# Main Steam Line Break Analysis with Failure of all Main Steam Isolation Valves in a BWR (Peach Bottom)

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## ABSTRACT

After the Fukushima accident, beyond design basis accidents (BDBA), even though their frequency is predicted by probabilistic safety assessment (PSA) to be low, have moved to the center of attention. Since then an increased number of analyses on the long term station blackout accidents have been performed. The present paper highlights selected aspects of another accident which, even if its frequency to occur is very low, may be severe in its consequences - a main steam line break with failure of all main steam isolation valves in the affected steam line in a Boiling-Water Reactor (BWR).

Following the accident the nuclear power plant (NPP) quickly loses coolant inventory directly to the environment (containment bypass). Without proper operator actions, core heat up and core damage may occur at an early stage in the accident. A plant input model which has been developed at the University of Pisa has been used to run three cases, using the best estimate thermal hydraulic system code Relap5 mod 3.3. The first case assumes no operator actions and availability of passive systems only. It is used as baseline to be compared to the two other cases. The second case assumes availability of active and passive systems, and plausible operator behavior following symptom based emergency operating procedures (EOP), considering plant states. The third case presents an optimized strategy, assuming that the operator was able to clearly identify the root cause of the accident.

The analysis confirms that, without proper operator actions, the accident can quickly progress to a severe accident (the situation is aggravated by the containment bypass). It shows further that procedures based on plant states, if this special case has not been considered, may not lead to desirable results. Finally it can be seen that, adopting a suitable strategy, the grace period can be extended up to a point where additional sources of coolant can be made available.

## KEYWORDS

Thermal hydraulics, Accident Management, Safety Analysis, Relap5

# 1. INTRODUCTION

The border between design basis and beyond design basis accidents (BDBA), in general, is drawn using a probability cut-off criterion for the initiating event. Therefore, BDBA sequences are not likely to occur. However, analysis of BDBA, which may have severe consequences and are still plausible, can help in planning procedures, and may contribute to reduce the residual risk of nuclear power plants (NPPs). This paper deals with selected aspects of a postulated main steam line break (MSLB) with failure of the main steam isolation valves (MSIV) in the affected main steam line at the Peach Bottom BWR. This, although of low predicted frequency, could result in severe consequences in respect to plant stability on one hand and radiological releases on the other. Loss of coolant accidents (LOCA) outside of the containment are usually covered in every Final Safety Analysis Report, but the functioning of at least one MSIV is generally (failure of a single MSIV to close would be the single active failure typically assumed in deterministic safety analyses).

Accident management relies on deterministic safety analysis, proper scoping of possible accident sequences and operator countermeasures. Aim of the present analysis is to draw attention to a potentially dangerous accident sequence, and to propose an accident management strategy for the situation. The paper focuses on mitigative accident management (up to core damage, or impending core damage). Therefore the best estimate thermal hydraulic system code Relap5 (US NRC version mod 3.3) has been chosen. Two outcomes of the analysis are possible; either Relap5 predicts peak cladding temperatures of more than 1200 deg C indicating that the core is damaged and beyond repair – the present analysis stops at this point; or the analysis shows that the plant can be led to a long term coolable configuration.

The paper presents three analyses. All cases assume double ended guillotine break (DEGB) of the main steam line (MSL) "A" outside the containment as initiating event. It is further assumed that both MSIV on the affected MSL will fail to close.

- Analysis a) (baseline scenario) assumes that only passive ECC components are available turbine driven pumps).
- Analysis b) assumes that all safety systems are functional, and that the operator classifies the scenario as LOCA (inside containment), based on plant symptoms
- Analysis c) assumes that all safety systems are functional, but that the operator is able to identify the root cause of the accident, and tries to minimize the loss of coolant to the environment by using a normal operation mode of the RHR system.

Peach Bottom, as reference BWR/4 reactor, is taken as example unit for the analysis.

## 1.1. Peach Bottom – Unit 2

Unit 2 of Peach Bottom Atomic Power Station is a General Electric BWR/4 with a Mark-I containment. The currently licensed thermal power of the reactor is 3514 MWth, uprated from 3458 MWth in 2002. The BWR/4 is fitted with 4 Main Steam Lines (MSL), allowing a total steam flow of 1880 kg/s at a nominal operating pressure of 7.1 MPa. Each MSL has two main steam isolation valves, one on the inside and one on the outside of the containment. Furthermore flow restrictors limit the steam flow of every MSL to about 200% of its standard throughput. The third engineered safety features along the MSL are the Safety Relief Valves (SRV), which open at a pressure above 7.81 MPa.

