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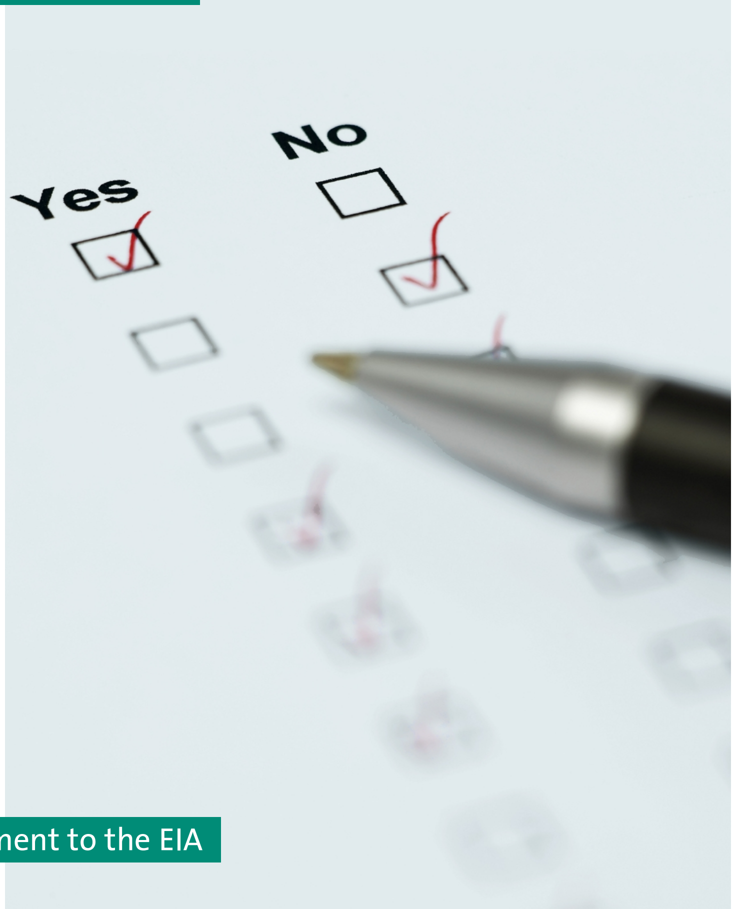
Hinkley Point C



Expert Statement to the EIA

HINKLEY POINT C

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Oda Becker

REPORT
REP-0413

Wien 2013

Project Management

Franz Meister, Umweltbundesamt

Author

Oda Becker

The author expresses her thanks to Dr. Helmut Hirsch for reviewing a draft of this report and providing helpful comments

Layout and typesetting

Elisabeth Riss, Umweltbundesamt

Title photograph

© iStockphoto.com/imagestock

Imprint

Owner and Editor: Umweltbundesamt GmbH
Spittelauer Lände 5, 1090 Wien/Österreich

Printed on CO_2-neutral 100% recycled paper

ISBN 978-3-99004-217-5

CONTENT

SUMMARY

A new nuclear power plant (Hinkley Point C – HPC), comprising two UK EPR™ is planned at Hinkley Point, Somerset. Under the National Regulations, the application for the Development Consent Order (DCO) must be subject to an Environmental Impact Assessment (EIA). With reference to the ESPOO Convention, Austria takes part in the transboundary EIA. The Umweltbundesamt (Environment Agency Austria) has assigned Oda Becker, scientific consultant, to elaborate an expert statement on the project documents presented by the UK. The review of the documents is focused on the safety and risk analysis. The goal is to assess if the EIA process leads to reliable conclusions about the potential impact of transboundary emissions.

On September 21, 2012, the National Planning Inspectorate (Examining Authority) concluded the examination of the Hinkley Point C application and sent its recommendation to the Secretary of State (December 19, 2012). The report of recommendation will be published once a decision has been made. A decision will be published on or before March 19, 2013.

On December 13, 2012, the Office for Nuclear Regulation (ONR) has issued a Design Acceptance Confirmation (DAC) for the UK EPR™ design. During GDA process, however, ONR has identified several “findings” that are important to safety and still need to be resolved (Assessment Findings).

According to EDF and AREVA, the UK EPR™ is a Generation 3+ reactor; its safety approach at the design level is based on an improved concept of defence in depth. EDF and AREVA claim that the plant’s safety concept meets ad vanced regulatory requirements so that, on the one hand, accident situations resulting in a core melt that would subsequently lead to large early releases are practically eliminated and, on the other hand, the consequences of low pressure core melt sequences that would require protective measures for the public are very limited both in area and time.

Taking into account all the presented facts of the application documents, the preservation of the containment integrity neither in the long-term nor in the short term during a severe accident is guaranteed by the proposed safety design and features yet.

The claimed “practical elimination” of a large early release is not sufficiently demonstrated.

Severe accidents with high releases of caesium-137 (>100 TBq) cannot be excluded, although their calculated probability is below 1E-7/a. Consequently, such accidents should have been included in the EIA since their effects can be widespread and long-lasting.

Many relevant factors are not included, because they fall outside the scope or are not addressed appropriately (for example, Common Cause Failure (CCF)). PSA results in any case should only be taken as very rough indicators of risk. All PSA results are beset with considerable uncertainties, and there are factors contributing to NPP hazards which cannot be included in the PSA. Therefore, for rare events, the probability of occurrence as calculated by a PSA should not be taken as an absolute value, but as an indicative number only.

In the Environmental Statement, EDF Energy claims: “Significant transboundary environmental effect arising from construction and operation of HPC are not considered likely.” In the transboundary screening document, the Secretary of State confirmed this view.

For the estimation of possible transboundary impact, calculations of the flexRISK project are used. The flexRISK project modelled the geographical distribution of severe accident risk arising from nuclear power plants in Europe. Using source terms and accident frequencies as input, for about 1,000 meteorological situations the large-scale dispersion of radionuclides in the atmosphere was simulated.

For each reactor an accident scenario with a large release of nuclear material was selected. For a severe accident at Hinkley Point B, a caesium-137 release of 53.18 PBq is assumed. This source term is comparable with UK EPR™ source terms calculated in the PSA 2.

This possible caesium-137 release at Hinkley Point C, would result in a considerable contamination of the Austrian territory. Most parts show depositions of about 1,000 Bq/m² which is beyond the thresholds (650 Bq/m²) that agricultural intervention measures trigger.

The presentation of the results of the analysis of transboundary impacts of a potential severe accident at the Hinkley Point NPP site illustrate that an impact on Central European regions (including Austria) cannot be excluded. The results indicate the need for official intervention in Austria after such an accident.

Recommendation of this Expert Statement:

A conservative worst case release scenario should have been included in the EIA. A source term, for example for an early containment failure or containment bypass scenario, should have been analysed as part of the EIA – in particular because of its relevance for impacts at greater distances. It is recommended that this should be taken into consideration before granting further permissions.

Austria should be kept informed regarding the ongoing progress resolving the “Assessment Findings” concerning severe accidents.

1 INTRODUCTION

The NNB Generation Company Limited[1], part of EDF Energy, plans to construct and operate a new nuclear power plant (NPP), comprising two UK EPR™ at the Hinkley Point NPP site. The electric capacity of each unit will be around 1,630 MWe. On October 31, 2011, an application for a Development Consent Order (DCO) was submitted.

Under the Planning Act 2008, this application must be subject to an Environmental Impact Assessment (EIA) in accordance with the Infrastructure Planning Regulations 2009 (EIA Regulations). With reference to the ESPOO Convention, the UK has submitted the application of this project to Austria. The EIA procedure is close to completion and the decision of the application as well as the Recommendation of the Examining Authority is intended to be published on March 19, 2013 (PLANNING INSPECTORATE 2013).

The Austrian Federal Environmental Agency "Umweltbundesamt" has assigned Oda Becker, scientific consultant, to elaborate an expert statement on the documents presented by the UK.

The review of the documents is focused mainly on the safety and risk analysis. The goal is to assess if the EIA process allows making reliable conclusions about the potential impact of transboundary emissions.

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[1] In the following, this company is referred to as EDF Energy.

