumps are not tripped) or AAD [SSS] in the unaffected SG.

As a second step, upon generation of signals “SG pressure drop > MAX2” or “SG pressure < MIN2” in the affected SG, the ARE [MFWS] low-load line is closed in that SG: the affected SG is detected by its further depressurisation after the VIV [MSIV] isolation performed in the first step, whereas the pressure in the unaffected SG increases. The purpose of the action on the affected SG is to fully terminate its normal feedwater supply. The isolation of the ASG [EFWS], which has been automatically actuated on low SG level, is performed by the operator, for which a 30 minute delay is assumed in the accident analyses.

<u>RBS [EBS] actuation</u>

The RBS [EBS] injection is automatically actuated in order to increase the margins in case of a steam line break event.

A steam line break leads to a fast depressurisation of the secondary side and a reactor trip is triggered upon “SG pressure drop > MAX1” or “SG pressure < MIN1” signal. This rapid overcooling could lead to a return to core criticality, should this event occur at hot shutdown. A fast injection of borated water limits the reactivity increase and helps maintain the core subcritical.

The RBS [EBS] is automatically actuated upon “SG pressure < MIN4” (see Sub-chapter 14.1 – Table 9) as this is characteristic of a steam line break event.

5.2. CORE RELATED I&C SIGNALS

The core related F1 signals that are considered in the PCC accident analyses for the core protection function are listed in Sub-chapter 14.1 - Table 10.

The corresponding time delays considered in the PCC accident analyses are listed in Sub-chapter 14.1 - Table 11.

A short description of the protection channels which implement these signals is given below, mainly in terms of the required inputs. Further details are presented in Sub-chapter 7.2.

<u>High linear power density</u>

The maximum linear power density value is directly derived from the in-core neutronic instrumentation using Self Powered Neutron Detectors (SPND). All 72 signals are distributed within each of the four divisions of the Reactor Protection System (RPR [PS]).

The fixed in-core instrumentation is described in more detail in the section related to the in-core instrumentation (see Sub-chapter 7.6).

This function initiates a Reactor Trip (and a Turbine Trip).

<u>Low DNBR</u>

The calculation of the minimum DNBR uses the following parameters:

- Hot channel power distribution
- Core inlet temperature,
- Pressure,
- Core flow rate.

The hot channel power distribution is directly derived from the in-core instrumentation using Self Powered Neutron Detectors (SPND). All 72 signals are distributed within each of the four divisions of the RPR [PS].

The core inlet temperature is derived from the cold leg temperature sensors.

The pressure is derived from the RCP [RCS] pressure sensors.

The core flow rate is derived from the Reactor Coolant Pump speed sensors.

This function initiates a Reactor Trip (and a Turbine Trip).

<u>Anti-dilution</u>

The detection and mitigation of a spurious boron dilution is based on a calculation of the boron concentration in the core, using the RCV [CVCS] charging line boron-meter and the RCV [CVCS] charging line flow measurement. The output signal is compared to a sliding threshold that depends on the primary system temperature. This protection channel is operational in all reactor states.

This channel isolates the main sources of spurious dilution from the Chemical and Volume Control System (RCV [CVCS]) by closing two redundant valves in series downstream of the Volume Control Tank.

In the event that the RCV [CVCS] becomes inoperable following a reactor trip, the operating procedure may require manual actuation of the RBS [EBS]. If the operator does not actuate the RBS [EBS] and boron dilution is detected, the RCSL limitation function (F2 classified) will isolate the water source from the REA [RBWMS] and automatically actuate the RBS [EBS].

Further information is given in the section describing the RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant (see section 13 of Sub-chapter 14.3).

<u>High neutron flux rate of change</u>

The following description concerns the high ex-core neutron flux rate of change channel as considered in the analysis of the uncontrolled RCCA bank withdrawal at power transients (see section 9 of Sub-chapter 14.3).

This channel uses the high neutron flux rate of change signal (the derived function Tp / (1+Tp)) provided by the ex-core instrumentation, where:

- p is the Laplace transform, and
- T is a time constant.

The derived function provides a step equal to the input step, and then decays to zero. As an example, the inverse Laplace transformation may be applied to this function using step and ramp inputs (the response to a constant signal is zero).

As this is a derived function, the setpoint value is specified as a percentage of Nominal Power.

This function initiates a Reactor Trip (and a Turbine Trip).

<u>High core power level</u>

The core power level signal is derived from an enthalpy balance using:

- Loop temperature measurements (from cold and hot leg temperature sensors),
- The primary circuit pressure,
- The core flow rate (derived from the Reactor Coolant Pump speed sensors),

This function initiates a Reactor Trip (and a Turbine Trip).

<u>Low-low reactor coolant flow rate (one loop)</u>

The partial loss of forced reactor coolant flow and the reactor coolant pump shaft break or locked rotor are detected using the loop flow rate measurement.

The loop flow rate measurements are provided by differential pressure sensors.

This function initiates a Reactor Trip (and a Turbine Trip).

Low Reactor Coolant Pump speed (loss of four Reactor Coolant Pumps)

To protect against transients due to events affecting the electrical supply of all the Reactor Coolant Pumps, a specific protection channel is required with high accuracy and as short as possible response time.

For such events, Reactor Coolant Pump speed information is used as a representative measure of the core coolant flow.

This function initiates a Reactor Trip (and a Turbine Trip).

High neutron flux (source range)

This function limits the consequences of increase in reactivity transients, particularly in the case of boron dilution due to a non-isolatable rupture of a heat-exchanger tube during shutdown conditions. The neutron flux signal is derived from the source range detectors. This function initiates a shutdown high neutron flux alarm.

Further information is given in the section describing the RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant (see section 13 of Sub-chapter 14.3).

6. SAFETY SYSTEMS CHARACTERISTICS

In the PCC accident analyses, the systems claimed to mitigate the consequences of an event are F1-classified:

- F1A-classified, to bring the plant to the controlled state.
- F1B-classified, to bring the plant to the long term safe shutdown state.

The list of F1 mechanical-systems considered in the PCC accident analyses comprises:

- The core control and shutdown rods, performing reactor trips,
- The RCP [RCS] and SG isolation valves,
- The RCP [RCS] and SG fluid systems performing injection,
- The RCP [RCS] and SG fluid systems performing relief,
- The RCV [CVCS] control tank isolation valves.

These systems are claimed according to the conservative PCC analyses rules defined in section 2 of Sub-chapter 14.0:

- Minimum guaranteed system performance,
- Consideration of the most conservative single failure,
- Consideration of the most conservative unavailability due to preventive maintenance.

Sub-chapter 14.1 - Table 13 to Table 22 provide the minimum and/or maximum characteristics of the F1 fluid-systems claimed in the accident analyses, with information on systems not only used in the PCC accident analyses but also in the overpressure protection (OPP) analyses of section 1.5 of Sub-chapter 3.4 and in the RRC-A accident analyses of Sub-chapter 16.1.

The F1A systems and functions that are considered in the accident analyses (excluding support systems such as the RRI [CCWS], SEC [ESWS]...) are:

- RT (reactor trip),
- Extra boration system (RBS [EBS]),
- Safety injection system (RIS [SIS]) (MHSI, LHSI, accumulators, IRWST),
- Pressuriser safety relief valves (PSV),
- Main steam relief train (VDA [MSRT]),
- SG safety valves [MSSV],
- Main steam isolation valve (VIV [MSIV]),
- Emergency feedwater system (ASG [EFWS]),
- Containment / RCP [RCS] isolation,
- ARE [MFWS] high-load line isolation,
- ARE [MFWS] low-load line isolation,
- SG blowdown isolation (APG [SGBS]),
- Reactor coolant pump trip,
- Main diesel generators start-up,
- Boron dilution detection (boron-meter) and isolation.

The F1B systems and functions that are considered in the accident analyses (other than the F1A ones previously listed) are:

- ASG [EFWS] passive headers,
- ASG [EFWS] isolation,
- LHSI in hot leg SI-mode (switchover to HL injection),
- LHSI in RHR-mode (RIS/RRA [SIS/RHRS] operation),
- Manual reactor coolant pump trip,
- Manual MHSI shutdown,
- Manual isolation of accumulators,

- Manual opening of VDA [MSRT],
- VIV [MSIV] bypass,
- SG blowdown transfer line between two SGs,
- Regarding the integrity or isolation of the Reactor Coolant Pump seals:
    - o the RRI [CCWS] seal cooling,
    - o the reactor coolant pump stand still seal system (DEA [SSSS]), and the isolation valves on the pump seal return lines.

The characteristics of the shutdown rods are already provided in Sub-chapter 14.1 - Table 8.

In addition to the above F1 systems, non-F1 systems may be considered in the PCC accident analyses in accordance with the PCC analyses rules (either as having a negative impact, or having a positive impact but not experiencing some discontinuity in their operation). These systems may also be used in the OPP or RRC-A accident analyses. Their main relevant characteristics are listed in Sub-chapter 14.1 - Table 23.

