Assessment of Phenomenological Uncertainties in Level 2 PRAs[1]

Hossein P. Nourbakhsh and Thomas S. Kress
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Abstract- *This paper presents an assessment of the phenomenological uncertainties associated with probabilistic accident progression and source term analyses (Level 2 PRAs). Development of risk importance measures for phenomenological issues is also explored. Risk importance ranking of the various phenomenological issues can be useful for assessing any potential accident management strategies as well as for developing research priorities to reduce the overall state-of-knowledge uncertainty*

I. Introduction

Probabilistic accident progression and source term analyses (Level 2 PRAs) address the key phenomena and/or processes that can take place during the evolution of severe accidents, the response of containment to the expected loads and the transport of fission products from damaged core to the environment. Such analyses provide information about the probabilities of accidental radiological releases (source terms) to environment. The analyses also indicate the relative importance of events in terms of possibility of radiological releases which provide a basis for development of plant specific accident management strategies.

The phenomenology of severe accidents is extremely complex. The severe accident evaluation methodologies are associated with large uncertainties. Thus quantitative evaluation of uncertainties associated with the results of Level 2 PRA requires, among other things, knowledge of the uncertainties in the severe accident phenomenology. Such epistemic (state-of-knowledge) uncertainties are the major source of uncertainty in the results of Level 2 PRAs.

The NUREG-1150 study [1] was a major effort to put into a risk perspective the insights into system behavior and phenomenological aspects of severe accidents. An important characteristic of this study was the inclusion of the uncertainties in the calculations of core damage frequency and risk that exist because of incomplete understanding of reactor systems and severe accident phenomena. The elicitation of expert judgment was used to develop probability distributions for many accident progression, containment loading, structural response, and source term issues. Five specific commercial nuclear power plants were analyzed in NUREG-1150: Surry Power Stations, a 3-loop Westinghouse PWR with a subatmospheric containment; Zion, a 4-Loop Westinghouse PWR with large dry containment**;** Sequoyah, a 4-loop Westinghouse PWR with ice-condenser containment; Peach Bottom, a BWR-4 reactor with a Mark I containment; and Grand Gulf, a BWR-6 reactor with a Mark III containment.

In spite of plant specific nature of NUREG - 1150 quantitative results (e.g., core damage frequency, containment performance, and risk), this study provides valuable insights on severe accident phenomenological issues and associated

[1]The views expressed in this paper are solely those of the authors and do not necessarily represent those of either the ACRS or NRC

state-of-knowledge uncertainties which are very useful to the study of plants with similar NSSS and containment design.

This paper begins with a general overview of the PRA process with an emphasis on the accident progression and source term analysis (level 2 PRA). It then presents an assessment of the phenomenological uncertainties associated with Level 2 PRAs. This assessment is based on the results of NUREG-1150 study, integrated risk assessment for LaSalle[2], and the Individual Plant Examination (IPE) Insights Program [3] as well as the more recent technical knowledge and understanding of severe accident phenomena.

While a formal propagation of the uncertainties is the best way to account for model uncertainties, under certain circumstances, it can be demonstrated that the model uncertainties associated with some severe accident phenomenological issues may not be important to the overall risk. Development of plant specific importance measures for phenomenological issues will also be explored in this paper. Such risk importance measures for severe accident phenomena can be useful for assessing any potential severe accident management strategies, as well as for developing research priorities to reduce the overall model uncertainty.

II. Overview of the PRA process

A PRA of a nuclear power plant is a multidisciplinary process that quantifies the potential risk (with regard to the health and safety of the public) associated with accident sequences that are functions of the design, operation, and maintenance of the plant [4,5]. Figure 1 displays schematically the major components of the PRA analytical process which consists of the following key elements:

- Systems analysis and models of plant response to various initiating events, identification and quantification of sequences of events leading to core damage;
- Analysis of the accident progression and containment performance to determine various possible ways the accident could evolve given core damage;
- Source term analysis, the release of radioactive material to the environment for various outcomes of the accident progression; and
- Consequence analysis, the environmental transport and the health impacts of each of the source terms.

Integrated risk is obtained by combining the frequency of core damage, the conditional probability of the release paths, and the value of the consequences of each source terms into a single risk measure.

A complete PRA involves three sequential analytical parts or “levels” as shown in Figure 1. Level 1 PRA analysis identifies the specific combination of system or component failures (i.e. accident sequence cutsets) which can lead to core damage. The outcome of the analysis is a group of accident sequences leading to core damage and their associated frequencies.

The number of cutsets generated by a level 1 analysis is very large. It is neither practical nor necessary to assess the severe accident progression, containment response, and fission product release for each of these cutsets. As a result, the common practice is to group the level 1 cutsets into a sufficiently small number of plant damage states (PDSs) to allow a practical assessment of severe accident risks. A PDS is defined in such a way that all accident sequences associated with it can be treated identically in the accident progression and containment performance analysis. All the plant model information on the operability status of active systems that are important to the timing and magnitude of the release of radioactive materials are also passed into the leve2 analysis via the

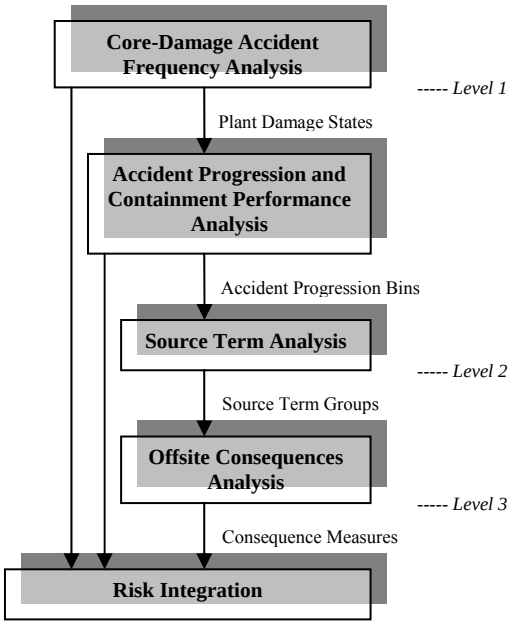

Figure 1. Elements of the PRA Analytical Process

definition of the PDS. The PDS definition should recognize all distinctions that matter in the level 2 analysis.

The accident progression and containment performance analysis considers the key phenomena and/or processes that can take place during the evolution of a severe accident and that can have an important effect on containment behavior. The accident progression event tree (known also as containment event tree) analysis generates conditional probabilities for a large number of end states (i.e., potential ways in which radioactivity could be released to environment). Some of these end states are either identical or similar, in terms of key radionuclide release characteristics. These end states are therefore, grouped to a smaller number of accident progression bins or release categories on the basis of appropriate attributes that affect radiological releases and potential offsite consequences.

The magnitude and composition of radioactive materials released to the environment and the associated energy content, time, release elevation, and duration of release are collectively termed the "source term." The source term analysis tracks the release and transport of the radioactive materials from the core, through the reactor coolant system (RCS), then to the containment and other buildings, and finally into the environment. The removal and retention of radioactive materials by natural processes, such as deposition on surfaces, and by engineered safety systems, such as sprays, are accounted for in each location. The source term analysis generates a set of source terms (source term groups) that have a one-to-one correspondence with each of release categories discussed above. These estimates of environmental source terms are needed to perform an offsite consequence assessment (Level 3 PRA).

The offsite consequences of accidental environmental release of radioactive materials can be expressed in several forms including impacts on health, the environment, or economics. The consequence measures of interest to a level 3 PRA generally focus on impacts on human health. These impacts are estimated both in societal terms and in terms of the most-exposed individual. Consequences to plant personnel are usually not included in a level 3 PRA.

The severe accident progression and the radionuclide source term analyses in the level 2 portion of the PRA, as well as the consequence analysis conducted in the level 3 portion of the PRA, are generally performed without regard to the absolute or relative frequency of the postulated accidents. An integrated risk assessment combines the results of the levels 1, 2, and 3 analyses to compute the selected measure of risk in a self-consistent and statistically rigorous manner. The uncertainty of the integrated risk can be carried out by assigning distributions to important variables (through a formal expert elicitation process, for example) and propagating them through the integrated risk analysis. The result is a distribution of risk for each of the selected consequence measures.

III. Phenomenological uncertainties in severe accident progression analysis

The evaluation of accident progression and the attendant challenges to containment integrity is an essential element of risk assessment. Containment performance plays an important role in the assessment of the risk associated with severe accidents. The primary concerns for containment performance are how well the containment can withstand the pressure and temperature loads associated with severe core damage accidents and whether or not the containment is bypassed. For scenarios in which containment integrity is maintained, fission product release will be small. For those scenarios leading to containment failure, fission product release depends on the timing as well as the size of the break in containment. Early containment failure can be important because it tends to result in shorter warning times for initiating public protective measures, and because radionuclide releases would generally be more severe than if the containment fails late. The mode of containment failure (i.e., gross failure versus leakages through failure of penetrations) also influences the amount of radioactive materials inside the containment that would be released to the environment.

