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SUB-CHAPTER 16.3 – PRACTICALLY ELIMINATED SITUATIONS

0. INTRODUCTION

It is recalled that in the EPR context, “Practical Elimination” refers to the implementation of specific design measures to reduce the risk of a large early release of radioactive material to the environment to an insignificant level. To achieve practical elimination, each type of accident sequence that could lead to a large early release of radioactivity is examined and addressed by design measures. Demonstration of practical elimination of an accident sequence may involve deterministic and/or probabilistic considerations, and must take into account uncertainties due to the limited knowledge of physical phenomena involved in severe accident analysis.

Conditions covered by specific treatment leading to their practical elimination are those which are liable to give rise to significant early releases; mainly high pressure core meltdown sequences. The sequences particularly considered using this approach are detailed as follows.

1. SITUATIONS RELATED TO SEVERE ACCIDENTS

As already outlined in section 1 of Sub-chapter 16.2 of the PCSR, the EPR pursues a two-staged approach to the control of severe accidents, i.e.

i. practical elimination of highly energetic phenomena which have the potential to breach the containment early in an accident and thus result in large early releases. Practical elimination is, therefore, also a precursor condition for the second stage,

ii. maintaining the containment integrity in the long-term.

The purpose of this chapter is to emphasise the importance of practical elimination of these phenomena to ensure the fulfilment of the safety objectives for severe accidents, see section 0 of Sub-chapter 16.2. More specifically, sections 1.1 to 1.3 below compare the relevant phenomena against the corresponding safety objectives set down in the Technical Guidelines and summarise the particular analyses of Sub-chapter 16.2 that verify the fulfilment of these objectives.

1.1. HIGH PRESSURE CORE MELT ACCIDENT AND DIRECT CONTAINMENT HEATING (DCH)

1.1.1. Safety objectives

The following safety requirements have been taken from section E 2.2.1 of the Technical Guidelines.

A design objective is to convert high pressure core melt sequences into low pressure sequences with a high reliability so that high pressure core melt situations can be "practically eliminated".

This objective implies the need to limit the pressure in the reactor coolant system in the range of 15 to 20 bar by the time of reactor pressure vessel (RPV) rupture. This objective can be ensured by installing two dedicated Severe Accident Depressurisation Valves (SADVs) with isolation valves to the depressurisation system of the pressuriser. The discharge capacity of the dedicated valves is determined by considering the following situations using best-estimate assumptions:

- loss of off-site power with unavailability of all diesel generators;
- loss of off-site power with unavailability of all diesel generators but with recovery of water supply during core melting;
- total loss of feedwater combined with the failure of primary feed and bleed[1].

Sensitivity studies regarding the discharge capacity of the SADVs, hot gas temperatures and the opening criteria of the SADVs are required to consider delayed bleeding of the pressuriser and late reflooding of the core as well as the uncertainties in the modelling of these phenomena. These sensitivity studies will consider the possibility of human error during the course of the accident.

Moreover, design measures have to be implemented to limit the dispersal of corium into the containment atmosphere in the event of a reactor pressure vessel melt-through in order to prevent direct containment heating (DCH). These design measures are related to the reactor pit and its ventilation as well as to the ex-core neutron measurements, to ensure that large quantities of corium released from the reactor pressure vessel cannot be carried out of the reactor pit.

1.1.2. Studies and conclusion

The UK EPR is equipped with dedicated severe accident depressurisation valve trains that are part of the primary depressurisation system (PDS). The designed bleed capacity amounts to 900 t/hr at 176 bar. Analyses assessing the performance of the PDS in severe accidents, see sub-section 2.2 of Sub-chapter 16.2, have shown that for

- *representative* core melt scenarios the capacity of the dedicated depressurisation lines is such that a low primary pressure (~ 5 bars abs.) will be reached well before reactor vessel failure. This pressure level is well below the safety objectives as defined in the Technical Guidelines and prevents, in combination with the tortuous path for the melt to enter the containment, see section 6 of Sub-chapter 6.2, any significant melt dispersal and subsequent direct containment heating.
- *bounding* core melt scenarios involving late in-vessel reflooding, the increase in the primary pressure induced by vaporisation of the injected water falls rapidly because of the high discharge capacity. At vessel failure, the pressure of the primary system is a maximum of 20 bars. The result is that the mechanical impacts on the reactor vessel and the reactor pit walls are negligible, as are the effects due to melt dispersal.

