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Executive Summary

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ABSTRACT

This document summarizes the Quantitative Uncertainty Estimate of the Source Term (QUEST) program. QUEST determined the magnitudes and natures of uncertainties associated with recent, NRC-sponsored, predictions of radioactive material releases (BMI-2104) for the Surry TMLB', Surry S_2D, and Grand Gulf TC accidents. Radioactive "source term" uncertainties arising from input uncertainties to the suite of computer codes used in BMI-2104 were estimated as were uncertainties due to poorly modeled or omitted phenomena in the code suite. Specific input or modeling uncertainties that most impact final source term uncertainties were identified through systematic sensitivity studies of the codes and by developing and/or applying improved models for those judged to be inadequate in the BMI-2104 suite. Methods used to define, combine, and propagate uncertainties are discussed at length in the detailed reports and are summarized herein.

QUEST demonstrated rather large uncertainties in current source term predictions, and concluded that source term ranges rather than specific values must be considered in severe accident analysis. However, even with uncertainties, the "envelope" of source terms in many cases falls well below the release values quoted in previous reports such as the Reactor Safety Study, WASH 1400. Releases reaching and sometimes exceeding WASH 1400 predictions, however, are shown to be possible within the range of uncertainties if early loss of containment integrity occurs. Even with uncertainties, at least a factor of ten decrease in source terms (and some cases much more) can be expected if containment integrity is maintained for a few (~10) hours after core melt. One important QUEST finding is that the volatile fission-products such as cesium and iodine which seem to receive the most attention in safety analyses may be important for only a short time after core melt because radionuclides such as tellurium, strontium, barium, lanthanum, etc., dominate the source term at late times. Finally the phenomenological uncertainties contributing most to source term uncertainties were identified in QUEST, thus providing a basis for establishing research priorities to reduce source term uncertainties.

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ACKNOWLEDGMENTS

This document is but a summary of the vast amount of work done in the Quantitative Uncertainty Estimation for the Source Term (QUEST) program. The technical analyses that led to this summary are found in Volumes II, III, and IV of the QUEST reports. Authors of these works are D. R. Bradley, J. E. Brockmann, J. M. Griesemeyer, R. J. Lipinski, P. K. Mast, K. K. Murata, D. A. Powers, J. B. Rivard, J. Tills, and D. C. Williams. The contribution to the conception and promotion of the QUEST program by M. J. Jankowski of the U.S. Nuclear Regulatory Commission is also acknowledged.

1. INTRODUCTION

Release of radioactive materials constitutes the principal threat to the public health and safety posed by severe nuclear power plant accidents. Evaluation of this threat requires knowing many of the features of the release from the plant: the mass, the chemical composition, and the physical characteristics of the released materials as well as the timing and the duration of the release process. These features of the release are called, collectively, the severe-accident "source term." The source term cannot be described now with great accuracy. The magnitudes and natures of uncertainties in current characterizations of the severe-accident source term are the subjects of this document.

Data have become abundant and models of the behavior of radioactive materials have improved dramatically since publication of source-term descriptions in the Reactor Safety Study (RSS).[1] The U.S. Nuclear Regulatory Commission has recently sponsored a reassessment of severe-accident source terms using these improved models and data. This reassessment effort was conducted by the Battelle Columbus Laboratory (BCL) and the results are described in a series of reports designated BMI-2104.[2] Because of the many improvements in the technology, the BCL study did not resort to the bounding, generic treatment of accident source terms adopted in the Reactor Safety Study. Hallmarks of the new source-term estimates are the mechanistic detail used to produce them and the sensitivity of the estimates to particular features of the accidents and the power plants in question.

Review of the results provided in BMI-2104 reinforced the perception that the most effective use of source-term estimates from reassessment calculations could be made if the source-term uncertainties were identified and estimates of their magnitudes were obtained. The Severe Accident Uncertainty Analysis (SAUNA) research program at Sandia National Laboratories, Albuquerque (SNLA) addressed the general identification problem by providing a list of phenomena for which inherent uncertainties would propagate to the source term.[3] To respond to the need for quantification, SNLA's QUEST (Quantitative Uncertainty Estimation for the Source Term) study was undertaken, with the specific objective of providing an estimate of the plausible range of uncertainty in certain selected source terms as calculated in the BCL study.

The scope of the QUEST study was limited. Only three combinations of plant and accident sequence were considered:

- The TMLB' accident sequence in the Surry plant. (Detailed results of this study are presented in Volume II of this report.)
- The S_2D accident sequence in the Surry plant. (Detailed results of this study are presented in Volume III of this report.)
- The TC accident sequence in the Grand Gulf plant. (Detailed results of this study are presented in Volume IV of this report.)

This volume provides an executive summary of the QUEST study. The major sections include summaries of the methodology and results, a discussion of the major findings, and a final summary. The methodology section identifies generic sources of uncertainties and describes a means of propagating the various uncertainties to arrive at a final estimate of the source-term uncertainty. The results section summarizes the findings of the three plant/accident-sequence analyses and identifies dominant uncertainties. The discussion section discusses how the results of the QUEST study affect the understanding and use of "best estimate" source-term calculations. The final section provides a short summary.

2. QUEST METHODOLOGY

2.1 Overview of Accident Source-Term Determination

A severe nuclear-reactor accident can progress through many stages that can contribute to and shape the radiological source term. These stages can include core heatup, core degradation, and vessel failure; in-vessel fission-product release from the fuel; in-vessel fission-product retention on primary system surfaces; release of molten core materials and fission products from the failed vessel; ex-vessel fission-product release and aerosol formation; fission-product retention in the containment by aerosol settling and phoresis mechanisms; scrubbing of aerosol-laden gases in a BWR suppression pool; and, finally, release of fission products to the environment through a failed containment. The suite of codes used in the BMI-2104 study models many of the features of the accident progression described above. Figure 1 displays the relationships among the various codes used in estimating the radiological source term. A description and assessment of each of these codes may be found in Reference 4.

A hypothesized severe fuel-damaging accident begins with a loss of coolant, leading to a gradual heatup of the core as the core-water inventory decreases. At high temperatures, the fission-product decay power is augmented by the highly exothermic oxidation of cladding in the hot steam environment. As the accident progresses, the core melts and slumps, failure of internal core structures such as grid plates occurs, and eventually molten (or nearly molten) core materials drop to the bottom of the reactor vessel. There, the heating of the vessel, coupled with the weight of the slumped core materials, can result in vessel failure and expulsion of core materials into the reactor cavity.

In both the BMI-2104 and QUEST studies, the MARCH 2.0 code was used to model these in-vessel phenomena. MARCH 2.0 handles the overall accident thermal hydraulics and calculates the rate of water removal from the reactor coolant system (RCS), the core temperature and clad-oxidation rates during the meltdown process, the time of vessel breach, and the temperatures of the molten materials released from the vessel.

During the in-vessel core-degradation process, fission products and structural materials vaporize at high temperatures and form aerosols upon condensation. The ORIGEN code is used to determine the fission-product inventory in the core as a function of previous operating history. The fuel temperatures calculated by MARCH, along with the fission-product

The CORCON code calculates temperature and gas-flow rates through the melt during core/concrete interactions using initial melt mass, composition, and temperatures determined by MARCH. The VANESA model calculates the aerosol-release rate during core/concrete interactions, using time-dependent melt temperatures and gas-flow rates from CORCON.

In a BWR, the aerosols released from the reactor vessel and the aerosols produced ex-vessel may flow through a suppression pool before entering the containment building. Scrubbing of aerosol-laden gases by the suppression pool can result in a substantial decrease in the amount of fission products that can escape to the environment. The SPARC code was used in both the BCL and QUEST studies to calculate the retention of fission products in the suppression pool.