2 DESCRIPTION OF THE PROCEDURE

On October 31, 2011, NNB Generation Company Limited[2], part of EDF Energy, submitted an application for a Development Consent Order (DCO) to the Infrastructure Planning Commission (IPC, now Planning Inspectorate) to construct and operate a new nuclear power station at Hinkley Point, Somerset, to be known as Hinkley Point C (HPC) (EDF ENERGY 2011b).

The Planning Act 2008 introduced a new planning regime for nationally significant infrastructure projects including nuclear power plants. Since March 1, 2010, such projects must be authorised by grant of a **Development Consent Order (DCO)** from the Planning Inspectorate (formerly Infrastructure Planning Commission, IPC). National Policy Statements (NPS), also introduced by the Planning Act 2008, provide the policy framework against which the Planning Inspectorate is required to make its decision. The NPS for Nuclear Power Generation (EN-6) sets out the Government´s assessment of the need for new nuclear power generating capacity and has been accompanied by a Strategic Siting Assessment (SSA). The SSA identifies sites that are considered strategically suitable for the construction of new nuclear power stations; Hinkley Point is identified as one of eight sites in the UK (EDF ENERGY 2011).

Under the Planning Act 2008, the application of a DCO must be subject to an Environmental Impact Assessment (EIA) in accordance with the Infrastructure Planning Regulations 2009. Under this EIA regulations, EDF energy was required to prepare an **Environmental Statement (ES)** that reports on the likely environmental effects arising from the construction and operation of HPC, and to identify appropriate measures to mitigate significant adverse impacts. The ES has been published as a set of eleven volumes together with the non-technical summary (NTS) (EDF ENERGY 2011).

The scope of the EIA was agreed with the Planning Inspectorate and other relevant authorities. The scoping process identifies the potentially significant environmental effects of the proposed development and defines the study area and methodology for assessing environmental impacts (EDF ENERGY 2011).

On September 21, 2012, the National Planning Inspectorate (Examining Authority) concluded the examination of the Hinkley Point C application. The Planning Inspectorate issued a report of recommendation to the Secretary of State on December 19, 2012. The Secretary of State has three months to issue a decision. The decision letter and report of recommendation will be published once a decision has been made. A decision will be published on or before March 19, 2013 (PLANNING INSPECTORATE 2013).

In response to EDF Energy's application to build two reactors at the Hinkley Point C site in Somerset, a Nuclear Site Licence was granted on November 26, 2012.

On December 13, 2012, the Office for Nuclear Regulation (ONR) has issued a Design Acceptance Confirmation (DAC) and the Environment Agency a Statement of Design Acceptability for the UK EPR™ design.

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[2] In the following, this company is referred to as EDF Energy.

As regards the procedure the open and transparent approach of the British authorities concerning the availability of relevant documents has to be highlighted.

2.1 Generic Design Assessment

In June 2006, the UK's Health & Safety Executive (HSE), which licenses nuclear reactors through its Office for Nuclear Regulation (ONR), suggested a two-phase licensing process similar to that in the USA. The first phase, developed in conjunction with the Environment Agency (EA), is the Generic Design Assessment (GDA) process (WNA 2013). The GDA, also referred to as *pre-licensing*, aims to assess the generic safety, security and environmental aspects of new designs of nuclear power plants (LARGE 2012a).

The ONR has undertaken a Generic Design Assessment (GDA) of the UK EPR™ nuclear reactor during the period from July 2007 to December 2012. On December 14, 2011, ONR issued an Interim Design Acceptance Confirmation (IDAC) for the UK EPR™ nuclear reactor. There were a number of open GDA Issues which had to be addressed.

The ONR report (2012) summarises the work undertaken to assess EDF and AREVA's responses to the 31 GDA Issues and documents why ONR is content to provide a Design Acceptance Confirmation (DAC).

The GDA Issues include resilience to internal hazards, adequacy of the structural integrity of the built-structures, doubts about I&C system and human factors. Although limited to 31 in number, the majority of the individual GDA Issues are a composite made up of a number of often quite involved tasks to be executed (LARGE 2012a).

Findings that were identified during the regulators' GDA assessment are important to safety, but are not considered critical to the decision to start nuclear island safety-related construction, are known as **Assessment Findings (AF)**. After GDA, the Assessment Findings will be subject to appropriate control as part of normal regulatory oversight (ONR 2012).

During GDA a total of 82 design change proposals have been identified; during the GDA close-out phase another 54 design improvements have been proposed by EDF and AREVA within their responses to the GDA Issues (ONR 2012).

These design and safety improvements have now been accepted within GDA by ONR. Further development of the details of these modifications will be progressed after GDA, during the site-specific phase. Examples of the design changes are, changes to the architecture of the control instrumentation and control (I&C) systems, including the addition of a non-computerised I&C safety back-up system as well as improvements to the spent fuel cooling pond (ONR 2012).

Although the changes were identified independently of the Fukushima lessons learnt review, many of them are considered by ONR to help provide additional protection in extremely challenging hazard or plant failure scenarios. In addition, as a result of the post Fukushima review, EDF and AREVA identified five design change proposals, covering 16 resilience enhancements (ONR 2012).

EDF and AREVA's safety case for GDA Step 4 was described in their March 2011 Pre-Construction Safety Report (PCSR). This was updated during the GDA Issue close-out phase to take account of new information, to improve the clarity of the safety arguments, and to include agreed design changes. The updates were incorporated into a final version of the PCSR which was submitted in November 2012 (ONR 2012).

Within GDA, ONR has also conducted a **security assessment** alongside the safety assessment. The provision of a DAC by ONR means that it is fully content with both the security and safety aspects of the generic design (ONR 2012).

2.2 Conclusion

On December 13, 2012, ONR has closed the generic design assessment (GDA) and has issued the Design Acceptance Confirmation (DAC) for the UK EPR^{TM}

During GDA process, however, ONR has identified several "findings" that are important to safety and still need to be resolved (Assessment Findings). In the important topics containment hydraulics performance / severe accident and Probabilistic Safety Analysis (PSA), ONR has raised 26, respectively 46, Assessment Findings.

In the next chapters, ONR´s assessments including Assessment Findings (AF) are described to some extent in order to evaluate the possibility of severe accidents at Hinkley Point C by which Austria could be affected.

3 SEVERE ACCIDENTS

3.1 Treatment in the Application documents

The UK EPR™ reactor is a four-loop Pressurised Water Reactor (PWR) with a rated thermal power of 4,500 MW and an electrical power output around 1,630 MW, depending on conventional island technology and heat sink characteristics. The reactor is designed for a lifetime of 60 years (UK EPR 2012, OV).

The UK EPR™ design complies with safety requirements formulated by the French and German nuclear safety authorities for the next generation of nuclear reactors. According to EDF and AREVA, the UK EPR™ is a Generation 3+ reactor and benefits through its evolutionary design from global international experience acquired at both PWR system operational level in western countries, and French and German engineering design experience (UK EPR 2012, 3.1).

The safety approach at the design level is based on an improved concept of defence in depth (UK EPR 2012, 3.1).

The Risk Reduction Category A (RRC-A) is introduced to complement the deterministic Design Basis Analysis by considering a set of Design Extension Conditions (DEC) involving multiple failure events. Analysis of the DECs is used to identify additional safety measures (so-called 'RRC-A features'), which make it possible to prevent the likelyhood of the occurrence of severe accidents in these complex situations. One RRC-A sequence is concerned with the Loss of Offsite Power (LOOP), combined with the total failure of the four Emergency Diesel Generators (EDGs), whilst at-power (state A). The RRC-A features associated with this functional sequence are the two Station Black Out (SBO) diesel generators which supply electrical power to the emergency supply system for the Emergency Feed Water System, trains 1 and 4. The operator switches to the SBO diesel generators manually (UK EPR 2012, 16.1).

The plant's safety concept meets advanced regulatory requirements so that, on the one hand, accident situations with core melt which would lead to large early releases are practically eliminated and, on the other hand, low pressure core melt sequences (Risk Reduction Category B, RRC-B) necessitate protective measures for the public, which are very limited both in area and time. RRC-B is concerned with preserving the containment integrity in the long-term. This task encompasses the prevention of

- Hydrogen risks for the containment in the long-term,
- Containment failure due to exposure of the concrete base-mat to core melt,
- Containment failure due to containment over-pressurisation.

