Regulatory Safety Review of the Source Term Assessment — Part II

Outline

Part 1: Basics of Source Term Assessment
Part 2: Radionuclides Important In Source Term Assessment
Part 3: Radionuclide Transport Mechanisms Under Normal Operating Conditions
Part 4: Estimating Radionuclide Transport Mechanisms Under Severe Accident Conditions
Part 5: Exercise on Critical Review of a Source Term Assessment for Accident Condition
Part 6: Spent Fuel Storage Source Term Calculations
Part 7: Criticality Accidents
Part 1: Basics of Source Term Assessment - Objective

To enable the reader to

• specify the role of radionuclide release (source term) assessment

• illustrate the necessity of recognizing and identifying the great uncertainties associated with performing a source term assessment

• understand the utility of the various source term estimation methods
There are five bases that may be used to estimate source term (radionuclide release) from a severe reactor accident (NUREG-1228). These are

• (1) effluent monitor readings
• (2) accident analysis contained in the safety evaluation report
• (3) various severe accident consequence studies
• (4) detailed analysis of plant conditions conducted during an accident
• (5) pre-calculated estimates that relate dominant accident conditions to potential radionuclide releases (source terms)
Source Term Estimates Based on Effluent Monitors

Obviously if a release is out of a monitored pathway, the monitor could provide useful information on the size of the release. However, a monitor cannot be used in review of SAR.
As part of the licensing process, analyses are conducted of various postulated accidents, some of which have the potential for offsite releases. These postulated accidents are listed in Section 15 of the Standard Review Plan (NUREG-0800). They include

- control rod drop accidents
- steam generator tube failures
- loss-of-coolant accidents (LOCAs)
- waste tank leak/failure
- fuel handing accidents
- spent fuel drop accidents
- anticipated transients without scram
Source Term Estimation Based on Accidents Analyzed as Part of Licensing

Codes to be used in the review of source term assessments in SAR.

- For light water reactors: US NRC SCALE computation system, including CSAS sequences (burnup), ORIGEN (isotopic and decay); etc.
- For CANDU: WIMS-AECL (burnup), RFSP (core simulation) and its dynamics version, ORIGEN; etc.
Source Term Estimation Based on Severe Accident Consequence Studies

• The Reactor Safety Study analyzed many specific accidents and the results were grouped into "release categories".
• These release categories are sometimes used to characterize the possible accident source terms by persons knowledgeable of probabilistic risk assessment (PRA) research.

Advantages of this basis are that core damage accidents are considered and various dominant accident scenarios can be compared.

However, in using this and all other source term estimation methods, the analyst must keep in mind the limitations and uncertainties involved.
Source Term Estimates Based on Detailed Analysis of Plant Conditions

Computer codes (e.g., MAAP-CANDU; TACT, NUREG/CR-3287, NUREG/CR-4722,) have been developed to predict releases (source terms) resulting from accidents.

These codes
- require considerable detailed information about the plant (e.g., containment volume), accident conditions (e.g., leak rates), and effectiveness of source term reduction mechanisms (e.g., sprays)
- are very powerful and flexible.

However:
- these codes require considerable time and information
- the analyst must keep in mind the limitations and uncertainties involved.
The basic assumptions of this method are that
• (1) there is a small set of accident conditions that dominate any severe accident release,
• (2) there are values that can characterize these dominant conditions, and
• (3) these conditions can be recognized/characterized during an actual accident.
The following are the basic steps for source term estimation:

• (1) Estimate the inventory of fission products in the core.
• (2) Estimate the amount of fission products released from the core.
• (3) Identify the dominant release pathway.
• (4) Characterize the dominant mechanisms that will act to reduce the release. These would include filters, pools of water, sprays, or natural processes.
• (5) Estimate the release rate.
Source Term Estimates Based on Pre-calculated Assumptions of Dominant Accident Conditions

Advantage: The results will be a set of precalculated doses that can be used to compare possible consequences of various accident sequences.

Disadvantage is the large uncertainties.
Part 2: Radionuclides Important In Source Term Assessment

Objectives of Part 2:
To enable the reader to
• identify the factors that have the greatest effect on radionuclide inventory in the reactor core
• identify the fission products that are important to offsite consequences.
Source of Fission Products:
- The term "fission product" as used in this document will include not only the isotopes produced directly in fission (primary fission products), but also those produced indirectly through primary and fission product decay (secondary fission products) and other methods.

