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FINAL REPORT

INTERACTIONS OF SEVERE ACCIDENT RESEARCH AND THE REGULATORY POSITIONS

(ISARRP)

CO-ORDINATOR

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EXECUTIVE SUMMARY

This project, started on April 1, 1999, had the specific objectives of determining whether; (i) the focus of the severe accidents (SA) research is consistent with that of the regulatory authorities, (ii) the results obtained so far by SA research satisfy the regulatory concerns, (iii) the future programs, envisaged will address the potential regulatory needs into next century, and (iv) how much weight in the future SA research should be placed on preventive versus mitigative accident management measures.

The project work consisted of Workshops to which the partners contributed. The partners represented their respective regulatory organizations or their technical support organizations. A Questionnaire based on the objectives, listed above, was prepared and sent to several European regulatory authorities. The Questionnaire was also sent to United States Nuclear Regulatory Commission, Japanese Nuclear Safety Commission and the regulatory authorities of Hungary, Slovakia, Slovenia and Czech Republic. Responses have been received from nine European organizations, four Eastern European organizations, Japanese Safety Commission and the United States Nuclear Regulatory Commission. The responses showed differences between the attitudes of the various regulatory organizations towards SA research accomplishment and needs. Clearly, the responses obtained have statistical value since a wide spectrum of regulatory organizations have contributed, although no statistical analysis was performed. Insights obtained from their responses have been combined and are reported here.

In addition to the analysis of the responses to the Questionnaire, a critical review of the severe accident phenomenological research conducted in the World for the past 20 years was performed. The accomplishments made by this research activity were examined and related to the needs of the regulatory organisations as evidenced by the responses to the Questionnaire, referred to above. The research accomplishments were also related to the requirements of the severe accident management guidance and its implementation. The impact of the more recent approaches e.g. the probabilistic safety assessment (PSA) and the risk informed regulations was examined and their needs related to the accomplishments of the severe accident research performed so far. In this context the accomplishments of the SA research, sponsored by the European Commission in the fourth framework program, were reviewed. This has lead us to summarize the state of resolution of the SA issues with respect to the needs of the regulatory organisations. It was found that most regulatory organisations state that they have employed the results obtained, and the insights gained, from the SA research for regulatory decision making. They have also used the same for establishing greater confidence in either the regulations they have proposed or in reviewing the actions that the licensees (plants) have proposed for preventing or mitigating the consequences of severe accidents. The specific needs for further research, that the majority of the regulatory organisations [or their technical support organisations (TSO´s)] have indicated, have been classified in the report and recommendations have been made on the future directions of the SA research. The highest priority for future SA research has been assigned to the resolution of the issue of ex-vessel melt/debris coolability to achieve stabilization and termination of the postulated severe accident.

The recommendations provided in the report, although, quite general, may not apply to all countries and plants. Some of the SA issues and recommendations, particularly with respect to the application of the severe accident guidance (SAMG), may require plant specific modifications.

A. OBJECTIVES and SCOPE

A.1. Objectives

The European Commission has been sponsoring research into the phenomenology of postulated severe accidents in reactor plants, and, in severe accident management measures, for several years. The Third Framework Program, primarily, collected information on the national research programs in several specific areas and produced state-of-the-art reports. The Fourth Framework Program dedicated substantial funds towards specific cost-shared research projects on severe accidents, and on accident management. Some European countries, e.g., France and Germany are pursuing large severe accident research programs with their own funds. Substantially similar and/or complementary research on severe accidents has been pursued in USA, Japan, Canada and a few other countries.

Concurrent with these efforts world-wide, the regulatory positions (concerns) have also been evolving. As an example, the requirements on containment integrity, and the environmental release of radioactivity, have been strengthened. The USNRC has stated the position that containment integrity should be maintained for at least 24 hours, however, the emergency planning includes evacuation of the population from the vicinity of a postulated accident. In Europe, the French-German safety approach goes even a step further and requires that for new reactor designs, after a postulated severe accident, there shall be no need for permanent relocation and evacuation from the immediate vicinity of the plant, and long restrictions in food consumption. In Germany, this requirement has been turned into an extension of the existing law (Atomgesetz). Differences in the U.S. and European regulatory approach are also apparent in the backfit considerations for an existing nuclear plant: the USNRC´s backfit rule requires a cost/benefit analysis, while the European regulatory authorities do not prescribe such a rule.

Clearly, the nuclear regulatory positions and the findings from the research programs influence each other. The former determine the research directions and topics, while the latter influence the regulatory thinking, positions and concerns. It should be pointed out that the severe accident research programs are to a large extent a function of the positions of the regulatory authorities regarding the safety design of the nuclear power plants. Thus, there is mutual interaction between the regulatory positions and severe accident research programs, that varies from country to country in Europe.

The specific measurable objectives of the ISARRP Project are to determine:

(a) whether the focus of the present directions in the severe accident research is consistent with that of the regulatory authorities,
(b) whether the results (findings) obtained in the severe accident research programs have provided the information necessary to satisfy the regulatory positions (concerns) e.g., with respect to SAMG and,
(c) whether the future research programs, envisaged, address the potential regulatory needs.

A.2. Scope and Procedure

The ISARRP is a Concerted Action (CA) Project requiring a series of working meeting (workshops) between the partners to develop the information to satisfy the objectives of the Project. It turned out that the scope of the project envisioned at the time of the submission to the E.C., and at the start of the Project work was increased considerably. Many more regulatory organisations were contacted than originally contemplated. The work expanded to a review of the SA research that has been performed so far and that which is scheduled to be performed in the near future. Most of the previous and current important research programs pursued in the World were reviewed and their accomplishments considered in the evaluations. The research results obtained and the regulatory needs, as expressed by the responding regulatory organisations and the TSO´s were compared. Intercomparisons were made between the needs expressed by the various organisations to arrive at the findings that have been described in the report.

The procedure for the work in the Project was to develop a Questionnaire of twenty-five questions after due discussions between the partners. The Questionnaire was sent to the regulatory and TSO organisations in Western Europe. Later, the Questionnaire was also sent to the USNRC, Japanese Safety commission and the regulatory organisations in Eastern Europe. The responses obtained were discussed and analysed by the partners in the workshops.

Each of the partners was assigned the responsibility for developing a short report on one or two of the work packages in the Project. These individual reports were reviewed by all the partners and served as the base-documents for this final report.

A trip was made by the Coordinator and the Swiss partner to Washington D.C. to discuss with the USNRC their current positions on severe accidents, in general, and on accident management, in particular. Discussions were also held on their views on the benefits of the SA research that they had derived and on the remaining SA issues for further resolution. These discussions helped to clarify the responses they had provided to the Questionnaire.

The partner group for the ISARRP project is much smaller than those for the other E.C. funded Concerted Action projects, however, the main nuclear countries in Europe are represented. We believe, the small size has helped in generating frank and lively discussions and afforded deeper analyses of the SA issues, the research results, their usefulness in general and their utilization by the regulatory and technical support organisations. In addition, the small group could openly analyse the responses of the various regulatory organisations and comprehend the different national philosophies and attitudes.

B. WORK PROGRAMME

The work Programme of the ISARRP Project was divided into several work packages. The work was conducted in the form of presentations and discussions, held during several meetings whose character was that of workshops. Short reports were prepared by the partners assigned to each task.

Work Package 1: *Critical review of the SA phenomenological research*
The objective of this work package was to consider the progress made world-wide in research on the resolution of the outstanding phenomenological issues posed by severe accidents.

Work Package 2: *Relevance of severe accident research to SAMG requirements and implementation*
The objective of this work package was to relate the progress made in the resolution of the SA issues to the practical matter of what results are required or have been used for the management of severe accidents. Clearly, the SAMG is the most important avenue employed by the regulatory organizations to assure themselves of the safe (from public perspective) performance of a nuclear plant in a postulated severe accident event.

Work Package 3: *Relevance of severe accident research to PSA and the risk informed regulatory approach*
The objectives of this work package is to relate the results obtained by the severe accident research to the requirements of a PSA and of the new trend of employing the risk informed approach in promulgating regulations. Clearly a PSA identifies vulnerabilities in the knowledge base, however, their importance is decidedly plant specific. Nevertheless the uncertainties in the phenomenology or in resolution of issues lead to uncertainties in the PSA conclusions and in the adoption of the risk informed approach.

Work Package 4: *Questionnaire and the evaluation of responses to the questions*
The purpose of this work package is to solicit the views of the regulatory organizations towards the results of the SA research and the benefits they have derived from it in terms of regulatory actions, or in the confidence they have gained in assessment of plant safety. This work package was also designed to distinguish the differences between the attitudes and approaches followed by the various regulatory organisations in Europe, Eastern Europe, U.S.A. and Japan.

Work Package 5: *Relevance of example PSA results to SA research*
The objective of their work package was to employ the results of some recent PSAs (preferably for a PWR and a BWR) and relate their findings to the results obtained in SA research, and to the effectiveness of the SAM measures already taken or contemplated.

Work Package 6: *The state of resolution of the SA issues with respect to the needs*
The objective of this work package is to have another look at the state of the resolution of the severe accident issues which have been identified over the years, and relate that to what the needs of the regulatory organizations are in terms of their functions.

Work Package 7: *Regulatory use of the results of severe accident research*
The objective is to identify the results of the SA research which the regulatory organizations, over the years, have used in either defining specific regulatory actions or in not taking specific actions.

Work Package 8: *Remaining issues and concerns*
The objective of the work here is to review the work in the previous work package and identify what are the remaining unresolved safety issues and concerns for which sufficient results of the SA research are not available.

Work Package 9: *Recommendations on future directions of severe accident research*
The purpose of this work package is to provide recommendations to E.U. (and to the readers) by the authors of this report on the directions that should be followed, in the future for the conduct of severe accident research. These recommendations are in essence the conclusions of this study.

C. WORK PERFORMED and RESULTS

The work performed and the results obtained are described below in the various subsections under Section C. We have not adhered one to one to the various tasks identified in the Work Programme, however, we have performed all the work described in those tasks.

C.1. Critical Review of the Severe Accident Phenomenological Research

C.1. 1. Introduction and Background

The light water reactor (LWR) systems engineered and constructed in the Western countries followed a definite design philosophy for ensuring a very low level of risk to the public. Briefly, the plant systems are designed with the defense in depth concept. The systems are designed to withstand a single failure and prevent a severe accident in which core damage could occur. The goals for core damage frequency range from 10^{-4} to 10^{-6}/reactor year. The plant systems are also designed to withstand the loadings due to the design-basis accidents and incidents, and specified external events, e.g., earthquakes, fires, tornadoes, floods etc. In addition, with characteristic foresight, the designers provided a strong containment system to contain any fission product radioactivity produced even in the beyond-the-design-basis accidents. The containment structures withstand pressures much beyond those imposed by the energy release during the design basis accidents. Mitigation measures are provided in the containment buildings e.g., the suppression pool in the boiling water reactors (BWRs) and the sprays, fan coolers and ice condensers in pressurized water reactors (PWRs) for long term heat removal from the containment buildings. The objectives of these containment safety systems is to keep the pressure low and protect the integrity of the containment in the design and the beyond-the-design-basis accidents.

In terms of public safety, it is perhaps self-evident that if containment integrity is not violated public safety is not compromised. The severe accident, even if it progresses to the core melt on the floor, will not be a life-threatening event from the point of view of public safety, if the containment remains intact and leak-tight. Adequate performance of the containment in the aftermath of a postulated severe accident, thus, is of vital concern. In particular, it has been determined that maintaining the integrity of the containment for the first few hours, after any fission product release in the severe accident, can reduce the containment airborne radioactivity by orders of magnitude. This is a direct consequence of the time constant for aerosol deposition on the containment walls and floors. Early containment failure, thus, has to be prevented by design or by accident management. Late failure of the containment has also been questioned recently. Perhaps, the public anathema to evacuation and to even a minor land and water contamination is forcing a re-examination of the regulatory attitudes and safety philosophy. Consideration of the requirement of 24 hours as the time for containment leak tightness for the new plants in USA and the enactment in Germany of the extension of the existing law (Atomgesetz) that there shall not be permanent relocation, and evacuation, from the immediate vicinity of a nuclear plant, are indicative of these new attitudes and philosophy. These containment performance goals, laudable as they are, for the new plants, will be difficult to achieve if the old evaluation philosophy of using conservatism at each step is employed. Thus, it is imperative, that the new containment performance goals are accompanied by rational evaluation methodologies.

A severe accident by definition involves severe damage to, and melting of the core, and release of radioactivity. Clearly, the phenomena involved in a core-melt accident are extremely complicated, since the main characteristics of the accident scenario are the interactions of the core melt with structures, and water, and the release, transport and deposition of the fission product carrying vapors and aerosols. The interactions of core melt may lead to (i) ablation of structures (ii) steam explosions, (iii) concrete melting and gas generation and (iv) dispersion of heat-generating melt (debris). These phenomena involve the disciplines of thermal hydraulics, high temperature chemistry, high temperature material interactions, aerosol physics, among others. Predictions of the consequences of a severe accident have to be based on experimentation and models whose veracity may be limited by the scale at which the information about the phenomenology is derived. Scaling considerations become very important since large scale experiments with prototypic melts are very expensive and difficult to perform.

Another aspect about severe accident consequences should be mentioned. The LWR safety systems for the design base accidents have an acceptance criterion: the peak-clad temperature has to be maintained below 1200 ^{o}C, while employing conservative methods of analyses. No such criterion exists for severe accidents, which would focus the research adequately. Recently, the core damage frequency (CDF) $\leq 10^{-4}$ to 10^{-6} and the conditional probability of containment failure < 0.1, are becoming criteria for severe accidents. This, however, is a probabilistic criterion and is subject to some interpretation. The CDF criterion also is not used as a design basis, but as a design goal. In the same vein, the research accomplishments are harder to evaluate, since there is no specific measure.

As mentioned above, it became clear quite early, and confirmed by the WASH-1400 {WASH-1400, 1975} and NUREG-1150 {NUREG-1150, 1987} studies, that the containment had a central role in protecting the public against the consequences of a severe accident. Thus, the focus of the severe accident research, became the evaluation of the survivability of the containment for the various severe accident scenarios. More recently, the focus has shifted a little, due to the accident mitigation perspective, from the survivability of the containment to that of the survivability of the vessel. Vessel external flooding has been adopted in the AP-600 design {Theofanous 1995}, and has been back-fitted in the containment of the Loviisa Power plant in Finland {Kymalainen-97}.

In this review, we will confine ourselves to describe the progress of the severe accident research, in relation to the public safety issues posed by the hypothetical severe accident scenarios. Several issues were identified previously and the research work was focused towards resolution of those. New issues have been identified due to the changing attitudes about public safety, and by the designs of new reactors. We will attempt to briefly describe the status of the research work focused on the resolution of the issues. We will not be able to provide references to the many many fine investigations performed. We apologize for this.

C.1.2. In-Vessel Accident Progression

It is perhaps instructive to delineate the time scales involved in the various phases of the in-vessel accident progression. The core boil-off and the initial heat-up process are relatively lengthy (1-3 hours), before significant core damage takes place. Accident termination during this time is relatively straightforward, if operator is able to add water to the reactor vessel.

Clad melting, fuel melting, core blockage and core melt pool formation are relatively shorter duration processes (1/2 to 1 hour), during which access of water to some of the blockages and debris beds formed may become limited. The interaction of the core melt with the lower head water and structure, and the failure of lower head may be relatively longer duration (3 hours) processes if the melt quenches and reheats. Alternatively, if melt cooling/quenching does not occur, the lower head may fail relatively fast (minutes). The character of the melt discharged to containment is different in the two scenarios.

C.1.2.1. Early Phase of In-Vessel Accident Progression

A severe accident in a PWR starts with core uncovery initiated by loss of reactor coolant inventory and failure of some of the reactor safety systems. The in-vessel progression of the accident, from that point on, is determined by thermal-hydraulics and material interactions. If accident management actions are not successful, the rise in core temperatures due to undercooling leads to exothermic Zircaloy oxidation transient which delivers heat to clad and fuel at a very large rate (upto 10 times the decay energy rate), a large amount of hydrogen is produced and released to the containment. Core temperatures rise at the rate of 1 to 10K/sec;

melting starts with the structural and control rod materials and progresses in turn to clad, fuel eutectic, and fuel. Substantial loss of geometry takes place, and a melt pool may be formed within the original core boundary as happened in the TMI-2 reactor. Eventually, the molten core material may be discharged, as a jet, to the lower plenum as occurred in TMI-2. Alternatively, the core slumps and eventually attacks, thermally and mechanically, the core support structure. Failure of the support plate or core barrel brings the corium (molten fuel-structure mixture) to the lower head. This ends the early phase of the in-vessel accident progression.

During the early phase of in-vessel accident progression the parameters of interest to the containment integrity are:

- the magnitude and rate of hydrogen generation,
- the elapsed time before the onset of core melting, and
- the temperature levels of the reactor coolant system (RCS),

Information about hydrogen generated (and released to containment) is required for its management and for establishing that strong deflagrations, transitions to detonation or detonations will not occur. Information about the elapsed time before onset of core melting provides the time window, available to the operator, for terminating the accident without core damage or fission product release. During core-heat-up, a considerable fraction of energy generated may be transferred to the RCS by natural circulation of the steam generated, which may become hot enough to induce local failures. This could change the risk-dominant high pressure accident scenario, thus, accurate prediction of RCS temperature levels is essential in determining the consequences of some of accident scenarios.

Much research has been performed for the early phase of the in-vessel melt progression. A representative experimental research program is CORA {Hagen-97} in which several bundles representing PWR and BWR fuel arrangements were heated electrically and observations on fuel degradation were obtained. Previously, experiments were performed with the PBF {McDonald-83} and LOFT {Carboneau-89} reactor facilities, and, currently, PHEBUS {Livolant-96} experimental program is directed towards in-vessel melt progression, and fission product release, transport and revolatalization.

Clearly, the above research programs have produced results which have reduced uncertainty. The state of knowledge with respect to the PWR in-vessel core melt progression confirms the picture conveyed by TMI-2. It is believed that a melt pool will form in the original core volume and will drain along the side of the core into the lower plenum to commence the loading on the lower head.

The state of knowledge regarding BWR in-vessel melt progression, in particular, for the higher probability depressurized dry core scenario, is relatively confused. Core wide blockage formation could occur similar to that for a PWR; however, there is not enough data, or analysis to delineate the conditions, under which it could occur or not occur. It is conceivable that the BWR in-core melt progression may terminate with failure of the core support plate. There are also possibilities of earlier relocation of control rod and other core material to the lower head for the dry core scenario.

The effects of accident management actions, e.g. water addition to a hot core, have been considered recently. It was found in the CORA tests {Hagen-97} that this increases the core damage and the hydrogen generation, due to the increase in Zircaloy oxidation by the steam produced. A new facility QUENCH {Sepold-99}was constructed with European funding to further investigate the increase in hydrogen generation as a function of the clad surface conditions. It was found that if a reasonably thick (~ 300 μm) oxide layer is present on the clad surface, the release of additional hydrogen during the quench process is not large. The converse is true if there is no oxide layer present on the clad surface. It is expected that the clad surface which has undergone some oxidation during normal plant operation and prior to the accident management action of bringing water to the hot core, will be covered by a relatively thick (~300 μm) oxide layer. The oxide layer in the QUENCH experiment suffered some cracks, which allowed some hydrogen generation. The fresh clad tested produced much hydrogen and damage to the fuel bundle resulted due to the exothermic energy generated. In all cases fuel bundle quenched eventually.

An in-vessel issue related to the BWR accident management is that of addition of unborated cold water to the partially damaged core in which the control rods may have melted and the boron-carbide accumulated on the core support plate. Investigations on the reactivity effects of this scenario have been pursued in an EU Project {Frid-99}. There are many uncertainties in this evaluation; nevertheless the Doppler and the void feed back mitigate the core damage. Adding boron separately, as it is prescribed for the anticipated transient without scram (ATWS) event may be beneficial.

C.1.2.2. Late Phase of In-Vessel Accident Progression

Accurate description of the late phase of the in-vessel severe accident scenarios has assumed greater importance lately, since it has become evident that the assumptions made in its modelling determine the composition, amount and the rate of corium discharged to the containment, to which the containment loadings are directly related. In particular, if the projected loadings are severe enough to fail a containment soon after the vessel failure, e.g., due to direct containment heating or hydrogen detonation, the "source term" consequences of a severe accident can be very severe indeed.

The late phase of in-vessel accident progression did not receive as much attention before, except for some specific evaluations e.g. that of the AP-600 in-vessel melt retention {Theofanous-95}. Recently more generic investigations have been pursued in, a recently concluded, EU Project in which the following questions were addressed {Sehgal-97a} {Sehgal-99b}:

1. Can the lower head fail immediately, in spite of the presence of water, due to the attack of a melt jet released from the core?
2. Can the melt debris be cooled by the water in the lower head to preclude vessel failure?
3. If the water can not be supplied can the melt be retained within the lower head by cooling the external surface with water?

4. In the absence of water, inside and outside of the lower head, how long will it take to fail the lower head by melting and creep processes?
5. What is the mode and location of lower head failure and is it affected by the presence of the penetrations in the lower head? and finally
6. What is the rate of enlargement of a local lower-head-failure-site caused by the flow of melt through it?

The melt jet discharged from the core during its interactions with the lower head water would fragment and could generate a steam explosion. The questions relevant to that process are:

- What is the fraction of the melt jet that fragments in water?
- Can the steam explosion cause the failure of the lower head?

It is recognized that there is a relatively broad consensus that an in-vessel steam explosion will not cause containment failure, however, there is no consensus that a steam explosion can not cause lower head failure, particularly at the location of a penetration.

The investigations performed for establishing the feasibility of the in-vessel melt retention for the Loviisa plant {Kymalainen-97} and for the AP-600 design {Theofanous-95} and those performed in the EU projects, Melt Vessel Interactions (MVI) and Molten Fuel Coolant Interactions (MFCI) have provided quite well-validated responses to some of the issues raised above. These are:

1. It appears {Sehgal-97a} that the immediate failure of the lower head due to the impingement of a melt jet dropped from the core is physically unreasonable. Only in the case of a long-running thin melt jet attacking the lower head wall without water, there could be an ablative failure. This, however, is a physically unreasonable occurrence.
2. The FARO experiments {Magallon, 1997} have shown that between 40 and 60% of the melt jet would fragment, and the remainder could form a cake of very low porosity at the bottom of the debris bed. The long-term coolability of such a bed has not been established
3. Much work performed recently {Sehgal, -98a} and ongoing in the RASPLAV Project {Asmolov-97} has clarified the limitations on the power level of a reactor which would be amenable to melt retention in lower head by the cooling of the vessel from outside. It appears that the plants with electrical power generation level beyond 1000 MWe may not have sufficient margin. Recent results from RASPLAV have added the uncertainty of melt pool stratification, whose effect on the margins has not been clarified so far.
4. Many experiments performed in the KROTOS facility {Huhtiniemi-99} with jets, and one very recently in the FARO facility, have failed to produce strongly-propagating steam explosions. On the contrary, spontaneous explosions have been observed when Al_2O_3 melt jets are employed. It appears from the experiments that the explosivity and efficiency of a steam explosion with UO_2-ZrO_2 melt interacting with saturated or subcooled water is much lower than that of Al_2O_3, which was, previously, considered as a good simulant for the UO_2-ZrO_2 corium mixture.
5. The ablation of the vessel failure site was measured and scaling analysis developed {Sehgal-97b}. It was found that a crust layer persists, reducing the heat transfer from the melt stream to the vessel wall. The most probable hole size, after ablation by the

melt in a prototypic scenario, may be in the range of 15 to 20 cms. These are much lower estimates than those derived earlier.

6. Considerable experimentation {Sehgal 1998a} {Sehgal, 1998b} and analyses {Sehgal 1999b} have indicated that global vessel failure is highly unlikely for both PWRs and BWRs. The most probable mode of failure for the vessel is the creep of the lower head and the likely location of failure would be around a penetration. For the scenarios in which melt pool convection is established in the lower head, the likely location of failure is near the upper elevations of the hemispherical head, where the temperatures are the highest.

The results described above have been obtained in the last 5-7 years and the technology developed provides a relatively good basis for the description of the processes occurring in the late phase of the in-vessel melt progression. More work is needed, in particular, to

1. understand the reasons for the low explosivity of UO_2-ZrO_2 melt. This is also necessary for the evaluation of the consequences of ex-vessel steam explosions,
2. explore the coolability, in vessel, by either gap cooling (for melt pool) or water ingression (for a debris bed),
3. determine the fragility of lower head against dynamic loads,
4. obtain confirmatory results on the timing, mode and location of the lower head failure for the commonly-used pressure vessel steels. It has been observed that the creep deformation laws for the various pressure vessels steels are quite different from each other and
5. determine if there are adverse chemical reactions between the melt/debris (crust) and the vessel wall which may cause vessel failure.

C.1.2.3. Fission product release and transport during in-vessel accident progression

The "source term", i.e., the magnitude, the chemical and the physical form of the fission product source distribution in the containment atmosphere received great attention right after the TMI-2 accident and currently the PHEBUS Project is providing confirmatory data on this subject. During the in-vessel accident progression phase, the parameters of interest are:

- the fraction of the core fission product inventory released
- the fission product chemical species
- the fraction of released fission products deposited on the reactor coolant system (RCS) surfaces
- the revaporization of the fission products from the RCS surfaces

The research work pursued made great progress and provided good estimates for the parameters above. It was found that, in general, 70 to 80% of the volatile fission products inventory is vaporized from the core, except for tellurium, some fraction of which is retained by the unoxidized Zirconium in the core and is released as Zr oxidizes. The fission product vapors change into aerosols as they cool down in the cooler parts of the RCS and aerosol physics determines the fission product deposition on the RCS surfaces. A substantial fraction of the fission products released from the core will deposit in the primary system before exit from the break location to the containment. The deposited fission products, thus, are not immediately available as the source term; however, as the temperatures in the RCS increase due to the continued decay heat generation by fission products, the revaporization of the deposited volatile fission products occurs and, in time much of the deposited volatile fission products will leave the RCS and enter the containment. Early on, the importance of the re-vaporization process was not fully realized, however it has become quite clear that re-vaporization plays a significant role in determining the fission product "source term" for the cases of late containment failure, and for some containment bypass sequences.

The total release of relatively low volatile fission products, e.g., oxides and hydroxides of Ba, Sr, Ru, Ce etc., during the early phase of in-vessel accident progression, is of the order of a few percent of the inventory at most. The Molybdenum is an exception since its release is significant. However, the release estimate is based on very uncertain knowledge about the chemistry of Molybdenum.

During the late phase of the in-vessel accident progression, the vessel lower head may be full of a convecting high temperature melt pool, which may contribute a release of the non-volatile fission products. The in-vessel melt retention accident management scheme results in the high temperature melt pool residing in the lower head for hours or days. There are very little data on the release of the less-volatile fission products from a high temperature melt pool. The melt pool upper surface will have a crust. The efficiency of the crust in stopping the fission products is not known. Such information will be needed for estimation of the source term if the in-vessel accident management scheme is adopted, for new or existing plants.

The chemical character of the fission products released is an important element in the estimation of the source term. The research work conducted after the TMI-2 accident identified the compounds formed by the various fission products during their release in the core and also during their transport in the RCS. The dominant species for Iodine and Cs releases were found to be CsI and CsOH, which are extremely soluble in the water present in the containment and the sump. The recent PHEBUS tests {Ktorza-99} have found that a few percent of the total Iodine release may be in the form of Iodine gas, and that silver Iodide may be formed. The small amount of the gaseous iodine, released from the core, was found to diminish rapidly during its stay in the containment. Nevertheless, the PHEBUS data indicates that interaction of the iodine with the various materials in the core to form different compounds needs greater resolution.

C.1.3. Ex-vessel accident progression

The ex-vessel accident progression is basically the interaction of the products of the in-vessel accident progression, namely steam, fission products, hydrogen and corium melt with the contents of the containment.The pressure (and temperature) loadings exercised during these interactions on the containment structure may cause failure of the containment, which as we discussed in Section 1 should be prevented. Thus, the study of the ex-vessel accident progression is primarily that of the containment loadings, and of the evaluation of the probability of its failure. In this respect two time zones can be defined namely "early" and "late" for the failure of the containment. This distinction results from the observations on the radioactive aerosol source in the containment, which diminishes, exponentially with time, due to its deposition on the containment floor and surfaces, and its dissolution in water. It has been observed that with steam in the containment atmosphere 99.9% of the aerosols in the containment atmosphere are removed in 4-6 hours. [Schöck-84] Thus, the time span of interest for the early failure of containment is 4-6 hours and for the late failure of containment more than 4-6 hours. It should be obvious that the greater public hazard is posed by the early failure of the containment.

C.1.3.1. Early failure of containment

After a prolonged review of the severe accident scenarios, initially by the Containment Loads working Group, formed by the USNRC and later by the expert panel working with the Sandia laboratories on the NUREG-1150 {NUREG-1150, 1987}, the following major challenges, which may lead to an early failure of LWR containments, were identified.

- direct containment heating as a result of melt discharge at high pressure from a vessel breach in a PWR.
- melt attack on the liner of the BWR Mark I containment,
- hydrogen detonation, and in-vessel and ex-vessel steam explosion.

Each of these challenges, in turn, became a severe accident issue and led to several years of concentrated research. Some of these issues are resolved, or close to resolution, while others still are far from resolution. By resolution, we mean a technical consensus is reached on either the adequacy of the existing containment systems to meet the challenge posed with a very high degree of confidence, or, a technical consensus is reached on the necessary measures (accident management and/or back fit), which would impart that character to the existing containment systems.

C.1.3.2. Late failure of containment

The time span of interest is beyond 4 hours after the initial release of radioactivity in the containment. In this time span, if the melt is discharged into the containment, it is essential that a heat transport system is established within the containment, i.e., the containment heat removal systems, e.g., fan coolers in PWRs and suppression pool coolers in BWRs are functioning. Otherwise, the slow pressurization resulting from either the prolonged heat addition to the containment atmosphere, or the generation of steam from melt (debris bed) cooling, or the non-condensable gases generated from the molten corium concrete interaction (MCCI) can reach pressure levels at which the containment may fail or leak excessively. This may occur after several hours (more than 4), or a few days, depending upon the water availability, the type of concrete and the pressure-bearing capacity of the containment.

Another potential radioactivity pathway to the environment can result from the containment basemat penetration when the melt can not be cooled and it keeps attacking the basemat. This may occur after a day, or after many days, depending upon the heat removal from the melt debris, the type of concrete, and the thickness of the basemat.

The outstanding safety issues, identified for this time span are:

- melt spreading
- melt (debris) coolability,
- concrete ablation rate,
- non-condensable gas generation rate,
- stabilization and termination of accident and
- performance of venting (filter) systems.

C.1.4. Direct containment heating

The direct containment heating (DCH) issue was around for a long time. Substantial experimental and analytical research, sponsored by the USNRC was performed in the '80s and early'90s. Accompanied by a stringent peer-review-process this resulted in a focussed effort whose results led to the resolution of this issue; for the Westinghouse pressurized water reactor, and more recently for some of the other PWR plants. This resolution is, however, plant specific and DCH loads model {Pilch-93} could be used for evaluation of this issue for individual plants.

Another finding {Denny-83} which has a direct bearing on the DCH issue is the high probability of unintentional depressurization occurring during the high pressure severe accident scenario. The reason is the establishment of natural circulation flow loops in the vessel, hot legs and the steam generators, which can transfer the energy from the core, during the heat-up phase, to the piping system. An elaborate program of 1/7 scale experiments performed at the Westinghouse laboratories, corresponding scaling analysis and the computer code simulations all point to the high expectation of the creep rupture of the surge line to the pressurizer before the vessel rupture. The depressurization induced will also bring water from the accumulators to the dry and hot core and change the high pressure scenario completely.

The DCH issue has been muted with the SAMG requiring depressurization in PWR plants by the operator and automatic depressurization systems available in BWRs. Reduction of vessel pressure to the level of ≤2Mpa reduces the potential of DCH very significantly.

C.1.5. Melt attack on BWR Mark-1 containment liner

This safety issue was raised due to the short distance between the vessel and the containment liner in the Mark-1 BWR dry well. The contention was that the corium melt will be able to traverse that distance and melt the steel liner to fail the containment., soon after vessel failure. This issue stood as one of the major sources of risk for the Mark-1 BWR. The expert opinion obtained during the NUREG-1150 probabilistic safety analysis (PSA) work split on the assignment of the probability of the liner melt-through. The probability values, with water present in the dry well, ranged from 0.001 to 1.0. The authors of NUREG-1150 averaged these results to obtain a point estimate of 0.33, which certainly was a very arbitrary estimate of the probability of a sequence which has major source-term consequences for the Mark-I BWRs.

The ROAAM methodology {Theofanous-93} was employed to decompose the scenario into the individual components of melt release, melt spreading, melt concrete interaction and attack on the liner. The formalism employed three causal relations and five probability distribution

functions to arrive at the probability of liner failure. The analysis was quite comprehensive and the causal relations employed phenomena models validated against experiments; with conservatisms added wherever model uncertainties dictated that. The conclusions derived were that the probability of liner failure, without water present in the dry well, is close to 1.0, while, with the water present in the dry well, the liner failure probability decreased to the range of 0.0001. After peer review, the
latter was subsequence changed too 0.001, which can be labeled as physically unreasonable. Thus, we believe this issue has been adequately resolved.

C.1.6. Hydrogen combustion

The hydrogen combustion loads on the containment were the first to be addressed by the USNRC, since the hydrogen combustion event in TMI-2 triggered a heightened awareness of these loads. The hydrogen rule requires management of hydrogen concentration in the containment resulting from the oxidation of up to 75% of the heated Zirconium clad. This has already been incorporated in the ice condenser, BWR Mark III and BWR Mark II and I plants. The BWR Mark I and II plant containment are inerted, while the ice condensers and BWR Mark III plant have been fitted with igniters. The large volume U.S. designed PWR containment were judged to be immune, since the hydrogen concentration did not reach high enough to produce combustion-induced pressure loads, which would threaten containment integrity. The hydrogen combustion loads issue for these plants relates to either high local concentration, or the transition to detonation, which can occur for special geometries (ducts, accelerating flow regions etc.) at relatively low (≅10%), compared to stoichometric hydrogen concentrations. Most European countries consider 100% of Zr clad content in the core for estimating the hydrogen generation during a severe accident.

Hydrogen mixing research has been performed at several laboratories and several large experiments have been performed {Takumi-93}{Wolf-93}. The overall conclusion derived from these experiments and from analytic studies is that hydrogen mixing is quite efficient and local non-homogenities do not persist for long periods, except when they are coincident with thermal stratification effects. Recently many calculations, including some very large scale CFD calculations have been performed for several accident events in the complex geometry of an actual containment. These calculations do indicate some local concentrations of hydrogen greater than the average. Such complex analyses have been employed to determine the preferred locations for hydrogen catalytic recombiners; the hydrogen control option that is preferred by Europeans. There has been extensive proprietary research, and testing, on the hydrogen catalytic recombiners to determine their performance in different environments that a containment may be subjected to during the course of a severe accident.

The current focus of hydrogen combustion research is on the issue of transition to detonation and for what geometrical conditions and hydrogen concentrations this phenomenon can occur. Experiments were performed at BNL {Ciccarelli-93} and are currently being performed at the RUT facility near Moscow, Russia. The main difficulty is in scaling the experimental results obtained to the prototypic geometries in containment, which could be prone to such transitions. Very recent work {Dorofeev-99} has indicated that flame acceleration and fast combustion (leading to detonation) can occur under favorable conditions, at sufficiently large scale, for only strong mixtures. Such mixtures have a value of expansion ratio greater than a critical value, which is a function of the Zeldovich and Lewis numbers. Measurements performed so far have already provided some estimates of the critical values, inspite of the uncertainties. More measurements are scheduled to cover the influencing parameters for which the data are lacking.

C.1.7. In-vessel and ex-vessel steam explosion

The steam explosion loads on the containment were first considered in the WASH-1400 and, because of the assumptions made about the nature of this event at that time, the failure of containment (due to in-vessel steam explosion generated missiles) contributed a substantial fraction of the probability for early containment failure. The work on steam explosions {Theofanous-87}, since that time, led to more realistic estimates of the probability of containment failure due to in-vessel steam explosions. A steam explosion review group (SERG) established in 1995 {SERG2-95}, deliberated on the phenomenology of the steam explosion and provided expert estimates on the probability of the containment failure as a result of an in-vessel steam explosion. Although there were some differences of opinion, the vast majority of the experts concluded that the conditional probability (i.e., if there is a core melt) is less than 0.001, i.e., the containment failure is physically unreasonable. Recent tests in the BERDA program at FZK, also, have shown that for a scaled upper vessel head subjected to impact loads, simulating those from a very strong steam explosion, the head and the bolts survived.

