REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**176-8089**
SRP Section:	**03.11 – Environmental Qualification of Mechanical and Electrical Equipment**
Application Section:	**03.11**
Date of RAI Issue:	**08/31/2015**

Question No. 03.11-11

10 CFR 50.49 and 10 CFR 50, Appendix A, criterion 4 require that certain components important to safety be designed to withstand environmental conditions, including the effects of radiation, associated with design basis events, including normal operation, anticipated operational occurrences, and design basis accidents.

Regulatory Guide (RG) 1.183 provides assumptions for evaluating radiation doses for equipment qualification. RG 1.183 indicates that, "EQ equipment located outside of containment may be exposed to (1) radiation from sources within the containment building, (2) radiation from activity contained in piping and components in systems that re-circulate containment sump water outside of containment (e.g., ECCS, RHR, sampling systems), (3) radiation from activity contained in piping and components in systems that process containment atmosphere (e.g., hydrogen recombiners, purge systems), (4) radiation from activity deposited in ventilation and process filter media, and (5) radiation from airborne activity in plant areas outside of the containment (i.e., leakage from recirculation systems). The amount of dose contributed by each of these sources is determined by the location of the equipment, the time dependent and location-dependent distribution of the source, and the effects of shielding."

SRP Section 3.11 indicates that the applicant's safety analysis report should be sufficient to support the conclusion that all items of equipment that are important to safety are capable of performing their design safety functions under all environmental conditions that may result from any normal mode of plant operation, anticipated operational occurrence, design basis events, post-design basis events, and containment tests.

Finally, SRP 12.2 states that, the description of radiation sources, during normal operations and accident conditions in the plant, is used as the basis for designing the radiation protection program and for shield design calculations. This description should include isotopic composition, location in the plant, source strength and source geometry, and the basis for the values.

The applicant's approach for calculating the accident dose rates for equipment qualification outside containment is unclear. In addition, the applicant does not provide any post-accident source terms for systems and components transporting design basis accident containment sump water outside of containment. Therefore, staff has the following questions.

1. Please update the FSAR to provide the maximum post design basis accident containment sump fluid source term, and the assumptions used to develop this source term (if the assumptions are already provided in the FSAR, the applicant may reference the appropriate FSAR section).

2. Please update the FSAR to include the maximum post design basis accident source term information for major components outside containment that would contain significant post design basis accident source terms, such as the shutdown cooling system pumps and main control room filters, and provide the assumptions used to develop these source terms in the FSAR. In addition, for any source terms provided, please ensure that FSAR Table 12.2-25 includes the parameters for the source term, or include them elsewhere in the FSAR.

3. Please identify each system that re-circulates containment sump water outside of containment during a design basis accident (e.g. shutdown cooling system and containment spray system), and for each of these systems:

    a. Describe the methods, models, and assumptions used to calculate the post-accident total integrated dose values, for rooms or cubicles in APR1400-E-X-NR-14001-P, Table 4, which contain a system or component transporting containment sump water outside of containment, and for rooms adjacent to such systems or components.

4. Please identify each system (if any) that will be used to process the containment atmosphere during design basis accidents, and for each of these systems:

    a. Describe the methods, models, and assumptions used to determine the radioactive contents within the systems.

    b. Describe the methods, models, and assumptions used to calculate the accident total integrated dose values, for rooms or cubicles in APR1400-E-X-NR-14001-P, Table 4, which contain a system used to process the containment atmosphere during design basis accidents, and for rooms adjacent to such systems or components.

5. Please identify the locations of post-accident ventilation process and filter media (e.g. emergency control room ventilation system filter media) and for each of these systems:

    a. Describe the methods, models, and assumptions used to determine the radioactive contents of the filter media.

    b. Describe the methods, models, and assumptions used to calculate the accident total integrated dose values, for rooms or cubicles in APR1400-E-X-NR-14001-P, Table 4, which contain post-accident filter media, and for rooms adjacent to such systems or components.

6. Please indicate if airborne activity associated with leakage from recirculation systems has been considered in the total integrated dose values for rooms and cubicles in APR1400-E-X-NR-14001-P, Table 4? If so, please describe the methods, models, and assumptions used in determining the airborne activity contributions associated with these sources to the total integrated dose values.

