A SIMPLIFIED APPROACH FOR PREDICTING RADIONUCLIDE RELEASES FROM LIGHT WATER REACTOR ACCIDENTS*

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ABSTRACT

This paper describes a personal computer-based program (GENSOR) that utilizes a simplified time-dependent approach for predicting the radionuclide releases during postulated reactor accidents. This interactive computer program allows the user to generate simplified source terms based on those severe accident attributes that most influence radionuclide release. The parameters entering this simplified model were derived from existing source term data. These data consists mostly of source term code package (STCP) calculations performed in support of Draft NUREG-1150. An illustrative application of the methodology is presented in this paper.

INTRODUCTION

The ability to predict fission product release characteristics to the environment following a reactor accident requires the detailed modeling of a wide range of physical and/or chemical phenomena associated with core melt progression, fission product release and transport, containment loading and failure mechanisms.

Significant research activity in the area of severe accidents has been undertaken following the accident at Three Mile Island Unit 2 (TMI-2). Updated fission product source term methods were developed under NRC sponsorship and initially published in BMI-2104 [1]. A technical assessment of severe accident source term technology for U.S. Light Water Reactors was published by the NRC in NUREG-0956 [2]. This reassessment involved reviewing experimental and analytical results from severe accident research programs sponsored by the NRC and the nuclear industry. As a result of these activities the source term code package (STCP) [3] was developed as an integrated tool for source term evaluation.

Draft NUREG-1150 [4] is a major effort by the NRC to put into a risk perspective the insights that have been generated as a result of recent research into system behavior and phenomenological aspects of severe accidents. One of the major activities of this study was the development of fission product source terms for a spectrum of accident conditions. A limited number of source term calculations were performed using STCP for those accident sequences found to be important to risk in Draft NUREG-1150. Radiological source terms for other accident scenarios in Draft NUREG-1150 were extrapolated from STCP results. Simplified methods of analysis were developed with adjustable parameters that could be tuned to approximately reproduce STCP results. This simplified approach led to the development of separate computer codes for several of the plants; these were labeled following a pattern of

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XSOR, where "X" identifies the plant. For example, SURSOR was used for Surry, SEQSOR was used for Sequoyah, and so on. In addition, the simplified source-term methods also included a parametric representation of a number of source term issues that were not treated mechanistically in the STCP, but were varied as part of an uncertainty analysis. It should be emphasized that the parametric models used in the XSOR codes are not time dependent. These codes generate source terms only in terms of early or late releases.

The objective of the work described in this paper is to review the available source term data related to accidents in LWRs and to develop a simplified time-dependent approach for predicting the radionuclide releases during postulated reactor accidents. This paper also includes a description of a personal computer-based program (GENSOR) that utilizes the proposed simplified approach for estimating source terms. The computing time requirement is on the order of a few seconds, thus providing a valuable tool for quick evaluation of order of magnitude insights into source terms and for quick evaluation of important mitigative features.

APPROACH TO DEVELOPMENT OF SIMPLIFIED SOURCE TERMS

The calculation of simplified source terms consists of two underlying assumptions. First, the fission product species are grouped according to their respective chemical forms and release characteristics. Secondly, the accident conditions will be categorized into appropriate categories somewhat similar to the approach utilized for the source term assessment used in Draft NUREG-1150 [4].

For simplicity the radiological release fraction into the containment can be represented by:

$$F_i = FRCS_i + FCCI_i + FDCH_i + FREV_i \tag{1}$$

where

- F_i = fraction of the initial core inventory of species i that is released into the containment.
- $FRCS_i$ = fraction of the initial core inventory of species i released from the reactor coolant system into the containment prior to vessel failure,
- $FCCI_i$ = fraction of the initial core inventory of species i released from the corium melt during core/concrete interaction,
- $FDCH_i$ = fraction of the initial core inventory of species i dispersed into the containment atmosphere during pressurized melt ejection, and
- $FREV_i$ = fraction of the initial core inventory of species i released to containment at later times (late revolatilization from RCS).