For the considered accident scenario water supply and water injection are the most safety relevant systems. The plant contains two main water sources for residual heat removal in case of normal shutdown or emergency core cooling (ECC). The condensate storage tank (CST) is located outside of the reactor building and provides 756 m³ of water for cooling. The main source of water in case of an emergency is the freestanding steel drywell torus suppression pool, located below the reactor. It is connected to all ECC Systems and contains up to 3605 m³ of water. Peach Bottom is fitted with two injection systems operating at high pressure as well as two systems injecting at lower pressure. The main characteristics and setpoints of these systems are summarized in Tables 1 to 3. The systems themselves are described in more detail in the following subsections.

**Table 1. Pump characteristics of the high pressure core cooling systems**

| | No. of pumps | Flow rate [kg/s] | pressure range [MPa] |
|---|---|---|---|
| **HPCI** | 1 | 315 | 1.034 - 6.895 |
| **RCIC** | 1 | 37.8 | 1.034- 6.895 |

**Table 2. Pump characteristics of the low pressure core cooling systems**

| | No. of pumps | Flow rate [kg/s] | @ pressure [MPa] | Pump shutoff [MPa] |
|---|---|---|---|---|
| **Core Spray** | 4 | 197 | 0.724 / | 1.99 |
| **RHR (LPCI)** | 4 | 687.5 | 0.138 | 2.034 |

**Table 3. Setpoints of the core cooling systems**

| Setpoint | Action | Level* |
|---|---|---|
| **Switch off** | Close RCIC Steam Supply Valve<br>Trip HPCI Turbine | + 1.435 m |
| **Switch on** | Initiate RCIC<br>Initiate HPCI | - 0.965 m |
| **Switch on** | Initiate RHR (LPCI Mode)<br>Initiate CS | - 3.366 m |

* measured from the instrumentation zero-level

### 1.1.1. Reactor core isolation cooling (RCIC) system

The RCIC system is a coolant makeup system operating at high-pressure. Its purpose is to support safe shutdown by providing core cooling when the MSLs are isolated. The capacity of the system is designed to compensate the initial decay heat boil-off until the residual heat removal system can be put into operation. Primary water source of the RCIC is the condensate storage tank. The torus suppression pool serves as an alternative source of water. The water is injected via the feedwater line by a steam turbine driven high pressure pump, delivering at least 37.8 kg/s in a pressure range from about 1 to 7 MPa. The steam supply of the turbine originates from the “C” main steam line upstream of the inboard MSIV. Two different automatic setpoints are implemented for the RCIC system. An initiation point at a reactor water level of - 0.965 m (measured from the instrumentation zero-level) and a close signal for the steam supply valve of the RCIC at + 1.435 m, to avoid reactor overfilling, which would result in liquid flow to the steam lines.

### 1.1.2. High pressure coolant injection (HPCI) system

The HPCI system is part of the emergency core cooling systems and is designed to provide water at high pressure in case of small break LOCAs and help depressurize the RPV in case of medium break LOCAs, so the low pressure emergency core cooling systems can be put in operation. The HPCI design is similar to the RCIC, but the capacity is about 10 times larger, with a steam driven pump delivering at least 315 kg/s in a pressure range from 1 to 7 Mpa. The pump discharge is via the “A” feedwater line, while steam supply for the turbine comes from the "B" main steam line upstream of the inboard MSIV. Water sources, as well as the setpoints are the same as for the RCIC. Therefore this high pressure system can also serve as a backup of the reactor core isolation cooling.

### 1.1.3 Core spray (CS) system

The purpose of the core spray system is to provide spray cooling to the reactor in case of a large break LOCA. It consists of two independent subsystems, each with two motor driven pumps, which deliver 197 kg/s at 0.724 MPa each. The system automatically starts at one of two signals indicating a loss of coolant, level 1 reactor vessel water level (-3.366 m) or high drywell pressure (+0.0117 MPa). Water source for the CS is the torus suppression pool. The injection starts when the pressure drops below the pump shutoff head of 2 MPa and the water is sprayed on top of the core. There is no automated signal to stop injection if the pressure is lower than the pump shutoff pressure.

### 1.1.4. Residual heat removal system (RHR)

The residual heat removal system is a multipurpose system, which is capable of operating in seven different modes.