7. Please update the FSAR or APR1400-E-X-NR-14001-P, as appropriate, to provide a description of the methodology used to calculate post-accident total integrated dose values for equipment qualification outside of containment (for example, identify the applicable systems and components and provide general information regarding the assumptions used in determining the dose contributions from these systems and components).

8. Please indicate if the source terms and methodology used in developing the post-accident zoning provided in FSAR Figures 12.3-20 through 12.3-51, is consistent with the assumptions used in determining the post-accident total integrated dose values for rooms in Table 4 of APR1400-E-X-NR-14001-P. If different source terms or a different methodology was used to calculate these different doses, please provide the differences.

Response – (Rev. 1)

1. The design basis accident that results in the maximum post-accident source terms in the containment sump fluid is the large break loss of coolant accident (LBLOCA) discussed in DCD Subsection 15.6.5. The major input parameters used in the radiological consequences analysis for LBLOCA are summarized in Table 15.6.5-13. The maximum post-LOCA source terms for the IRWST sump will consist of 40% halogens, 30% of alkali metal, and small fractions of other fission products in the core inventory as summarized in DCD Tier 2 Table 15A-1.

   The maximum post-LOCA source terms inside the containment consist of two parts: (1) airborne radioactivity in the containment, and (2) radioactivity contained in the in-containment refueling water storage tank (IRWST) sump water.

   (1) Airborne radioactivity in the containment

   The airborne radioactivity in the containment is calculated based on the fission products released from the core into the containment at a core power of 4,062.66 MW_t, which is two (2) percent higher than the expected power of 3,983 MW_t and corresponds to a three-cycle burnup of 56.4 GWD/MTU. The maximum core inventory and fraction of fission products released into containment are given in DCD Tables 15A-1 and 15A-2, respectively. Assumptions used to develop this source terms in the containment are as follows:

   - The fission products released into the containment are assumed to exist only in the atmosphere and to mix instantaneously and homogeneously throughout the free air volume inside the containment;

   - Effect of radioactive decay during holdup in the containment is included; and

- Removal effect by the containment spray and natural deposition is included.

(2) Radioactivity contained in the IRWST sump water

As with airborne radioactivity in the containment, the radioactivity of the fission products dissolved in the IRWST sump water is also calculated based on the fission products released from the core into the containment. Therefore, the source terms for the calculation of the IRWST sump water are similar to those for the airborne source in the containment. Assumptions used to develop this source terms in the IRWST sump water are as follows:

- Initial source term in IRWST consists of 40% of halogens in the core inventory, 30% of alkali metal, and small fractions of other fission products;
- All of the fission products (except for noble gases) released from the core to the containment are assumed to be instantaneously and homogeneously mixed in the IRWST water as provided in Subsection 15.6.5.5.1.2;
- Effect of radioactive decay during holdup in the containment is included; and
- Effect of deposition and plate-out on the wall of the IRWST is not considered.

The above discussion, including the basis and assumptions for the development of the post-accident containment source terms, is added to the end of DCD Subsection 15.6.5.5.1.2.

2. The EQ TIDs for components/equipment should consider the expected worst-case environment condition (i.e., bounding) taking into account all post-accident conditions. Depending on the location of components, the maximum post-DBA source terms for major components located outside containment (e.g., components in the shutdown cooling (SC) system, containment spray (CS) system, safety shutdown (SI) system, and emergency HVAC filters) are determined from the followings:

- Loss of coolant accident (LOCA) for SSCs inside/outside reactor containment building (RCB) except for main steam valve house and fuel handling area in the auxiliary building (outside containment building);
- Main steam line break (MSLB) for SSCs in the main steam valve house (MSVH) in the auxiliary building; and
- Fuel handling accident (FHA) for SSCs in the fuel handling area in auxiliary building.

In calculating TIDs for components in the MSVH, the MSLB accident environment is chosen as the representative condition based on following justification:

- For the MSLB accident, the fluid from the broken steam line containing the radioactivity is released into the MSVH. For other accidents (e.g., SGTR) except for the LOCA and FHA, the steam produced in the SGs for plant cooldown is directly released to the environment via the MSSV or ADV in the auxiliary building, where the release points of the MSSV/ADVs are located outside the MSVH.

