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Reactor Safety Study Applied to the Forsmark 3 Power Plant - A Sensitivity Analysis

G Ericsson and L I Tirén
AB ASEA-ATOM, Västerås, Sweden

Abstract

A reactor safety study of the Forsmark 3 BWR power plant has been carried out for the purpose of calculating core melt probabilities using WASH-1400 methods. A sensitivity analysis shows that the calculated core melt probability is changed by approximately a factor of 10 depending on assumptions made with respect to the probability of human error. The importance of the availability of off-site power and the influence of common cause failure is also discussed.

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1. Introduction

On behalf of the Swedish Energy Commission ASEA-ATOM has carried out a reactor safety study of the Forsmark 3 nuclear power plant, which incorporates an ASEA-ATOM reactor of the standard BWR-75 design. A specific reactor safety study of this plant was considered to be of interest not only to Forsmark 3 but also to future nuclear power plants of the standard BWR-75 design.

Methods used in the Reactor Safety Study, WASH-1400 (1), conducted by Professor Norman Rasmussen were applied. Our task, however, was limited to the calculation of probabilities of a core melt in Forsmark 3. No independent calculation of the release rates after core melt was made. It was reasoned that the release categories given by WASH-1400 are also valid, at least approximately, in the Forsmark 3 case. The study involved an effort of about 5 man-years. The work was summarized in a report and this was published by the Energy Commission in February, 1978, along with critical comments and ASEA-ATOM's replies to these comments (2). The full report was written in Swedish. A summary of the study was submitted to the recent NUCLEX technical meeting in Basel (3).

A sensitivity analysis has now been performed to indicate to what degree the results depend on various assumptions and how the results would change when certain individual probability figures are altered. In this analysis attention was focused on issues which were either identified as having a strong influence on the final result or which have been debated in the context of WASH-1400:

- Availability of off-site power
- Emergency cooling functionability (ECF)
- Common cause failures (CCF)
- Human error

2. Brief Description of the Forsmark 3 Nuclear Power Plant

Forsmark is located on the Baltic coast 120 kilometers north of Stockholm. At present, the site includes three nuclear units in various stages of completion. Forsmark 1 is currently undergoing hot commissioning tests. Forsmark 2 is a repeat of Forsmark 1 and is scheduled for commercial operation in 1980. Construction work on Forsmark 3 has begun. The reactor pressure vessel is presently being manufactured at Uddcomb Sweden AB. All three units incorporate BWR reactors of the ASEA-ATOM design. STAL-LAVAL Turbin AB, Sweden, is responsible for the delivery of the turbines.

Forsmark 3 has an electrical rating of approximately 1000 MW, corresponding to 3000 MW thermal power. Its design is a development of that of the previous units. General information about the plant is given in reference 4. The reactor core consists of 700 fuel assemblies of the 8x8 fuel rod type. Coolant flow is provided by eight internal recirculation pumps (see Figure 1). The primary containment is made of prestressed concrete with an embedded steel liner (see Figure 2). The condensation pool at the bottom of the containment acts as a heat sink on steam relief from the reactor.

The emergency core cooling and residual heat removal systems are divided into four separate subsystems, which are made redundant in such a way that, in most cases, two subsystems have an adequate capability for the intended safety function. The various subsystems have been separated physically in order to prevent the occurrence of common cause failures as a result of a fire or a missile. These principles are described in more detail in references 5 and 6.