The aerosols released from the reactor vessel and the aerosols produced ex-vessel (that make it through the suppression pool in a BWR) mix in the containment atmosphere, agglomerate, and can deposit onto containment surfaces. Containment sprays can sweep aerosols and fission-product gases from the containment atmosphere. In the BMI-2104 study, NAUA-4 was used to treat both the agglomeration and settling of aerosols within the containment building and the removal of aerosols by containment spray. NAUA-4 uses aerosol sources from CORSOR (after attenuation by TRAP-MELT) and VANESA, and steam-condensation rates from MARCH. Both the BMI-2104 and QUEST studies used NAUA-4 in their assessment of containment phenomena. QUEST also used the newer CONTAIN code, which models aerosol phenomena and containment thermal-hydraulics in a coupled, self-consistent manner.

The net result of the processes described above is a time-dependent amount of suspended radioactivity in the containment atmosphere. How much of this amount is actually released to the environment was shown in QUEST to depend strongly on the timing and character of containment failure. Catastrophic containment failure results in rapid containment depressurization and release of a large fraction of the suspended aerosols and gases. Localized containment failure with a small breach results in slow depressurization and release of only a small fraction of the suspended materials, because of the added time for further settling of aerosols inside containment.

After containment failure, the amounts of radioactive material released to the environment depend upon the amounts that are generated from melt/concrete interactions, that are resuspended by gas flow or by water entrainment from flashing pools of water, and that are revaporized by continued

heating of primary system surfaces. The cumulative release to the environment following containment failure constitutes the radiological source term from the severe accident.

2.2 Components of the Source-Term Uncertainty

There will always be uncertainties in the estimates of radioactive source terms arising from hypothetical severe accidents. Models describing complex sequences of events in a severe accident rely on imprecisely known material properties. Also, models often make use of parametric representations and user-supplied values for the parameters to describe phenomena that are not mechanistically modeled. Such parameters may take on any value within an acceptable range of values. Thus, two equally competent and responsible code users would not necessarily formulate identical code-input sets to analyze the same problem.

In addition to this code-input uncertainty, there is the uncertainty with respect to both accuracy and completeness that is associated with the representation of phenomena in any set of codes. For example, the release-rate coefficients for CORSOR are calculated using experimental correlations obtained from data relating fission-product release to fuel temperature. Scatter in the data creates uncertainties in the correlation expressions. Further, the correlations ignore the effects on release of pressure, coolant composition (oxidizing or reducing), coolant thermal hydraulics, and of other factors thought to be important to the thermodynamics and kinetics of fission-product release.

The existence of this phenomena uncertainty in the BMI-2104 suite of codes is one of the reasons newer codes to model severe-accident phenomena are being developed. Some of the important NRC-sponsored efforts to develop improved codes include SCDAP, MELPROG, CONTAIN, and MELCOR.[5-8]

The code-input and phenomena uncertainties are components of the source-term uncertainty. Uncertainties in specific accident features are ultimately manifested in the source term uncertainty by propagation. Uncertainty introduced into the analysis in a previous stage will be propagated through the present stage and amplified or attenuated by the physical processes considered. For example, uncertainties in fuel-temperature history produced by MARCH can yield very large uncertainties in fission-product release from CORSOR.

The QUEST study considered the two components of source-term uncertainty:

- The code-input component refers to the source-term uncertainty that can be attributed to variations in the inputs to the computer codes used in BMI-2104.

- The phenomena component refers to the source-term uncertainty that can be attributed to uncertainties in known phenomena and to the omission of important phenomena in the BMI-2104 code suite.

Currently, phenomena uncertainty is the dominant component of the overall source-term uncertainty, simply because of the large number of phenomena neglected or only approximately modeled in the BMI-2104 code suite. In the future, with improvements in the computer codes used to analyze severe reactor accidents, it should be possible to reduce phenomena uncertainties by utilizing the insights gained from studies such as QUEST.

2.3 Method for Determining Source-Term Uncertainty

The determination of both code-input uncertainty and phenomena uncertainty involves a two-stage process. First, the important uncertainties are identified and the ranges over which they might be expected to vary are determined. Second, these important uncertainties are combined and propagated through the suite of codes to determine their effects on the amount of suspended or released radionuclides. The method for doing this depends upon the nature of the uncertainty. For those instances where the uncertainty can be addressed directly through code-input parameters in the codes, simply varying those code-input parameters will solve the problem. For those cases where phenomena uncertainties exist, but these uncertainties are not addressable by code-input parameters, simple code modifications can be made to add new input parameters. In the event that important phenomena are missing from the suite of codes, there are three choices: the phenomena can be added to the appropriate code, a more advanced code that already includes the phenomena can be used, or extensive separate-effects calculations can be performed. All of these approaches were used in QUEST.

2.3.1 Determination of Important Uncertainties and Their Ranges

Only the most important uncertainties, i.e., those that strongly influence the suspended radionuclides and the release of suspended material from containment, were considered in QUEST. To identify those, it was necessary first to identify potentially important uncertainties; second, to specify the ranges over which they might vary; and then, based on the results of sensitivity studies for these ranges, to determine which uncertainties contributed most to the uncertainty in the overall source term.

Compiling a list of potentially important uncertainties involves reviewing literature on accident progression,

examining experimental results for unexpected effects, and considering alternative models for accident behavior. Fortunately, much of this work had been carried out in a recent SNLA research program that provided a review and listing of phenomena uncertainties in severe accidents.[3]

Determination of appropriate ranges for uncertain inputs and parameters is a critical task in any uncertainty analysis and one that occupied much of the time devoted to the QUEST program. Meaningful results are obtained in an uncertainty analysis only if variations in parameters and inputs are over realistic ranges and do not violate rules of consistency or physical possibilities. An essential ground rule for the QUEST program was that ranges of uncertainty be technically defensible and not just the product of speculation. These defensible ranges were defined for QUEST in two ways:

- When data were available, the uncertainty ranges were defined from scatter in the data.
- When no data were available, the uncertainty ranges were derived from predictions of alternative models.

When alternative models were necessary to define uncertainty ranges, these models were developed in many cases as part of the QUEST effort. Some important models developed in the program were

- A model of resuspension in the containment
- A model of fission-product entrainment during flash boiling of water pools
- A model of aerosol formation during pressurized melt ejection
- A model of turbulent energy dissipation in the reactor containment
- A model of the evolution of aerosol shape factors as particles grow by agglomeration

In all cases, extreme limits of the uncertainty ranges for parameters were avoided. Representative uncertainty ranges were sought. An illustration of avoiding extreme uncertainty ranges is provided by the selection of release-rate coefficients for iodine during core degradation. A representative cross section of available data for iodine release is shown in Figure 2. Data obtained in out-of-pile experiments were used to define the original CORSOR model release-rate coefficients and the factor of ten uncertainty range in the coefficients adopted in QUEST. The recent and much lower release-rate data from in-pile release coefficients could be used to