- low-pressure coolant injection
- containment spray
- suppression pool cooling
- shutdown-cooling
- steam condensing
- standby coolant supply
- fuel pool cooling

The system has two separate loops, each containing two pumps, a heat exchanger and the necessary piping. The four pumps of the RHR are motor-driven and have a design capacity of 687.5 kg/s each at 0.138 Mpa. They take suction from the torus suppression pool and are designed for water temperatures from 4°C to 177°C.

The relevant systems for the considered accident scenario are the low-pressure coolant injection (LPCI) mode and the shutdown-cooling and head spray mode.

The LPCI is designed to maintain an appropriate water level in the reactor vessel in case of a LOCA. It automatically initiates at level 1 reactor vessel water level (-3.366 m) or high drywell pressure (+0.0117 MPa). Injection is performed via the recirculation system's discharge piping with pressure below the pump shutoff head of 2 MPa.

The shutdown cooling and head spray mode has the purpose of removing decay heat during normal reactor shutdown. The system is designed to reduce the temperature below 52°C within one day and to keep it below that value to allow refueling. Therefore the water is taken from the recirculation loop, cooled by the heat exchanger and discharged back to the RPV.

### 1.2 Relap5 Nodalisation

To simulate the thermo-hydraulic plant behavior a model of the Peach Bottom vessel components and coolant loop was adopted for the RELAP5/Mod 3.3 code, modeling the core with 33 T/H channels. The balance of plant was substituted by boundary conditions obtained utilizing a combination of calculated results and test data in order to reproduce the actual steady

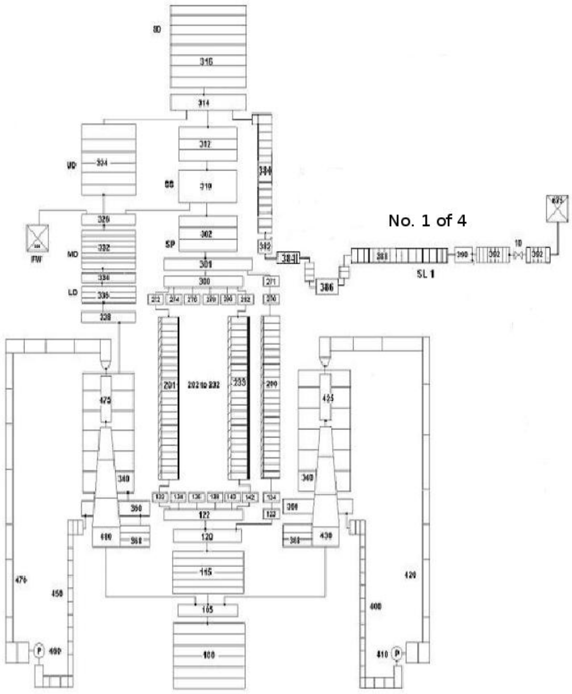


**Fig. 1 Relap5 mod 33 Nodalization Scheme.**

State conditions. The boundary conditions chosen for the RELAP5 model are:

| | |
|---|---|
| **Reactor Power** | 3458 MWth |
| **RPV pressure** | 7.1 MPa |
| **DC collapsed water level** | 11.42 m |
| **Core mass flow rate** | 12144.9 Kg/s |
| **Core inlet temperature** | 276.9 °C |
| **Core outlet temperature** | 285.76 °C |
| **Core outlet quality** | 14.5% |
| **Recirculation pump flow** | 2120 Kg/s |
| **Feedwater mass flow rate** | 1884 Kg/s |
| **Feedwater temperature** | 220°C |
| **Turbine inlet pressure** | 6.72 MPa |

**Table 4. Relap5 boundary conditions**

The existing model comprised a detailed nodalization of the RPV including the recirculation loops plus one steam line (Fig. 1). In the used nodalization all the four steam line are modeled. HPCI and RCIC are modeled considering a constant mass flow rate in the whole range of operation, while for the LPCI and CS pumps a linear characteristic is assumed. The water temperature of ECCs is assumed equal to 50°C. The capability of the ECCs simplified model has been tested against a 2A LOCA in one of the recirculation loop. A satisfactory agreement respect to the results reported in [10] was observed.

### 1.3 Boundary Conditions

Three cases are presented. All three cases assume as initiating event a main steam line break outside the containment of MSL “A”. In addition failure of both main steam isolation valves of the affected steam line is assumed.