The possibility of hydrogen combustion in the long-term is avoided by installing autocatalytic recombiners in the containment. An ex-vessel core melt stabilisation system avoids the penetration of the liner and concrete base-mat, and, subsequently, the interaction between molten core and subsoil, and long-term ground-water contamination. By maintaining the melt in a cooled configuration, the stabilisation system further prevents the heat-up of the concrete in the lower containment region. This eliminates the risk of thermal deformation and induced crack formation in the concrete slab. For long-term decay heat removal, the UK EPR™ has a dedicated containment heat removal system (CHRS) (UK EPR 2012, 16.2).

Each type of accident, which has the potential to breach the containment early in the accident, could result in large early releases. Practical elimination of these accidents is achieved by specific engineered safety features that concern the following phenomena (UK EPR 2012, 16.2).

- Core melt under high pressure and direct containment heating
- Large steam explosions which can threaten the containment
- Hydrogen combustion phenomena potentially critical to containment integrity

3.2 Discussion

3.2.1 Safety Aspects

Origin and Objectives of the EPR Project

The conceptual design, based on the French reactor N4 and the German reactor Konvoi, was completed in 1994, with a planned output at that time of 1,450 MWe. The design combined development from some parts of N4, such as the containment, and some parts of Konvoi such as the instrumentation.

The global goal of the French safety approach for operating reactors was to ensure that design, assessment and control of the reactors should guarantee a probability of a major accident with severe damage to the core of less than 10E-5 per reactor and year (/yr) and a probability of an event that could lead to unacceptable consequences for the population of less than 10E-6/yr. The concepts of the EPR remain based on the same approach of probabilistic assessment and increased depth of defence. The improvement objective with the EPR can be summarized by getting these probabilities down, respectively to 10E-6/yr and to 10E-7/yr.

The basic design phase started in 1995 and was completed in August 1997. While the French nuclear safety authority stated in September 1999 that it expected to give its conclusions on a final design certification in the coming months, the generic design approval of the EPR, still not final, was only issued in September 2004 (MARIGNAC 2011).

EPR Projects in France, Finland and the US

The Finnish and French regulators both agreed for orders to be placed for the construction of EPRs respectively at Olkiluoto in Finland in 2003 and Flamanville in France in 2005. At that time, the level of review of the EPR detailed design did not reach that of a comprehensive generic safety assessment. As a result, while construction is going on, although experiencing major delays partly due to the complexity of the reactor, the final generic approval was still not granted. In 2010, the French nuclear safety authority pointed out that it would not be in a position, should the construction work in Flamanville be completed at that time, to give approval for the operation of the EPR to start – and this was before the Fukushima accident (MARIGNAC 2011).

The process of granting approval for the EPR design is also slower than expected in the United States. In the US, AREVA NP was expecting some generic approval by 2008 when it started discussions with the nuclear safety regulator (NRC) about the EPR design in 2004. In December 2007, when AREVA NP submitted a Standard Design Certification Application to the NRC, it was then expecting the technical review to end by 2010 (MARIGNAC 2011). The application is still under review by the NRC and will probably only be completed in 2014 (LARGE 2012b).

One major pending issue that was explicitly a reason for delays is the Instrumentation & Control (I&C) system. The EPR design includes a fully computerised I&C system which is a significant development. The same was already tried in the development and construction phase of the N4 reactors in France, but was eventually dropped in favour of an already proven system, contributing to four years of delays in the completion of these reactors. The concerns with the proposed I&C lies in negative interactions that could arise from its complexity and redundancy (MARIGNAC 2011).

In late 2008, the Finnish nuclear safety regulator (STUK) made public its concern over unresolved issues centring around the incomplete design and its reservations about I&C architecture. When this was subsequently addressed by the French regulator (ASN), the I&C issues become known to the ONR (2009) and the NRC (2010) (LARGE 2012b).

Other issues are still being discussed within the process of safety assessment conducted by the regulators or through independent expertises of some aspects of the design. These include the assessment of the probability of a steam explosion due to the very energetic reaction between the melted core and the water that might be found in the “core catcher” in the process of an accident, the concerns with potential failures of the emergency cooling systems including sump clogging. Some concerns were also raised through the anonymous disclosure of studies by EDF about the possible reactivity of the core in transients linked to the project to allow fast change of the reactor power to follow power demand (load following) (MARIGNAC 2011).

Olkiluoto-3 will likely not be in commercial operation until 2016, seven years behind the original contract schedule, largely because of continuing problems with design of digital instrumentation & control. STUK has required that there be a hard-wired analogue backup for the safety functions of the digital system. The French regulators did not require such a backup, nor did Chinese regulators for the two EPR units being built at Taishan and scheduled to operate at the beginning of 2014 (NUCLEONICS WEEK 2013).

Meanwhile, concerns about the costs associated with new safety features have led to **increase the output** of the plant. Another way to try to improve the economics of the EPR is to aim for better fuel performance. This includes a design objective of burning the uranium oxide fuel (UOX) up to an unprecedented level (70 GWd/t), which posing specific problems. This also includes the possibility to use as much as 100% mixed oxide fuel (MOX) in the core; irradiated MOX fuel has a heat output up to four times higher than UOX fuel and poses significant reactivity problems, and it contains more plutonium which is highly toxic (MARIGNAC 2011).

Finally, some issues were raised regarding the progress made by EPR in terms of security, and its capacity to withstand some kinds of malevolent attacks that have become credible after September 11, 2001. This particularly relates to the resistance of the containment to a commercial plane crash. No details are known since this comes under national defence secrecy (MARIGNAC 2011).

For the UK-EPR™ as well as the EPR being built at Flamanville in France, the risk of an aircraft crash has been assessed on the basis of French regulatory requirements. Those regulations cover accidental crashes and do not require protection against the crash of a commercial airliner.

European Stress Tests and the UK EPR™

As the GDA process moved through the final stage, the accident at the Fukushima Daiichi NPP in March 2011 occurred. Following that accident, the European Council of 24/25 March 2011 requested that the safety of all EU nuclear power plants should be reviewed, on the basis of a comprehensive and transparent risk and safety assessment ("stress tests"). For these stress tests, the conditions and parameters for re-evaluation of the NPP resilience to and management in the aftermath of an extreme external event were set out.

Both the French and the Finnish nuclear safety regulator required the operators to address all issues arising from the Stress Tests requirement as these applied to the EPRs under construction at Flamanville and at Olkiluoto.

Despite the imposition of the European Council's requirement for re-evaluation of NPP performance (existing and planned) via the stress tests, ONR requires EDF and AREVA to evaluate these issues as a separate GDA Issue. EDF and AREVA identified five design change proposals including (ONR 2012):

- Improved flood protection for emergency electrical supplies
- Extension of the capability and autonomy of emergency electrical supplies
- Identified connection points for proposed mobile diesel generators
- Addition of spent fuel pool (SFP) instrumentation into the severe accident management I&C systems
- Identification of a reserve ultimate water supply
- Delivery of water via mobile pumps for SFP make-up and containment pressure control

For instance in the frame of the Multinational Design Evaluation Programme (MDEP[3]), there was information exchange on the lessons learnt from Fukushima and how these could affect the UK EPR™.

[3] The aim of MDEP is to promote international sharing of information between regulators on their new nuclear power station safety assessments and to promote consistent nuclear safety assessment standards among different countries (ONR 2012).

Critical Role of Station Black-Out (SBO)

To provide the necessary electrical power for safety relevant systems in case of loss of offsite power, the EPR is equipped with four emergency diesel generators (EDG). A loss of offsite power combined with the failure of the four EDG would lead to the unavailability of various safety relevant systems. The EPR is equipped with additional power sources, the so called SBO-diesel generators (SBO-DGs).

The SBO-DGs are diversified with regard to the EDGs. Therefore, according to AREVA, a common cause failure (CCF) of the SBO-DGs together with the EDGs had not to be considered before the accident at Fukushima.

The diesel buildings, each housing two EDGs and one SBO-DG, are designed to withstand earthquakes and explosions. However, the EPR diesel buildings' protection against aircraft crash is provided exclusively by the different positions of the buildings on the site, which are separated by the reactor building. A physical protection of the buildings is not implemented for the EPR. This is different to the Konvoi plants (HIRSCH 2011).