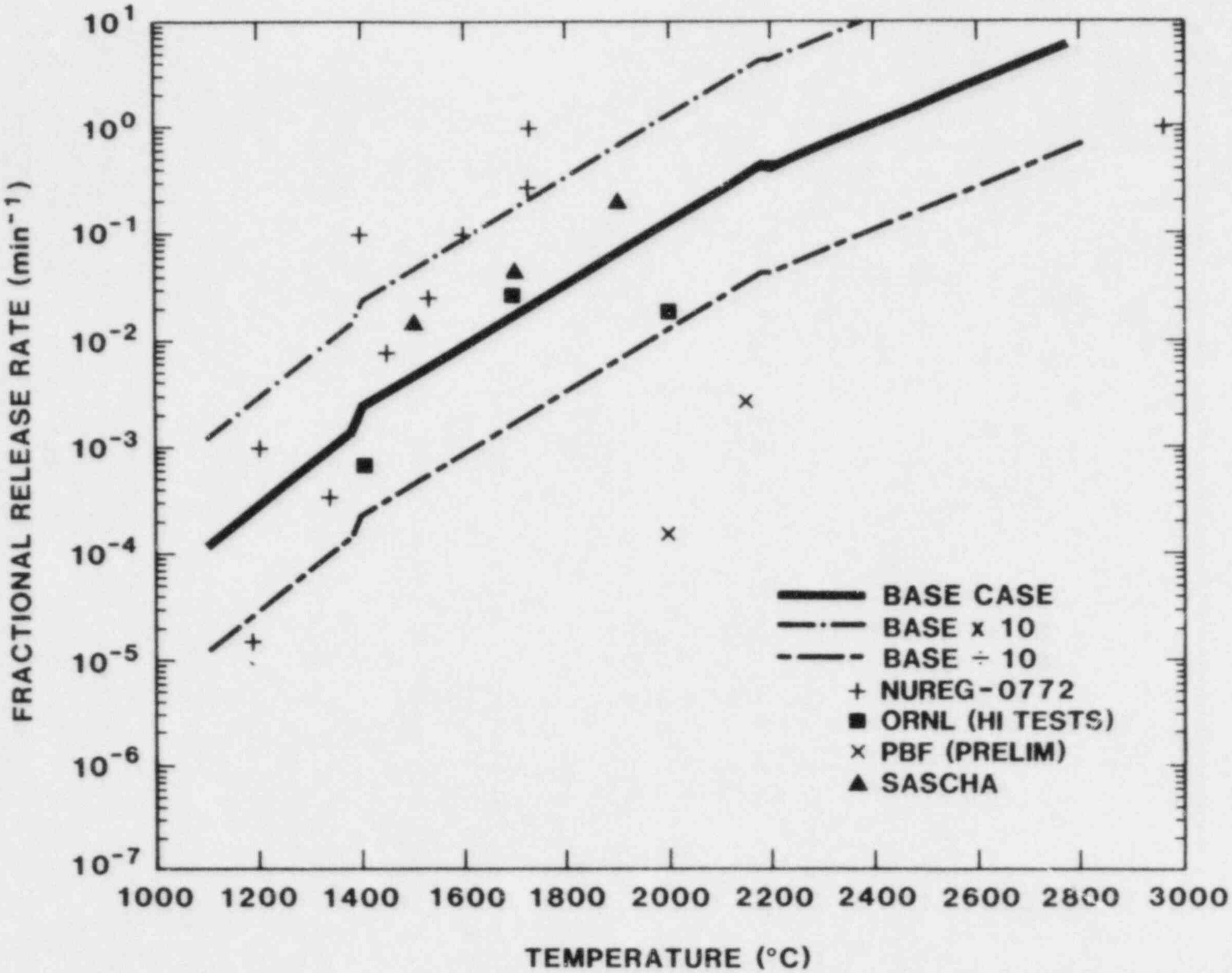

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Figure 2. Iodine In-Vessel Release Rates Used in QUEST

define an uncertainty range very much broader than that used in QUEST.

Having identified potentially important uncertainties and their ranges, the determination of the more important ones was made by carrying out sensitivity studies for each phase of the accident in reverse chronological order. That is, the phenomena and parameters that strongly affected the suspended aerosol concentration in the containment building were determined first. These were input parameters to and models in the NAUA and CONTAIN codes, which were used for the containment analyses. Next, the phenomena and parameters that strongly affected the input parameters to CONTAIN and NAUA were identified. These were input parameters to and models in the VANESA code. This reversed progression was continued until chains of parameters and phenomena that strongly affected the suspended aerosol concentrations in containment were identified.

Code sensitivity studies aid understanding of tools available for estimating accident source terms and are essential for quantitative evaluation of uncertainty ranges for source-term estimates. Volumes II and IV of the QUEST program documentation provide, in some cases for the first time, detailed sensitivity studies of the computer codes

- MARCH
- CORCON/VANESA
- SPARC
- NAUA-4
- CONTAIN

2.3.2 Combining and Propagating Uncertainties

Three methods of combining and propagating code-input and phenomena uncertainties could be used in various stages of an uncertainty study. These methods were all used to a varying degree in QUEST. First, a single input parameter was varied about a given base case (specifically, a BMI-2104 case), and the effect of the variation on the estimated radiological source term was examined. This approach gave a measure of the marginal importance of each parameter. At the end of this process, however, it was difficult to combine the individual parameter uncertainties into an overall uncertainty estimate. This approach will fail to detect synergistic effects that arise when two or more parameters vary simultaneously. Therefore, QUEST made little use of this approach.

Second, sensitivity studies were carried out for the component codes of the suite under consideration. This approach indicated which parameters were important to each code. In addition, the influence of these parameters on the source term was estimated by linking the sensitivity studies of the various codes. Again, the resultant uncertainties could not be readily combined into an overall uncertainty statement. Nonetheless, this approach was useful, because the sensitivity studies helped to define important components of uncertainty.

In the third approach, a specific value was selected from within the uncertainty range for each input parameter and phenomenon. The resulting sets of values denoted consistent, plausible alternative descriptions of the accident sequence. The sets of values could then be used to calculate alternative source-term estimates. These estimates then provide an indication of the magnitude and disposition of the uncertainty range for the source term. This is the approach used in QUEST for final uncertainty predictions.

2.4 Choice of Calculations for Determining Source-Term Range

To estimate the source-term uncertainty using the third method described above, sets of values for input parameters for the various codes could be selected for full calculations with the code suite, with each set yielding a specific radiological source term. A large number of such calculations would be required to investigate completely the source-term uncertainty band. This would be costly, time-consuming, and well beyond the scope of the QUEST study.

An alternative approach is to consider only the most influential parameters and carefully choose code-input sets that, within reasonable bounds, yield high or low values for the source term. Providing that each input set is inspected to ensure that no inconsistent assumptions exist and that input parameters are within technically justifiable ranges of uncertainty, then the high and low source terms calculated provide an estimate of the source-term uncertainty range. This was the approach used in QUEST.

It should be emphasized that this approach did not provide the absolute highest and lowest bounds on source-term uncertainty. Even though the parameter sets were chosen with the intention of calculating source terms near the bounds of the source-term envelope, these bounding cases do not reflect the combination of a series of highly unlikely model/parameter alternatives. Extreme ranges for the various parameter ranges were avoided; reasonable and technically defensible ranges were used in all cases. Thus, the calculated high and low source terms should be considered as part of a "best-estimate" envelope rather than as low-probability extrema.

For each of the three plant/accident-sequence combinations considered, a high and a low source term were calculated. Because of its dependence on containment failure time and mode, the source term (for the TMLB' and S_2D sequences) was characterized by the time-dependent masses of various radionuclides suspended in containment. The question of containment failure was considered separately. In many sequences, the possibility of containment failure is a maximum at early (near the time of vessel failure) and late times (due to gradual containment pressurization). Thus, typically, four characteristic source terms were calculated, with parameter ranges within the uncertainty bounds:

- Early-High: Chosen to lead to high suspended radioactivity shortly after vessel failure
- Early-Low: Chosen to lead to low suspended radioactivity shortly after vessel failure
- Late-High: Chosen to lead to high suspended radioactivity late in time
- Late-Low: Chosen to lead to low suspended radioactivity late in time

These four typical source terms yield, obviously, an estimate of the magnitude of the uncertainty in the severe-accident source term. They also provide an indication of how the range of uncertainty is disposed around nominal cases such as those considered in the BCL study (BMI-2104). The disposition of the uncertainty band around the nominal case can be as important as the magnitude of the uncertainty. Quite clearly an uncertainty band centered around the nominal case can lead to conclusions very different than would follow from an uncertainty band that ranges to predominantly lower values than the nominal case. Ascertaining the position of the uncertainty band relative to the "point" results obtained in the BCL study was viewed as an essential task in QUEST.

3. SUMMARY OF RESULTS

An assessment of the uncertainty in the source term was performed for the TMLB' and S_2D sequences in Surry and for the TC sequence in Grand Gulf. As in the BMI-2104 assessment of these accidents, "hands-off" sequences with no operator intervention were assumed. Complete code-input uncertainty calculations and phenomena uncertainty calculations were performed for the Surry TMLB' sequence. Phenomena uncertainty calculations were performed for the other two sequences. This section describes the phenomena uncertainty calculations performed for each sequence and summarizes the source-term uncertainty ranges found. A discussion of these results, emphasizing important insights and conclusions, is provided in Section 4.

