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Study of the Accidental Risk of the German Fast Breeder Prototype Reactor SNR-300

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Summary

A fact-finding committee of the German Federal Parliament in July 1980 recommended to perform a "risk-oriented study" of the SNR-300, the German 300 MWe fast breeder prototype reactor being under construction at Kalkar. The main aim of this study was to allow a comparative safety evaluation between the SNR-300 and a modern PWR, thus to prepare a basis for a political decision on the SNR-300. Methods and main results of the study are presented in this paper.

In the first step of the study six groups of accidents have been identified which may initiate core destruction. By reliability analyses, expected frequency of each group has been estimated.

Conditional probabilities for conceivable reactor tank failure modes have been analysed. Tank failure after core destruction leads to release of energy and radioactive material into the containment system. Such accident sequences have been pursued further. Based on a number of core destruction initiators and tank failure modes and various combinations of success and failure states of the containment systems, detailed calculations of different containment scenarious were carried out. From the results of the plant systems analyis, five release categories have been defined.

Possible effects of external events and releases of radioactivity from the spent fuel storage pool have also been analysed.

In order to quantify the degree of uncertainty of the calculated frequencies, subjective probability distributions of fixed, but inaccurately known quantities have been propagated through the calculations.

Using the release categories as input, accident consequences were calculated for the site Kalkar.

Though the uncertainty bandwidths for the accident frequencies estimated in the SNR-300 analysis are much wider than for the PWR, the analysis indicates that frequencies of severe accidents, and consequences, are smaller for the SNR-300 than for the PWR as analysed in the "German Risk Study".

1. Charter of the Study

A fact-finding committee on "Future Nuclear Energy Policy" of the German Federal Parliament in July 80 recommended to perform a study on the accidental risk of the SNR-300, the German fast breeder prototype reactor being under construction at Kalkar. The main aim of this study was to perform a comparative safety evaluation between the SNR-300, and a modern pressurized water reactor of the type Biblis B, taking into account probabilistic aspects, in order to prepare a basis for a political decision on the SNR 300.

The Federal Minister for Research and Technology awarded a contract to GRS, the Gesellschaft für Reaktorsicherheit, to perform this study. Essential parts of the study have been performed together with the Nuclear Research Center Karlsruhe, especially with respect to analysis of core disruptive accidents and accident consequences. GRS has also been supported by a number of other institutions and persons. The study has been completed in April 1982 and published, after presentation to and discussion with the fact-finding committee, in October 1982.

2. The Plant

The SNR-300 is designed as a 3-loop plant, with a fast core, and sodium as coolant. The core is equipped with two independent and diverse shut down systems, which are actuated by independent protection systems.

The thermal energy generated in the core is transferred by three parallel primary sodium circuits, located in separate cells, via three intermediate heat exchangers per loop to three parallel secondary sodium circuits. Via evaporators and superheaters heat is transferred to the tertiary water-steam circuit.

Nearly all sections of the primary circuits are installed above the emergency coolant level in the reactor tank, thus preventing syphoning in case of component leakage. Beneath low sections of the primary system, mainly at pumps and heat exchangers, cavities are installed which limit the amount of possible coolant loss.

The decay heat is normally removed via the three main heat transfer loops. Decay heat can be removed also via heat exchangers submerged in the reactor tank. Even in case of failure of all active components of the cooling systems, the core can be sufficiently cooled after shutdown by passive decay heat removal.

These safety features make a core destruction highly unlikely. However, since the SNR-300 is a prototype reactor and since operating experience with sodium cooled breeder reactors is limited, the SNR-300 has been designed to prevent dangerous releases of radioactivity into the atmosphere even after core destruction accompanied by a release of mechanical energy.

This implies, that reactor tank and primary system are designed to withstand a mechanical energy release up to 370 MJ, and that the plant will be equipped with a double containment system.