- The impact on TIDs of components from the direct radiation in pipings of the MSSV/ADVs through the MSVH is expected to be insignificant compared to radioactivity released from the MSLB because of the shielding effect of pipe wall.
- In addition, the results of radiological consequences of these accidents, which are given in Table 1 below, have shown that the TEDE at the EAB due to a MSLB accident was higher than a SGTR accident, which means that the amount of radioactivity released from the MSLB accident would be higher than the SGTR accident.

Table 1. Summary of Radiological Consequences of MSLB and SGTR accidents

Accident	TEDE (mSv) @ EAB
MSLB	4.91E+01
SGTR	8.07E+00

The resultant maximum source terms for components are calculated using the RUNT-G code, and the corresponding parameters and assumptions are described in detail below in the responses to sub-questions 3 through 6.

Radioactive source dimensions and parameters in DCD Table 12.2-25 are used for the determination of the radiation zone maps and shielding wall thickness, not for the TIDs for equipment qualification. The parameters for the determination of the source terms are separate from DCD Table 12.2-25 and are also included in the responses to sub-questions 3 through 6.

3. The calculation methods, models, inputs and assumptions for the determination of post-accident TIDs are delineated in Appendix B (provided in Attachment 2) which will be added to Technical Report (TeR) APR1400-E-X-NR-14001-P. The responses to sub-questions 3 through 6 provide a general response; however, more specific detail for each can be found in Appendix B.

   There are two systems, the safety injection system (SIS)/shutdown cooling system (SCS), and the containment spray system (CSS) that recirculate containment sump water outside of containment during a LOCA. The SIS/SCS and the CSS are part of the ESF systems used for the mitigation of a LOCA. TIDs for the cubicles containing the ESF components/equipment and for the rooms adjacent to the components are calculated based on the post-LOCA environment.

   The determination of accident TIDs for the SIS/SCS and CSS systems is described in Section B.2 of Appendix B, *TIDs in ESF System Areas in Auxiliary Building (outside containment)*. The detailed and specific input parameters and assumptions are summarized in Subsection B.2.1 and the methods and models are described in Subsection B.2.2. The integrated RUNT-G and ISOSHLD computer program is used to calculate the accident TIDs. The computer model includes contributions of direct radiation and airborne radioactivity associated with the source component inside the target room.

4. There are no systems in the APR1400 design that are used to process the containment atmosphere during design basis accidents.

5. Post-accident ventilation process and filter media related systems consist of the auxiliary building controlled area emergency exhaust air cleaning units (ABCAEEACUs) and the control room emergency makeup air conditioning units (CRE ACUs).

   The ABCAEEACUs are used to process the atmosphere in various cubicles inside the auxiliary building, including those that contain ESF systems, during and after a design basis accident. They are part of the AB Controlled Area HVAC system and consist of two 100% trains.

   a. The inputs and assumptions, method, and model to determine the radioactive contents within the ABCAEEACUs are included in Subsections B.2.1, and B.2.2.1 and B.2.2.2, and Table B-3 in Appendix B. The RUNT-G and ISOSHLD models for TIDs are depicted in Figures B-4 and B-7, respectively. A discussion of the direct radiation is presented in Subsections B.2.2.3, B.2.2.4, and B.2.2.5.

      The TIDs for the filter and charcoal bed rooms are included in the revised DCD Table 3.11-3 and Table 3 in TeR APR1400-E-X-NR-14001-P. For determination of the accident TIDs, the indirect radiation from adjacent and surrounding rooms is not included.

   b. The response for this item is included in sub-item a above.

   The APR1400 design includes four control room supply air handling units (CR AHUs) and two CRE ACUs, which contains medium and high efficiency particulate filters and carbon adsorbers to remove radioactive contaminants in emergency conditions. Two CR AHUs in each train are dedicated for air supply during post-accident conditions.

   TIDs for filter loading of the CRE ACUs are bounded by those of ABCAEEACUs since the inlet concentrations in the CRE ACUs are significantly lower compared to those in the ABCAEEACUs. The CRE ACUs are conservatively assumed to have the same TIDs for the ABCAEEACUs. Therefore, TIDs resulted from filter loadings in the CRE ACUs are not specifically analyzed.