**Case A** assumes in addition a loss of all active ECC. Only HPCI and RCIC, which are both turbine driven, are assumed to be available. The case is, due to its low probability, of academic interest only, and constitutes in this study a reference case to be compared against the two other cases.

**Case B** assumes that active and passive ECC systems are available, and assumes that the operator, based on falling pressure, identifies the initiating event as LOCA inside the containment, and functional containment isolation.

**Case C** assumes, like case B, that active and passive ECC systems are available, but this time the operator adopts a strategy to limit the losses through the break – he switches RHR to shutdown-cooling and head spray after 30min.

## 2. CALCULATED RESULTS

### 2.1. Description Results Case A

Following initiating event the pressure in the RPV and steam lines decreases rapidly, and turbine valves and main steam isolation valves start to close. At 1.0 s the main steam isolation valve open area is lower than 90%, and reactor scram (and shutdown of recirculation pumps) is initiated. At the same time, pressure in the RPV and level continue to decrease.

At 28 s conditions for switch on of HPCI and RCIC are met. Within a period of 154 s 54895 kg of water are injected. However, the pressure quickly drops below 1.034 MPa, which is below the minimum pressure needed for operation of the turbine driven ECC systems. From this point on the water inventory boils of due to the decay heat. Roughly 2100s after beginning of the transient the water level in the core is so low that dryout conditions are reached, and cladding temperatures start to rise. Roughly 1500s later, at 3500s, cladding temperatures exceed 1200°C, which was set as indication for core damage and the calculation was terminated. Please refer to Figure 2 and Figure 3 for break mass flow, RPV pressure, RPV level and PCT.