3. Methods and Results of Analysis

3.1 Core destruction initiators

In the first step of the study, accidents which may initiate core destruction have been identified (fig. 1). The core can be destroyed either through rapid nuclear power excursion, usually called "core disruptive accident", or through slow melt down. Core destruction through nuclear power excursion can only occur through disturbances, which add significant amounts of reactivity within short time, and if this combines with failure to scram. In this case, power increases rapidly until the excursion is limited by inherent physical mechanisms. There are three potential causes for such reactivity increases, namely displacement of sodium from core, unprotected reactivity addition, and fuel pin failure propagation.

Potential causes of displacement of sodium could be gas bubbles in the core area, or sodium boiling. While entering of gas bubbles into the core is prevented by passively operating measures, prevention of sodium boiling requires active measures. Sodium boiling may occur if there is insufficient coolant flow through the core or insufficient heat removal to the heat sink, and if this combines with failure of reactor scram. These events correspond to an "Unprotected Loss of Flow" (ULOF) an "Unprotected Loss of Heat Sink" (ULOHS), respectively.

Unprotected reactivity addition and fuel pin failure propagation have been analyzed in the study. Their effects are covered by the treatment of ULOF and ULOHS; they are not further discussed in this paper.

The core could also be destroyed after reactor scram, due to an imbalance between the decay heat generated in the core, and heat removed from the primary coolant. Possible causes are loss of active and passive decay heat removal capability without loss of coolant, or loss of decay heat removal capability in case of primary coolant boundary leakage, leading to partial loss of coolant.

It should be pointed out that the core destruction initiators mentioned in the above sections (groups 1 to 6 in fig. 1), comprise all conceivable courses, potentially leading to core destruction. Each of these core destruction initiators itself can be caused by various accident initiators.

The analysis investigated the various modes under which accident initiators can lead to one of the core destruction initiators. Expected frequencies of such sequences were estimated. Five classes of accident initiators have been defined. They are representative for all conceivable individual accident initiators.

Event trees have been developped for the following accident initiators: (general) transient, loss of normal ac-power, leak in the primary coolant system, reactivity addition and degraded local cooling conditions.

The systems important to prevent core destruction are the reactor shut down systems and the decay heat removal systems.

Looking at the reactor shutdown system, failure of the signal to be actuated (either automatically by the reactor protection system or manually) has been distinguished from failure of the mechanical scram systems.

If the signal fails after occurence of a transient the reactor remains in the operating state. If the heat removal capability is impaired by the initiating event, the reactor temperature begins to rise, and can reach values where the main coolant pumps fail, unless limiting effects due to inherent physical effects or manual measures take place. If the pumps fail, the event sequence ends at core destruction, corresponding to an ULOHS. If the signals are present and the primary coolant pumps are shut off as designed (or pump coast down is the initiating event), but both mechanical scram systems fail, then the event sequence leads to an ULOF.

If the reactor is shut down, decay heat has to be removed. If decay heat removal systems fail, the event sequence leads to "failure of decay heat removal system with scrammed reactor".

3.2 Reliability analysis

Expected frequencies of core destruction initiators have been calculated by reliability analyses. The most important results are shown on fig. 2.

The table shows frequencies of initiating events and failure probabilities of systems needed to keep the reactor in safe state. From these values, the expected frequencies of the core destruction initiators ULOF (group 1), ULOHS (group 2) and "Loss of decay heat removal" (group 3) are determined. Other groups are not shown since their contribution is insignificant.

Dominant initiating events for ULOF and ULOHS are general transients, estimated to occur 12 times a year. With an estimated failure probability of 10^{-7} per year of the mechanical scram system, an expected frequency of $1.2 \cdot 10^{-6}$ is obtained for the ULOF. This case is used as the basis case of the subsequent accident analysis.

In the analysis of the ULOHS credit has been taken for inherent physical effects or manual actions which can keep the reactor in safe state even after failure of the automatic reactor shut down signal. An expected frequency of $1.2 \cdot 10^{-7}$ per year has been estimated for an ULOHS. In this case the coolant temperature is about 650 °C at the onset of core destruction, compared to a nominal maximum temperature of 550 °C in the coolant. As far as this high temperature influences the course of the accident, it has been considered in the subsequent analysis.