6. The airborne activity associated with leakage from recirculation systems is included in the determination of TIDs for rooms and cubicles in the revised DCD Table 3.11-3 and Table 3 of APR1400-E-X-NR-14001-P.

   The calculation pathway is depicted as “Due to Containment Leakage” and “Due to SI/SC/CS Components Leakage” in Figure B-1 in Appendix B. The input parameters and assumptions, method, and model are described in Subsections B.2.1, and B.2.2.1 and B.2.2.2, and Table B-2 in Appendix B. The RUNT-G and ISOSHLD models for

TIDs are depicted in Figures B-4 and B-5. A discussion of the direct radiation is included in Subsections B.2.2.3, B.2.2.4, and B.2.2.5.

The airborne TIDs for the associated rooms and cubicles are included in the total TIDs in the revised DCD Table 3.11-3 and Table 3 of TeR APR1400-E-X-NR-14001-P.

7. This response, which contains general information regarding the basis, inputs, and assumptions used in determining the dose contributions, will be included as Appendix B in TeR APR1400-E-X-NR-14001-P.

8. Different source terms and methodologies are used in developing the radiation zone maps, the minimum shield wall thicknesses, and the total integrated dose for equipment qualification. The comparison between the source terms and methodologies for these analyses is summarized in the following table, Table 2.

Table 2 Comparison of Methodologies used in Radiation Zoning and Equipment Qualification

Items		Dose Rate for Radiation Zoning	Total Integrated Dose for Equipment Qualification
Normal Condition	Source Term	0.25% Fuel Failure with no gaseous stripping operation, DCD Chapter 12	1.0% Fuel Failure with continuous gaseous stripping operation, DCD Chapter 11
	Computer Code Applied	MicroShield code	MicroShield code
	Radioactive Decay	No Decay	No Decay
Accident Condition	Source Term	LOCA Event Condition	LOCA Event Condition MSLB Event Condition FHA Event Condition
	Computer Code Applied	• Source Term: Hand Calculation • Dose rate calculation: MicroShield Code	• Source Term: RUNT-G Code • TIDs calculation: ISOSHLD Code
	Radioactive Decay	Decay	Decay

Note) All of the shielding geometries available in ISOSHLD computer code is included in the RUNT-G computer code

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Impact on DCD

DCD Subsection 15.6.5.5.1.2 will be revised as indicated in Attachment 1.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

Technical Report APR1400-E-X-NR-14001-P/NP will be updated as indicated in Attachment 2.

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IRWST water pH remains at greater than 7.0 for duration of the accident including the effect of acids and bases created during the LOCA event and the radiolysis products. Consequently, the re-evolution of dissolved iodine from the IRWST is not credible and is therefore not considered in the analysis.

15.6.5.5.1.2 Engineered Safety Feature (ESF) System Leakage

The ESF systems that recirculate IRWST water outside containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. The radiological consequences from the postulated ESF leakage are analyzed and combined with consequences postulated for other fission product release paths to determine the total radiological consequences from the LOCA.

According to the NRC RG 1.183, the initial source term in IRWST consists of 40% of halogens in the core inventory, 30% of alkali metal, and small fractions of other fission products, which is

Post-LOCA Sump Water Iodine Source Term

~~NRC RG 1.183 requires that, with the exception of noble gases, all of the fission products released from the fuel to the containment are~~ assumed to instantaneously and homogeneously mix in the IRWST water. Consistent with this guidance, a total of 40 percent of the core iodine released during the gap and early in-vessel phases is assumed to mix in the IRWST water.

Effect of radioactive decay during holdup in the containment is only considered without removal effect by deposition and plate-out on the wall of the IRWST.

ESF Leakage Release Path

The ESF pumps including the containment spray (CS), safety injection (SI), and component cooling water (CCW) pumps are located in the auxiliary building (AB). The ESF leakage is assumed to be retained on the floor of the equipment compartments in the AB and the iodine in the ESF leakage flashes and becomes airborne in the AB and the iodine is released to the environment through the AB ventilation exhaust system.