3. Summary of Results of the Core Melt Analysis

The safety study for Forsmark 3 consists mainly of a probabilistic core melt analysis. The study began with a qualitative analysis in order to assess the applicability of WASH-1400 results. It was found at this stage that most accident sequences identified as significant or dominant in WASH-1400 might be correspondingly significant in the case of Forsmark 3, at least after minor modifications in some cases. It was also concluded that compound events and unlikely initiating events not found to be important in WASH-1400 would not make a significant contribution in the Forsmark 3 case either. These conclusions implied that the quantitative analysis could be limited to accident sequences starting with "likely initiating events" and loss of coolant accidents (LOCAs). The following transient events and accidents underwent a detailed analysis:

- Failure to shut down the reactor (TC)
- Failure to provide make-up water to the reactor (TQUV,TPQUV)
- Overpressurization transients
- Failure to remove residual heat to cooling water canals (TW)
- Large LOCA
- Small LOCA

In the course of the study it was found that the probability of a core melt accident in Forsmark 3 due to random failures is negligible. Attention was therefore focused on common cause failures, which were treated in a conservative manner. The net core melt probability was calculated to be 3.1×10^{-6} per year (point estimate). Significant contributions to this value are illustrated in Figure 3. The corresponding core melt probability value of the WASH-1400 reference plant is approximately 2.5×10^{-5} per year. The lower probability level in Forsmark 3 is primarily a consequence of reduced failure rates with respect to reactor shut-down (TC) and residual heat removal (TW). These reductions are attributed mainly to the dual shut-down mechanism and the redundancy of residual heat removal systems and associated power sources, respectively. For details and explanations, see references 2 and 3.

4. Sensitivity Analysis of Core Melt Probabilities

In general the approach taken in the sensitivity analysis was to evaluate the result of changes of probability numbers on the event tree level. Since the probability numbers sometimes depend on various assumptions about failure rates of systems or components care was taken to include possible interactions, which would imply that several probability numbers are affected by the change of a certain assumption.

4.1 Availability of off-site power

When analyzing the probability of failure of residual heat removal to the ultimate heat sink, it was found that the availability of auxiliary power strongly affects the calculated probability figure. In Forsmark 3 auxiliary power is exclusively supplied by electrical sources. There are no pumps driven directly by a steam turbine. Emergency power is supplied by four on-site diesel-generator sets, one or two of which are sufficient in various situations involving loss of off-site power. However, the fact that off-site power sources include gas-turbine-driven generators at Forsmark improves the overall availability figure for auxiliary power. The probability of failure of the gas-turbines was estimated to be 10^{-2} per demand, the failure being dominated by a common generator breaker for both generators. To illustrate the sensitivity of assumptions about off-site power, a calculation was performed in which the gas-turbine generators were not credited at all, although no technical reason for discounting them was identified. In this case the probability of failure of long-term residual heat removal increased from $P_W = 3 \times 10^{-8}$ (TW, see Figure 3) to $P_W = 1.7 \times 10^{-6}$ per year. This figure is composed as follows:

$$P_W = P_1 \times P_2 \times P_3 \times Q = 1.7 \times 10^{-6}$$

where P_1 = probability of total loss of off-site grid, 0.2 per year

P_2 = probability of failure of plant main generator to provide house load = 0.3 per demand

P_3 = probability of non-return of grid within 5.5 hours, or failure to recover turbine condensor = 5.5×10^{-2}

Q = non-availability of residual heat removal systems including on-site power = 5.1×10^{-4} per demand.

The availability of the gas-turbine generators has a less strong effect on the short-term cooling of the reactor, mainly because the loss of short-term cooling is dominated by other causes. As a result of discounting the gas-turbine generators the overall core melt probability increased from 3.1×10^{-6} to 5.6×10^{-6} per year. For details about the reliability analysis of the auxiliary power supply, see reference 7.

4.2 Emergency cooling functionability

The functionability of ECCS upon a loss of coolant accident was discussed in WASH-1400; comments and criticism were dealt with primarily in Appendix XI of that report. This item refers not to the initial operability of the systems but to the possible loss of functionability (ECF) due to the distortion or structural failure of reactor internals as a result of dynamic forces experienced in connection with a large LOCA.