The likelihood and severity of many challenges to containment integrity such as direct containment heating (DCH) and core-concrete interaction, which occur because of core-debris relocation to the reactor cavity, is highly dependent on the conditions inside the reactor vessel at the time of vessel breach. These conditions are determined by the accident sequence and the in-vessel meltdown progression of the core. The kind of information needed is the failure size/mechanism of the reactor pressure vessel (RPV) bottom head, the mass of melt available for release as well as its temperature and composition, the RCS pressure, and the amount of hydrogen in the RCS at the time of vessel breach.

Phenomenological issues surrounding the in-vessel core melt progression are highly complex. Included are the clad oxidation behavior, fuel and clad melting and relocation mechanisms, crust formation in the lower (colder) portions of the fuel bundles and thermal loading and failure of the crust. Significant research activity in the area of severe accidents has been undertaken following the accident at the Three Mile Island (TMI) Unit 2. Over the past twenty five years, major experimental and model development studies have been performed that provide an improved understanding of core melt progression phenomenology. A number of computer codes (e.g., MELCOR, SCDAP/RELAP) have been developed for analyzing various stages in severe accidents

including in-vessel melt progression. However, there are practical limits to the feasibility of deterministically modeling all aspect of core melt progression. Therefore, the use of expert judgment may become necessary for analysis of some stages of severe accidents. This may include the structure of the model as well as uncertainty concerning the quantification of the variables involved.

For the NUREG-1150 study, accident progression was analyzed using a single accident progression event tree developed for each plant, which was evaluated with the EVNTRE code [6]. The accident progression event trees, developed for this study, made extensive use of the available severe accident computer code calculations and experimental observations. The elicitation of expert judgment was used to develop probability distributions for fourteen accident progression, containment loading, and structural response issues shown in Table 1. Probability distributions for many other issues believed to be of less importance to risk were also developed by analysts on the project staff or by phenomenologists from national laboratories using techniques like those employed with the expert panels.

The key phenomena and /or processes that can take place during the evolution of a severe accident and that can have an important effect on containment behavior are plant specific. There are plant specific features and operational practices that may influence the likelihood and the severity of specific events or phenomena during the progression of severe accidents. For example, the results of individual plant examinations (IPEs) [3] indicate that specific containment features may lead to unique and significant failure modes. For instance, the probability values of early containment failures for both Palisades and Davis Besse were largely attributed to the special features of the particular designs of the plants. The location of the sump in Palisades causes the flow of molten core debris from the reactor cavity into this sump and subsequently into the ESF recirculation piping. In the IPE analysis, the debris was assumed to eventually melt through the pipe wall and enter the auxiliary building, resulting in a large containment failure area. For Davis Besse, one of the few PWR plants that have large dry containments of steel construction, the largest fraction of early containment failure was associated with the potential failure of the containment wall via direct contact with core debris.

Figure 2 presents a logic structure for how typical PDS attributes influence the severe accident phenomenological issues and containment failure mechanisms. The following sections provide a discussion of the state-of-knowledge uncertainties associated with different phenomenological issues and containment failure mechanisms.

III.A Reactor pressure vessel bottom head failure

A characteristic accident sequence leading to severe core damage would be one in which a combination of system failures results in loss of water from the reactor coolant system and in the failure of the emergency core cooling system to function properly. In such an event, the loss of coolant inventory would result in core uncovery with subsequent heatup and damage to the fuel rods. In the case of delayed operation or partial performance of the emergency core cooling system, core damage may be arrested as occurred in the TMI-2 accident. However, if coolant flow is not restored in time, complete meltdown of the reactor core and subsequent vessel breach (bottom head failure) could result. For a typical U.S. pressurized water reactor design, credit for in-vessel arresting of the accidents has been given for cases where water flow was restored within 30 minutes of the onset of core damage. If cooling is restored within 30

Table 1 Accident progression and containment performance issues presented to NUREG-1150 Expert Panels

In-Vessel Accident Progression Panel

Temperature-induced hot leg failure (PWRs)
Temperature-induced SGTR (PWRs)
In-vessel hydrogen generation in BWRs
In-vessel hydrogen generation in PWRs
Mode of temperature-induced bottom head failure in BWRs
Mode of temperature-induced bottom head failure in PWRs

Molten Core-Concrete Interaction Panel

Mark I drywell melt-through at Peach Bottom
Pedestal erosion from core-concrete interaction at Grand Gulf

Containment Loads Panel

Hydrogen combustion at Grand Gulf
Hydrogen combustion at Sequoyah
Hydrogen combustion in reactor building at peach Bottom
Loads at vessel breach at Grand Gulf
Loads at vessel breach at Sequoyah
Loads at vessel breach at Surry
Loads at vessel breach at Zion

Structural Response Panel

Static failure pressure and mode at Zion
Static failure pressure and mode at Surry
Static failure pressure and mode at Peach Bottom
Static failure pressure and mode at Sequoyah
Strength of reactor building at peach Bottom
Ice condenser failure due to detonation at Sequoyah
Drywell and wetwell failure due to detonations at Grand Gulf
Pedestal strength during concrete erosion at Grand Gulf

minutes, the probability of successful arrest of accident progression, without the vessel breach, was assumed to be 1.0. This time period was estimated based on core heatup characteristics and the potential for core coolability.

In order to characterize the rate of discharge of core debris from the reactor vessel it is necessary to know the mode of bottom head failure. Several types of failure modes have been suggested, ranging from the local failure of

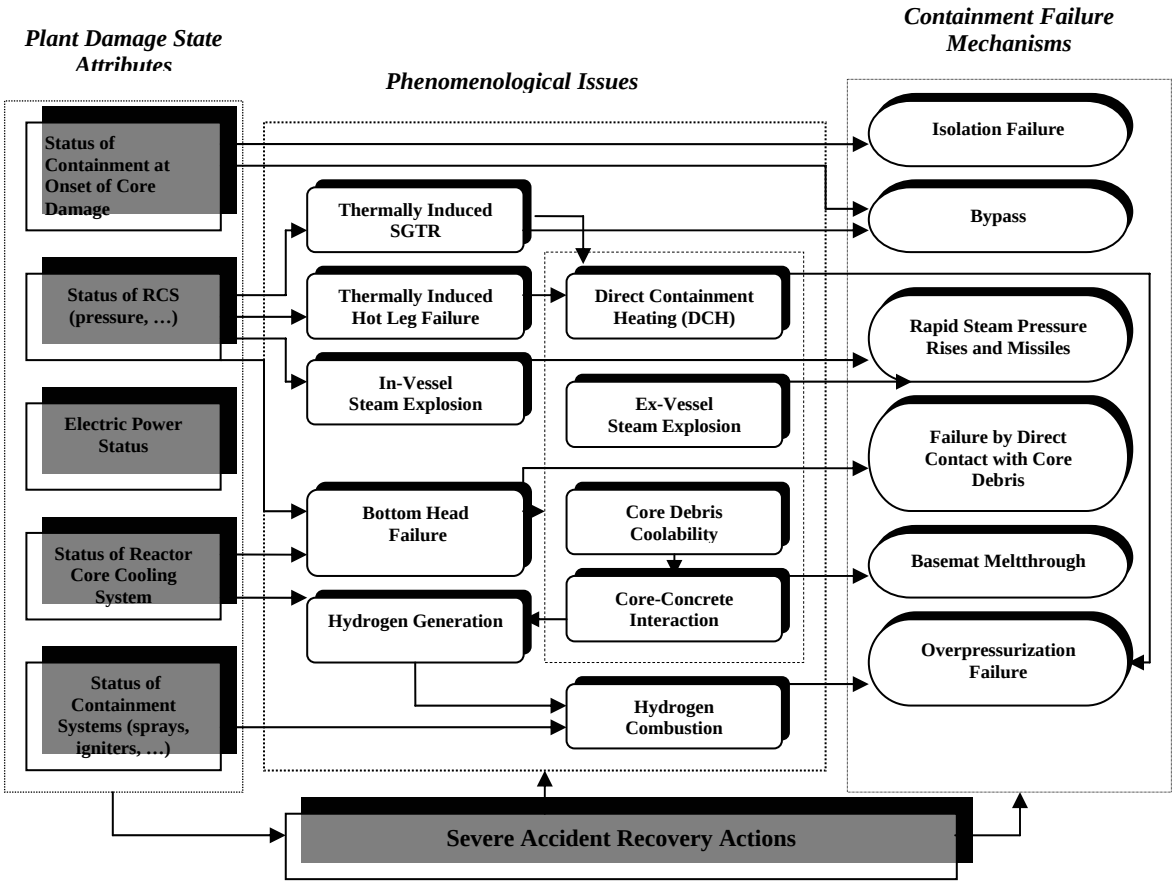

Figure 2. Logic structure of how typical PDS attributes influence the severe accident phenomenological issues and containment failure mechanisms

instrument tube penetrations, to the gross creep rupture of the entire lower head. The mode of bottom head failure depends on the nature of thermal and pressure loading that the lower head experiences, which in turn is dependent on the accident sequence and the characteristics (quantity, composition, temperature and timing) of the molten core material that arrives on the bottom head.