The dedicated primary depressurisation valves ensure a pressure close to 5 bar in the primary system at the time of RPV failure for representative scenarios and limits the pressure to 20 bar for bounding scenarios. RPV failure at such pressures has a negligible impact on the mechanical integrity of the vessel supports.

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[1] It is assumed that the pressuriser valves are not available; the dedicated valves and their isolation valves remain available.

In terms of melt dispersal, a failure pressure of 20 bar is deemed to be sufficiently low to preserve integrity of the containment even for wide reactor pits with large and direct flow paths to the containment.

In the EPR the geometry of the reactor pit is designed with only small openings to the loop compartments. There is no direct passage to the upper volume of the containment. Therefore, this design provides a tortuous path for the melt to enter the containment and hence promotes the retention of the corium in the reactor pit. In effect, the design of the EPR reactor pit further limits melt dispersal and consequential effects relative to PWRs with large open containments.

In conclusion, with the implementation of the dedicated engineered RPV safety features, it is deemed that the risk of high pressure core melt scenarios with subsequent vessel failure at high pressure, direct containment heating and missile formation can be considered as "practically eliminated”.

1.2. STEAM EXPLOSIONS LEADING TO FAILURE OF THE CONTAINMENT

1.2.1. Safety objectives

The following safety requirements have been taken from section E 2.2.3 of the Technical Guidelines.

1.2.1.1. In-vessel phenomena

A release involving high mechanical energy would be necessary to threaten the reactor pressure vessel and the containment; nevertheless, the design must address the potential for in-vessel steam explosions linked to core melt. Due attention has to be paid to:

- the justification of the maximum mass of material that might be involved, taking into account the specific design of the lower core support plate and the uncertainties related to core relocation and behaviour in the vessel lower head. In this context, reflooding scenarios have to be precisely assessed;

- the application of experimental results[2] to the specific design of the nuclear power plant;

- the extent of the temperature rise of the vessel upper internal structures and head during core melt sequences, and their consequences;

- the behaviour of the reactor coolant system (including the steam generators) in the event of an energetic water slug being ejected through the downcomer as a result of an energetic in-vessel melt/water interaction.

1.2.1.2. Ex-vessel phenomena

The amount of water which could be present in the reactor pit and in the spreading compartment at the time of the reactor vessel melt-through has to be limited by the design. The possibility of a large steam explosion during corium flooding must be prevented and loads resulting from melt-water interaction must be taken into account in the design.

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[2] Including results from the BERDA facility (see section 1.3 of Sub-chapter 16.2)

1.2.2. Studies and conclusion

Section 1.3.1 of Sub-chapter 16.2 revisits the findings of several studies, such as that performed by the Steam Explosion Review Group (SERG) and concludes that the failure of the containment by an in-vessel steam explosion is highly unlikely. This conclusion is supported by relevant experiments, e.g. BERDA, which show that the mechanical energy of a melt slug formed by an in-vessel steam explosion is insufficient to breach the upper head of the RPV upon slug impact.

Ex-vessel steam explosions are practically eliminated by avoiding the ingress and presence of water in the reactor pit before melt release from the RPV and in the core catcher before melt spreading commences. Water ingress into the reactor pit is prevented by restricting and shielding the entry cross-sections around the Main Coolant Lines.

In addition, induced failure of the hot leg of the reactor coolant system, which would otherwise be a potential precursor for water ingress into the pit, is prevented by primary system depressurisation.