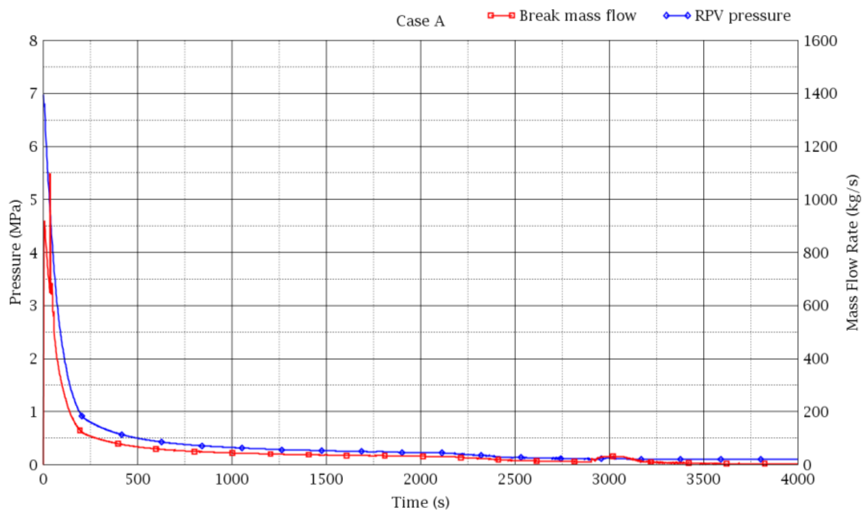


**Fig. 2 Case A mass flow through the break and RPV pressure**

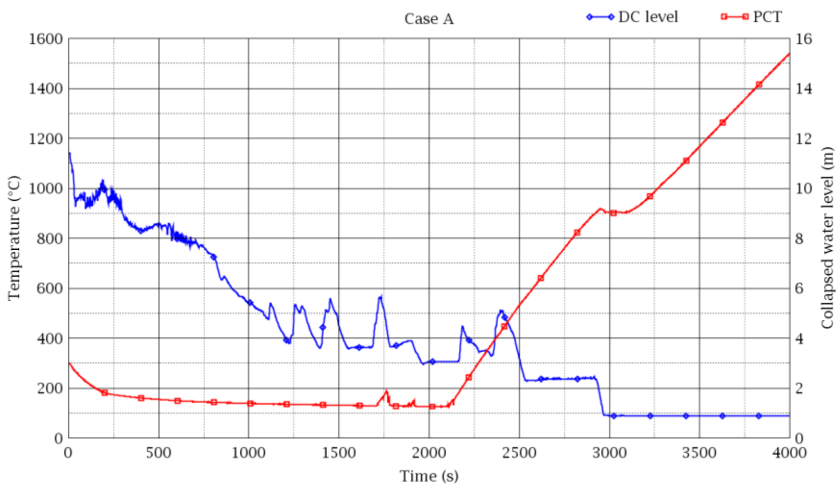


**Fig. 3 Case A DC collapsed water level and PCT**

## 2.2. Description Results Case B

The initial phase of Case B evolves like Case A, but since also active ECC is assumed to be available, LCPI and CS start injection and the trend of the RPV level is inverted at roughly 800s (see Figure 5). The RPV is quickly refilled. However, since no signal for switch off of the system is assumed to be initiated, the steam lines fill up with liquid, and coolant is lost through the break (see Figure 4). After roughly 2400s the inventory of condensate storage tank and the torus is depleted and injection stops. From this time, water is evaporated through the break, which takes, since the RPV is filled up to the steam lines, up to 13000s. At this time the water level of the core is such that dryout and core heat-up starts, which continues up to 15500s. After this time cladding temperatures exceed 1200°C, which is taken as indication for core damage and the calculation has been interrupted.

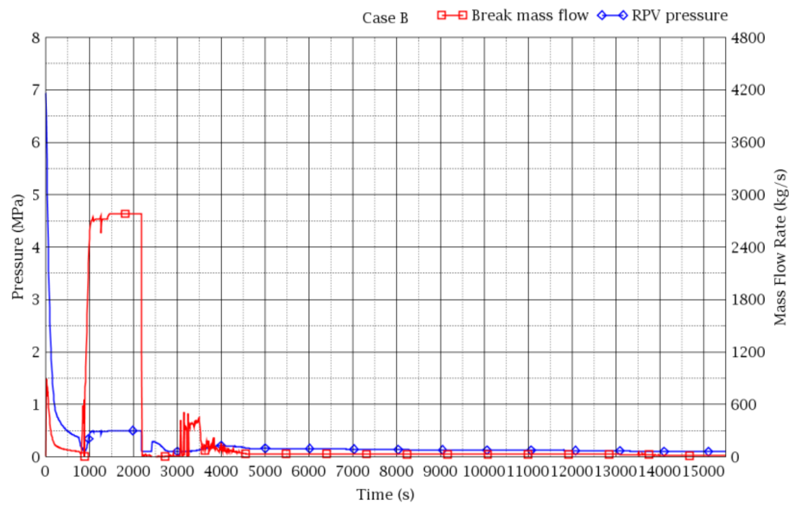


**Fig. 4 Case B mass flow through the break and RPV pressure**

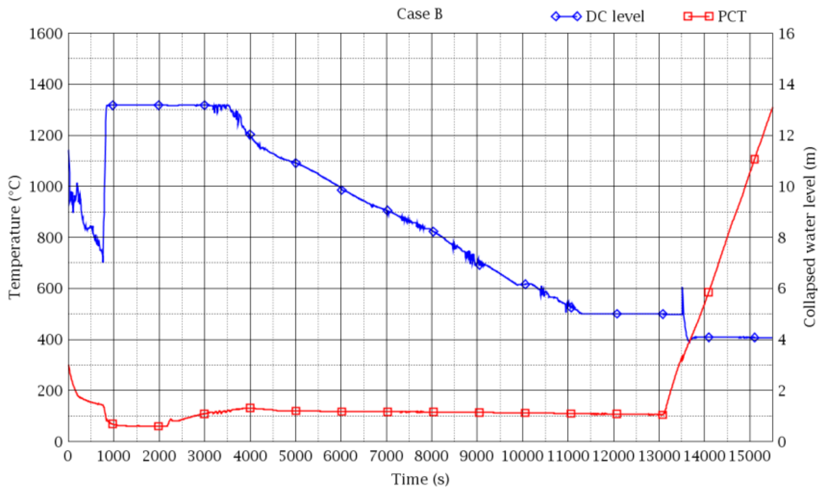


**Fig. 5 Case B DC collapsed water level and PCT**

## 2.3. Description Results Case C

Case C develops like case B, for the first 1800s. As can be seen from Figure 5, the large reservoir of water, combined with the capacity of the LPCI and CS, allows filling the RPV completely within a short period of time. The strategy under evaluation is to switch RHR into shutdown cooling. In this configuration suction is taken from the recirculation line (before the recirculation pump), and injected after the recirculation pump. This way a closed recirculation loop is established. The heat exchanger can operate up to pressures of 8.6 bar and 177°C. LPCI, however, is capable to reduce RPV pressure and temperature below the values mentioned above. As can be seen in Figure 6, the procedure practically reduces the coolant lost through the break to zero. After 3500s of calculation showing that all parameters are stable (Figure 6, Figure 7), the calculation has been stopped and the strategy has been considered to be successful.