The following initiating events contribute to the frequency of "Loss of Decay Heat Removal": loss of normal ac-power, steam generator failure, and the general case of decay heat removal. If the accident is initiated by loss of normal ac-power or by steam generator failure, the availability of the decay heat removal system is already impaired by the accident initiator.

The reliability analysis took into account the passive decay heat removal capability. A conditional probability of 10^{-2} has been estimated for failure of passive decay heat removal. All together, an expected frequency of $3 \cdot 10^{-7}$ per year has been estimated for this core destruction initiator, including a small contribution from the failure of emergency core cooling in the case of primary system leakage.

3.3 Accident analysis

The accident analysis deals with the course of core destructive accidents. The first step of the accident analysis investigates the potential failure modes of the reactor tank and guard vessel. Core destruction may be accompanied by the release of significant amounts of mechanical energy. The primary coolant system of the SNR-300 is designed to withstand mechanical energy releases up to 370 MJ. Other design features like the submerged heat exchangers make it possible to cool the molten core inside the reactor tank, so that melt-through of the tank can he prevented.

Although these features make a destruction of the tank extremely unlikely, the accident analysis had to estimate the conditional probability that releases of mechanical energy beyond the design value of 370 MJ occur, and the conditional probability of reactor tank failure due to mechanical or thermal overload at energy releases below 370 MJ.

According to present understanding the amount of mechanical energy possibly released in such accidents is expected to be far below the design value of 370 MJ. However, for the probabilistic risk assessment it was necessary to arrive at a quantification of probabilities of energy release exceeding certain values.

In order to put this quantification on a basis as broad as possible, an international expert questionning was conducted on phenomena influencing the release of mechanical energy after an ULOF.

Incorporating the results of this action, subjective complementary probability distributions for release of mechanical energy have been obtained. The totality of these distributions reflects the degree of uncertainty exhibited in the experts' answers. A reference complementary probability distribution was also generated.

This analysis resulted in a 0.95 conditional probability that the release of mechanical energy in an ULOF is less than 50 MJ. The conditional probability for exceeding 400 MJ, which is about the design value, was estimated to be three tenth of a percent. The analysis also showed that there is a conditional probability of about one half that there is no mechanical energy release at all in an ULOF.

In case of a mechanical energy release exceeding 400 MJ, failure of the plug system due to sodium impact has been assumed. This impact can be excluded with the "failure of decay heat removal", where the sodium level is low at initiation of core destruction. For the ULOHS the probability of plug system failure is slightly higher than for the ULOF (see fig. 3), since structural integrity is weakened due to higher temperatures.

The reactor tank could fail mechanically due to an annular rupture of the tank barrel, resulting in a failure of the guard vessel and of the core catcher through impact. Heat removal from the containment would be impaired in this case. This failure mode may be expected if, at the time of a mechanical energy release, the reactor tank had been at temperatures significantly above the design value for an extended period of time. This is mainly the case with ULOHS and with "failure of decay heat removal."

The reactor tank fails due to thermal overload if molten material deposited on internal structures is insufficiently cooled. Conditional probability is seven percent after ULOF and about 1 resp. 0.5 for the other groups.

As fig. 3 shows, there is an overall conditional probability of seven tenth that the tank remains intact after core destruction. The conditional probability for mechanical failure is about one tenth. Investigations on the failure probability of components under extreme load conditions are described more detailed in [3].

Tank failure after core destruction leads to release of energy and radioactive material into the containment system. Such accident sequences have been pursued further in the risk analysis.

If the plug system is destroyed by high mechanical energy release it is assumed that the cover between inner and outer containment is lifted and sodium and core material are spilled directly into the outer containment. Through sodium fire the pressure increases quickly, leading to containment failure because of overpressurization within a few minutes, and subsequently to massive release of radioactivity into the environment. There is very little influence of the state of most containment systems on the course of this extreme accident sequence.