In Appendix V of WASH-1400 estimates of the probability of failure of the ECF range from 10^{-2} to 10^{-5} per occurrence for large LOCA. In our study the geometric mean value of these extremes was used, i.e. 3×10^{-4} per occurrence. Replacing this figure by the upper limit, 10^{-2}, changes the overall core melt probability in Forsmark 3 from 3.1×10^{-6} to 4.1×10^{-6} per year.

A higher probability level than 10^{-2} for ECF failure is not considered as realistic, since reactor internals are designed to withstand dynamic loads induced by a LOCA.

4.3 Common Cause Failures (CCF)

Particular attention was focused in the study on the problem of common cause failures (CCF). The importance of this item in probability analyses of the present kind has been stressed by several groups (8, 9, 10). In the design of Forsmark 3, efforts were made to minimize the likelihood of failures due to a common cause. In the core melt analysis, specific mechanisms that might involve a risk of common cause failures were identified in a few cases only. However, the possibility of material defects, the use of similar components or designs in many parts of the plant, and the risk of human error are regarded as implicit causes of failures of this nature.

Consequently, CCF contributions were always assumed to be present and were calculated on the component level as well as on system and inter-system levels. In most cases this was done by using the geometric mean value of the calculated probabilities, assuming complete independence and complete dependence, respectively. The result of introducing this CCF contribution is illustrated for the auxiliary feedwater system, which consists of four identical trains or subsystems. In the following table the failure probability (per demand) is given in the case for which 1, 2 and 3 of the four subsystems are required to function.

Functioning requirement No. of parallel subsystems (trains)	Independent failures only	CCF contribution included
1-of-4	1.4×10^{-8}	3.5×10^{-6}
2-of-4	1.0×10^{-5}	1.0×10^{-4}
3-of-4	2.5×10^{-3}	2.9×10^{-3}

These results are typical of emergency cooling systems, which in Forsmark 3 are consistently divided into four subsystems. In the safety analysis report the proper functioning of two of the four subsystems is generally credited. In many situations, however, realistic assumptions and calculations of fuel cladding temperature show that one subsystem is sufficient. For such situations it is overconservative to use the prescriptions of the licensing calculations as well as a large CCF contribution on the system level.

4.4 Human error and negligence

The argument about human fallibility has been much used by some nuclear critics to make the point that safety studies of the kind discussed here are incomplete or inconclusive. In fact the Rasmussen Report as well as the present core melt analysis include a thorough evaluation of the risk of human errors. Two types of failure that may be caused by human error or negligence can be distinguished:

- Safety functions fail due to systematic miscalibration of logic equipment in the reactor protection system (RPS).

- A safety function should be initiated by the operator but he fails to take action.

Systematic miscalibration of RPS equipment was treated in the same basic manner as in WASH-1400. The contribution to the probability of a core melt originating from miscalibration was assumed to be 1.9×10^{-6} per demand, the failure implying that automatic reactor trip (scram) does not occur. Since the hardware contribution to RPS failure is small in Forsmark 3 due to the presence of dual shut-down systems, the miscalibration contribution becomes quite dominant. It should be possible to eliminate this failure mechanism once it has been identified. Its elimination would reduce the probability of reactor shut-down failure from 2.5×10^{-6} to 0.6×10^{-6}. The resulting change in the total core melt probability would be from 3.1×10^{-6} to 1.2×10^{-6} per year in Forsmark 3.

With regard to the operator failing to take action (human negligence), it was found that such an event did not significantly affect LOCA sequences. This result partly reflects the Swedish 30-minute rule, which stipulates that important safety-related actions needed within 30 minutes upon accidents must be automatic.

In transient sequences a set of operator actions of importance were identified. The required actions were divided into two categories, a) those which have to be taken within 10 to 30 minutes after the initial event and b) those which must be taken within about 24 hours. The two categories include:

Category a)

- Initiate boron system upon control-rod insertion failure
- Actuate manual shut-down upon reactor protection logic system failure
- Restart feedwater system upon initial loss of feedwater flow
- Actuate reactor depressurization system upon auxiliary feedwater system failure
- Initiate reclosure of relief valve upon failure to reclose automatically

Category b)

- Restore feedwater system upon initial loss of main heat sink (turbine condensor) or loss of off-site power

- Restore turbine condensor as heat sink upon initial loss.