Inventory of Fission Products in the Core
- The first consideration in determining the contribution of a particular fission product to the overall source term is how much of the radionuclide is available in the core at the time of the accident.
- This is a difficult question to answer because the fission product inventory is influenced by a number of factors as shown in Table 2.1. You need to run ORIGEN-S!
Fission Products Important to Offsite Consequences

• Many fission products in the core do not need to be considered in source term estimation because they contribute little or nothing to offsite consequences. Many studies have been performed to determine which fission products are the most important in terms of offsite consequences during severe core damage accidents.

• A rough ranking of the importance of each group of radionuclides (e.g., iodine) from a health effects perspective can be obtained by summing the assigned scale values for each radionuclide.

• The most important are listed in Table 2.2.
Inventory Assumptions Used for Source Term Estimation

For each of the radioisotopes listed in Table 2.2, a specific Inventory expressed in Ci/MWt is provided. These data are the standard starting Inventory cited in many source term studies, such as NUREG-0956, and are in agreement with other computer codes, such as CINDER results (NUREG/CR-3108). The computer codes used to estimate core Inventory are considered to be the most accurate of all the codes used in estimating source term.

- The inventories in Table 2.2 are based on a burnup of 30,000 MWD/MTU. Reviewer should adjust the inventory of radionuclides that have a half-life exceeding one year to account for burnup.
Part 3: Radionuclide Transport Mechanisms Under Normal Operating Conditions

Objective:
To enable the reader to
• describe the fission product barriers
• describe the changes that occur in fuel during normal operation and what effect these changes have on fission product transport from the core
• identify the fission products most likely to be released from the fuel and plant during normal operation
• recognize normal versus accident release rates;
• specify the coolant concentration typical of an operating reactor that could be used to estimate the source term for a coolant release assuming no fuel damage and how these concentrations can change during rapid power changes (spikes)
Barriers to Fission Product Release

For the fission products generated within the core to reach the environment, they must pass through four fission product barriers:

- The first barrier is the fuel pellet often referred to as the fuel matrix.
- The second barrier is the fuel pellet cladding.
- The reactor coolant system provides a third barrier to fission product release.
- The final and ultimate barrier to fission product release is the reactor containment.
The fuel used in reactors undergoes thermal distortion and cracking because of the large temperature differences that exist between the center line and the surface of the fuel pellet.

As gaseous fission products are formed, they will move to cracks where they can escape the fuel pellet.
Radionuclide Transport from the Fuel into the Coolant

- A small fraction of fuel pin cladding will leak during normal operation because of manufacturing flaws, irradiation-induced creep, and other mechanisms.
- The American Nuclear Society (ANS) Standards Committee Working Group has prepared a set of typical radionuclide concentrations for estimating the non-accident radioactivity in the principal fluid streams of an LWR over its life time [American National Standards Institute (ANSI)/ANS-18.1-1984]. The expected coolant concentrations for the reference plant types in the ANS standard are shown in Table 3.2.
- For each reactor, the actual licensed power (MWt) has to be divided by the reference power (3,400 MWt), then multiplied by the mass of the water (kg from the table) to estimate the coolant mass used in Table 3.2.
Routine Effluent Releases

• Radionuclides are routinely released in nuclear power plant effluents during normal operation or as a result of anticipated operational occurrences.

• The ranges of airborne effluent releases (curies) from BWRs and PWRs for the year 1980 are summarized in Table 3.3 and can be used during the review.
Part 4: Estimating Radionuclide Transport Mechanisms Under Severe Accident Conditions

Objective
To enable the reader to
• describe the phenomena that occur as a result of core heatup, beginning with normal operating conditions and ending with melting of the fuel pellets, and the relationship of these phenomena to release of fission products from the core
• describe pathways that allow radionuclides to be transported from the core to the environment and explain what effect the pathway has on the fraction of radionuclides released
• describe the dominant mechanisms that act to reduce an offsite release and how to characterize those mechanisms during an event
• describe the procedure for bounding the source term for a potential release during an accident
Estimating Radionuclide Transport Mechanisms Under Severe Accident Conditions

Part 5 will outline a very simple method for estimating source terms for various severe accident conditions. This method is based on two assumptions:

• (1) that there is a limited number of accident conditions that characterize the size of source terms and
• (2) that these accident conditions can be estimated during an accident.

Part 4 will identify these conditions and characterize

• their effects on source terms and discuss how these accident conditions can be estimated based on observable plant conditions. The hope is to relate actual plant conditions to the range of possible source terms.
Major Considerations

• To make the first approximation of a severe accident source term the analyst must:
  • (1) estimate the Inventory of fission products in the core
  • (2) estimate the fraction of the fission product inventory released from the core
  • (3) estimate the fraction of the fission product Inventory released from the core that is removed on the way to the environment
  • (4) estimate the amount of the available fission product inventory with potential for release to the environment.
To estimate the inventories of fission products in the core. Table 2.2 can be used. Specific plant inventory can easily be estimated by multiplying Table 2.2 values by the long-term steady-state power level (MWe or MWt, as applicable) at the time of the accident.