When using Equation 1 appropriate decontamination factors (DFs) must be applied to account for retention of fission products at various stages in the release path. For example aerosol fission products would be retained in BWR suppression pools and in any water that might be overlying core debris interacting with concrete.

The categorization of radionuclide releases to the containment were determined by four key characteristics, namely;

1) Reactor type (BWR versus PWR),
2) RCS pressure prior to reactor pressure vessel breach (high versus low),
3) Concrete aggregate/composition (limestone versus basalt), and
4) Cavity/pedestal condition (dry versus flooded).

In the simplified formulation for the appearance rate into the containment, the fission product releases were treated as being proportional to time after the initial release (see Figure 1).

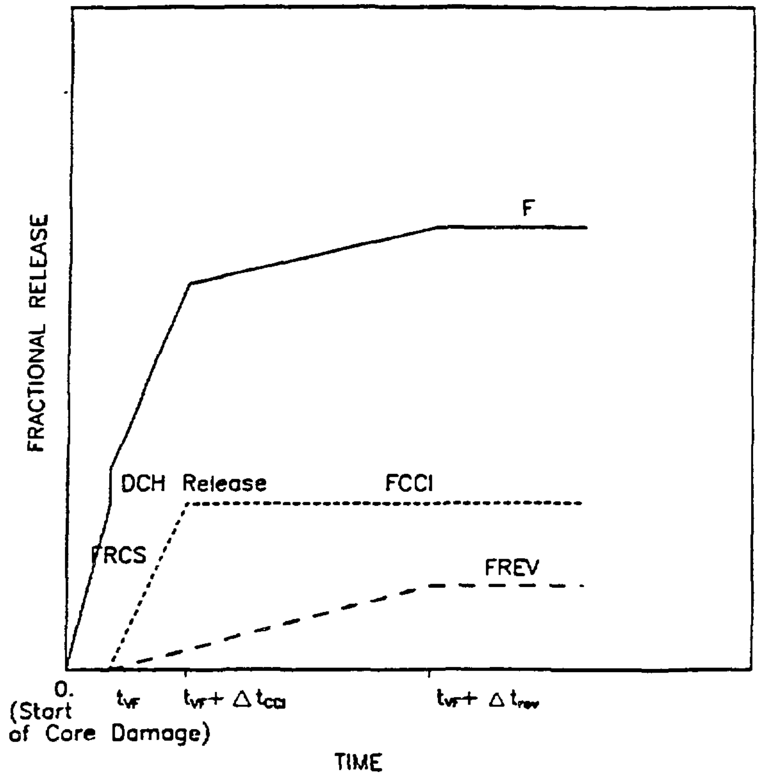

FIG. 1. Time variation of simplified radionuclide releases into the containment.

Assessing the results of STCP calculations, generic fission product releases from the reactor coolant system and from the melt during core/concrete interaction have been developed [5]. The release fraction for each radionuclide group which are assigned to an accident category generally are taken as the highest STCP calculated fraction from all those accident sequences assigned into the release category. The duration of these releases to containment have also been selected through an assessment of the existing STCP calculations.

Once the simplified in-vessel and ex-vessel release characteristics into the containment are specified a single volume containment model for a large volume PWR and a two volume compartment model for PWR ice condensers and BWR containments is used to analyze the fission product behavior in the containment. The key parameters which have been considered in the development of the simplified containment model include:

a) Containment type (Mark I, II, III versus large dry and ice condenser), and
b) Status of Engineered Safety Features (sprays, fans, etc.).

In characterizing containment behavior, appropriate geometrical data (volume, depositional surface areas, etc.), together with containment leak rates must be provided.

The parameters entering the simplified model were derived from existing source term data. These data consist mostly of STCP calculations performed in support of Draft NUREG-1150.

GENSOR

An interactive personal computer based code, GENSOR [6], was developed, incorporating the simplified approach introduced in the previous section. This computer program allows the user to generate simplified source terms based on the severe accident attributes that were found to be important for radionuclide release. The solution procedure involves an explicit Euler type numerical integration of the airborne concentration equation for each radionuclide group over time. The computing time requirement for a typical calculation on a personal computer is on the order of a few seconds, thus providing a valuable tool for quick evaluation of order of magnitude insights into source terms and for quick evaluation of important mitigative features. In addition, the computer program is very simple to use and can be readily updated if the analyst wishes to incorporate new data.