The issue of bottom head failure was assessed by the NURG-1150 in-vessel accident progression panel. The results of expert panel elicitation are reported in Ref. 7. Three mode of bottom head failure were considered: (1) high-pressure melt ejection (HPME) in which core debris is ejected, possibly through a penetration failure, as a liquid jet, driven by the gas pressure in the vessel; (2) gravity pour in which core debris flows under gravity; and (3) "Dump" in which bottom head failure occurs by creep rupture, allowing large quantities of debris to fall into the cavity/pedestal region of the containment. There was a wide difference of opinion for the mode of bottom head failure. For example, one expert was certain that the vessel failure in PWRs would always be a failure of a penetration, resulting in HPME. Only one expert thought that

a “dump” or gross failure of bottom head was possible.

Since the completion of NUREG-1150, more analytical and experimental studies [8, 9] have been performed to address the issue of lower head failure in light water reactors (LWRs). The recent lower head failure experimental results indicate that the vessel creep can result in failure of penetration welds before global rupture of the lower head could occur. Mode of lower head failure has also been addressed as a part of quantification of important variables toward resolution of DCH and “Mark I Liner Attack” issues (see Sections III.D and III.I).

Cooling the vessel from outside in order to retain the core debris inside the vessel has been used as a severe accident management measure. With the reactor intact and debris retained in the lower head, phenomena such as ex-vessel steam explosion and core-concrete interaction, which occur as a result of core debris relocation to the reactor cavity, could be prevented. This is the so-called severe accident management strategy of In-Vessel Melt Retention, which has been approved for the Loviisa plant in Finland and has been incorporated in the design of the AP600 and AP1000 passive plants. Reactor vessel integrity is assumed if RCS is depressurized and the cavity adequately flooded. The Risk Oriented Accident Analysis Methodology (ROOAM) [10] were used to quantify the margins to vessel failure for these designs [11, 12]. Accounting for the uncertainties in the in-vessel melt progression parameters, the heat fluxes to the vessel wall and reactor vessel internals from the debris pool were calculated. These heat fluxes were then compared to the critical heat flux limit for the downward-facing curved surfaces. Vessel failure was assumed if the critical heat flux was exceeded. These evaluations are based on scenarios for degradation of the core that avoids consideration of direct contact by metallic core debris with the reactor vessel. However, in view of the results of more recent experimental studies performed as a part of RASPLAV project and its follow-up MASCA program, this assumption may be questionable. The implications of these experimental results and the potential for molten pool stratification and relocation of metallic melt to the bottom of lower head on in-vessel retention of core debris should be further investigated.

III. B. Thermally induced failure of RCS

The severe accidents at high RCS pressure represent the most significant contributors to risk. The initial stages of core degradation involve coolant boiloff and core heatup in a steam environment. It has been argued [13] that at high pressure, naturally circulating steam can redistribute the core thermal load throughout the primary system leading to failure of the pressure boundary prior to the occurrence of large-scale core melt. The most vulnerable locations for such failure were identified as the hot leg and steam generator tubes. The location and the size of failure, however, remain uncertain.

The issue of temperature-induced PWR hot leg or surge line failure before vessel breach was assessed by the NURG-1150 in-vessel accident progression panel. Two cases were considered: (1) Classic TMLB′ sequence; and (2) Early induced pump seal loss-of-coolant accident (LOCA) with failure of auxiliary feed water. For case 1, the NUREG-1150 aggregated distribution indicates that there is approximately a 14% probability that an induced hot leg failure will never occur, and a 44% probability that hot leg failure always occur. For case 2, there is an 83% probability that hot leg failure will never occur.

The likelihood of a thermally induced creep rupture of steam generator tubes and associated containment bypass depends on several factors including the thermal-hydraulic conditions at various locations in the primary and secondary systems, which determines the temperature and pressure to which steam generator tubes are subjected as the accident progresses. Other

relevant factors include the effective temperature required for creep rupture failure of the steam generator tubes and the presence of defects in the steam generator tubes, which increases the likelihood of rupture.

NUREG-1150 in-vessel accident progression panel also considered the issues of temperature-induced steam generator tube rupture (SGTR) before vessel breach for the classic TMLB′ sequence. There was a wide difference of opinion for the uncertainty associated with temperature induced SGTR. Some panel members correlated the SGTR frequency distribution to the hot leg failure frequency distribution. One panel member argued that, due to the large time lag between temperatures in the hot leg and those in the steam generator tubes, the conditional probability of an SGTR is so small that it can be expressed as a constant value of 10^{-4}. The aggregated distribution for probability of SGTR, however, was correlated to the distribution of probability for hot leg failure.

In 2001, an Ad Hoc Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) concluded that the issue of possible evolution of severe accidents to involve gross failure of steam generator tubes and bypass of the containment has not yet been resolved [14]. The U.S. NRC is currently investigating this issue as a part of its Steam Generator Action Plan.

III.C. Steam explosion

During progression of severe accidents, molten debris from the damaged core would at some point begin to fall into the lower plenum of the reactor vessel. If an applicable amount of water remained in the lower plenum, molten core material falling into it could potentially cause a steam spike and if severe enough, an explosion. Rapid steam pressure rise and missile, resulting from in-vessel steam explosion (alpha mode failure) has been identified as a potential challenge to the containment in the past studies. However, a more recent assessment of this issue by an NRC sponsored steam explosion review group [15] concluded that alpha mode failure is of very low probability that is of little or no significance to the overall risk.

III.D Direct containment heating

The concern for DCH arises only if vessel breach occurs while the reactor is at elevated pressure. For such cases, the expulsion of molten core debris could lead to very rapid and efficient heat transfer to the containment atmosphere, possibly accompanied by oxidation reactions and hydrogen combustion that further enhance the energy transfer. The pressurization accompanying this process is referred to as direct containment heating (DCH).

The potential for DCH to cause containment failure depends on several factors, such as the primary system pressure, the size of opening in the vessel, the temperature and composition of the core debris exiting the vessel, the containment pressure and composition before the vessel breach, the amount of water in the cavity, and the dispersive characteristics of the reactor cavity.

The issue of DCH was assessed by the NURG-1150 containment loads panel as a part of load at vessel breach. The distributions for total pressure rise at vessel breach for various cases were developed [16]. These distributions were aggregated distributions from several experts, who based their judgments on experimental data, results of code calculations, and engineering judgment. The distributions were wide in part because of difficulty in extrapolating the limited data obtained at small scale with non-prototypical materials to reactor scale accidents.

Since the completion of NUREG-1150, advances have been made in understanding of the DCH phenomena. The U.S. Nuclear Regulatory Commission (NRC) has identified DCH as a major issue for resolution in its revised severe

accident research plan [17] and has sponsored analytical and experimental programs for understanding the key physical processes in DCH. A number of experiments have been performed in support of DCH issue resolution for PWRs. These experiments included both separate effect tests and integral effects tests, simulating the DCH processes in scaled models of the Zion, Surry, and Calvert Cliffs containments. The results of an assessment of the probability of containment failure by DCH for the Zion nuclear power plant were published in NUREG/CR-6075 and its supplement [18]. By using ROAAM, it was concluded that the containment failure probability by DCH at Zion is so low as to be considered physically unreasonable. The basic understanding upon which the approach to quantification of DCH loads is based is that intermediate compartments trap most of the debris dispersed from the reactor cavity and that the thermal-chemical interactions during this dispersal process are limited by the incoherence in the steam blowdown and melt entrainment processes. With this understanding, it was possible to reduce most of the complexity of DCH phenomena to a single parameter: the ratio of the melt-entrainment time constant to the blowdown time constant, which is referred to as the coherence ratio. Reference 19 provides further discussions on application of ROAAM to address the DCH issue for 34 Westinghouse plants with large volume containments. Reference 20 presents the estimates of the probability of containment failure by DCH in BWRs with Mark I containment.

As a part of a study to assess the risk significance of containment and related engineered safety features (ESF) system performance requirements [21], the accident progression event trees (APET) for Zion that had been used for NUREG-1150 was modified to reflect the more recent understanding of DCH phenomena in Zion. This also included incorporation of the containment fragility curve that was used in the NUREG-6075 study [18] and the Zion IPE [22]. The results of updated evaluation of the conditional probability of accident progression bins for internal events, as compared with the original results of NUREG-1150, are summarized in Figure 3. The conditional probability of early containment failure is very low for the Zion plant. This is due to the fact that the Zion containment capacity is high and the expected containment loads from the core melt accidents are not high enough to threaten the integrity of the containment during the early stage of an accident. In addition, a large fraction of the plant damage states are expected to result in low RCS pressure at the time of vessel breach, which lowers the potential for loads associated with HPME.