Other accident sequences, where the inner containment is still intact at the time of tank failure, would require a detailed consideration of the state of the containment systems. Since it was deemed unwarrented for a risk-oriented study to investigate a huge number of different events, distinguished by various combinations of success or failure states of the containment systems, the analysis was restricted to the simulation of some typical sequences, including those with the most severe consequences. With the exception of the functions "isolation between inner and outer containment", "isolation of outer containment", "exventing system", and "nitrogen natural circulation inside the inner containment", all containment systems pessimistically have been assumed to be in the failed state (except for one reference case calculation).

Based on a number of core destruction initiators and tank failure modes and various combinations of success and failure states of the containment systems mentioned above, detailed calculations of eight different containment scenarious were carried out. They considered the amounts of radioactive material released from the reactor core into the containment system and the portions thereof held back in the containment system due to plate-out processes, and determined amount and time of radioactivity release into the environment.

As the results of the plant systems analysis, five release categories have been defined (fig. 4). They are distinguished by different release portions of the radionuclide inventory, by the release time after accident initiation, by the duration of the release, and by the thermal energy carried with the release. The expected frequencies of occurence of the release categories are obtained from the expected frequencies of core destruction and from conditional probabilities of tank failure modes and containment isolation failure.

Category 1 comprises the most severe release after plug system failure and overpressurization failure of the outer containment.

The main release occurs a few minutes after accident initiation and is combined with considerable release of thermal energy. On table 3, only the release fractions of noble gases and actinides are shown. For other radionuclides release fractions between five and fifteen percent of the inventory have been calculated for this category.

Category 2 comprises cases with failure of heat removal from inner containment, while the outer containment is isolated up to 22 hours. At that time, hydrogen explosion occurs in the inner containment, destroying the integrity of the containment sytem. Releases (except noble gases) are significantly lower than in category 1.

For category 3 thermal tank failure, failure of containment isolation and unfiltered exventing of containment atmosphere is assumed.

Categories 4 and 5 contain cases with lower releases, category 5 being quite similar to the design basis accident.

3.4 External events

Besides accidents caused by internal initiating events, possible effects of external events have been investigated. Flooding, tornadoes, lightning, gas cloud explosions, effects of hazardous materials and missile generation in the turbogenerator building have been analysed qualitatively. The analysis showed that no significant risk contribution is to be expected from such events.

Effects of airplane crashes and earthquakes have been analysed quantitatively. Due to design measures, and low frequencies of these events contributions from air plane crashes are small. Earthquakes, which may threaten the availability of the reactor scram systems, of the decay heat removal systems and of containment isolation, contribute 50 percent to the expected frequency of the most severe release category 1 and about 40 percent to the expected frequency of category 3. Details of the seismic risk analysis are presented in [4,5].

The reason for this significant contribution is not a particular sensitivity of the SNR-300 to earthquakes, but the fact that the expected frequency of core destruction caused by internal events, is as low as the expected frequency of extremely strong earthquakes at the site Kalkar.

Since simultaneous core destruction and failure of containment structures have been assumed for a very strong earthquakes, these events contribute mainly to release categories 1 and 3.

3.5 Spent Fuel Pools

Besides accident sequences in the reactor core, releases of radioactivity from the spent fuel storage pool have been analysed. There is a sodium cooled and a nitrogen cooled storage pool. Consequences and expected frequencies of a total failure of their cooling systems have been determined. Based on a rather pessimistic approach, a frequency of 4×10^{-5} per year has been determined for radionuclide release from the sodium cooled storage pool. Release fractions are in the range of category 3 for the more volatile isotopes and in the range of category 4 for the actinides. However, the inventory in the pool is by a factor of about 6 smaller than in the core. Releases from the gas-cooled pool are insignificant for the risk.