3.1 Surry TMLB' Analysis

A detailed assessment of both code-input and phenomena uncertainties was performed for Surry TMLB' using the BMI-2104 code suite, the CONTAIN code, and various separate-effects calculations. The code-input-uncertainty calculations and their results are described in Volume II of this report. For the phenomena-uncertainty assessment, the BMI-2104 base case was recalculated, along with four other cases that were designed to estimate the magnitude and the shape of the source-term uncertainty range.

3.1.1 TMLB' Cases Considered

Table 1 gives the general conditions for the various stages of a TMLB' accident that would be expected to result in high or low source terms at early or late time. An early-high source term would result from a high release of fission products from the fuel within the calculated uncertainty band for in-vessel fuel release, combined with conditions that permit little aerosol retention within the RCS (such as might occur if the physical characteristics of the aerosols led to little retention in-vessel, or if there were aerosol resuspension either from reheating of surfaces or from gas flow at vessel failure). Extensive aerosol generation during high-pressure melt ejection would augment this high in-vessel release to containment. For the early-high case, ex-vessel aerosol production and retention are not as important to the source term as are the in-vessel processes.

The conditions expected to result in an early-low source term are the opposites of those expected to result in an early-high source term, except for melt/concrete aerosol production.

Table 1

Conditions* Expected to Result in High or Low Source Terms at Early or Late Time for Surry TMLB' Sequence

Accident Phenomena	Early-High	Early-Low	Late-High	Late-Low
In-vessel fission-product production	Large	Small	Small	Large
Net fission-product retention in RCS	Small	Large	Small	Large
High-pressure melt ejection	Large	Small	Small	Large
Melt-concrete aerosol production	Small	Small	Large	Large
Aerosol retention in containment	Small	Large	Small	Large

*Parameters and inputs were chosen so that conditions prevailing during the calculations were as indicated in the table.

Because the inventory of volatile and refractory radionuclides in the melt is large, aerosol production from the melt/concrete interaction is potentially important.

Several combinations of conditions could result in a high source term at late time. One such combination is shown in Table 1. Temperatures at the low end of their uncertainty range in the MARCH calculations and release-rate coefficients at the low end of their uncertainty range in CORSOR would result in low in-vessel release. If the in-vessel fission-product release is low, the fission products might be released later, during the core/concrete interactions. Those fission products released late would have less time to settle out in the containment building. If the ex-vessel aerosol characteristics lead to little settling, the aerosols will remain suspended in containment for a long time.

The conditions that might lead to a late-low source term are, for the most part, the opposites of those that might lead to a high source term at late time. Thus, the high in-vessel fission-product release with maximum in-vessel retention would lead to low releasable inventory for late-time ex-vessel release. Those fission products that did escape from the vessel would have considerable time to settle out in the containment.

The various parameters and their values that actually defined the four cases summarized in Table 1 were chosen based on the results of separate code-sensitivity studies. These sensitivity studies, along with a more detailed description of the four TMLB' cases, may be found in Volume II of this report. The technical justifications of the code-input values chosen are discussed at length in Volume II.

3.1.2 TMLB' Results Summary: Quantitative Aspects

For the four cases, CONTAIN was used to calculate the time dependence of the aerosol masses suspended in containment of CsI, CsOH, tellurium, refractory fission products (RFP), and inert materials, as well as the total time-dependent suspended aerosol radioactivity. These results are shown in Figure 3. Note that masses of suspended CsI and CsOH are large at early times, but settle quickly. Tellurium and RFP dominate the inventory of suspended radioactivity at later times. The total time-dependent suspended aerosol radioactivity is shown in Figure 4.

Inspection of these results shows that the range in suspended radioactivity at early time, approximately a factor of 50, is dominated by uncertainties in the suspended CsI and CsOH at the time of vessel failure. The upper value for suspended radioactivity at that time is 1.2 GCi, which includes contributions from RFP, cesium, iodine, and tellurium. The peak drops off quickly because of the enhanced aerosol agglomeration and settling caused by the large mass of inert aerosols. However, if the time of containment failure coincides with the pressure spike at vessel failure and a large hole is created, then a large fraction (close to 1.0) of this peak suspended radioactivity could be released from containment, depending on the failure mode.

The suspended aerosol radioactivity at late time varies by about a factor of 150. The uncertainties at this time are dominated by uncertainties in RFP and tellurium. These materials are suspended in the containment as a result of core-debris/concrete interactions. In each case, the suspended aerosol radioactivity decreases by a factor of ten within four hours after vessel failure, indicating the importance of ensuring early containment integrity. Most of the uncertainty range found in this analysis results in calculated source terms that are lower than those in the BMI-2104 base case.

3.1.3 TMLB' Results Summary: Qualitative Aspects

The Surry TMLB' analysis provided the most comprehensive sensitivity study of the codes in the BMI-2104 code suite and the most detailed assessment of source-term uncertainty.

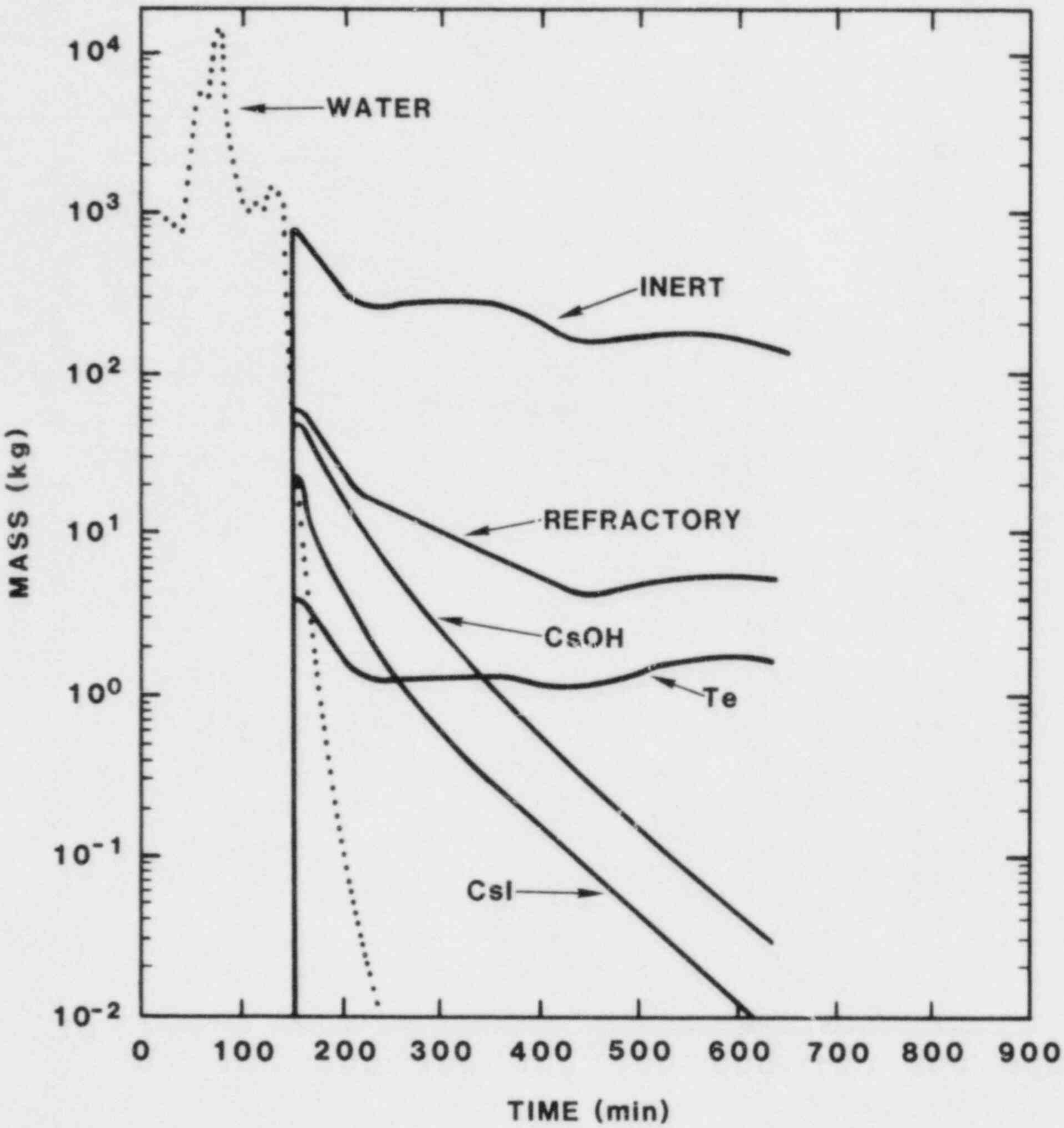

Figure 3. Suspended Aer... l Masses versus Time, Surry TMLB'