III.E Hydrogen generation

The issue regarding hydrogen generation centers on the rate and quantity of hydrogen production and the associated hydrogen-steam mass and energy release rates into the containment during both in-vessel and ex-vessel phases of severe accidents. These parameters strongly influence the flamibility of the containment atmosphere and the magnitude, timing, and location of potential hydrogen combustion.

The in-vessel hydrogen generation is primarily due to high-temperature steam oxidation of the zircaloy cladding surrounding the fuel rods. Key phenomenological uncertainties that influence hydrogen generation include flow blockage, by relocation of molten zircaloy and fuel, and natural circulation flow within the RCS.

In those accident scenarios in which the reactor vessel fails, high temperature core debris is likely to relocate into the reactor cavity/pedestal where it will interact with structural concrete and any water that may be present. The interaction of core debris with concrete in the cavity/pedestal region constitutes the ex-vessel phase of hydrogen generation. At high temperature, concrete decomposes. The ablation products commonly include water vapor and carbon

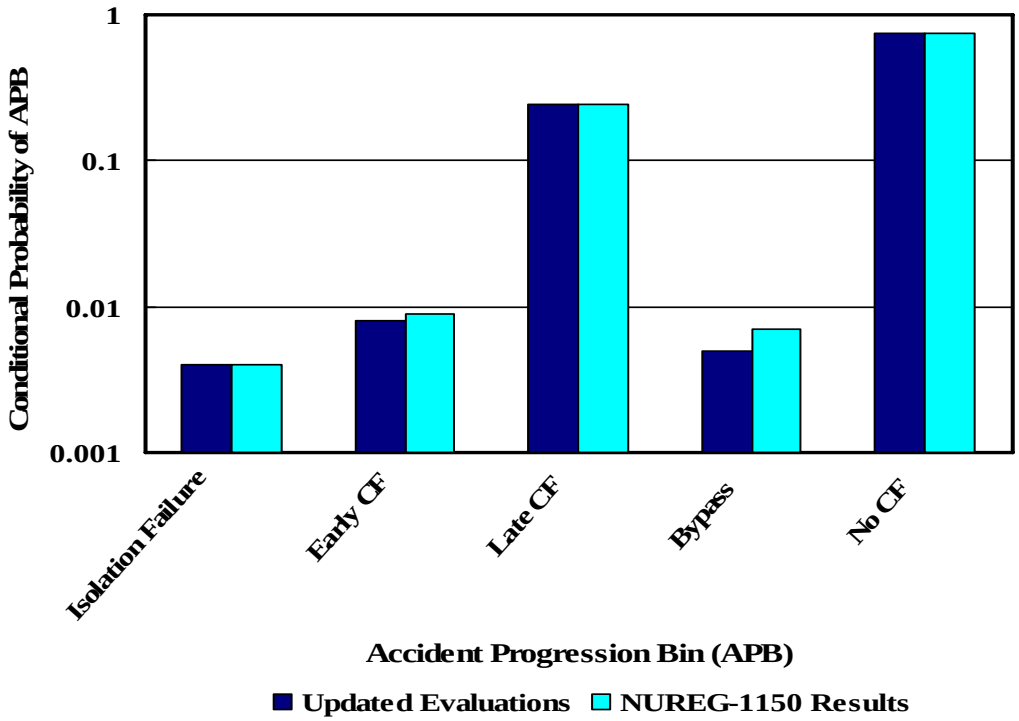

Figure 3. Summary results of updated conditional probability of accident progression bins for internal events at Zion, as compared with the original results of NUREG-1150

dioxide. As H_2O and CO_2 bubble through the debris pool, they may react with liquid metals to form H_2 and CO. Uncertainties affecting the ex-vessel hydrogen generation include the phenomenological uncertainties related to corium-concrete interaction and the debris coolability in the reactor cavity (see Section III. G). If coolability of core debris cannot be achieved, the in-vessel and ex-vessel metal oxidation could produce a quantity of hydrogen equivalent to more than a 100% clad oxidation due to additional oxidation of other metallic components in the core debris.

The in-vessel hydrogen generation issue was assessed by the NURG-1150 in-vessel accident progression panel. The results of expert panel elicitation for the various cases that was considered are reported in Ref. [7]. The distributions for the percent of hydrogen produced relative to that produced when all of the zirconium in the core is oxidized were developed. These distributions were aggregated distributions from several experts, who based their judgments on the results of code calculations and engineering judgment. The aggregate distributions are very wide. This is a result of widely differing opinion among the panel members considering this issue. The differences among experts appear to be as large as or larger than the differences between cases considered. For PWR cases, the median values of the aggregate distributions range from 30 to almost 50% zirconium oxidation.

III.F Hydrogen combustion

Hydrogen combustion in the containment building could produce pressure and thermal

loads that may threaten the integrity of the containment boundary.

The extent of hydrogen combustion is largely a function of the rate and magnitude of hydrogen generation and the nature of combustion of this hydrogen. Uncertainties associated with hydrogen loading arise from an incomplete state of understanding of various phenomena associated with hydrogen generation and combustion. These phenomena include in-vessel hydrogen generation, hydrogen transport and mixing, hydrogen deflagration, deflagration-to detonation transition (DDT), hydrogen detonation, and diffusion flames.

The degree of mixing and rate of transport of hydrogen in the containment building is an important factor in determining the mode of combustion. Hydrogen gas released during an accident can stratify, particularly in the absence of forced circulation and if there are significant temperature gradients in the containment. Hydrogen released with steam can also form locally high concentration in the presence of condensing surfaces. Should the hydrogen accumulate in a locally high concentration, then flame acceleration and detonation could occur. Hydrogen mixing and distribution in containment is sensitive to the hydrogen injection rate and the availability of forced circulation or induced turbulence in the containment. The result of large scale hydrogen combustion tests performed at the Nevada Test Site appear to qualitatively support the notion that operating the spray system will result in a well mixed atmosphere [23].

Deflagrations are the most likely mode of combustion during degraded core accidents. In fact, the deflagration of a premixed atmosphere of hydrogen-air-steam occurred during the Three Mile Island (TMI) Unit 2 accident. The likelihood and nature of deflagration in containments is strongly influenced by several parameters, namely, composition requirement for ignition, availability of ignition sources, completeness of burn, flame speed, and propagation between compartments. In addition combustion behavior is influenced by the effects of operating sprays [24].

The issues of hydrogen combustion at Grand Gulf (a BWR with a Mark III containment) and at Sequoyah (a PWR with ice-condenser containment) were presented to the NUREG-1150 containment loads panel [16]. For Grand Gulf, the experts were asked to provide the distributions characterizing the uncertainty in the probability that hydrogen combustion occurs in the containment. Given that hydrogen combustion occurs in the containment, the experts were further asked to provide distributions that characterize: the peak static pressure attained in the containment; the maximum impulse loading to the drywell wall; and the completeness of combustion. Level of hydrogen concentration were used to represent the continuous range of possible hydrogen concentrations.

III.G Debris bed coolability

Coolability of the core debris released into a flooded cavity/pedestal region of containment is an important issue. If the debris forms a coolable geometry, the only source for containment pressurization will be the generation of steam from boiloff of the overlaying water. Under these circumstances if containment heat removal systems are available, then late containment failure would be prevented. Even with the absence of containment heat removal, pressurization from water boiloff is a relatively slow process and would result in a very late containment failure allowing time for remedial actions.

There is a significant likelihood, that even if water is available, the core debris will not be coolable, and therefore will attack the structural concrete. Formation of coolable bed depends on many factors, such as mode of contact between the core debris and water, the size and

distribution of core debris particles, and the depth of debris bed.

III.H Core-concrete interaction

In the absence of a coolable configuration, the core debris will interact with the concrete structural materials of the cavity/pedestal region of the containment. The core-concrete interaction generates large quantities of noncondensible gases (including hydrogen and carbon monoxide) that contribute to pressurizing the containment atmosphere. Downward erosion of the basemat concrete may also lead to basemat penetration (depending on the thickness of the concrete) with the potential for ground water contamination and subsequent discharge of radionuclides to the surface environment. Also, thermal attack by the molten corium of retaining sidewalls could produce structural failure within the containment causing damage to vital systems and perhaps to failure of containment boundary.

III.I Containment failure by direct contact with core debris

Direct contact with the core debris is a significant containment challenge that can lead to early containment failure. Very briefly, the problem is concerned with the possibility that the molten core debris released from the reactor vessel will come in contact with the containment wall and cause a breach in it shortly after.