Sub-chapter 14.1 - Table 24 provides the list of safety functions that are required to ensure safe control of the plant, as well as the systems associated with each of these functions.

Sub-chapter 14.1 - Figure 4 represents a simplified functional diagram of the main F1A fluid systems: RIS [SIS], ASG [EFWS], pressuriser and SG safety valves, VDA [MSRT] (atmospheric steam dump), steam isolation, main feedwater isolation.

7. COMPUTER CODES USED

Appendix 14A provides summaries of some of the principal computer codes used in the transient analyses.

The relevant accident analyses sections also provide summaries of the other codes used. These include, in particular, the specialised codes in which the modelling has been developed to simulate one specific phenomenon for a particular accident, such as those used in the analysis of the hydraulic loads following pipe rupture (see Sub-chapter 3.4).

The computer code used for the relevant calculations is mentioned for each accident analysis presented in Chapter 14.

8. APPROACH USED IN ACCIDENT ANALYSIS WITH RESPECT TO DNB

The overall approach is detailed in the section related to the thermal-hydraulic design (see Sub-chapter 4.4).

A transient analysis with respect to DNBR depends on:

- The type of protection function actuated (low DNBR or specific protection),

- The LCO function used for limiting the initial conditions (low DNBR LCO or other),
- The methodology used to combine the DNBR uncertainties.

Three types of transients have to be considered:

- The transients actuating the low DNBR protection:

  For such transients, the initial DNBR can be chosen at any value (no fixed initial DNBR), and the DNBR design limit is equal to 1.

  These transients are referred to as "type-1 transients".

- The transients actuating a specific protection:
  - If the initial state is power operation, the initial min DNBR is fixed to the so-called "DNBR limiting value" in such a way that, for the worst transient of this type, the minimum DNBR value during the transient meets the DNBR design limit (equal to 1).

    These transients are referred to as "type-2 transients".

    Taking into account all the uncertainties (U) concerning the DNBR calculated by the surveillance system, the "DNBR limiting value" allows the on site DNBR-LCO (Limiting Condition of Operation) to be set at: DNBR-LCO = DNBR limiting value x U
  - If the initial state is zero power, the minimum DNBR is calculated using the thermal-hydraulic conditions, modelling the uncertainties in a deterministic way.

    These transients are referred to as "type-3 transients".

Sub-chapter 14.1 - Table 25 gives the overall approach for these 3 types of transients.

The DNBR analyses consider the following definitions:

- Safety criterion: Radiological limits
- DNBR decoupling criteria:
  - PCC-2 : No DNB (DNBR > DNBR design limit)
  - PCC-3, 4: Percentage of rods in DNB not higher than 10% of the total core
- $DNBR_{RT}$: Actuation threshold of the low DNBR protection function without considering uncertainties (see Sub-chapter 4.4)
- On site $DNBR_{RT}$: Actuation threshold of the low DNBR protection function considering uncertainties (see Sub-chapter 4.4)
- DNBR limiting value: Initial DNBR for type 2 transients (see Sub-chapter 4.4)

- $DNBR_{LCO}$: Actuation threshold of the low $DNBR_{LCO}$ function considering uncertainties (see Sub-chapter 4.4)

- DNBR design limit: For each transient, being above this value ensures that the DNBR criterion is met.

  The design limit is the value to which the DNBR calculated by the thermal-hydraulic design code is compared.

The PCC transients which relate to DNB are classified as follows:

Type-1 transients:

- Uncontrolled RCCA bank withdrawal at power (PCC-2) (Sub-chapter 14.3, section 9)
- RCCA misalignment up to rod drop, without limitation (PCC-2) (Sub-chapter 14.3, section 11)
- Uncontrolled single control rod withdrawal (PCC-3) (Sub-chapter 14.4, section 13)
- RCV [CVCS] malfunction that results in a decrease in boron concentration in the RCP [RCS] [1] (PCC-2) (Sub-chapter 14.3, section 13)
- Excessive increase in secondary steam flow [1] (Sub-chapter 14.3, section 3)

Type-2 transients:

- Uncontrolled RCCA bank withdrawal at power, at very fast reactivity insertion rates (PCC-2) (Sub-chapter 14.3, section 9)
- Inadvertent closure of all VIV [MSIV] (PCC-3) (Sub-chapter 14.4, section 7)
- Inadvertent closure of one VIV [MSIV] (PCC-3) (Sub-chapter 14.4, section 7)
- Short term loss of offsite power (PCC-2) (Sub-chapter 14.3, section 6)
- Forced decrease of reactor coolant flow (PCC-3) (Sub-chapter 14.4, section 9)
- Partial loss of core coolant flow (Loss of one Reactor Coolant Pump) (PCC-2) (Sub-chapter 14.3, section 8)
- Reactor coolant pump shaft break (PCC-4) (Sub-chapter 14.5, section 9)
- Reactor coolant pump seizure (locked rotor) (PCC-4) (Sub-chapter 14.5, section 8)
- RCCA ejection at power [2] (PCC-4) (Sub-chapter 14.5, section 5)

Type-3 transients:

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[1] Regarding DNB behaviour during the transient, these events are covered by the uncontrolled RCCA bank withdrawal at power.
[2] This transient is studied in a conservative way as a type 3 transient.

- Uncontrolled RCCA bank withdrawal in states B to D (PCC-3) (Sub-chapter 14.4, section 12)
- RCCA ejection from zero power (PCC-4) (Sub-chapter 14.5, section 5)
- Steam system piping break (PCC-4) (Sub-chapter 14.5, section 2)

SUB-CHAPTER 14.1 - TABLE 1

MAIN GEOMETRICAL DATA [REF-1]

CORE

- Number of fuel assemblies	241	(type 17x17)
- Number of fuel rods per assembly	265	
- Active height	4.20 m	

RCP [RCS] FLUID VOLUMES

- Reactor vessel	150 m^3	
. Reactor vessel downcomer + lower plenum		53.5 m^3
. Core		27.5 m^3
. Reactor vessel upper plenum + upper head		68.5 m^3
- Pressuriser	75 m^3	
- RCP [RCS] loops	235 m^3	
. Surge line		2.4 m^3
. Hot/cross-over/cold legs (incl. Reactor Coolant Pumps)		14.9 m^3 x 4
. SG plenum and tubes		43.1 m^3 x 4
- TOTAL RCP [RCS]	460 m^3	

RCP [RCS]

- Surge line diameter (inside)	0.325 m
- Hot/cross-over/cold legs diameter (inside)	0.780 m

SG (per SG)

- Number of tubes	5980
- Heat transfer area	7960 m^2
- Tube (in diameter/out diameter/thickness)	16.87/19.05/1.09 mm
- Secondary side fluid volume	238 m^3
- Steam flow limiter section at SG outlet	0.13 m^2

SUB-CHAPTER 14.1 - TABLE 2

PLANT INITIAL CONDITIONS [REF-1]

Maximum steady-state errors

Parameter	**Nominal value at thermal-hydraulic design flow rate**		**Maximum steady-state error**
	at 0% FP No fouling No plugging	**at 100% FP** No fouling No plugging	
- Core power [1]	0% FP	100% FP	± 2% FP
- PZR pressure	155 bar	155 bar	± 2.5 bar
- RCP [RCS] avg. temp.	303.3°C [2]	312.7°C [3]	± 2.5°C [4]
- PZR level	31% R (23 m^3)	56% R (40 m^3)	± 8.5% R (5.8 m^3)
- SG level	49% NR (15.7 m)	49% NR (15.7 m)	± 5% NR (0.35 m)
- SG water mass	107 te	77.2 te	

FP = Full Power

R = Range

NR = Narrow Range

[1] 100% FP = 4500 MWth.
[2] The corresponding SG saturation pressure is 90.0 bar a.
[3] The corresponding SG saturation pressure is 78.0 bar a.
[4] Below 25% NP, the maximum uncertainty is ± 4°C.

SUB-CHAPTER 14.1 - TABLE 3

PLANT INITIAL CONDITIONS [REF-1]

Other pertinent parameters (at 100% Full Power)

Nominal Nuclear Steam Supply System thermal power output	4524 MWth	
Nominal core thermal power	4500 MWth	
ARE [MFWS] flow temperature at SG inlet	230°C	
Main steam flow rate (per SG)	638.1 kg/s	
SG saturation pressure	78 bar a	
Steam moisture at SG outlet (max)	0.25%	
Core average linear power [1]	163.4 W/cm	
	Thermal-hydraulic design flow rate No fouling No plugging	**Best-estimate flow rate** No fouling No plugging
RCP [RCS] -loop flow rate	27185 m^3/h	28320 m^3/h
RCP [RCS] flow rate (4 loops)	22235 kg/s	23140 kg/s
Core flow rate bypass [2]	5.5%	3.46%
Core inlet temperature	295.6°C	296.0°C
Core outlet temperature	331.6°C	330.0°C
Vessel outlet temperature	329.8°C	328.9°C
Vessel dome temperature [3]	329.8°C	328.9°C

Note: Minor deviations from these values, due to the specific nature of the codes, may be observed in the accident analyses.