Much experimental and analysis-development work is in progress, presently, on in-vessel steam explosions. Experiments have been performed with several kilogram quantities of simulant material heated particles and molten materials. Elaborate three-field analysis code: MC3D {Berthoud-97}, IVA {Kolev-99}, ESPROSE.m {Theofanous-96a} and PM-ALPHA {Theofanous-96b} have been developed. Some of the insights gained are (1) steam explosion probability is much reduced due to the extensive water-depletion that occurs around the fragmented particles of a jet in the premixture, (2) super-critical steam explosions, however, can not be excluded.

Ex-vessel steam explosion loads on PWR and BWR containments are also an issue, since a) in some PWRs, water discharged from the reactor primary system accumulates in the reactor cavity under the vessel and b) in some BWRs, a deep water pool is established under the vessel, prior to vessel failure: an accident management strategy employed in the Swedish BWRs. The ex-vessel water is generally highly subcooled and the extensive voiding, that develops in the premixture in a saturated pool, may not occur in the subcooled pool. Additionally, it has been found that the median particle size, obtained during the break-up process, may be much smaller for the subcooled water than for the saturated water. Contrary to these effects, which may argue, on heuristic grounds, for a larger probability of a steam explosion, there are the effects of cooling and solidification which argue for a reduction in the probability of a steam explosion. The corium melt may be a complex mixture of metals and oxides, however, predominantly it is a mixture of UO_2-ZrO_2-Zr, whose phase diagram, in general, shows a liquidus curve and a solidus curve, which are apart from each other by at most 200 to 300 K. For the UO_2-ZrO_2 mixture the difference between the liquidus and the solidus curve is only 50 to 75K. As the corium mixture solidifies its properties change radically. In particular, the viscosity, which is infinite in the limit of solidus, changes radically. The process of break up of a corium melt jet during its interaction with water results in many corium melt droplets of complex shape undergoing solidification from the exterior surface to the interior of the droplets. The changes occurring in the physical properties of the droplets affect the potential for the participation of the droplets in the steam explosion process. For example, it has been found that a thin high viscosity layer on the surface of a spherical droplet will greatly impede its subsequent fragmentation by a pressure wave, or shear forces.

The most remarkable experimental observations derived from the experimental program employing prototypic corium melt (UO_2-ZrO_2) in the FARO {Magallon-99} and (UO_2-ZrO_2) and Al_2O_3 in the KROTOS {Huhtiniemi-99} facilities at Ispra, Italy are:

- UO_2-ZrO_2 melt jets dropped in subcooled and saturated water at low pressure do not generate spontaneous steam explosions
- strongly-triggered UO_2-ZrO_2 melt jets in subcooled and saturated water at low pressure may develop a propagating event, however, of very low efficiency ($\leq$ 0.15%)
- Al_2O_3 melt jets (serving as a simulant for the corium fuel) generally experience spontaneous strong steam explosions when dropped in low pressure subcooled water
- Al_2O_3 melt jets dropped in saturated water at low pressure, in general, have to be triggered to experience strong steam explosions.

These significant observations point to the important role that the melt physical properties may be playing in the steam explosion process. Much research on this aspect is being pursued in Europe under the auspices of the European Commission. Some physical mechanisms have been identified. Nevertheless, it appears that the prototypic corium mixtures may not be as explosive (very low efficiency and /or explosivity) as previously assumed to be.

C.1.8. Melt spreading

In a dry or practically dry containment, the melt discharged from the vessel will spread on the concrete floor, which is the basemat for the LWRs. The spreading process determines the height of the melt pool that will have to be cooled subsequently. The importance of the spreading process is in its connection to the melt cooling process. A well spread melt will be of lower depth than an ill-spread melt and, thus, easier to cool. With this objective, efficient melt spreading has been employed as an accident management scheme in the proposed containment of the European pressurized water reactor (EPR). The EPR containment contains a special area where the melt discharged from the vessel, and held in a concrete crucible, is spread, after the failure of the holding crucible.

The corium spreading process is controlled by the hydrodynamic flow behavior which is a function of the melt pouring rate, the surface tension and the viscosity, and by the melt solidification process controlled by the heat loss from the melt to its surroundings. The heat is lost from the melt by radiation at its upper surface, and by convection, conduction and ablative process at its bottom surface. The heat of fusion and the increase in the melt viscosity as the freezing-crystallization processes start with the melt temperature dropping below the liquidus temperature are important parameters. The physics of all these process acting together is very complex and it is very difficult to predict the dynamics of the spreading process e.g. in terms of the position of leading edge (in 1-D) or the surface area (in 2-D) as a function of time. On the other hand, it is possible to predict the average thickness of the spread melt (and from there, the spreading length in 1-D and the spreading area in 2-D) {Sehgal-98c}. Such a scaling analysis was developed and validated against data obtained from spreading of various melt materials, ranging from cerrobend at low temperatures to corium at prototypic temperature {Dinh-98}. The scaling analysis was normalized to one parameter, which is that the melt loses 1/2 of its heat of fusion to stop spreading. This implies that the increase in melt viscosity is so large at the leading edge or at the surface of the spreading melt, when it loses 1/2 of its heat of fusion, that the melt can not move any more.

It was found both from experiments performed by different researchers and from analyses that the 2-D melt spreading is much more efficient than 1-D melt spreading for the reason that the melt has one more degree of freedom to move in the transverse direction {Sehgal-98c}.

The data base on melt spreading {Konovalikhin-99} has increased greatly in the last 3 years, obtained under the auspices of the European Commission in the CSC Project. The database has very large melt property variations, since many different melts were employed.

C.1.9. Molten corium concrete interactions (MCCI)

In a dry containment, the melt discharged from the vessel, after the short-time-spreading process, will attack the basemat concrete. The concrete ablation (melting accompanied by gas generation) occurs at much lower temperature than the melt temperature, resulting in substantial erosion of the basemat. The ablation process can continue, indefinitely, if a crust is formed on the melt upper surface, practically eliminating the heat loss from the melt upper surface. The rate of ablation in this limit would be governed by the melt heat generation rate and the ablation enthalpy of the concrete employed in the basemat. Thus, basemat melt-through can be envisioned. Concurrently, the gas generated during the concrete ablation process keeps pressurising the containment and late containment failure can be envisioned.

Molten corium concrete interactions (MCCI) research has been conducted over many years. A substantial body of experimental data have been accumulated from quite expensive programs e.g. SURC, BETA, ACE, where experiments were performed with heated corium and iron melts. Analysis development culminated in the codes CORCON {Cole, 1984} and WECHSL (Reimann, 1990), which have employed 2-D and 1-D analysis with primarily empirical heat transfer correlations. These codes have also represented the major chemical reactions taking place during the interactions.

The experience in validating these codes has been, basically, not as satisfying as one would like. The codes predict the measured ablation rate and total ablation within 30%. The same is true for the prediction of the combustible (H_2,CO) and non-combustible (CO_2,steam) gas generation rates. There are several uncertainties in the choice of parameters and there is the fear that some phenomenon is not being modelled or incorrectly modelled.

One phenomenon, which has been recently identified {Froment-99}, is that of melt segregation, which may have a greater contribution in the late phase of concrete ablation than in the early phase. This phenomenon may lead to higher concentration of Uranium oxide near the bottom of the melt pool resulting in non-uniform heat generation in the pool. Inclusion of the melt segregation modelling in the overall MCCI process has led to prediction of pool temperatures which were close to those measured in the ACE tests employing prototypic melt compositions. Complete influence of the melt segregation phenomenon on the consequences of the MCCI process has yet to be determined.

C.1.10. Basemat Melt-Through

The melt deposited in the containment, if uncooled, will continue to ablate the concrete basemat. The MCCI process ablates the basemat in radial and axial directions and can lead to sufficient axial ablation that the melt penetrates the soil below the concrete basemat. This condition called "basemat melt through", although not as severe as the release of aerosol, vapor and gaseous radioactive source term to the environment, in the event of containment failure, has to be avoided since it leads to ground contamination and, possibly, contamination of the groundwater.

It is important to predict, reasonably well, the long term progression of the MCCI process so that (a) any structural damage in the containment due to its radial ablation of concrete can be assessed and (b) the time to basemat melt-through can be estimated for the purposes of the management of the accident consequences through emergency evacuation and/or other measures to cool the melt, and terminate the accident.

The currently available MCCI codes, i.e. WECHSL and CORCON provide very different predictions for the MCCI progression process in the long term. The WECHSL code predicts much greater ablation of concrete in the axial direction than the radial direction from that predicted by the CORCON code. Unfortunately, except for the MACE scoping test, there is no MCCI experiment in which two dimensional ablation has been measured. Certainly, there are no tests where the long term MCCI process (low heat generation rate and insulated) has been modelled. There is a need to perform carefully-designed low decay heat, two dimensional ablation tests for a long duration to provide bench mark data for validation of the models in the CORCON and WECHSL codes. This need has been recognized and there is a proposal to perform such tests with the limestone-common sand and the silecous concretes in the MACE facility at ANL. Such an experimental program is, presently, being considered at OECD for initiation in 2001.

C.1.11. Melt debris stabilization and coolability

Melt coolability is perhaps the most vexing issue impacting severe accident containment performance in the long term. As mentioned earlier, melt coolability is essential to prevent both the basemat melt-through and the continued containment pressurization, thereby, to stabilize and to terminate the accident, without the fear of radioactivity release from the containment.

Provision of deep (or shallow) water pools under the vessel may not assure long term coolability/quenchability of the melt discharged from the vessel. Interaction of the melt jet may lead to very small particles (in the event of a steam explosion), which may be difficult to cool in the form of a debris bed of low porosity. Incomplete fragmentation will lead to a melt layer on the concrete basemat under a particulate debris layer and a water layer.

Coolability of a melt pool interacting with a concrete basemat by a water overlayer has been under intense investigation in the MACE Project \cite{Sehgal-92}, sponsored by an international consortium and managed by EPRI. The experimental work is being performed at ANL. Three experiments were performed successfully in which melt pools of 30 cm x 30 cm x 15 cm depth, 50 cm x 50 cm x 25 cm depth and 120 cm x 120 cm x 20 cm depth were generated on top of concrete base-mats and water added on top. The melt material contained Uranium oxide, Zirconium oxide, Zirconium and some concrete products. The decay heat generation in the melt was simulated through electrical heating. It was found that for these three tests, the effect of the sidewall dominated the phenomena, since an insulating crust was formed, which attached itself to the sidewalls. The crust prevented intimate melt-water contact and the heat transfer rate slowly decreased from approximately 2 to 0.1 MW/m^2, which is less than the decay heat input to the melt.

Three modes of heat removal from the melt pool have been identified. These are the (1) initial melt-water contact (2) the conduction through the crust and (3) melt eruptions into water, when the heat generated in the melt is greater than that removed by conduction through the crust. In the large test (120 x 120 x 20 cm), it appears that significant water ingression occurred since after the test the crust (or cooled melt) was 10 cm thick, i.e., about half the melt was cooled. Continued concrete ablation leads to the separation of the melt pool from the suspended crust, and the conduction heat transfer decreases substantially.

A 50 x 50 x 25 cm integral melt coolability test with siliceous concrete was performed recently whose results were approximately the same as for the earlier tests. Further separate-effects tests are planned. Presently, no definite experimental proof of melt pool coolability with a water overlayer can be offered. However, it appears that crust can not be maintained as a solid body for spans of several meters found in prototypic-geometry containments.

Melt coolability has been investigated at FZK in the COMET facility (Alsmeyer, 1998) employing water entry at the bottom of the melt pool. This new approach works since it has been found that the injected water creates sufficient porosity in the melt pool to cool the melt in a relatively short time. Several experiments have been performed at different scales with Al_2O_3 and iron melt pools to prove the concept. The concept has been directed towards the design of a core catcher for a new containment design at FZK. The core catcher top face is made of some tens of millimeters of sacrificial concrete, under which nozzles are embedded in the basemat. These nozzles open when the concrete is ablated and inject water from the bottom into the melt pool. The COMET concept has been optimized through many experiments. No steam explosions have been experienced. It appears that addition of the sacrificial concrete in the Al_2O_3-iron melt considerably reduces the explosivity of the melt.

Presently, the physical mechanism that creates porosity in the melt with water injection from below is not known. Research towards understanding of this physical process is underway with the support of European Union.

C.1.12. Fission product release and transport during ex-vessel accident progression

The fission products and the core materials released during the core heat up process arrive into the containment, as aerosols. Their transport in the containment is governed by aerosol physics, which determines the fission product concentration in the containment atmosphere as a function of time. As mentioned earlier, if there is steam atmosphere in the containment (as it should be for a severe accident), the fission product aerosol concentration in the containment atmosphere decreases exponentially with time, largely due to the process of aerosol particle size growth (due to steam condensation), agglomeration and sedimentation. Another aerosol deposition process active is that of Stefan flow carrying aerosols to the walls of the containment where the steam is condensing. As mentioned earlier, typically, fission product concentration in the containment atmosphere can decrease by a factor of 10^{-4} in about four hours.

The release of fission products during the ex-vessel accident progression can occur during the MCCI due to the gas sparging and the high temperatures in the melt. The releases of interest are those of the less-volatile fission products e.g. Ba, Sr, Ce, Ru, MO, since the volatile fission products have already been released.

The ACE experiments provided systematic data on the release of the above-mentioned fission products. In general, it was found that the releases were much smaller than what were previously calculated. The measured values for releases were less than 1% of the inventory for all of the less-volatile fission products. Recently an analysis of the ACE experiments points out that these releases occurred after all of the Zr contained in the melt had been oxidized. If such was not the case, the fission product releases could be larger. Thus, some uncertainty has been created with respect to the implications of data obtained in the ACE tests. One or two transpiration experiments may be able to determine the effect of the unoxidized Zirconium on the release rates of the less-volatile fission products.

Management of the iodine concentration in the containment immediately after the accident and for the long term is essential in order to reduce the potential of harmful releases due to containment leakage or other events. In this respect, the processes of concern are (i) the interaction of iodine with paints on containment surfaces to form organic iodine, which is difficult to remove and (ii) the radiolytic formation of iodine. Thus, iodine chemistry in the containment is important and the use of p-H control to reduce the iodine concentration is needed for the long term management of the iodine concentration.

There has been much research performed on the iodine chemistry over the years, particularly in Canada. Recently some additional work on iodine chemistry in the containment has been initiated in France. A thorough review of the past and currently on-going research is needed. The iodine-paint reaction chemistry may be a plant-specific issue.

C.1.13. Filtered-vent performance

The PWRs and BWRs in many European countries have all been fitted with filtered vents. Similar plans are under consideration at other plants. The performance of different filtered vent designs was confirmed in the tests performed during the LACE Project supported by a consortium of international organizations and managed by EPRI. Full-scale prototypic filters were employed and the decontamination factors (DF) measured were very large (10^3-10^5). Further tests with specific full scale filters have been performed, more recently, e.g., at Paul Scherrer Institute. In general filtered vents provide a relatively safe way of relieving the pressure in containment. Some large PWRs, however, have not chosen to consider filtered vents.

C.1.14. Conclusions

The intensive research work on severe accidents initiated world-wide after the TMI-2 accident has borne fruit in several ways. The work identified new vulnerabilities for the LWR vessel and containments, but also provided answers to several questions and increased knowledge to the extent that a majority of the in-vessel and ex-vessel accident progression issues are resolved.

The most important remaining issues for the current plants are concerned with accident stabilization and termination, and with the containment loads. In the former category are the in-vessel and ex-vessel coolability; whether the corium is in the form of a melt pool or of a debris bed or a combination thereof. In the latter category are the steam explosion and the hydrogen transition to detonation or detonation loads. Current indications are that with more experimentation, understanding of the phenomena and model validation, the steam explosion and the hydrogen loads issues could be resolved. Of particular importance is to understand the reason for the very low explosivity of the UO_2-ZrO_2 melt.
In-vessel melt retention concept is relatively well-investigated for medium power reactors (≤600MWe). Regulatory approval has been granted for the Loviisa plant (≤500 MWe). Its feasibility for reactors of high power density (≥1000 MWe) does not, currently, appear to be promising. Ex-vessel melt retention concepts based on similar ideas as the in-vessel melt retention concept hold promise. Their feasibility for new reactor designs is worth examining.

In-vessel coolability, if it can be proven to be effective would, perhaps, be the best solution for melt stabilization and accident termination since it precludes the consideration of the melt-containment interaction processes and the loads they impose on the containment. The concept of water ingression in the corium debris, and of gap cooling which may occur, while the vessel may be undergoing creep deformation, should be investigated further.

The concept of spreading the melt and cooling a low-depth melt pool with a water overlayer has merit. The work performed so far on melt spreading has provided reasonable assurance for this concept. However, for the EPR, the holding crucible design and time of failure are quite crucial for the success of the spreading concept.

The issue of ex-vessel melt pool coolability by a water overlayer has not been resolved yet. The COMET concept appears to have promise for achieving melt quenching and stabilization and its application could be considered for new plants. Perhaps, backfits involving downcomers may be effective for current plants.

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C.2. Relevance of SA Research to SAM Requirements and Implementation

C.2.1. Introduction

Severe accident management (SAM) is the use of existing and alternative resources, systems and actions to arrest and mitigate accidents that exceed the design bases of nuclear power plants (8)

Most European nuclear plants have implemented or are in the process of implementing SAM measures. The objective of this chapter is the analysis of the relevance of SAM and the results of severe accident research. Also, this task will attempt to determine whether additional research results in the field of severe accidents are needed to back up the SAM.

C.2.2. Implementation of Severe Accident Management

A report published by NEA-OECD in 1996 (1) gives a general overview of the state of SAM implementation and technical approaches followed by some NEA-member countries. Also, it contains an evaluation of uncertainties and open items related to SAM.

Westinghouse and BWR Owners Groups (6, 7) have published generic severe accident guidelines (SAMG) which were adapted to individual plants. Some nuclear plants in Europe will closely follow these guidelines produced by these Owners Groups, and the implementation of SAMG for those plants will be essentially limited to adapting the set points, curves and computing aids to the specific plant features. No substantial plant backfitting is foreseen. In France, Germany, Sweden (for BWRs) and Finland, the approach is much more open, since no generic standard guidelines are used and more individual plant work is required to devise and implement SAM. Some other countries are using a combined approach which is open to the possibility of backfitting the plant with new equipment specially designed to deal with severe accidents. The SAMIME EU-Concerted Action is providing information on this question.

In the following paragraphs, the structure presented in the OECD report has been employed to group the SAM actions under four main functions: Cooling a degraded core, managing combustible gases, managing containment temperature, pressure and integrity, and managing the release of radioactivity.

<u>C.2.2.1. Cooling a Degraded Core</u>

Adding water to the reactor vessel (RPV) is an action that is very similarly implemented in many countries. There is a general agreement that the hazards posed by increased hydrogen generation; possible recriticality and increased steam production do not outweigh the benefits of retaining the degraded core in vessel. The criteria generally followed for this action is to supply to the reactor vessel with water as soon as injection capability is available. Westinghouse Owners Group (WOG) standard guidelines contain warnings about the side effects of increased hydrogen production, and their computing aids take into account in a simplified way the additional risk of hydrogen combustion in the containment. The issue of recriticality is generally considered to affect more the BWR, where borated water sources are less available and early control rod material meltdown and relocation is a possibility. General electric (GE) standard guidelines specify the use of the Liquid Control System in case of core melt criticality, but no criteria are given on the water flooding rate.

RCS depressurisation is also a generic SAM action that can be accomplished in a variety of ways. The preferred way for PWR is the "feed and bleed" system, adding water to the steam generators and depressurising the secondary side thereby cooling down the primary side and reducing its pressure. If this action is ineffective, depressurization can be accomplished by direct opening of pressurizer valves. There are numerous benefits to intentional depressurisation, i. e. alternate means of cooling become available, and high pressure melt ejection is avoided, although there are also possible drawbacks, like increased H2 production and higher probability of in-vessel energetic fuel-coolant interaction. All PWRs have pressurizer valves that can be used, although sometimes pressurizer spray is a possibility. All BWR are designed to be easily depressurised through dedicated systems (Automatic Depressurisation System) and can be manually depressurised in case of ADS failure.

The action of containment initial flooding in order to delay vessel failure by means of cooling through the vessel wall, is one where there is considerable variation among countries. It is recognised that the action can not by itself guarantee vessel integrity, especially for reactors with higher power (3), but the action may delay vessel failure.
The reactor cavity of PWR, or the drywell of BWR, must be flooded up to the upper level of the active core in the reactor vessel, for this action to be effective. The existence of non-vented vessel skirts in most BWRs will imply the accumulation of non-condensable gases below the vessel wall and preclude wall-water contact in some areas. Plant specific questions, like specific heat transfer phenomena, and potential of fuel-coolant interactions must be addressed in any assessment of this action. Heat transfer must consider the plant specific design, the reduction of heat transfer due to degradation of vessel insulation, and the unavailability of flow path (3) to evacuate steam. Containment flooding to several levels is recommended in the standard WOG Severe Accident Management Guidelines (5), although specific implementation will depend on the design of the reactor cavity. Generic GE standard guidelines (6) recommend drywell or primary containment flooding as an integral SAM action that could provide a means of core cooling through the vessel wall, and also as a possibility of alternative vessel flooding through the relief valve tail-pipes. German plants do not consider cavity flooding and continue the concept of “dry cavity”. Finland has implemented the strategy in Loviisa plant. Swedish and Finish BWR have also implemented the strategy that a water pool is created under the vessel as soon as the water level may falls below the top of the core. However, the level of water does not reach the vessel and the vessel wall is not cooled. In case of BWRs, there is a forest of control rod guide tubes under the vessel, with their drives (both hydraulic and electrical in some plants). Degradation of their operation may occur due to submergence in water.

C.2.2.2. Management of Combustible Gases

There are considerable variations in the strategies followed to reduce H_2 and CO inventory in the containment, because of the differences in existing equipment and the status of implementations. Many countries have decided on the use of catalytic recombiners in PWR containments, which can reduce H_2 and CO concentrations while keeping containment pressure low. Some BWRs and some PWRs use igniters to produce intentional H_2 or CO burns. Venting of the containment is a strategy considered also for the reduction of combustible gas inventory.

Catalytic recombiners have demonstrated their capability of reducing H_2 concentration under steam-inerted atmospheres, very low H_2 concentrations, and presence of aerosols (1). Installation of recombiners has been decided in Belgium, Germany, France and the Netherlands and in some Eastern European countries. Finland has decided on the installation of a new H_2 management system using recombiners, although currently igniters are being used. G.E. BWRs with Mark I containments and KWU German BWRs of old design are inerted and do not use ignition devices. G.E. BWRs with the larger Mark III containments have ignition systems.

C.2.2.3. Management of Containment Temperature, Pressure and Integrity

Automatic or manual initiation of containment sprays to condense steam released exists in most BWRs and PWRs although there is a significant variation in the equipment dedicated to the implementation of this action. Sprays are also used, in the longer term, in conjunction with heat exchangers, which can extract heat from the containment to avoid pressurisation. Spray systems, usually, are an operational mode of the safety grade core cooling system. German plants do not have spray systems. Many plants have alternate spray sources, such as the fire protection system. Swedish plants have an independent dedicated spray system. Loiivisa in Finland and Zorita in Spain have external spray systems for their steel containments. External sprays have been installed in two Belgium plants with steel containments.

Fan cooler systems in PWRs can extract heat and avoid late pressurisation due to release of non-condensable gases during MCCI, but not all plants have fan coolers as qualified safety grade equipment. The initiation of fan coolers for SAM in PWR containments is considered in Belgium, Spanish and UK plants, and it is included as a standard action in WOG SAMG.

Containment flooding is considered both in PWRs and BWRs. Also, a consensus is developing that initial containment flooding will improve the chances of ex-vessel melt coolability (12) in case of vessel breach, in spite of the higher risk of energetic ex-vessel melt water interactions, and will reduce ex-vessel radioactive releases. Here we have to distinguish between PWRs with their larger and relatively strong containment and BWRs with their small containments and perhaps vulnerable vessel support structures, whose integrity may be threatened by a highly energetic steam explosion.

There is a considerable variation in the actual implementation of the preemptive containment flooding strategy. Some considerations which must be taken into account are: (1) time needed to flood a large volume, (2) side effects of the type of water available, (3) structural capability of the containment, and (4) effects of containment venting. Build-up and leakage of contaminated water is a concern in the long-term (2), management of the accident.

Flooding of the containment up to the level of the active fuel is considered as a standard action in WOG, although its implementation will depend on the plant specific design. GE standard SAMG also recommend flooding of the drywell or primary containment above the level of the active fuel, coupled with containment venting if necessary to facilitate water injection and reduce containment overpressure. Belgium, France and Finland do not, presently, consider this as an accident management option. In Germany, injecting cold water is recommended, in combination with filtered venting, to prevent sump water evaporation due to de-pressurisation and to reduce vent opening duration. However, the containment cavity itself is not filled with water. In Finland and Sweden for their BWRs this strategy is implemented using an independent dedicated system.

Many European plants include the strategy of containment venting, to avoid late failure due to over-pressurisation. Scenarios like complete loss of containment heat removal capability, or full power ATWS in BWR, are typical examples where containment venting becomes essential. This accident management action can avoid late failure due to pressurisation by non-condensable gases released during MCCI, for which containment heat removal systems are ineffective. Venting can be used also to ease containment flooding, and to reduce the inventory of combustible gases. Considerable variation exists in the implementation of this SAM feature (1). The standard WOG SAMG do not include containment venting as a SAM strategy. Venting of containment with specially designed filtered vent systems is implemented in all PWRs and BWRs in France, Germany, Sweden, Netherlands and Switzerland. U.K., Belgium and Spanish PWRs do not have venting. Spanish BWR have a dedicated manually operated venting system, which connects the suppression pool airspace to the off-gas stack, without filtering.

C.2.2.4. Management of Radioactivity Releases

Standard strategies for mitigating the rate of radioactivity release through openings in the containment boundary include reducing the containment pressure, by means of available containment heat removal systems and through the venting systems. At later times in a severe accident revolatalization releases from the deposited aerosols in the RCS become a concern. Mitigation of those releases will involve cooling of the RCS walls.

A common strategy, for reducing the inventory available for release in the containment, is the initiation of containment sprays in PWR and BWR. Sprays were designed for early operation and steam condensation after LOCA, and not for long term operation during severe accidents. However, sprays can produce effective aerosol deposition (8) due to interception of droplets. Also, sprays can remove some of the gaseous molecular Iodine as long as they do not become saturated with I. The effectiveness of sprays will depend on the availability of AC power and the extent of the area covered by the spray system. Iodine volatility in many PWR is reduced by means of additives that are included in the design of containment sumps, or the containment spray system.

Engineered filtering systems are installed in most PWR and BWR, with HEPA filters generally designed for conditions of normal operation. Use of engineered filtering systems during severe accident environmental conditions is possible, but the efficiency of the filtering may be reduced

(8), if additional technical features have not been provided (i. e. emergency filtering systems). A number of containments have a filtered venting system (see above) specially designed to deal with severe accident situations.

Removal of radioactive aerosol, by means of scrubbing in BWR suppression pools, is a beneficial side effect of the suppression pool functional design. Aerosol scrubbing by means of a water pool overlying the core debris is also considered, in standard WOG and GE standard SAMG, as a strategy to reduce ex-vessel releases to the containment.

Secondary side flooding is a standard strategy, included in WOG SAMG, for mitigation of releases to the environment due to SGTR accidents, and protection of SG tubes from creep ruptures.

C.2.3. Uncertainties and Open Issues in SAM

According to (1) there is sufficient information available to proceed with the implementation of SAM, but some issues are not yet closed, and countries should perform periodic reviews and updates of their SAMG, to incorporate new information. New knowledge about uncertainties probably will not change the presently recommended operator actions substantially, however, further research results will increase confidence in the robustness of current SAM strategies, reduce uncertainties about SAM actions, and also improve SAM training (1).

The perception of existing uncertainties is influenced by the different approaches followed by member countries and, consequently, expert opinion on uncertainties can vary considerably. However, there is a consensus that areas where research programs may provide information that could influence SAM implementation are:

- In-vessel core debris coolability
- RPV failure
- Ex-vessel flooding to provide in-vessel core debris cooling,
- Ex-vessel debris coolability and
- Fuel-coolant interactions

According to (1) there are also a number of open SAM issues that should be potentially considered in SAM guidelines, and they could cause current SAMG to be supplemented or modified. Examples are accidents occurring from a low power or shutdown condition or accidents where the reactor is not scrammed. Long term recovery aspects of accident management have been discussed in a recent OECD report (2), and they should also be considered in SAM decision-making, as well as the effects of plant ageing.

C.2.4. Severe Accident Management Objectives and Their Relationship to Severe Accident Research

A comprehensive list of individual SAM objectives is included here, both for PWRs and BWRs. Each objective is compared to the state of knowledge obtained from research, and a judgement is made on the need of additional data. The information on research knowledge has been obtained from programs which are generally available to European countries, such as the NRC-supported CSARP, the OECD-NEA reports, and the EU 4th Framework Program.

Some phenomena discussed here are directly related to SAM effectiveness. In other cases, phenomena are related to SAM only in that they can cause a dynamic change in the accident conditions or the integrity of the safety barriers: RPV and reactor containment.

1st Objective: Achieve a coolable controlled core state

A coolable controlled core is defined as core conditions under which no significant short term or long term physical or chemical changes would be expected to occur. The core temperature must be well below the point where chemical or physical changes might occur, and a long-term heat sink must be available for transferring all energy being generated in the core.

OBJECTIVES	RELATIONSHIP TO SEVERE ACCIDENT RESEARCH
The core is totally within the reactor vessel	
1. Maintain sub-critical degraded core.	The issue of re-criticality has been addressed in severe accident research, especially for the case of BWR, where unborated water sources are more readily available, and early control material melt-down may be expected. Examples are EU-SARA project, and NRC-CSARP. Conclusion of CSARP research (7) is that an eventual recriticality event would result in low power spike, which in turn would induce faster core degradation. The nuclear fission reaction would be finally stopped, either due to boron injection, or to depletion of water inventory in the core or to loss of core critical geometry due to core degradation. However, a continuous supply of unborated water might result in a power level in the core, limited by the thermal balance defined by in-vessel water steaming and fission power generation. The SARA Project results are similar, however of more severe consequences. For example for water addition rates of 500kg/sec, it is possible to have the core power reach ~50% of nominal. The initial power spike may also be large enough to exceed the threshold values for the energy deposition of 280 kcal/gram for fresh fuel and 70 kcal/gram of burned-up fuel, which may have the consequences of fuel fragmentation and dispersal.

OBJECTIVES	RELATIONSHIP TO SEVERE ACCIDENT RESEARCH
	Some additional evaluative type research may be necessary.
2. Cooling the damaged core by in-vessel flooding.	The effect of water addition to a core which may have suffered local damage but has kept its geometry relatively intact, has been addressed in the EU supported QUENCH experimental program, which investigates the increase in H_2 generation due to quenching as a function of the clad surface conditions. The issue of coolability of a core which has melted and the melt may have relocated to the lower plenum is much more uncertain. The FARO experimental results show that deep low porosity debris beds may be formed for whom the dryout heat flux is very low and they may be difficult to cool, even if reflooding is accomplished at that stage. In case of continued lack of water availability the debris bed will dry out and remelt and a convecting oxidic pool topped by a metal rich layer may be formed. Reflooding at that point in time could involve the possibility of a stratified stem explosion whose yield and consequences have not been completely evaluated. Studies have been and are being performed on the mechanism of gap cooling as may have occurred in the TMI-2 accident. Experiments performed in U.S.A. and Japan have indicated the possibility of quenching the segment of the vessel wall just below the water overlayer. Experiments in Germany with a partially filled vessel and a pre-existing few millimeter gap have indicated successful cooling of the vessel wall. The Swedish experiments have indicated that even with up to 10% creep, no gap could be maintained between the crust of the molten pool and the vessel wall. Experiments on gap cooling are currently being performed in Korea with high temperature aluminium oxide in steel vessels. Experiments on gap cooling will also be performed in Sweden. The present expert opinion on gap cooling, as a practical in-vessel coolability mechanism for the existing reactors, is that it is subject to too many uncertainties to be considered as a robust reliable concept for prevention of vessel failure. Further experiments on both the quenching of the in-vessel low porosity heat generating particulate beds and on the gap cooling of the vessel wall should be pursued, to obtain definitive information about this concept.

OBJECTIVES	RELATIONSHIP TO SEVERE ACCIDENT RESEARCH
3. Cooling of a damaged core by external reactor vessel flooding.	This is the accident management strategy of in-vessel melt retention (IVMR) which has been promulgated in Loviisa and is in the design of the AP-600. The research results on melt pool convection obtained with simulant materials have been complemented by the RASPLAV(14) experiments which have exhibited the possibility of melt pool stratification. Stratification increases the thermal loading on the vessel wall and may affect the focussing effect. Experiments are being performed in the SIMECO facility (16) at RIT in Sweden to obtain quantitative data on these differences introduced by the melt pool stratification. The MASCA program (15) will perform experiments to determine the existence of stratification in various reactor compositions. The experimental programs MASCA, SIMECO, COPO may attempt to obtain some relevant data in next 2-3 years. Additional simulant material and prototypic material tests may be required. Physical properties of the stratified layers would have to be measured. The solidus temperature of the upper layer determines the radiative heat transfer to the vessel environment above the melt pool.

2nd Objective: Maintain Containment Integrity

The objective is to keep the containment intact so that the last barriers to release of the radioactivity to the environment is maintained.

4. In-vessel steam explosions.	In vessel steam explosion research results has led to the consensus of experts that the alpha mode failure of the containment is extremely unlikely (conditional, on core melt, probability of $<10^{-4}$). Some ongoing programs e.g. BERDA and ECO at FZK are directed towards a deterministic demonstrating of the same conclusion. Experiments have been performed in the BERDA project with a high velocity slug impacting on the upper head. The continued integrity of the upper head and the bolts for very large nomentum slugs provide a convincing argument against alpha mode failure. The effect of the in-vessel steam explosion on the vessel penetrations has not been evaluated. This could lead to an early vessel failure, if the weld around the penetrations fails.

OBJECTIVES	RELATIONSHIP TO SEVERE ACCIDENT RESEARCH
When a substantial amount of the core has relocated out of the vessel	
5. Ex-vessel steam explosions	Ex-vessel steam explosions can not be ruled out, if a pool of cold water is maintained, either in the lower dry wall of a BWR, or in the cavity of a PWR. The conditions of low pressure and the high subcooling of the coolant, actually are more conducive to the occurrence of a steam explosion. The experiments conducted in FARO and KROTOS facilities have found that the prototypic fuel mixture of UO_2+ZrO_2 is very hard to explode even with a strong trigger, while discharged into subcooled water at low pressure. Further research on this aspect is essential to support low yield estimates for ex-vessel steam explosions, otherwise for the ABB BWRs it could be an early containment failure issue. Consequences have not been evaluated for the Westinghouse PWRs in which an explosion occurs in the cavity very close to the failed vessel.
6. Cooling of core debris in the reactor cavity, and corium melt spreading.	The issue is critical, since it defines melt stabilization, accident termination and containment survival in the long term. There are several scenarios of core melt coolability (17) depending on the timing of the availability of water in the cavity of a PWR or in the dry well of a BWR. If there is no water available, or it can not reach the cavity (as possibly in the German PWRs), the core melt/debris will attack the basemat and melt-through may occur at a time depending on the thickness of the basemat. If there is substantial pre-existing water in the PWR cavity, or in the BWR drywell, then, a particulate debris bed could be created, whose porosity could be quite low. Occurrence of a small steam explosion would generate very small size particles which could worsen the porosity and create stratification. Such beds are not easy to cool since the dryout heat flux is very low. Concurrent with the formation of a particulate debris bed, there could be some fraction of melt which did not fragment and forms a melt pool. Alternatively, a poorly cooled particulate debris bed could melt and form a melt pool. Addition of water from top would lead to the configuration, which has been investigated in the MACE and the COTELS experiments (12). The MACE experiments have not demonstrated complete coolability, while the COTELS

OBJECTIVES	RELATIONSHIP TO SEVERE ACCIDENT RESEARCH
	experiments have shown coolability for the specific configuration employed. Melt pool coolability has also been demonstrated in the COMET experiments in which the water is added to the pool from bottom. The mechanisms of coolability are quite different between water injection from top and bottom. Employing the COMET concept would require backfits to the existing LWRs. Downcomers have been suggested (18) as relatively simple backfits in the LWR containments. They have been found to enhance the coolability of particulate beds. Melt spreading has been employed in new designs (EPR) to reduce the height of the melt layer that would have to be cooled. This is helpful towards coolability potential. The physical properties of the melt are important for its coolability. The MACE experiments show that addition of SiO_2 from the concrete into the melt composition make the crusts formed tough and almost impermeable so that a crust layer is able to isolate melt pool from the water coolant and preclude melt coolability. Melt/particulate debris bed coolability is a critical issue and further research work is highly warranted
7. Obtain a short-term heat sink, for certain severe accident sequences	Containment spraying, or vent opening, are the actions to mitigate a short-term pressure and temperature rise. No special research programs or additional date are deemed necessary to better delineate this particular challenge.
8. Obtain a long-term heat sink	A long-term heat sink can be either the containment fan coolers, or the RHR exchangers via ECCS and/or spray recirculation. Containment venting is also available for avoiding long term overpressure for most plants in Europe. No special research programs or additional data are considered necessary to better delineate this particular challenge.
Phenomena which can contribute to dynamic changes of containment conditions	
9. H_2 burns.	Concerning the generation of combustible gases, there are sufficient data to cover generation during initial core damage. Existing models to calculate H_2 generation during late phase degradation or core quenching are still uncertain. Additional data could be useful to make a more realistic assessment of H_2 generation. All the metal will be oxidized eventually.