Drywell liner melt-through (caused by direct contact with core debris) has been found to be the most important contributor to early containment failure for Mark I containments. This failure mode is only possible for Mark I containments because the pedestal and drywell floor are at the same level, and core debris can easily reach the containment liner. The steel liner is the containment pressure boundary, and such a breach (i.e. drywell melt-through) would constitute an early containment failure.

The results IPEs [3] indicate that, during severe accidents at full power operation, the leading cause of early containment failure in some of the PWR ice condenser containments is direct impingement of core debris on the containment cylinder wall in the seal table room of the containment. In this scenario, core debris is swept out of the reactor cavity during high pressure melt ejection (HPME) and comes in contact with the containment pressure boundary in the seal table room.

The issue of Mark I drywell shell (liner) melt-through at Peach Bottom was assessed by the NURG-1150 molten core-containment interaction panel. The results of expert panel elicitation are reported in Ref. 16. There were two schools of thoughts on this issue. Some experts felt that melt spreading is a hydrodynamically limited phenomenon and that there is a high probability of drywell failure, even with the presence of water. Other experts felt the movement of the debris is thermodynamically limited and will be impeded by crust formation and the presence of water.

The experts provided subjective probability of drywell failure for several different scenarios, characterized by five parameters. These parameters were the RPV pressure, debris flow rate from the vessel, debris superheat, unoxidized metal content of the debris, and the presence of water on the drywell floor.

Since the completion of NUREG-1150, U.S. NRC has sponsored analytical and experimental programs to address and resolve this so-called "Mark I Liner Attack" issue. The results of an assessment of the probability of Mark I containment failure by melt attack of the liner were published in NUREG/CR-5423 [25] and NUREG/CR-6025 [26]. It was concluded that in the presence of water the probability of early containment failure by melt-attack of the liner is so low as to be considered physically unreasonable.

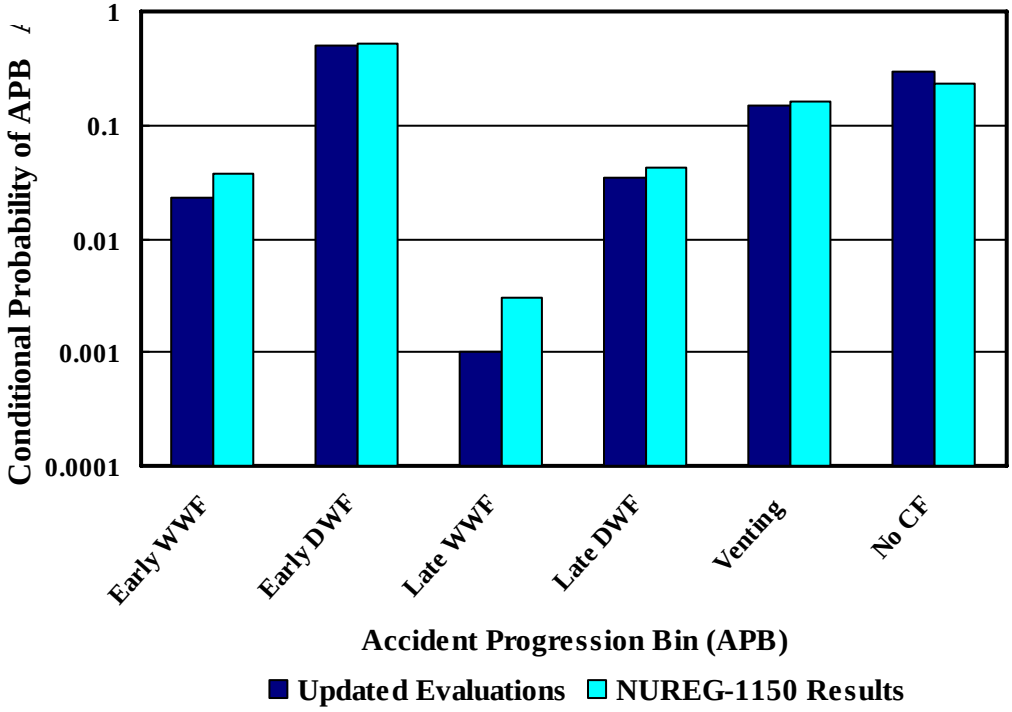

Figure 4. Summary results of updated conditional probability of accident progression bin for internal events at Peach Bottom, as compared with the original results of NUREG-1150

As a part of a study to assess the risk significance of containment and related ESF system performance requirements [21], the Peach Bottom APET that had been used for NUREG-1150 study was modified to reflect the more recent understanding of the drywell melt-through mechanism. The results of updated evaluation of the conditional probability of accident progression bins for internal events, as compared with the original results of NUREG-1150, are summarized in Figure 4. The results of updated evaluations indicate a decrease in the probability of containment failure but not as much as one might expect. This is because multiple failure modes are possible. Also in the ATWS and SBO accident groups (the risk dominated plant damage states), there is a significant probability that the vessel failure will occur when there is no water in the pedestal.

IV. Uncertainties associated with source term analysis

A realistic assessment of severe accident source term requires the detailed modeling of a wide range of phenomena associated with core melt progression, containment performance, and fission product release and transport. Figure 5 presents a logic structure for how the various phases of severe accident progression, containment failure mode and mechanism, and in-containment removal mechanisms influence the environmental source term.

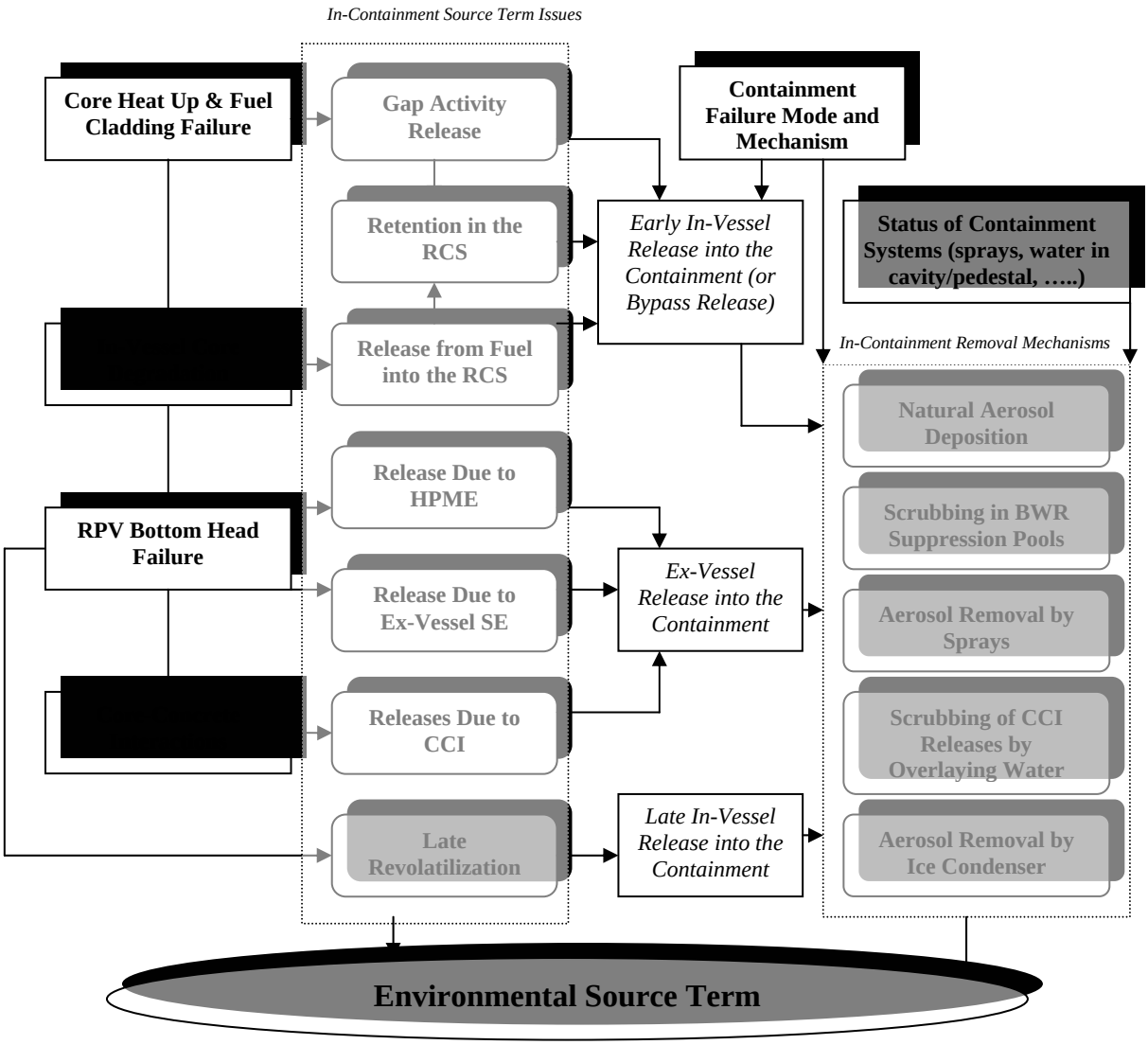

Figure 5. Logic Structure of the impact of severe accident progression phases, containment failure mode and mechanism, and in-containment removal mechanisms on environmental source term.