[1] Related to power generated inside the fuel, corresponding to 97.4% of total power.
[2] Related to hot dome design
[3] RPV dome temperature is conservatively taken at RPV outlet temperature, the maximum possible, while the realistic temperature is close to the average RCP [RCS]-loop temperature.

SUB-CHAPTER 14.1 - TABLE 4

Reactivity Coefficients and fuel managements [Ref-1]
Conservative data set for point kinetic model
enveloping all UO_2 and MOX fuel managements

MODERATOR DENSITY COEFFICIENT [1]

MIN [2]	MAX
0 ΔK/K per g/cm^3	0.515 ΔK/K per g/cm^3

DOPPLER TEMPERATURE COEFFICIENT

MIN	MAX
- 4.03 pcm/°C	- 1.98 pcm/°C

DOPPLER POWER COEFFICIENT

MIN	MAX
- 10.5 pcm/% FP at 100% FP	- 5.1 pcm/% FP
- 11.7 pcm/% FP at 75% FP	- 5.4 pcm/% FP
- 16.1 pcm/% FP at 50% FP	- 5.9 pcm/% FP
- 20.9 pcm/% FP at 25% FP	- 6.0 pcm/% FP
- 28.3 pcm/% FP at 5% FP	- 6.4 pcm/% FP

FP = Full Power

BORON DIFFERENTIAL WORTH

MIN	MAX
hot shutdown / cold shutdown	**hot shutdown / cold shutdown**
- 11.0 / - 15.1 pcm/ppm for 25 ppm	- 4.6 / - 7.1 pcm/ppm
- 10.1 / - 13.3 pcm/ppm for 1000 ppm	- 4.3 / - 6.5 pcm/ppm
- 9.3 / - 12.3 pcm/ppm for 2000 ppm	- 4.1 / - 6.0 pcm/ppm
- 8.7 / - 11.6 pcm/ppm for 3000 ppm	- 3.9 / - 5.6 pcm/ppm

[1]. More realistic values for a nominal coolant density are:
- BOL: 0.09 Δk/k per g/cm^3
- EOL: 0.32 Δk/k per g/cm^3

[2] Decoupling value.

SUB-CHAPTER 14.1 - TABLE 5

RCP [RCS] BORON CONCENTRATION [REF-1]

(uncertainties)

Plant operating conditions	EOL		BOL	
	UO_2	MOX	UO_2	MOX
Nominal operating conditions (full power, incl. maximum Xenon build-up)	10 ppm (140 ppm)	10 ppm (150 ppm)	485 ppm (140 ppm)	1535 ppm (150 ppm)
RIS/RRA [SIS/RHRS] connecting conditions (core subcritical at 150°C, N-1 rods inserted)	244 ppm (200 ppm)	367 ppm (200 ppm)	692 ppm (200 ppm)	1412 ppm (200 ppm)

This data refers to natural boron, not accounting for B10 enrichment. This conservatism maximises the difference in boron concentration between nominal operation at full power and RIS/RRA (SIS/RHRS) connection conditions (150°C in cold legs in natural circulation, ensuring 180°C max in hot legs).

Accounting for the following B10 enrichments retained for UO_2 and MOX, the correspondence between "enriched boron C%" and "natural boron C%" is respectively:

	UO_2	MOX
B10 C% in enriched boron (atomic)	31.3%	33.0%
B10 C% in natural boron (atomic)	19.9%	19.9%
Enriched boron C% (ppm)	natural boron C% (ppm) x 0.625	natural boron C% (ppm) x 0.592

SUB-CHAPTER 14.1 - TABLE 6

RESIDUAL DECAY HEAT (TERM B+C) [REF-1]

Term B+C = decay of fission products and actinides

MAXIMUM DECAY HEAT (**MAX 2σ**) = refers to "ORIGEN-S results + uncertainties"

Uncertainties for UO_2 :	+ 15%	(t < 30min),	+ 10%	(30min < t < 1day)
Uncertainties for MOX:	+ 15%	(t < 30min),	+ 12.16%	(30min < t < 1day)

based on a one-sided probability distribution at a 97.5% confidence level

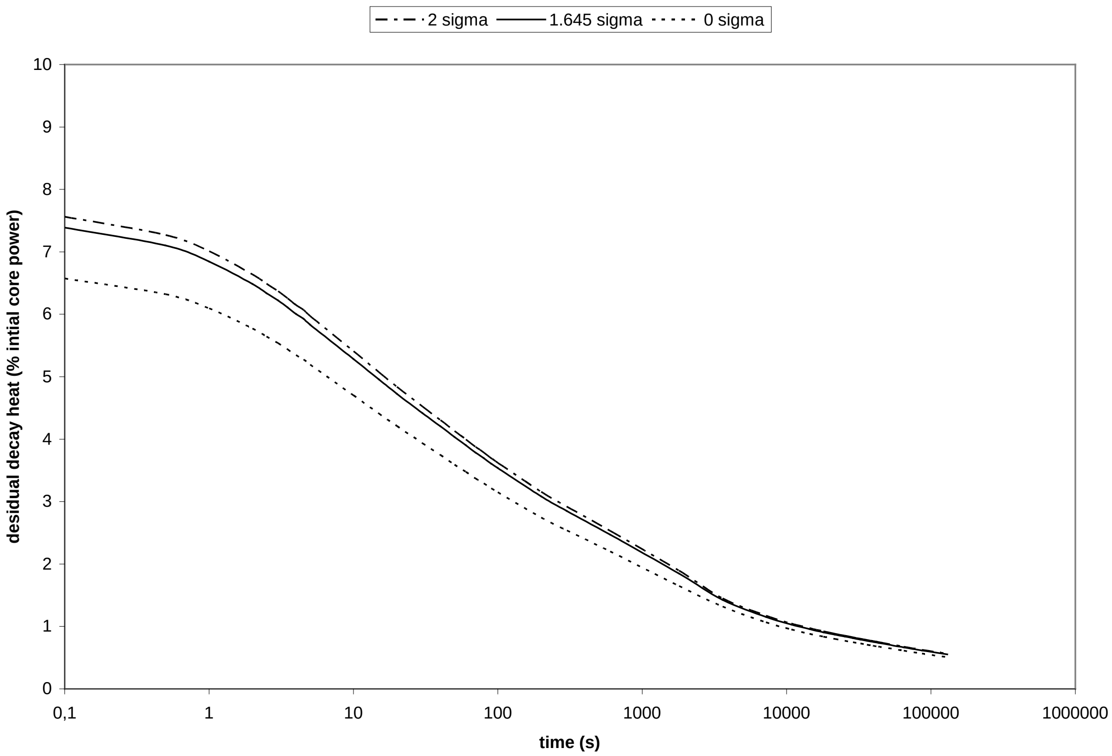
MAXIMUM DECAY HEAT (**MAX 1.645σ**) = refers to "ORIGEN-S results + uncertainties"

Uncertainties for UO_2 :	+ 12.34%	(t < 30min),	+ 8.22%	(30min < t < 1day)
Uncertainties for MOX:	+ 12.34%	(t< 30min),	+ 10%	(30min < t < 1day)

based on a one-sided probability distribution at a 95% confidence level

BEST ESTIMATE DECAY HEAT (**BE**) = refers to "ORIGEN-S without uncertainties"

For each case, the conservative curve between UO_2 and MOX fuel management is chosen