OBJECTIVES	RELATIONSHIP TO SEVERE ACCIDENT RESEARCH
	Ex-vessel H_2 generation during corium quenching by water has been identified as a source where uncertainties remain (11) and additional data are necessary, due to the generation potential of this phenomena. Concerning H_2 mixing and distribution, research programs have been conducted in different countries, and analytical methodologies for detailed mixing calculations have been developed, which have been employed to determine location of passive recombiners (PAR) in the containment. Efficiency of PAR under adverse environmental conditions seems proved. Additional research programs in this area are not needed. Concerning H2 combustion, the issue of possible transition from deflagration to detonation (DDT) in containment rooms is considered of significance for SAM. The issue has been and it is still being investigated extensively [19]. Simple empirical models (Dorofeev, Sherman-Berman) are available for evaluating the probability of DDT in a given room, considering geometry and mixture composition. Additional understanding and review are needed to extrapolate research results to prototypic geometry. Research programs currently active should be pursued to completion.
10. Debris-liner contact	This safety issue was raised for Mark I BWR, due to the short distance that the corium melt would have to traverse to attack the containment liner, in case of vessel rupture at low pressure. The conclusion of research done in the frame of CSARP was that the probability of liner failure with water present in the drywell was too low to consider the phenomena physically reasonable. No additional research is necessary.
11. MCCI	Prediction of basemat attack can be performed by means of available simplified models (CORCON and WECHSL). A new model (12) exists that couples thermal-hydraulic and physical chemical effects. Additional work would be needed to implement the new model in existing predictive codes. Also research is needed on the development of predictive models for long-term MCCI.
12. Direct Containment Heating Containment	The issue has been resolved (CSARP) for Westinghouse plants, with finding of no significant failure probability, based on evaluation of containment loads and fragility. Models are available to determine containment pressurisation and heating due to DCH, which have been developed in the frame of CSARP for PWR-typical geometry. SAM action of vessel depressurization for mitigation of DCH is effective. No additional research programs are needed

3rd Objective: Fission Product release prevention, termination and mitigation

SAM actions are directed at reducing the inventory of radioactivity available for release, or reducing the release rate.

CHALLENGES	RELATION SHIP TO SEVERE ACCIDENT RESEARCH
13. Reduction of airborne inventory in the containment by "natural" deposition	Research about deposition mechanisms [20] inside the containment is being conducted in the frame of Phebus FP project and other facilities in the EU, such as PITEAS and AHMED. Project STU has performed research on uncertainties related to source term, and concludes that aerosol deposition physics in containment is fairly well understood, and main uncertainties arise from the prediction of flow conditions and the coupling of aerosol physics to thermal-hydraulics in multicompartment geometry. EU project APC identifies hygroscopy and diffusiophoresis as sources of uncertainty. In large containments, gravitational sedimentation is the predominant "natural" decontamination mode. From the point of view of SAM, additional research is only justified if a substantial reduction of uncertainties can be achieved, because natural deposition is a relatively slow decontamination process, and usually engineered systems (sprays) are initiated for SAM, when there are threats to containment integrity (8).
14. Removing suspended aerosol by means of spray.	Sprays were originally intended for reducing steam pressurisation of the containment, and not for aerosol removal, and they are not designed for the prolonged release of radioactivity expected in a severe reactor accident (8). However, they can be very effective in particle removal. Physics of aerosol removal by sprays seems to be well understood, and validated models are available (EU-STU project). However, chemistry effects are less well understood, and it is important from the point of view of SAM to preserve the removal efficiency of the spray system over the long period of a reactor accident (8). Chemistry plays a major role in spray efficiency in the long term. An example of uncertainty is removal of organic I species, which according to result of Phebus FP tests could be the species predominant in the long term. There is also uncertainty on the long-term capability of spray systems, after they become saturated with I (8). Additional research on the efficiency of sprays in the long-term is justified.

15. Reduce release rate by FP scrubbing	Scrubbing is applicable to the flooded secondary side of the SG during a SG tube rupture accident and to BWR steam suppression pools. FP scrubbing is a very complex phenomenon. According to EU project STU, pool-scrubbing models are well validated for low-pressure differential injection into containment pools, as in BWRs. Good experimental studies have been conducted (8). There are less data available to validate models in the jet injection regime applicable to SGTR accidents, where pressure differential is greater. Research on retention of FPs in the pool for high injection velocities, and the influence of surfaces present in the injection area will be performed in the EU-SGTR project
18. I removal, including reduction of long term release of gaseous I	Some plants have been traditionally equipped with systems able to remove Iodine expected under DBA, when only some limited gap release is expected (i. e. HEPA filters). Other plants rely only on natural deposition mechanisms to mitigate airborne I. During a severe accident a substantial I release can occur, and long term generation of gaseous I is a concern because of the ability of I to partition from water into the containment atmosphere. Most plants have installed measures to control the pH or to fix volatile I in water, by means of additives to the spray or sump water, in order to limit the gaseous iodine generation. Maintaining high values of pH in the water bodies present in the containment, is an effective measure to prevent conversion of I compounds into gaseous volatile form (8). According to EU-STU project, I speciation could be of significance only for accident sequences where filtered venting is expected, due to the different filtering efficiencies of I gas species compared with aerosol species. However, presence of gaseous I forms in the containment atmosphere is always a concern, especially for plant site operators. I behaviour in the containment has been the subject of two current EU projects (IC, OIC). Also, much work has been done to explain findings of Phebus FP tests FPT0 and FPT1. Generation of volatile organic I compounds, in chemical reactions with containment paints, has been identified in Phebus FP tests. Additional research would be needed to delineate the phenomena, and to study efficient ways of reducing the release of organic species.

4th Objective: Maintain Monitoring and Forecasting Capability. Maintain Equipment Capability.

These objectives are concerned with the usability of the instrumentation to monitor and forecast the progression of a severe accident. Several factors determine the usability of instrumentation: especial environmental conditions created by the severe accident, submergence in water, and the availability of electrical power. Other factors are the range of the validity of instrumentation and the capability to repair equipment.(21)

EU project ASIA of the 4th FWP has addressed the development of a methodology to assess the survival potential of certain instruments during severe accidents, and also the development of algorithms for the purpose of signal validation and accident identification. However, no other projects in the research literature have been found. In principle, research on the survivability of specific instrumentation and equipment is the responsibility of vendors. However, the question of forecasting and diagnosis capability of plant operators during severe accidents, is more generic and can be the subject of research.

New projects, addressed to the development of generally applicable algorithms and tools to improve forecasting and diagnosis capabilities, would help in the implementation of SAM and operator training.

C.2.5. Conclusions

- There is sufficient information available on the subject of severe accident phenomena to implement adequate SAM guidelines for existing plants.
- However, additional knowledge gained in some areas where uncertainty still remains, will contribute to a better assesment of SAM, and to an increase of confidence in SAM effectiveness, and also to an improvement in plant personnel training.
- Some phenomena discussed here are directly related to SAM effectiveness. In other cases, phenomena are related to SAM only in that they can cause a dynamic change in the accident conditions or the integrity of the safety barriers: RPV and reactor containment.
- The relationship between SAM challenges, for both PWRs and BWRs, and the state of "open" severe accident research programs has been analysed. Research programs exist on most phenomena which constitute challenges to the effectiveness of SAM for present reactors. However, needs of additional data and analyses which would help to clarify some questions on SAM effectiveness have been identified.

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Projects of EU 4th Framework Program

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CSC: Corium Spreading and Coolability
SARA: Severe Accident Recriticality Analysis
STU: Source Term Uncertainties
HDC: Multidimensional Simulation of H2 Combustion and H2 Distribution in Containment
IC: Iodine Chemistry

OIC: Organic lode Chemistry
APC: Aerosol Physics in Containment
ASIA: Algorithm Support for Accident Identification and CSF Signal Validation.

C.3. Relevance of SA Research to PSA and Risk-Informed Regulatory Approaches

C.3.1. Background

Historically, the Western nuclear reactor licensing process has been based on deterministic regulatory requirements, involving defence-in-depth, and the use of multiple barriers to fission product release (i.e., fuel, reactor coolant system boundaries, and the containment system). To account for the uncertainties associated with the design, operation, and phenomenological processes impacting the conformance with deterministic regulatory requirements, sufficient conservatism were built into the analysis tools, and the deterministic regulatory criteria. Plant design requirements have been derived through the analysis of Design Basis Accidents (DBAs), supplemented by the single failure criterion to ensure an adequate level of reliability for safety systems [1].

DBAs are a set of occurrences selected to envelope credible accident conditions, and to ensure that these accidents could be accommodated within the design envelope. The probabilistic safety assessments (PSAs) have confirmed that the risks of nuclear reactor accidents result from events that occur outside of the design basis domain, and are due to multiple failures, human errors, and external initiating events [1].

The western nuclear regulatory process has evolved from the initial “engineering judgment” framework of the 1960's, the prescriptive deterministic requirements of 1970s, the transition years of the 1980s, to the present day movement toward risk-based approaches.

In recent years, events that were and would be contributors to risk in PSAs have become a greater focus of regulatory efforts than in the past, but quantifying the significance of these events and a subsequent reallocation of utility and regulatory resources have not fully been realized. The deterministic design basis of the 1970s is just being revisited in some of the western countries (e.g., United States). This is because; the deterministic design basis has grown to cover the ever-broadening events and requirements, whose benefits are not always clear. A few examples might be appropriate:

1. Fires are typically found to be significant contributors to risk (especially in older plants). In the United States a very prescriptive requirement has been developed (10 CFR50.48/Appendix R) and implemented, with the objective of reducing the consequential impact of fires in nuclear power plants. Despite all the requirements and plant changes, safety assessments conducted since implementation of this rule still shows fires to be significant contributors to risk in some plants.

2. Environmental Qualification (EQ) of equipment requires a massive testing and documentation, using licensing basis accident conditions, and allows no recognition,

that not all safety related components are equally important. The reality is that the relative importance of components and equipment subject to the EQ varies by orders of magnitude. The risk is dominated by a relative handful of equipment. In addition, the radiation doses used for EQ-purposes have been established based on non-mechanistic bases. These could be re-evaluated based on the current knowledge-base.

3. Because of the large degree of conservatism in the original design, many plant systems operate on the edge of the “acceptable performance.” This has resulted in a large expenditure for testing, maintenance, and replacement of major components to persevere performance at licensing-based limits not related to risk.

In recent years, as PSAs are becoming more wide spread and their benefits more transparent; PSAs are being used with increasing effectiveness to enhance plant safety in a more integrated and consistent framework. Use of risk assessment in regulatory safety would allow the industry and regulators to focus on the "important" systems and issues and, at least by implication, stop wasting those resources, which have been devoted to unimportant uses.

C.3.2. Risk-Informed Regulation

Regulation is defined as the entire process of interactions between the licensee and regulatory authority, encompassing legal, design, and operating requirements; inspection activities; and performance assessment [2].

Risk-informed regulation involves the whole area from implicit probabilistic considerations in the traditional deterministic requirements, to an intensive use of probabilistic safety and risk analysis results in optimization of regulatory attention, enforcement of regulatory requirements, and for more efficient utilization of resources to enhance safety improvements by plant owners and utilities.

Closely related to risk-informed regulations is the concept of performance-based safety requirements or rules, which are intended to focus the regulatory process on the desired safety results instead of on the methods used to achieve those results [3]. Such requirements or rules do not specify the process, but instead they establish the desired goals to be reached, and how the achievement of such goals can be judged [4]. The inspection and enforcement activities are then to focus on the confirmation of whether the overall goals have been attained.

The general objective of risk-informed regulation is to define requirements, which are consistent with the risk importance of the equipment, events and procedures to which the requirements apply. The stringency of the risk-informed requirements should be directly related to the risk and safety importance of the contributor being regulated.

The general objectives of risk-informed concepts are consistent with the overall objectives of the existing nuclear power regulations and safety philosophies in many European countries. However, the concept of risk-informed regulation differs from the existing, by in large, deterministic approach to regulations, in terms of the approach to implementation of nuclear safety objectives. Most current regulations have been devised without explicit consideration of risk importance of the contributors. Instead, the existing regulations have been developed based on a relatively ad hoc qualitative perception of important contributors, using subjective engineering judgment with large degree of conservatism built into the deterministic regulatory requirements. What sets apart the concept of risk-informed regulation from the existing approach is that risk-informed regulation is based on a strong dependence of the regulatory decisions on insights based on plant-specific PSA results. The regulatory requirements are specifically tied to the risk importance of the contributors through the PSA; therefore, providing a systematic, consistent, justifiable and audit-able basis for regulatory requirements.

The objectives of risk-informed regulation dictate that the regulatory requirements are commensurate with the risk contributors (i.e., regulations should be more stringent for risk important contributors, and less stringent for risk unimportant contributors). Therefore, provided risk-informed regulatory criteria are appropriately developed, a systematic and efficient expenditure of resources are to be expected, while, simultaneously, a balance in overall safety of nuclear power plant can be achieved. These objectives would further strengthen the traditional multi barrier (i.e., "defense-in-depth") safety philosophy, and provide a quantitative means of demonstrating compliance (or degree of non-compliance) with regulations.

There is a difference between risk-based and risk-informed regulatory concepts. In the risk-based regulatory concept, the focus is placed predominantly on the results of plant-specific PSAs/PRAs in a more stringent and rigid fashion. On the other hand, the concept of risk-informed regulatory process would not necessarily demand such a rigid reliance on the PSA results; instead, the insights drawn from plant-specific PSAs/PRAs could be utilized to assess the relative importance of the various safety issues. In such an approach, the uncertainties associated with the numerical results of the PSAs/PRAs are expected to have a less significant impact on the decision-making process. In either concept, increased emphasize is placed on the quality, completeness, and the methodological vintage of these studies. Therefore, it is very crucial that the methodological and scope of PSAs/PRAs is consistent with established PSA/PRA standards. This necessitates the development of risk standards for use in performance of plant-specific studies to be utilized in regulatory applications. This is not be interpreted that all the existing PRAs/PSAs will need to be redone, but it is to emphasize that the living PSAs/PRAs need to stay abreast of the recent development, and by definition, they need to be consistent with established methods, data, and other standards.

To summarize, the overall objectives of risk-informed regulations include [5]:

1. Enhancement of safety by focusing regulatory attention and licensing resources in areas commensurate with their importance to public health and safety.

2. Process by which regulatory oversight can be rendered within a framework that uses risk insights and information.

3. Process by which risk information can be utilized to provide flexibility in plant operation and design, which can result in reduction of burden without compromising public health and safety.

The goals of risk analysis are to estimate the severity and likelihood of harm to human or the environment occurring from exposure to a risk agent. Mathematically, risk is defined as:

R = Frequency of undesirable events x Consequences

Specifically, for a complex system (i.e., a nuclear reactor), the frequency of the undesirable event could be divided into its constituents, and provided the consequence measure is defined, the risk can then be estimated.

Development of a PSA model is very specialized. It requires the build-up of Boolean logic models of the various systems; collection and analysis of historical events/data; collection and use of component failure data; detailed knowledge of engineering systems and their operations; development of system success criteria and analysis of accidents using models based on uncertain physical and chemical processes; and incorporation of man-machine interactions (i.e., human factors), to list a few. One of the most difficult problems in the quantification of risk is the assessment of consequences of severe accidents. These consequence measures can be defined in many forms, including:

1. Typical consequence measures defined based on the results of the so-called "level-2" PSAs such as:

- Radiological release quantities
- Activity associated with radiological releases (which can be estimated by extending the calculations of releases to include radioactive decay and transmutation leading to assessment of release activities).

2. Typical consequence measures defined based on the results of the so-called "level-3" PSAs such as:

- Ground contamination level
- Prompt and latent fatalities
- Radiation dose
- Economic impacts

Therefore, it is obvious that the uncertainties in quantitative estimates of risk are directly related to the uncertainties in the severe accident and source term phenomenological prediction capabilities, among other things.

Risk-based regulations require well-identified risk targets, which can be and used to assess plant and design performance. The probabilistic safety criteria (i.e., safety goals) should be viewed as economic optimums [6-7]. These criteria are typically viewed as aspiratory targets. Therefore, risk reduction well below these targets will impose great economic burdens, including large capital and operating costs. On the other hand, exceeding these criteria significantly could have large economic and social consequences as a result of nuclear accidents, with the constraints that an adequate level of safety that must be assured without regards to cost. However, beyond this level of safety, cost and social implications must be considered in dealing with safety improvements.

Safety goals and probabilistic safety criteria have been proposed by various regulatory organizations, and for use in various industries. Results of recent PSA studies show that the risk of severe accidents are for the most part, mitigated by the existing defense-in-depth approach to design, and by-in large, the existing approach to regulations has been effective in protecting public health and safety. On the other hand, these studies also show that the potential for very large, rare, radiological releases cannot be ruled-out. Therefore, given the public perception of nuclear reactor accidents, it is difficult to gain the confidence of the general public, unless both the frequency and the consequences of such accidents can be shown to be at insignificant levels with a large degree of confidence. Accident and radioactivity release frequencies could be reduced substantially, through a balanced, risk-informed improvements and/or design, commitment to a defense-in-depth, proper attention to accident mitigation, and operational safety culture. Accidents of relatively large frequency could be prevented through plant modifications and/or design. On the other hand, research results will be very useful in providing additional confidence that accidents of increasing consequences can be shown to be very improbable (based on acceptable probabilistic methods), and to the extent economical, should be designed to be mitigated by engineered systems.

The PSA studies have also shown that the regulatory decision-making could be piece-meal and there is a need for achieving a balance in the approach to the implementation of the existing licensing requirements in most of the Western countries. On the other hand, the same studies have also largely confirmed the adequacy of the defense in-depth safety philosophy which is certainly qualitatively probabilistic, and risk-based, even though the actual implementation of this process has been based on deterministic requirements.

If risk-informed and performance-based regulation is to take hold, there will be greater reliance on the results of plant-specific PSAs, and the use and application of PSA numerical values. This necessitates greater emphasis on the better understanding of the governing uncertainties, including phenomenological uncertainties. The most prudent approach to closure of severe accident and source term uncertainties, could be based on the principle that even though the resolution of all outstanding uncertainties may not be an achievable objective; enough knowledge should be gained through continued research so that the potential benefits and/or detriments of accident management measures, could be assessed.

C.3.3. Current Trends and Out-Look in Europe

The existing regulatory framework in most European countries has always been qualitatively, risk- and probabilistic-based, even though, by establishing the fundamental deterministic licensing criteria, it was argued that the probabilities of occurrence of severe accidents is zero or acceptably low [10].

Most of the European regulatory authorities do not use formal risk-based acceptance criteria with respect to the final numerical results of plant-specific PSAs, even though several European countries have formulated and established probabilistic safety criteria, not so much as mandatory safety limits [10], but rather as aspiratory targets supporting the overall licensing decisions. These criteria are typically not legally binding requirements. The Netherlands is by far the most progressive with respect to the adaptation of a global safety policy, where the national "acceptable risk limits" need to be respected by all potentially hazardous industries and activities, even though these risk limits do not have any legal status [10].

In most European countries, as in the United States, plant-specific risk results have been utilized by the regulators to justify regulatory actions (e.g., requirements for plant-specific or generic improvements), or by the utilities, to justify plant-specific modifications (e.g., justifications to support arguments against specific regulatory decisions and/or actions). All in all, the current approach could be viewed as "risk-informed" but not "risk-based." So long as potential changes in the plants or exemptions from regulatory requirements, do not alter the fundamental deterministic bases on which the operating licenses are granted, risk-informed arguments have been found to be generally acceptable. However, in instances where the "the fundamental deterministic bases" are to be altered, problems have surfaced. For instance, if a change in the plant technical specifications can be justified on the basis of risk arguments, it becomes difficult for the regulators to grant the modifications, as this may violate the original deterministic basis. A fundamental change in the existing regulatory basis similar to that currently underway in the United States, has to be discussed and agreed to.

The results of the severe accident research are starting to be utilized in some of the European countries. These results are generally included as part of the Severe Accident Management Guidelines (SAMGs) that are being implemented, by varying degrees by various European utilities. The regulatory authorities have also been instrumental in requiring certain plant hardware and procedural modifications that have been guided by the results of severe accident research. Examples include, cavity flooding to provide for external cooling of the lower head, external containment cooling (for long-term heat removal), in Finland, to name a few.

C.3.4. Potential Requirements on Severe Accident and Risk Assessment Research

The degree by which the regulatory decision-making process is tied to the quantitative risk information, will dictate the requirements that will be placed on the results of severe accident research.

In the early, in-vessel phase of severe accidents, the unresolved issues that require additional research deal with release, transport and chemical forms of volatile fission products. In general, even though the existing uncertainties are not fully quantifiable with the results of the available data; nevertheless, it is believed that from the standpoint of accident management actions and regulatory decision-making, it would be very difficult to expect that additional research will necessarily alter the decision-making process. On the other hand, for accidents occurring at shutdown (with the reactor head removed), or accidents involving spent fuel pools, the potential for air-ingress and additional oxidation of the cladding and revolatization of some of the otherwise non-volatile fission products (e.g., Ru), requires additional research, as there is currently no information available in the literature that could be useful for the purpose of risk studies.

In the late, in-vessel phase of severe accidents, the most significant outstanding issues that could benefit from additional research and are also relevant to regulatory decision-making, especially in evaluating the impact of SAM actions, include (See Table 1):

1. Melt/Debris Coolability in the Lower Plenum by Internal Flooding
2. Melt/Debris Coolability in the Lower Plenum by External Cooling of the Lower Head
3. Mode, Location & Size of Lower Head Failure

In addition, the current database for analysis of the release of volatiles and semi-volatile fission products inside the lower plenum is not adequate, and it is desirable to develop the additional data and models for release of fission products in a molten pool configuration. This was also recommended as part of the recent peer review of the Phebus experimental program [11]. On the other hand, this issue is not as important to the overall regulatory evaluation of SAM strategies, and/or other relevant regulatory actions, because, it is not expected that any of the perceived SAM actions could exacerbate the melt conditions that could lead to higher releases of fission products, nevertheless, it would be desirable to have a better technical basis.

In the ex-vessel phase of a severe accident, the issue of molten debris pool coolability remains unresolved. Provisions for a deep or shallow water pools under the reactor pressure vessel cannot be demonstrated to ensure long term debris coolability, following melt discharged from the reactor pressure vessel. Interaction of the melt jet may lead to very small particles (in the event of a steam explosion), which may be difficult to cool in the form of a debris bed of low porosity [12]. Incomplete fragmentation will lead to a melt layer on the concrete basemat under a particulate debris layer and a water layer.

Coolability of a melt pool interacting with a concrete basemat by an overlaying water pool has been under intense investigation in the MACE Project, sponsored by an international consortium and managed by the Electric Power Research Institute (EPRI). Three experiments using melt material containing Uranium oxide, Zirconium oxide, Zirconium and some concrete products [12] that have been performed to date that have shown that the sidewall dominated the phenomena, since an insulating crust was found to form that attached it to the sidewalls. The crust prevented intimate melt-water contact and prevented long-term debris coolability. Therefore, given the importance of achieving a stable debris configuration following a severe accident with the resultant impact on containment integrity, this issue remains one of the vexing and unresolved severe accident issues with potential impact on the manifestation of the effectiveness of the potential ex-vessel SAM actions, and regulatory resolution of severe accident issues. On the other hand, it should be noted that the effectiveness of overlying water in retaining fission product aerosols is established and DF factors of the order of 10 [13] have been reported (for typical ex-vessel pool depths); however, any additional research would only help in narrowing the range of uncertainties for the expected decontamination factors at prototypic depths, water subcooling, and gas sparing rates.

Other issues with a high importance to regulatory decisions, and where additional research could improve our technical basis include:

1. Retention of FPs in the secondary side of damaged steam generators (following a SGTR event)
2. Fuel Failure Criteria
3. Fission Product Revaporization
4. Volatization Potential of Refractory Groups

Since SGTRs have been found to be major risk contributors in most PWR risk assessments, and the current SAM strategy involves the refill of the damaged steam generators. Substantial retention of aerosols due to this added water is predicted using any of the available computer codes. There is no reason to suspect that additional experiments would not generally confirm these predictions; however, in order to buttress the current position, it is desirable to provide an adequate basis of the current understanding. Studies underway in Switzerland in an EU project are expected to contribute to our understanding of fission product retention due a water-filled steam generator.

It would be worthwhile to assess the current requirements for fuel failure criteria that could impact the more realistic prediction of design basis accidents, and radiological activity analysis.

A substantial fraction of the fission products released from the fuel are expected to be deposited on the reactor coolant system (RCS) structural surfaces during transport of fission products from the core into the containment. Some of these fission products may be chemically absorbed, while others may adhere as aerosol deposits to various surfaces. Following reactor pressure vessel failure, these fission products will continue to heat-up the various surfaces, eventually leading to conditions where vaporization of non-chemically absorbed species and/or compounds will lead to their release into the containment, and possibly into the environment. PSA studies have shown that this revaporization component has a major contribution to accident source term, especially for accidents involving late containment failure. The current modelling of fission product revaporization is not supported by an adequate database, and needs to be improved. This could also be helpful to establish additional containment protection and mitigation strategies that would circumvent the uncertainties associated with the late radiological releases.

The increased use of risk assessment in regulatory decision-making requires improvements of probabilistic safety assessment studies in a number of areas, including modelling of:

1. Organizational and safety culture aspects
2. Passive components (e.g., piping, steam generators, reactor pressure vessel, etc.)

3. Equipment aging
4. Operator actions (i.e., human reliability),
5. Software reliability and digital systems (e.g., digital control and protections-systems, and
6. State-of knowledge uncertainties.

Management direction, control and oversight plays an important role on both equipment and human reliability estimates. Quantitative inclusion of organizational and safety culture aspects are important factors leading to completeness of PSAs, and assessment of plant performance (14). Methods have been proposed)e.g., see (14) that require specific data collection and analysis, including development of alternative and improved approaches to inclusion of organizational and safety culture influences in PSAs. This research could also help in the development of approaches to risk-based or risk informed, performance indicators for assessment of performance of nuclear power plants as well as regulatory authorities (e.g., assessment of regulatory effectiveness and regulatory oversight process).

The reliability of passive components and equipment aging require additional research in order to clearly establish a more mechanistic basis to the current approach. Modeling improvements would enhance the current statistically empirical approach.

Inclusion of human reliability and operator actions with PSAs is currently based on softer and less rigid scientific modelling approaches. It is unlikely that future research in this area could eliminate the judgmental and subjective aspects of this important PSA element. However, improved methods for incorporation of plant and operator performance data, including methods for collection and application of increasingly more sophisticated simulators, would benefit PSAs. This includes development of semi-empirical models for inclusion of a broad spectrum of human errors including errors of commissions and omission.

The increased use of digital instrumentation, control and protection systems in nuclear power plants is introducing some unique reliability and risk issues. In addition, incorporation of software and digital reliability issues within the PSA modeling framework needs additional research. Experience in other technologies and industries (e.g., aircraft, aerospace, etc.) may be useful to development of any future research programs.

Use of risk insights for regulatory decision-making dictates and adequate knowledge of the range of uncertainties associated with estimated results of plant-specific PSAs. Assessment of uncertainties in most of the current PSAs is either non-existent, or focused primarily on data uncertainties. Quantification of modeling and state-of-knowledge uncertainties has received considerably less attention. However, increased use of risk results, especially, the bottom-line risk contributors demand a much better treatment of all potential uncertainties within plant-specific PSAs. A longer-term research aimed at developing mathematical techniques, and procedures for quantification and propagation of uncertainties within a PSA framework is desirable.

In addition, in anticipation of the move towards risk-based decision-making, it is important to further develop the technical basis for PSA applications by developing:

- Detailed procedural guides for full power, non-full power and external events PSAs, including issues relevant to new generation of power reactors
- Guidelines for PSA quality assurance and peer review requirements
- Graded approach to risk-informed regulatory activities, including In-Service Inspection (ISI), In-Service Testing (IST). And plant maintenance and backfit actions
- Guidelines for cost/benefit analyses associated with any proposed changes in plants either to modernize or to enhance safety (based on new insights of safety research)

This will ensure direct application of risk and severe accident research results by end-users, including regulatory organizations and power plant operators.

C.3.5. Summary and Concluding Remarks

There is a real impetus to complement the current traditional, deterministic-based approach to reactor regulations. Even though in many countries, an explicit change in the regulatory approach has not been codified, nevertheless, risk-insights and results of plant-specific PSAs are increasingly being utilized for regulatory decision-making, and to respond to request from utilities for exemptions from certain deterministic-based regulatory requirements.

This trend will pose additional requirements on the quality, and the technical foundations of the existing PSA results.

Substantial progress has been made over the last 20 years in resolving many of the important severe accident issues. In addition, generic procedures have been developed by various owners groups and are being implemented by various plants, in order to circumvent the existing severe accident uncertainties, and as a way to overcome the potential severe accident vulnerabilities. In some cases, these procedural modifications are also implemented along with hardware changes.

Therefore, the current severe accident research needs to be focused towards addressing those issues that will help increase the confidence in the viability of the various severe accident management actions.

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C.4. Questionnaire and the Evaluation of Responses to Questions

C.4.1. Questions and Responses

A Questionnaire based on the objectives of the Project was prepared and sent to the nuclear regulatory authorities or their technical support organizations in Czech Republic, France, Finland, Germany, Hungary, Japan, the Netherlands, Slovakia, Slovenia, Spain, Sweden, Switzerland, United Kingdom and the United States. All of the country representatives contacted provided responses to the questionnaire.

The following provides an overall summary of the responses provided by the respondents to each question listed in the ISARP Questionnaire.

1. What organisations are supporting you?

All the authorities are supported either by dedicated and independent institutions or industrial contractors and universities. It should be noted that, some of the respondents only provided their funding sources, instead of listing their Technical Support Organisations (TSOs).

2. Who is responsible for funding the SA research?

Most respondents indicated that SA research funding is either provided through governmental budget or shared by utilities. However, it was pointed-out by some, that ultimately, all costs fall upon the licensees. All countries commented that modest funding has been obtained through the CEC (4^{th} Framework Programme), or other CEC-sponsored programs.

3. What is the safety policy/philosophy to support decision making?

All institutions concord that safety policy is based on guidelines, i.e., a prescriptive approach, developed either independently on state-of-the-art knowledge, or based on guidelines developed in the countries of origin of the plants. The Finnish approach also requires an adequate understanding of the plant behaviour under all applicable operational conditions. In addition, it is apparent that one of the outcomes of SA research is to buttress the "defence-in-depth" that is embodied within the current Western reactor design and licensing basis. However, the overall regulatory decision-making process continues to be based on "deterministic" philosophy, which is being increasingly supported by risk insights.

4. Would you like to have the support of the SA research?

Most responding institutions favour the continuation of nuclear safety research to support safety decision-making. The focus of safety research varies amongst the various countries, due to the influence of national nuclear safety research budget, organisational and other national priorities.

5. How do you use the results of SA research?

Most of the responding institutions appear to employ the results of SA research to improve the models embodied within SA computer codes. In addition, SA research is found to be useful for the development and implementation of plant specific Severe Accident Management (SAM) strategies and guidelines, as well as instrumentation and mitigative systems (e.g., locations of recombiners in containments, filtered vent installation, etc.) for use under severe accident conditions.

6. Are you satisfied with the SA results that you have used so far?

All institutions are only partially satisfied with past results. It is worthwhile to note that some of the respondents believe that the results of past research may have actually contributed towards *increasing* phenomenological uncertainties, rather than *reducing* them.

7. How have such SA research results affected your decision making related to protection against severe accidents?

Opinions are divided amongst the responding institutions. Some believe that uncertainties in key phenomena e.g., steam explosions, in-vessel and ex-vessel melt coolability, are still large thus impeding regulatory decision making in some areas. The others contemplate applications of SA research results mainly to the development and implementation of SAM measures (pre- and post-core damage) e.g., RCS depressurisation, containment venting, hydrogen control, in-vessel melt retention, and melt spreading, etc.

8. Where and why do you see further needs of SA research?

Several respondents do not foresee the need for continued general research just to gain further academic phenomenological understanding. While most others foresee the need for continuation of research as a way to improve our understanding and to reduce overall uncertainties associated with key phenomena. Overall, there appears to be general support for continuation of some SA research in order to maintain competence and expertise in reactor safety. However, the focus of such research should be increasingly on issues related to accident management, including a better understanding and qualification of instrumentation and systems during severe accidents.

9. Which areas of SA research would you like to be investigated further and why? Could you prioritise?

All respondents follow the response to the previous question. There appears to be a general consensus on the need for additional research as related to in-vessel and ex-vessel melt coolability, in-vessel corium retention, steam explosions, and phenomena related to potential issues arising from SAM implementation.

10. What are your requirements with respect to severe accidents?

There is a significant variation amongst the various countries regarding requirements with respect to severe accidents. In most countries, SAs are considered to be beyond the design basis accident envelope; while in other countries, specific guidelines exist regarding SAs. Overall, SA risk is to be demonstrated to be low, and the vulnerabilities to severe accidents are expected to be eliminated through procedural and/or plant-specific modifications.

11. Is the focus of SA research consistent with what you deem appropriate?

Some responses were negative, either because the research carried on thus far has been too generic, or because it was pointed out through their responses, that the current SA research is unduly focused on the needs of future reactor designs, not addressing the needs of the existing operating plants. However, several respondents indicated that the closure of some of the more important SA issues has been achieved primarily based on the results of the past SA research programs; and therefore, it was judged by some of the respondents that the SA research has been appropriately focused over the last few years.

12. What is your view: should SA research focus on prevention or mitigation actions?

Prevention (especially of energetic phenomena which could challenge containment integrity) is of primary interest to most of the respondents. However, most recognise that research on mitigative phenomena (including improved operator training and instrumentation for use during SAs) is essential in order to provide the technical bases for management of potential accidents and mitigation of radiological releases.

13. What is your view of backfits in existing plants to enhance safety and on the issue of benefits/costs?

In general, all appear to agree that backfits based on risk reduction are, and have been, implemented. Some backfits have been performed on the basis of deterministic considerations, some on the basis of systems analysis (PSA level 1) only. However, even though cost-benefit arguments are useful, nonetheless, most institution agree that to some degree, cost-benefit arguments are of relatively lower importance, especially considering the prevailing public opinion.