Radiological releases under severe accident conditions can be generally categorized in terms of phenomenological phases associated with the degree of core damage and degradation, reactor pressure vessel integrity, and, as applicable, attack upon concrete below the reactor cavity by molten core materials.

As it was noted earlier, the loss of coolant inventory during severe accidents would initially result in core uncovery with subsequent heatup and damage to the fuel rods. Upon failure of the cladding, a small quantity of fission products that resides in the gap between fuel pellets and the fuel cladding would be released. This

release, which is termed the gap release, would consist mostly of the volatile nuclides, particularly noble gases, iodine, and cesium.

As the accident progresses, core degradation begin, resulting in loss of fuel geometry accompanied by melting and relocation of core materials to the bottom of the reactor pressure vessel. During this period, the early in-vessel phase, significant quantities of the volatile nuclides in the core inventory as well as small fractions of the less volatile nuclides are released into containment.

The fission products and other materials, which are released from the fuel, are likely to be transported through the various portions of the RCS before reaching the containment (or environment if containment bypasses). As they move through the RCS, fission products may be retained as a result of various types of interactions.

If the bottom head of the reactor pressure vessel fails, two additional release phases (ex-vessel and late in-vessel) may occur. Following the bottom head failure, the molten core debris will be released from the reactor pressure vessel to the containment. Contact of molten core debris with the concrete structural materials of the cavity/pedestal below the reactor could lead to core-concrete interactions (see Section III.H). As a result of these interactions, significant quantities of the volatile radionuclides not already released during the early in-vessel phase as well as lesser quantities of non-volatile radionuclides are released into the containment atmosphere.

Two other phenomena that affect the ex-vessel release of fission products into containment could also occur. The first of these is HPME. If the RCS is at high pressure at the time of failure of the bottom head of the reactor pressure vessel, quantities of molten core materials could be ejected into the containment at high velocities. In addition to a potentially rapid rise in containment temperature (see Section III.D), a significant amount of radioactive material could also be added to the containment atmosphere, primarily in the form of aerosols. The occurrence of HPME is precluded at low RCS pressures. A second phenomenon that could affect the release of fission products into containment is a possible steam explosion (SE) as a result of interactions between molten core debris and water. This could lead to fine fragmentation of some portion of the molten core debris with an increase in the amount of airborne fission products. While small-scale steam explosions are considered quiet likely to occur, they will not result in significant increases in the airborne activity already within the containment. Large-scale steam explosions on the other hand, could result in significant increases in airborne activity, but are much less likely to occur. In any event, releases of particulates or vapors during steam explosions will also be accompanied by large amounts of water droplets, which would tend to quickly sweep released materials from the atmosphere.

Following the failure of the bottom head of the reactor pressure vessel, some of the volatile nuclides may be released into containment as a result of re-volatilization of the material, which had deposited on RCS structures during the in-vessel phase, or volatilization of material remaining in the reactor pressure vessel after vessel breach. This late in vessel release phase proceeds simultaneously with the occurrence of the ex-vessel phase. However, the late in-vessel releases have generally a longer duration than that of the ex-vessel releases.

The containment represents the final barrier between the fission products and the environment. Fission products released into the containment will be removed by a combination of mechanisms. The combination of removal mechanisms varies among different plant types and accident sequence conditions, but common to all is the aerosol removal that would occur as

Table 2 Source Term Issues Presented to NUREG-1150 Source Tem Panel

In-vessel fission product release and retention
Ice condenser decontamination factor (DF)
Revolatilization from RCS/RPV
Core-concrete interaction release
Release of RCS and CCI species from containment
Late source of iodine at Grand Gulf
Reactor Building DF at Peach Bottom
Release during direct containment heating

a result of natural transport and deposition processes, as well as aerosol removal by operating engineered safety systems. The effectiveness of these removal mechanisms, however, depends on the containment failure mode and mechanism (see Section III).

One of the major activities of NUREG-1150 study was the development of fission product source terms for a spectrum of accident conditions. The source terms were calculated using simplified parametric algorithm. The parametric equations describe the source terms as the product of release fractions and transmission factors at successive stages in the accident progression for a variety of release pathways, a variety of accident progressions, and nine classes of radionuclides. This approach led to development of separate computer codes for each plant, i.e., the XSOR codes [27]. The parametric models used in the XSOR codes are not time dependent. These codes generate source terms only in terms of early and delayed releases.

None of the basic parameters used in the XSOR codes are internally calculated. The values for the parameters must be specified by the user. The input data on the more important parameters, shown in Table 2, were constructed in the forms of probability distributions. Such distributions were developed using the elicitation of expert judgment to augment the analytical results to reflect the model uncertainties. For a few parameters that were judged of lesser importance or not considered as uncertain, single-valued estimates were used in XSOR models.

In 1995, the U.S. NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" [28], which defined a revised accident source term for regulatory applications. The early in-vessel, ex-vessel, and late in-vessel release fractions for the revised accident source terms were derived from the simplification of NURG-1150 source terms documented in NUREG/CR-5747 [29]. NUREG/CR-5747 utilized the current technical knowledge and understanding of the source term phenomenology to develop simplified formulation for realistic estimates of source term release into containment in terms of timing, nuclide types and quantity. The results of uncertainty analyses for releases into containment, utilizing the NUREG-1150 probability distributions for source term parameters, are also reported in Reference 29.

The following sections provide a discussion of the state-of-knowledge uncertainties associated with source term issues.

IV.A Release of fission products from the fuel into RCS before the vessel breach

There are large uncertainties associated with the data and fission product release models. The data taken from different sources span several orders of magnitude of release rates for the same fuel temperature [30]. The release of medium and low volatility fission products is even more uncertain. The variability of data may be most important for mitigated accidents, since the fuel could be maintained at temperatures that do not imply total release for relatively long periods of time. There are other factors such as fuel geometry (surface-to-volume ratio), system pressure, chemical potentials and flow velocities that can influence the release rates of fission products.

In addition to uncertainties in the data base for fission product release, there is a significant uncertainty in the prediction of the fuel temperature, melt relocation, and the extent of local oxidation during core degradation. Even if there were excellent fission product release models, the boundary conditions, and the parameters fed into these release models would be highly uncertain.

The issue of in-vessel release of fission products from fuel was assessed by the members of the NUREG-1150 source term expert panel. The results of expert panel elicitation are presented in Reference 31. Two experts concluded that there were no significant differences between PWRs and BWRs as far as release to RCS was concerned. The other two not only made distinction between BWRs and PWRs, but they also considered the extent of Zircaloy cladding oxidation, as a subcase, in quantifying the uncertainty distributions for the in-vessel releases. Tellurium release was given an additional dependency on Zircaloy cladding oxidation, based on the results of Oak Ridge National Laboratory tests indicating that the tellurium is sequestered in the tin in the Zircaloy cladding and it is not released until a high fraction of the cladding is oxidized. However, the most recent French tests (i.e., VERCORS, PHEBUS-FP), completed since the publication of NUREG-1150, indicate that the tellurium release could be similar to that of iodine release.

The NUREG -1150 aggregated distributions indicate that the fractional releases depend strongly on the volatility of fission products as might be expected. The range of release estimates for the volatile nuclides, such as the noble gases, iodine, cesium, and to some extent tellurium, spans about one order of magnitude. In contrast, the range for low volatile nuclides, such as barium, strontium, cerium and lanthanum, spans about 4 to 6 orders of magnitude.

IV.B Fission product retention in the RCS

The extent of retention in the RCS depends on the fission product chemical and physical form and the thermal hydraulic conditions along the flow path. The more volatile fission products would tend to enter the RCS as gases while the less volatile elements would tend to condense. The released fission product gases could absorb or condense onto aerosols and RCS surfaces, or react chemically with other species in the RCS atmosphere or with RCS surfaces. The amounts of fission products released into containment during the early in-vessel release phase are influenced by the residence time of the radioactive material within the RCS during core degradation. High-pressure sequences result in long residence times and significant retention of fission products within the RCS, while low-pressure sequences result in relatively short residence times and less retention within the RCS and consequently higher releases into containment.

Retention within the RCS was considered by the NUREG-1150 source term expert panel. The uncertainty distributions for fission product transmission within RCS for different RCS

pressure level and for each type of reactor (PWR vs. BWR) were developed [31].