Flashing of Iodine from ESF Leakage

NRC RG 1.183 requires that if the temperature of the ESF leakage exceeds 100 °C (212 °F), the fraction of total iodine in the liquid that becomes airborne is assumed to be equal to the fraction of the leakage that flashes to vapor. This flash fraction (FF) is determined using a

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APPENDIX A - CALCULATION METHOD OF DETERMINING NORMAL CONDITION TIDs FOR ENVIRONMENT QUALIFICATION
APPENDIX B - CALCULATION METHOD OF DETERMINING POST-ACCIDENT CONDITION TIDs FOR ENVIRONMENT QUALIFICATION

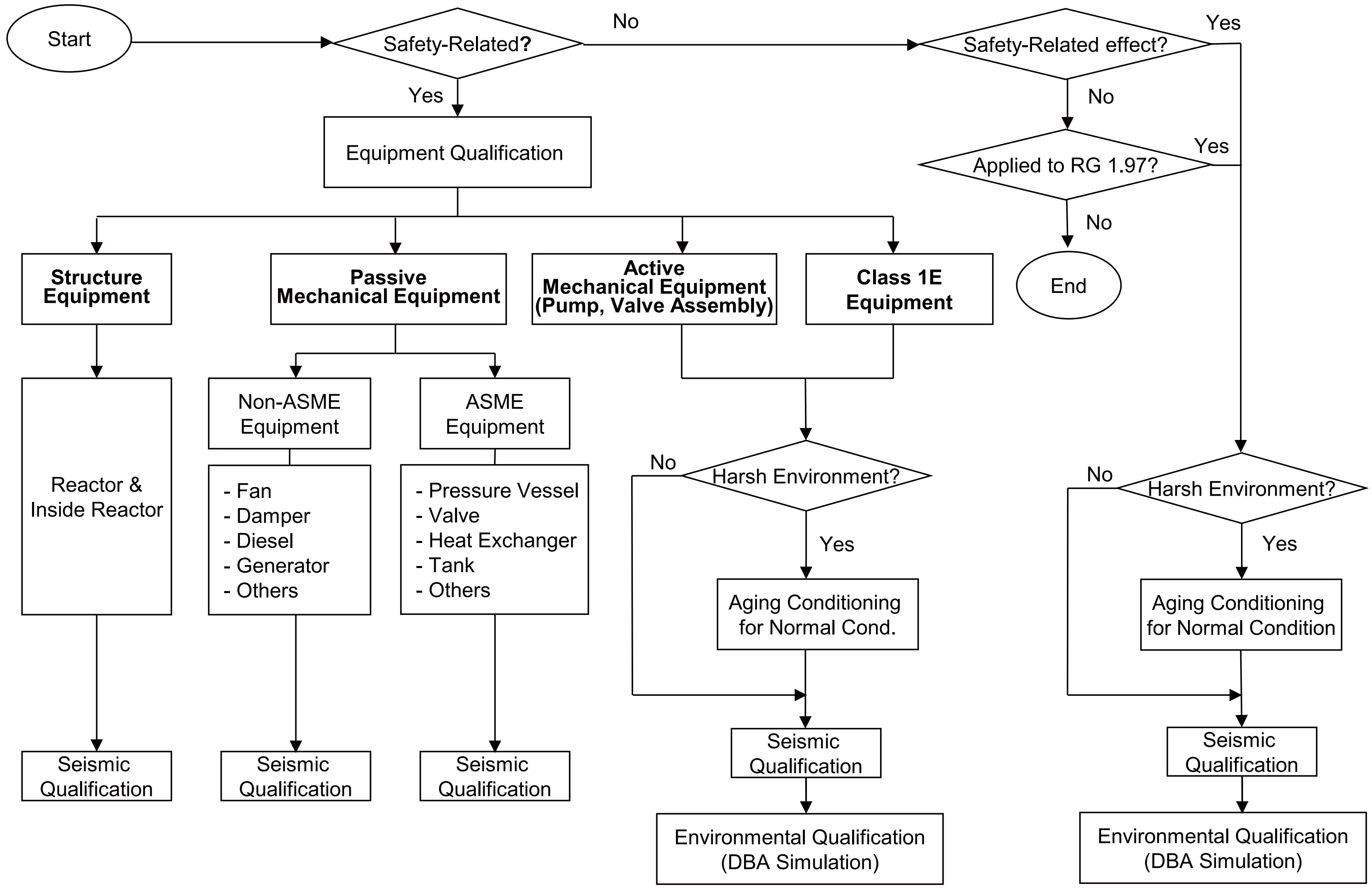

Figure 3 Equipment Qualification Flow Diagram By Type of Safety-Related Equipment