14. Do you feel that there are sufficient SA research results to satisfy your needs?

Most respondents felt that either the existing data is not totally sufficient, or it is of limited use to regulatory applications. Instead, some of the respondents believe that SA research to date has not been responsive for application to plant specific issues, and that research in the future should concentrate more on the needs for SAMs. Some of the respondents noted that some of the accident research database has been proprietary, and the results of those programs have not been shared expeditiously amongst the nuclear safety community.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

In general, the respondents indicated that any new research program must pass a test of importance of the expected results in light of actual current and future safety needs, commensurate with any future nuclear power developments. There is a general consensus for continued research in order to maintain expertise and to enhance overall technical understandings. Additional emphasis should be placed to co-ordinate future research programs, in order to ensure their relevance to issues of regulatory and safety concern. Future research may also be needed to support upgrading of older generation of Eastern European nuclear power plants. In addition, as the safety requirements on future reactors are increasingly focused on the inclusion of severe accidents within the design basis envelope, additional research to support the qualification of severe accidents design bases for future reactors will become necessary.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

Most respondents do not foresee a change, since either risk information has already been used to support regulatory decision-making, or simply because regulatory decisions are being, and expected to be, made solely on deterministic bases. On the other hand, the move towards risk-informed decision-making may entail additional demands for reduction of SA uncertainties and thereby continued research in some areas of SA may be warranted.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

There appears to be a general consensus that at least to some extent gaining knowledge to reduce uncertainties in risk assessment is of value to regulatory decision making, provided that research is appropriately focused.

18. Some regulatory and research organisations have concluded that the following SA issues have been resolved:

- **a mode failure**
- **DCH for Westinghouse PWRs**
- **liner failure for G. E. Mark 1, BWRs**

while some of the following issues are considered unresolved:

- **vessel failure modes**
- **fuel-coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatalisation, ex-vessel release)**

What are your views on each of the above statements? Please prioritise the importance of each one of these and the needs for further research if any?

All institutions feel that alpha mode failure, DCH for Westinghouse PWRs, and liner failure for G.E. Mark I BWRs have been resolved. However, all point out that the last two are not generic issues, and can be resolved only on a plant-specific basis. The combined result of prioritisation of the other issues to be resolved appears to be that:

1. Most respondents attach the highest priority to the issues of in-vessel and ex-vessel cooling.
2. At least 4 respondents attach a relatively high priority to resolution of hydrogen issue.
3. Two respondents attach a high priority to resolution of fission product release and transport issues.
4. Most respondents include all of the listed issues where there is a need for additional reduction of uncertainties.

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development of SA codes?

All of the surveyed organisations appear to be using SA codes to support their decision making process. Most consider that SA codes should be improved or further developed, using results from experiments for validation, and thus reducing uncertainties in calculated results. None of the institutions, however, recommended development of a new large system code, since existing codes appear to be satisfactory for the purposes of analysing SAM strategies.

20. What is your current estimate of the contribution of human error to the SA Risk? Do you feel further research is necessary or appropriate to reduce this contribution?

All respondents identify human error as being a large contributor to severe accident risk in nuclear power plants. However, most point out that the probability of human error used in the analyses may be significantly overestimated, due to lack of appropriate credit to recovery actions, in most existing analyses. Furthermore, the respondents appear to support the continued research in human reliability analysis.

21. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it required to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

Some feel that since AM is yet to be completely defined or agreed upon between regulators and operators, it may be premature to indicate which outstanding issues should be addressed. Debris coolability or hydrogen control (but in some cases not both at the same time) appear to be at the top of some of the respondents lists. The issue of DF in the SGTR was only addressed by some of the respondents. The respondents were divided on the need for additional research to improve our understanding of DF values for flooded SGs to within more than one order of magnitude.

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam and aerosol concentrations?

Most respondents point to the fact that the catalytic recombine technology is already developed and tested (or should be tested) by the vendors under prototypic accident conditions; therefore, research in this area should be left to the vendor organisations and not through publicly funded programmes.

23. What should be the main focus of safety, as it related to SAs? High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain with respect to frequency of release) high consequence (i.e., large releases) accidents?

The respondents to this question differed in opinion. Some indicated that the focus should be on risk and risk alone. Others believe that the focus should be on high consequence low frequency accidents. Yet, others indicated that accident prevention should be the top priority. However, most of the respondents indicated that mitigation of large releases is an important aspect of safety considerations.

24. Do you think that the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

There is a general consensus that circumvention through SAMs is the main objective of SA research.

25. What design related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of SAs?

Some respondents list hardware provision to introduce core catchers as an important measure to reduce the risk of severe accidents. While others list hydrogen control devices, provisions for flooding as useful fixes to reduce SA risk.

C.4.2. Conclusions

Conclusions reached during discussions between the partners are described below.

- The responses from the regulatory organizations show definite attitudes which are quite different from each other, but also similar.

- Most regulatory organizations appreciate the results achieved through SA research, e.g. the resolution of many of the severe accident issues and the knowledge and understanding achieved of the severe accident phenomenology. Most regulatory organizations recognize that the "sharpening of pencil" that research work has achieved, generally, has led to reduction of the risk of the severe accidents.

- Regulatory organizations also appreciate that some of the results of SA research have been employed in devising AM measures e.g. filtered vents, depressurization, feed and bleed, hydrogen recombiners, in-vessel melt retention etc.

- Regulatory organizations which deal with a large number of plants tolerate more generic or phenomenology-oriented research in which understanding and reduction of uncertainties is emphasized.

- It appears, however, that regulatory organizations which deal with 1 to 6 plants, would like to conduct more plant-specific research, or receive research results, or methodology, which can be readily applied to resolve plant-specific issues.

- The regulatory organizations are unhappy about non-applicability of some research or analysis results which should be applied for formulation of accident management (AM) guidelines or resolution of AM issues, since they believe this to be their primary responsibility in terms of safe guarding the public from the consequences of severe accidents.

- The situation is somewhat confusing, since devising AM guidelines and decision- making e.g. on depressurization, feed-bleed, containment venting, hydrogen recombiners, measures for melt (debris) coolability etc. may require broad research results, but more importantly it requires analyses for many scenarios for specific plants, which have to be performed with codes employing models for those plants.

- The regulatory organizations want very much to have improved, validated, less uncertain and quality-assured codes for the resolution of plant-specific AM or safety issues, and for the Level 2 Probabilistic safety Analyses (PSA).

- It is also the aim of the SA research to develop realistic models which would have the same qualities as above. These models could be used to construct system codes having the attributes mentioned above.

- It appears that the regulatory organizations recognize the contributions of the SA research towards that purpose, however they think that:
  - The research aims are not as well-focussed as they could be,
  - Research results are not timely,
  - The research some times raises new questions,
  - Research coordination is not optimum.

- Another aspect, becoming clear, is that the regulatory organizations, although still asking for deterministic analyses, with realistic parameters and models, are getting used to PSA, and have developed a positive attitude towards it. This is, perhaps, mainly due to the success of level 1 PSA, which is almost universally useful in finding plant system vulnerabilities.

- The regulatory organizations want similar successes from Level-2 analyses. Level 2 analyses require SA system codes. Existing codes still have large uncertainties in their models and in the interactions of the models. Perhaps some generic conclusions could be drawn about the efficacy of accident management measures and procedures. However there is a very large user effect.

- Regulatory authorities think that some safety issues have been neglected by research organizations, e. g. SA risk during reactor shut-down and refueling operations, containment bypass.

- Some regulatory organizations would like more focussed research on fission product release since that is of primary concern to the public, and to them, for emergency planning and for developing safety goals, which some countries have established as limits of the releases to the environment. In general the regulatory organizations do not consider that there are any outstanding fission product release and transport issues for which extensive further research is needed. They would like to obtain research result on short and long term management of fission products in the containment e.g. through sprays, through p-H control, through specification of paints which reduce formation of organic iodine etc.

- Most regulatory organizations state that any SAM measures or backfits should not be evaluated only from the point of view of cost/benefit, but from the perspective of risk reduction.

- Regulatory organizations have tougher attitudes about new plants. In general, the regulatory organizations would like the designers to incorporate design features, which would circumvent the severe accident concerns. In this respect, design of improved containment, core catcher, containment vent, passive heat removal systems, melt (debris) coolability measures etc. are all encouraged. Both preventive and mitigative measures would be considered for the new designs and passive safety measures would be preferred.

C.5. Relevance of Example Level 2 PSA Results to SA Research

C.5.1. Introduction

Results from PSA-level 2 from individual plants could be used in principle for the prioritisation of research issues and the allocation of limited resources available. However, specific design features may be the cause of specific plant vulnerabilities to severe accidents, and they should be taken into account in the assessment of research issues.

Level-2 PSA studies of individual nuclear plants have the following objectives: assessment of severe accident challenges and containment response, quantification of containment failure likelihood and risks of radioactive releases to the environment. Also, level 2 PSA must assess the uncertainties impacting the containment response and radiological releases.

An integrated full scope system code is generally used for the quantification of severe accident progression, containment loads and source term.

USNRC sponsored NUREG 1150 (1) presents summary results of PSA level 2 performed in the late eighties for 5 commercial plants in the US. It was already recognised in this document that PSA-based information, because of its integrated nature and discussion of uncertainties, could be used to guide and focus activities designed to improve the state of knowledge.

According to NUREG 1150, the importance of a given research issue can be evaluated in terms of the number of plants affected, the risks impact of each plant, the effect of modifications in reducing the risk, and the effect of additional knowledge on improving the prediction of plant risk or on defining or reducing the associated uncertainties.

C.5.2. Summary of Level-2 PSA Results

Level 2 PSA results for two specific plants will be presented in this chapter:

- BWR of 1000 Mwe, GE design with Mark III containment
- PWR of 1000 Mwe, Westinghouse design with large, dry containment.

Results presented here are specific and they are not necessarily applicable to other plants in Europe. The end products of PSA level 2 studies are the probabilities of containment failure for each of the identified containment failure modes, and also the frequency of each "release class", in terms of events per reactor-year. A measure of the risk posed by individual plants is the risk of release of activity to the environment in the vicinity of the plant, for each release class. The risk of release of activity is defined as the product of the release class frequency (events/year) times activity (Bq) released. Activity released is calculated in PSA level 2 by means of consequence codes.

Risk of released activity includes the aerosols and noble gases, but aerosols are major contributors to radiological risk and they are a better indication of the risk posed by each failure mode. Here, the risk from aerosol release will be presented for each release class.

C5.2.1. General Electric BWR with Mark III containment

The following containment failure modes were considered. In every case, the effect of possible SAM actions was not considered.

In-vessel steam explosion

PSA shows that the probability of containment failure due to in-vessel steam explosion is very low and it is not considered any further. This is due to the effect of the lower plenum structures typical of BWR designs, which reduce the fraction of the core which would mix with water and cause the explosion.

Ex-vessel steam explosion

PSA considers the loss of integrity of the drywell and containment penetrations, as a result of an ex-vessel steam explosion that can damage the RPV pedestal and impose mechanical stresses on the pipes connected to the reactor. However, mitigation of the damage due to the existence of intervening structures is considered in the analysis. Conditional failure probabilities for the drywell and containment are calculated to be 0.06 and 0.01 respectively.

Containment Pressurisation loads

In the time frame after RPV breach, loads due to the flashing of water inventory are considered, but this load is not expected to be great, unless the RPV remains at high pressure, due to the failure of the Automatic Depressurisation System. In the late time frame after RPV breach, the release of gases due to MCCI will continue to pressurise the containment, and analyses show that over-pressure cannot be stopped with available systems, and late venting will be necessary. For high energy anticipated transients without scram (ATWS) sequences, containment failure is dominated by suppression pool saturation and boil off, leading to possible loss of the containment function prior to core damage.

Direct Containment Heating (DCH)

Loads due to DCH after vessel breach is also considered, but it is assumed that debris will be trapped in the pedestal and drywell areas, and cannot produce containment failure, due to DCH. The reason for this assumption is that the debris entering the suppression pool is quenched in this area. However, there is a lack of models of the mitigating effects of the suppression pool in the available codes (i. e. CONTAIN) which can be used to validate this assumption.

Molten Corium Concrete Interactions (MCCI)

Containment failure due to axial and radial erosion of the RPV pedestal, due to molten debris, is also considered. The analysis shows that the pedestal region is flooded with water in all of the scenarios considered, but it is assumed that debris will not be coolable. Calculations predict axial melt-through after 4.3 days and complete radial melt-through after 3.4 days.

Hydrogen Combustion

The drywell is not threatened with failure due to H_2 combustion, since little or no H_2 reaches the drywell during the in-vessel phase, and H_2 is inerted by steam almost immediately after vessel breach. The impact of H_2 combustion on containment failure is only an issue for the small set of cases dominated by station black out (SBO) accidents where the Hydrogen Ignition System (HIS) has failed. The operation of the HIS is not challenging to the containment or drywell integrity, and late build up of H_2 to challenging levels can be prevented by containment venting.

Containment failure mode/Release class	**Frequency (Events per reactor-year)**	**Risk of aerosol radio-activity released* (Bq/reactor-year)**	**Percent of total risk of release %**
Very Early failure, prior to core damage, due to	9.17 E-7	5.9 E12	71.0

suppression pool failure caused by ATWS			
Early failure, at the time of vessel breach, by DCH or ex-vessel steam explosion.	2.3 E-9	5.7 E9	0.07
Early failure around the time of vessel breach due to H2 combustion.	2.8 E-9	7.0 E9	0.08
Late failure due to pedestal melthrough, suppression pool bypass and penetration failure	3.4 E-9	3.7 E9	0.04
Late failure due to H2 combustion (major contributor from SBO accidents)	3.0 E-9	3.7 E9	0.04
Late containment venting. Sup. Pool not bypassed	7.6 E-7	1.2 E11	1.44
Isolation failure due to SBO	1.7 E-7	1.5 E12	18.0
Bypass due to interfacing system LOCA outside containment	4.7 E-8	7.2 E11	8.67
Intact containment, 48 hours after core damage	1.0 E-6	4.1 E8	0
Total risk		8.3 E12	

* Risk of aerosol radioactivity released/reactor year = events/reactor year x aerosol radio-released from containment for the particular event (scenario)

More than half of the release of aerosols is represented by accidents involving failed containment prior to core damage. The second largest contributors to risk are the SBO accidents with non-isolated containment and the containment bypass sequences. All other failure modes are found to be much less important in terms of radioactivity released.

C.5.2.2. Westinghouse PWR with large dry containment

The following containment failure modes were considered in the evaluation. As before, no SAM actions were considered.

In-vessel steam explosion

PSA takes into account the probability of containment failure due to a missile generated by in-vessel steam explosion. The probabilistic approach takes into account the processes from the time of core relocation up to the time of missile impact with the containment wall. Parameters which are subject to quantification of uncertainty are: fraction of core relocating to the lower plenum, melt thermal energy, energy conversion ratio, dissipation of slug energy by the in-vessel structures before they fail. Even with conservative initial conditions, the calculated conditional probability of containment failure is below 1.0E-3.

Ex-vessel steam explosions

If there is water in the cavity at the time of vessel breach, there is a possibility of explosive ex-vessel melt-coolant interaction, which would be confined to the cavity and would not threaten the containment. However there could be an indirect mode of containment failure due to shaking of the reactor vessel, piping and steam generators following cavity wall damage. This indirect mode of containment failure is caused by the failure of the containment penetrations. The impulse loads on the cavity wall can exceed the wall fragility, especially for cases when the lower head fails at a side location. However, since the piping and the steam generators are seismically qualified, the shaking failure of the steam generator supports is extremely unlikely, and the probability of containment failure is extremely low. The existence of water in the cavity at the time of vessel breach is dependent on the design of the cavity and the strategy followed for severe accident management. In the case of dry cavities, water will be present only if a decision is taken to flood the cavity prior to vessel breach.

RCS Blowdown and Generation of Non-condensable gases

Early-phase containment loads are dominated by steam, while in the late time frame the pressurisation loads include non-condensable gases produced during MCCI. However, pressurisation due to steam could become more significant if water is made available for ex-vessel debris cooling. The operation of the containment heat removal system reduces the pressurisation loads significantly. Spray can effectively condense steam, and fan coolers are able to reduce pressure loads imposed by non-condensable gases. The highest contribution to the probability of containment failure comes from a small LOCA not involving the operation of containment heat removal systems.

H_2 and CO Combustion

There are considerable uncertainties associated with in-vessel Zr oxidation and generated H2. Variations between 20 – 40 % of initial core mass are noted, the lower figures correspond to low-pressure accidents. Ex-vessel H_2 generation accounts for the remainder of Zr inventory, plus the contribution from oxidation of steel. As a best estimate, in the late phase of severe accident, close to 100% of the Zr is assumed to be oxidised and, in the absence of steam, it is almost certain that a H_2 combustion event will occur if an ignition source is available. In addition, large quantities of CO are produced from MCCI. However, calculations with simple combustion models show that the probability of containment failure is zero, both in the early and late phases of the accident. There is a small conditional probability of early containment failure due to core reflood (0.002), and of late failure due to starting the containment heat removal system (0.006).

The containment has a considerable height, and this produces significant entrainment of gases and mixing with the H2 plume is expected, and the concentration of H2 in the dome is not expected to be much different. Therefore the risk of failure due to local combustion events is very low.

DCH

The analysis of high pressure melt ejection (HPME)-induced DCH, based on the work carried out in the US for Westinghouse plants, show that the conditional probability of containment failure due to DCH is almost zero. This result is obtained even without the consideration of natural circulation-induced creep rupture that would lead to RCS depressurisation prior to RPV breach.

Vessel rocketing

Following failure of the vessel head at high pressure, the debris and gases in the lower plenum can escape through the hole at the bottom of the RPV, and the vessel can be accelerated upwards. If the thrust forces exceed the restraining forces, there is a potential of the vessel impacting the containment boundary. The factors that impact the energy release are the vessel pressure at vessel breach and the size of the lower head failure. The analysis show that the probability of containment failure due to vessel thrust forces is almost zero.

Liner attack

Debris dispersed from the RPV at high pressure can be transported to the lower compartment of the containment through the instrument tunnel. Thus, there is a possibility of direct contact between the dispersed debris and the liner in the annular compartment. Heat up and melting of the liner is possible without the mitigating effect of any water adjacent to the liner. The principal uncertainties are the floor area of concrete over which the debris can be assumed to spread, and the heat transfer coefficients. Liner failure would not necessarily involve large releases, since the containment structure would be present as a barrier.

Basemat melt-through

For PSA, it is assumed that cooling of the molten debris by an overlying pool of water remains uncertain and therefore it is difficult to assign a higher than 0.5 likelihood for debris coolability. Analysis of concrete ablation with available codes show that melt-through of the basemat can be expected between 4 to 5 days. Therefore, melt-through is very unlikely within 48 hours after core damage, which is the period normally covered by PSA. Melt-through before 48 h is assigned a conditional probability of 0.01.

Containment Failure Mode (Release Class)	**Frequency (events per reactor-year)**	**Risk of release of aerosol radioactivity Bq/reactor-year**	**Percent of total risk of release, %**
Containment bypass events: SGTR accidents initiated or temperature induced, containment isolation failure	2.39 E-6	9.92 E12	72.9
Rupture at or before vessel breach	4.04 E-8	2.3 E11	1.69
Liner failure	5.68 E-8	2.09 E10	0.15
Late rupture (within 48 hours from core damage)	10.96 E-6	3.45 E12	25.36
Basemat penetration (within 48 hours from core damage)	4.74 E-7	1.04 E9	0.00

Containment intact 48 hours after core damage	5.1 E-5	8.2 E10	0.60
Total risk		1.36 E13	

From the point of view of risk the bypass category contributes to 73 % of the total risk of activity released, due to the high frequency of containment bypass events, and the large release of associated radioactivity. Although the activity released in early containment failure events is large, the risk of activity released to the environment corresponds to only 1.69 % of the total radioactivity released risk, due to the low frequency of early containment failure events. The liner failure category is a negligible contributor. The late failure category is a substantial contributor (25 %) to the total risk, especially for sequences without the operation of containment sprays. This is due to the fact that for sequences corresponding to 66% of the core damage frequency, containment sprays are not available to scrub the releases, and the cavity is dry for sequences corresponding to 65 % of the core damage frequency.

C.5.2.3. Impact of accident management on risk for PWR plants

The impact of successful SAM strategies on the calculated frequency and risk of release can be evaluated. An example of evaluation will be presented here. For each SAM strategy considered in the analysis, information is given about the impact on risk reduction for each release category.

Isolation of failed SG before in-vessel releases occur can reduce the risk of activity for bypass sequences by 63 %. Flooding of SG secondary, prior to significant in-vessel release, with a constant decontamination factor of 100 assumed for all releases, reduces also significantly reduces (by 62 %) the risk associated to bypass sequences. The reduction of overall risk of each SAM action is about 40 %.

Recovery of water injection before substantial core relocation occurs, has been considered in the PSA, but there is considerable uncertainty associated to degraded core coolability. A conditional probability of 0.5 was allocated to degraded core coolability by reflood after recovery of AC power. One of the detrimental effects is increased H_2 production, which has been calculated to increase the conditional probability of early containment failure, and to decrease significantly the risk associated with late containment failure. The overall impact on total risk is about 10 %.

Flooding of the cavity before vessel breach, with the spray systems has been considered, assuming that the action is unconditionally successful in cooling the core in-vessel. The risk of radioactivity release from late failures decreases by a factor of 33, and the risk of radioactivity release from early failure decreases by a factor of 3. The total risk decreases by 3.6 %. However, if it is assumed that flooding is independent of the availability of AC power, the overall risk is reduced by 25 %.

Cavity flooding after vessel breach reduces the risk of radioactivity release for the late containment failure category by a factor of 10, due to the increased probability of core debris coolability and reduced generation of non-condensable gases. The impact on overall risk is substantial, with a reduction of 23%.

The impact of RCS depressurisation after core damage reduces the risk of radioactivity release from early containment failure and liner failure to zero. However the overall impact on risk is negligible, due to the very low contribution of these failure modes to total risk.

C.5.3. Relevance to SA Research

For this specific PWR (Westinghouse), the research issues which impact the overall risk significantly are related to containment bypass accidents, followed by issues related to late containment failure. In particular, issues of retention and re-vaporisation in the RCS, failed SG, melt attack and ex-vessel coolability are important. Also in relation to SAM, efficacy of SG secondary flooding to reduce releases, of reactor cavity flooding to prevent vessel rupture, and of reactor cavity flooding to cool debris discharged from the vessel, are important for reduction of risk of releases.

For this specific BWR (GE MARK-III), research issues which impact the overall risk significantly are related to accidents in which containment isolation fails and the full power ATWS sequences. Issues related to the need of late containment vent, i. e. generation of non-condensable gases during MCCI have some impact. Issues related to early containment failure, such as DCH, steam explosion or H_2 combustion do not have a large impact since the probability of early containment failure is very small. Issues related to late containment failure by RPV pedestal melt-through due to MCCI, or H_2 combustion also not have a large impact on overall risk.

SAM action	**Release Category**	**Risk of radio-activity release, considering SAM** Bq/reactor-year
Isolation of faulted SG	Containment bypass events: SGTR accidents initiated or temperature induced, containment isolation failure	3.70 E12
Flooding the steam generator secondary side	Containment bypass events: SGTR accidents initiated or temperature induced, containment isolation failure	3.76 E12
Recovery of sprays before vessel breach	Late rupture (within 48 hours from core damage)	1.50 E12
Intentional RCS depressurisation after core damage		3.1 E3
Cavity flooding after vessel breach		3.6 E11
In-vessel recovery by core flooding before vessel		4.57 E8

breach		

C.6. State of Resolution of Severe Accident Issues with respect to Regulatory Needs (concerns)

C.6.1 . Objectives

The aim of this task is to evaluate the state of resolution of the severe accident issues with respect to regulatory needs (concerns). Research results from the Fourth Framework Programme of European Union have been employed. Other sources of results are the national programmes developed in countries of European Union or abroad. Some of these results are available through international agreements because some of these programmes were conducted under multilateral collaboration or through international organisations such as OECD (e.g., the RASPLAV programme).

In the following, different topics relevant to the severe accidents are discussed.

C.6.2. Issues

<u>C.6.2.1. In-Vessel Core Degradation Progression</u>

The uncertainties are on the accident scenario (corium progression) and on some physical phenomena.

As uncertainties on the scenario of the accident, it may be noted :

- the relocation of the corium in the vessel (e.g., side or bottom discharge, metal content of jet)
- physical state of the corium in the vessel bottom head (melt pool, debris bed)
- the timing of vessel failure

Several uncertainties remain also on physical phenomena :

- physico-chemical properties of the corium (composition, chemical reactions, viscosity...),
- structure heat up and structure-corium interaction,
- 3 D behaviour of the molten corium and of the cooling,
- water progression in the debris bed and between vessel bottom head and corium.

Taking into account the actual knowledge, it seems necessary that the studies cope on the one hand with the different phases of relocation of the molten corium from the core to the vessel bottom head and on the other hand with the form of the corium in the vessel bottom head. Emphasis has to be placed also on composition and physico-chemical properties of the corium.

The conditions of the failure of the vessel are strongly related to the characteristics of the corium: amount, composition, physico-chemical properties.

As actions of the 4th Framework Programme, the Projects COBE (investigation of core degradation) and CIT (corium interaction and thermochemistry) can be cited.

Experiments will also bring additional knowledge as, for example, MADRAGUE (IPSN, France), PHEBUS (IPSN, France in collaboration with EU and the most important nuclear countries), CORA and QUENCH(FZK, Germany), RASPLAV and MASCA (KI, Russia under the auspices of OECD).

At the moment, numerous results exist or will be available in the short term and they have to be interpreted and reconciled. Also sufficient facilities are available for conducting experimental research. No new facilities are needed.

C.6.2.2. Retention of the Corium in the Vessel Bottom Head and Ex-Vessel Cooling

Retaining the molten core material in the vessel can be an objective for the severe accidents management even if, up to now, the knowledge is insufficient to assure that the vessel bottom head would not fail. To reach this aim it is necessary, if putting water inside the vessel is not possible, to have an ex-vessel cooling.

The possibility of flooding is plant specific because it depends on the reactor pit geometry, the vessel bottom head form and the possible supply of water. It could be studied in order to cope with an optimal solution for severe accident management and new reactor concept.

Beside the design related aspects of the external cooling of the vessel, the physical phenomena concern essentially the heat exchange coefficients, between water and vessel lower head bottom. Some experiments have been made about this question with the projects SULTAN and CLIAU (CEA, France), CYBL (SNL, USA) and ULPU. Some results are available also from the 4th FP action IVCRS (In Vessel Core Retention Strategy).

Concerning the possibility of cooling of a debris bed inside the vessel bottom head, large uncertainties remain. Some experiments are under development, but the uncertainties about the ingression of water inside the debris bed and of gap-cooling are so large that it seems difficult that these phenomena could be taken into account in the frame of the safety analysis.

An issue of some concern is the chemical interaction of the melt/debris with the wall of the vessel at the prevailing temperatures during melt pool convection. Recent data from RASPLAV indicates mild or very little interaction.

C.6.2.3. Rupture of the Vessel

The objective for the studies related to the rupture of the vessel is to know when and how this rupture could occur during a severe accident. This problem is related to the in-vessel cooling of the corium and the failure of the vessel at high pressure.

The failure modes of the vessel can be creep of the steel, corium jet and interaction between steel and corium.

There are uncertainties about the mode of failure. Some results will be available through the 4^{th} FP actions MVI (Core Melt Pressure Vessel Interactions During a LWR Severe Accident), RPVSA (On the Prediction of the Reactor Vessel Integrity under Severe Accident Loading) and REVISA (Reactor Vessel Integrity in Severe Accidents). The LHF programme first developed by the USNRC at the Sandia National Laboratory and now continuing in the frame of an OECD programme will bring some results about the vessel failure. The FOREVER programme conducted under the 4^{th} and 5^{th} FP programme will provide information on vessel creep failure.

The mechanical behaviour of the vessel is investigated in several experiments and the information derived may be sufficient for regulatory purposes.

C.6.2.4. Direct Containment Heating

The direct containment heating is a problem strongly related to the specific geometry of the reactor. The corium dispersion is related to the reactor pit design and ways to communicate between the reactor pit and other part of the containment.

Studies and experiments have been made about several designs and it seems now that there is no need for additional action for the actual reactors.

This issue has been closed by the NRC for the US plants. For the European safety authorities either this issue is closed or the tests under development will provide sufficient knowledge for the actual and future reactors.

C.6.2.5. Steam Explosion

The phases of a steam explosion are successively :

- the pre-mixing phase : the molten corium is mixed with the coolant. The characteristic time of this phenomenon is about one second and the size of the debris is the order of few millimeters,
- the triggering phase and fine fragmentation of the corium with a characteristic time of around one millisecond,
- the propagation of the explosion to all the material leading to the production of a high amount of steam.

In the domain of the steam explosion, uncertainties remain because the studies have been made with simplified computer codes whose qualification is based on a relatively limited number of experiments.

The main uncertainties affect particularly :

- the fraction of the total amount of corium fragmented in the water,
- the trigger,
- the efficiency of the steam explosion,
- the probability of creation of a missile,
- the consequences of a missile impact on the structures and on the walls of the containment.

It has been observed that the efficiency of the steam explosion depends on the material. For example, thermite creates very efficient steam explosion, whereas corium containing UO_2 leads to very low efficiency steam explosion, even with a strong trigger, as the last FARO test has demonstrated in July 1999. This behaviour has been observed, but the phenomenology has to be further demonstrated.

To take this difference of behavior into account in the safety analysis, it is necessary to well understand the physical phenomenon and demonstrate that it is not possible to obtain an explosion as such efficient with corium, whatever the composition, as in the case of other materials like thermite.

Several experiments are under way in order to study explosion and to qualify the modelling of the severe accident codes. One can cite the MFCI action of the 4^{th} Framework Programme and tests like PREMIX (FZK, Germany), ALPHA (JAERI, Japan), MSWI (RIT, Sweden), TREPAM and MICRONIS (IPSN, France).

A special mention must be made about the JRC programmes FARO and KROTOS. Both these experiments have up to now brought numerous results about steam explosions. FARO particularly was the largest test facility in the world to perform fuel coolant interaction experiments with real materials. Unfortunately it has been closed. No other installation is able now to make such large scale tests. Some results could be obtained with KROTOS but at a smaller scale. This facility has been moved to CEA-Cadarache.

In conclusion it seems that there are sufficient experiments for separate effect test facilities. The question remains open for the effect of material composition and properties on steam explosion and for the effects of scale, both geometry and material.

C.6.2.6. Hydrogen Explosion, Mitigation

The physical phenomena to be taken into account for the hydrogen issue are :

- magnitude and release rate to the containment,

- distribution in the compartments of the containment,
- combustion : deflagration, deflagration to detonation transition (DDT), detonation,
- loads on the structures.

Uncertainties remain on the amount of oxidation of the metals inside the vessel. The actual state of the art do not allow to define an upper limit of the oxidation of the zircaloy which may be considerably less than 100 % of this material present in the vessel.

One important topic for the analysis is the production of hydrogen in case of late reflooding. Some experiments are under way, i.e., QUENCH at FZK, to study the production of hydrogen during quenching of the fuel. This phenomenon is also observed during the PHEBUS tests.

In the domain of the distribution of hydrogen, the uncertainties, are concerning the concentration in the compartments, which depends on the release rate from the primary circuit, the mixing of the gases, the influence of spray and the effect of the mitigation devices i.e., recombiners or igniters.

Numerous tests have been performed studying the mixing and the distribution of hydrogen e.g. tests in Germany, or NUPEC tests in Japan. Other are planned in order to study the mixing behaviour of steam, air and helium (representing hydrogen), particularly the condensation phenomena including the effect of spray. One can cite TOSQAN and MISTRA in France or PANDA in Switzerland.

All these tests will contribute to the knowledge of the physical phenomena and to the validation of the codes.

The combustion of the hydrogen includes the phenomena of deflagration and detonation and the transition from deflagration to detonation.

The knowledge-base on the deflagration process appears to be adequate. The flammability limits depend on several parameters (gas proportions, pressure, temperature, turbulence). Fast deflagration could damage the structures and the containment.

The knowledge-base on the detonation phenomena for hydrogen mixtures also appears to be adequate, i.e., the conditions necessary for a detonation to develop are reasonably known. What is not known is whether conditions in the containment will exist which will supply the energy to initiate a detonation event for a particular hydrogen-air-steam concentration.

The exact conditions under which the deflagration-detonation transition occurs are, up to now, not well understood. Consequently, no complete modelling, of DDT exists. However, limiting conditions for DDT are being formulated based on data obtained recently.

In terms of experiments, many tests have been made under various conditions and others are being performed at the RUT facility on hydrogen combustion, sponsored by IPSN and FZK and performed by the Russian Kurtchatov Institute. This very large scale experiment allows to study phenomena in a more representative size even if the volume of the compartments inside reactor containment is still larger. It is hoped that on completion of the experimental programmes under way, adequate knowledge-base would be obtained. The modelling of the DDT combustion phenomena, however, may not be completed.

In the domain of the mitigation of the hydrogen during a severe accident, many experiments have been made about recombiners, like the KALI and H2PAR experiments in France. It is not necessary to develop new experiments.

The measurement of the concentration of hydrogen during an accident remains an unresolved issue. Some instruments have been developed, but none of them is entirely suitable. Some research is necessary in this domain, but it should be the responsibility of the plant operator.

C.6.2.7. Corium Spreading, Molten Corium Concrete Interaction (MCCI)

The phenomena related to the behaviour of the corium out of the vessel are :

- drop of the corium in the reactor pit with or without water and possibility of a steam explosion,
- spreading of the corium is the containment with crust formation and physico-chemical phenomena,
- interaction between corium and concrete and release of gases and aerosols, particularly long term interaction,
- basemat melt-through

The pertinence of the calculation of behaviour of the corium outside the vessel, as inside it, is strongly related to the knowledge of the physico-chemical properties of the corium which is a complex mixture of core components.

The remaining uncertainties on the scenario of the accident are :

- the state of the corium in the reactor pit : molten bath, debris bed,
- the risk of steam explosion, as examined above in paragraph 2.5,
- the time at which the basemat melt through could occur.

Here are uncertainties in the physical phenomena e.g. in:

- corium properties,
- cooling of the corium by injection of water inside the reactor pit,
- spreading and formation of a debris bed, consequences on the integrity of the containment and the basemat, melt through.

For the future reactors, use of a core catcher is envisaged, in order to avoid the corium concrete interaction and the possibility of the basemat, melt through.

The research related to this new system should be the responsibility of the vendors.

Considerable efforts have been spent in the 4th framework program of EU on melt spreading research. This has completed some of the programs started earlier in national programs e.g. KATS in Germany and CORINE in France. Large scale (~2 tonnes) experiments, employing nearly prototypical melt were performed by Simplekamp and at lower scale at the FARO and the VULCANO facilities. Simulant material experiments were performed at Royal Institute of Technology (RIT) in Sweden. A number of codes were developed and analysed the experiments. A predictive scaling methodology was developed at RIT in Sweden.

It appears that sufficient knowledge-base has been accumulated for adequate predictions of melt spreading in one-dimensional spreading areas. However, it is not possible so far to predict the dynamics of spreading. It has been found that the spreading configuration can assure very complicated geometry e.g. long fingers in different directions.

Regarding molten corium concrete interactions (MCCI) in a dry cavity, several experimental programs e.g. BETA, SURE, ALPHA, ACE has provided a data base using thermite (Al_2O_3-Fe) and UO_2-ZrO_2-Zr melts. All of these data are from one-dimensional experiments. The CORCON and WECHSEL codes developed, respectively in USA and Germany have been employed for productions of concrete erosion in seven accident scenarios.

There are, however, no data available for 2-D-concrete ablation except for that available from the test Mo, the MACE scoping test. Thus the 2-D basemat melt through predictions for the prototypic scenarios are lacking the validation needed.

C.6.2.8. In-Vessel and Ex-Vessel Melt Coolability

In-vessel coolability has been addressed in the FARO program, however, only for the time period of the initial relocation of corium to the lower head. For the long term when the particulate/melt debris bed created in the vessel dries out and reheats, there is the possibility of quenching the bed with water addition. Experiments are underway at RIT, Sweden on quenching of dry particulate beds. Here are also investigations on-going on coolability of the vessel wall by gap cooling. The gap cooling, so far, has not been as effective as needed.

Ex-vessel melt coolability is perhaps the most important issue needing data and models for resolution, since melt coolability is essential for the stabilization and termination of the severe accident. The MACE integral experiments employing prototypic melt composition at large scale (120x120 cm, 2 tonnes of melt) has shown partial coolability of the melt layer and has not been able to stop the continued ablation of the concrete basemat. The MACE program is being directed to separate effect tests, in order to develop a model for the coolability process with a water overlayer.

In contrast to the MACE experiments the COMET experiments have employed water addition to the melt from the bottom of the melt pool. These experiments have employed thermite melt and have shown that sufficient porosity is created in the melt pool to obtain coolability and quenchability in relatively short time.

C.6.2.9. Fission Products Release Transport and Chemistry

The amount of FP release to the environment must be as low as possible in quantity and in half-life in order to respect the requirements of the Safety Authorities in case of severe accident.