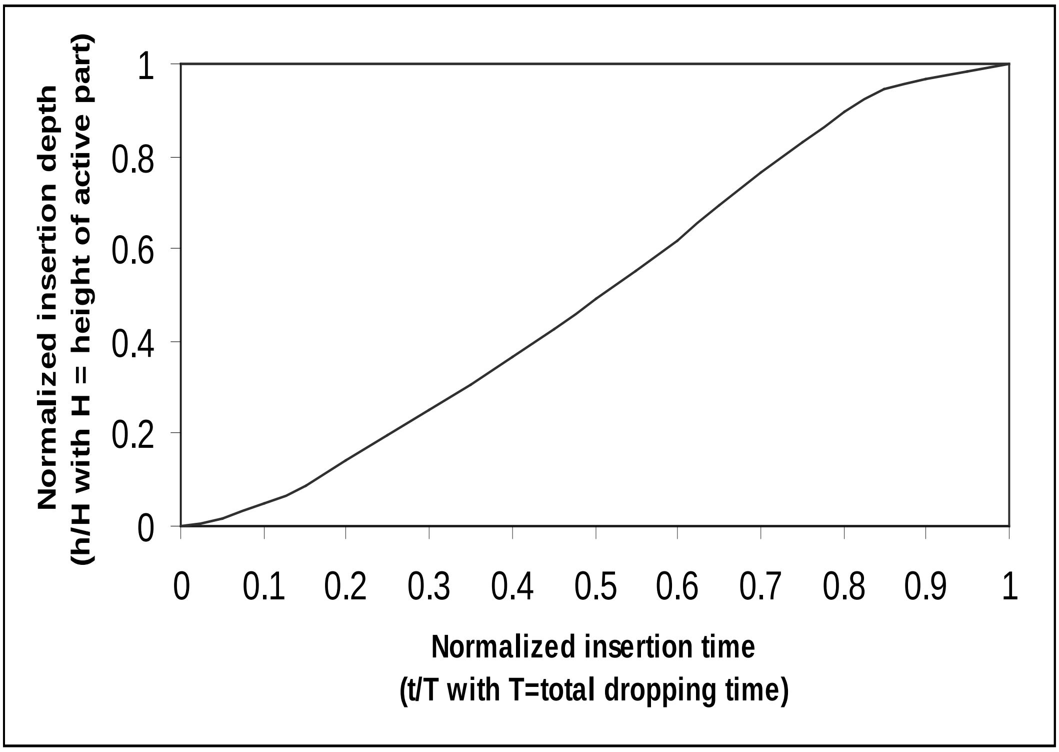


- "MAX 2σ": used in PCC LOCA-short term analyses (PCT concern)
- "MAX 1.645σ": used in PCC accident analyses, except LOCA-short term
- “BE”: used in RRC-A accident analyses.

SUB-CHAPTER 14.1 - TABLE 7

RCCA INSERTION CHARACTERISTICS (RT) [REF-1]

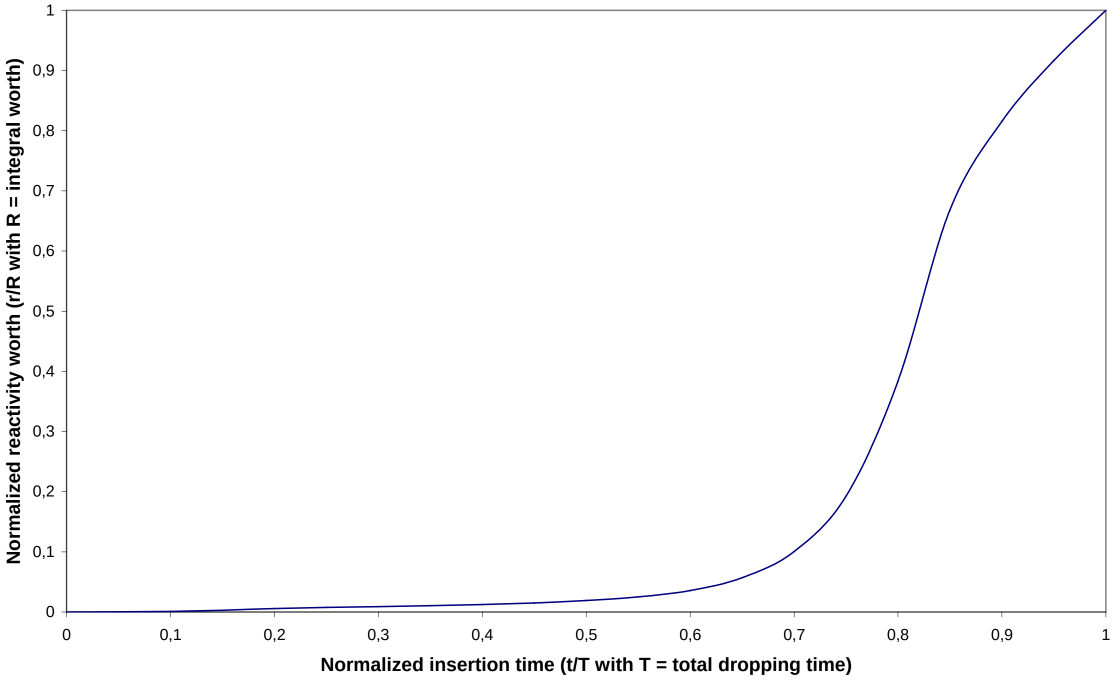
- Total drop time (maximum)	3.5 s	without earthquake
	5 s	with earthquake
- Height of active core	4.20 m	
- Insertion depth versus time (minimum)	see Figure, below	



SUB-CHAPTER 14.1 - TABLE 8

RCCA CHARACTERISTICS (RT) [REF-1]

- Total drop time (maximum)	3.5 s [1] without earthquake 5 s [1] with earthquake
- Integral reactivity worth (minimum)	
. with N-1 rods	5100 pcm for UO_2 , 5100 pcm for MOX
. with N rods	6100 pcm for UO_2 , 5700 pcm for MOX
- Reactivity worth versus time (minimum)	see Figure, below



[1] This is not a safety criterion.

SUB-CHAPTER 14.1 - TABLE 9 (1/7)

F1 SIGNALS (P/S RELATED) [REF-1]

PRESSURISER PRESSURE SCALE

	ACTION	SETPOINT	UNCERTAINTY
[1]	3rd PSV opening (closing)	181 bar (152 bar)	± 1.5 bar
[1]	2nd PSV opening (closing)	178 bar (149.5 bar)	± 1.5 bar
[1]	1st PSV opening (closing)	175bar (147 bar)	± 1.5 bar
MAX2	RT, TT	166.5 bar	± 1.5 bar
MIN2	RT, TT	135 bar	Normal conditions ± 1.5 bar Degraded conditions ± 3 bar
MIN3	RIS [SIS] [2] Partial cooldown [3] RCP [RCS] isolation	115 bar	Normal conditions ± 1.5 bar Degraded conditions ± 3 bar

[1] Indicated, however not being I&C related (hydraulic opening).
[2] In state A (actuation of MHSI + LHSI).
[3] Decrease of all VDA [MSRT] setpoints from 95.5 bar to 60 bar, with a rate of 250°C/h (preliminary value).

SUB-CHAPTER 14.1 - TABLE 9 (2/7)

F1 SIGNALS (P/S RELATED) [REF-1]

PRESSURISER LEVEL SCALE

	ACTION	SETPOINT	UNCERTAINTY
MAX1	RT, TT	85% of Measured Range (10.1 m from bottom of PZR)	Normal conditions ± 2% Degraded conditions ± 5%

RCS LOOP LEVEL SCALE

	ACTION	SETPOINT	UNCERTAINTY
MIN	MHSI [1]	(see section 7 of Sub-chapter 14.5)	(see section 7 of Sub-chapter 14.5)

[1] In states C and D, with reactor coolant pumps off (large mini flow line of MHSI already open).

SUB-CHAPTER 14.1 - TABLE 9 (3/7)

F1 SIGNALS (P/S RELATED) [REF-1]

SG PRESSURE SCALE

	ACTION	SETPOINT	UNCERTAINTY
[1]	SG safety valves opening in affected SG	105.0 bar	± 1.5 bar
MAX1	RT, TT MSRIV opening in affected SG	95.5 bar	± 1.5 bar
MIN1	RT, TT VIV [MSIV] closure ARE [MFWS] high load line isolation in all SGs	50 bar	± 1.5 bar
MIN2	ARE [MFWS] low load line isolation in affected SG	40 bar	± 1.5 bar
MIN3	VDA [MSRT] isolation in affected SG (if VDA [MSRT] has opened)	40 bar	± 1.5 bar
MIN4	RBS [EBS] actuation	35 bar	± 1.5 bar

[1] Indicated, however not being I&C related (spring-loaded valve).

SUB-CHAPTER 14.1 - TABLE 9 (4/7)

F1 SIGNALS (P/S RELATED) [REF-1]

SG PRESSURE DROP SCALE

	ACTION	SETPOINT	UNCERTAINTY
MAX2	ARE [MFWS] low load line isolation in affected SG	- 5 bar/min [2] Variable limit, setpoint 17 bar below the actual pressure in steady state Maximum value 65 bar	± 1.5 bar (on setpoint)
MAX1	RT, TT MSIV closure, ARE [MFWS] high load line isolation in all SGs	- 5 bar/min [2] Variable limit, setpoint 7 bar below the actual pressure in steady state Maximum value 75 bar	± 1.5 bar (on setpoint)

[2] See Sub-chapter 14.1 - Figure 3 and section 5.1.2

SUB-CHAPTER 14.1 - TABLE 9 (5/7)

F1 SIGNALS (P/S RELATED) [REF-1]

SG LEVEL SCALE

	ACTION	SITE SETPOINT	UNCERTAINTY
MAX2	VIV [MSIV] closure and VDA [MSRT] setpoint increase in affected SG RCV [CVCS] charging line isolation [1] (if partial cooldown finished)	18.0 m (85% NR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MAX2	Partial cooldown [2] in all SGs	18.0 m (85% NR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MAX1	ASG [EFWS] isolation in affected SG (if ASG [EFWS] has started)	17.0 m (89% WR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MAX1	RT, TT ARE [MFWS]-HL isolation, ARE [MFWS]-LL isolation in affected SG after time delay (approximately 3 s)	17.0 m (69% NR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MIN1	RT, TT	13.8 m (20% NR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MIN2	ASG [EFWS] actuation in affected SG APG [SGBS] isolation	7.85 m (40% WR cold side)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR

WR : Wide Range, calibrated at hot standby on cold side
NR : Narrow Range, calibrated at 100% power
MR : Measuring Range
ARE-HL : ARE [MFWS] high-load line
ARE-LL : ARE [MFWS] low-load line

[1] Reactor coolant pump seal injection remains available.
[2] Decrease of all VDA [MSRT] setpoints from 95.5 bar down to 60 bar with a rate of -250°C/h.