ILLUSTRATIVE EXAMPLE

The accident sequence selected to illustrate the simplified approach is a station blackout in a PWR subatmospheric containment design. This sequence results in failure of the heat removal capability of the secondary system. The primary system water boils off at the pressure setpoint of the pressure relief valves. In the absence of onsite and offsite power, none of the active engineered safety features (ESFs) operate. As a result the core begins to uncover, melt and eventually slump. Because of the large mass of hot core materials in the bottom head coupled with the high primary system pressure, failure of the bottom head is expected to occur rapidly leading to depressurization of the primary system. The containment is assumed to fail early (at vessel breach) due to high pressure caused by direct containment heating.

GENSOR calculations for a sequence simulating high pressure RCS failure in a PWR with a subatmospheric containment constructed of basaltic concrete were performed. No attenuation of fission product release to the containment by the containment spray system was assumed (sprays: off). The natural deposition removal rate constant for the radionuclides (except noble gases) was assumed to be λ_{nd} = 0.24 (1/hr). This removal rate is the default value used in GENSOR, which is calculated internally based on the appropriate geometrical data (volume, deposition surface) of the Surry containment and a deposition velocity of 9.6 m/hr. Due to accumulator water discharge into the corium following vessel failure, the reactor cavity was assumed to be flooded with water for a substantial period of time while the core-concrete interaction takes place. The effective DF for the cavity pool was assumed to be DF_{cav} = 3. The containment was assumed to fail at vessel breach, (i.e., early failure)) due to the high pressure caused by DCH. This selection of sequence features in GENSOR, allows for a direct comparison with a detailed STCP calculation [7] of the TMLB'δ sequence in Surry.

The results of GENSOR calculations for I-Cs are presented in Figure 2. The results of the cumulative radionuclide releases obtained by the detailed STCP calculation for the Surry TMLB'-δ [7] are also displayed. In order to make an assessment of the simplified methodology proposed in the present study, a comparison between the total environmental releases obtained by the

GENSOR calculation and the ranges of radionuclide releases obtained by the NUREG-1150 methodology for the same accident sequence in Surry (using the SURSOR Parametric Code) are presented in Figure 3. The results of an STCP calculation for a Surry TMLB′-δ are also presented in Figure 3. There is good agreement between GENSOR predictions and STCP results. Higher Ru-La-Ce release predicted by GENSOR are due to DCH releases which were not modeled in the STCP.