The spreading compartment that houses the core catcher is a dead-end compartment and in combination with the remote location, is only very limitedly affected by convection and the corresponding transport of sprays, steam and humid air in the containment. Flooding of the free, upper surface of the spread core melt at a controlled rate has been experimentally shown not to result in steam explosion.

In summary, both in- and ex-vessel steam explosions do not constitute a viable threat for the containment and thus failure of the containment by these phenomena is considered to be practically eliminated.

1.3. HYDROGEN COMBUSTION PROCESSES ENDANGERING CONTAINMENT INTEGRITY

1.3.1. Safety objectives

The following safety requirements have been taken from section E 2.2.4 of the Technical Guidelines.

The possibility of local high concentrations of hydrogen must be prevented as far as achievable by the design of the containment internal structures. When it is not possible to demonstrate that the hydrogen local concentration remains below 10%, specific criteria[3] could be used (insofar as they are fully justified and validated) to demonstrate that deflagration to detonation transitions and fast deflagrations do not occur. Otherwise, adequate provisions need to be implemented such as reinforced walls for corresponding compartments and for the containment.

A systematic and deterministic approach must be undertaken by the designer to select relevant scenarios in terms of hydrogen release rates, taking into account mitigation measures. Further, it has to be demonstrated that the selected scenarios are bounding.

Mitigation using re-combiners only (i.e. igniters not being used) with direct discharge of the primary circuit into the containment is acceptable in principle and has the potential to fulfil the safety objectives mentioned above.

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[3] Such as the σ criterion (see section 2.3 of Sub-chapter 16.2)

Discharge via a large pressuriser relief tank with two discharge pipes equipped with rupture disks discharging to two reactor coolant pump compartments is also acceptable. However, this arrangement must be optimised and the methodology and the analysis tools used to demonstrate its effectiveness have to be fully justified and validated.

However, it is underlined that significant uncertainties exist concerning the production of hydrogen during severe accident sequences. These uncertainties are intimately linked to such phenomena as late flooding of a partially damaged core at high temperature, slumping of molten core material into residual water in the lower head of the RPV and interactions between corium and sacrificial materials. These uncertainties require investigation with various codes and models.

Notably, scenarios with passive or active reflooding as well as scenarios characterised by multiple release locations have to be addressed in the demonstration of the effectiveness and robustness of the hydrogen risk mitigation concept.

It is emphasised that the consequences of the decrease in steam partial pressure following actuation of the containment heat removal system on the flammability of the hydrogen mixture must be carefully assessed. Various start-up times of the EVU [CHRS] system must also be considered.

1.3.2. Studies and conclusion

The engineered approach to the swift reduction of high local hydrogen concentrations is based on good mixing of the containment atmosphere. Mixing is achieved by establishing global convection in the atmosphere of both the accessible and inaccessible rooms of the containment. This is achieved by transforming the two-zone containment into an one-zone containment with the hydrogen distribution and mixing system (CONVECT) of the Combustible Gas Control System (ETY [CGCS]), see section 4 of Sub-chapter 6.2, in combination with the direct discharge of the RCP [RCS] coolant inventory into the lower compartments of the containment, and with the strategic arrangement of the re-combiners of the ETY [CGCS] system, which fosters convection.

Given the reliable transformation of the two-zone containment into a one-zone configuration, the analysis results presented in section 2.2.3 of Sub-chapter 16.2 show that the atmospheric mixing is highly efficient and thus, hydrogen concentrations prone to fast combustion processes are limited to the vicinity of the hydrogen release locations, notably the steam generator compartments. Assessment of the corresponding combustion processes revealed that dynamic pressure loads on the containment shell are moderate and thus are bounded by the static Adiabatic Isochoric Compete Combustion (AICC) pressure. For representative scenarios this pressure is predicted to remain well below 5.5 bar and below 6.3 bar for bounding scenarios.

For any of the analysed scenarios including those with reflooding, the global hydrogen concentration in the containment atmosphere remains below 10 vol%.