According to the Stress Tests for Olkiluoto 3, in case of SBO, if countermeasures were unsuccessful, the uncovering of the core would take place within 3 hours with extensive fuel damage within 4 hours and pressure vessel melt-through within 7 to 8 hours after an accident starts (ENSREG 2012).

Figure 1 illustrates the resulting Time Line for Post Accident Management (SBO = Total Loss of AC Power for Power States) (UK EPR 2011, 16.6).

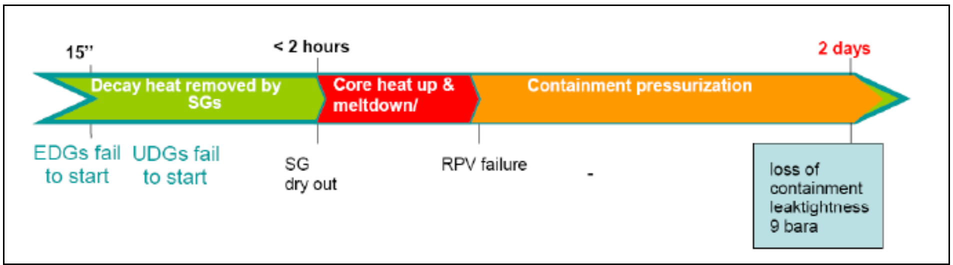

Figure 1: Time Line Post Accident Management (SBO at Power States)

3.2.2 Severe Accidents

Containment Sump Clogging

In design-basis faults, reactor coolant inventory is generally replenished by safety injection from the in-containment refuelling water storage tank (IWRST). However, this has a limited size and ultimately will empty. In the largest loss-of-coolant accidents this can happen in a matter of hours. Under these circumstances, the operator is required to realign the injection pump suction lines to take water from the containment sump. It is necessary to ensure that debris in the containment building is not swept into the primary circuit where it would impair cooling (ONR 2011b). This problem is not sufficiently resolved. The UK EPR^{TM} project should identify a design which reduces risks in this area as far as reasonably practicable (ALARP) (AF-UKEPR-CSA-07).

Primary Depressurisation System (PDS)

The containment design takes into account consequences related to a severe accident, but without considering loads induced by High Pressure Melt Ejection (HPME). In the context of severe accidents, the primary depressurisation system aims to avoid the possibility of HPME and the potential for Direct Containment Heating (DCH), phenomena which can lead to early containment failure.

The manual operation of the PDS introduces a degree of uncertainty into the time and rate of depressurisation. The PCSR does not fully describe the functional requirements of the PDS during design basis and severe accidents. The successful initiation of the PDS is a key step within the severe accident management procedures in preventing high pressure accident scenarios leading to a HPME (ONR 2011b).

The operator may depressurise the Reactor Cooling System (RCS) at various stages during the fault conditions. Depressurisation is anticipated to be activated by the operator when the core outlet temperature reaches 650°C. The core outlet temperature is also proposed to be used for initiation of severe accident management procedures associated with control of debris and containment performance. The measurement systems indicating core conditions used to initiate the accident management procedures have to justify, in particular concerning common cause failure (CCF) (AF-UKEPR-CSA-08).

Ex-Vessel Cooling of Molten Core

To stabilize the molten core in a severe accident, the EPR relies on an ex-vessel strategy. The intent of the design is that the molten material will be spread sufficiently evenly so that it can be cooled efficiently and retained in a stable configuration where it cannot damage the structure of the containment building. The design is also intended to minimise the release of gas from concrete materials as a result of melt-concrete interaction.

In-vessel melt retention by outside cooling of the reactor pressure vessel (RPV) was dismissed because the high power rating of the reactor leads to low margins for heat transfer. The molten material from the RPV is first collected in the reactor pit. In the pit, the corium is temporarily retained by a layer of sacrificial concrete. The time delay and the admixture of the concrete leads to a collection of melt in the pit and a more uniform spectrum of possible melt states at the end of the retention process. Finally, the melt will penetrate the melt plug consisting of concrete and a metal plate (of Al/Mg-alloy) and flow into the core catcher properly.

Because of the retention and collection in the pit, the subsequent spreading and the stabilisation measures are largely independent of the uncertainties associated with in-vessel melt pool formation and RPV failure; there is a one-step release into the spreading area. There, the spread melt is to be stabilised by flooding and external cooling.

The cooling of the melt in the core catcher by the overflow of water from the in-containment refuelling water storage tank (IRWST) is fully passive and triggered by the arrival of melt in the core catcher. The water first fills the central supply duct underneath the core catcher, then enters the horizontal cooling channels and submerges the space behind the sidewalls. After filling, it will overflow onto the surface of the melt.

Alternatively to the IRWST, the containment heat removal system (CHRS) can be used to actively deliver cooling water. Solidification of the melt is to be achieved within a few days.

If the ex-vessel cooling of the molten core is functioning as planned, this new feature would have the potential to reduce the probability of large releases in case of a severe accident. However, the ONR´s assessment emphasised uncertainties regarding the functionality of different steps of the Core Melt Stabilisation System (ONR 2011b)**:**

The mass of ablated concrete is one of the key factors affecting the corium viscosity influencing the spreading capability and potentially the layer inversion. According to ONR (2011b), the presence of the layer inversion phenomenon for the bounding scenario of the minimum ablated concrete quantity has to demonstrate. This justification is required to ensure that the risk associated with any significant interactions between water and the metallic layer is avoided. The response should also demonstrate that the resultant corium viscosity is appropriate for the bounding scenario of the maximum ablated concrete quantity (AF-UKEPR-CSA-13).

The claim, that he potential presence of chunks of concrete above the melt plug at the time of bottom head failure has no significant consequences on the melt plug opening has to be justifyed (AF-UKEPR-CSA-15).

In the opinion of ONR (2011b), a blockage of the cooling channels under the spreading plate is not adequately examined by EDF and AREVA. ONR (2011b) therefore made an Assessment Finding requiring that this has to be addressed (AF-UKEPR-CSA-19).

In order to examine the effectiveness of corium spreading from the melt plug to the spreading compartment, EDF and AREVA employed the CORFLOW code and a complementary analysis based on a phenomenological spreading model developed by the Royal Institute of Technology (RIT), Stockholm.

The European Severe Accident Research Network (SARNET[4]) in 2007 questioned the applicability of the simplified approach raising some technical points with regard to the RIT model.

In order to examine the claims made for spreading of the core melt within the spreading compartment, ONR commissioned the German Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) to perform a set of independent confirmatory analyses to develop an appreciation of the extent of the uncertainties (ONR 2011b).

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[4] Forty-seven partners from Europe, Canada, India, South Korea, United States and Japan participate in the SARNET consortium. This network of excellence has been launched in 2004 in the frame of the EC FP6. A 2nd project has been defined in continuity in the FP7 frame in order to reach the self-sustainability after its end in March 2013.

The confirmatory analysis demonstrated a shortfall in some assumptions made in the PCSR methodology. Updated spreading calculations for bounding scenarios have to be provided (AF-UKEPR-CSA-20).

Steam Explosion in Accident Conditions

The possibility of steam explosions constitutes a problem during severe accidents. Such explosions, which can damage the containment, can occur when the molten core falls into a pool of water. In this case, the melt can fragment into small particles; heat transfer to the water is very fast, with abrupt vaporisation as a result. According to ONR (2011b), a steam explosion is not a totally incredible event, and so there is a need to assess the damage potential. Therefore, the risk of a steam explosion in the RPV bottom head (in-vessel) and in the reactor pit and spreading compartment (ex-vessel) have been considered.

ONR (2011b) concluded that EDF and AREVA have presented a safety case based on current international understanding such that the probability of an in-vessel steam explosion sufficiently energetic to breech the RPV is very low. But ONR (2011b) pointed out that this assessment is based on subjective views on melt progression and conversion efficiencies, supported, in part, by limited modelling and the experimental database.

Melt can also contact water ex-vessel, either in the reactor pit, transfer channel or spreading compartment. The design intention is that the reactor pit and transfer channel are maintained dry. However, in some accident scenarios water may accumulate in the reactor pit. Measure(s) and arrangement(s) for inspection in order to ensure that the reactor pit is kept sufficiently dry are required by ONR (AF-UKEPR-CSA-21).