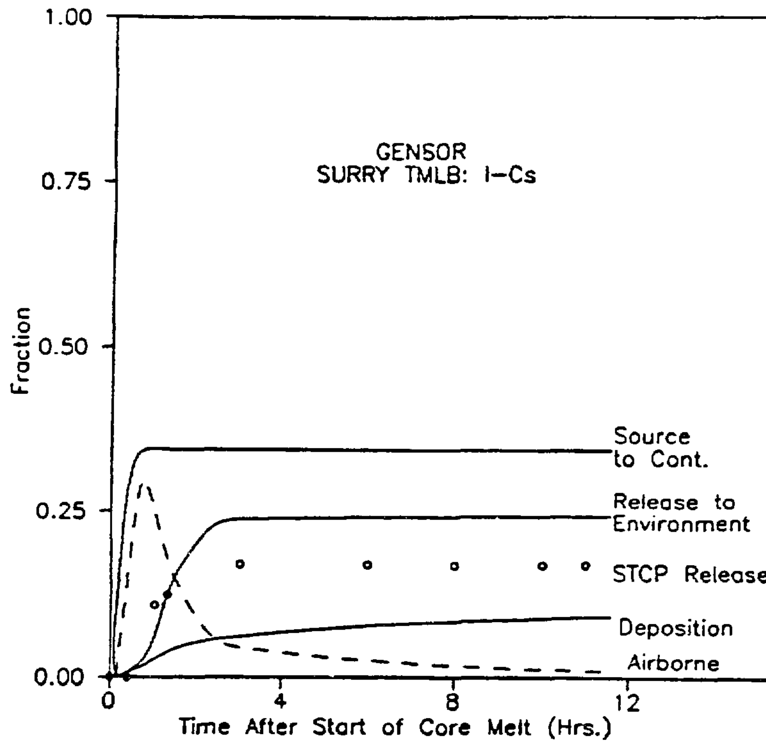

FIG. 2. I-Cs behavior in the Surry containment for a TMLB accident sequence.

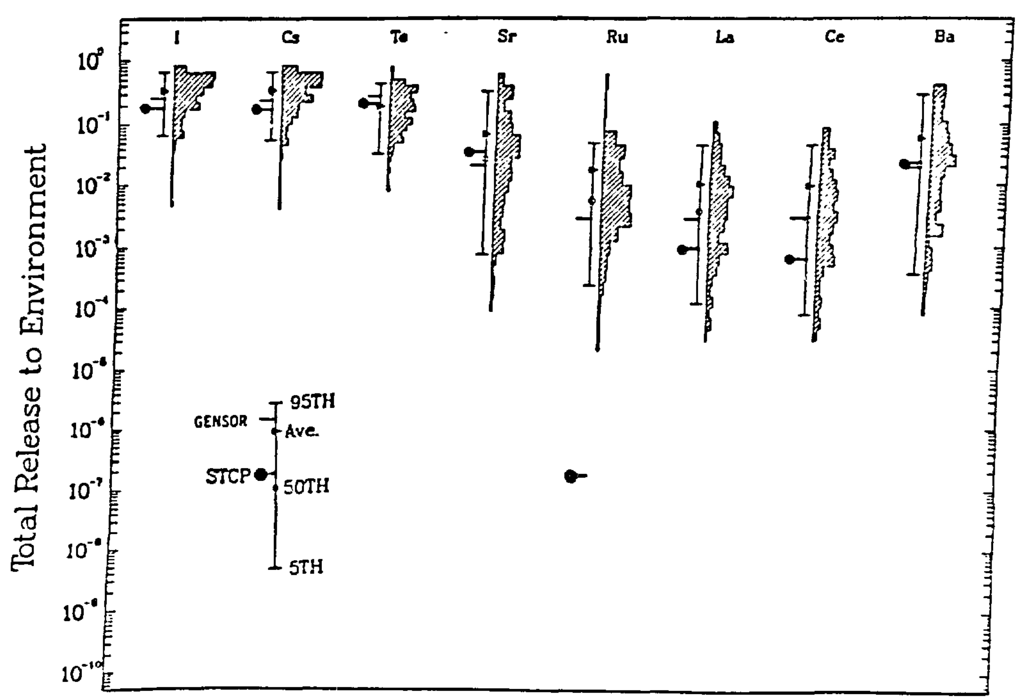

FIG. 3. Uncertainties in radionuclide releases to environment, using NUREG-1150 methodology, compared with STCP and GENSOR results (Surry, TMLB, early containment failure).

The similarities in environmental release distributions between the I and Cs, between the Sr and Ba, and between the Ru, La and Ce groups justifies the grouping, which was adopted for the present simplified approach. In addition (as shown in Figure 3) the GENSOR predictions agree with the 50th percentile of release distribution obtained by NUREG-1150 methodology.

SUMMARY AND CONCLUSIONS

A simplified approach has been developed for predicting radionuclide releases from light water reactor accidents.

A personal computer-based program (GENSOR) has been written based on the present simplified approach. This computer program allows the user to generate simplified source terms based on those severe accident attributes that most influence radionuclide release. The advantage of using this approach is that it allows flexibility, and it employs up-to-date methods and information. The computing time requirement for a typical GENSOR calculation on a personal computer is an order of a few seconds, thus providing a valuable tool for sensitivity and uncertainty studies. In addition, the computer program is very simple to use and can be readily updated if the analyst wishes to incorporate new data.

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