In the next page, "A" will be added

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APPENDIX ~~3A~~ – ~~CALCULATUION METHOD OF POST-ACCIDENT TIDs FOR SYSTEM INSIDE AND OUTSIDE CONTAINMENT~~

CALCULATION METHOD OF DETERMINING POST-ACCIDENT CONDITION TIDs FOR ENVIRONMENT QUALIFICATION

The appendix illustrates the methods, calculation models, inputs and assumptions for the determination of accident TIDs. A simplified flow chart for calculating the general TID for systems and components inside and outside the containment is presented in Figure ~~3A~~-1.

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In general the method, including the structure of the computer model, for calculating the TIDs is based on the RUNT-G and ISOSHLD computer codes, which determines the post-DBA radioactive source terms and the corresponding accident TID values. The ISOSHLD computer code is incorporated into the RUNT-G code to run as one computer program. The integrated RUNT-G and ISOSHLD computer model is used for the determination of TIDs for systems that re-circulate containment sump water outside the containment, AB controlled area emergency exhaust air cleaning units (ABCAEEACUs) and emergency control room ventilation system filter media, and for the inclusion of the airborne activity associated with leakage from recirculation systems and containment leakage following a LOCA.

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~~3A~~.1. ~~Source Terms~~ TIDs inside containment

~~3A~~.1.1. Input parameters and assumptions

TIDs for environmental qualification of mechanical and electrical components important to safety under the post-accident radiological environment inside the containment, which ~~consist of CS/SC and SI systems,~~ include the CS/SC, SI systems, accident radiation monitors, etc., are calculated for one (1) year following a LOCA event. The input parameters, assumptions, and evaluation model for running the RUNT-G code are described below.

- **Released Source Term**: The source terms of two release phases (gap and early in-vessel release) as described in RG 1.183 are considered as the effective source terms for post-accident equipment qualification analysis. The core inventory release fractions for each radionuclide group at the gap release and early in-vessel release phases for the LOCA are listed in DCD Table ~~15A-2~~ Table 15.6.5-13. Iodines in the forms of elemental, particulate and organic iodine in the containment atmosphere are assumed to be 4.85%, 95%, and 0.15%, respectively. With the exception of elemental and organic iodine and noble gases, the fission products are assumed to be in particulate form as specified in RG 1.183, Appendix A, Section 2. The maximum core inventory of the APR1400 is shown in Table 15A-1. The source term activities for gap release and early in-vessel release are separately calculated for all radionuclide groups.

- **Containment Data** (extracted from DCD):
  - Free volume= 3.128×10^6 ft^3
  - Internal radius = 75 ft
  - Effective height = $3.128 \times 10^6 / (\pi \times 75^2) = 177.0$ ft
  - Sprayed region = 2.346×10^6 ft^3 assuming 75% of containment free volume

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- **Surface Area**: Total surface area available to be deposited on the walls of containment is assumed to be the same as that inside containment (700,963 ft^2). Of this area, the surface area of containment wall and operating floor is 9.05×10^4 ft^2.

- **Containment Leakage**: No leakage from the reactor containment building to the environment is assumed in order to maximize the TIDs inside containment.

- **Containment Spray**: According to RG 1.183 guidance, the airborne radioactivity in the containment may be removed by natural deposition and the containment spray system. Their removal rates are a function of time after accident, which is described in Subsection 15.6.5.5.1.1

- **IRWST Volume**: The minimum volume of water sources in containment is 8.6×10^4 ft^3 (2.44×10^9 cm^3).

- **IRWST Source term**: Replace with B ~~The initial source term in the IRWST consists of 40% of halogens in the core inventory, 30% of alkali metal, and small fractions of other fission products~~

- **Radioactive Decay**: The effect of radioactive decay during holdup in the containment is included.

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~~3A~~.1.2. Calculation Method and Model

The radioactive nuclides released from the core escape from the reactor coolant pressure boundary (RCPB) into the containment during a LOCA, are dispersed throughout the containment. This analysis consists of two steps; the first is to determine the activity distribution as a function of time, and the second is to determine the dose contribution from each source to each dose point.