Corium Re-criticality

One essential safety function which needs to be addressed is the ability to shut down the chain reaction and retain the core subcritical. The potential for re-criticality is one of the hazards to be considered when the core configuration is lost (ONR 2011b). This requires consideration of the pool of molten debris formed once the core has relocated to the RPV lower head and the corium melt as it moves from the RPV into ex-vessel positions. According to ONR (2011b), the risk of re-criticality due to the relocated molten material and its progression within the Core Melt Stabilisation System (CMSS) should receive further examination (AF-UKEPR-CSA-22).

Prevention of Hydrogen Combustion

The containment has a dedicated combustible gas control system (CGCS) with two subsystems to avoid containment failure:

- The hydrogen reduction system consists of 47 passive autocatalytic recombiners (PAR) installed in various parts of the containment.
- The hydrogen mixing and distribution system.

The EPR containment is designed based on a two-region concept; inner containment (inaccessible) and outer containment with limited access to equipment while the reactor is operating at power. This is facilitated by the provision of radiation shielding within the containment and also thin contamination barriers.

This separation is convenient for plant operations, but complicates the combustible gas management during an accident by delaying dilution and mixing (ONR 2011b).

Several of the equipment rooms surrounding the Reactor Coolant System (RCS) are isolated from the rest of the containment during normal operation. In the event of an accident, communication is established between these equipment rooms, thereby eliminating any potential dead-end compartments where non-condensable gases could accumulate. A series of mixing dampers and blowout panels would open to transform the containment into a single volume.

ONR sees the need to consider whether it is ALARP to take additional measures to limit peak hydrogen concentrations (AF-UKEPR-CSA-23).

Furthermore, ONR (2011b) emphasised that there are a number of observations made with regards to the overall ventilation philosophy during normal operating and fault conditions relating to the foils and dampers which are responsible for the separation of the two atmospheres within containment (AF-UKEPR-CSA-01 to AF-UKEPR-CSA-05)

Performance of PAR

Additional confirmatory experimental work is required to provide greater assurance that fission product poisoning of passive autocatalytic recombiners (PARs) is unlikely to adversely influence their operational capabilities (AF-UKEPR-CSA-24).

Measures against Containment Overpressure

The ex-vessel core cooling system has to be seen in connection with the Containment Heat Removal System (CHRS). This system controls the containment pressure. It consists of a spray system and allows recirculation through the cooling structure of the molten core retention device to mitigate the consequences of the considered accident scenario. The CHRS serves to avoid containment failure while the molten core is stabilised in the core catcher. It also aims to avoid venting of the containment; Konvoi and N4 plants are equipped with filtered venting systems for containment pressure control.

ONR highlighted that UK EPR design does not have a filtered discharge facility to vent the containment. EDF and AREVA indicated that the EOPs recommend discharging into the adjacent buildings as an alternative to a filtered discharge. Although no additional information is provided to justify this alternative venting route, ONR consider that this strategy could lead to increased radiological releases following a severe accident to the peripheral buildings, limiting access for recovery and potential use of equipment.

ONR stated that it will expect that the EPR project should identify a design which reduces risks in this area as far as reasonably practicable and, therefore, raising an Assessment Finding requesting that a potential licensee demonstrate why the proposed design is ALARP (AF-UKEPR-CSA-25).

Hydrogen Analysis Codes

ONR (2011b) emphasised that the conclusions of a joint EU research project with the goal to develop verified and commonly agreed physical and numerical models for the analysis of hydrogen distribution, turbulent combustion and mitigation were generally positive. However, the present combustion models do not allow fully quantitative predictions of the detailed containment loads under all conditions.

A comprehensive set of documentation for the GASFLOW and the COM3D codes used in support of the PCSR (including substantiation of the codes' validity by comparison against measurements and independent analysis) has to be provided (AF-UKEPR-CSA-26).

EOP and OSSA

The proposed Emergency Operating Procedures (EOP) and the Operating Strategies for Severe Accident (OSSA) management are determinated as out of the scope Items of the GDA process (ONR 2011b).

3.3 Conclusion

The EPR, which relies on an increased power, aims for higher burn-up of fuels and for the use of MOX, increasing the potential of danger in comparison with the latest Generation II plants.

The EPR was conceived as a reactor with the capability to better withstand various types of threats and events while reducing the consequences of serious accidents. Nonetheless, its design basis needs to be re-examined in the light of the Fukushima accident (MAKHIJANI 2012). Regarding SBO, backfitting measures are necessary and planned, but the actual design problems remain. The relatively high thermal power of the EPR, for example, reduces the time for the operator to react properly during accident sequences to avoid a severe accident.

If the ex-vessel cooling of the molten core is functioning as planned, this new feature would have the potential to reduce the probability of large releases in case of a severe accident. However, the ONR´s assessment emphasised uncertainties regarding the functionality of the Core Melt Stabilisation System; in several Assessment Findings the need for further examination of nearly all important safety issues is addressed. Taking into account all the facts, the preserving of the containment integrity neither in the long-term nor in the short term is guaranteed by the proposed safety design and features yet.

Currently, it cannot be proven beyond doubt that a large release (>100 TBq) cannot occur. **Severe accidents with high releases cannot be excluded.**

A conservative worst case release scenario should have been included in the EIA. A source term, for example for an early containment failure or containment bypass scenario, should have been analysed as part of the EIA – in particular because of its relevance for impacts at greater distances. It is recommended that this should be taken into consideration before granting further permissions.

Austria should be informed continuously regarding the progress resolving the "Assessment Findings" concerning severe accidents.

4 PROBABILSTIC SAFETY ANALYSIS

4.1 Treatment in the Application documents

The PSA for the UK EPR™ is described in Chapter 15 of the Pre-Construction Safety Report (PCSR). The PSA is noted as a contribution to a key objective ensuring that the risk of release of radioactive products to the environment is reduced to As Low As Reasonably Practicable (ALARP) (ONR 2011a).

The PSA has been carried out at Level 1, 2 and 3.

The PSA considers all modes of operation including low power, shutdown and refuelling. The Plant Operating States (POS) are summarised below (ONR 2011a):

- States A and B, the plant is assumed to be at full power (i.e. 4,500 MWth), with all systems available, all controls in operation, and the core thermal power being removed via the steam generators.
- State Ca is representative of cold shutdown with the residual heat removal system in operation for reactor cooling. The reactor is pressurised and full of water.
- State Cb is representative of ¾ loop operation (usually called mid-loop operation), with the reactor pressure vessel head in place.
- State D represents ¾ loop operation with the reactor pressure vessel head removed. As the vessel head is open, the secondary side systems cannot be used for residual heat removal.
- State E is representative of core loading and unloading operations.

All sources of radioactivity are included in the PSA documentation. The sources of radioactive releases are (ONR 2011a):

- The reactor core
- The spent fuel storage pool
- The spent fuel handling facilities
- The radioactive waste storage tanks

The last three sources are not considered in the Level 1 PSA, but are considered in the overall PSA, feeding into Levels 2 and 3.

PSA Level 1

The Level 1 PSA considers both internal events and internal and external hazards that, together with total or partial failure of protection or mitigation measures, can lead to core damage, and evaluates the resulting core damage frequency (CDF). Other end points that do not result in core damage but may lead to potential releases, including those relating to the spent fuel pool, are included (ONR 2011a).

The calculated core damage frequencies (CDF) are summarised in Table 1.

Table 1: Core damage frequencies (CDF) (ONR 2011a)

CDF internal events	5.31E-7/yr
CDF internal hazards	1.01E-7/yr
CDF external hazards	7.59E-8/yr
CDF total	**7.08E-7/yr**

Main contributors of CDF for **internal events** are (UK EPR 2012, 15.1):

- Partial or total loss of cooling systems (LOCC): 22.3%
- Loss of primary cooling accident (LOCA): 20.3%
- Loss of offsite power (LOOP): 19.8%
- Primary transients (e.g. boron dilution): 15.4%
- Induced LOOP:..8.0%
- Anticipated transients without scram (ATWS): 4.0%
- Secondary transients: 3.4%
- Breaks on secondary side and steam line rupture with steam generator tube(s) rupture: 3.2%
- Reactor pressure vessel failure: 1.9%
- LOCA leading to containment bypass: 0.9%
- Steam generator tube(s) rupture (SGTR): 0.8%

Level 2 PSA

According to EDF/AREVA, the Level 2 PSA results show that the strong containment and dedicated severe accident mitigation measures of the EPR plant are efficient in reducing the frequency and magnitude of releases to the environment in the case of a severe core damage event (UK EPR 2012, 15.4).