The phenomena involved in the study of the FP release transport and chemistry are complex and there remain many uncertainties, mainly because all parameters are not sufficiently known, in particular the thermal hydraulic conditions, the scaling effects, the chemical reactions and the scenarios.

For the release of FP from the core, there is substantial agreement between experts about the most volatile materials. There remain uncertainties about for the transuranium elements and the materials of the control rods and of the structures. For example, the results of the test calculations for the FPT4 test, made by the different partners of the PHEBUS programme, are in a range of 2 to 3 decades for the transuranium elements. The VERCORS programme which includes separate effects tests at temperature up to the melting of UO_2 pellets and the integral tests of the PHEBUS programme should provide data for validation of models for such releases.

For the retention of the fission products in the pipes, the amount deposited on the tube wall are generally limited in comparison to the total FP rate in the tube and of the inventory in the containment. The experimental programmes have provided sufficient knowledge-base conducted for adequate predictions.

The knowledge about the release of the less volatile fission products from the corium concrete interaction is actually sufficient for the short terms. The amount of FP released by this mean is relatively limited.

The resuspension of the fission products deposited in the primary system piping has been found to be negligible. The revolatilization above of the deposited fission products, on the other hand, could be occuring when the containment integrity is in jeopardy and thus could be important.

The knowledge-base on the deposition of the aerosols in the containment is quite sufficient. The uncertainties as the results of the FPTO and FPT1 of the PHEBUS programme are mainly related to the thermohydraulic calculations inside the containment. Thus, the publication of the thermal hydraulic conditions insdie the containment are the source of uncertainty.

The behaviour of iodine is actually not well known. The chemical phenomena are very complex because the affinity of iodine for many bodies is great. Studies must be done on the chemical forms of iodine, on the radiolysis, on the influence of pH and temperature of the sump, the behaviour with organic materials like paints and the interaction of silver coming from the degradation of the control rods.

In conclusion, the behaviour of the iodine, is not known sufficiently. This is an important problem because iodine is the most significant radionuclide for the short term release. Many separate effect tests have been performed up to now and other programmes are under completion, for example at IPSN/France the facilities EPICURE and CAIMAN are investigating chemical behaviour and radiolysis.

The PHEBUS programme will measure the global test of the behaviour of iodine from the core to the containment including the primary circuit, the walls and the sump.

For the release of the transuranium elements, the separate effect facility VERCORS and the PHEBUS programme should provide data.

Up to now programmes on the fission products behaviour until all conclusions to be drawn from the ongoing programme have yet to be realised. It appears though that additional experiments have to be performed in order to determine the chemical behaviour of some elements, particularly of the iodine.

C.6.2.10. Instrumentation and Diagnostics

Additional instrumentation is needed to know the situation inside the reactor during a severe accident and to provide information to apply the management procedures.

The main areas in which instrumentation is required:

- to measure the proportion of hydrogen in the containment and particularly in some compartments where it would be possible to get deflagration or DDT or detonation,

- to measure inside the containment the level of radio activity and the composition of the atmosphere during the accident,

- to know the beginning of MCCI in order to have a correct view of the accident progression.

The objective of the research related to the instrumentation is to design devices and not to study physical phenomena. Of particular importance is the survivability of the instrumentation during and its qualification. Thus, the instrument to be designed for has to provide for the conditions that may be experienced during the severe accident and experiments would have to be performed to validate the measurements as recorded.

C.6.2.11. Human Factor

Human factors are important elements during a severe accident. Their contribution to the risk is clearly demonstrated in the Probabilistic Safety Assessment.

It demonstrates itself in the form of error of the operator or unexpected delay for the realisation of an action of mitigation.

The human factors include many other aspects e.g., the level of automation of the procedure: what is the most efficient level of automation, and the wording of the procedure sheets must be clear and easy to understand and to apply by the operators under a great deal of stress during the accident.

Some research has been made in this area, but the uncertainties in the specification of the human factor scenario are very large.

C.7. Regulatory Use of Severe Accident Research

Regulatory authorities are customers for severe accident research. They have monitored and sponsored severe accident research. Over the years many regulatory positions (concerns) have been addressed by the results obtained by severe accident research. The research results lead to backfits and accident management actions and procedures which have enhanced the safety of the plants, or provided the rationale for the deliberate decisions of not requiring any backfits or SAM measures. A representative list is provided below:

- hydrogen control with ignitors and catalytic recombiners
- improved safety valves on PWRs
- no inerting of Mark-3 BWRs
- water addition to the Mark-1 drywell to prevent liner failure
- vessel depressurization for DCH protection
- no backfits for protection against alpha mode failure
- use of BWR suppression (condensation) pools for FP decontamination
- hard vents for BWRs from the suppression pool
- flooding of PWR vessel cavity for Westinghouse PWRs
- flooding of drywell for Swedish BWRs
- additional water delivery sources for accident termination
- reinforcement of containment penetrations
- realistic ex-vessel source term specification
- pressurized thermal shock prevention procedure
- filtered venting
- long term management of Iodine in the containment
- bottom coolant injection to ensure ex-vessel melt coolability for future plants
- in-vessel melt retention strategy through ex-vessel cooling for AP-600 and Loviisa
- melt spreading in a special compartment as an accident management scheme for EPR
- water delivery through downcomers in the cavity as possible backfits for enhancing debris coolability.

Presently, the regulatory authorities have focussed the amelioration of their safety concerns regarding severe accidents through the management guidelines for such accidents.

C.8. Remaining Issues and Concerns

What are the remaining issues or the remaining areas of uncertainty? We will address this in the following paragraphs and in this we will include the accident management measures, which are now the back bone of the severe accident safety philosophy for the existing plants.

C.8.1. In-vessel Core Melt Progression and In-Vessel Melt Retention (IVMR)

The current understanding of the early phase of core melt progression is more than adequate. For the late phase of core melt progression, there is insufficient understanding of the transient effects in the scenario. Those effects can be bounded, however, it may be possible to derive greater margins on the in-vessel melt retention strategy if greater knowledge-base (including modelling) is gained on the following aspects during core relocation from the original core boundary to the lower head:

- the role of molten metal (steel, Zr) in the transient process
- the history of the development of metal layer in the lower head melt pose

In terms of accident management actions, greater definition of the effects of water addition to a damaged core in its original configuration is needed. In addition, the effects of addition of water to the vessel when a melt pool is covered with a metal layer in the lower head have not been investigated so far. Questions exist about the possibility of a stratified steam explosion.

Other questions regarding the retention of melt in the vessel are concerned with the observations in the RASPLAV experiments where melt pool stratification has been observed. Such stratification will increase the thermal loading on the vessel wall and possibly reduce the focussing effect of the metal layer. Experiments should be performed to assess the magnitude of the effects of stratification.

It appears that the IVMR strategy will not be successful for a high power (>1000 MWe) plant because of the focussing effect on the vessel wall. This effect can be decreased substantially if the metal layer is cooled by water at its upper surface. Integral experiments comprising of a pool with a metal layer whose upper boundary condition is varied from adiabatic to isothermal should be performed to determine the magnitude of the reduction of the focussing effect due to the cooling at the upper surface of the metal layer.

The composition of the metal layer, which may have several percent of oxides in its composition, determines its liquidus and solidus temperature. The radiative heat flux from the upper surface of the metal layer depends on the fourth power of the temperature of the crust that will be formed on the upper surface of the metal layer. Thus it is necessary to perform experiments with prototypic materials and determine the composition of the "metal" layer.

C.8.2. Melt-Water Interactions

In-vessel melt-water interactions do not pose a danger to the integrity of containment. They, however, could cause an early failure of the vessel. This, by itself, may not be an unexpected event, however, accident recovery (management) may become difficult.

The ex-vessel steam explosion can not be ruled out for plants which establish a water pool under the vessel (either in the BWR drywell or in the PWR cavity). In particular, the steam voiding which inhibits steam explosion is absent for the ex-vessel melt-water interaction and the pressure is low. Research on limiting mechanisms that may be active in the ex-vessel steam explosions is needed. Ex-vessel explosions pose a higher risk to the integrity of BWR containments than for the PWR containments due to the smaller volume and construction configuration of the former.

Another issue in the melt-water interaction physics is that of the very low propensity for a steam explosion when a UO_2-ZrO_2 melt jet is injected in water versus the very high propensity (spontaneous) for the steam explosion when a Al_2O_3 melt jet is injected in water. Understanding of this difference in explosivity between those two melts, which must be due to the physical properties of the melt has to achieved. This may be the key to the resolution of the steam explosion issue. Research on the effects of the melt physical properties should continue.

C.8.3. Vessel Failure

The understanding of the phenomenology which governs:

- time to vessel failure
- mode and location of vessel failure
- rate, amount and composition of the melt discharge to the containment

is not adequate. The information above is needed for accident management (the time that the operator has to prevent vessel failure) and for estimation of the loads on the containment. The scenario of concern is that at lower pressure (≤25 bars), assuming that the vessel can be depressurized by the accident management measures. The scenario should also consider the convection of the melt pool and the azimuthal variation in temperatures that it produces in the vessel wall. Scaleable experiments should be performed and the data should be employed for the validation of coupled thermal-hydraulic and structural codes.

C.8.4. Melt-Structure Interactions

The interactions of interest are:

- impingement of a melt jet on vessel wall
- vessel hole ablation

Sufficient knowledge-base has been accumulated for these processes.

C.8.5. DCH

The DCH issue is resolved for most PWRs. DCH is not a concern for BWRs.

C.8.6. Hydrogen

This issue has three components

(i) the release magnitude and rate
(ii) distribution in various compartments of containment
(iii) combustion modes: deflagration, DDT and detonation

Perhaps, the main gap in knowledge-base of sufficient adequacy is in the reliable predictions of conditions for strong deflagrations and DDT.

C.8.7. MCCI and Base-mat Melt Through

This issue is of great concern for plants with dry containment cavities. There is hardly any data base on long term, 2-D ablation of a concrete basemat. Such data should be acquired and codes validated. The standard codes should be modified to include melt segregation models.

C.8.8. Melt Coolability

It is clear that the corium melt has to be cooled in order to stabilize and terminate the accident. In-vessel melt coolability and retention with external cooling of the vessel have been amply investigated. In-vessel melt coolability without external cooling, the so called gap cooling is an interesting concept and, perhaps, helped to cool the vessel wall in the TMI-2 accident. It may be effective in those scenarios, where only a fraction of the core melt (20 to 30 tonnes) relocates to the lower head. Thus, research on gap cooling should be pursued with only those scenarios in mind. Similar remarks apply to the coolability and quenchability of in-vessel debris beds. FARO experiments show in-vessel beds could be stratified with a low porosity region at the bottom and a higher porosity region on top. Coolability of deep debris beds of such configuration has not been established so far.

Ex-vessel melt coolability research has been performed over many years in the MACE Project. The integral experiments performed so far have shown partial coolability. The research may be re-directed to separate-effect experiments and the development of a model for the various heat transfer mechanisms involved in the coolability process. This issue is unresolved.

C.8.9. Fission Product Release and Transport

The PHEBUS data has provided information which may be somewhat contrary to the understanding reached earlier on this topic. At issue are the various compounds formed with iodine released and the Molybdenum release fractions.

C.8.10. Iodine Chemistry and Long Term Management of Containment and Sump Radioactivity

Iodine chemistry, in particular, the formation of the large amount of organic iodine in the containment, has been of concern since the data obtained from the PHEBUS experiments. Understanding of the processes involved with consideration of the paint on the containment surfaces is needed for (a) the estimation of the source term and (b) the proper filters to install in the containment. Scaling analyses of the PHEBUS experiments should be performed in order to extend the validity of the PHEBUS data to the prototypic plant conditions.

C.8.11. Accident Management Guidelines

Severe accident management guidance has been, or will be, provided to personnel in most plants. In general, there are not too many options available, except to add water. However, there are some accident management actions which need further understanding. An example is:

- Core Quenching: Is it beneficial to depressurize and add water at a very high rate? Small rates of addition could produce large amounts of hydrogen. The small rates may, however, be beneficial in later phases of core melt progression when there is a melt pool in the lower head covered by a metal layer.

C.8.12. Existing Computer Code Updating and Validation

There is much severe accident knowledge base incorporated in the existing computer codes. These codes serve several purposes and it is imperative that they be improved and updated to reflect the advances in the severe accident knowledge base. A serious effort is required in validating these codes against new data obtained from for the severe accident experimental programs. Quality assurance of codes is also of concern for regulatory applications.

C.8.13. Instrumentation and Diagnostics

Currently there is very little instrumentation and diagnostics which can identify for the operator the progression of a severe accident. The instrumentation vendors should be encouraged to produce reliable and robust instrumentation systems, which can withstand the harsh environment and provide information on a range of parameters, e.g. (i) the hydrogen concentration in the containment, (ii) the time of melt pool formation in the lower head, (iii) time of vessel failure (iv) time for start of basemat penetration and (v) the containment atmosphere radioactivity level.

C.8.14. Prioritization of Remaining Issues

The remaining issues and concerns described above are not all of equal importance to severe accident safety. It is imperative that the risk associated with each issue should be factored in prioritising the various issues for resolution through further research. A reasonable rationale is that the highest priority should be given to those research areas which represent a high safety risk and for which insufficient knowledge-base has been acquired so far.

C.9. Conclusions and Recommendations on Future Directions of Severe Accident Research

C.9.1. Introduction

The tempo of severe accident research has diminished greatly over the last few years all over the world. For example, the European Union severe accident research in the fifth framework program is about 40% of that in the fourth framework program. The emphasis of the program has also changed from phenomenological research to that related to obtaining the knowledge-base necessary to assure the success of the severe accident management measures. This emphasis is also supported by the responses by the regulatory authorities or their technical support organisations (TSO´s) to the Questionnaire that the authors of this report prepared. Another change in emphasis is that towards the risk-informed regulation, primarily by the USNRC, but also being increasingly considered in Europe.

One aspect, which has been pointed out by the responses by the regulatory authorities or the TSO´s is that concerning the aging of the plants. This aspect is not directly considered in the severe accident phenomenology; but it could play a very important role in two particular scenarios, namely (a) steam generator tube rupture during the high pressure accident scenario, which can lead to containment bypass and, possibly, a severe release to the environment and (b) vessel rupture at high pressure in pressurized thermal shock scenario due to the very low value of the nil ductility temperature. The latter accident scenario is not part of the severe accident phenomenology so far, and perhaps, can be managed if the crack in the vessel is not very large and not at too low location in the vessel. The steam tube rupture accident management actions have been devised, which consist of flooding the secondary side of the faulty steam generator to reduce the release of the fission products and to reduce the primary side pressure by depressurization.

C.9.2. Recommendations

In the following, we will make recommendations on the future directions of the severe accident research, based on the considerations that have been described in the previous sections of the report. The overall guiding principal is that:
Highest priority should be given to those research areas which represent a high safety risk in the nuclear power plant operations and for which insufficient knowledge-base has been acquired so far.

The recommendations below are similar to those arrived by USNRC instituted expert panel in the CSARP meeting held in May 2000, in which two of the authors of the present report participated, along with the other experts. In the following paragraphs, we have provided the recommendations for the remaining issues pertaining to the present and the future LWR plants.

C.9.2.1. Prioritization for Current Plants

C.9.2.1.1. Priority 1

9.2.1.1.1. Ex-Vessel Debris/Melt Coolability

Ex-vessel debris/melt coolability is essential for timely stabilization and termination of a postulated accident and for assuring the public that this is so. The current research programs have not reached that goal. New innovative ideas may be needed for assuring ex-vessel melt/debris coolability. Development of a model for melt coolability will require experiments with prototypic materials and simulant materials at different scales.

C.9.2.1.1.2. Ex-Vessel Steam Explosions

These can lead to early containment failure for some BWRs and possible leakage in the containments of some PWRs. There is a connection between ex-vessel steam explosions and coolability. Lack of former can provide the credible accident management option of establishing a pool of water under the vessel and forming a coolable particulate debris bed.

Recent data have shown that oxidic corium may be resistant to triggering and propogation of steam explosions. This may be the key to the resolution of the steam explosion issue. Thus, a fundamental understanding of these observations is essential, in particular for the ex-vessel conditions of low pressure and high subcooling.

C.9.2.1.1.3. Basemat Failure

This issue is important for those plants where the access of water to the ex-vessel melt/debris is not available. Basemat failure may imply contamination of ground water supplies and spread of radioactivity to the environment. The technical issue is that of the prediction of the time to basemat failure due to the long term multidimensional erosion of the concrete basemat.

C.9.2.1.2. Priority 2

C.9.2.1.2.1. Lower Head Failure and Timing

The mode of lower head failure is needed for specification of the initial conditions of melt discharge for containment loadings, in particular for the ex-vessel steam explosion analyses. The timing is important for the feasability of SAM actions to prevent vessel failure.

C.9.2.1.2.2. Core Quenching

This refers to the accident management actions of delivery of water to the vessel for (1) flooding a damaged but not yet relocated core (ii) flooding the lower head when an oxidic melt pool, covered by a metallic layer, is present. The former action may produce more hydrogen and the latter could produce stratified steam explosions.

C.9.2.1.2.3. Iodine Chemistry

The PHEBUS data indicate formation of organic Iodine, which may need additional systems for removal. Its presence can increase the environmental release in case of leaky containments and for filtered-vent releases.

C.9.2.1.2.4. Instrumentation and Diagnostics

Instrumentation to identify for the operator, the progression of the accident will facilitate proper accident management actions.

C.9.2.1.3. Priority 3

C.9.2.1.3.1. Steam Generator Tube Failure

This is of concern for the high pressure scenarios and for aged steam generators. The accident management actions are to flood the secondary side and reduce the primary pressure. Evaluation of this bypass sequence should be completed with the effects of the accident management measures.

C.9.2.1.3.2. Hydrogen Mixing and Combustion

Much work has been performed. Further evaluation and validation of the calculations for mixing and distribution of hydrogen in the containment compartments, coupled with the containment thermal hydraulics, is needed.

C.9.2.1.3.3. Existing Computer Code Updating and Validation

A serious effort is required in updating the existing codes to incorporate new knowledge gained and to validate them against new data obtained.

C.9.2.2. Prioritization for Future Plants

C.9.2.2.1. Priority 1

C.9.2.2.1.1. In-Vessel Melt Retention

Much work has already been performed, The remaining questions are on (i) the effects of melt stratification on vessel wall thermal loading (ii) the composition of the metal layer and its effect on the focussing of the heat flux, (iii) the reliability of the gap cooling mechanism and (iv) the plant maximum power level that can be reliably certified for IVMR. Some of these questions need additional experiments, while others need evaluation type research.

C.9.2.2.1.2. Core Melt Spreading and Retention in a Core Catcher

This is the severe accident management scheme, which has been employed for the EPR. The main uncertainty is in the process of retention in a crucible for mixing sacrificial material in the corium melt and its subsequent failure. It is necessary to generate high flow rates in order to assure spreading over the whole surface area of the core retention device. The other uncertainty is in the long term cooling of the spread melt. The remaining research is of evaluative type.

C.9.2.2.1.3. Core Melt Retention in an External Vessel

This is the concept of having the core melt of a large power LWR discharge from the vessel into a much larger diameter steel vessel housed in the containment below the reactor vessel. This external vessel may be lined by a ceramic material and is cooled by water. This concept has been promoted by Westinghouse-Atom for a future BWR design. Evaluation of this and similar concepts is needed.

C.9.2.2.1.4. Innovative Ex-Vessel Melt/Debris Coolability Concepts

These are new concepts for stabilizing the core melt in the containment. A prominent concept is that of adding water to the melt layer from the bottom. This concept appears to help in cooling and quenching even relatively deep layers of melt. For this concept only evaluation work is needed. Another concept is of employing downcomers which have been shown to increase the dry out heat flux in particulate debris beds. Further experimental and analytical research is needed for this concept. Other innovative in-vessel and ex-vessel melt stabilization and coolability concepts have been advanced. Some of these may need extensive research investigations.

C.9.2.3. Knowledge-Base and Readiness

It is essential to negotiate some key strategic needs that into the pursuit of above research. These include the orderly building of the knowledge-base and readiness. Knowledge-base refers to basic understanding, that is robust and transmittable. Readiness refers to immediate availability of reliable expertise and competence (not computer codes only) should ever the need for such arise. Those, together with the technical base for severe accident management, i.e. the hardware, procedures and documentation will not only safeguard public safety but also build public confidence that this is the case.

Appendix A

Responses to the Questionnaire

A:1 Answers from Belgium (AVN)

1. What organizations are supporting you?
There are no organizations supporting us on a regular basis. On a ad-hoc basis, technical support can be obtained from universities and from the SCK-CEN research centre.

2. Who is responsible for funding the SA research?
SA research in Belgium is funded by the government and by the utilities. On top of that, modest funding has been obtained from the CEC.

3. What is the safety policy/philosophy to support decision making?
The safety policy in Belgium is based on the guidelines developed in the country of origin of the plants, i.e. the US. PSA models of the plants are being developed in order to complement the deterministic approach.

4. Would you like to have the support of the SA research?
We are of the opinion that SA research should be continued in order to solve remaining safety problems, _rioriti the accident management strategy and guidelines, and maintain an adequate knowledge level needed in case of a real accident.

5. How do you use the results of the SA research?
SA research results are used by the utilities to develop the severe accident management strategy, and by the regulatory body to judge the adequacy of the strategy developed by the utilities.

6. Are you satisfied with the SA research results that you have used so far?
Globally yes, even if some results have not (yet?) been used (e.g. fission products behaviour) and others are still missing (e.g. corium coolability).

7. How have such SA research results affected your decision making related to protection against severe accidents?
SA research results have been used for the development and the implementation of SAM measures (e.g. the installation of the hydrogen catalytic recombiners), and for the construction of the containment event trees in the PSA.

8. Where and why do you see further need of SA research?
We feel further SA research is needed to reduce remaining uncertainties in well selected key phenomena (e.g. corium coolability for existing reactors), and also to maintain competence.

9. Which areas of SA research would you like to be investigated further and why? Could you prioritise?

In our opinion, high priority should be given to SA research on the following topics: in-vessel and ex-vessel coolability, and steam explosions (in particular the understanding of the differences between prototypic materials and simulants)

10. What are your requirements with respect to severe accidents?

The requirements we have with respect to severe accidents are that each plant has SA guidelines implemented, that the staff is well trained in using those guidelines, and that the strategy and guidelines be updated and optimised at regular intervals (at least every 10 years).

11. Is the focus of SA research consistent with what you deem appropriate?

Until now: yes, because it has supported the requirements identified in the question 10. For the future, it may be necessary to put again more focus on existing reactors because the easy problems have been solved and the most difficult ones are even more difficult to solve given the constraints of existing reactors (less freedom in the choice of solutions).

12. What is your view: should SA research focus on prevention or mitigation actions?

Prevention is of course the preferred strategy, but in line with the defence-in-depth concept, mitigation is also important. Moreover, mitigation is more difficult and may therefore require more research.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?

Backfits based on risk reduction have been and still are implemented. We feel that if the safety is not constantly improved (safety culture) it may quickly degrade. Deterministic arguments (including feedback of operating experience, from Belgium and from abroad) as well as insights from the level 1 PSA (including from PSA based event analyses) are used. Benefits/costs is not an issue, but of course backfits have to be "reasonable" in comparison with their safety importance. This is a qualitative concept, and therefore good engineering judgement has to be exercised.

14. Do you feel that there are sufficient SA research results to satisfy your needs?

No: we feel that research could be more focused on plant application (especially for existing plants) in order to further _rioriti SAM and also to refine the containment event trees of the PSA.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

I do not have a sufficiently broad view on future SA programs in order to give a decent answer, but as long as those future programs concentrate on the research needs identified in the previous questions, I believe they will satisfy our needs.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

No.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

Yes, especially when the reduction in uncertainties allow to have more convincing safety demonstrations, or when it allows to better quantify the safety margins, thereby giving the possibility to eliminate undue (and possibly counter productive) conservatisms.

18. Some regulatory and research organizations have concluded that the following SA issues have been resolved

- **a mode failure**
- **DCH for Westinghouse PWRs**
- **liner failure for G. E. Mark 1, BWRs**

while some of the following issues are considered unresolved

- **vessel failure modes**
- **fuel-coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatilization, ex-vessel release)**

What are your views on the above statements? Please prioritise the importance of each one of these and the needs for further research if any.

I agree with the list of resolved and unresolved issues. Concerning the unresolved, the priority for further research should be the following:

- melt debris coolability in-vessel and ex-vessel
- fuel-coolant interaction (steam explosions)
- vessel failure modes
- source term (revolatilization, ex-vessel release)
- hydrogen combustion (DDT, global detonation)

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development the SA codes?

No: SA codes are used only by the utilities and their engineering company.

20. What is you current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?

We feel that the contribution of human intervention (errors and recovery actions) to SA risk is important but has large uncertainties. More research is necessary in order to have a more balanced and realistic view of the risk profile of a plant and on the potential and priorities for further risk reduction.

21. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

Implementation of SAMGs in the Belgian plants is not yet completed; therefore it is somewhat early to identify outstanding issues. Debris coolability is certainly an issue. The DF in the secondary side of a steam generator in case of SGTR is important, not so much to develop SAMGs but to assess their effectiveness. Belgian PSAs do not (yet) consider the source term and therefore information on DF is not needed at this time, but we believe that it is an important parameter with large uncertainties and therefore that further research is needed.

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?
We believe that enough is known, except maybe on long term behaviour. We feel that the ageing characteristics of the catalytic plates are in need of further research.

23. What should be the main focus of safety, as it related to Sas - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?
We have no policy in that regard, but we are of the opinion that all reasonable preventive measures have to be taken before talking about mitigation. In other words, it is not acceptable to justify the absence of a preventive measure on the basis of the presence of a mitigative measure.

24. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?
We believe that both should be pursued. Circumvention is nice if it is done by developing robust SAMGs. By robust, we mean not too sensitive to specific assumptions. On the contrary, if circumvention is obtained by making the SAMGs more complex in order to cover all imaginable cases, it is better to reduce uncertainties.

25. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of Sas?

In existing plants where the presence of water is not guaranteed in the cavity below the vessel, hardware modifications to allow water to flow to the cavity may be implemented if it allows to significantly increase to probability to save the basemat without introducing a high risk of steam explosion.

A:2 Responses from Czech Republic (State Office for Nuclear Safety, Prague)

1. What organizations are supporting you?
Nuclear Research Institute Rez, Czech Republic

2. Who is responsible for funding the SA research?
Formerly SONS now NPPs and research institutes and international cooperation.

3. What is the safety policy/philosophy to support decision making?

Prevention of SA, preparation of EOP and SAMG or SACM.

4. Would you like to have the support of the SA research?
Yes.

5. How do you use the results of the SA research?
On the base of severe accident analysis to review some vulnerabilities of NPP.

6. Are you satisfied with the SA research results that you have used so far?
We have not results of SA analysis performed by NPPs. We have the results from PHARE project and results paid by SONS only. These results are very useful.

7. How have such SA research results affected your decision making related to protection against severe accidents?
We have it, but protection against SA is in responsibility NPPs. We can recommend this protection only.

8. Where and why do you see further need of SA research?

9. Which areas of SA research would you like to be investigated further and why? Could you prioritise?
Hydrogen phenomena, accident during low and zero power, accident in spent fuel pond, long term SA progression.

10. What are your requirements with respect to severe accidents?
Technical measures for prevention of SA, SAMG preparation.

11. Is the focus of SA research consistent with what you deem appropriate?
Yes.

12. What is your view: should SA research focus on prevention or mitigation actions?
Prevention.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?
To enhance safety with respect to cost benefit approach.

14. Do you feel that there are sufficient SA research results to satisfy your needs?
No, we need all results of SA analysis for the purpose of decision and recommendation making.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?
I think, the programmes of OECD and EC in SA is sufficient, but programme and financial support in Czech is not sufficient.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

No.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

Yes.

18. Some regulatory and research organizations have concluded that the following SA issues have been resolved

- **a mode failure**
- **DCH for Westinghouse PWRs**
- **liner failure for G. E. Mark 1, BWRs**

while some of the following issues are considered unresolved

- **vessel failure modes**
- **fuel-coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatization, ex-vessel release)**

What are your views on the above statements? Please _rioritise the importance of each one of these and the needs for further research if any.

Hydrogen combustion, source term, mode failure of containment and the items mentioned in the points 8 and 9 of this questionnaire

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development the SA codes?

Formerly Czech research _rioritise_ns used STCP code (implemented for VVER reactors by NRI Rez (Czech subject). STCP code is from NRC and SONS was owner. Now we have MELCOR implemented and verified for VVER reactors and now we want to participate in US NRC research programme CSATP (NRI Rez). MAAP code was used in PHARE project for NPP Dukovany.

20. What is you current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?

I think the human error has important contribution for SA risk. Research is necessary. It shows some probabilistic study. EOP and SAMG and their implementation and personal training can lead to reduce of contribution of human error.

21. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

The DF level is detemined by SONS Decree No. 184/1997

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?

Catalytic recombiners are instalated in Czech both NPPs, but for design basis accident, no for severe accidents. The knowledge about recombiners during severe accident conditions can be useful probably.

23. What should be the main focus of safety, as it related to Sas - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?

For selection of SA sequences for analysis and technical measures and SAMG preparation we use the combination of two criteria – high frequency and high consequences in balance.

24. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

Remaining uncertainties can be circumvented through SAMG procedures. I think that SAMG and technical measures are final product of the solution of SA phenomena. SAMG and measures must be verified a validated by additional analysis with respect SAMG and performed measures and checked by fullscope simulator. This is our practice.

25. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of Sas?

I think that it is individual approach for different type of NPPs. Additional requirements for SA research can result from the SAMG and technical measures preparation and implementation.

A:3 Responses from Finland (STUK)

1. What organizations are supporting you?

VTT (Technical Research Centre of Finland), LUT (Lappeenranta University of Technology), HUT (Helsinki University of Technology)

2. Who is responsible for funding the SA research?

Utilities as far as their backfitting plans are concerned, Ministry of Trade and Industry, VTT and STUK for more fundamental and/or regulatory-oriented research.

3. What is the safety policy/philosophy to support decision making?

That plant behaviour must be adequately understood and controlled to an appropriate degree in all situations, including severe accidents.

4. Would you like to have the support of the SA research?

We need primarily phenomenological information, not more integrated code packages – the existing codes appear more or less adequate for our purposes. Where they are inadequate we decide on other grounds (typically relying on experimental data base).

5. How do you use the results of the SA research?

See 4. In making regulatory decisions regarding requirements and utility designs to fulfil them.

6. Are you satisfied with the SA research results that you have used so far?

Research results that the utilities employed as the basis for designing and implementing their severe accident management strategies were quite satisfactory.

7. How have such SA research results affected your decision making related to protection against severe accidents?

Key elements of Loviisa SAM (in-vessel melt retention, external containment cooling, and the hydrogen management scheme for an ice condenser containment) were approved on the basis of primarily experimental information.

8. Where and why do you see further need of SA research?

Some items such as passive recombiner catalyst poisoning under both normal and accident conditions have received too little attention, even though such devices are marketed as solutions to the hydrogen problem. See also our response to item 22.

Fuel-coolant interactions and containment structural response to resulting dynamic loads are relevant to our BWRs but not yet thoroughly understood.

The phenomenology of (not just the containment pressurisation due to) high-pressure melt ejection may become significant for possible future plants, as may the confinement of severe accidents at shutdown conditions where the primary containment may need to be open (as in most BWR designs).

9. Which areas of SA research would you like to be investigated further and why?

Could you _rioritise?

See 8. List order is approximately the priority order.

10. What are your requirements with respect to severe accidents?

That the containment withstands them and the plant can be brought to a controlled state for long-term confinement. This, we think, will guarantee maintaining the releases within present release limits.

11. Is the focus of SA research consistent with what you deem appropriate?

Whose research? Our national? Nordic? EU? USA? All these have different priority lists, some match our priorities better than others.

12. What is your view: should SA research focus on prevention or mitigation actions?

Both capabilities should be available. Prevention is not in our view a “severe accident” research item.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?

Our experience is that effective backfits can be implemented with quite reasonable costs, and that it pays to develop and implement more at a steady rate. Steady development also helps maintain competence.

14. Do you feel that there are sufficient SA research results to satisfy your needs?

For the decisions made so far, this was the case. Of the future we know nothing certain.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

Envisaged by whom, or at what level? National? Nordic? EU? USA? New needs may arise, so there is no certainty that any present programs would provide satisfaction in the future. Likewise, where current priorities do not match ours, or other design-related factors preclude application in our plants, we anticipate little satisfaction from the results.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

We make use of risk insights also, but regulatory decision making can be *based* on risk analysis only if the risk analysis can be shown complete enough for the purpose, which we do not think can be done.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

No. It may even be counterproductive, because there are many different types of uncertainties, and reducing the reducible uncertainties about the presently considered questions in no way alleviates our concerns about the unknown or unaddressed issues, which fundamentally limit the usefulness of risk analysis.

18. Some regulatory and research organizations have concluded that the following SA issues have been resolved

- **a mode failure**
- **DCH for Westinghouse PWRs**
- **liner failure for G. E. Mark 1, BWRs**

while some of the following issues are considered unresolved

- **vessel failure modes**
- **fuel-coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatization, ex-vessel release)**

What are your views on the above statements? Please _rioritise the importance of each one of these and the needs for further research if any.

We have no Westinghouse or GE plants so we do not comment on those issues, but we note that high-pressure melt ejection which initiates DCH can have consequences other than containment pressurisation which yet are significant (such as challenging the integrity of penetrations and dispersed melt long-term coolability).

Of the other issues we think that it is not possible to categorically claim any of them "resolved" or "unresolved". Whether they are resolved or not is to us a plant-specific question. For example, we consider the current data base more than adequate for the demonstration of the feasibility of in-vessel melt retention in Loviisa, and unfeasibility for large power density (above, say, 1800 MWth in a typical today's PWR vessel) reactors.

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development the SA codes?

We do not see much progress can be made here.

20. What is you current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?

We feel the most serious human error possibility relates to mishandling or negligence of prevention-related safety issues among all parties (utilities, regulators, research).

21. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

We think that each SAM strategy (that the SAMG's should reflect) should be consistent and the required accuracy follows then from what is generally achievable for the various release paths.

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?

We have strong reason to believe enough is *not* known of recombiner catalyst performance and poisoning characteristics under normal operation, or possible de-poisoning phenomenology.

23. What should be the main focus of safety, as it related to Sas - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?

Main focus of safety is to maintain independent barriers, but for less likely event the barriers may not need to be as stringently designed as for more likely events.

24. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

Our experience especially with Loviisa shows that it is most efficient to circumvent uncertainties by requiring adequately demonstrated SAM strategy (of which the SAMGs are a part, and plant modifications another).

25. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of SAs?

The risk of SA's is best reduced by reducing the core damage likelihood, and this is beyond (or rather "before") the severe accident research. More robust systems to deal with events in the transient/accident categories directly reduce to severe accident risk, but these fixes are often plant(type) specific. On the other hand, SAM is an essential part of defence-in-depth and hence the risk of not meeting safety goals due to a severe accident can also be efficiently lowered by specific SAM hardware such as, but not limited to, catalytic recombiners in noninerted containments (assuming catalytic recombination fits within the overall SAM strategy of the plant in question).

A:4 Responses from France (IPSN)

1- **What organizations are supporting you?**
IPSN is the technical support of the French safety authority, DSIN.

2- **Who is responsible for funding the SA research**
The funding of the SA research is made by a subvention of the government with the participation of French (EDF) and foreign partners (European Union, USA, Japan...) for some programs (PHEBUS...)

3- **What is the safety policy/philosophy to support decision making ?**
In the domain of the SA, as the knowledge has to be developed and rules and criteria has to be established, the objective of the research at IPSN is to get a well understanding of the physical phenomena in order to have a good estimation of the risks and give a well based judgment on the methodology of safety demonstration.

4- **Would you like to have the support of the SA research ?**
Yes, but SA research has to be as far as possible oriented to give a answers useful for the plants assessment.
One important task of the experts on SA research is to conclude for transposition of research results to the reactor case.
The links between the analysis needs and the experimental programs must be strong enough to support the safety demonstration.

5- **How do you use the results of SA research ?**
IPSN strategy is oriented to the development and validation of SA codes, supported by a SA experimental program (analytical and global experiments).
This position results from the fact that generally experimental results are not directly transposable to the reactor case (scaling effects, more complex situation,...).