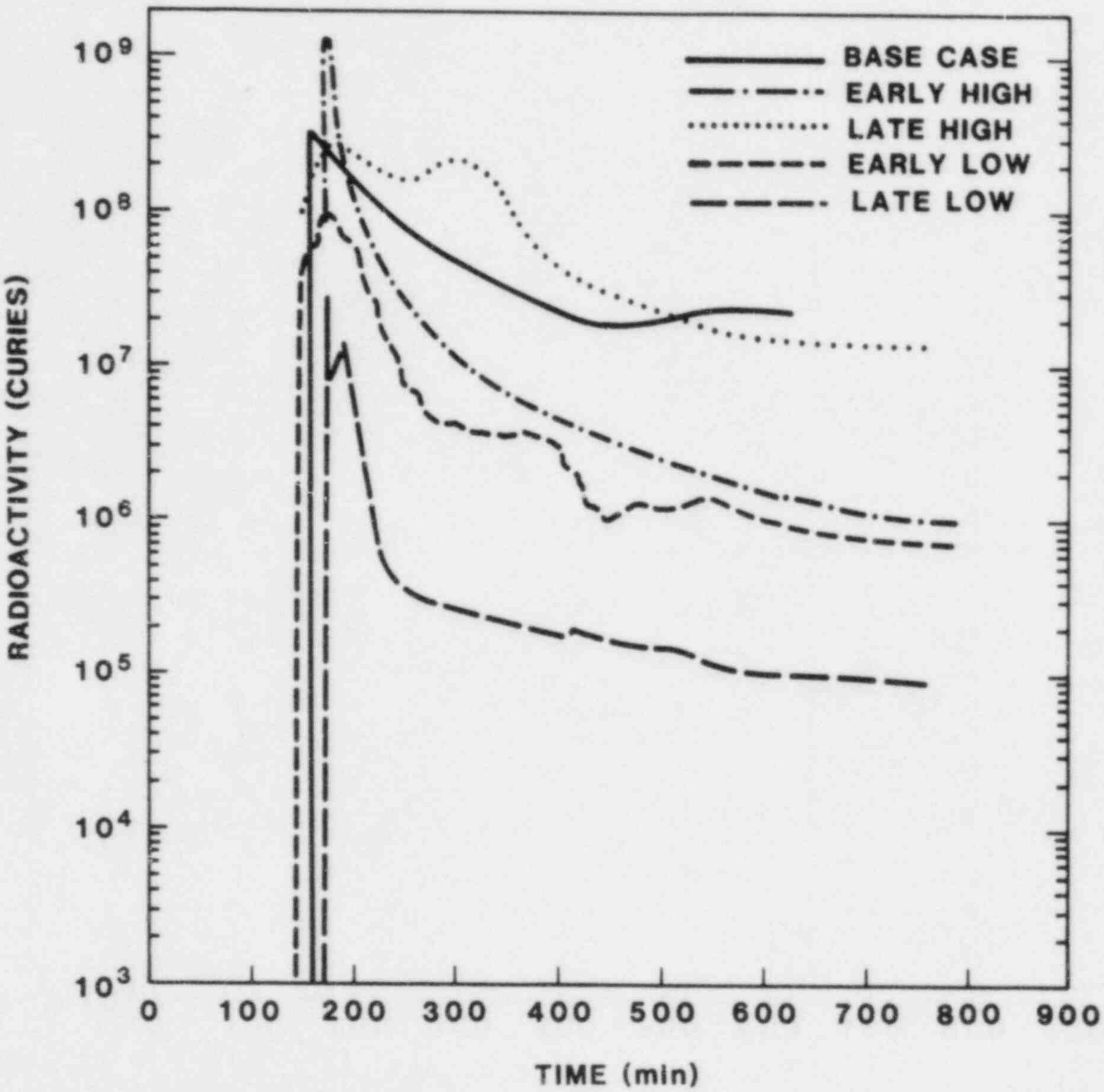

Figure 4. Total Time-Dependent Suspended Aerosol Radioactivity, Surry TMLB'

Results of this analysis indicate that the source-term uncertainty near the time of a vessel failure is dominated by in-vessel processes. Particularly important uncertainties include

- Core-temperature histories
- Release rates from fuel for given temperature and gas-flow conditions
- Natural circulation within the RCS
- Retention of fission products within the RCS (including effects of revaporization and resuspension)

The source-term uncertainty at late time (about 15 hours into the accident) was dominated primarily by ex-vessel processes. Particularly important uncertainties are

- Debris temperature and gas generation during melt/concrete interactions
- Aerosol shape factors in low-humidity conditions
- Turbulent agglomeration rates

Though the in-vessel processes do not affect directly the suspended mass in the containment atmosphere late in an accident, the extensive sensitivity studies done in QUEST show that the in-vessel processes do affect the nature of ex-vessel phenomena. For instance, the extent of zirconium oxidation in-vessel establishes an important initial condition for core-debris/concrete interactions and consequently affects the magnitude of aerosol production during these interactions. Similarly, the in-vessel processes can determine whether the containment atmosphere is steam-saturated. This can affect the shapes of aerosol particles and consequently the rate of aerosol settling in containment.

3.2 Surry S_2D Analysis

A detailed assessment of the source-term phenomena uncertainty was performed for the Surry S_2D sequence, using the BMI-2014 code suite, the CONTAIN code, and various scoping calculations. The BMI-2014 base case was recalculated, along with three other cases that were designed to provide an estimate of the shape and magnitude of the source-term uncertainty range. Unlike the TMLB' analysis, only one low case was considered. Early-low and late-low cases were determined to result from similar sets of assumptions in an S_2D accident sequence.

3.2.1 S_2D Cases Considered

Three self-consistent source terms were calculated: a high source term at early time (just after vessel failure), a high source term at late time (nominally 15 hours into the accident), and a low source term, which turned out to be low for both early and late containment-failure times.

Table 2 gives the general conditions for the various stages of an accident that would be expected to result in high or low source terms at early or late time. For the early-high case, high core temperatures (as calculated by MARCH) and CORSOR release-rate coefficients at the high end of their justifiable uncertainty ranges are used. In addition, fission-product retention in the RCS is assumed to be at the low end of its uncertainty range. These conditions result in a high release of radionuclides to containment at the time of vessel failure.

Table 3-2

Conditions Expected to Result in High or Low Source Terms at Early or Late Time for Surry TMLB' Sequence

Accident Phenomena	Early-High	Late-High	Low
In-vessel fission-product production	Large	Small	Small
Net fission-product retention in RCS	Small	Small	Large
Melt-concrete aerosol production	Large	Large	None
Aerosol retention in containment	Small	Small	Large

The time of vessel failure is also the time when the containment experiences a pressure spike that can threaten containment integrity. This pressure spike could result, for example, from one or more of the following processes: the quenching of ejected molten materials in either water or the containment atmosphere, hydrogen burns ignited by the hot melt, and release of high-pressure steam from the vessel. (Uncertainties in such containment-loading phenomena were not addressed in QUEST.) Thus, the early-high case is of particular interest for the S_2D sequence in the Surry plant.