In the report (2) human failure rates were derived from WASH-1400 data. A sensitivity study was made to investigate the effect of assuming, for category a) and b) events, respectively:

1) The probability of operator failure = 0

2) The probability of operator failure = 1

The following results were obtained:

1) If the operator is always assumed to take correct action upon the initial event, the total core melt probability is reduced from 3.1×10^{-6} per year to 0.6×10^{-6}. The remaining core melt probability is dominated by LOCA sequences, including pressure vessel rupture.

2) If the operator is assumed to take no action within the first 30 minutes upon the initial event, the core melt probability is increased to 3.9×10^{-5} per year. This probability is dominated by his failure to initiate manual shut-down and his failure to take action to supply feed-water to the reactor. This need arises upon initial loss of the normal feedwater system combined with the unlikely complete failure of automatic start of the auxiliary feed-water system.

Assuming reactor shut-down and short-term cooling to be successful, core melt can still occur due to continued operator passivity. Several sequences are involved but are dominated by the case of initial loss of off-site power. In this case residual heat is transferred from the reactor to the containment condensation pool. The pool is cooled by the emergency residual heat removal chain to the sea. If the cooling chain does not operate, the pool water temperature will exceed 95°C after about 5.5 hours. At this stage emergency core cooling pumps are assumed to become inoperable due to the loss of their NPSH. In the meantime, however, the operator can make use of the return of offsite power in order to supply water from the normal feedwater system and to remove residual heat by other systems not connected to the on-site sources. These actions are here assumed not to be taken and so core melt will eventually occur.

Using the value $Q = 5.1 \times 10^{-4}$ for the non-operability of the residual heat removal chain (cf section 4.1), the probability of core melt in this sequence was estimated to be 3.1×10^{-5} per year. The change in total core melt probability is from 3.1×10^{-6} to 3.6×10^{-5} per year.

5. Summary and Conclusions

The results of the sensitivity analysis are summarized in the table below. It should be pointed out that a discussion of absolute values of probability is beyond the scope of this work. The study was undertaken on the basis that data and methods used in WASH-1400 are in principle accepted. Given this framework, the relative values of probabilities and the changes in these values when subject to altered assumptions are thought to be of considerable interest.

The analysis indicates that Forsmark 3 possesses design features which render plant safety relatively insensitive to the effects of human negligence. If the maximum impact of omitted manual action is assumed, this will increase the overall probability of a core melt by approximately a factor of ten.

Parameter change	Core melt probability per reactor year
Reference value	3.1×10^{-6}
Power from gas-turbine generators not credited	5.6×10^{-6}
Emergency Core Cooling Function failure probability = 10^{-2}	4.1×10^{-6}
No systematic miscalibration occurs	1.2×10^{-6}
No operator failure occurs	0.6×10^{-6}
No operator action until 30 minutes after initial event	3.9×10^{-5}
Short-term cooling successful but no operator action during first 24 hours	3.6×10^{-5}