6- **Are you satisfied with the SA results that you have used so far ?**
For safety assessment purpose, it is necessary to spent time on synthesis and extraction of key phenomena. Moreover efforts devoted to experimental results interpretation in order to improve code validation or to assess plant behaviour has to be sufficient. Generally large uncertainties remain for the evaluation of the reactor case.

7- **How have such SA research results affected your decision making related to protection against severe accidents ?**

- Hydrogen risk : decisions have been taken to limit the hydrogen concentration in the containment of some plants by the implementation of autocatalytic recombiners, in order to prevent a containment tightness in case of hydrogen deflagration. The effectiveness and the absence of drawback (self ignition criteria) of recombiners in a severe accident conditions have been checked by an experimental program (H2PAR, KALI). Experiments on the RUT facility are going on in order to evaluate criteria for fast deflagration and DDT with application to the reactor case.

- High pressure core melt prevention : decision has been taken to prevent high pressure core melt ejection by a voluntary RCS depressurization. For this purpose it is necessary to determine with margin a pressure level below which DCH risk is prevented. In this way, DCH tests (especially US tests) give a good understanding of the DCH phenomenology (especially for cavity with no direct exit to the containment like Zion).

- Core concrete interaction tests gave orders of magnitudes of H_2, CO release in the containment (H2 risk), of fission products releases during the short term (first hour) of the MCCI.

- Accident management of the French CRUAS PWR : use of iodine tests results gave the recommendation to have a basic Ph water sump.

- In vessel corium cooling : tests on external critical heat flux gave good results for in-vessel core cooling capability.

- Corium spreading tests are giving data to validate spreading codes, used mainly to analyze the future core catcher of the European EPR reactor.

8- **Where and why do you see further needs of SA research ?**

- Accident progression : there are important uncertainties related to the later phase of in-vessel core degradation, and weak validation of calculation tools.

Important aspects for the analysis are :

- RCS pressure: increase by later reflooding,
- Hydrogen production : in case of later reflooding (SI or accumulators), corium relocation in the lower part of the vessel,
- Fuel-coolant interaction in case of water injection with a molten pool (in core location or at the lower vessel head),
- Specially important measure could be to stop the accident progression by effective in vessel corium retention system (for example ex-vessel cooling or others).
- Improvement of in-vessel thermalhydraulic knowledge.

- Ex-vessel behaviour : there are important uncertainties related to the ex-vessel corium cooling possibility and associated risk :

    - Water ingression in the reactor pit : effect of water in the vessel cavity pit at the vessel bottom head failure. The question is a key accident management one : should we introduce (if available) water in the vessel cavity pit during an accident ? benefit ? (on corium partial freezing,...) versus risks (consequences of ex-vessel steam explosion,...) ? This question makes the link between steam explosion research (at low pressure) and ex-vessel corium cooling experiments such as the present MACE tests (water on corium interacting with concrete).

- In containment phenomena : some uncertainties related to potential accident management measures :

    - Hydrogen distribution : improvement in 3D tools validation (multi-compartment containment codes such as the French TONUS code for example and associated validation tests). Studies on H2 stratification,
    - Potential effects of spray actuation (H2 homogenization, containment loss of inertization,..),
    - Hydrogen management : igniter effectiveness and without drawback.

Research on specific instrumentation : research on pertinent instrumentation needed to manage severe accident is also a subject of interest. Particularly, one challenge is to be able to assess the activity and the composition (gas, iodine) of the containment atmosphere during the accident.
Of interest also is to detect the beginning of MCCI, in order to have a correct view of the accident progression.
This instrumentation may be useful to improve decision making for crisis teams.

- Systems behaviour during severe accident conditions : notably systems involved in containment isolation function (safety injection and spray systems, containment pass-through, airlock...).
- Understanding of the phenomenology and potential source for volatile organic iodine formation (air phase and water phase). Reduction of in containment organic iodine will be an objective (paints). More effective filters for organic iodine will be of great interest.
- More specific understanding of the chemistry in the primary circuit during in-vessel fission product release.

9- **Which areas of SA research would you like to be investigated further and why ? Could you _rioritise ?**

Areas are mentioned in point 8.
Priorities :

1. later reflooding effects,
2. in-cavity water ingression,
3. H2 distribution and igniters behavior,
4. organic iodine production and potential measure to reduce or retain ,

5. in-vessel corium retention,
6. long term MCCI behavior.

10- **What are your requirements with respect to severe accidents ?**
It seems that no physical phenomena are missing. The main problem is quantification.

11- **Is the focus of SA research consistent with what you deem appropriate ?**
The research has to meet the items presented in point 8.

12- **What is your view : should SA research focus on prevention or mitigation actions ?**
The French approach concerning the severe accident was from the beginning to prevent the risk of severe accident associated to an early fission product release because the emergency plans are not consistent with these kind of scenario. Then severe accident with containment bypass and containment failure resulting from all energetic phenomena leading to an early FP release must be prevented as far as possible (global H2 detonation, high pressure core melt sequences, steam explosion leading to containment failure). For the other scenarios of severe accident, mitigation measures will be developed.

13- **What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs ?**
See questions 7 and 8.

14- **Do you feel that there are sufficient SA research results to satisfy your needs ?**
There is an important research program on severe accident, but progress in the severe accident safety enhancement remains insufficient. On one hand the results are not directly applicable to the reactor case and the expert has difficulties to conclude on the issue, on the other hand there is little availability of results for aspects concerning severe accident management.
A more important effort on synthesis and interpretation of results for the reactor case must be performed. The most important demonstration is the evaluation of steam explosion risk, and the absence of models and results of a consequence of a later reflooding.

15- **Do you see whether future SA programs envisaged will satisfy your needs in the next century ?**
The program of IPSN research will be oriented to the needs of the safety analysis.

16- **Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis ?**

17- **Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making ?**
The reduction of the uncertainties to a level consistent with the objectives of accident management is important for regulatory decision.

18- **Some regulatory and research organizations have concluded that the following SA issues have been resolved**

7. **alpha mode failure ?**
8. **DCH for Westinghouse PWRs**
9. **liner failure for G. E. Mark 1, BWRs**

while some of the following issues are considered unresolved

- **vessel failure modes**
- **fuel-coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatization, ex-vessel release)**

What are your views on the above statements ? Please prioritise the importance of each one of these and the needs for further research if any ?

DCH : Corium dispersion in the containment is strong dependent on the reactor design cavity and communication ways between reactor cavity and containment. Representative test are necessary. For French reactor cavity, the proposed test for lower pressure would be sufficient.

Priority :
For the operating plants, priority will be defined for an actual concept, and for the impact on accident management :
1- melt debris coolability in-vessel or ex-vessel (including steam explosion risk),
2- fuel-coolant interaction (in-vessel steam explosion),
3- hydrogen combustion (DDT, global detonation),
4- vessel failure modes,
5- source term (revolatization, ex-vessel release).

19-Are you using any of the SA computer codes or models to support your decision making ? What are your views on further development the SA codes ?
We use SA accident codes and engineer judgment to support our decision making.
Concerning SA codes, a much greater effort must be devoted to the validation aspects of existing models in place of new models development.

20-What is your current estimate of the contribution of human error to the SA risk ? Do you feel further research is necessary or appropriate to reduce this contribution ?
In the level 1 PSA, the weight of the sequences which include at least one human error is very important (> 80 %). However this does not mean that operators are the “weak point” of nuclear power plants. Usually, these “errors” are just the lack of recovery actions. The weight of the sequences which include at least one system failure is 100 %. Thus, research efforts should be oriented in every aspect of risk assessment, including but not restricted to human factors.

21-What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

For the general question, see above point 8. Furthermore accident management in France is under discussion between the utility and the safety authority.

Concerning the DF, the potential advantage for FP retention is evident, however the mechanical behavior of over-heated SG tubes and plate sheet in case of reflood with cold water have to be assessed. Moreover in case of late SGTR occurrence (empty SG and advance core degradation), detection means of SGTR has to be assessed.

22-Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?

An extensive experimental program has been performed by IPSN in order to evaluate the effectiveness of recombiners under severe accident conditions. Effectiveness of recombiners seems to be proved under steam and aerosols concentration, only some complementary tests will be necessary to check the effect of potential poisoning like iodine. Concerning the self-ignition risk (depending on steam and hydrogen concentration), complementary studies will be necessary in order to understand the phenomenology.

23- What should be the main focus of safety, as it related to Sas - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?

There is no formal policy on this topic in France. Each situation is considered case by case.

24- Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

If it is possible, devise a way that uncertainties could be circumvented through SAMG, is the ideal way, but in order to check the effectiveness of SAMG, reducing uncertainties on the potential loads will be necessary.

25- What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of Sas?

A:5 Responses from Germany (GRS)

1.What organizations are supporting you?

GRS is financed exclusively by order of projects.
The main employers of the GRS are

- Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) and
- Federal Ministry of Economics (BMBF, now BMWi)

2.Who is responsible for funding the SA research?

In Germany, experimental research connected to reactor safety is supported by the BMWi, while investigations for plant applications are financed by the BMU.
SA research for plant specific applications is carried out additionally by industry.

3.What is the safety policy / philosophy to support decision making?

-

4.Would you like to have the support of the SA-research?

yes

5.How do you use the results of the SA research?

- modeling of specific SA-phenomena
- SA-code validation and the qualification of code users for plant specific investigation
- direct use of SA-research as part of SAM component qualification (e.g. PAR)
- plant specific investigations for the development, lay-out and demonstration of the efficiency of SAM-measures
- training of plant personal
- proposals for the requirements and recommendations concerning SA related safety aspects of new reactor concepts

6. Are you satisfied with the SA research results that you have used so far?

- SA research is limited by use of real materials and some simulation techniques (e.g. chemical reaction kinetics, decay heat)
- range of remaining uncertainties is always large
- SAM-measures and new reactor concepts to cope with SA must be simple and robust

10. How have such SA results affected your decision making related to protection against severe accidents?

A technical solution for how to cope with a specific SA-phenomena or a SAM-measure should not be realized in case of large uncertainties for a very sensitive parameter.

11. **Where and why do you see further need of SA research?**

- properties and chemical reaction kinetics for real plant materials (e.g. vessel failure mode → chemical interactions of corium with RPV-wall, melt coolability, FCI, long-term MCCI)
- slow gas flows and natural convection pattern of gases in the reactor system and the containment (e.g. structure heat-up, gas- and aerosol distribution, decay heat removal
- SA instrumentation and process indicators
- debris – or spreaded melt coolability.

SA research is necessary because of still existing large uncertainties and to find optimal solutions for SAM-measures and new reactor concepts.

12. **Which areas of SA research would you like to be investigated further and why? Could you prioritize?**

(see answer to question 8.)

13. **What are your requirements with respect to severe accidents?**

- Development and installation of SAM measures
- Lay-out of provisions to cope with SA in new reactor concepts

In general:
concerning SA, there shall be no feed for permanent relocation, evacuation outside the immediate vicinity of the plant and long restrictions in food consumption. In Germany this general requirement is fixed by law (Atomgesetz).

14. **Is the focus of SA research consistent with what you deem appropriate?**

Not in all aspects (material properties, chemical processes, SA instrumentation and indicators).

15. **What is your view: should SA research focus on prevention or mitigation actions?**

Prevention has a higher priority than mitigation.

16. **What is your view on backfits in existing plants to enhance safety and on the issue of benefits / costs?**

Reactor safety has always to be assured based on the actual status of science and technique. Cost-benefit guidance for safety related backfits is always problematic to maintain public acceptance for the use of nuclear energy.

17. **Do you feel that there are sufficient SA research results to satisfy your needs?**

The existing international data base of experimental results is large and was never used in total by experts of the different countries. This is due to restrictions in data exchange agreements or due to proprietary rights. In addition research results gained in the past will not satisfy to days requirements (e.g. use of CFD-tools). SA research is still needed (see also answer to question 8).

18. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

No. This is due to the tendency of budget reductions in many countries and the decisions taken to close specific issues. For the lay-out of future plants to cope with SA the knowledge base today is insufficient.

19. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

Risk analysis could only support specific decision making processes, but should not be the base for regulatory demands only.

20. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

It should be, while still existing uncertainties will always raise the question about an conservative approach for safety aspects.

21. Some regulatory and research organizations have concluded that the following SA issues have been resolved

- **a mode failure**
- **DCH for Westinghouse PWRs**
- **liner failure for GE Mark 1, BWRs**

while some of the following issues are considered unresolved

- **vessel failure modes**
- **fuel coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatization, ex-vessel release)**

What are your views on the above statements? Please prioritize the importance of each one of these and the needs for further research if any.

The first statements on closed issues are in general plant specific. Concerning the list of unresolved issues, this requires a detailed discussion.

22. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development of the SA-codes?

GRS is using a number of available computer codes for risk assessment and the development of plant specific preventive and SAM measures.

Due to still ongoing SA research, including research for new reactor concepts, the SA computer codes have to be always on the actual status of science and applied technique.

23. What is your current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary for appropriate to reduce this contribution?

The contribution of human errors to the SA risk is considerable. The knowledge base today gained by research is not appropriate to further reduce this contribution.

24. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

Concerning additional research for SAMG:

- Finalize the research on hydrogen production during reflood of a highly damaged core region
- Capabilities to arrest molten core materials in-vessel (debris coolability and thermo-chemical interaction corium → vessel wall)
- Vessel failure mode related to different SA-situations
- Ex-Vessel long-term debris coolability in different cavity designs
- Long-term aging effect of systems and components, relevant to maintain a controlled plant state after the occurrence of a SA
- SA phenomenological indicators, which could be used for plant state diagnostics
- Reliable SA-instrumentation.

Concerning steam generator secondary side flooding:

In case of a steam generator tube rupture event which could lead to the loss of the steam line integrity, the effectiveness of scrubbing fission products in water by flooding the steam generator secondary side is an essential SAM-measure.

25. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations ?

In general, the knowledge-base is sufficient to decide an implementation of PARs. Existing uncertainties:
PARs leading to an ignition due to high recombination rates, start-up under cold conditions (e.g. ice-condenser containment, PAR-qualification for DBA-use).

26. What should be the main focus of safety, as it related to Sas – High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e. large releases) accidents?

Concerning HFLC it has to be checked whether there is a deficit in the design (area of DBA → prevention).
LFHC should be the main focus of SAM! (→ mitigation !)

27. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

Reducing existing large uncertainties will lead to more reliable and effective SAM-measures, also concerning SAMGs.
SAM-measures and SAMG should not be sensitive to the range of uncertainties (robust solutions).

28. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of Sas?

In case, the integrity of the pressure vessel could be kept by SAM-measures, all phenomenological aspects of the ex-vessel SA-phase could be prevented, drastically reducing the risk (long-term gas production due to MCCI, basemate-melt-through, etc.). When this could not be assured, the long-term ex-vessel debris coolability is of main importance.

A:6 Responses from Hungary (HAEA NSD)

1. What organisations are supporting you?
A: Mainly our Technical Support Organisations, i.e. the Institute for Electric Power Research Co. and the Atomic Energy Research Institute, both in Budapest.

2. Who is responsible for funding the SA research?
A: HAEA NSD is one of the main financiers, obtaining resources from the state budget, for R&D purposes. The TSOs also use some of their financial resources and certain projects are financed by the utility.

3. What is the safety policy/philosophy to support decision-making?
A: Our policy is to use as much expertise of the country as we can (including experts of the TSOs) for our regulatory decisions. R&D resources have been used to keep the knowledge-level of the TSOs high, while TSO-agreements ensure the availability of this knowledge in the regulatory decision-making process.

4. Would you like to have the support of the SA research?
A: Our policy in case of any support (not restricted to the SA research) is that although support is most welcome, the Hungarian regulatory authority tries to solve its problems by using its own resources and according to its well defined safety philosophy and priorities. In case of SA research application of this principle is limited by the high costs associated with such research activities.

5. How do you use the results of the SA research?

On one hand, these results are primarily used in assessing safety analysis reports, and in decisions related to safety enhancement of the power plant. On the other hand they form a basis of the procedures as well as of software tools used in emergency preparedness activity. Assessing and approving accident management guides shall also require such results.

6. **Are you satisfied with the SA research results that you have used so far?**
   A: Yes.

7. **How have such SA research results affected your decision-making related to protection against severe accidents?**
   Cf. Answer 5. Especially in emergency preparedness issues, SA knowledge is essential in decision-making.

8. **Where and why do you see further need of SA research?**

A: A comprehensive re-evaluation of the safety of the Hungarian nuclear plant has been performed in 1992-1994 (AGNES project), resulting in a number of conclusions and recommendations. (The current safety upgrading Program of the NPP is based on these results.) The analysis has shown that the most important factor influencing the outcome of a severe accident is the containment. Its week points from SA point of view is that it needs passive and active tools for maintaining pressure suppression on one hand and that it exhibits a relatively high leak-rate on the other hand. Consequently we have to make efforts:

- to prevent core damage and keep the integrity of the RPV by using accidnt management tools
- to reduce pressure and to decrease and/or to delay over-pressure time periods, and
- to reduce the volume of volatile fission products and aerosols in the containment.

The in-vessel phase is still interesting from the viewpoint of the preparation of the Accident Management Guides (e.g. bleed and feed) not yet ready for the plant.

For emergency preparedness purposes calculation of source terms from the possible SA sequences as well as the results of Level 2 PSA are relevant.

Application of new fuel types, as well as the use of the existing fuels for longer period, also pose SA questions.

9. **Which areas of SA research would you like to be investigated further and why? Prioritise!**
   Cf. Answer 8.

10. **What are your requirements with respect to SAs?**
    Cf. Answer 8.

11. **Is the focus of SA research consistent with what you deem appropriate?**
    A: In the sense that what we immediately need we can have performed by one of our TSOs, the answer is yes. On the other hand a small country like Hungary – among others also for financial reasons – may not conduct a well defined, concise SA R&D project covering every important issue. Thus some of our needs may only be satisfied from results of international projects.Its is to be noted, however, that – since only a small portion of the results can be directly used for our

VVER plant – better understanding of the phenomena is what generally deemed as the main benefit from the international SA research.

12. What is your view: should SA research focus on prevention or mitigation actions?

A: Since the leakage from the VVER-440 containments is quite high, prevention of an SA is more important for us than mitigation.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benetfits/costs?

A: Safety enhancing measures and the related backfitting are the only possible solutions to raise the safety of an older design plant to the current requirements. This has long been realised in Hungary and the NPP and the regulatory body considers the performance of such measures of primary importance. As for benefit/cost considerations, the present legal regulations in Hungary have no indications to that, on the other hand the HAEA NSD policy requires the application of the ALARA principle to the risk reduction.

14. Do you feel that there are sufficient SA research results to satisfy your needs?

Cf. Answer 11.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

Cf. Answer 11.

16. Will there be a change in regulatory demands on SA research if the regulatory decision-making is based on risk analysis?

A: Very likely yes, however the implications of such a change are not clear to us as yet.

17. Is gaining knowledge to reduce uncertainties in risk assessment is of value to regulatory decision-making?

Yes, and also increasing the quality of PSAs.

18. Some regulatory and research organisations have concluded that the following SA issues have been resolved: alfa-mode failure, DCH for Westinghouse PWRs, liner failure for G.E. Mark 1 BWRs, while some of the following issues are considered unresolved:

- **Vessel failure modes**
- **Fuel-coolant interaction (steam explosion)**
- **Melt-debris coolability in-vessel, ex-vessel**
- **Hydrogen combustion**
- **Source term**

What are your views on the above statements? Prioritise!

A: As for the listed possibly solved problems, we are not interested in the matter, thus we have no opinion. The listed, possibly not solved problems indeed need further investigation, our preference is for the last three topics. (Cf. also Answer 8.)

19. Are you using any of the SA computer codes or models to support your decision-making? What are your views on further development the SA codes?

A: With the help of our TSOs we are using the codes MELCOR, CONTAIN, SCADAP/RELAP, CATHARE. We also use MAAP4/VVER and some other simple codes for emergency preparedness. We believe that further development of these codes is necessary, however only in a well defined, strategic, world-wide co-ordinated manner. Actually several such projects are running parallel, (USNRC CSARP, EU PHARE, EU SRR, OECD NEA CSNI, IAEA etc.) what makes the situation difficult.

20. What is your current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?

A: According to our statistics, the direct contribution of human errors to ordinary (INES 1 or below) events in our NPP was about 25% in the last years. On the other hand, since no symptom oriented SAMGs are in force in Hungary as yet, and no thorough training of the personnel in SA situations has yet been conducted, the chance of unexpected human errors in SAs is unpredictable. Further research in this field is certainly needed.

21. What are the outstanding SA issues that require additional research to ensure proper credit for SAMGs? (E.g. decontamination factor in SF reflood?)

A: We do not have enough experience with SAMGs to answer this question.

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam and aerosol concentration?

An EU PHARE project (PH 2.07/94) addressed the question. I do not know whether the answers were comprehensive enough.

23. What should be the main focus of safety, as it related to SAs – High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e. large releases) accidents?

A: To my best knowledge a SA is per definitionem a low frequency, high consequence event, thus safety as related to SA may only concern the latter case. If it is about whether which events are more important from safety point of view ("ordinary" or SA) I have the feeling that prevention of high frequency, low consequence events represent the overwhelming majority of practical safety issues, hence automatically these are in the focus.

24. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

Since, as mentioned, we do not have proper SAMGs in Hungary, obviously the development of SAMGs is of primary importance, the question can only be properly answered after that.

25. What design-related fixes do you foresee that requires additional SA research, which if implemented could substantially reduce the risk of SAs?

Cf. Answer 8. No major design modification of the plant (related to SA risk reduction) is foreseen, except certain seismic measures, and a computer-based, digital reactor protection system refurbishment.

A:7 Responses from Japan (Nuclear Safety Commission)

1.What organizations are supporting you?

The Nuclear Safety Commission (NSC) utilizes available knowledge obtained from domestic and foreign organizations when NSC makes a decision for severe accident issues. NSC establishes five-year safety research plan on nuclear safety, from which NSC can get specific knowledge if necessary. Japan Atomic energy Research Institute (JAERI) makes a major contribution as a leading organization for LWR related severe accident research in the five-year safety research plan. NSC refers also to outcome from research and technology being performed by Nuclear Power Engineering Corporation (NUPEC) and industries.

2.Who is responsible for funding the SA research?

Science and Technology Agency (STA), Ministry of International Trade and Industry (MITI) and utilities respectively fund research and development activities in JAERI, NUPEC and industries.

3. What is the safety policy/philosophy to support decision making?

Many years of excellent operation record clearly demonstrates the high level of safety of nuclear power plants in Japan. We therefore do not think that the risk of severe accident is considerably high. However, to be as careful as possible, NSC issued a statement in May 1992, which strongly encourages all the utilities to establish accident management measures voluntarily to reasonably reduce the risk of severe accident.

4.Would you like to have the support of the SA research?

Since the risk of NPPs to the public is dominated by severe accidents and there is large uncertainty in the estimation of the severe accident risk even now, it is natural for the NSC to continue to support severe accident research. However, the priority of severe accident research in the field of nuclear safety is relatively lowered at present since phenomenological understanding has been significantly improved by worldwide research activities accelerated after the TMI-2 accident. It is important to focus the areas of severe accident research so as to more effectively satisfy the regulatory needs.

5. How do you use the results of the SA research?

SA research results have been reflected to PSAs that were performed for assisting in the planning of AM strategies and PSAs for Periodic Safety Reviews (PSRs) of NPPs. Through these PSAs, they have contributed to confirm safety of existing plants and to identify plant vulnerable points. They are also being referred to in the process of updating the site evaluation guideline. For the guideline for containment vessel design for future plants established by industries in May 1999, SA research results by JAERI and NUPEC were utilized in addition to those by industries.

6. Are you satisfied with the SA research results that you have used so far?

We are basically satisfied in the sense that the issue of severe accidents, which has been one of the most important regulatory issues, has been resolved (at least from a regulatory point of view) for existing plants by the implementation of accident management measures based on the results of severe accident research. However, we have to point out that not all issues have been resolved and not all research programs on severe accidents seem to be effective for resolution of remaining issues. A thorough discussion seems to be necessary on the direction of the severe accident research.

7. How have such SA research results affected your decision making related to protection against severe accidents?

The SA results were used for the evaluation of overall AM strategy in 1995. In the near future the SA research results will contribute to the evaluation of the effectiveness of the AM measures actually implemented at existing plants.

8.Where and why do you see further need of SA research?

Considering the fact that a wide range of SA research has been made in the 20 years after the TMI accident, we think it is the time to review the past efforts and clarify what is known and what is not in order to prioritize the remaining issues and focus the research programs on the most important areas.

9.Which areas of SA research would you like to be investigated further and why? Could you prioritize?

Check and review of current SA research results to contribute to provide necessary solutions for regulatory issues would be more important than continuation of research to clarify "involved phenomena" in discursive manner. Priority should be given for researchers to improve predictive capability of models and computer codes for analysis of accident progression and source terms for the full scale nuclear power stations. In this respect, cooperative, research with PSA and human-factor specialists should be encouraged.

10.What are your requirements with respect to severe accidents?

NSC has no legal requirement for additional protection against severe accident. However NSC is encouraging the utilities to continue their efforts to introduce severe accident countermeasures to their plants as a part of their integrated efforts to further reduce the risk of their facilities.

11.Is the focus of SA research consistent with what you deem appropriate?

Recent domestic and foreign SA research activities are not necessarily adequate, because sometimes the strategies for using the research results for the evaluation of phenomena and/or scenario of severe accidents in actual plants is no clear. For example, small-scale experiments and their analyses are solely performed without scaling strategy necessary to connect their results to the full-scale evaluation.

12.What is your view: should SA research focus on prevention or mitigation actions?

From the regulatory viewpoint, prevention takes priority to mitigation in principle. However, it is difficult to answer in a general way, because he priority depends on uncertainty and risk impact of specific issues.

13.What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?

In Japan, backfits are not made from the regulatory viewpoint. We rely on the efforts which will be made by utilities themselves.

14.Do you feel that there are sufficient SA research results to satisfy your needs?

We feel that most of SA research results are satisfactory for understanding SA phenomena. In Japan, we believe further regulatory requirements for severe accident is not necessary. However, we encourage severe accident research as reasonable effort to further reduce risk of nuclear power plants.

15.Do you see whether future SA programs envisaged will satisfy your needs in the next century?

The future SA research programs are now under discussion for establishing the next five-year (FY2001~FY 2005) safety research plan. NSC will promote SAS research which meets NSC´s needs.

16.Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

Regulatory decision making is always made reflecting risk information. Here, the risk information is not limited to PSA results. It is a matter of course that the prioritization of SA research should be made based on the risk information.

17.Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

We expect that PSA results will be applied to regulatory decision making more broadly in the future. Thus, research works to reduce uncertainties in PSA are useful. It is necessary to verify how much the severe accident research has directly helped the reduction of uncertainties in PSA.

18.Some regulatory and research organizations have concluded that the following SA issues have been resolved

- **a mode failure**
- **DCH for Westinghouse PWRs**
- **liner failure for G. E. Mark 1, BWRs**

while some of the following issues are considered unresolved

- **vessel failure modes**
- **fuel-coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatization, ex-vessel release)**

What are your views on the above statements? Please prioritize the importance of each one of these and the needs for further research if any.

We have to review to what extent each issue has been resolved. We expect that specialists in SA research summarize and evaluate the current status of extent of resolution.

19.Are you using any of the SA computer codes or models to support your decision making? What are your views on further development the SA codes?
NSC does not use any SA code directly. JAERI is developing and using SA codes and NSC gets useful information which is obtained through analyses with SA codes. The SA research results so far have been reflected in the computer codes but sometimes the use of such results is limited to small scale codes which deal with individual phenomenon. We hope that SA research results be properly reflected to integrated codes.

20.What is you current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?
In general, human errors largely contribute to the SA risk. Modeling development of human errors, especially, errors of commission during post-accident operation seems to be of value. Potential human errors in severe accident managment, however, should be studied not in the context of human factors research but in the context of providing clearly understandable and reliable Severe Accident Management Guidelines (SAMGs).

21.What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?
Since SAMGs are currently being prepared by the utilities and they will be reported to the regulatory authority (MITI) in the year 2000, we can not define the outstanding SA issues at this moment.

22.Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?
Feasibility study of catalytic recombiners is now being implemented by the utilities.

23.What should be the main focus of safety, as it related to SAs - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?
Both should be avoided by using any reasonably achievable technical measures. It is not an issue that can be answered as general consideration.

24.Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?
You may decrease predicted uncertainty and/or make the uncertainty irrelevant to the consequences, depending on the issues or related phenomena. It is thus difficult to answer this question in a general way.

25.What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of SAs?

A research will be needed to confirm appropriateness of Passive Safety features being designed for next-generation plants, though it is not a kind of fix.

A:8 Responses from Netherlands

Note: regulatory body is abbreviated as 'RB'

1. What organizations are supporting you?
No specific organisations are supporting us, as the RB is a governmental organisation.

2. Who is responsible for funding the SA research?
N/A, as the RB has no own research programme.

3. What is the safety policy/philosophy to support decision making?
See 2.

4. Would you like to have the support of the SA research?
We have no direct need for severe accident research, as we put almost all questions that arise to the licensees, as they are responsible for safety. In some areas we try to obtain sufficient insights in the associated risk before we put such questions to the licensees. Here we sometimes consult Technical Support Organisations or similar institutes that have expertise in severe accidents and/or are involved in such research themselves.

5. How do you use the results of the SA research?
In the Dutch regulations, severe accidents are addressed as items that licensees must consider in their applications. They then develop mitigation means and guidelines which we review and approve on the basis of severe accidents insights.

6. Are you satisfied with the SA research results that you have used so far?
Partly, as much severe accident research appears to be knowledge-driven, rather than application- and user-driven. Or driven by national policies. But the results that have been used to date were of good quality (e.g. re containment venting, hydrogen recombination, introduction of SAMG).

7. How have such SA research results affected your decision making related to protection against severe accidents?
As mentioned, the containment filter, the hydrogen recombiners and the SAMG that were recently introduced in our plant have been accepted by us using insights from such research. The decision to have these countermeasures was, however, made before such results were available; we anticiapted that they would become available.

8. Where and why do you see further need of SA research?
Main items are in- and ex-vessel debris coolability, local hydrogen accumulation and combustion, RPV failure and failure mode, ex-vessel FCI, MCCI in small/dry cavities to find timing of foundation failure, I&C behaviour during severe accidents, programmes that support further development of A/M, codes to help authorities to find source terms.

Reason: to reduce the areas where there are still large uncertainties in A/M, enhance A/M as such, and support Emergency Planning.

9. Which areas of SA research would you like to be investigated further and why? Could you prioritize?

See item 8. Priority is with local hydrogen phenomena (risk for early containment failure) and with I&C for A/M, as that is the main uncertainty in effective A/M.

10. What are your requirements with respect to severe accidents?

Plants must consider severe accidents and take appropriate action. This includes both hardware (equipment) and software (procedures and guidance that cover a full core melt event). It is a part of our regulation.

11. Is the focus of SA research consistent with what you deem appropriate?

Not really; few severe accident research programmes have been initiated after a careful inventarisation of the needs of regulators and utilities. Many have started because they fitted into the capabilities of an existing research centre or met national policies.

12. What is your view: should SA research focus on prevention or mitigation actions?

Prevention of core damage is not an area for severe accident research, prevention of releases is. In general, mitigative actions are the domain for severe accident research.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?

If you mean backfitting severe accidents hardware, that should be done if substantial risk reduction can be achieved. It even must be done if the risk otherwise would be too high (according to some predefined standard). Cost-benefit is difficult, as the costs of a severe accident are unknown. On the other side, backfitting to any cost is not feasible. Hence, cost-benefit should be considered in a qualitative way.

14. Do you feel that there are sufficient SA research results to satisfy your needs?

No. There are still some large uncertainties in SAMG. But as it concerns events with a very low probability, there is no immediate need for their resolution either.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

Assuming that the next century is this century: yes. We anticipate that many if not most outstanding questions will have been treated to a sufficient depth after completion of the 5th Framework Programme ('sufficient' in terms of needs of regulators and industry). Measures should be taken to preserve the corpus of know-how in the field after that time.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

Our regulatory decision making is already largely based on risk insights. In general, the shift to the use of risk insights makes it possible to define close-out criteria: if risk is reduced below a certain level, further work to fill gaps in knowledge or to reduce remaining uncertainties is not warranted.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

Partly, if the calculated mean value of the risk is sufficiently low, the uncertainties may remain still high from a regulatory viewpoint. An example is a steam explosion that leads to an early containment failure: the phenomena involved are not fully understood, but the probability is judged to be very low.

18. Some regulatory and research organizations have concluded that the following SA issues have been resolved

- **a mode failure**
- **DCH for Westinghouse PWRs**
- **liner failure for G. E. Mark 1, BWRs**

while some of the following issues are considered unresolved

- **vessel failure modes**
- **fuel-coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatization, ex-vessel release)**

What are your views on the above statements? Please prioritize the importance of each one of these and the needs for further research if any.

We agree on most of the above with the following precautions. FCI should be restricted to ex-vessel phenomena. Global hydrogen combustion is not risk relevant, local still is. Which includes flame acceleration. Source term issues do not drive A/M decisions at the plant; they are mainly relevant for Emergency Planning. Source term insights have contributed little to effective SAMG so far.

Priority should be with phenomena that may lead to an early containment failure. For NL that is hydrogen combustion (but it is understood that this is plant specific - for many plants hydrogen is not a risk).

Reference is made to item 8 where we specified the perceived needs of these and other fields, plus some other articles where we specified research needs.

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development the SA codes?

Our licensee uses a.o. MAAP and MELCOR; we usually accept the results of their calculations. Codes should be mechanistic where possible, they should be less dependent on the user experience (and be equipped with appropriate user manuals...) and be capable of integral simulation of severe accidents, including the effect of SAMG execution. They should be capable of supporting the development of simulators in the severe accident domain, in order to enhance SAMG exercises and drills. For gas distribution calculations in containments with compartments, the CFD type of codes should be further developed. Care should be exercised that codes can be validated by appropriate tests and experiments.

20. **What is you current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?**

Human error is one of the known major contributors to severe accident risks, although it is probably difficult to quantify it. Operator behaviour in mitigation of severe accidents is at present not modelled in PSAs; without appropriate guidance (SAMG type) the event may even get worse. A useful type of study would be to model operator behaviour during the execution of SAMG, in oder to enhance its effectiveness (some work is already underway).

21. **What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?**

The issues mentioned above are considered relevant from the SAMG point of view. The effectiveness of SG secondary reflood is one of the issues recognized as warranting further work. We should have a reasonable estimate of the DF, but real accuracy is not needed: the flooding is and will remain an useful A/M action. In general, the negative consequences asociated with the Candidate High Level Actions - which are the main body in existing SAMG - are fairly well understood.

22. **Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?**

Yes. They are fully qualified for those circumstances. However, some of these recombiners tend to become igniters at high hydrogen concentrations; this is an area where further work may be useful (and already is being performed).

23. **What should be the main focus of safety, as it related to SAs - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?**

This is not a very useful criterium. The cut-off should be based on risk, not on either of its components alone. One should, however, confirm that appropriate SAMG is in place to cover all scenarios with higher probabilities.

24. **Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?**

SAMG should be in place anyhow, regardless of any actual or perceived uncertainties. Optimizing SAMG can be done in both directions: to reduce the uncertainty by better understanding the processes (wich will lead to more effective countermeasures), or to introduce hardware changes to circumvent the problems. The research should be directed in the way where the best chance is to reduce risk, under the prevailing boundary conditions (which may include e.g. that no substantial hardware changes will be introduced at the plant). This is not an automatism: in the past, many research efforts have been spent in acquiring more knowledge, rather than developing effective SAMG through hardware and/or software changes.

25. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of SAs?

For new plants one could think of design related measures (but this is not relevant for the Netherlands in the absence of any new NPP development). For existing plants, possible hardware changes are highly plant specific. For some plants with a compartmentalized containment countermeasures against high local hydrogen concentrations could reduce uncertainties, maybe even risk (igniters, inertisation, dilution). Plants with a dry cavity should investigate how they can flood that cavity and how much water may be needed for the flooding to be effective. Where ex-vessel FCI is relevant, simple corium spreading devices may be more effective than expensive research in FCI phenomena (not relevant for existing NPP in NL).