The uncertainties in retention of fission products (except for noble gases) within the RCS are high. There is much uncertainty as to the kinetics and mechanics of interactions of volatile fission products within RCS gases and on solid structures. These uncertainties are compounded by uncertainties about aerosol agglomeration and deposition rates and chemical interactions of fission products on the RCS structural surfaces.

IV.C Releases into the containment due to CCI

The uncertainties in the ex-vessel release of fission products during CCI are high. There is much uncertainty in the concrete erosion processes and in the interaction between the molten pool and any water that may present in the reactor cavity or pedestal region during core concrete interaction. These uncertainties are compounded by uncertainties associated with the thermochemistry and kinetics of the vaporization of melt species into gases.

The issue of release of fission products during core-concrete interaction was assessed by the members of the NUREG-1150 source term expert panel. The experts were asked for the distributions that they believe would characterize the uncertainty in the fraction of each radionuclide group that is released from the molten fuel during CCI. For each type of reactor (PWR vs. BWR), different cases were considered based on the type of concrete, the amount of Zr in the molten core, and the presence or absence of water in the reactor cavity or pedestal region during CCI.

IV.D Late revolatilization release from RCS

Revaporization of radionuclides retained in the RCS and their subsequent release into the containment has a potential impact on the severe accident source term. The importance of this issue is the timing of radionuclide revaporization. The additional activity associated with the revaporized fission products is not expected to dramatically increase the overall releases except for some BWR accident sequences since the revolatilization release may not pass through the suppression pool. The impact of radionuclide revaporization is more noticeable and significant if the timing of revaporization is delayed. If fission products revaporize early, (shortly after vessel breach) the revaporized material will enter the containment and could interact with a concentrated source of aerosols generated by CCI (and enter the suppression pool for BWR cases). The result is some mitigation of revaporization process.

Fission product revaporization is affected by post-vessel-failure thermal hydraulics, reactor coolant system heat transfer, and the chemical form of retained radionuclides. Extensive RCS retension during the in-vessel phase of the accident, high temperature of RCS structures, and high flow inside the RCS after vessel failure (to carry vaporized fission products to the containment) are prerequisites to fission product revaporization.

Fission product releases into the containment by late revolatilization from the RCS were considered by the NUREG-1150 source term expert panel. Only three groups: iodine, cesium, and tellurium were considered for the late revolatilization release. The uncertainty distributions for fraction of the each group of fission product retained in the RCS (at the time of vessel breach) which is released to containment at later time were obtained. The uncertainty distributions were developed for each type of reactor, for the number of large openings in the RCS after vessel breach (for PWRs), and for the level (high vs. low) of drywell temperature (for BWRs). For PWRs, the revolatilization is higher if there are two RCS breaches (e.g., in the vessel and in the hot leg) due to establishment of natural circulation between the RCS and the containment.

IV. E In-containment removal mechanisms

Combination of mechanisms that remove fission products from the containment atmosphere with consequent mitigation of source term varies among different plant types and accident sequence conditions, but common to all is the removal that would occur because of natural transport and deposition processes, as well as removal by operating engineered safety features (ESFs). The systems that remove fission products include containment sprays, BWR suppression pools, and ice condenser.

In NUREG-1150 study, uncertainty distributions for decontamination factors for containment sprays, suppression pools, and water overlaying corium during CCI were developed. The effectiveness of these removal mechanisms, as evaluated in NUREG-1150, is discussed in Reference 32.

The effect of containment sprays decontamination factors on overall risk estimates for the NURG-1150 plants was found to be not significant, because for these plants, risk is dominated by accident sequences in which sprays are not operational (i.e., station blackout accidents)or which result in containment failure modes that cause sprays to be less effective (i.e., containment bypass) [21].

The effectiveness of the suppression pool as a fission product cleanup system is largely determined by the amount of bypass flow. Suppression pool bypass can significantly influence overall plant risk. The effect of suppression pool decontamination factors on overall risk is small, since risk tends to be dominated by accident sequences that result in early containment failure and subsequent suppression pool bypass [21].

V. Risk importance measures for phenomenological issues

It is desirable to assign some ranking of "risk importance" among the various phenomenological issues that are considered in a plant PRA model. The purpose of such ranking is to arrange these issues in the order of importance to the overall risk of a particular plant. Risk importance ranking of the various phenomenological issues can be useful for assessing any potential accident management strategies as well as for developing research priorities to reduce the overall state-of-knowledge uncertainty.

The importance measures that have been frequently used for ranking PRA basic events or structures, systems and components (SSCs) with respect to their risk significance [33] include:

Risk Achievement Worth, RAW

$$RAW_i = R_i^+ / R_0$$

Fussell – Vesely, FV

$$FV_i = (R_0 - R_i^-) / R_0$$

Risk Reduction Worth, RRW

$$RRW_i = R_i^- / R_0$$

where:

R_i^+ = overall risk with the probability of basic event i set to 1 (the event has occurred or the equipment is failed),

R_i^- = overall risk with the probability of basic event i set to 0 (the event is impossible or the equipment is totally reliable), and

R_0 = overall base-case risk

Similar importance measures can also be defined for various phenomenological issues [34]. For

example, a risk importance measure of "Risk Significance Worth", somewhat similar to Risk Achievement Worth (RAW) used for risk importance ranking of various plant components, can be defined as:

$$\text{Risk Significance Worth} = R\ (\text{issue i})^{+}/R_0$$

Where R (issue i)$^{+}$ is the calculated risk, using the bounding (conservative) assumptions in quantifying the phenomenological issue i, and R_0 is the base-case risk. Other risk importance measures somewhat similar to Fussel-Vessly (F-V) or Risk Reduction Worth (RRW) can also be defined. It should be noted that various risk metrics could be used for defining these risk importance measures. For example, Figures 6 and 7 show the large release frequency (LRF) importance measures for post core-damage mitigation systems and for occurrence of severe accident phenomena that has been addressed in the containment event tree (CET) used in AP1000 PRA [35].

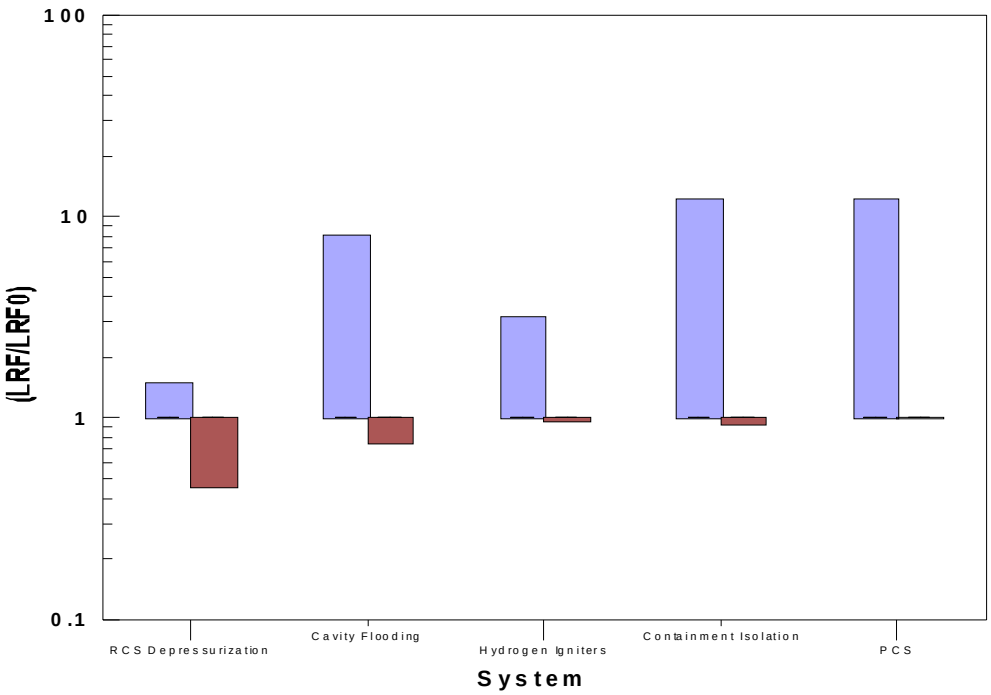

Figure 6. Risk (LRF) importance measures for post core-damage mitigation systems in AP1000 Design