The calculated large release frequency and the large early release frequency including all states and the spent fuel pool are summarised in Table 2 (UK EPR 2012, 15.4).

Table 2: Large release frequency (LRF) and large early release frequency (LERF)

LRF	7.69E-8/yr	10.8% of CDF
LERF	4.07E-8/yr	5.7% of CDF

Release Risk

The Level 2 PSA results were also presented in terms of “release risk”, which is the frequency of a given release multiplied by its magnitude (UK EPR 2012, 15.4). For the purpose of presenting, three isotopes which are known to be important for consequences are considered. These are Cs-137, I-131 and Sr-90.

Spent fuel pool accidents contribute significantly (86%) to the Cs-137 release risk. The second most contributing events (9%) are bypass events: interfacing system LOCAs and steam generator tube ruptures (SGTR).

Level 3 PSA

The Level 3 PSA estimates the likely impact of radiologically significant faults.

For the UK EPR, Safety Design Objectives have been adopted for risks to members of the public and workers which correspond to the Basic Safety Objective (BSO) risk targets from the HSE Safety Assessment Principles (SAP)

To meet the requirements of UK Health and Safety legislation, it is necessary to show that the radiation doses to workers and the general public due to EPR operation, taking into account the possibility of accidents, will be as low as reasonably practicable (ALARP). This requires that all reasonable measures are taken in the design, construction and operation of the plant to minimise the radiation dose received by workers and the general public, unless such measures involve disproportionate cost.

The UK release targets are expressed in terms of doses to persons on-site and off-site, and mortality risk. Although there is no release target as such, it can be instructive to refer to a large release frequency (LRF) or a large early release frequency (LERF). The LRF would be the sum of the frequencies of release categories exceeding some release threshold. A release of 100 TBq of Cs-137 is used as a guide to define "large release".

The total frequency of all release categories that fall within each dose band are presented for comparison with SAP Target 8 in tableTable 3. These results show that the calculated frequency in each dose band is consistently below the corresponding BSO and in most cases by more than an order of magnitude.

Based on these doses, an estimate of risk of death is made. The result is an estimate of total risk of death of 1.7×10^{-7}/yr which can be compared with the BSO of 1×10^{-6} /yr from SAP Target 7.

A screening approach, based on previous accident consequence assessments of UK power stations, is used to determine which release categories from the Level 2 PSA are likely to result in significant off-site consequences, i.e. in 100 or more deaths. The result is a risk of 8×10^{-8} /yr which is just below the BSO for Target 9 (10^{-7}/yr).

Table 3: The total frequency within each dose band

Effective Dose (mSv)	Total Frequency (per yr)	
	EDF and AREVA Target	Result
0.1 – 1.0	10^{-2}	1.4×10^{-3}
1.0 – 10	10^{-3}	1.3×10^{-5}
10 – 100	10^{-4}	1.2×10^{-6}
100 – 1,000	10^{-5}	1.5×10^{-7}
>1,000	10^{-6}	8.0×10^{-8}

4.2 Discussion

PSA results are of considerable value for the orientation of NPP designers and regulators (for example, to identify weak points in a reactor design).

On the other hand, the inherent limitations of PSA should not be forgotten – such analyses are beset with considerable uncertainties, and some risk factors are difficult to include in a PSA, or cannot be included at all:

- Unexpected plant defects or unforeseen physical or chemical processes could not be included in the PSA.
- Ageing phenomena can only be incorporated in PSAs in retrospect.
- Complex forms of human error are extremely difficult to model.
- Due to the complexity of an NPP, some accident initiators or sequences are simply bound to be overlooked or omitted.

In the following, the specific limitations of the UK EPR^{TM} PSA are described:

Out of Scope Items

The following items have been agreed with EDF and AREVA as being outside the scope of the GDA process and hence have not been included in the assessment by ONR.

- Any requirement on the PSA modelling that needs detailed design information or site-specific data
- Failure Modes and Effects Analysis (FMEA) for initiating event analyses
- Test frequencies of key components

List of Initiating Events (IEs) is not complete yet

According to ONR, there are a number of IEs identified related to plant systems that are not yet included in the PSA, due to lack of design detail. As mentioned above, the Failure Mode and Effect analysis (FMEAs) supporting IE derivation is out of scope.

Influence of the HVAC not considered

Loss of ventilation/room coolers (Heating, Ventilation and Air Conditioning, HVAC) during other accident sequences was also not included. The potential impact of the inclusion of HVAC based on the French EPR study could be a 6% increase in the CDF.

Generic LOOP not confirmed bonding

Regarding the initiating event frequencies, the generic loss of offsite power (LOOP) frequency is not confirmed (AF-UKEPR-PSA-019). Since LOOP situations have a considerable contribution to the CDF, this is important.

Review of the Modelling of the I&C required

There will be further development of I&C that will need to be incorporated into the PSA during post GDA phases. ONR requires that the modelling of the I&C in the PSA is reviewed. This should include explicit consideration of I&C based initiating events (including spurious signals) and the potential dependencies between such initiators and the safety mitigation systems and potential dependencies between the cues for operator action and signals used for the automatic I&C (AF-UKEPR-PSA-015). It is also required by ONR that future updates of the model explicitly include the actuators associated with the compact model, and also take account any CCF related to the actuators (AF-UKEPR-PSA-016).

Human Reliability Analysis (HRA) are not substantiated

The inclusion of pre-initiating Human Failure Events (HFEs) is incomplete. Only misalignment of manual valves is considered explicitly, motor operated and solenoid valves, automatically realigned on a system demand and manoeuvrable from the main control room (MCR), are not considered.

The SPAR-H approach for the Level 2 PSA is different from the approach used for the Level 1 PSA HEPs. This introduces an inconsistency into the analysis. The SPAR-H model is being used outside of the context for which it was developed.

The HRA in the UK EPR^{TM} PSA is largely assumption-based, with no underlying substantiation. ONR requires that substantiation for the Human Reliability Analysis (HRA) in the form of task analyses, procedures and training is provided to underpin the numerical Human Failure Event (HFE) values used in the PSA. The substantiation should include further consideration of pre-initiating HFEs and the potential for HFE dependencies (pre & post fault) (AF-UKEPR-PSA-017).

Common Cause Failures (CCF) are not considered appropriate

Only global CCF parameters are used, which provide no discrimination between different CCF groups for overall risk estimates (AF-UKEPR-PSA-025).

Scope of the internal and external hazards PSA is limited

- The potential dependency between combinations of extreme weather events (snow and wind) and consequential LOOP has to be taken into account and, if necessary, the PSA has to be amended (AF-UKEPR-PSA-028).
- Concerning external hazards only those leading to the loss of ultimate heat sink (LUHS) are effectively addressed in all PSA levels. The other external hazards have not been included due to their low occurrence frequency and consequences. This assumption has to be confirmed (AF-UKEPR-PSA-029).
- The use of an appropriate loss of ultimate heat sink frequency for the site is not confirmed yet (AF-UKEPR-PSA-030).
- Hazards such as internal explosion, turbine missiles and animal infestation are considered and, if necessary, have to be included in the PSA model (AF-UKEPR-PSA-031).
- Full scope Internal Fire PSA as well as a full scope Internal Flooding PSA has to perform as the detailed design evolves (AF-UKEPR-PSA-034; AF-UKEPR-PSA-036).
- Internal hazards that might be caused by a seismic event, such as fire or flooding, have to be analysed in detail and to be included in the PSA model supporting the Seismic Margin Assessment (SMA) (AF-UKEPR-PSA-037).

- The impact of seismic faults during shutdown has to be addressed in a consistent manner with other contributions to the risk during shutdown (AF-UKEPR-PSA-038).
- The scope of the PSA has to be expanded to include hazards such as fire and flooding during non power operating states (AF-UKEPR-PSA-002).
- Initiating faults due to intentional mal-operation or sabotage are not considered.
- Also, terror attacks such as an intentional aircraft crash are not considered.

Limitation of the PSA 2

An UK-EPR specific containment structural analysis has to be performed which addresses all potential modes of containment failure, including penetration and leakage failures (AF-UKEPR-PSA-042).