3.6 Uncertainty of calculated frequencies

In order to quantify the degree of uncertainty of the calculated frequencies, subjective probability distributions of fixed, but inaccurately known quantities have been propagated through the calculations. Fig. 5 shows subjective confidence intervals for expected frequencies of occurence of release categories 1 to 5 as far as they result from internal events. No great changes of the bandwidths would be expected from external events. The bandwidth approximately corresponds to a 90 percent snbjective confidence interval.

3.7 Accident Consequences and Comparison SNR-300/PWR

Using the release categories as input, accident consequences were calculated for the site Kalkar. In order to make the results comparable to the German Risk Study for pressurized water reactors [2], the consequence model used therein was employed, after some modifications.

Early fatalities due to acute lethal radiation doses occur only above a threshold dose. This threshold is not reached after accidents of the SNR-300 within populated areas around the reactor site. Tbis even remains true if no credit is taken for countermeasures.

Late fatalities have been determined on the basis of a linear dosis-effect relation without threshold. The result of such a model is that even very low doses are assumed to lead to late fatalities, if only a big enough population is exposed to them. Fig. 1 shows the complementary probability distribution for late fatalities. For comparison, the analogous results of the German Risk Study for PWR [2] are also shown. These results have been calculated for 25 plants at 19 different sites. For purpose of comparison, frequencies have been reduced to one plant. Consequences reflect the whole spectrum of 19 sites.

Fig. 6 shows, that the maximum number of late fatalities is about a factor of ten smaller for the SNR than for the PWR. Expected frequencies are also lower for the SNR. The uppermost curve for the SNR relates to release from the sodium-cooled storage pool, the other four curves relate to releases from the core.

Similiar relations between SNR-300 and PWR have been obtained for other kinds of consequences.

The number of late fatalities caused by release category 1 is increased by a factor of about 3 if plutonium from light water reactors would be considered instead of plutonium from Magnox reactors, as has been done in the study. This is due to the fact that light water reactor plutonium contains higher portions of the radialogically important isotope Pu 238.

Uncertainty analysis was not carried out for the consequence modelling. However, significant differences with regard to uncertainty analysis are not to be expected between the SNR-300 and the PWR of the "German Risk Study".

The bandwidth of the confidence intervals obtained for the frequencies of occurence of release categories for the SNR-300 is about 10 times as large as for the PWR. This confirms the expectation that a risk analysis for a prototype plant necessarily will end up with wider margins of uncertainty than a risk analysis for an existing plant with operating experience.

The main task of the analysis was a comparative safety evaluation of the SNR-300, and of a modern PWR. Though the uncertainty bandwidths obtained in the SNR-300 analysis are much wider than in the PWR analysis, it is indicated that both frequencies and consequences of severe accidents are smaller for the SNR-300 than calculated in [2] for the PWR.

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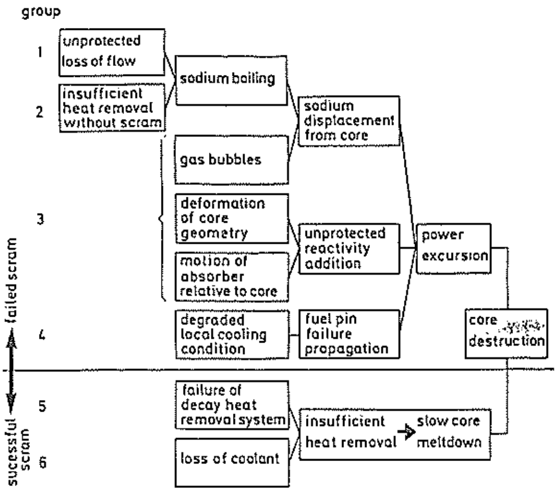