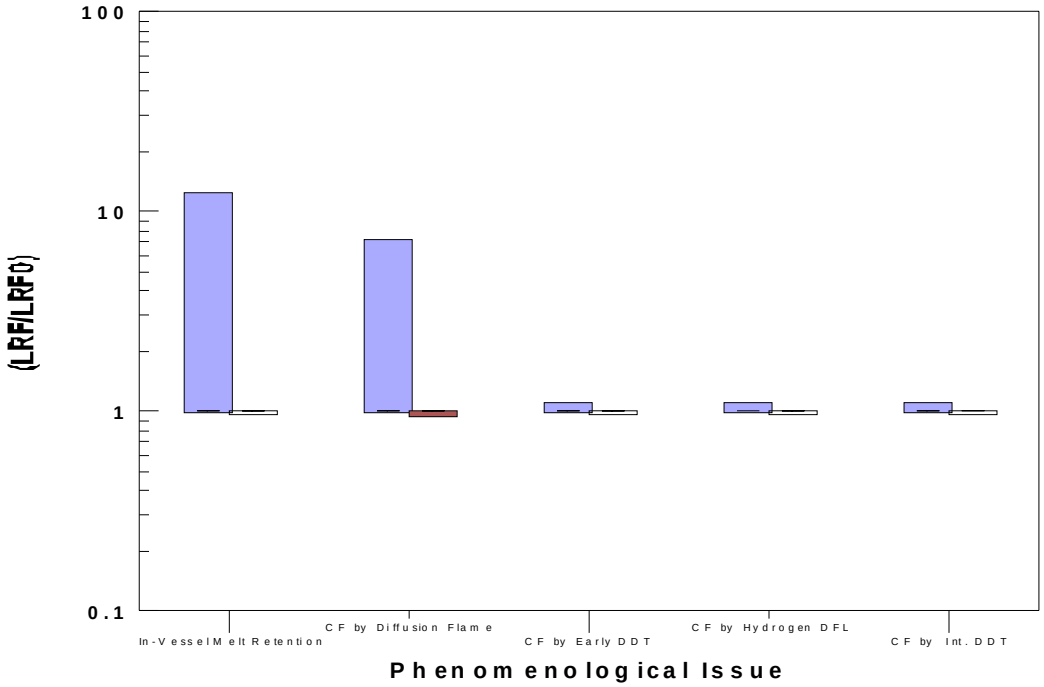

Figure 7. Risk (LRF) importance measures for level 2 phenomena in AP1000 Design

VI. Summary and conclusions

In this paper, a general overview of the PRA process with an emphasis on the accident progression and source term analysis (level 2 PRA) was presented. An assessment of the phenomenological uncertainties associated with level 2 PRA was also presented. Development of risk importance measures for phenomenological issues is also explored. This assessment is based on the results of NUREG-1150 study and the Individual Plant Examination (IPE) Insights Program as well as the more recent technical knowledge and understanding of severe accident phenomena.

Development of risk importance measures for severe accident phenomenological issues was also discussed. Risk importance ranking of the various phenomenological issues can be useful for assessing any potential accident management strategies as well as for developing research priorities to reduce the overall state-of-knowledge uncertainty.

VII. References

1. U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

2. BROWN, T.D., et al., "Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant," Sandia National Laboratories, NUREG/CR-5305, Vol. 1, August 1992.

3. U.S. Nuclear Regulatory Commission, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," NUREG-1560, November 1996.

4. U.S. Nuclear Regulatory Commission, "Probabilistic Risk assessment (PRA) Reference Document," NUREG-1050, September 1984.

5. U.S. Nuclear Regulatory Commission, "Severe Accident Risks for VVER Reactors: The Kalinin PRA Program," Volume 3: Procedure Guides, NUREG/CR-6572, Vol. 3 Part 1, September 1999.

6. GRIESMEYER, J. M. and L. N. SMITH, "A Reference Manual for the Event Progression Analysis Code (EVNTRE)," NUREG/CR-5174, SAND88-1607, September 1989.

7. HARPER, F. T., et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Expert Opinion Elicitation on In-Vessel Issues," Sandia National Laboratories, NUREG-CR-4551, SAND86-1309, Vol. 2, Rev. 1, Part 1, December 1990.

8. REMPE, J. L., et al., "Light Water Reactor Lower Head Failure Analysis," NUREG/CR-5642, 1993.

9. CHU, T. Y., et al., "Lower Head Failure Experiments and Analysis," Sandia National Laboratories, NUREG/CR-5582, 1999.

10 THEOFANOUS, T. G., "Dealing with Phenomenological uncertainty in Risk Analysis," *Proceeding of Workshop I in Advanced Topics in Risk and Reliability Analysis – Model Uncertainty: Its Characterization and Quantification,* Annapolis, Maryland October 20-22, 1993, NUREG/CP-0138, 1994.

11. TUOMISTO, H. and T. G. THEOFANOUS, "A Consistent Approach to Severe Accident Management," *Nuclear Engineering and Design,* **148**, p. 171-183, 1994.

12. THEOFANOUS, T. G., et al., "In-Vessel Coolability and Retention of Core Melt," U. S. Department of Energy, DOE-ID-10460, July 1995.

13. NOURBAKHSH, H.P., e. al., "Natural Circulation Phenomena and Primary System Failure in Station Blackout Accidents," *Proceedings of the Sixth Information Exchange Meeting in Debris Coolability*, EPRI-NP-4455, March 1986.

14. U.S. Nuclear Regulatory Commission, "Voltage-Based Alternative Repair Criteria," A report to the Advisory Committee on Reactor Safeguards by the Ad Hoc Subcommittee on a Differing Professional Opinion, NURG-1740, March 2001.

15. BASU, S. and T. GINSBERG, "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of Broader Fuel-Coolant Interaction Issues," U.S. Nuclear Regulatory Commission, NUREG-1524, August 1996.

16. HARPER, F. T., et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Experts' Determination of Containment Loads and Molten Core Containment Interaction Issues," Sandia National Laboratories, NUREG-CR-4551, SAND86-1309, Vol. 2, Rev. 1, Part 2, April 1991.

17. U.S. Nuclear Regulatory Commission, "Severe Accident Research Program Plan Update," NUREG-1365, Rev. 1, December 1992.

18. PILCH, M. M., H. YAN, and T.G. THEOFANOUS, "The Probability of Containment Failure by Direct Containment Heating in Zion," NUREG/CR-6075, SAND93-1535, December 1994.

19. PILCH, M. M., et al., "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containment or Subatmospheric Containment," U.S. Nuclear Regulatory Commission, NUREG/CR-6338, February 1996.

20. ESMAILI, H., et al., "The probability of High Pressure Melt Ejection-induced Direct Containment Heating Failure in Boiling Water Reactors with Mark I Design," *Proceeding of the 11th International Topical meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11),* Avignon, France, October 2-6, 2005.

21 NOURBAKHSH, H., et al., "Risk Importance of Containment and Related ESF System Performance Requirements," Brookhaven National Laboratory, NUREG/CR-6418, BNL-NUREG-52489, November 1998.

22. Zion IPE, Commonwealth-Edison Zion Station: Individual Plant Evaluation, NRC Doc. 9204290315, April 1992.

23. THOMSON, R. T., "Large-Scale Hydrogen Combustion Experiments, Volume 1: Methodology and Results," EPRI NP-3878, Electric Power Research Institute, October 1988.

24. NOURBAKHSH, H., et. al., "Effectiveness of Containment Sprays in Containment Management," Brookhaven National Laboratory, NUREG/CR-5982, BNL-NUREG-52354, May 1993.

25. THEOFANOUS, T.G., et al., "The Probability of Liner Failure in a Mark I Containment," NUREG/CR-5423, 1989.

26. THEOFANOUS, T.G., et al., "The Probability of Mark I Failure by Melt –Attack of the Liner," NUREG/CR-6025, November 1993.

27. JOW, H.N., W. B. MURFIN, and J. D. JOHNSON, "XSOR Codes User's Manual," Sandia National Laboratories, NUREG/CR-5360, December 1993.

28. SOFFER, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

29. NOURBAKHSH, H. P., "Estimate of Radionulide Release Characteristics into Containment under Severe Accident Conditions." NUREG/CR-5747, BNL-NUREG-52289, Brookhaven National Laboratory, November 1993.

30. ALLISON, C., et al., "Severe Core Damage and Associated In-Vessel Fission Product Release," *Progress in Nuclear Energy,* Vol **20**, No. 2, pp. 89-132 (1987).

31. HARPER, F. T., et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Experts' Determination of Source Term Issues," Sandia National Laboratories, NUREG-CR-4551, SAND86-1309, Vol. 2, Rev. 1, Part 4, 1991.

32. NOURBAKHSH, H.P., and S.E. PEREZ, "Assessment of In-Containment Aerosol Removal Mechanisms," Brookhaven National Laboratory, Technical Report, L-1535, December 1992.

33. CHEOK, M.C., G.W. PARRY, and R.R. SHERRY, "Use of Importance Measures in Risk-Informed Regulatory Applications," *Reliability Engineering and System Safety*, Vol. **60**, pp 213-226, 1998.

34. NOURBAKHSH, H. P., "Assessment of Model Uncertainties In Regulatory Decisions Involving PRA Results," *International Topical Meeting on Probabilistic Safety Analysis, PSA'05,* San Francisco, California, September 11-15, 2005.

35. Westinghouse Electric Company, "AP1000 Probabilistic Risk Assessment," APP-GW-GL-022, Revision 0, March 2002.