When using the fission product inventories specified in Table 2.2, keep in mind that these values will greatly overestimate long half-life fission product (e.g., cesium) inventory in a new core.
(2) estimate the fraction of the fission product inventory released from the core

- Tables 4.1 - 4.4 show the fraction of fission products that can be assumed to be released from the core for each of the fuel damage states.
- Recall that these fission products were identified in Section 2 because of their importance to early health affects.
(3) estimate the fraction of the fission product Inventory released from the core that is removed on the way to the environment

To estimate the amount of fission products released from the fuel that reach the atmosphere, one must:

(1) estimate the pathway the fission products will follow through the plant and

(2) estimate the effectiveness of the various fission product removal mechanisms encountered. On the basis of this information, the reduction factor (RDF) for that particular release pathway can be estimated.

Tables 4.5 – 4.6 lists those that have been selected to represent the dominant removal mechanisms.

Consideration of the **Containment Bypass and Hydrogen Detonation/Burns** is very important
(4) estimate the amount of the available fission product inventory with potential for release to the environment.

- The final step is to calculate the fission products actually released from the containment. Table 4.6, which summarizes typical escape fractions (EFs) released in 1 hour, can be helpful.
- A set of event trees has to be developed that provides adequate estimates.
Objective is to calculate source terms for three types of spent fuel storage accidents:

- (1) releases from spent fuel stored in a pool when the water drains from the pool causing the fuel to become uncovered, overheating the fuel, and causing cladding damage,
- (2) releases from spent fuel stored in a pool when the fuel is damaged while it is under water, and
- (3) releases from spent fuel in a dry storage cask when an accident causes both damage to the cladding of the fuel and loss of the integrity of the cask.
Basic Method to Calculate Spent Fuel Source Terms

- The method to calculate source terms for spent fuel accidents is similar to the method for the nuclear power plant accident source terms, but it is simpler. The method is simpler because the model assumes that the entire release from the spent fuel is released instantly.

- This approach does not account for the time that it takes for radionuclides to escape from the damaged spent fuel, but the amount escaping is correct even though the timing is not realistic.
To calculate the inventory of each radionuclide in the spent fuel at the time of the accident the reviewer starts with the inventories per MWt in Table 2.2. The inventories of radionuclides with a half-life of longer than one year from Table 2.2 are then adjusted for burnup using Equation 4.1. The default burnup for spent fuel is 50,000 MWD/MTU, but the reviewer can adjust this value if desired.
Fractions of Inventory Available for Release in Spent Fuel Accidents

The fractions of the radionuclide inventories that are available for release during an accident $AFi$ are shown in Table 6.1 for different types of accidents:

- Spent Fuel Pool Water Drained;
- Fuel Damaged Under Water;
- Release from a Dry Storage Cask.
Release Pathways and Reduction Factors

- For spent fuel pool accidents, the release from the spent fuel is assumed to be into a building.
- If the release pathway to the environment passes through filters, a reduction factor of 0.01 is applied to all radionuclides except noble gases.
- For spent fuel damaged underwater in a spent fuel pool, a reduction factor of 0.01 is applied to all radionuclides except noble gases to account for scrubbing by the water in the pool. This factor is in additional to reduction by building filters.
- For dry cask storage accidents, there are no reduction mechanisms to reduce the amount of activity released.
Leakage Fractions

- For all spent fuel accidents, the reviewer assumes the leak rate to the environment in terms of %/hour with a maximum rate of 100%/hour.
- For fuel casks stored outdoors, the reviewer would normally select 100%/hour to indicate a very fast transfer rate to the environment.
Inadvertent Out-of-Core Criticality Accidents

• A criticality accident results from the uncontrolled release of energy from fissile material. For reviews, a criticality accident may be modeled using the physical system scenarios in NUREG/CR-6410 (SAIC 1998) or using criticality data assumed directly by the reviewer.

• The physical systems modeled are listed in Table 7.1 along with the assumed number of fissions in the first burst and the total yield.
To calculate the source term, reviewer first determines the initial activity of each radionuclide present as the product of the yield of the initial burst (FI) (in $10^{19}$ fissions) and the activity per $10^{19}$ fissions listed in Table 7.2

**Release pathway:**

The criticality is assumed to take place inside a building. A leak rate to the atmosphere from this building is to be selected.