The actuation of the EVU [CHRS] spray results in an increase in hydrogen concentration due to condensation of steam. However, this adverse effect is more than compensated for by enhanced mixing of the containment atmosphere and by hydrogen depletion by the re-combiners.

In summary, the analysis results demonstrate the robustness of the hydrogen control system with respect to the particular risks associated with fast combustion processes. Therefore, these risks are considered as “practically eliminated”.

2. RAPID REACTIVITY INSERTION

The safety case for heterogeneous boron dilution faults is presented in section 7 of Sub-chapter 16.4.

3. CONTAINMENT BYPASS

3.1. SAFETY OBJECTIVES

The following safety requirements are stated in section E 2.2.5 of the Technical Guidelines.

Accident sequences with core damage involving containment bypassing have to be ”practically eliminated” by design provisions aimed at ensuring reliable isolation and preventing failures.

In order to ”practically eliminate” core damage with containment bypass due to a significant leak through the two isolation check valves, the designer must justify the capability of the motor operated isolation valves located outside the containment on the safety injection lines to prevent reverse flow (including two-phase flow). The pipe sections of the RIS/RRA [SIS/RHRS] outside the containment up to and including the motor-operated isolation valves must be designed so that their integrity would be maintained under maximum primary coolant pressure and temperature conditions.

Adequate provisions must be implemented to guarantee the integrity of the affected parts of the Safety Injection System (RIS [SIS]) outside the containment in the event of leakage through the check-valves located on the LHSI suction line from the IRWST and on the Medium-Head Safety Injection (MHSI) line inside the containment. Stringent design requirements must be implemented on the parts of the RIS/RRA [SIS/RHRS] outside the containment so as to prevent large breaks from occurring. Additionally, the closure capability of the isolating valves must be proven for all break sizes (up to a guillotine break), including two phase flow[1]. For possible breaks inside the reactor coolant pump thermal barriers and inside the Chemical and Volume Control System (RCV [CVCS]) high pressure cooler, the designer must justify the maximum break size taken into account as well as the provisions implemented for the detection and isolation of such a break, again including two phase flow conditions.

For possible breaks inside the RIS/RRA [SIS/RHRS] heat exchangers, the designer also needs to justify the maximum break size considered and the consequences (in terms of pressure and temperature) of such breaks on the Component Cooling Water System (RRI [CCWS]) circuits must be evaluated.

For core damage accident sequences which could occur during shutdown states with the containment building open, the designer has to specify the different phases of shutdown states for which an open containment is permitted. It must be demonstrated that, for representative accident sequences, the containment building would be reliably closed before a significant radioactive release could occur inside the containment. This requirement primarily concerns the containment hatch.

1 Note that the guillotine break of the largest pipe is a reference accident (PCC-4).

For core damage accident situations with a significant leak in the steam generator tubes (up to multiple steam generator tube rupture), the following situations must be investigated:

- single or multiple steam generator tube rupture with loss of the protection systems normally intended to deal with such a rupture;
- single or multiple steam generator tube rupture with the failure to close of the corresponding main steam isolation valve;
- steam line rupture with leaks in the associated steam generator tubes;
- spurious opening of a secondary safety valve coincident with a leak in the associated steam generator tubes.

As core damage sequences with consequential steam generator tube failures have to be “practically eliminated”, scenarios leading to natural circulation flow through the primary loops and the steam generators must also to be investigated in detail using suitably validated codes.

3.2. STUDIES

Core damage accidents with containment bypass cover all core damage accidents which lead to direct contact between the primary fluid and the external environment (whether the bypass occurs before or after core damage):

- containment bypass accidents induced by single initiating events of the “LOCA leading to containment bypass” type (bypass via the systems connected to the Reactor Coolant System (RCP [RCS]));
- containment bypass accidents induced by accident sequences (multiple failures) leading to core damage;
- containment bypass accidents leading to a core damage accident.