SUB-CHAPTER 14.1 - TABLE 9 (6/7)

F1 SIGNALS (P/S RELATED) [REF-1]

ΔP ACROSS REACTOR COOLANT PUMPS AND SI SIGNAL

	ACTION	SETPOINT	UNCERTAINTY
MIN	Reactor coolant pump trip	80% of nominal ΔP across Reactor coolant pump	Normal conditions ± 3% Degraded conditions ± 5%

HOT LEG SATURATION MARGIN ΔPSAT

	ACTION	SETPOINT	UNCERTAINTY
MIN	MHSI+LHSI actuation + Partial cooldown MHSI actuation	(see section 5 of Sub-chapter 14.4 and section 6 of Sub-chapter 14.5) [3] (see section 7 of Sub-chapter 14.5) [4]	(see section 5 of Sub-chapter 14.4 and section 6 of Sub-chapter 14.5) [3] (see section 7 of Sub-chapter 14.5) [4]

[3] In state B.
[4] In state C with reactor coolant pumps on (large mini flow line of MHSI already open).

SUB-CHAPTER 14.1 - TABLE 9 (7/7)

F1 SIGNALS (P/S RELATED)

STEAM LINE ACTIVITY

	ACTION	SETPOINT	UNCERTAINTY
MAX	None[1]	To be defined	To be defined

SG SECONDARY SIDE SAMPLING SYSTEM ACTIVITY

	ACTION	SETPOINT	UNCERTAINTY
MAX	None[1]	To be defined	To be defined

[1] This F1A alarm is used to detect activity increase in case of SGTR so that the operator can trigger a manual reactor trip (Sub-chapter 14.4, section 6)

SUB-CHAPTER 14.1 - TABLE 10

F1 SIGNALS (CORE RELATED)

CORE PROTECTION

F1 SIGNAL	LIMIT	ACTION	SETPOINT
High linear power density	MAX	RT, TT	590 W/cm
Low DNBR	MIN	RT, TT	[1]
Anti dilution (boron-meter)	MIN	Stop/start RCV [CVCS] isolation	
High neutron flux rate of change	MAX	RT, TT	13% NP
High core power level	MAX	RT, TT	120% NP
Low-low loop flow rate (loop 1)	MIN	RT, TT	25% NF
(nuclear power level above a permissive threshold)			
Low-low loop flow rate (loop 2)	MIN	RT, TT	25% NF
(nuclear power level above a permissive threshold)			
Low-low loop flow rate (loop 3)	MIN	RT, TT	25% NF
(nuclear power level above a permissive threshold)			
Low-low loop flow rate (loop 4)	MIN	RT, TT	25% NF
(nuclear power level above a permissive threshold)			
Low reactor coolant pump speed - Loss of reactor coolant pump	MIN	RT, TT	91% NS
High neutron flux (source)	MAX	Alarm	[2]

NP: Nominal Power level
NF: Nominal Flow rate
NS: Nominal Speed

[1] Setpoint used in safety analyses.
[2] Flux value equal to three times the current flux in shutdown conditions.

SUB-CHAPTER 14.1 - TABLE 11 (1/2)

I&C SIGNAL DELAYS (P/S RELATED) [REF-1]

Type of channel	$T_{I\&C}$ (max)	T_{RT} (max)	T_{SA} (max)
Pressure	0.9 s	1.2 s	0.9 s + T_D
Level	1.5 s	1.8 s	1.5 s + T_D
Temperature	4.5 s	4.8 s	4.5 s + T_D

$T_{I\&C}$ = Response time of I&C channel, including sensor (up to the actuator)

T_{RT} = Total delay to start RCCA dropping ($T_{I\&C}$ + 0.3 s)
0.3 s = RT breakers opening + gripper release

T_{SA} = Total delay to reach the full completion of a safeguard action ($T_{I\&C}$ + T_D)
T_D = see Sub-chapter 14.1 - Table 12

SUB-CHAPTER 14.1 - TABLE 11 (2/2)

I&C SIGNAL DELAYS (CORE RELATED) [REF-1]

Type of channel	T (I&C) (max)	T (RT) (max)
High linear power density	0.6 s	0.9 s
Low DNBR (including SPND response time)	0.6 s	0.9 s
Anti dilution (boron-meter)	0.5 s	[1]
High neutron flux rate of change [2]	0.3 s	0.6 s
High core power level [3]	0.5 s	0.8 s
Low reactor coolant flow rate (loop i)	0.9 s	1.2 s
Low reactor coolant pump speed - Loss of reactor coolant pump	0.3 s	0.6 s
High neutron flux (source)	0.3 s	0.6 s

T (I&C) : response time of I&C channel, including sensor (up to the actuator)
T (RT) : total delay to start RCCA dropping (T (I&C) + 0.3 s)

[1] The isolation of the RCV [CVCS] is done in 40 s.
[2] Time constant = 30 s.
[3] Not including sensor response time.

SUB-CHAPTER 14.1 - TABLE 12

SAFEGUARD ACTION DELAYS [REF-1]

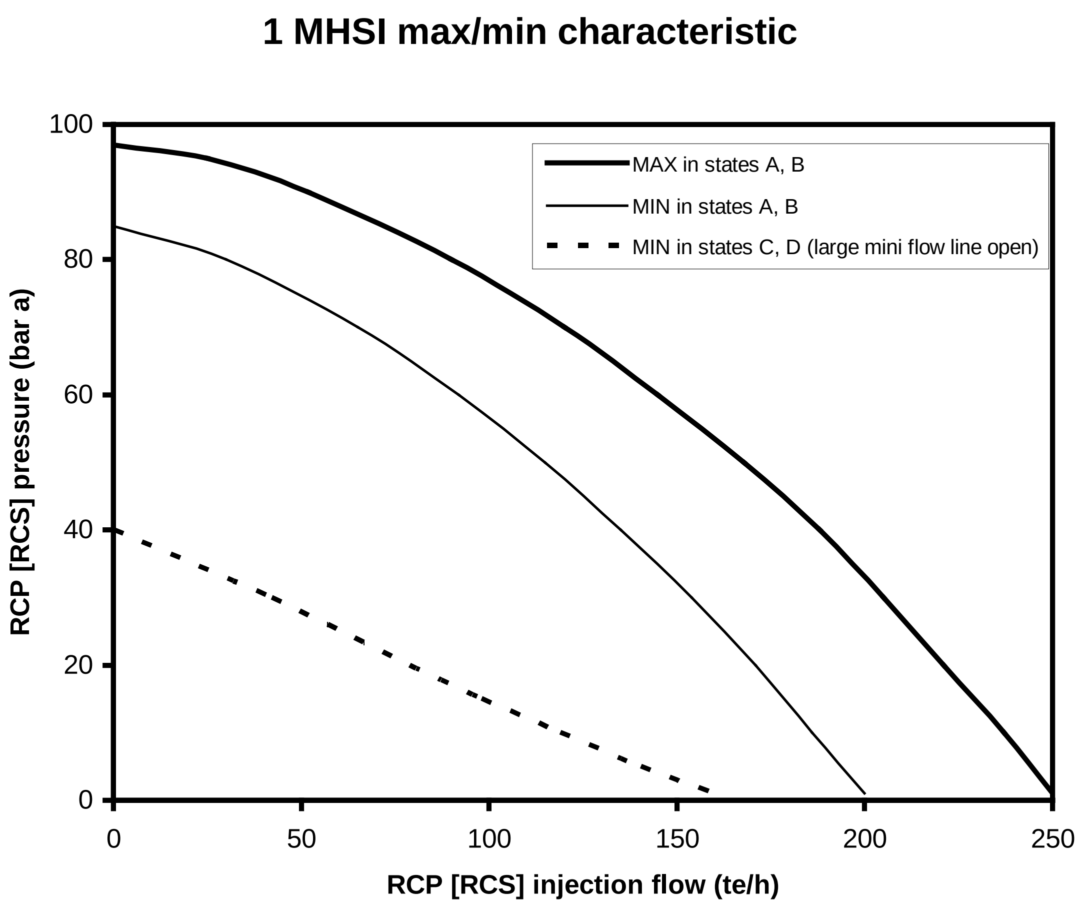
Safeguard action	Delay T_D (max)	Reason
RIS [SIS] actuation (MHSI & LHSI)	15 s w/o LOOP 40 s with LOOP	Pump start-up to full flow rate EDG reloading sequence
ASG [EFWS] actuation	15 s w/o LOOP 50 s with LOOP	Pump start-up to full flow rate EDG reloading sequence
ARE [MFWS] high load isolation	15 s	Valves closing delay
ARE [MFWS] low load isolation	15 s	Valves closing delay
ASG [EFWS] isolation	5 s	Valves closing delay
VIV [MSIV] closure	5 s	Valve closing delay
VDA [MSRT] opening	Dead time 1.5 s Opening time 0.5 s	MSRIV opening delay
VDA [MSRT] isolation	5 s	MSRIV closing delay
Cont^{t}/RCP [RCS] isolation	30 s	Valves closing delay
Reactor coolant pump trip	0.15 s	Breaker opening delay
Turbine trip	0.3 s	Turbine valves closing delay