The ex-vessel aerosol production and retention are not as important to the early-high case as are in-vessel phenomena. Nonetheless, their effect on how much material is available to escape a failed containment is not negligible. Thus, conditions from CORCON that lead to high ex-vessel aerosol production are used, as are conditions leading to low retention in the containment. Therefore, inputs that lead to higher melt temperatures are used in the CORCON calculation. In addition, it is assumed that there is no water pool above the molten pool (the melt-ejection process and small steam explosions are assumed to blow out the water). Low ex-vessel retention occurs by assuming that the sprays fail when containment fails from overpressurization. Because spray failure is linked to containment failure, the early-high case is not extended to late times as was done in the TMLB' analysis.

A high source term at late time will result if significant amounts of fission products are not released extensively from the fuel until melt/concrete interactions begin. Therefore, the core temperatures (MARCH) and release-rate coefficients (CORSOR) at the low ends of their uncertainty ranges are used for the late-high case. Conditions leading to low in-vessel retention are also used. The ex-vessel melt/concrete aerosol generation as calculated by VANESA is at the high end of its range. The high-end calculation assumes that no water pool exists above the molten pool, because occasional small steam explosions blow the water out. Finally, low aerosol depletion is achieved by assuming a spray-drop size at the high end of its uncertainty range. Spray failure would yield even less depletion, but no reasonable means of failing the sprays (without also failing the containment building) could be identified.

For the low case, core temperatures and release-rate coefficients at the low ends of their uncertainty ranges are used. Conditions were chosen so that RCS retention was at the high end of its uncertainty range. Conditions for low ex-vessel aerosol production and large ex-vessel retention are assumed in this case. Indeed, for this case, there is no melt/concrete aerosol production, because the ejected melt is assumed to quench in water in the reactor cavity to form a permanently cooled debris bed.

3.2.2 S_2D Results Summary: Quantitative Aspects

The CONTAIN code was used to calculate the range in the time-dependent masses of CsI, CsOH, tellurium, RFP, and inert solids suspended in containment and to calculate the total suspended aerosol radioactivity. The time-dependent total suspended aerosol radioactivity is shown in Figure 5.

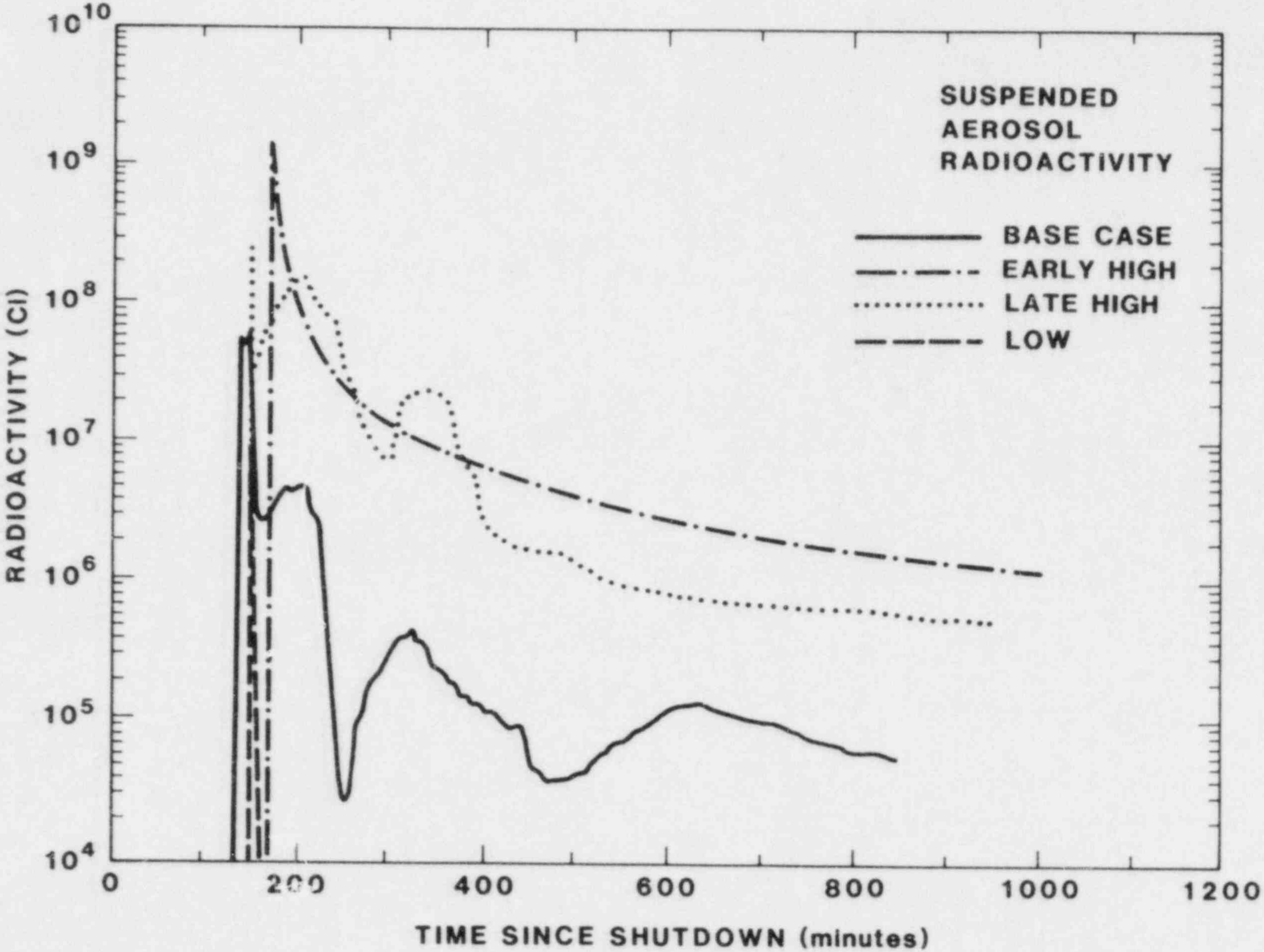

Figure 5. Total Time-Dependent Suspended Aerosol Radioactivity, Surry S_2D

The suspended radioactivity at vessel failure ranges from about 1 GCi down to a factor of 1000 lower. The high case is composed about equally of volatile and refractory fission products. The low case is dominated by CsI. At late times, the high case is about 1 MCi and is a factor of ten higher than the base-case value reported in BMI-2104. The low case is many orders of magnitude lower than the base case; this occurs because the ex-vessel debris is assumed to be coolable. In addition, any aerosols released at the time of vessel failure are rapidly removed from the containment atmosphere by the containment sprays.

3.2.3 S_2D Results Summary: Qualitative Aspects

Many of the findings of the TMLB' study are also applicable to the results of the QUEST assessment of the Surry S_2D sequence. The source-term uncertainty near the time of vessel failure is dominated by in-vessel processes. Particularly important uncertainties are

- Core-temperature histories
- Release rates from fuel for given temperature and gas-flow conditions
- Natural circulation within the RCS
- Net retention of fission products within the RCS (including effects of revaporization and resuspension

The source-term uncertainty at late time (about 15 hours into the accident) is dominated by ex-vessel processes. Particularly important uncertainties are

- Whether the ex-vessel debris is coolable, and the nature of ex-vessel core-debris interactions with concrete when the debris is not coolable
- Effectiveness of the containment sprays in removing airborne aerosols

As can be seen from the above list of important uncertainties, the early source term in S_2D near the time of vessel failure is dominated by the same phenomena uncertainties that dominate the early source term in TMLB'.

For the S_2D accident, a very important uncertainty for later times is whether the core materials in the reactor cavity are coolable by the continuous flow of water from the sprays. This uncertainty is due primarily to uncertainty in the configuration of debris after it contacts water in the reactor cavity.

The second major uncertainty relative to the late-time source term in the S_2D sequence concerns the effectiveness of the containment sprays, which is strongly dependent on the drop size. There is also significant uncertainty concerning the effectiveness of these sprays given a known droplet size. The combined effect of these uncertainties leads to a factor of ten variation in the late-time suspended radioactivity in containment. Even with the most pessimistic assumption concerning spray effectiveness, however, the sprays could reduce the total suspended aerosol radioactivity by a factor of 100 in 100 minutes or less.