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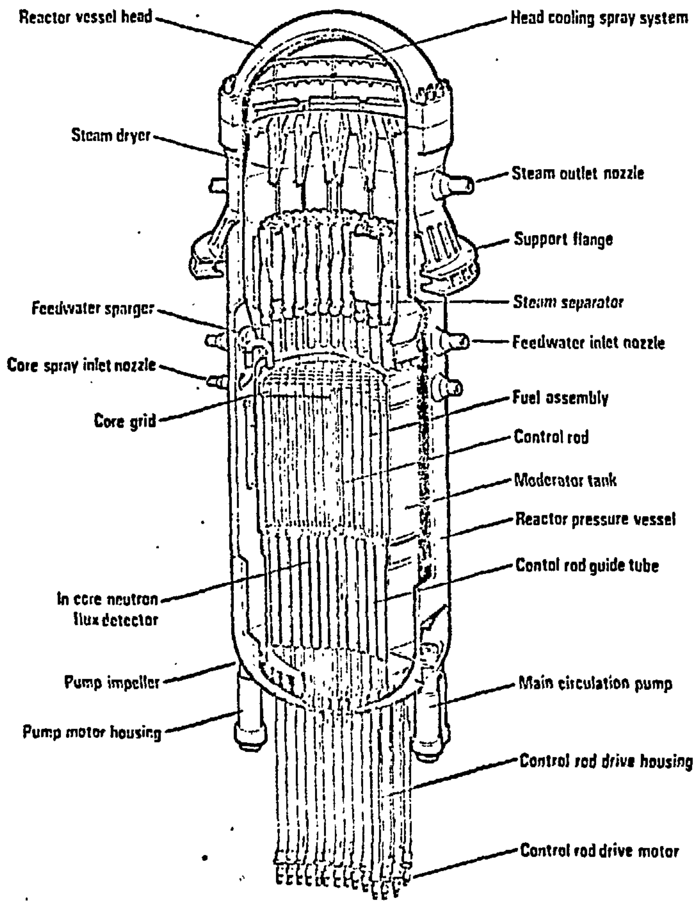