A:9 Responses from Slovakia (Nuclear Regulatory Authority)

1. What organizations are supporting you?

The Nuclear Regulatory Authority of the Slovak Republic (UJD SR) is the independent central state administrative office responsible for the supervision of nuclear facilities in Slovakia in the area of nuclear safety. There is an internal support (in-house) and external support (Slovak and foreign organizations) for the UJD SR in the severe accidents. The Department of Safety Analysis and Technical Support of UJD SR provides an in-house support; the Nuclear Power Plant Research Institute, Trnava (VUJE), IAEA, and research organizations from Germany, France, USA provide an external support in the severe accidents. There are not any experimental facilities on severe accident research in Slovakia. So, Slovak contribution to severe accident research is focused mainly on the analytical support (accident analyses, development of emergency operating procedures and guidelines). Slovak organizations are also involved in the PHARE projects, bilateral and multilateral cooperation, international meetings, workshops and training courses. The international meetings, workshops and training are oriented on the sharing the experience and information in the area of severe accident research.

2. Who is responsible for funding the SA research?

The severe accident research in Slovakia is founded from the state budget. The Slovak organizations are also involved in the PHARE projects financed by European Community (EC).

3. What is the safety policy/philosophy to support decision making?

The safety policy/philosophy is defined in UJD SR legislative documents and guidelines. Any regulatory decision has to be prepared in compliance with legislation, properly supported and documented by analytical results, experimental results or engineering judgement.

4. Would you like to have the support of the SA research?

The UJD SR requires a support in the severe accident research. The UJD SR has not got enough experts to cover all areas of nuclear safety.

5. How do you use the results of the SA research?

The results of severe accident research is used at UJD SR mostly for:

a) the development of emergency procedures to be applied at the Emergency Response Center of UJD SR during the emergency drills or emergency situations at the nuclear power plants;
b) the review of accident management procedure applied at the nuclear power plants;
c) the preparation of nuclear power plant upgrading.

6. Are you satisfied with the SA research results that you have used so far?

The UJD SR requires a support in the severe accident research. However, the existing support is not considered to be sufficient. Some expected severe accident phenomena are not properly experimentally investigated. There is still a room for the continuation in severe accident research.

7. How have such SA research results affected your decision making related to protection against severe accidents?

The results of severe accident research are used for the development of emergency procedures, severe accident management guidelines, emergency planning and for the preparation of nuclear power plant upgrading.

8. Where and why do you see further need of SA research?

The UJD SR requires a support in the severe accident research. However, the existing support is not considered to be sufficient. Some expected severe accident phenomena are not properly experimentally investigated (reactor core degradation and creation of molten pool, vessel failure modes, retention of core degradation, melt debris coolability, chemical reactions between radionuclides, their transportation and release into the environment). Mathematical description of some severe accident phenomena in used computer codes (MELCOR, MAAP) is not adequate and consequently, some calculated results are not reliable. There is still a room for the continuation in severe accident research to understand the plant response to severe accidents, prevent and mitigate the severe accidents.

9. Which areas of SA research would you like to be investigated further and why? Could you prioritize?

Some expected severe accident phenomena are not properly experimentally investigated (reactor core degradation and creation of molten pool, vessel failure modes, retention of core degradation, melt debris coolability, chemical reactions between radionuclides, their transportation and release into the environment). Mathematical description of some severe accident phenomena in used computer codes (MELCOR, MAAP) is not adequate and consequently, some calculated results are not reliable. The continuation in severe accident research is needed to understand the plant response to severe accidents, prevent and mitigate the severe accidents. The prioritization will be specified after the completion and review of PSA studies level-2 for Slovak NPPs.

10. What are your requirements with respect to the severe accidents?

Severe accidents belong to the beyond design accidents. UJD SR does not prescribe any specific requirements for severe accidents with respect to the plant design. In case of emergencies, the severe accident management guidelines and emergency plans are required. They are mostly focused on the mitigation actions.

11. Is the focus of SA research consistent with what you deem appropriate?

Financial resources for severe accident research are limited in Slovakia. They do not cover the needs of research. The definition and solution of research tasks through the PHARE projects is time consuming due to EC administration and in many cases the results are out of time.

12. What is your view: should SA research focus on prevention or mitigation actions?

There is not common meaning whether severe accident research should be focused on prevention or mitigation actions. The important is to have an acceptable level of risk from radioactive releases for staff, public and environment.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?

The plant backfitting is guided to enhance the NPP safety to the internationally accepted level and to protect the public and environment. However, a certain specific level of plant safety has to be reached in any way, without cost-benefit consideration and discussion. Plant backfitting contributes to the severe accident prevention.

14. Do you feel that there are sufficient SA research results to satisfy your needs?

No, there is still a lack of severe accident research results to cover our needs. Some expected severe accident phenomena are not properly experimentally investigated (reactor core degradation and creation of molten pool, vessel failure modes, retention of core degradation, melt debris coolability, chemical reactions between radionuclides, their transportation and release into the environment) and they required a continuation in severe a accident research.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

Yes, we see.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

The Slovak regulatory body decision making is deterministic. An application of risk based decisions could change the priorities in severe accident research.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

The PSA studies level-2 for Slovak nuclear power plants are now in preparation. After completion of these studies and their careful regulatory review, the UJD SR will receive an actual specific information about the nuclear power plant response to the severe accidents, containment failure mode, risk profile, radioactive releases and impact of selected accident management measures. UJD SR will see the impact of uncertainties on calculated results, weak points of plant design and operation. The results will be used in the decision making, accident management, emergency planning, plant backfitting and formulation of needs for further severe accident research.

18. Some regulatory and research organizations have concluded that the following SA issues have been resolved:

- **a mode failure,**
- **DCH for Westinghouse PWRs,**
- **liner failure for G. E. Mark 1, BWRs,**

while some of the following issues are considered unresolved:

- **vessel failure modes,**
- **fuel-coolant interaction (steam explosions),**
- **melt debris coolability in-vessel and ex-vessel,**
- **hydrogen combustion (DDT, global detonation),**
- **source term (revolatization, ex-vessel release).**

What are your views on the above statements? Please prioritize the importance of each one of these and the needs for further research if any.

The PSA studies level-2 for Slovak nuclear power plants are now in preparation. After completion of these studies and their careful regulatory review, the UJD SR will receive an actual specific information about the nuclear power plant response to severe accidents. UJD SR will see the impact of uncertainties on calculated results, weak points of plant design and operation. UJD SR will prioritize the tasks for severe accident research.

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development the SA codes?

Mostly MELCOR, MAAP and ADAM computer codes are used in Slovakia for the modeling of severe accidents. Mathematical description of some severe accident phenomena in used computer codes is not adequate and consequently, some calculated results are not reliable and their uncertainty is high. A continuation in computer code development and validation is recommended to understand the plant response to severe accidents, prevent and mitigate the severe accidents.

20. What is you current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?

The PSA studies level-2 for Slovak nuclear power plants are now in preparation. After completion of these studies and their careful regulatory review, the UJD SR will receive an actual specific information about the nuclear power plant response to severe accidents and impact of human factor on accidents. The results of PSA study level-1 show a significant impact of human factor on calculated results. Further validation and verification of methodology and data used for the modeling and description of human behavior is recommended.

21. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

The PSA studies level-2 for Slovak nuclear power plants are now in preparation. After completion of these studies and their careful regulatory review, UJD SR will be able to answer this question.

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?

Catalytic recombiners are installed in some Slovak NPPs to increase the plant safety but do not cope with severe accidents. The installation of catalytic recombiners in the containment is not obligatory in Slovakia.

23. What should be the main focus of safety, as it related to SAs - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?

We look at both frequency of releases and release consequences to protect staff, public and environment.

24. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

A reduction of uncertainties in severe accident is important. This will avoid an excessive conservatism and will make the results of severe accident analyses more realistic. The existing uncertainties are circumvented through SAMG procedures.

25. What design-related fixes do you foresee that requires additional SA research that if implemented could substantially reduce the risk of SAs?

The PSA studies level-2 for Slovak nuclear power plants are now in preparation. After completion of these studies and their careful regulatory review, the UJD SR will be able to answer this question.

A:10 Responses from Slovenia

1. What organizations are supporting you?

VUJE (Nuclear Power Plant Research Institute, Trnava)

2. Who is responsible for funding the SA research?

NPPs

2. **What is the safety policy/philosophy to support decision making?**

No specific requirements

3. **Would you like to have the support of the SA research?**

Yes

4. **How do you use the results of the SA research?**

SAMG development (setting of preventive and mitigate measures for SA).

5. Are you satisfied with the SA research results that you have used so far?

Yes

7. How have such SA research results affected your decision making related to

protection against severe accidents?

e.g. Some preventive accident management measures have been verified. List of SA analyses -was modified according to results of previous project related to BDBA and accident management.

8. Where and why do you see further need of SA research?

(l)For development of SAMG,

(2) verification of proposed preventive and mitigative measures in Accident Management

(3) setting of hardware measures (recombiners) resulting from SA analysis.

9. Which areas of SA research would you like to be investigated further and why? Could you prioritize?

Steam explosion (in-vessel molten pool water interaction), vessel failure and direct containment heating, molten cerium-concrete interaction in the reactor cavity due to large uncertainties.

10. What are your requirements with respect to severe accidents?

To have acceptable models and codes to analyze all significant phenomena to be expected during SA and to have a validated SAMG for all NPPs operated in Slovakia

11. Is the focus of SA research consistent with what you deem appropriate?

Yes

12. What is your view: should SA research focus on prevention or mitigation actions?

Both

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?

Backfits were identified and seem reasonable.

14. Do you feel that there are sufficient SA research results to satisfy your needs?

No

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

To be determined.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

Yes.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

Yes

18. Some regulatory and research organizations have concluded that the following SA issues have been resolved

- **(x mode failure**

- **DCH for Westinghouse PWRs**
- **liner failure for G. E. Mark I, BWRs**

while some of the following issues are considered unresolved

- **vessel failure modes** 3.
- **fuel-coolant interaction (steam explosions)** 2.
- **melt debris coolability in-vessel and ex-vessel** 1.
- **hydrogen combustion (DDT, global detonation)** 4.
- **source term (revolatization, ex-vessel release)** 5.

What are your views on the above statements? Please prioritize the importance of each one of these and the needs for further research if any.

19. Are you using any of the SA computer codes or models to support your decision making?

Yes, Melcore.

20. What are your views on further development the SA codes?

They should be improved.

21. What is you current estimate of the contribution of human error to the SA risk?

There is a discussion about it, but seems to be significant.

22. Do you feel further research is necessary or appropriate to reduce this contribution?

Yes.

23. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

To be determined during SAM.G development.

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?

No

23. What should be the main focus of safety, as it related to SAs - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?

Both

24. Do you think the focus of SA research should be on reducing the remaining

uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures ?

Yes

25. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of SAs?

It will be based on PSA 2 results.

A:11 **Responses from Spain**

1. What organizations are supporting you?	Spanish Research organizations: CIEMAT (national research center) Polytechnical University of Madrid Also: CSARP, EU, and NEA programs
2. Who is responsible for funding the SA research?	CSN (regulatory agency) and UNESA (consortium of utilities) are funding organizations. CIEMAT is also partially funded from the national Spanish budget. Minor funding has been obtained from the IV Framework Program
3. What is the safety policy/philosophy to support decision making?	Our objective is the identification of risks due to SA in existing plants, the elimination of specific SA vulnerabilities when justified, and the implementation of SA management procedures. Concerning regulatory decision-making, licensees have to follow decisions made in the countries of origin of the nuclear plant main technology (USA and FRG).
4. Would you like to have the support of the SA research?	Yes,
5. How do you use the results of the SA research?	We use it to assess PSA level 2 of existing plants. We will probably use it for the assessment of specific SA management topics in the future.

6. Are you satisfied with the SA research results that you have used so far?	Not fully satisfied.
7. How have such SA research results affected your decision making related to protection against severe accidents?	Protection against SA will be based on 1. SA management procedures, which will be implemented according to generic guidelines developed by vendors and owners groups. 2. Identification of SA vulnerabilities through PSA level 2 Generally speaking, SA research has improved our knowledge of phenomenology and codes, which we are using to assess PSA's level 2. Decisions which have been or will be made, as a consequence of the assessment, are thus affected by SA research results. We have developed an emergency analysis tool named MARS, based on the MAAP code. Results of SA research are currently being used to define uncertainties of the code.
8. Where and why do you see further need of SA research?	We see further need of research, principally: 1. To define and reduce if possible the uncertainties of integrated codes MELCOR and MAAP. 2. Phenomenology areas we would like to be investigated should be related to proving the effectiveness of SA management procedures, and also to reducing uncertainties of containment failure modes which are dominant in PSA analysis
9. Which areas of SA research would you like to be investigated further and why? Could you prioritize?	SA phenomenology areas: TOP priority: hydrogen control measures

	MEDIUM: exvessel steam explosions, vessel failure modes, melt coolability inside/outside the vessel LOW: some topics concerning FP behaviour in containment (I chemistry), and also scrubbing of FP in SGTR sequences
10. What are your requirements with respect to severe accidents?	All plants must identify severe accident vulnerabilities through PSA level 2 (deadline to submit PSA of all plants is year 2000), and also propose backfitting measures if deemed appropiate, but there are no established quantitative criteria for this. Backfitting has been done in some cases as a result of PSA level 1 All plants must implement SA management procedures. Deadline is year 2001.
11. Is the focus of SA research consistent with what you deem appropriate?	Not entirely. Sometimes, new international research programs are set up with a view to answer to the needs of important labs, or try to answer to questions posed by the design needs of "future" reactors. It happens that those programs may not answer to needs of existing reactors. Thus, too much attention is given to programs which are not be useful to define or reduce the main uncertainties existing in PSA level 2 or SA management procedures. However, it is recognised that programs intended primarily for advanced reactors have given data useful for existing reactors.
12. What is your view: should SA research focus on prevention or mitigation actions?	It is not possible to give a single answer to this question. We think that reasearch of actions aimed at stopping SA progression is more important in general, but in some research fields, mitigation is the key. Examples: Reactor Pressure Vessel: prevent vessel failure.

	Exvessel steam explosions: prevent energetic explosions H2: prevent dangerous situations leading to DDT MCCI: prevent basemat failure by mitigating thermal load FP: mitigate FP inventory in containment, and mitigate FP release outside containment
13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?	SA are outside the design basis of operating plants in Spain, therefore backfitting would only be required if PSA showed "unacceptable" risk to the public. However, there is no agreed or established quantification on what is unacceptable. Qualitative considerations would be given to the cost and benefits involved, if a backfitting were ever considered. Backfits have been done as a result of PSA level 1 analysis, when a fault-tree analysis has shown that a relatively simple modification can reduce considerably the failure probabilities of a given system .
14. Do you feel that there are sufficient SA research results to satisfy your needs?	Not entirely. Anyway, we want to clarify that in our opinion, research funding is probably enough, but it should be allocated to programs which really answer to the needs of regulatory agencies and existing plants.
15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?	We feel that any new research program must pass a test of importance of the expected results in the light of actual safety needs of the plants. Research only for the sake of advancing the knowledge is not

	needed. A consultation group might be set up to coordinate future research programs.
16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?	We already use a risk informed approach in the topic of SA prevention and mitigation.
17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?	Yes. Some SA research issues have been solved after uncertainties have been reduced.
18. Some regulatory and research organizations have concluded that the following SA issues have been resolved	
- a mode failure	Yes, no more research needed
- DCH for Westinghouse PWRs	Yes, no more research needed
- liner failure for G. E. Mark 1, BWRs	Yes, no more research needed
while some of the following issues are considered unresolved	
- vessel failure modes	We have entered the LHF project. Research is needed to better define mechanical properties of vessel materials at high temperatures and creep behaviour. Medium priority.
- fuel-coolant interaction (steam explosions)	Research is needed to define the uncertainties of ex vessel steam explosions. Medium priority
- melt debris coolability in-vessel and ex-vessel	Research is needed on cooling mechanisms inside the vessel: gap cooling, entrance of water in the debris mass, upward heat flux. Also melt coolability Medium priority. We consider that Spanish plants probably will not implement ex vessel flooding. Concerning exvessel melt coolability, reasearch is needed on the possibility of cooling the melt by, pouring water on top of the melt.
- hydrogen combustion (DDT, global detonation)	High priority, especially the issue of DDT. Global detonation seems unlikely and does not need additional research.
	Additional research on the chemistry of FP in containment. Also FP scrubbing in the

- **source term (revolatization, ex-vessel release)** **What are your views on the above statements? Please prioritize the importance of each one of these and the needs for further research if any.**	secondary side of SG. Low priority
18. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development the SA codes?	Yes, we are using a number of codes, especially Melcor and Gasflow for the assessment of PSA level 2. Other important code we are using is Maap, which is the base of the MARS emergency response software. We think that further developments should try to better define and reduce if possible the uncertainty bounds of integrated codes. Development of detailed models is probably less necessary, taking into account our needs and, in any case, it should obbey to the validation needs of integrated codes.
19. What is you current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?	The impact of human error in sequences leading to core damage is well addressed in PSA level 1, although certain aspects, such as commission errors require better treatment. Procedures to deal with severe accidents are not so detailed, and also instrumentation may have been failed, or give wrong indications, therefore the probability of human errors is higher. However, without a detailed study of SA management procedures, it is difficult to estimate the contribution of human error.
20. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative	Research is needed on the efficacity of measures to control H2 concentration in the containment of PWR. We feel that it is not required to know the DF associated with SG reflood in more than an order of magnitude. We feel that a DF of 80-90 % should be conservative enough

value of water DF level for which confirmation is not needed?	
21. Do you feel that enough is known about the operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?	Catalytic recombiners are commercial devices, which probably have different performances. Probably tests have been already made on the operation of recombiners under these conditions, but little has been published. As far as we know, available effective surface of recombiners is very huge, and it would be very difficult to reduce substantially their performance due to high concentration of aerosols. The combined effect of aerosols and steam might be worth exploring, however.
22. What should be the main focus of safety, as it related to SAs - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?	Prevention of the consequences of SA should be top priority. We think the main focus of safety should be on low frequency-high consequence accidents
23. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?	Uncertainties can only be reduced, beyond a certain limit, at a very high cost. We feel that this is the case with some research fields. Therefore, in this case, the focus should be on devising ways that SA uncertainties be circumvented through SA management procedures. Maybe assessments of new research programs should include an analysis of this kind.
24. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of SAs?	Core catchers might be considered in the future, if exvessel cooling proves to be ineffective. H2 control devices in the containment, especially recombiners.

A:12 **Responses from the Sweden (SKI)**

1. What organizations are supporting you?

There is no dedicated support organization for the authority m Sweden. The authority has funds that are used to conduct research to support regulation. Universities, various national and international research organizations, and consultants are frequently used for support of the authority.

2. Who is responsible for funding the SA research?

Both industry and authority has such responsibilities. According to legislation and government decree the industry shall maintain and develop competence needed for preparedness, emergency operating procedures and guidelines for accident management updated. The authority has an obligation to fund research to enforce safety improvements and to maintain competence. The authority normally does more fundamental research than the industry. There has been a tradition of cooperation between the industry and the authority on severe accident phenomena.

3. What is the safety policy/philosophy to support decision making?

The safety policy is to support the defence-in-depth policy that implies defence and maintenance of the physical barriers to prevent releases. The government has also established quantitative safety goals for use as design bases for the mitigation of offsite radiological consequences.

4. Would you like to have the support of the SA research?

Yes, support of severe accident (SA) safety research is needed.

5. How do you use the results of the SA research?

When the Swedish severe accident consequence mitigation strategy was established, it was realised that there were large uncertainties associated with severe accident phenomena. It was therefore attempted to choose solutions that were robust against changes in the knowledge base. However, new insights should continuously be followed up in order to identify potential flaws or weaknesses in the strategy. Examples of such areas are effects of energetic fuel-coolant, interactions, coolability of corium in the containment and hydrogen deflagrations and detonation. A recent finding is that the possibility to retain the core in the vessel may be higher that earlier anticipated. This could lead to changes in the Swedish mitigation strategy.

6. Are you satisfied with the SA research results that you have used so far?

Research results have in several areas revealed even larger uncertainties than earlier anticipated. It has been difficult to converge on final conclusions. The difficulty to reach consensus among experts on specific conclusions on certain phenomena is a disappointment. However, if this disagreement reflects a more realistic view of the uncertainties of the phenomenon considered, the disagreement must in principle be considered as positive.

7. How have such SA research results affected your decision making related to protection

against severe accidents?

It has been difficult to resolve and close certain questions. Research results have for instance been used to verify the function of the emergency filters and scrubbers.

8. Where and why do you see further need of SA research?

It is needed to reduce uncertainties, in particular for risk dominating phenomena. Additional measures may be needed to reduce core failure probability, to investigate core melt chemistry and release, relocation phenomena, in-vessel coolabilitv, mode of vessel melt through, integrity of the containment with respect to fuel-coolant interactions, ex-vessel coolability, and measures to prevent containment failure. In addition one should also consider, for instance, features of more passive systems.

9. Which areas of SA research would you like to be investigated further and why? Could you prioritize?

First priority is perhaps further investigations of the possibility for early measures to limit core melt and to retain the molten core material in the vessel, since this would directly affect potential releases. The second objective would probably be to conclusively verify that the ex-vessel phenomena are taken care of in the case of a core melt. Characterization of risk dominating releases such as I, Cs, ruthenium, etc. would probably also be: requested.

10. What are your requirements with respect to severe accidents?

The requirements are outlined in a bill to the Swedish Parliament in 1980/81. It was emphasised that, although die probability of release of large quantities of radioactive material is small, measures in order to further reduce the risk should be taken. Specifically it was stated that land contamination, which impedes the use of large areas for a long period, shall be prevented, that fatalities due to acute radiation disease shall not occur, that the specified maximum release of radioactive substance shall apply to all reactors irrespective of site and power, and that extremely improbable scenarios have not to be considered for fulfilling the requirements.

In order to comply with these guidelines it was further required that any release must be limited to noble gases and at most 0.1% of the inventory of the caesium isotopes 134 and 137 contained in a reactor core of 1800 MW thermal power, assuming that other nuclides of significance in regard of land contamination are released to lesser or, at most, equal extent.

11. Is the focus of SA research consistent with what you deem appropriate?

Most of the research on SA is deemed as consistent with what we think is appropriate.

12. What is your view: should SA research focus on prevention or mitigation actions?

The main focus should be on preventing failure of physical barriers. Some research is also needed to support assessment of mitigation actions.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?

Backfits will be a normal part of reactor operation in Sweden. The reactors were backfitted.

with vented filtering and enhanced containment cooling. Large modernization programs will be carried out in order to improve safety of the plants in order to fulfill modern criteria.

14. Do you feel that there are sufficient SA research results to satisfy your needs?

The amount of SA research is probably adequate. One problem is whether we focus on the right issues. One problem is the difficulty to reach conclusions that can be internationally agreed upon and used by utilities and authorities.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

For the time ahead, the normal situation will be that there will be more questions than answers. There will not be sufficient funding, and actually not very much hope, to reach conclusive results on major areas. Decisions, for instance, on backfits will have to be made on limited information. Technical solutions will tend to be more conservative or "robust"

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

The Swedish regulatory position is mostly determined by the regulations decided by the government. The risk concept is in a way already a part of the regulations. No essential changes are expected because of a new view on the risk analysis.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

Gaining knowledge to reduce uncertainties on phenomena and thereby on the risk assessment is a key objective of the research sponsored by the regulatory authorities.

18. Some regulatory and research organizations have concluded that the following SA issues have been resolved

- **a mode failure**
- **DCH for Westinghouse PWRs**
- **liner failure for G. E. Mark 1, BWRs**

while some of the following issues are considered unresolved

- **vessel failure modes**
- **fuel-coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatization, ex-vessel release)**

What are your views on the above statements? Please prioritize the importance of each one of these and the needs for further research if any.

We believe that the potential for alpha-mode failure is resolved as well as the DCH for Westinghouse reactors. The liner problem has not really been addressed for the Swedish reactors. We thought that vessel failure mode was close to finally being resolved for Swedish BWRs, The discussion on the possibility to retain the core melt in the vessel will probably need a close review of this conclusion. Recent proposed explanations to the reason why some materials trigger steam explosions while other materials do not, and molten corium is one of the latter, have created anticipations that this question may be resolved within reasonable time. Melt debris coolability in-vessel and particularly ex-vessel is associated with a large uncertainty. The ex-vessel coolability is very important since the assumption that the core melt can be cooled inside the containment, and not cause containment failure, when it falls into water is a key element in the Swedish strategy for severe accident consequence mitigation. If the conclusion is that the melt will not be coolable, backfits will be necessary. Hydrogen will be an issue for two of the Swedish PWRs, and actions to eliminated hydrogen are probable. Source terms and ex-vessel releases are important. The results from testing in the Phebus reactor, reveal unexpected chemistry, in particular the forms of I and Cs, which probably will need a closer review with respect to applicability for Swedish reactor conditions.

The first priority is thus in- and ex-vessel coolability and retention of the molten core. If it can be concluded that threat to containment integrity by fuel-coolant interaction has a very low probability, the next priority is probably the source term assessment closely followed by the other issues. One has to bear in mind that it is not only the risk associated with certain phenomena which is important for research prioritization but also the probability to reach results of sufficient quality and reliability.

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development the SA codes?

We use computer codes to support our decision making. Typical uses are in PSA-level 2. The SA codes should be further developed. It is of particular importance that the codes are assessed systematically by comparisons with appropriate experiments so that the uncertainties of the code results can be judged.

20. What is you current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?

Human errors are only partly considered m the risk assessment. Mostly the human errors are treated very conservatively. If the requirement on a component is that it should have a passive function for a specific number of hours, operator actions are normally not credited for that period. The problem is that such conservatism dominates the results to such a degree that technical improvements do not significantly affect the overall picture of risk.

21. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

There are probably several SA issues that could warrant additional research to establish confidence in SAMGs. One is obviously the in-vessel coolability and retention. Decontamination factors are also important for scenarios that involve water scrubbing of the release. We do not quote any DF that would need no confirmation (obviously that level is 1.0). The scrubbing factors should be determined and assessed by experiments. For assessment of releases during preparedness we use the source term handbook which gives values of the order of O.OO1.

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?

We will need such devices in Sweden. The operation and function of the devices will be shown by the utility by experiment and analyses. We feel for the moment that the recombiner technology is well established.

23. What should be the main focus of safety, as it related to SAs - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?

The main focus should be on the high frequency - low consequence side. There should be a basic protection against the low frequency - high risk accidents as stated in the regulations, but the focus should be on safety improvements to minimize in particular those accidents which, through an additional human or base load error, could develop into a full blown core melt.

24. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

Ideally the focus of SA research should be on the remaining uncertainties. Given the situation that it is difficult to design and conduct experiment and analyses that actually addresses the key questions, the second option could be attractive. Devising ways that SA. uncertainties could be circumvented by SAMG procedures could be a goal for the SA research. The problem with the latter option is that it may lead to SAMGs that are far from optimal.

25. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of SAs?

Core melt cooling and retention in the vessel is very important which probably will be addressed by technical improvements such as, for instance, adding an independent water source or external cooling. Improving the overall defence-in-depth may be important. If ex-vessel coolability proves to be a dominant problem we need to start to build devices to cool the melt. Perhaps passive cooling could be implemented in existing containments. Provisions to ensure and monitor long time core melt cooling may be needed,

A:13 Responses by Switzerland (HSK)

1. What organisations are supporting you?

Ourselves.

2. Who is responsible for funding the SA research?

HSK, Paul Scherrer Institute (PSI), and utility organisation.

3. What is the safety policy/philosophy to support decision making?

According to Swiss Atomic Law, safety of nuclear installations must follow the State-of-the-Art in science and technology, and from this point of view, research must be followed very closely. Decision making, however, is supported by a number of guidelines which have been developed over a period of more than twenty years. For the most part, these guidelines follow the US NRC philosophy of the 1970s and early 1980s (i.e., deterministic and/or conservative). Currently, most of the guidelines are being revisited, in view of the fact that some inconsistencies are present (e.g., not all plants in Switzerland are licensed with exactly the same criteria), and changes are being introduced, which reflect also recent IAEA guidelines. In addition, more emphasis will be put on decision making based on results of living probabilistic safety assessments, specific for each installation.

4. Would you like to have the support of the SA research?

HSK is already funding several SA research projects, including experiments performed at the Paul Scherrer Institute (PSI), the Phébus experiments, experiments at the Royal Institute of Technology (RIT) in Stockholm, studies at Halden, the ETH Zürich, and other private institutions (see attached report).

5. How do you use the results of the SA research?

Currently, there is no direct use of the results of SA research in the activities of HSK. Some of the results, however, are factored in the codes used for regulatory decision making (e.g., SCDAP/RELAP; MELCOR, CONTAIN, etc.). In addition, in the past some research results (Revent, ACE, experiments on cables) have been used for regulatory decision making.

6. Are you satisfied with the SA research results that you have used so far?

Not completely. As mentioned, some results are factored in the generic codes used. However, when plant specific calculations are performed with these codes, a great deal of engineering judgement has still to be exercised in order to interpret the results in view of safety analyses. It must be remembered that most of SA research is conducted on separate effect tests, and the results are used to benchmark portions of the codes, which for the most part are integral models (e.g., integral representation of core degradation and fission product release, integral representation of containment phenomena, etc.). Results from the few integral tests which have been performed so far are not always easy to interpret, and thus incorporation of the data into the codes is much more problematic.

7. How have such SA research results affected your decision making related to protection against severe accidents?

As mentioned (see response to Question 5), so far most decision making has been based on deterministic rules, and severe accidents have played a very small role in HSK activities. In addition, also for matters concerning severe accidents, decisions have been mostly taken with deterministic/conservative considerations.

HSK on several occasions has used the results of severe accident research to achieve closure of outstanding issues. For instance, the results of the US NRC experiments on DCH have used to assess the DCH-induced containment failure likelihood for large dry containments (e.g., Goesgen). Other examples include, in-and ex-vessel steam explosions, and uncertainties in source terms (Phébus FP experiments). It should be noted that many of the decisions made so far, have also been based on deterministic rules, and it is only recently that HSK has started to focus on risk-informed decisions on plant-safety improvements. For instance, interesting variations on the potential benefits of containment venting, as compared with the original deterministic-based thinking, have emerged, that are based on risk methods that have generated a more serious interest on re-evaluating the deterministic-based process of decision-making at HSK.

8. Where and why do you see further need of SA research?

For the most part, HSK is of the opinion that SA issues as generic safety issues have been resolved or cannot be resolved within a reasonable time frame. This has been borne out by the plant specific PSAs performed for all Swiss installations. On the other hand, more plant specific research is needed to resolve plant related issues, which have been identified in PSAs.

9. Which areas of SA research would you like to be investigated further and why? Could you prioritise?

HSK is currently proposing to use frequency of exceedance of releases (CCDF curves) for safety criteria. All PSAs performed by HSK for the Swiss plants have shown that the overwhelming source of uncertainties in releases is associated with uncertainties in radionuclide release and transport phenomena, both in-vessel and ex-vessel, while uncertainties in accident progression phenomena inside containment play a smaller role. Therefore, research aimed at reducing or eliminating some of these uncertainties should be continued.

On the other hand, in-vessel accident progression is still not very well understood. And subject to large uncertainties. This, however, is an area where HSK is of the opinion that uncertainties will not be resolved within a reasonable time frame, therefore, if research is to be continued, it should be pursued for the sake of scientific interest, but we feel that it would likely have a limited potential application.

In addition, the Swiss plants are currently mandated to provide in the near future SA Management Guidance, therefore it is foreseen that research in this area should be increased also.

10. What are your requirements with respect to severe accidents?

That they should be of direct use for plant specific applications. As already mentioned, most of the past research which has found its way in applications has to be carefully re-evaluated every time a plant specific study is performed.

11. Is the focus of SA research consistent with what you deem appropriate?

One of HSK contractors has performed a survey for the DG XII of the CEC on the research performed during the 3rd Framework Programme (FWP) for Nuclear Safety. Only a very limited part of research founded in that entire program has been found to have been of use in SA related safety issues. Research in the just finished 4th FWP appears to have been continued under the same lines of the previous program. Therefore, the answer to this question is that more input is needed from end-users to properly focus the contents and aims of SA research.

12. What is your view: should SA research focus on prevention or mitigation actions?

Both aspects are very important. Prevention should be for the most part achieved through the use of passive systems and on improvements in operators training and procedures, or alternatively in an enhanced independence from operator interventions. Therefore, research in these areas should be enhanced. On the other hand, SAs cannot be ruled out deterministically, therefore, mitigation techniques and systems devoted to accident management should be more thoroughly investigated.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?

All Swiss plants have undergone to some extent in the past ten years to backfits related to SAs. Decisions for backfits, as mentioned, was based more on deterministic considerations than on safety research implications. All backfits introduced for preventive purposes have been proved by the plant specific PSAs to have been very effective in reducing the risks from SAs, and therefore justifiable from the point of view of costs/benefit. Some backfits, mandated for mitigating purposes, on the other hand, have been proved in some cases by the PSAs, not to be as effective as originally foreseen based on deterministic reasoning. The most noticeable case in this category is a mandated Filtered Containment Venting System for all plant designs.

14. Do you feel that there are sufficient SA research results to satisfy your needs?

As mentioned, future needs from the point of view of HSK have been identified in reducing uncertainties in some of the fission product related phenomena and in SA Management. Therefore, research activities should be carefully focussed in these areas, especially to support SA Management decisions.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

It is very hard to say about the future of the industry. A moratorium is already in effect in Switzerland, and the future appears to be primarily driven by political decisions.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

As mentioned, more focussed, plant specific research is foreseen to be needed in Switzerland.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

See the answer to Question 9.

18. Some regulatory and research organisations have concluded that the following SA issues have been resolved

1. **a mode failure**
2. **DCH for Westinghouse PWRs**
3. **liner failure for G. E. Mark 1, BWRs**

while some of the following issues are considered unresolved

1. **vessel failure modes**
2. **fuel-coolant interaction (steam explosions)**
3. **melt debris coolability in-vessel and ex-vessel**
4. **hydrogen combustion (DDT, global detonation)**
5. **source term (revolatization, ex-vessel release)**

What are your views on the above statements? Please prioritise the importance of each one of these and the needs for further research if any.
As already mentioned, HSK is of the opinion that most generic issues have either been resolved, or cannot be easily resolved. Based on plant specific studies, the statement on the first 3 issues appear correct. About the unresolved issues, only source terms issues (especially from the point of view of emergency planning) have been proven to be of importance for the Swiss plants. In addition, retention of debris in vessel by either reflooding (in-vessel) or by external cooling appears still to be the best mitigative measure, and this issue should be resolved also.

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development the SA codes?
See the answer to Question 5. As mentioned, results from these codes have still to be interpreted with much engineering judgement. Therefore, continuation of development and improvement of the existing codes, based on specific experimental results, should be supported.

20. What is you current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?
The PSAs which have been performed show that human errors are the major contributors to plant risks, therefore research in this area is still needed. However, most of the operator errors which contribute to risk are dependent on specific plant and systems designs. Generic research is of limited use, with the possible exception of the area of cognitive errors, since errors of commission have yet to be addressed in PSAs. In addition, the balance between human action and full automation should be investigated.

21. What are the outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it require to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than one order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

- In-vessel and ex-vessel debris coolability.
- Venting strategies
- Effect of plant ageing on the behavior of severe accidents

Typically, all DFs provided by water have associated uncertainties which may span two (e.g., for water in the steam generators) to 5 orders of magnitude (e.g., for pressure suppression pools), and is too dependent on accident conditions and specific plant designs. A conservative value which may be sweepily used without confirmation does not really exist.

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?

Operation of recombiners under SA conditions still has to be proven.

23. What should be the main focus of safety, as it related to SAs - High frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?

The focus should be based on expected risk. Any severe accident, irrespective of the consequences, would currently have a disastrous effect politically. Under these conditions, prevention of the most probable accidents (High Frequency, Low Consequences, HFLC) should have priority. On the other hand, low frequency accidents with large consequences (Low Frequency, High Consequences, LFHC) are for the most part accidents where preventive measures cannot be easily devised; moreover, the large uncertainties associated with their frequencies is inherent in the estimate of the initiator frequencies, and very little can be done to prevent core damage. For these, focus should be on mitigative measures and SAMG procedures.

24. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

The main aim should be on circumventing uncertainties, mostly by backfitting or through the development of SAM guidance (see answers to some of the previous questions, especially Question 13).

25. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of SAs?

Design-related modifications would be based on the priorities identified in Question 23. It can be broadly stated that, fixes for HFLC accidents would mostly include hardware modifications, which would include:

- Addition of passive safety systems.
- More capabilities for alternate in-vessel injection systems.
- Development of very fast running tools for evaluation of effectiveness of EOPs.