3.3 Grand Gulf TC Analyses

An estimate of the final source-term phenomena uncertainty was made by considering only the dominant uncertainties and applying these uncertainties to the base-case results reported in BMI-2104. The dominant uncertainties considered were those associated with in-vessel fission-product release and retention (in the RCS) and with suppression-pool scrubbing effectiveness. Though radionuclide releases from core-debris/concrete interactions were high for the Grand Gulf TC sequence, they proved not to be highly variable for the parameter and input ranges considered in the study.

3.3.1 TC Cases Considered

Because containment failure actually precedes vessel failure in this accident sequence, there is a continuous release of fission products to the environment. Consequently, separate early and late source-term estimates were not made. However, there is a possibility that suppression-pool disruption due to high-pressure melt ejection might occur at the time of vessel failure. This case is considered separately and is designated "early-high."

Table 3 gives the general conditions produced by the choices of parameters and inputs for the various stages of an accident that would be expected to result in low, high, and early-high source terms. As in previous analyses, these combinations of uncertainties are not unique in defining high or low source terms. Thus, other combinations of parameter uncertainties could also result in high or low source terms, possibly even outside the ranges defined here. Nonetheless, the combinations of Table 3 are thought to lead to estimates of reasonable high and low values for the source-term.

There are several ways in which a final low release to the environment may occur. The low case chosen here has high in-vessel production coupled with high in-vessel retention, so that the bulk of the fission products remains within the

Table 3

Conditions Expected to Result in High or Low Source Terms for Grand Gulf TC Sequence

Accident Phenomena	Low	High	Early-High
In-vessel fission-product production	Large	Large	Large
Net fission-product retention in RCS	Large	Small	Small
Aerosol production by core/concrete interactions in drywell	Small	Large	Large
Aerosol retention in drywell	Large	Small	Small
Aerosol retention in suppression pool	Large	Small	Small
Suppression pool disruption	No	No	Yes
Aerosol retention in containment	Large	Small	Small

RCS. Retention of aerosols in the drywell, in the suppression pool, and in the containment is high. No leakage through the drywell that bypasses the suppression pool was considered.

The high case defined here corresponds to high in-vessel fission-product production and high aerosol production in the drywell. Use of low-retention conditions throughout the accident sequence also leads to the high-release case. Again, no leakage of aerosols that bypasses the suppression pool was considered.

The early-high case is identical to the high case, except that conditions leading to suppression-pool disruption due to high-pressure melt ejection are adopted.

3.3.2 TC Results Summary: Quantitative Aspects

In the TC uncertainty analysis, it was assumed that phenomena uncertainties associated with ex-vessel aerosol production, drywell fission-product retention, and containment building fission-product retention were of secondary importance relative to other uncertainties. Thus, the RCS retention and suppression-pool retention uncertainties dominated.

The results of the analysis for the base, low, and high cases are summarized in Tables 4 through 6, respectively. The details of how these results were obtained are described in Volume IV. The range in release fractions is almost a factor of 1000, with the base case falling near the middle of this range.

3.3.3 TC Results Summary: Qualitative Aspects

The three dominant uncertainties in the determination of the source term for the Grand Gulf TC sequence are

- Retention of fission products within the RCS (including the effects of revaporization and resuspension)
- The effectiveness (decontamination factor) of the suppression pool in removing fission-product gases and aerosols from the gases passing through it (including the effects of pool bypass)
- The likelihood and potential consequences of a high-pressure melt ejection causing severe disruption of the suppression pool, such that most of the fission products released in-vessel and during the high-pressure melt ejection would enter the containment building atmosphere

The phenomena uncertainties associated with in-vessel fission-product release and retention within the RCS are essentially the same as for the Surry accident sequences.

The suppression pool in the Grand Gulf BWR design is an extremely important means of reducing radioactivity release to the environment. Uncertainties associated with the modeling of aerosol removal mechanisms during suppression-pool scrubbing can lead to an order-of-magnitude uncertainty in the decontamination factor. This has a significant effect on the radioactivity release to the containment building, because most of the fission products released from the reactor vessel are trapped in the suppression pool. Even for the most pessimistic assumptions on suppression-pool effectiveness, however, suppression-pool scrubbing is so effective that total release fractions are much lower than the high estimates for the Surry TMLB' and S_2D sequences.

Table 4

Final Distribution of Fission Products in Grand Gulf TC Sequence - BMI-2104 Base Case

Species	Fraction of Core Inventory				
	RCS	Drywell	Pool	Containment	Environment
CsI	0.19	3.6×10^{-2}	0.77	1.4×10^{-4}	6.8×10^{-3}
CsOH	0.51	1.4×10^{-3}	0.49	9.2×10^{-6}	3.5×10^{-4}
Te	0.22	0.32	0.45	4.3×10^{-4}	8.8×10^{-3}

Table 5

Final Distribution of Fission Products in Grand Gulf TC Sequence - Low Case

Species	Fraction of Core Inventory				
	RCS	Drywell	Pool	Containment	Environment
CsI	0.90	0.004	0.096	3×10^{-6}	9.7×10^{-5}
CsOH	0.90	0.0003	0.099	2×10^{-7}	7.1×10^{-6}
Te	0.90	0.008	0.092	1×10^{-5}	1.9×10^{-4}

Table 6

Final Distribution of Fission Products in Grand Gulf TC Sequence - High Case

Species	Fraction of Core Inventory				
	RCS	Drywell	Pool	Containment	Environment
CsI	0.1	0.04	0.78	2×10^{-3}	7.6×10^{-2}
CsOH	0.1	0.0025	0.89	2×10^{-4}	6.4×10^{-3}
Te	0.1	0.069	0.65	1×10^{-2}	1.7×10^{-1}

The largest single uncertainty associated with the Grand Gulf TC sequence concerns suppression-pool disruption. It has been predicted that high-pressure melt ejection and the subsequent vessel blowdown could overpressurize the drywell to such an extent that gas-flow rates through the suppression pool could exceed the fluidization velocity. The suppression pool would then be dispersed into a droplet flow regime. If this were to occur, it is likely that most of the fission products released in-vessel and during the high-pressure melt ejection would enter the containment building atmosphere. A much more detailed analysis of this phenomenon than was done for QUEST, including accounting for the tortuous flow path for ejected core melt from vessel to drywell, would be required to clarify the consequences of suppression-pool disruption.

4. DISCUSSION AND IMPLICATIONS OF QUEST ANALYSES

The QUEST program provided a quantitative description of the uncertainty in the radiological source term for three specific severe LWR accidents. The overall results of the uncertainty evaluations were summarized in the previous section. However, there are many other important conclusions and insights beyond a simple quantification of source-term uncertainty that can be obtained from those analyses. This section summarizes the most important of those conclusions and insights.

Upon inspection of the source-term uncertainty estimates in Section 3, one important conclusion is obvious; the uncertainty ranges, although large in some cases, are definitely quantifiable. Thus, the uncertainty ranges found in the QUEST study can be thought of as envelope of "best-estimate" source terms for those sequences. As the data base improves, and as "second-generation" models, reflecting an ever-improving understanding of accident phenomena, are exercised, the envelope should be considerably narrowed. The degree to which this happens will be a measure of the effectiveness of severe-accident research programs currently underway.

Another important observation is that for a time-dependent source, as presented for TMLB', the importance of uncertainties is also time dependent. Uncertainties in cesium and iodine release have a very large impact on the potential source term should containment failure coincide closely with vessel failure. For later containment failure, these uncertainties are less important, because by then essentially all the cesium and iodine will have been released and deposited. However, at these later times uncertainties associated with revaporization become important.

Comparison of the BMI-2104 calculated source terms with the WASH-1400 calculated source terms had shown a significant reduction in the radiological threat to the environment in some cases. An important question to be answered by the QUEST study was whether this conclusion still held if uncertainties were considered. Comparison of the early-high source terms summarized in the previous section with the WASH-1400 results shows that, with major uncertainties taken into account, it is possible to calculate source-term values that approach (and in a few instances exceed) WASH-1400 values. However, this is true only in certain specific instances involving early containment failure. In all cases examined, the calculated suspended radioactivity, even for particularly bad combinations of uncertainties, shows order-of-magnitude reductions within a few (~10) hours after core melt. This illustrates the importance of demonstrating with

high confidence that early loss of containment integrity has low probability, if reduced source terms are to be adopted for a particular plant without further reductions of source-term uncertainty bands. The conclusion that an order-of-magnitude decrease in the source term is potentially realizable must be tempered, however, by noting that revaporization of radionuclides deposited in the reactor coolant system is one mechanism that can reverse this trend and has not been fully explored in QUEST.