Note: Except VDA [MSRT] opening, all the safeguard actions are modelled by step-wise opening or closing at time T_D (conservative approach).

SUB-CHAPTER 14.1 - TABLE 13

RIS [SIS] CHARACTERISTICS (MHSI) [REF-1]

- Number of MHSI trains	Four
- Location	Separated divisions
- Suction from	IRWST
- Injection into	RCP [RCS] cold leg
- MHSI injection temperature (min/max)	10°C/50°C (if no IRWST heating)
- MHSI injection flow rate (min/max)	See Figure, below



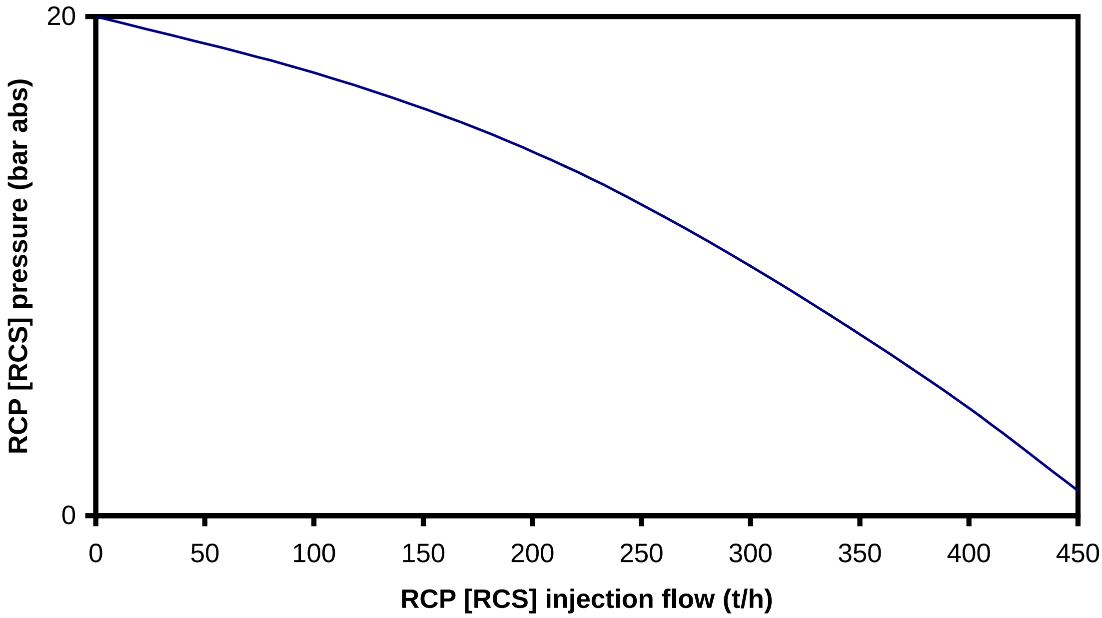
SUB-CHAPTER 14.1 - TABLE 14

RIS [SIS] CHARACTERISTICS (LHSI) [REF-1]

- Number of LHSI trains	Four
- Location	Separated divisions
- Suction from	IRWST
- Injection into	RCP [RCS] cold leg (in LOCA short term) RCP [RCS] hot leg (in LOCA long term)
- Heat exchanger (min)	1.1 MW/°C
- LHSI injection temperature (min/max)	10°C/50°C (if no IRWST heating)
- LHSI injection flow rate (min)	See Figure, below
- LHSI flow rate towards IRWST (min)	180 m^3/h at RCP [RCS] ≥ 20 bar

Note: LHSI flow rate in RHR-mode is 150 kg/s.

1 LHSI min characteristic



SUB-CHAPTER 14.1 - TABLE 15

RIS [SIS] CHARACTERISTICS (ACCUMULATORS) [REF-1]

- Number of accumulators	Four
- Location	Separated trains
- Injection into	RCP [RCS] cold leg
- Accumulator injection temperature (min/max)	10°C/50°C
- Accumulator initial pressure (min/max)	45/50 bar
- Accumulator initial water volume (min/max)	30/35 m^3
- Accumulator total volume	47 m^3
- Accumulator line flow resistance (min/max)	1700/2500 m^{-4}
- Accumulator boron concentration (min/max)	see IRWST boron concentration in Sub-chapter 14.1 - Table 16

SUB-CHAPTER 14.1 - TABLE 16

RIS [SIS] CHARACTERISTICS (IRWST) [REF-1]

- IRWST initial water volume (min/max)	1850 m^3 / 1940 m^3
- Maximum volume that could be trapped in the reactor building (excluding reactor pit) in course of LOCA	400 m^3
- IRWST initial water temperature (min/max)	10°C/50°C

- IRWST initial boron concentration (min):

	UO_2	MOX
• Natural boron	Min : 2450 ppm Max : 2750 ppm	Min : 2700 ppm Max : 3000 ppm
• Enriched boron[1]	Min : 1530 ppm Max : 1720 ppm	Min : 1600 ppm Max : 1780 ppm

[1] The conversion of the boron concentration (in ppm) from natural boron to enriched boron is given by :
- C (enriched boron) = 0.625 x C (natural boron) for UO_2,
- C (enriched boron) = 0.592 x C (natural boron) for MOX.

SUB-CHAPTER 14.1 - TABLE 17

ASG [EFWS] CHARACTERISTICS [REF-1]

- Number of ASG [EFWS] trains	Four (one per SG)
- Location	Separated divisions, with "passive" cross-headers
- Suction from	Dedicated ASG [EFWS]-tank, with "passive" cross-headers
- Injection into	Dedicated SG
- ASG [EFWS] injection temperature (min/max)	10°C/50°C
- ASG [EFWS] tank effective water volume (min)	400 m^3 per tank (division 2,3) 440 m^3 per tank (division 1,4)
- ASG [EFWS] injection flow rate (min/max)	See Table, below

SG pressure		ASG [EFWS] injection flow rate per train (min)	(max)
106.5 bar	(MSSV setpoint, with +1.5 bar uncertainty)	28 m^3/h	Not relevant
97 bar	(MSRV setpoint, with +1.5 bar uncertainty)	90 m^3/h	130 m^3/h [1]
1 bar		90 m^3/h	200 m^3/h [2]

[1] If the active flow limitation is OFF.
[2] If the active flow limitation is ON, the maximum EFWS flow rate is 120 m^3/h per train, irrespective of the SG pressure.