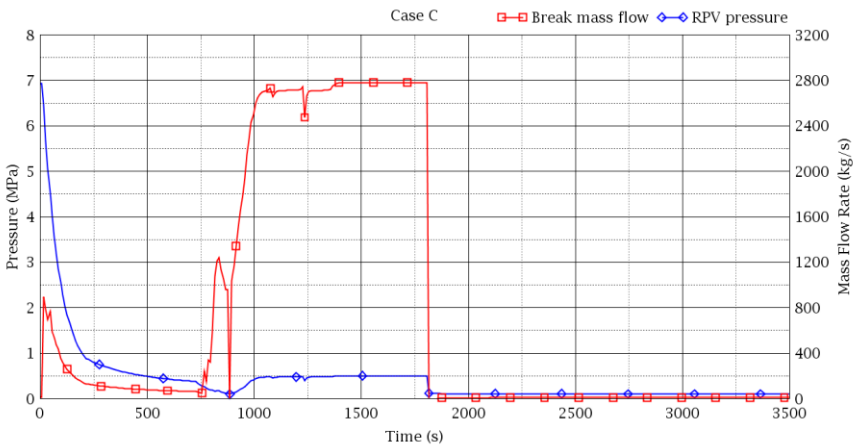


**Fig. 6 Case C mass flow through the break and RPV pressure**

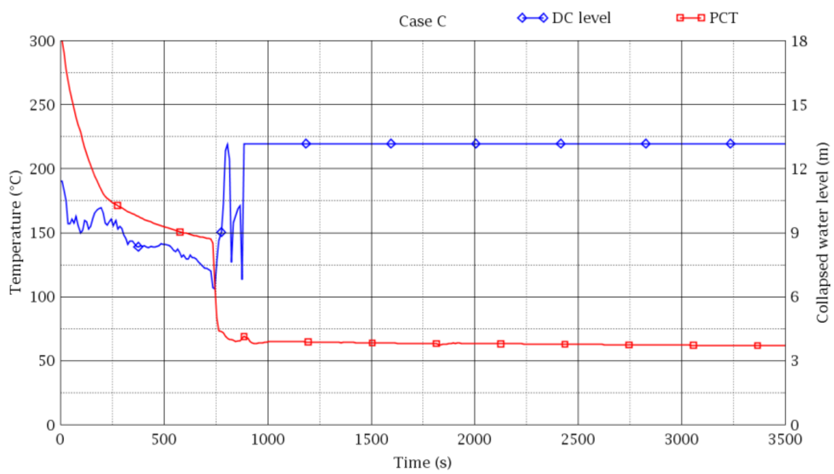


**Fig. 7 Case C DC collapsed water level and PCT**

## 3. CONCLUSIONS

The present analysis investigated the consequences of a beyond design basis accident (non isolable break main steam line break outside containment). In a first analysis, only passive ECC systems have been considered to be available. Analysis shows that due to the fast depressurization passive ECC are of reduced effectiveness, and come into operation only for a short period of time. After the coolant inventory present in the RPV has been boiled off, dry-out and core heat-up cannot be avoided. The analysis (case B) assumes also active ECC to be available. The large inventory present in torus and condensate tank allows supply coolant for roughly 4-5 hours. It should be noted that the calculation tried to establish the maximum time that would be available – peach bottom power upgrade has not been considered in the analysis, ECC pump characteristic curves are biased towards lower injection rates and maximum possible inventory of torus and condensate tank has been considered.

Case C, the last analysis, shows that switching the RHR into shutdown cooling is successful as strategy. Recirculation inside the RPV can be established, look term cooling can be assured, and the losses through the break can be, basically, eliminated. However, one has to stress that the analysis restricted itself to indicate a strategy, and that many points still have to be covered, in case the strategy should be turned into a procedure. For example – what are the consequences of break for the turbine building? What are the consequences for the control room, which is situated close to the turbine building in Peach Bottom? Are there signals in the control room, which allow to clearly identifying the event as non isolable break? These and other questions have to be further investigated in order to turn the strategy into a procedure. However, it can be concluded that the event, without mitigating operator actions, has the potential to loose the available coolant through the break within 5 hours, and core damage would be especially severe since there is a open containment bypass. Through adoption of an optimal accident management strategy, core damage can be avoided.

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