Principal concerns with the validity of the claimed risk figures

Off-site radiological consequence assessments have been carried out by EDF and AREVA for comparison with HSE Safety Assessment Principles (SAP) Targets 7, 8 and 9.

The total frequency of all release categories that fall within each dose band is consistently below the corresponding BSO and in most cases by more than an order of magnitude. However, ONR (2011a) criticised that the results do not include the specific calculation of early or late health effects based on organ doses. Based only on these doses, an estimate of risk of death is made.

The result is an estimate of total risk of death of 1.7 x 10^{-7}/yr which can be compared with the BSO of 1 x10^{-6} /yr from SAP Target 7. However, ONR (2011a) pointed out that the result does not include variability in meteorological conditions.

As EDF and AREVA's claims risks below the BSOs, ONR (2011a) emphasised principal concerns with confirming the validity of the claimed risk figures. The actual doses that have been estimated are not presented in the PCSR, and the results of the UK Health Protection Agency (HPA) study suggest that some may result in a number of deaths that significantly exceeds 100.

4.3 Conclusions

Generally, PSA results should only be taken as rough indicators of risk. All PSA results are beset with considerable uncertainties, and there are factors contributing to NPP hazards which cannot be included in the PSA

In the specific PSA of the UK EPR™, many factors are not included, because they are out of scope or not addressed appropriately (for example, Common Cause Failure (CCF)).

Therefore, for rare events, the probability of occurrence as calculated by a PSA should not be taken as an absolute value, but as an indicative number only. Hence, it is problematic in practice to reliably demonstrate the fulfilment of a probabilistic goal by PSA.

The claimed "practical elimination" of a large early release is not sufficiently demonstrated by the UK EPR^{TM} PSA. To practically exclude the occurrence of severe accidents requires a deep knowledge of a certain situation.[5]

Therefore, a conservative worst case release scenario should have been included in the EIA. As mentioned above, a source term, for example for an early containment failure or containment bypass scenario, should have been analysed as part of the EIA – in particular because of its relevance for impacts at greater distances.

[5] A situation is practically excluded when its occurrence is either physically impossible (deterministic prove) or can be seen as extremely unlikely with a high degree of trust (probabilistic prove).

5 POSSIBLE TRANSBOUNDARY IMPACTS

5.1 Treatment in the Environmental Statement (ES)

In EDF Energy´s Environmental Statement (ES, Volume 1 chapter 7.10) it is stated that, under regulation 24 of the EIA Regulation and the Espoo Convention and EU Directive 85/337/EEC, the Infrastructure Planning Commission (IPC, now Planning Inspectorate) "is obliged to form a view on the potential for transboundary impacts and consult with relevant European Member States.

The IPC Advice (June 2011) sets out how the IPC will meet its obligations in this regard. As detailed in Appendix 7E, EDF Energy has undertaken a screening exercise to determine the potential for transboundary impacts and concluded that no such impacts are likely." (EDF ENERGY 2011a)

According to the ES, Appendix 7E ("Assessment of Transboundary impacts"), the likely impacts determined through a thorough EIA do not extend beyond the county of Somerset and the Severn Estuary. Furthermore, EDF Energy pointed out that significant transboundary effects arising from the construction of new NPPs are not considered likely. Due to the robustness of the regulatory regime there is a very low probability of an unintended release of radiation (EDF ENERGY 2011a).

The Non Technical Summary (NTS) of the ES claims the potential for transboundary effects on other countries has been considered particularly in terms of emission and air quality impacts, marine water quality and ecology impacts on the Severn Estuary and radiological impacts. "Significant transboundary environmental effect arising from construction and operation of HPC are not considered likely." (EDF ENERGY 2011)

The Planning Inspectorate stated in the transboundary screening document (April 11, 2012): "Under Regulation 24 of the Infrastructure Planning (Environmental Impact Assessment) Regulations 2009 and on the basis of the current information available from the Developer, the Secretary of State thinks that the proposed development is **not likely** to have a significant effect on the environment in another EEA State." It is noted that the Secretary of State's duty under Regulation 24 of the Infrastructure Planning (EIA) Regulations 2009 continues throughout the application process (PLANNING INSPECTORATE 2012).

5.2 Discussion

Severe accidents at HPC with considerable releases of caesium-137 cannot be excluded, although their calculated probability is below 1E-7/a. There is no convincing rationale why such accidents should not be addressed in the Environmental Statement (ES); quite to the contrary, it would appear rather evident that they should be included in the assessment since their effects can be widespread and long-lasting and Austria can be affected.

Concerning safety and accident analysis, Austria should assess a possible future impact on its territory caused by accidental radioactive releases from the HPC to develop a catalogue of countermeasures.

5.3 Analysis of Transboundary Impacts

For the estimation of possible impact of transboundary emission of Hinkley Point C, calculation of the flexRISK project is used (FLEXRISK 2012). The flexRISK project modelled the geographical distribution of severe accident risk arising from nuclear facilities, in particular nuclear power plants in Europe. Using source terms and accident frequencies as input, for about 1,000 meteorological situations the large-scale dispersion of radionuclides in the atmosphere was simulated.

For each reactor, an accident scenario with a large release of nuclear material was selected. To determine the possible radioactive release for the chosen accident scenarios, the specific known characteristics of each NPP were taken into consideration. The accident scenarios for the dispersion calculation are core melt accidents and containment bypass or containment failure; the release rates are in the range of 20% to 65% of the core inventory of caesium.

The dispersion of radioactive clouds as a consequence of serious accidents in nuclear facilities in Europe and neighbouring countries is calculated for selected accidents with varying weather conditions.

Using the Lagrangian particle dispersion model FLEXPART both radionuclide concentrations in the air and their deposition on the ground were calculated and visualised in graphs. The total caesium-137 deposition per square-meter is used as the contamination indicator.

For a severe accident at Hinkley Point B, a caesium-137 release of 53.18 PBq is assumed. This source term is comparable with source terms of the UK EPR^{TM} calculated in the PSA 2 (see Table 4).

Table 4: Calculated caesium- 137 releases of severe accident at Hinkley Point C

Cs-137 release [PBq]	Containment failure mode
44.5	Isolation failure in-vessel recovery
45.8	Isolation failure (debris not flooded)
46.8	Isolation failure (debris flooded)
40.6	SGTR unscrubbed
438.0	Large ISLOCA, unscrubbed
1,780.0	Spent fuel pool accident

Figure 2 illustrates the calculated caesium-137 depositions after a possible severe accident at Hinkley Point B (FLEXRISK 2012a).