SUB-CHAPTER 14.1 - TABLE 18

VIV [MSIV] BYPASS [REF-1]

- Number of VIV [MSIV] bypass trains	Four (one per SG)
- Location	In parallel with the VIV [MSIV] (MSS warm-up line)
- Suction from	Upstream VIV [MSIV]
- Relief into	Downstream VIV [MSIV]
- Equipment per train	Two isolation valves in series
- VIV [MSIV] bypass relief flow rate (minimum)	4 kg/s saturated steam at SG = 5 bar (upstream MSIV-BP) and MSH = 1 bar (downstream MSIV-BP)

SUB-CHAPTER 14.1 - TABLE 19

VDA [MSRT] CHARACTERISTICS [REF-1]

- Number of VDA [MSRT], per SG	One
- Location - Suction from - Relief into	MS line, outside containment MS line, upstream VIV [MSIV] Environment
- Equipment, per train	1 isolation valve (MSRIV), initially closed, 1 control valve (MSRCV), initially open, in series
- VDA [MSRT] relief flow rate per train min / max	50% / 55% nominal steam flow 1150 t/h / 1270 t/h saturated steam under 100 bar
- VDA [MSRT] opening dynamics (max) refers to MSRIV, closed at standby MSRCV being fully open at standby	1.5 s dead time 0.5 s opening time
-Partial cooldown characteristics	From 95.5 bar to 60 bar, with a rate corresponding to -250°C/h

SUB-CHAPTER 14.1 - TABLE 20

STEAM GENERATOR SAFETY VALVES CHARACTERISTICS [REF-1]

- Number of MSSV trains, per SG	Two
- Location - Suction from - Relief into	MS line, outside containment MS line, upstream VIV [MSIV] Environment
- Equipment, per train	One spring-loaded safety valve
-SG safety valves relief flow rate per train minimum / maximum	25% / 27.5% nominal steam flow 575 / 635 te/h saturated steam under 100 bar
- SG safety valves opening dynamics (max)	3% accumulation

SUB-CHAPTER 14.1 - TABLE 21

RBS [EBS] CHARACTERISTICS [REF-1]

- Number of RBS [EBS] trains	Two
- Location	Separated trains
- Injection into	RCP [RCS] cold leg (between vessel and reactor coolant pumps) Each RBS [EBS] train injects into two cold legs, with one isolation valve per cold leg injection line
- RBS [EBS] tank water content (min) per train	36 m^3
- RBS [EBS] injection flow rate per train (min/max)	2.8 kg/s / 3.2 kg/s up to 180 bar RCP [RCS] pressure
- Injected boron concentration [1] (min)	Enriched boron : 7000 ppm Natural boron UO_2 : 11200 ppm Natural boron MOX : 11825 ppm

[1] The conversion of the boron concentration (in ppm) from natural boron to enriched boron is given by:
- C (enriched boron) = 0.625 x C (natural boron) for UO_2,
- C (enriched boron) = 0.592 x C (natural boron) for MOX.

SUB-CHAPTER 14.1 - TABLE 22

PRESSURISER SAFETY VALVES CHARACTERISTICS [REF-1]

- Number of PSV trains	3
- Location	Top of pressuriser
- Suction from	Pressuriser steam phase
- Relief into	Pressuriser relief tank (PRT), then Reactor Building
- PSV relief flow rate per train (min)	300 te/h sat. steam at 176 bar 450 te/h sat. liquid at 176 bar
- PSV opening / closing dynamics (max)	0.5 s dead time on opening 0.1 s opening time 5 s dead time on closing 1 s closing time

SUB-CHAPTER 14.1 - TABLE 23

NON-F1 SYSTEMS AND PRESSURISER NORMAL SPRAY (F1B) CHARACTERISTICS [REF-1]

"Partial trip" (PT)	
- Total rod drop time	3.5 s
- Integral reactivity worth	Depends on the I&C limitation function actuating it
- Reactivity worth versus time	Same as for RT
Pressuriser normal spray (3 stages)	
- Setpoints	156 bar / 158 bar / 160 bar
- Capacity per stage (min)	2x10 kg/s / 23 kg/s / 23 kg/s
- Capacity per line (min/max)	23 kg/s / 35 kg/s
- Opening time (max)	10 s / 2 s / 2 s
Pressuriser heaters	
- Total heating power	2592 kW
- Emergency power supplied heating	576 kW
GCT [MSB]	
- Steam relief capacity (min)	50% nominal steam flow (4600 te/h saturated steam under 75 bar)
ARE [MFWS]	
- Flow rate of low-load line after RT (min/max)	0% / 30% nominal ARE [MFWS] flow
- Closing time of high-load line after RT (min/max)	0 s / 15 s (step-wise)
AAD [SSS]	
- Injection rate (only one pump) (min/max)	350 m^3/h / 400 m^3/h
RCV [CVCS] [1]	
- Net RCP [RCS] injection flow rate, 1st step	
PZR level > reference + dead band	- 10 kg/s
PZR level < reference + dead band	+ 10 kg/s
- Net RCP [RCS] injection flow rate, 2nd step	
PZR level > 85% R (~ 60 m^3)	- 20 kg/s (injection isolation)
PZR level < 12% R (~ 10 m^3)	+ 20 kg/s (2 injection pumps)

[1] The simulation of the RCP [RCS] - inventory control as presented is simplified.

SUB-CHAPTER 14.1 - TABLE 24

SAFETY FUNCTIONS AND ASSOCIATED SYSTEMS

	C/R		C/I	C/T	C/P	C/C	
	Control of fuel integrity at power	*Control of core reactivity at shutdown*	*Control of RCP [RCS] Water inventory*	*Control of RCP [RCS] Water temperature*	*Control of RCP [RCS] pressure*	*Control of Containment*	
	dedicated RT	**dedicated RT dedicated isolation**	**Dedicated RT dedicated isolation** [1]	**dedicated RT dedicated isolation**	**dedicated RT dedicated isolation**	**dedicated RT dedicated isolation**	**F 1**
P							
		Short term:	**Reactor coolant pump trip**	Hot shutdown:	Overpressure protection:	Cont[t] heat removal:	**S**
	RCCA	**RCCA Boron-meter**		**ASG [EFWS]+VDA [MSRT]**	**RT+ SG and PZR valve**	**LHSI**	**Y**
C			SB-LOCA:				**S**
			MHSI+VDA [MSRT]			SGTR bypass:	**T**
					Depressurisation	**dedicated design of MHSI,VDA [MSRT], SG valve**	**E**
C		Long term:	IB/LB-LOCA:	Cold shutdown:	to cold shutdown:		**M**
		RBS [EBS]	**MHSI+Accumulator+LHSI**	**LHSI in SIS/RHR mode**	**PZR valves**	**BP MSIV, APG [SGBS]**	**S**
			SB-LOCA w/o MHSI:	Loss of ASG [EFWS]:	Loss of RT:	Loss of LHSI:	
			Accumulator+LHSI+VDA [MSRT]	**ARE [MFWS], AAD [SSS]**	**VDA [MSRT] + SG and PZR valve**	**EVU [CHRS]**	
R	Loss of RCCA:	Loss of RBS:	SB-LOCA without VDA [MSRT]	ARE [MFWS]+AAD [SSS]+ASG [EFWS] loss :	Loss of PSV:		**B**
	Core reactivity	**RCV [CVCS], RIS [SIS]**	**MHSI+GCT [MSB]**	**RIS [SIS]+RCV+PZR valve**	**PZR spray, PT/RT**		**A**
	Feedback,						**C**
R	**RCV [CVCS], RBS [EBS]**			Loss of VDA [MSRT] :	Loss of PSV	SGTR bypass:	**K**
				GCT [MSB]	for depressurisation:	**GCT [MSB]**	**-**
					PZR/ RCV [CVCS] spray		**U**
C				Loss of LHSI in SIS/RHR mode:	**RCV [CVCS] letdown**		**P**
				ASG [EFWS]+VDA [MSRT]			
				MHSI + EVU [CHRS]			

[1] RCP-seal integrity: **RRI** (seal cooling) **+ DEA / GMPP-seal return-lines isolation** (seal leak-tightness).

SUB-CHAPTER 14.1 - TABLE 25

PRINCIPLE APPROACH USED IN ACCIDENT ANALYSES WITH RESPECT TO DNB

Category of the transients		Aim of the transients analyses	Uncertainties consideration		DNBR decoupling criterion	DNBR protection setpoint	Initial DNBR value	site $DNBR_{RT}$	site $DNBR_{LCO}$	DNBR design limit
						(used for transient analyses, including uncertainties)		setpoints + uncertainties (U)		(used for transient analyses)
Transients actuating the low DNBR protection function **TYPE-1**		Demonstration of the efficiency of the low DNBR protection function for the largest initial domain: * time constant setting * definition of this protection function operating limits, e.g.: - max reactivity insertion rate - min. initial DNBR * accuracy	statistical (all of them grouped together, including the DNB predictor one)	All of them within the $DNBR_{RT}$ threshold	PCC-2 DNBR > DNBR design limit	1.	Not fixed (any value compatible with meeting the DNBR criterion)	1. *U	-	1.
Transients actuating a specific protection function	The DNB_{LCO} I&C function is used when the transient starts **TYPE-2**	Demonstration of meeting the DNBR criterion, when the transient starts from the $DNBR_{LCO}$		All of them within the $DNBR_{LCO}$ threshold	PCC-3 number of rods which experience DNB < 10%	Not considered	DNBR Limiting value	-	DNBR limiting value * U	1.
	The $DNBR_{LCO}$ I&C function is not used when the transient starts (power level=0) **TYPE-3**	Demonstration of meeting the DNBR criterion when the transient starts from the worst initial conditions within the operating domain bounded by LCOs.	deterministic	*part of them on initial conditions (e.g. on pressure, temperature, …) *part of them on the DNBR design limit (e.g. DNBR predictor, rod bow): Up: This part depends on the initial pressure		Not considered	Not significant	-	-	High pressures 1. *Up=1.21 Low pressures 1. * Up=1.12