Breaks outside the containment which may be compensated for by the RCV [CVCS] charging flow are not dealt with in the assessment. Indeed, several levels of defence can compensate for this small leak (RCV [CVCS] initially and then RIS [SIS]) which gives times to the operator to detect, locate and cancel the leak.

More precisely, the transient situations studied are the following:

1 - BYPASS SITUATION INDUCED BY SINGLE INITIATING EVENTS

This refers to “LOCAs leading to containment bypass”

- LOCA in the RIS/RRA [SIS/RHRS] outside containment
- LOCA leading to containment bypass via the MHSI
- LOCA leading to containment bypass via the RBS [EBS]
- LOCA leading to containment bypass via the RCV [CVCS]

- LOCA leading to containment bypass via the REN [NSS]
- LOCA leading to containment bypass via the RPE [NVDS]
- LOCA leading to containment bypass via the RRI [CCWS]
- LOCA leading to containment bypass via the EVU [CHRS]

2 - BYPASS INDUCED BY ACCIDENT SEQUENCES

- SGTR associated with a steam line break
- SGTR with a safety valve VVP [MSSS] or a safety relief valve VDA [MSRT] failed in the open position
- Risk of bypass via the RIS/RRA [SIS/RHRS] lines in accident situations

3 - BYPASS INDUCED BY SEVERE ACCIDENT SEQUENCES

- SGTR induced by core damage
- Breaks on lines connected to the primary system or on the thermal barrier of a RCP [RCS] pump

3.2.1. Bypass Situations Induced by Initiating Events

The analysis of potential initiating events induced by a LOCA in the RIS/RRA [SIS/RHRS] or by a LOCA leading to containment bypass via the MHSI, the RBS [EBS], the RCV [CVCS], the REN [NSS], the RPE [NVDS], the pressuriser relief tank (PRT) or the RRI [CCWS] (via the thermal barriers of the primary motor-driven pumps) is presented in section 5.2 of Sub-chapter 15.1. The study addresses the dependence between the levels of defence (components and operator actions). In Sub-chapter 15.2 potential bypass of the containment caused by aircraft crash is analysed.

This analysis is based on a functional analysis and quantification of the frequency of potential initiating events and evaluates the risk of bypass for each of the selected scenarios while taking account of the design features adopted in relevant systems. The corresponding overall risk is low: 4.8E-09/ reactor year.

The risk of bypass induced by an SGTR is also analysed. Two design features adopted on the EPR are of particular importance: the design of the MHSI (discharge pressure below the set pressure of the relief valves on the secondary side); and automatic shutdown of the RCV [CVCS] charging pumps if "high water level in the secondary side of the steam generators and partial cooldown finished” is detected. These features lead to a reduction in the risk of SG water overspill. The quantification of the risk associated with these accident sequences is given in Chapter 15.

3.2.2. Bypasses induced by accident sequences

The risk of bypass due to "SGTR induced by another initiating event" is also analysed. For both sequences "SGTR with a steam line break" and "SGTR with a Main Steam Safety Valve (VVP [MSSS]) or a Main Steam Relief Train VDA [MSRT] failed in the open position", the risk of core damage is low (assessed at around 4.3E-09/reactor year).

Finally, in accident situations where mitigation requires the use of the RIS [SIS] and EVU [CHRS] systems, given the evaluation presented in section 5.2 of Sub-chapter 15.1, it is deemed that the risk of bypass via these systems can be considered as “practically eliminated”.

3.2.3. Bypasses induced by severe accident sequences

Containment bypass due to SGTR resulting from severe accident sequences has also been studied (Sub-chapter 15.4). These situations are considered as highly unlikely because of the high reliability of primary depressurisation via the dedicated discharge line.

As regards core damage accident situations in shutdown mode with the containment open, it has been shown that either their frequencies were very low or that it is possible to take advantage of the period of time preceding significant discharge into the reactor building to re-close the equipment access hatch and the personnel air locks in a reliable manner.