Even with early loss of containment integrity, the early-low estimates are in general well below WASH-1400 values. Phenomena uncertainties, which must be reduced before the validity of the lower estimates can be determined, have been identified in QUEST. This identification provides a basis for establishing priorities for the research needed to validate the lower source-term estimates, even if low containment-failure probabilities cannot be established with the desired confidence.

The total number of curies from all radioactive species available for release in the event of late containment failure is uniformly much lower than would be predicted using the models of the Reactor Safety Study. In some case, factors that are currently uncertain could lead to total suspended radioactivity late in an accident that is lower than that reported in BMI-2104. These observations for total radioactivity do not necessarily apply, however, to each radionuclide.

The makeup of the source term is often as important as the total release of radioactivity. Much of the previous work on accident source terms has emphasized the release of only a few, relatively volatile, radionuclides--cesium, iodine, and tellurium. In some cases, the source-term issue is reduced to a question of the behavior of iodine. Analyses of the fate of cesium and iodine are then used to draw conclusions about the potential releases of all radioactive materials from the plant.

The QUEST study showed the makeup of the releasable inventory of radioactive material suspended in the containment atmosphere to be complex and to evolve in time. Cesium and iodine are dominant constituents of the suspended radioactivity only early in a reactor accident. These volatile radionuclides are rapidly removed from the atmosphere as a result of natural processes such as aerosol settling or by engineered plant features such as containment sprays. The volatile radionuclides are replaced, however, by more refractory radioactive species released from the core debris during interactions with the concrete. These core/concrete interactions and the attendant release of refractory radionuclides can persist for long periods of time. Consequently, the

refractory radionuclides can be dominant contributors to the source term when containment failure occurs late in an accident. Further, the analyses done for the QUEST program showed that refractory-radionuclide release during core-debris/concrete interactions can be quite efficient in some cases. Many instances of refractory releases an order-of-magnitude greater than estimated in the Reactor Safety Study were encountered in QUEST.

An important conclusion from the QUEST study is, then, that all radionuclides must be considered in analyses of accident source terms. Restriction of attentions to just the volatile species of cesium and iodine will lead to conclusions applicable only to brief periods in a severe accident and will miss the important contributions that can be made by the refractory radionuclides.

Some phenomena were not explicitly addressed in the BMI-2104 study because their treatments were not included in the existing code suite. The QUEST study developed models for many of these phenomena and quantified the importance of their omission. Three examples of such phenomena are high-pressure melt ejection, transmutation by radioactive decay, and aerosol resuspension after containment failure.

The most important aspect of the high-pressure melt ejection, which was included in the TMLB' assessment, is the extent to which direct heating of the containment building atmosphere may contribute to early containment failure. In terms of direct radiological consequences, the high-pressure melt ejection does contribute somewhat to the early source term, but, in so doing, it actually reduces the late source term because early increased aerosol concentrations in containment lead to early increased aerosol-depletion rates. Phenomena uncertainties in other potential early containment-failure mechanisms, for example, in-vessel steam explosions and containment hydrogen detonation, also become important if it is necessary to establish low probabilities for early containment failure.

The lack of treatment in BMI-2104 of transmutation by radioactive decay of radionuclides had been thought by some to be an important uncertainty and was investigated as part of the QUEST study. The decay of tellurium-132 to iodine-132, in particular, could constitute a large source of radioactivity released by the melt/concrete interactions. This phenomenon is most important late in time and would affect the late-high source term. In the late-high case, ~40% of the tellurium remains in the melt. This tellurium would decay into a volatile, highly radioactive iodine-132 source, which increases the late-high source term by ~50%.

Resuspension after containment failure can occur from steam or gas flashing of water pools during rapid depressurization of containment. Analysis of the phenomena in QUEST, however, showed that a catastrophic failure of containment with a breach area of greater than about 100 ft^2 would be required for appreciable resuspension.

Discussion of a number of other phenomena not treated in BMI-2104 but included in QUEST can be found in Volumes II, III, and IV of this report.

A final conclusion from the QUEST study, and in many respects a most important conclusion of the work, concerns the applicability and utility of the mechanistic approach to source-term estimation taken in BMI-2104. It was found that this approach was readily adopted by independent investigators. The suite of computer models was, for the most part, useful for sensitivity and uncertainty analyses. Further, the codes and the approach may be easily modified to take into account results of recent and presumably future developments in source-term research. The approach to source-term estimation can be pursued to any level commensurate with the skill and desire of the user. Altogether, the BMI-2104 approach represents a considerable advance over the bounding, generic methods used in the Reactor Safety Study.

5. SUMMARY

The QUEST study has provided quantitative estimates of uncertainties in the source term for three particular accident sequences utilizing the severe-accident analysis methodology introduced in the BMI-2104 reports. It was demonstrated that there are significant uncertainties in the mechanistic evaluation of the radionuclide source term for these particular accident sequences. It is thus unwise to speak of specific values for the source term. Rather, ranges for the source term must be considered. However, even with the uncertainties identified, the analyses performed for QUEST only reconfirm the usefulness of the BMI-2104 methodology. Further, QUEST has demonstrated that taking into account phenomena uncertainties in the analysis, the current "envelope" of source terms for a number of accident sequences can fall well below the release values quoted in the Reactor Safety Study. Likewise, source terms exceeding Reactor Safety Study values are possible for specific accident sequences when containment failure is early.

One of the useful aspects of the QUEST program has been the identification of those uncertainties that lead to source terms on the high end of the range. These results can be used to help define research priorities within the severe-accident assessment program so as either to reduce peak source-term estimates or to demonstrate that the high estimates are likely. (In either case, the uncertainty is reduced.)

The QUEST methodology as it has developed is a useful means of identifying source-term uncertainty and can be a valuable quantitative tool for monitoring progress in reducing uncertainty.

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13. ABSTRACT (200 words or less)

Abstract:

This document summarizes the Quantitative Uncertainty Estimate of the Source Term (QUEST) program. QUEST determined the magnitudes and natures of uncertainties associated with recent, NRC-sponsored, predictions of radioactive material releases (BMI-2104) for the Surry TMLB', Surry S$_2$D, and Grand Gulf TC accidents. Radioactive "source term" uncertainties arising from input uncertainties to the suite of computer codes used in BMI-2104 were estimated as were uncertainties due to poorly modeled or omitted phenomena in the code suite. Specific input or modeling uncertainties that most impact final source term uncertainties were identified through systematic sensitivity studies of the codes and by developing and/or applying improved models for those judged to be inadequate in the BMI-2104 suite. Methods used to define, combine, and propagate uncertainties are discussed at length in the detailed reports and are summarized herein.

QUEST demonstrated rather large uncertainties in current source term predictions, and concluded that source term ranges rather than specific values must be considered in severe accident analysis. However, even with uncertainties, the "envelope" of source terms in many cases falls well below the release values quoted in previous reports such as the Reactor Safety Study, WASH 1400. Releases reaching and sometimes exceeding WASH 1400 predictions, however, are shown to be possible within the range of uncertainties if early loss of containment integrity occurs. Even with uncertainties, at least a factor of ten decrease in source terms (and some cases much more) can be expected if containment integrity is maintained for a few (~10) hours after core melt. One important QUEST finding is that the volatile fission-products such as cesium and iodine which seem to receive the most attention in safety analyses may be important for only a short time after core melt because radionuclides such as tellurium, strontium, barium, lanthanum, etc., dominate the source term at late times. Finally the phenomenological uncertainties contributing most to source term uncertainties were identified in QUEST, thus providing a basis for establishing research priorities to reduce source term uncertainties.

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