SUB-CHAPTER 14.1 - FIGURE 1

SG DRAWING [REF-1]

{CCI Removed}

a

SUB-CHAPTER 14.1 - FIGURE 2

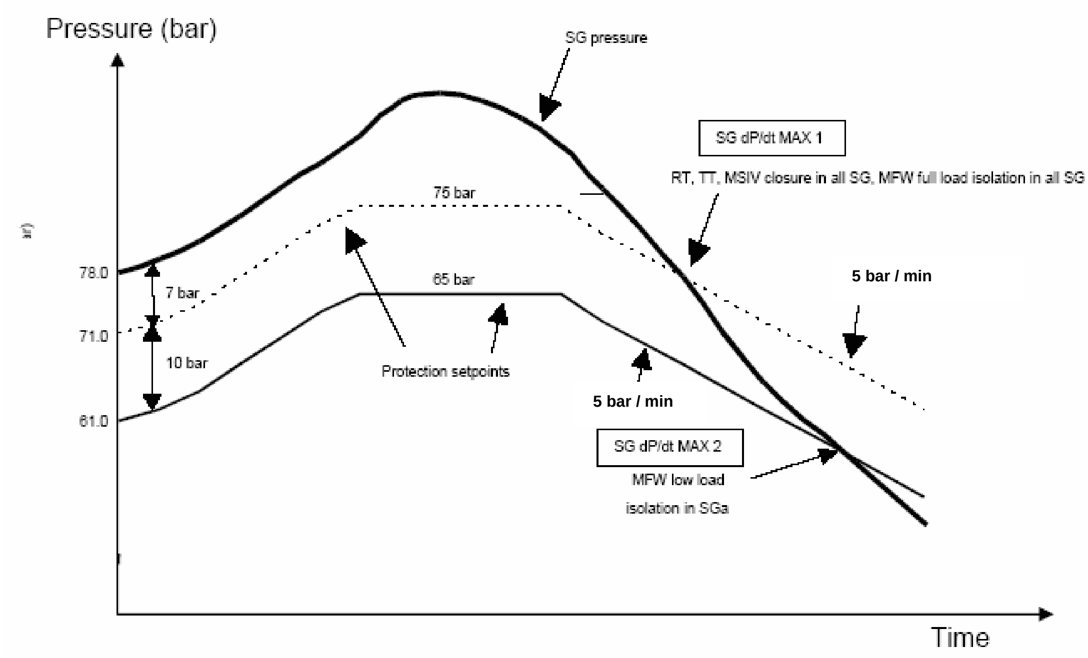
PRESSURISER DRAWING [REF-1]

{CCI Removed}

a

SUB-CHAPTER 14.1 - FIGURE 3

SG PRESSURE DROP PRINCIPLE



SIGNALS "SG pressure drop"

SG dP/dt MAX1 = actual SG pressure - 7 bar
decrease limited at 5 bar/min
maximum absolute value limited at 75 bar
⇨ RT, TT, ARE [MFWS] high-load line isolation in all SGs
⇨ VIV [MSIV] closure in all SGs

SG dP/dt MAX2 = actual SG pressure -17 bar
decrease limited at 5 bar/min
maximum absolute value limited at 65 bar
⇨ ARE [MFWS] low-load line isolation in SGa (SG related)

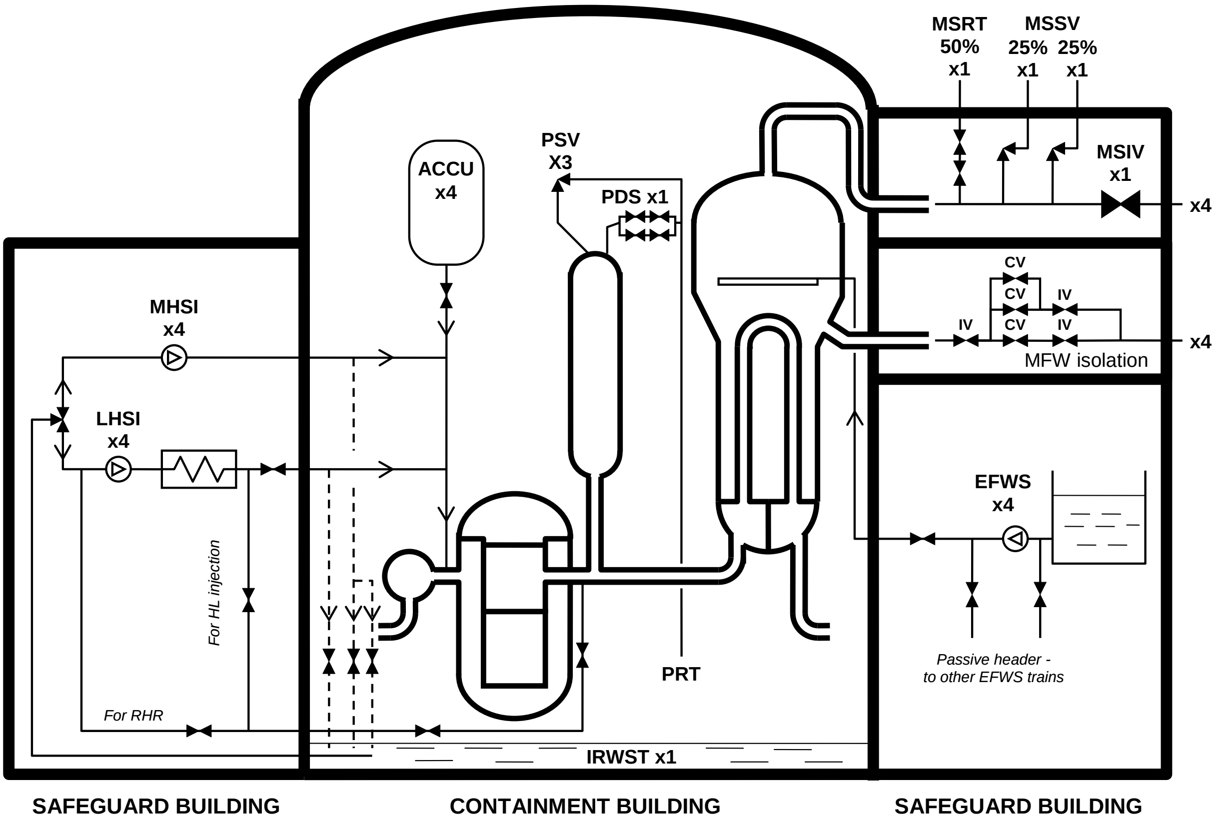
SIGNALS "SG pressure low"

SG pressure MIN1 = 50 bar
⇨ RT, TT, ARE [MFWS] high-load line isolation in all SGs
⇨ VIV [MSIV] closure in all SGs

SG pressure MIN2 = 40 bar
⇨ ARE [MFWS] low-load line isolation in SGa (SG related)
⇨ VDA [MSRT] isolation in SGa (SG related)

SUB-CHAPTER 14.1 - FIGURE 4

MAIN F1A FLUID SYSTEMS (SIMPLIFIED FUNCTIONAL SKETCH)



SUB-CHAPTER 14.1 – 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.

2. PLANT INITIAL CONDITIONS

[Ref-1] R. Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

[Ref-2] N. Goreaud. EPR – Lower Plenum Hydraulics Design of the Flow Distribution Device. EPRR DC 1651 Revision D. AREVA. December 2001. (E)

4. FISSION POWER AND DECAY HEAT AFTER REACTOR TRIP (RT)

[Ref-1] S. Ebalard. ORIGEN-S. Qualification manual. EPDS DC 150 Revision C. AREVA. November 2005. (E)

[Ref-2] S. Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

5. I&C SIGNALS

5.1. PRIMARY AND SECONDARY (P/S) I&C SIGNALS

5.1.2. Short description of specific F1A P/S I&C functions

ARE [MFWS] isolation

[Ref-1] R. Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

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[Ref-1] R. Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

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[Ref-1] S. Laurent. Neutronic data for transient analyses (update 4500 MWth). NFPSC DC 286 Revision B. AREVA. February 2006. (E)

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[Ref-1] C. Hove. Core Reactivity Control (update 4500 MWth). NFPSC DC 284 Revision B. AREVA. January 2006. (E)

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[Ref-1] S. Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

SUB-CHAPTER 14.1 - TABLE 7

[Ref-1] R. Gagner. RPV Thermal-hydraulic Design: Rod Drop Time. EPRR DC 1690 Revision C. AREVA. July 2005. (E)

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SUB-CHAPTER 14.1 - TABLE 9 (1/7 TO 6/7)

[Ref-1] R. Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

SUB-CHAPTER 14.1 - TABLES 11 TO 23

[Ref-1] R. Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

SUB-CHAPTER 14.1 – FIGURE 1

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

SUB-CHAPTER 14.1 - FIGURE 2

[Ref-1] R. Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)