Fixes for mitigation of LFHC accidents would typically involve more of software modifications, including:

- Analysis of accident scenarios with SA Management measures.
- Development of SAM Guidance.
- Development of SA diagnostics and management tools.

Some hardware modifications of relative low cost for these would include:

- Core catchers in designs which allow for such systems (plant specific issue).
- Hydrogen control devices (if proven effective).
- Containment spray systems (if feasible).

Table 1 summarises the response.

Table 1 Typical hardware and software modifications envisaged to prevent, or to mitigate different types of SAs.

SA Accidents	Software needs	Hardware needs
HFLC	Evaluation of effectiveness of EOPs using very fast-running codes. Plant specific research in operator responses.	Addition of passive safety systems More capabilities for alternate in-vessel injection systems. Automation of operator actions.
LFHC	Analysis of accident scenarios with SA Management measures. Development of SAM Guidance. Computerised SA diagnostics and management tools.	Core catchers Hydrogen control devices Containment spray systems

A:14 Responses from UK (NII)

1. What organisations are supporting you?

HSE's Nuclear Safety Directorate (NSD) and its Nuclear Installations Inspectorate (NII) contracts a number of organisations to perform research on its behalf, through a competitive tendering process when appropriate. There is no preferred contractor acting as a dedicated technical support organisation.

Generic safety research is commissioned and managed through an Industry Management Committee that contains representatives of both the nuclear site licensees and NII. The safety issues that are to be addressed by this research are identified by NII in a Nuclear Research Index (NRI) in consultation with representatives of interested bodies, such as the Health & Safety Commission's Nuclear Safety Advisory Committee (NuSAC). The involvement of both licensees and the regulator in the management of this research is intended to ensure that the results of the research are relevant to real safety issues on-site.

Research relating to specific regulatory requirements is commissioned and managed directly by either the licensees or NII. Research that licensees undertake to develop or support plant safety cases as required under licence conditions is contracted directly by the licensees. Research needed by NII for safety issues related to specific regulatory decisions is contracted directly

through NII's Nuclear Safety Studies (NSS) programme. This research is conducted by a variety of technical consultancy organisations and universities, as well as by the licensees themselves, e.g. British Nuclear Fuels plc, Magnox Electric plc, AEA Technology, National Nuclear Corporation, and the Universities of Bristol and Manchester.
2. Who is responsible for funding the SA research? Plant- and site-specific research contracted directly by licensees and the NII is funded directly by whichever places the contracts NII recovers its costs, however, from the appropriate licensees. Generic research conducted under the IMC arrangements may either be contracted and funded directly by the licensees, or contracted by NSD, in which case the cost is recovered via a levy on the licensees. Ultimately, therefore, the cost of all nuclear safety research falls upon the licensees.
3. What is the safety policy/philosophy to support decision making? A general requirement of UK health and safety legislation is that the risk presented by work activities must be reduced to a level as low as reasonably practicable (ALARP). Nuclear site licensees are responsible for the safety of their operations, and conditions attached to the licences require them to demonstrate this through written safety cases. These safety cases are assessed by NII, using guidance published by HSE (Safety Assessment Principles for Nuclear Plant [SAPs]). Research may provide evidence on the nature of challenges to plant safety, and indicate ways of eliminating or reducing those challenges. The results of research may therefore be used by NII to probe plant safety cases and test whether the risk has indeed been reduced to a level that is ALARP.
4. Would you like to have the support of the SA research? Most international severe accident research programmes relate to LWR technology. The safety case for Sizewell B, the UK's first civil PWR, includes severe accident analysis, and the licensee was able to demonstrate to NII that severe accident issues had been adequately addressed in the design and operating instructions for that reactor. This safety case was built upon the results of over a decade of severe accident research, both within the UK and internationally. The NII does not require any further severe accident research to confirm the basis of that safety case. No results of more recent severe accident research have challenged or undermined that basis. The benefit of further severe accident research is therefore considered to be limited in comparison to that already realised from past research.
5. How do you use the results of the SA research? Results of severe accident research have been used to challenge assumptions and analyses presented in the Sizewell B safety case, to ensure that risks have been reduced to as low a level as reasonably practicable. The results of continuing research programmes are monitored to ensure that action would be taken if the basis of this case were undermined by changes in understanding of severe accident phenomenology.
6. Are you satisfied with the SA research results that you have used so far? The severe accident research results that have been used to date have provided satisfactory support to the process of NII assessment of the licensee's severe accident analyses.
7. How have such SA research results affected your decision making related to protection against severe accidents?

In the UK, the licensee is responsible for developing and maintaining adequate safety standards. The NII as regulator challenges the licensee's proposals to ensure that these will result in plant designs and operations with a risk reduced to a level that is ALARP. In the case of Sizewell B, the behaviour of the reactor under severe accident conditions was also examined during a lengthy public inquiry. NII initially required a study of degraded core accidents in order to:

(a) demonstrate that there is no sudden escalation of consequences just beyond the design basis;
(b) estimate the overall risk to the public of adverse health effects from such accidents;
(c) estimate the benefit of the containment in reducing the frequency of uncontrolled release of radioactivity;
(d) establish the requirements for accident management procedures (pre- and post-core-damage) and to assess the usefulness of further plant modifications;
(e) identify instrumentation and equipment required to operate in a post-accident environment and to determine the levels of qualification of this equipment.

This study became part of the station safety case, and NII used the results of severe accident research to inform its decisions on the adequacy of the severe accident assumptions and analyses presented in this safety case.

8. Where and why do you see further need of SA research?

The potential benefits of further severe accident research appear limited, since there are no proposals for construction of new nuclear power reactors in the UK. There is therefore no need for research to assist with the development of new designs. There are currently no unresolved regulatory issues associated with severe accident analysis in the Sizewell B safety case, so there is no pressing need for further research to support this either. The only area where further research might be beneficial is into instrument and plant behaviour in severe accidents, to assist with severe accident management.

9. Which areas of SA research would you like to be investigated further and why? Could you prioritise?

Apart from research into instrument and plant behaviour in severe accidents as noted above, there are no areas of PWR severe accident research that NII regards as necessary for current regulatory decision-making.

10. What are your requirements with respect to severe accidents?

NII's assessments of licensees' safety cases are guided by its published SAPs. Those relevant to severe accident analyses are: SAPs 42, 44 and 45, which set out numerical criteria for accident frequencies, SAPs 28 to 31, which provide guidance on certain aspects of severe accident analysis, and SAPs 331 to 333, which relate to severe accident management.

1 **SAP42**: the total predicted frequency of accidents on the plant that would give a maximum effective dose to a person outside the site of > 1000 mSv should be less than 10^{-4}/yr (Basic Safety Limit - BSL) and 10^{-6}/yr (Basic Safety Objective - BSO).

2 **SAP44**: the total predicted frequency of accidents on the plant with the potential to

give a release to the environment of more than: 10000 TBq of Iodine 131; or 200 TBq of Caesium 137; or quantities of any other isotopes which would lead to similar consequences to either of these, should be less than 10^{-5}/yr (BSL) and 10^{-7}/yr (BSO).

3 **SAP45**: the total predicted frequency with which the plant suffers damage and a significant quantity of radioactive material is permitted to escape from its designed point of residence or confinement, in circumstances which pose a threat to the integrity of the next physical barrier to its release, should be less than 10^{-4}/yr (BSL) and 10^{-5}/yr (BSO).

4 **SAP28:** fault sequences beyond the design basis which have the potential to lead to a severe accident should be considered, and analysed (by means of bounding cases if appropriate - SAP19). The analysis should identify the failures which could occur in the physical barriers to the release of radioactive material or in the shielding against direct radiation, and should determine the magnitude and characteristics of the radiological consequences.

5 **SAP29:** the analysis of severe accidents should be sufficiently realistic to form a suitable basis for the accident management strategies in SAP331 et seq. Where the uncertainties are such that a realistic analysis cannot be performed with confidence, reasonably conservative assumptions should be made to avoid optimistic conclusions being drawn.

6 **SAP30:** the severe accident analysis should also provide information relevant to the preparation of the site emergency plan for the protection of people outside the site in the event of a large release of radioactivity.

7 **SAP31:** where severe accident uncertainties are judged to have a significant effect on the assessed risk, research aimed at confirming the modelling assumptions should be performed.

8 **SAP331:** accident management strategies should be developed to reduce the risk from severe accidents. The strategies should primarily aim to prevent the breach of barriers to release or, where this cannot be achieved, to mitigate the consequences. The ultimate objective should be to return the plant to a controlled state in which it can be maintained in a safe condition.

9 **SAP332:** the strategies should identify any instrumentation needed to monitor the state of the plant and the level of severity of the accident, and any equipment to be used to control the accident or mitigate its consequences. Where additional hardware would facilitate accident management, this should be provided if reasonably practicable.

10 **SAP333:** provision should be made in the strategy for training plant personnel in accident management procedures and implementing the accident management strategies, utilising appropriate instrumentation and items of plant that are qualified for operation in severe accident environments.

11. Is the focus of SA research consistent with what you deem appropriate?

Much current severe accident research appears to be related to the development of designs for future reactors, which it is inappropriate for the NII to support in the absence of any proposals to construct such plant in the UK.

12. What is your view: should SA research focus on prevention or mitigation actions?
The licensee's primary objective should be to prevent accidents. NII interprets a severe accident as an event or sequence of events that, through loss of control of plant conditions, creates a potential for the release of sufficient nuclear matter to the environment to enable a person off-site to receive a dose equivalent of 100 mSv or greater. Severe accident research can therefore only be relevant in the event that the licensee fails with this primary objective, since provided that this objective is met, there are no severe accidents. Future severe accident research has therefore to address mitigation.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?
Backfits to existing plant to reduce risk should be implemented if it is reasonably practicable to do so, ie. the cost of the backfit is not grossly disproportionate to the risk that it averts.

14. Do you feel that there are sufficient SA research results to satisfy your needs?
In general, there are already sufficient PWR severe accident research results to satisfy NII's current needs.

15. Do you see whether future SA programmes envisaged will satisfy your needs in the next century?
The need for severe accident research in the next century will depend on whether or not proposals are made to construct and operate new nuclear reactors.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?
NII's approach to assessment of nuclear plant has been informed, but not dominated, by risk analysis for many years. As current plant safety cases already contain a significant amount of risk analysis, it is not anticipated that there will be any changes to regulatory requirements arising from new risk insights.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

1 NII requires PSAs provided as part of plant safety cases to be based upon best-estimate methods, and accompanied by extensive sensitivity studies . In making regulatory decisions, it considers the significance of the uncertainties and seeks assurance that these could not have an adverse impact on the overall results of the PSA.

2 Research which reduces uncertainties and demonstrates that a particular risk is lower than assessed in a safety case may be important from the viewpoint of scientific understanding, and of benefit to a plant operator wishing to demonstrate the safety of an operation, but it is unlikely to result in the operator making changes to reduce the risk further. While the regulator may take comfort from the reduction in the assessed risk of the operation, this is tempered by the knowledge that the actual risk presented by the operation is unaffected by the research because the operation itself has not been changed. From the regulatory perspective, the value of such research is therefore limited.

18. Some regulatory and research organisations have concluded that the following SA issues have been resolved:
- ? mode failure
- DCH for Westinghouse PWRs
- liner failure for G.E. Mark 1 BWRs

while some of the following issues are considered unresolved
- **vessel failure modes**
- **fuel-coolant interaction (steam explosions)**
- **melt debris coolability in-vessel and ex-vessel**
- **hydrogen combustion (DDT, global detonation)**
- **source term (revolatilization, ex-vessel release)**

What are your views on each of the above statements? Please prioritize the importance of each one of these and the needs for further research if any.

These severe accident issues tend to be plant-specific, and have all been resolved from a regulatory perspective for the UK's one civil PWR. Sizewell B was designed and constructed in a period when these issues had already been identified, and its design was optimised so far as reasonably practicable to address them. They do not generally apply to the UK's gas-cooled reactors.

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development of the SA codes?

Severe accident computer codes are not currently being used by NII to support any regulatory decisions. It is the responsibility of the licensee to demonstrate that the analyses it submits as part of safety cases use adequately verified and validated methodologies. This demonstration may include inter-comparisons of computer code predictions with those from alternative codes, performed by independent contractors. If NII could not be persuaded of the validity of the calculations, the licensee would need to improve the analysis. This could include further development of the modelling. However, no need for further development of severe accident codes is considered necessary for regulatory purposes at present.

20. What is your current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?

Severe accident risk is not quantified as such in the Sizewell B PSA. In terms of core damage frequency, the fractional contribution(1) for operator error, including maintenance errors, is in the region of 50%. This figure is dominated by risks associated with the reactor at shutdown.

(1) The fractional contribution is the sum of minimal cutsets(2) containing the item of interest (i.e. a human error contribution in this case) divided by the sum of all of the minimal cutsets.

(2) A minimal cutset is a unique combination of equipment and/or human failures which in combination with an initiating fault, leads to the undesired event, in this case core damage. There are hundreds of thousands of minimal cutsets in the PSA.

A:15 Responses from U.S.A. (NRC)

1. What organizations are supporting you?

The Office of Research (RES) at the U.S. Nuclear Regulatory Commission (NRC) conducts research to resolve safety issues, and to support regulatory decisions by NRC program offices, notably, Office of Nuclear Reactor Regulations (NRR) and Office of Nuclear Materials Safety and Safeguard (NMSS). In carrying out its mission, RES is supported by national laboratories, universities, and international research organizations.

2. Who is responsible for funding SA research?
RES funding is allocated by NRC from its annual budget, which is appropriated by the U.S. Congress through the Omnibus Budget Reconciliation Act of 1990. This act requires NRC to recover its budget through licensing fees. SA research is funded through RES budget process.

3. What is the safety policy/philosophy to support decision making?
The Commissions safety policy and philosophy in support of decision making are based on a defense-in-depth strategy. A key element of this strategy is accident prevention. Safety systems are designed and installed in a plant to prevent accidents. Furthermore, a containment is provided to confine the radionuclide release in the event of an accident. In the event of a containment failure, plants are required to implement emergency procedures and accident management strategies. Thus, the defense-in-depth philosophy incorporates a multiple barrier concept for the protection of public against radionuclide releases.

The above philosophy provides a clear and logical structure for the safety research mission covering four major program areas: reactor licensing support, reactor regulation support, nuclear materials licensing and regulation support, and radioactive waste management support. The Commissions safety policy with regard to severe accidents are promulgated in the following policy statements:

Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (10 CFR Part 50), August 8, 1995. (Federal Register 50 FR 32138).

In the policy statement, the Commission said that it had concluded that existing plants pose no undue risk to public health and safety and saw no basis for immediate action on generic rulemaking or other regulatory actions to deal with severe accidents. However, the Commission indicated its intention to initiate a systematic examination of each nuclear power plant for possible significant risk contributors. In the policy statement, the Commission also said that it fully expected that designers of new plants would achieve a higher standard of severe accident performance than prior designs.

Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (10 CFR Part 50) August 21, 1986 (Federal Register 51 FR 30028)

SECY paper, SECY-97-171, "Consideration of Severe Accident Risk in NRC Regulatory Decisions," dated July 30, 1997, provides the Commission with a summary and background on how severe accident risk has been considered by the Commission in past regulatory decision making and how severe accidents are being considered for potential future actions.

In this policy statement the Commission established two qualitative safety goals based on the principle that nuclear risks should not be a significant addition to other societal risks.

4. Would you like to have the support of the SA research?
Yes.

5. How do you use the results of the SA research?

The results of SA research are used to define systematic closure of previously identified severe accident issues. Examples are: alpha-mode failure, direct containment heating (DCH), and liner meltthrough in Mark I containment. The results are also used to perform licensing reviews of advanced and passive light water reactors for design adequacy to mitigate and/or prevent severe accidents. Examples are: passive autocatalytic recombiners (PARs) research and in-vessel melt retention by external cooling. Where phenomenological knowledge is not adequate or where uncertainties in some severe accident phenomena are unacceptably large, SA research offers improvement in understanding and reduction of uncertainties. An important product of SA research is a suite of SA codes. Recent examples where SA codes have been used to support decision making are DCH issue resolution, evaluation of steam generator tube rupture, and assessment of the impact of revised source term. Finally, SA research results are deemed useful in the transitioning process to risk-informed regulations.

6. Are you satisfied with the SA research results that you have used so far?

NRC has been conducting SA research since 1981 following the TMI-2 accident. There were eight main areas of the program: core melt progression, core-concrete interactions, direct containment heating, hydrogen combustion, steam explosions, fission product behavior, containment performance, and severe accident codes. Following the publication of SECY-88-147, "Integration Plan for Closure of Severe Accident Issues" in 1988, a more focused SA research program designed to resolve risk significant issues was put in place. As stated in response to the previous question, results of SA research were useful in defining systematic closure of a number of severe accident issues. The results were also useful in reducing uncertainties or improving the phenomenological understanding. For example, results from the core melt progression research provided an understanding of uncertainties associated with initial conditions used in the resolution of the Mark I liner failure and the DCH issues. Results of FCI phenomenological research led to a better assessment of steam explosion potential of prototypic core melt. Likewise, results of hydrogen combustion research led to a better assessment of containment threats from DDT and of hydrogen control measures to mitigate such threats. Therefore, SA research over the years has been very productive.

7. How have such SA research results affected your decision making related to protection against severe accidents?

The Commission has considered severe accidents in its regulatory decisions and actions since its early days. For example, accidents more severe than the design basis accidents were clearly a consideration in the Commission's decision on reactor siting criteria. The source term used for the assessment of the Part 100 dose guidelines was based upon a "substantial meltdown of the core." The risk insights provided by WASH-1400 were considered in the staff's development of recommendations for emergency planning requirements.

Following the accidents at TMI, the Commission's regulatory decisions and actions utilized greater consideration of risks from severe accidents. Anticipated Transient Without Scram (ATWS) and Station Blackout (SBO) rules were issued by the Commission in the 80's in consideration of severe accidents and evaluation of potential new requirements for plants to deal with such accidents. Both ATWS and SBO were identified in the safety studies as important contributors to risk. In 1988, NRC issued Generic Letter 88-20 which required licensees to conduct Individual Plant Examinations

(IPEs) for vulnerabilities to severe accidents. In 1990, NUREG-1150 was published which provided a quantitative assessment of plant risks against severe accidents. NUREG-1150 also provided a model for subsequent PRA studies used in the design certification reviews of advanced and evolutionary plant designs.

Much of the ongoing SA research activities is being coordinated under the PRA Implementation Plan, a major element of which is the development of a risk-informed decision making framework. Improvement in the understanding of severe accident phenomena and reduction of uncertainties through continued SA research would be beneficial in that regard.

8. Where and why do you see further need of SA research?

Currently, we have adequate understanding of SA phenomena to make regulatory decisions. However, some SA phenomena are still treated conservatively which may hinder systematic implementation of risk-informed regulations. Therefore, further need of SA research is perceived where phenomenological knowledge is inadequate or where uncertainties in severe accident phenomena are unacceptably large. The following documents summarize our views of the status and plans for severe accident research.

SECY-98-131, "Status of the Integration Plant for Closure of Severe Accident Issues and the Status of Severe Accident Research, " dated June 8, 1998.

Memorandum to the Commission from W. Travers, "Schedule for Closure of Severe Accident Issues and Severe Accident Research Activities," November 9, 1998.

The severe accident research program, described in SECY-98-131, consists of activities in six areas: (1) hydrogen combustion, (2) direct containment heating (DCH), (3) fuel-coolant interactions, (4) lower head failure/vessel integrity, (5) fission product release and transport, and (6) code development, assessment and maintenance. The most recent focus of hydrogen combustion research has been the investigation of detonability of hydrogen-air-steam mixtures in order to obtain data for model verification, and mitigation of hydrogen release through passive autocatalytic recombiners (PARs) in support of the AP600 licensing review. We now believe that significant information exists on the hydrogen combustion issue, which is sufficient to assess possible threats to containment integrity. We also note that there is enough information about the operation of passive autocatalytic recombiners to adequately design the system to account for the effects of high steam and aerosol concentrations. All experimental hydrogen combustion research programs are considered complete at this point.

Direct Containment Heating (DCH) research has resulted in the closure of this issue for all Westinghouse, Combustion Engineering, and Babcock & Wilcox large dry and subatmospheric PWR plants. Additional research is currently being performed to address resolution of DCH for ice condenser plants. Historically, DCH has not been considered to be as risk significant for BWR plants.

Steam explosion research conducted thus far was useful in resolving the alpha-mode failure issue from a risk perspective. Research has also produced data demonstrating that it is difficult for a prototypic core material to explode under most severe accident conditions of interest. However, some residual issues remain concerning mixing and triggering of a large mass of prototypic material under subcooled and low pressure conditions, typically associated with ex-vessel scenarios. Continuation of FCI research at a modest level would be useful in this regard. With regard to ex-vessel quenching and debris coolability, demonstration of successful quenching at reactor scale is needed to terminate accident progression and confirm the effectiveness of accident management strategies. Past and ongoing coolability programs have not provided definitive demonstrations; thus, a focused effort is needed to resolve theex-vessel coolability issue.

Thermal loads imposed on the lower head by molten corium is the subject of the RASPLAV program, performed at the Russian Research Center. Examination of lower head failure modes is the objective of the Lower Head Failure (LHF) program at the Sandia National Laboratories. The LHF program is relevant to an accident management issue involving reflood of a partially depressurized reactor vessel following a core melt accident. Reflooding may cause significant repressurization when the vessel is approaching failure temperatures thus increasing the likelihood of vessel failure. Results of the LHF program are expected to shed some light into the efficacy of the reflood strategy. Both the LHF and the RASPLAV programs are being conducted under the auspices of OECD.

The scope of fission products research is confined to participation in the PHEBUS-FP program and, otherwise, to support the regulatory side of NRC in the implementation of the revised source term. Future PHEBUS tests may examine issues such as high burnup effects, mixed oxide fuel behavior, and fission product behavior following shutdown accidents.

The analysis of plant response to severe accidents is a key component of severe accident research in support of risk informed regulatory initiatives and resolution of safety issues. As the NRC experimental programs are gradually terminated, the emphasis is placed more on developing and maintaining analytical capabilities. Over the next several years, code capabilities will be consolidated to reduce the number of codes actively maintained while sustaining vital expertise in key phenomenological areas.

In summary, very few experimental SA research programs will continue, primarily under the auspices of international cooperative programs, in the short term to address risk significant phenomenological issues, . In the long term, work on severe accident codes will continue in support of risk informed regulatory initiatives. Also, in the long term, new emerging issues such as MOX and high burnup fuel behavior and future design issues may create new avenues of SA research, especially, in light of our transition to a risk-informed regulatory framework.

9. Which areas of SA research would you like to be investigated further and why? Could you prioritize?

The current thrust of NRC's SA phenomenological research is in-vessel severe accident phenomena (e.g., RASPLAV and OECD LHF), the rationale being that if in-vessel core melt coolability or retention is assured, there need not be further concern about ex-vessel issues. There is a recognition, however, that for high power reactors, in-vessel core melt retention may not be assured thus creating the likelihood of RPV failure. Also, as stated in connection with the LHF program in Question 8, accident management strategy of reflooding a partially depressurized RPV following a core melt accident may create the likelihood of RPV failure. Therefore, certain ex-vessel issues (e.g., ex-vessel FCI and coolability) may require further investigation. However, as mentioned elsewhere, facilities required to conduct the needed research are being closed down. As a result, remaining severe accident issues may not attain the same degree of resolution as the resolved issues. With regard to NRC's SA research prioritization, ongoing phenomenological research, particularly the cooperative international programs, will be brought to an orderly closure in the short term. In the longer term, emphasis will be directed toward improvement and assessment of severe accident analytical tools or codes.

10. What are your requirements with respect to severe accidents?

As implied in the Commission's policy statements referenced above, currently there are no regulatory requirements with respect to severe accidents for operating plants though the Commission has placed an increased emphasis on severe accident risks following the TMI accident. However, the Commission fully expects that designs of new plants would achieve a higher standard of severe accident performance than prior designs. In particular, the policy statements reaffirmed the Commission's belief that a new design could be shown to be acceptable for severe accident concerns if it: (1) demonstrated compliance with 10CFR50.34(f) requirements, (2) demonstrated technical resolution of all unresolved safety issues (USIs) and medium to high priority generic safety issues (GSIs), and (3) contained a design-specific PRA with consideration of severe accident vulnerabilities. The guidance was subsequently codified in 10CFR52.47. The subject of generic rulemaking on severe accidents for new reactor designs is discussed in SECY-97-148, which concludes that such an action is not needed at present.

11. Is the focus of SA research consistent with what you deem appropriate?

Formulation of the current focus of SA research followed the recommendations in SECY-88-147, i.e., resolution of risk-significant severe accident issues. Still consistent with the defense-in-depth safety philosophy, the focus, however, emphasized early containment failure. Important SA issues were identified and phenomenological research was performed to close these issues in a systematic manner. The closure process relied on best estimate tools which have inherent uncertainties, albeit, acceptable ones on basis of some risk measure (perspective). In that sense, the focus of NRC's SA research is consistent with what has been deemed appropriate. With the transition to a risk-informed regulatory framework, there is a stipulation that some SA issues may not pose as much risk as previously estimated. However, risk studies are not complete yet to verify the stipulation. Resolution of other issues may require that the best estimate tools to be used be more accurate than those originally designed. Future SA research should take this into consideration.

12. What is your view: should SA research focus on prevention or mitigation actions?

The Commission's Severe Accident Policy stated the desirability of performing a systematic examination of each nuclear power plant in order to identify potential plant-specific vulnerabilities to severe accidents. The policy led to the issuance of the Generic Letter 88-20 establishing the IPE program. One stated purpose of the IPE is to gain a better understanding of severe accidents in order to prevent or mitigate the same. Whenever possible, emphasis is placed on prevention through a reduction of the overall probabilities of core damage and fission product release. This may be achieved through a better understanding of the phenomenology and the use of best estimate tools to reduce undue conservatism in the estimation of severe accident risk. Or, it may be achieved by modifications of hardware and/or procedures. Mitigation of severe accidents is the next order of priority.

Generally, for new reactor designs, there are opportunities and provisions for design features and/or modifications to prevent severe accidents. For operating plants also, accident prevention takes priority over mitigation. However, when prevention cannot be assured, the focus shifts to mitigation by devising accident management strategies.

13. What is your view on backfits in existing plants to enhance safety and on the issue of benefits/costs?

The NRC promulgated its first Backfit Rule in 1970 which set forth a standard governing when the NRC could require a plant previously licensed to incorporate a new safety feature. The rule excepted from this standard any backfit that was necessary to bring a facility into compliance with its license or a Commission rule or regulation. A backfit of this kind was apparently always required. The Final Backfit Rule, which included cost impact in the consideration of backfits, was issued in 1985. The rule stated that: The Commission shall require the backfitting of a facility only when it determines ... That there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection. The Final Backfit Rule was further amended in 1988 and published as 10CFR50.109. Implementation of the Backfit Rule continues to evolve due to ambiguity concerning such languages as "substantial increase in the overall protection" and "costs ... are justified." Processes for measuring substantial increase in protection and verifying cost justification are also evolving as the PRA technology improves.

In the context of severe accidents and related safety issues, SECY-97-171, discussed above in response to Question 7, provides the Commission with a background on how severe accident risk has been considered by the Commission in regulatory decision making. Following the accidents at TMI, the Commission's regulatory decisions and actions utilized greater consideration of risks from severe accidents. The Generic Letter 88-20 was issued in 1988, which required licensees to conduct Individual Plant Examinations (IPEs) for vulnerabilities to severe accidents. Insight gained from IPEs was useful in the development of severe accident management guidance (SAMG). In the United States, management of severe accidents is regarded as an industry initiative. Utilities are expected to develop plant-specific SAMG documents, however, there are no regulatory requirements at present and backfitting is not imposed on the industry as a part of severe accident management strategies.

The two design-related fixes that came about as a result of SA research and that may be considered in the backfit category are: hardened vent for BWR Mark I containment and implementation of hydrogen control measures. We are not aware of any other backfit requirement resulting from SA research.

14. Do you feel there are sufficient SA research results to satisfy your needs?

To the extent research results were used to close some of the SA issues, the needs were adequately met. There are some areas where our understanding is not as mature as desired. In those areas, if a need should exist to reduce uncertainties further to meet specific future requirements as, for example, may be imposed by the transition to a risk-informed regulatory framework, it will be accordingly addressed.

15. Do you see whether future SA programs envisaged will satisfy your needs in the next century?

With the completion of the current SA experimental research programs, in particular, the LHF program at Sandia and the RASPLAV program at RRC, we expect to have an adequate understanding of severe accident issues which may be important risk contributors for current vintage of designs. The extent of knowledge from over 18 years of SA research is expected to enable plant operators to devise strategies to prevent severe accidents from occurring or mitigate such accidents, should they occur. However, in the course of implementing risk-informed regulations, one may need to reduce phenomenological uncertainties further thus requiring additional focused research. Furthermore, as we continue to exercise the SA codes, we may identify areas where the current understanding of the phenomena precludes adequate prediction of the consequences. In these cases, a focused research effort to improve the understanding and subsequently, to improve the codes may be warranted. Also, as stated in response to Question 8, new emerging issues such as MOX and high burnup fuel behavior and future design issues may create new avenues of SA research, especially, in light of our transition to a risk-informed regulatory framework.

16. Will there be a change in regulatory demands on SA research if the regulatory decision making is based on risk analysis?

The answer, in principle, is yes. The nature and extent of changes will depend on the results of risk analysis. SECY paper, SECY-98-300, "Options for Risk Informed Revisions to 10 CFR Part 50 - Domestic Licensing of Production and Utilization Facilities," dated December 23, 1998, proposed high-level options for modifying regulations in 10 CFR Part 50 to make the regulations risk informed and laid out associated policy issues for Commission consideration. Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated July 1998, describes a consistent approach to regulatory decisions in areas where the results of risk analysis will be used to justify regulatory action.

As a follow on to the above initiatives, the staff will start the process of making Part 50.59, dealing with licensee-initiated changes to the facilities and designs already in FSAR, risk-informed. As a possible outcome of this process, some current regulatory requirements may not be warranted based on risk significance and hence, a burden reduction may be in order. Yet, other regulations may need to be revisited if the process demonstrates that the issues covered by the regulations are more risk significant than previously considered. In these cases, results of SA research will be used to determine additional regulatory requirements.

17. Is gaining knowledge to reduce uncertainties in risk assessment of value to regulatory decision making?

For operating plants which meet existing regulatory and policy requirements, the answer is no unless a licensee requests amendment to its license using risk-informed regulations. The staff is also exploring changes to the body of the Part 50 regulations to incorporate risk-informed attributes. As stated in SECY-98-300, these changes could involve such actions as developing a new set of design-basis accidents, revising specific requirements currently in Part 50, or deleting unnecessary or ineffective regulations. Some of these actions may require further reduction of uncertainties. For future designs, gaining knowledge to reduce uncertainties is certainly of value since the new designs are fully expected to achieve a higher standard of sever accident performance.

18. Some regulatory and research organizations have concluded that the following SA issues have been resolved
- alpha mode failure
- DCH for Westinghouse PWRs
- liner failure for G.E. Mark I, BWRs
while some of the following issues are considered unresolved
- vessel failure modes
- fuel-coolant interaction (steam explosions)
- melt debris coolability in-vessel and ex-vessel
- hydrogen combustion (DDT, global detonation)
- source term (revolatization, ex-vessel release)
What are your views on the above statements? Please prioritize the importance of each one of these and the needs for further research, if any.
NRC is in complete agreement with the statements concerning the status of resolution of the alpha mode failure, the DCH (for PWRs), and the Mark I liner failure issues. Additionally, NRC is of the opinion that adequate research has been performed on hydrogen combustion and in-vessel steam explosions. We believe significant information exists on the hydrogen combustion issue, which is sufficient to assess possible threats to containment integrity. Ongoing programs (LHF program at Sandia, FAI in-vessel coolability program, RASPLAV program) are expected to yield additional information to close some of the remaining unresolved issues on in-vessel coolability and lower head integrity. PHEBUS program likewise is expected to yield information needed to resolve specific source term issues. With the unexpected termination of the FARO program, data on ex-vessel steam explosions and FCI will be severely limited. Finally, if the MACE program is discontinued, data on ex-vessel melt coolability (by top flooding) will be limited and consequently, the issue may remain unresolved.

19. Are you using any of the SA computer codes or models to support your decision making? What are your views on further development of the SA codes?
SA codes are used to support decision making, but they are not the only tools used. Recent examples of the use of SA codes in decision making are: licensing review of AP600, evaluation of steam generator tube rupture scenarios and effect of SA conditions on repair failures, calculations in support of the DCH issue resolution, assessment of the impact of the revised source term, and evaluation of spent fuel pool zirconium fire. As mentioned elsewhere, current emphasis of NRC's severe accident research is on SA code improvement and assessment. Further development of codes, when deemed appropriate, is considered an integral part of this current thrust.

20. What is your current estimate of the contribution of human error to the SA risk? Do you feel further research is necessary or appropriate to reduce this contribution?

We believe that human error is a significant contributor to severe accident risk, and that further research is needed in this area. NRC sponsors research on the incorporation of human errors of commission into PRAs. Such errors are generally not considered in the current PRAs. In addition, NRC is participating in two international efforts related to human reliability and risk analysis: PWG-5 work in this area and COOPRA study of organizational influences on risk.

21. What are outstanding SA issues that require additional research to ensure proper credit for Severe Accident Management Guidelines (SAMGs) being considered by various Owner's Groups? For instance, is it required to know the decontamination factor (DF) associated with water, under conditions of steam generator reflood following a tube rupture scenario, within more than an order of magnitude? What is the conservative value of water DF level for which confirmation is not needed?

SAMG considered by various owner's groups involve certain generic actions based on symptoms that are available to the operators. Such actions include depressurization of RPV, RPV flooding, containment flooding, hydrogen control, radioactivity release control and reactivity control. As mentioned elsewhere, SAMG is an industry initiative. In developing generic SAMG or plant specific guidelines, industry has made use of available SA research results. Additional research on some of the unresolved SA issues, identified elsewhere, is expected to aid in further improvement of SAMGs. As for the specific example in question, it is believed that an order of magnitude accuracy in DF calculations is adequate for most, if not all, applications.

22. Do you feel that enough is known about operation of catalytic recombiners under atmospheric conditions with high steam, and aerosol concentrations?

Yes, there is enough information about the operation of passive autocatalytic recombiners to adequately design the system to account for the effects of high steam and aerosol concentrations. This conclusion is drawn from the test program that we conducted at the Sandia Surtsey facility under conditions expected during severe accidents.

23. What should be the main focus of safety, as it relates to SAs - high frequency low consequence (but highly uncertain with regards to fission product releases) accidents, or low frequency (highly uncertain w.r.t. frequency of release) high consequence (i.e., large releases) accidents?

In conjunction with the transition to a risk-informed regulatory framework, NRC is looking into the use of Large Early Release Frequency (LERF) as a basis for PRA acceptance guidelines, in addition to the use of core damage frequency (CDF). Certainly, large release is a concern and, as such, remains an important focus of safety. Under a risk-informed regulatory framework, the focus would continue to be placed on credible events and not on speculative scenarios. Also, in implementing LERF and CDF bases, greater care will be exercised to properly quantify risks and to assure consistency with the Commission's Safety Goal Policy Statement.

24. Do you think the focus of SA research should be on reducing the remaining uncertainties or on devising ways that SA uncertainties could be circumvented through SAMG procedures?

Reducing uncertainties happens to be one aspect of SA research. However, diminishing benefits from an open-ended SA research addressing uncertainties alone should be weighed against risks.

25. What design-related fixes do you foresee that requires additional SA research, that if implemented could substantially reduce the risk of SAs?

For operating plants, we have concluded that these plants do not pose an undue risk to the public and no design-related fixes are required except for hardened vents and hydrogen control measures. Moreover, for operating plants, SAMG is an industry initiative and there is no regulatory imposition on the industry for major design-related fixes that may require additional SA research. We are not aware of any other design fixes proposed by the industry that warrant additional SA research.

For new plants, however, there is expectation that the designs shall achieve a higher standard of SA performance than the operating plants. One design feature that has been investigated extensively in relation to the AP600 design is the in-vessel retention capability through external flooding of the vessel. Provision of cavity flooding is considered as well in the ABWR design to mitigate core-concrete interactions, promote debris cooling, and provide fission product scrubbing. Similarly, reliable depressurization systems have been considered for the System 80+ design to address issues associated with high pressure melt ejection.