3.3. CONCLUSION

Given the arguments presented above based on design and operating features and the associated probabilistic results, it is deemed that the risk of bypass induced by initiating events and by accident sequences with or without core damage, can be considered as “practically eliminated”.

4. FUEL DAMAGE IN THE SPENT FUEL POOL

4.1. SAFETY OBJECTIVES

The following safety requirements have been taken from section E 2.2.6 of the Technical Guidelines.

Noting that the fuel pool is not located in the containment building, it has to be demonstrated that spent fuel damage conditions in the pool are "practically eliminated”.

The designer must provide justifications for this “practical elimination”, including the results of probabilistic safety studies.

The designer must make provisions to avoid the total loss of the cooling system of the spent fuel pool and to avoid pool draining. The likelihood of boiling occurring in the spent fuel pool should be reduced by appropriate design solutions, particularly of the support systems of the pool cooling system.

4.2. STUDIES

The spent fuel pool water cooling system (PTR [FPCS]) removes residual heat from the spent fuel assemblies stored in the pool. The PTR [FPCS] system consists of:

- Two identical principal trains, each equipped with two pumps and a heat exchanger,
- A third train equipped with one pump and one heat exchanger diverse from the principal trains; the heat exchanger is cooled by a completely separate cooling system.

The third cooling train has been specially designed to meet the two following objectives:

- To significantly reduce the risk of boiling in the spent fuel pool,
- To help in the practical elimination of the risk of fuel damage in the spent fuel pool.

The design of the PTR [FPCS] system with two pumps per train enhances the availability of the PTR [FPCS], thus reducing the probability of boiling occurring in the pool. In addition, the third PTR [FPCS] train is completely independent with respect to electrical supply and cooling system. Following a LOOP, the third train can be supplied by either the Emergency Diesel Generators (EDGs) or the Station Blackout (SBO) Diesel Generators. This independence provides a reduction of the risk of water boiling in the event of the loss of the two principal trains.

To reduce the risk of accidental draining of the spent fuel pool, design features such as the automatic isolation of the lines connected to the bottom of the pool have been implemented.

In order to study the design of the PTR [FPCS] in probabilistic terms, the following events have been considered:

- Incidents or accidents affecting the principal PTR [FPCS] trains and/or their support systems.

- Loss of the PTR [FPCS], corresponding to the simultaneous unavailability of the two principal trains, with a risk of losing the third train.

- Degradation of the fuel assemblies in the spent fuel pool because of water depletion resulting either from evaporation due to the total loss of cooling or following accidental draining of the pond.

The probabilistic assessments shown in sections 3 and 4 of Sub-chapter 15.3 confirm that the design allows:

- The risk of boiling occurring in the spent fuel pool to be significantly reduced.

- The risk of fuel assembly damage following a total loss of cooling or an accidental draining of the spent fuel pool to be considered as "practically eliminated". The risk of damage is evaluated at 2.54E-10/reactor year for loss of cooling and 2.3E-09/reactor year for accidental draining.

The UK EPR safety case has been revised to include non-isolable gross failures of piping penetrations in the Spent Fuel Pool and connected compartments (presented in section 8 of PSCR Sub-chapter 16.4). The current PSA model in Sub-chapter 15.3 has not been updated to take into account these new pool draining events since, taking account of their low frequency of occurrence and safeguards systems available to protect against them, the fuel damage frequency contribution due to the new pool draining events is expected to be similar to the fuel damage frequency due to other draining events that is currently calculated. These events will be incorporated into the PSA (Chapter 15) during the site licensing phase, which will also consider the risk reduction measures proposed for the UK EPR design.

4.3. CONCLUSION

On the basis of the design features for the mitigation of accidents likely to affect the spent fuel pool (loss of cooling and accidental draining) and the probabilistic assessments described above, it is deemed that the risk of fuel assembly damage in the spent fuel pool can be considered as "practically eliminated".