Figure 1. FORSMARK 3 - Reactor

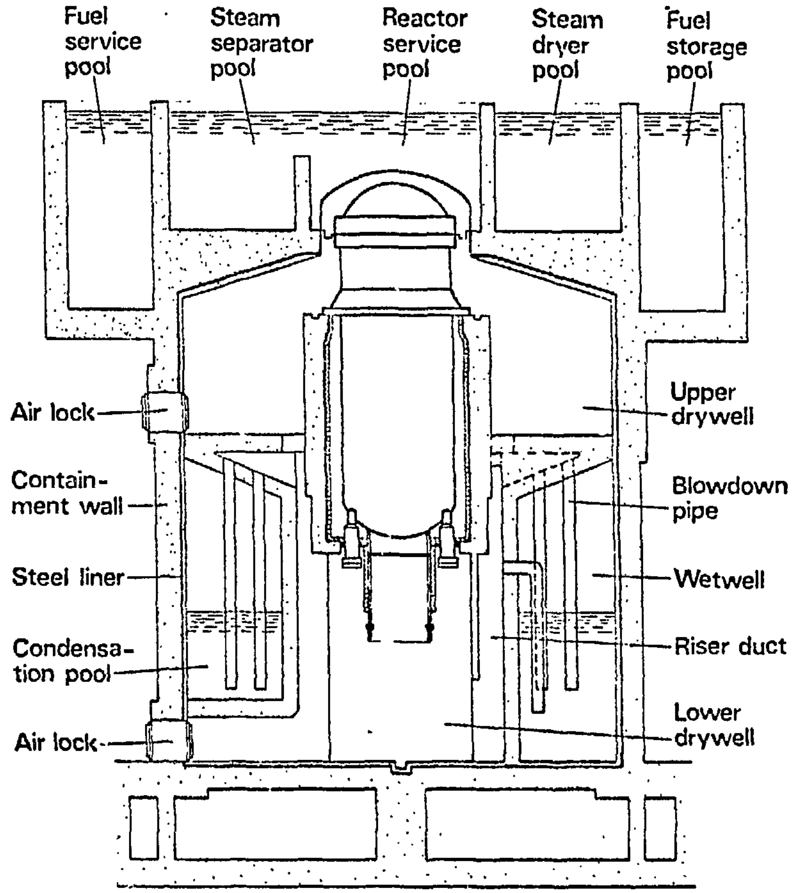
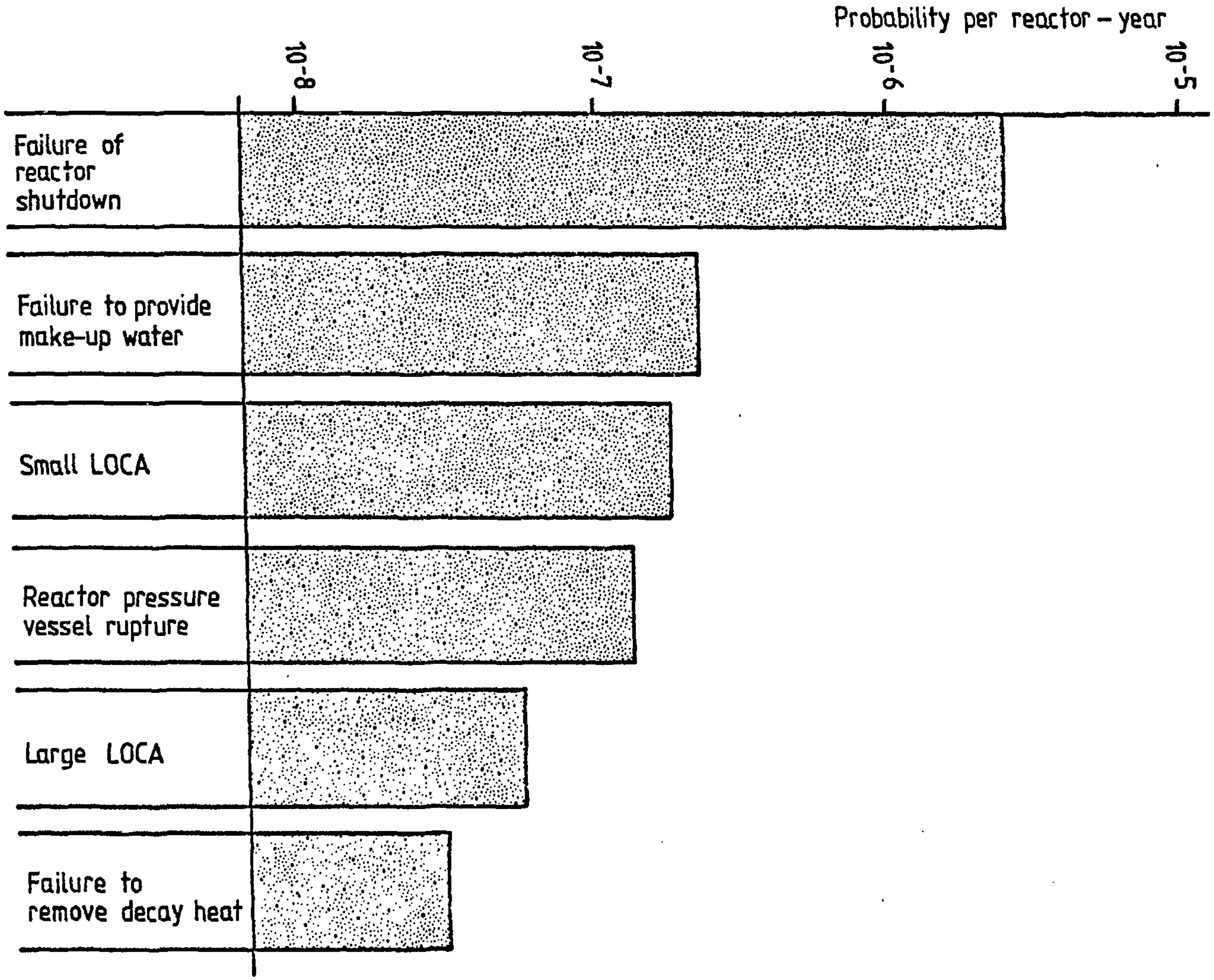


Figure 2. FORSMARK 3 - Primary Containment



Figur 3 Probability of core melt in Forsmark 3

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