Fig. 1: Core destruction initiators

group	accident initiator frequency	failed system function conditional probability		core destruction initiator
1	transient 12 /a	reactor scram system (mechanical) 10^{-7}		$1.2\cdot10^{-6}$/a
2	transient 12/a	manual shut down 10^{-1}	reactor shut down signal 10^{-7}	$1.2\cdot10^{-7}$/a
5	loss of power 0,07/a	decay heat removal system active 10^{-4}	passive 10^{-2}	$7\cdot10^{-8}$/a
	steam generator failure 1/a	decay heat removal system active $1.5\cdot10^{-2}\cdot5\cdot10^{-4}$	passive 10^{-2}	$8\cdot10^{-7}$/a
	general case of decay heat removal 11 /a	decay heat removal system active $1,7\cdot10^{-3}\cdot5\cdot10^{-4}$	passive 10^{-2}	10^{-7}/a
sum	12 /a			$3\cdot10^{-7}$/a

Fig. 2: Expected frequencies of core destruction initiators

core destruction initiator	conditional probability of tank failure		
	plug system failure	mechanical failure	thermal failure (melt through)
unprotected loss of flow	$3\cdot10^{-3}$	$3\cdot10^{-4}$	$7\cdot10^{-2}$
insufficient heat removal with failure to scram	$5\cdot10^{-3}$	$1\cdot10^{-2}$	0.99
failure of decay heat removal with reactor scram	—	0.5	0.5
sum	$2.6\cdot10^{-3}$	$9.5\cdot10^{-2}$	$2.2\cdot10^{-1}$

Fig. 3: Conditional probabilities of tank failure

release category nr.	description	time of main release h	released thermal energy 10^6kJ/h	expected frequency of release per year	released portion of reactivity inventory	
					noble gases	actinides
1	core destruction, plug system failure, overpressurization failure of outer containment	0 - 1	530	10^{-8}	1	0.05
2	core destruction, mechanical tank failure, damaged core catcher, loss of power	22 - 33	15	$2\cdot10^{-7}$	1	$5.5\cdot10^{-4}$
3	core destruction, thermal tank failure, unfiltered exventing	0 - 48	-	$2\cdot10^{-8}$	1	$4.1\cdot10^{-4}$
4	core destruction, thermal tank failure, loss of power, containment isolated	48 - 100	-	$2\cdot10^{-7}$	1	$1,8\cdot10^{-5}$
5	core destruction, thermal tank failure, containment systems functioning	240 - 320	-	$3\cdot10^{-7}$	$2\cdot10^{-2}$	$1.4\cdot10^{-11}$

Fig. 4: Release categories

release category	frequency / a		
	lower 5 % value	best estimate	upper 95 % value
1	$2\cdot10^{-10}$	$6\cdot10^{-9}$	$5\cdot10^{-7}$
2	$2\cdot10^{-9}$	$1\cdot10^{-7}$	$5\cdot10^{-6}$
3	$2\cdot10^{-10}$	$1\cdot10^{-8}$	$1\cdot10^{-6}$
4	0.0	$1\cdot10^{-7}$	$3\cdot10^{-6}$
5	$7\cdot10^{-9}$	$2\cdot10^{-7}$	$8\cdot10^{-6}$

Internal initiating events only

Fig. 5: Confidence intervals for expected frequencies of release categories

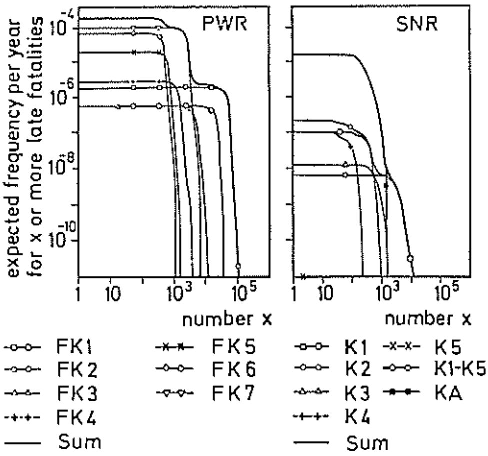

Fig. 6: Complementary frequency distribution for late fatalities (PWR and SNR-300)