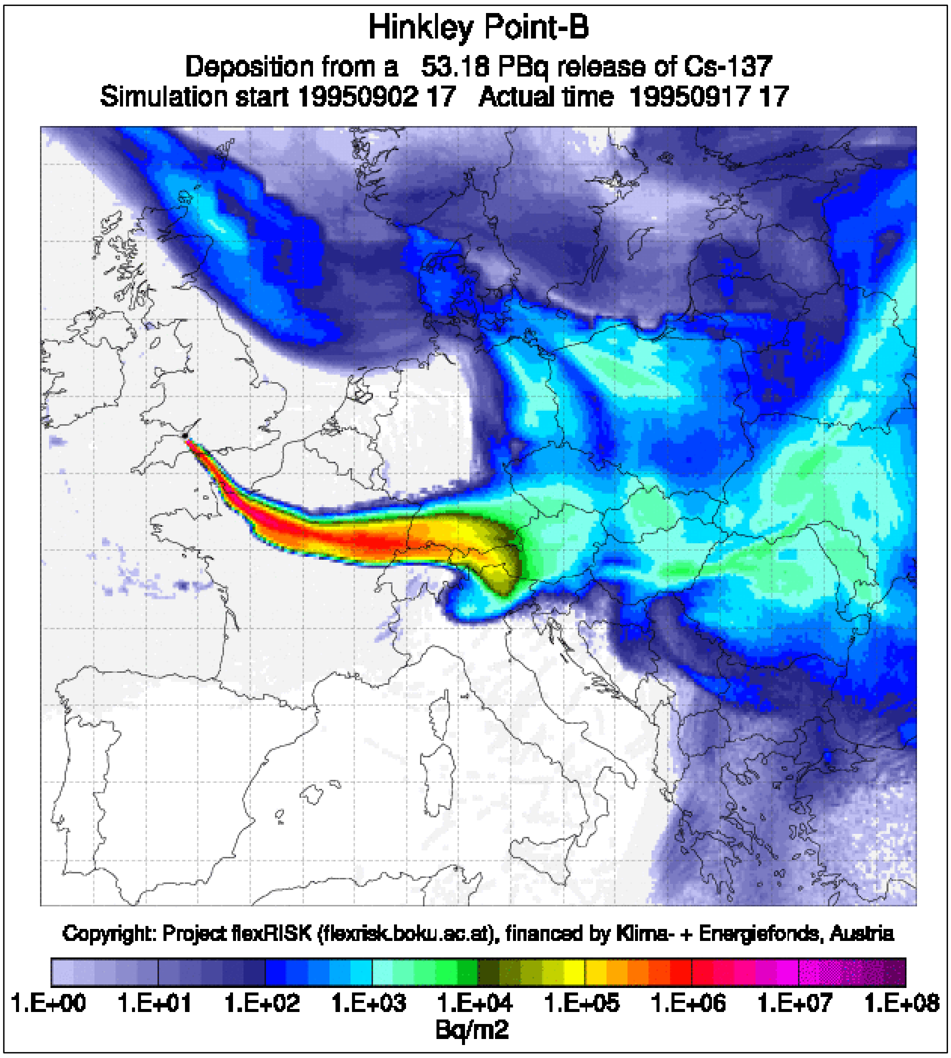

Figure 2: Caesium-137 deposition after a severe accident at Hinkley Point B

For a potential caesium-137 release of 53.18 PBq at Hinkley Point NPP under conditions comparable with those on September 2, 1995, a considerable contamination of the Austrian territory would result. Most parts show depositions of about 1E+03 Bq/m². However, in some areas the values are between 1E+04 and 1E+05 Bq/m², even up to 2E+05 Bq/m².

If a contamination of ground (and air) beyond certain thresholds can be expected, a set of agricultural intervention measures is triggered. These measures include earlier harvesting, closing of greenhouses and covering of plants, putting livestock in stables etc. For these measures, Austrian and German authorities defined a threshold for caesium-137 ground deposition of 650 Bq/m² (SKKM 2010; SSK 2008). These agricultural measures are quite complex and take some time. Reactions are especially difficult if there is only very little time between the onset of an accident and the arrival of the first radioactive clouds (FLEXRISK 2013b). For the calculated scenario, ground depositions of all areas are higher than this threshold, i.e. Austria would be highly affected.

It is important, however, to keep in mind that accidents with much more severe releases cannot be excluded. Other accident scenarios (failure of reactor pressure vessel at high pressure or containment bypass via uncovered steam generator tube leakage) can lead to caesium releases of more than 50% of the core inventory.

According to the PSA 2 results of the UK EPR™, a possible severe accident of the spent fuel pool could result in a release of 1,780 PBq, which is more than 30 times higher in comparison to the assumed release of Hinkley Point B.

Figure 2 shows that Austria and many other countries (including Germany and Switzerland) could be affected by a severe accident at Hinkley Point C.

5.4 Conclusion

The presentation of the results of the analysis of transboundary impacts of a potential severe accident at the Hinkley Point NPP site demonstrates that an impact on central European regions (including Austria) cannot be excluded. The results indicate the need for official intervention in Austria.

Moreover, the results emphasise the importance of a serious evaluation and discussion of the severe accident scenarios for Hinkley Point C in the framework of the transboundary EIA.

The information contained in the EIA procedure so far does not permit a meaningful assessment of the effects that conceivable accidents at Hinkley Point C could have on Austrian territory. The analysis of a severe accident scenario would close this gap and allow for a discussion of the possible impact on Austria. This should be taken into consideration before granting further permissions.

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7 GLOSSARY

AF .... Assessment Findings
ALARP .... As low as is reasonably practicable
ASN .... Autorité de Sûreté Nucléaire (French nuclear safety authority)
ATWS.................... Anticipated Transient without SCRAM (Reactor Shutdown)
Bq.......................... Becquerel
BSO .... Basic Safety Objective (in SAPs)
CCF....................... Common Cause Failure
CCWS .... Component Cooling Water Systems
CDF .... Core Damage Frequency
CGCS .... Combustible Gas Control System
CHRS .... Containment Heat Removal System
CMSS .... Core Melt Stabilisation System
DAC .... Design Acceptance Confirmation
DBA .... Design Basis Analysis
DCH .... Direct Containment Heating
DCO .... Development Consent Order
DG......................... Diesel Generator
ECCS .... Emergency Core Cooling System
EDF and AREVA .. Electricité de France SA and AREVA NP SAS
EDG .... Emergency Diesel Generator
EEA....................... European Economic Area
EFWS.................... Emergency Feedwater System
EIA .... Environmental Impact Assessment
EOP .... Emergency Operating Principles
EPRI...................... Electric Power Research Institute (USA)
ES .... Environmental Statement
ESWS .... Essential Service-Water System
FlexRISK .... Flexible Tools for Assessment of Nuclear Risk in Europe
FMEA .... Failure Modes and Effects Analysis
GDA .... Generic Design Assessment
GRS .... Gesellschaft für Anlagen- und Reaktorsicherheit (German)
HF .... Human Factors
HFE....................... Human Failure Event
HFIP .... Human Factors Integration Plan
HPC .... Hinkley Point C
HPME .... High Pressure Melt Ejection
HRA .... Human Reliability Analysis
HSE .... Health and Safety Executive
HVEA .... Heating, Ventilation and Air Conditioning
I&C .... Control and Instrumentation
IAEA .... International Atomic Energy Agency
IDAC .... Interim Design Acceptance Confirmation

IE ....Initiating Event
IPC ....Infrastructure Planning Commission (now Planning Inspectorate)
IRWST ....In-Containment Refuelling Water Storage Tank
ISLOCA ....Interfacing System LOCA
IVR ....In-Vessel Retention
LERF ....Large Early Release Frequency
LOCA ....Loss of Coolant Accident
LOCC ....Loss Of Cooling Chain
LOOP ....Loss of Off-Site Power
LUHS ....Loss of Ultimate Heat Sink
MCCI ....Molten Core Concrete Interaction
MDPE ....Multinational Design Evaluation Programme
MOX ....Mixed Oxide Fuel
ND ....The (HSE) Nuclear Directorate
NNB ....New Nuclear Build (as in NNB GenCo)
NPS ....National Policy Statements
NTS ....Non Technical Summary of ES
ONR ....Office for Nuclear Regulation
OSSA ....Operating Strategies for Severe Accident
PAR ....Passive Autocatalytic Recombiners
PBq........................Peta Becquerel = 10E+15 Bq
PCER ....Pre-Construction Environment Report
PDS ....Primary Depressurisation System
POS.......................Plant Operating State
PSA ....Probabilistic Safety Analysis
PWR ....Pressurised Water Reactors
RCS ....Reactor Coolant System
RIT ....Royal Institute of Technology
RPV ....Reactor Pressure Vessel
RRC.......................Risk Reduction Category
SAP ....Safety Assessment Principles
SARNET ....European Severe Accident Research Network
SBLOCA ....Small Break Loss of Coolant Accident
SBO.......................Station Black-Out
SBO-DG ....Station Black-Out Diesel Generator
SFP ....Spent Fuel Pool
SG ....Steam Generator
SGTR ....Steam Generator Tube Rupture
SMA.......................Seismic Margins Assessment
SSA ....Strategic Siting Assessment
STUK.....................Finish Nuclear Safety Authority
UOX.......................Uranium Oxide Fuel
US NRC ....Nuclear Regulatory Commission (United States of America)
WENRA ....Western European Nuclear Regulators' Association

ENVIRONMENT AGENCY AUSTRIA umweltbundesamt

Umweltbundesamt GmbH
Spittelauer Lände 5
1090 Wien/Österreich

Tel.: +43-(0)1-313 04
Fax: +43-(0)1-313 04/5400

office@umweltbundesamt.at
www.umweltbundesamt.at

EMAS
Geprüftes
Umweltmanagement
REG.NO. AT- 000484

ISBN 978-3-99004-217-5