As illustrated in Figure B-3, the ~~The~~ activity distribution, or the locations inside containment at which the dose rate is calculated are as follows:

- Center of Containment Atmosphere (X1)
- Containment Wall Surface (X2)
- Bottom of Containment (Radioactivity in sump water contributes to the exposure rate at the location in containment air space through the concrete shield) (X3)
- Center of Containment IRWST Sump (X4)

And the radioactive source terms that contribute to radiation exposure at any location are as follows:

- Airborne Fission Products in Containment Atmosphere
- Deposited Fission Products on Containment Wall
- Fission Products in IRWST Sump Water

Airborne nuclides in containment are readily absorbed by the spray droplets and thereby removed from the containment atmosphere. The aerosol removal by containment spray, natural deposition, and radioactive decay are considered. The dose rate in containment due to radioactive airborne is calculated by the RUNT-G code. As described above, the following three (3) processes would affect the airborne activity:

- Radioactive decay and sub-sequent daughter products are calculated in the RUNT-G model;

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The initial source term in the IRWST except for noble gases consists of 40% of halogens in the core inventory, 30% of alkali metal, and fractions of other fission products which are addressed in detail in Table 15.6.5-13.

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Depending on the above locations, X1, X2, X3, and X4, the resultant TIDs assigned to specific components or equipment are determined for the bounding analysis as follows:

- Components located below elevation 100' = TIDs at X4 location,
- Components located above elevation 100' = Maximum gamma TID + Maximum beta TID at any location.

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- Removal by the containment spray system; and
- Plate-out on walls and other surface inside containment by natural deposition.

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As shown in Figure ~~2~~ below, the RUNT-G model is developed to simulate the removal of the radioactivity in the containment atmosphere, in the IRWST sump, and on containment walls after the LOCA event.

TIDs for the concentrations of radionuclides at different locations inside containment are determined using the ISOSHLD computer code, which is incorporated in RUNT-G. As illustrated in Figure 3, the containment and the IRWST are modeled as a right cylinder with a free volume of 3.128×10^6 ft^3 (8.86×10^{10} cm^3) with an inner ~~diameter~~ Radius of 75 ft, and a right cylinder with a volume of 8.61×10^4 ft^3 (2.44×10^9 cm^3) and effective height of cylinder of 4.87 ft, respectively. The radioactive materials released in the containment and IRWST sump are assumed to be uniformly distributed throughout the containment atmosphere and the IRWST sump.

Radioactive source terms on the containment wall surface are calculated using the removal rate by natural deposition. The removal by natural deposition consists of gravity settling, thermophoresis, diffusiophoresis, and turbulent diffusion, of which the most dominant process is the removal by gravity settling. The radionuclides deposited on the containment wall and operating floor area are assumed to contribute to the dose at the center of containment. For calculating the dose rate at the center of containment (i.e., dose point of X1), the plate-out of concrete walls is modeled as a point source with a total activity equal to the total activity plate-out on the walls. For the dose rate on the concrete wall, the dose contribution of the plate-out radionuclides is determined by modeling the source as a large disk with a radius of 2.70×10^3 cm. The dose point is set 1.0 cm away from the wall to avoid the singularity at X = 0.

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Table 1 indicates the main input parameters of the ISOSHLD code.

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~~3A~~.2. TIDs in ESF system areas in auxiliary building (outside the containment)

There are two systems, the safety injection system (SIS)/shutdown cooling system (SCS), and the containment spray system (CSS) that re-circulate containment sump water outside of containment during a design basis accident. The analysis of accident TIDs for these systems is discussed below, including the input parameters, assumptions, methods, and models.

The SIS/SCS and the CSS are part of the ESF systems used for mitigation of a LOCA. TIDs for the cubicles containing the ESF components/equipment and from the rooms adjacent to the components are calculated based on the post-LOCA environment. The input parameters, the assumptions, and the evaluation model for running the RUNT-G code are described below.

~~3A~~.2.1. Input parameters and assumptions

- **Source Term**: Source terms for the systems that re-circulate containment sump water outside the containment are based on the source terms that are used for equipment qualification inside containment. Please refer to item ~~3A~~.1.1 above.

- ~~**Radioactive Decay**: The effect of radioactive decay during holdup in the containment is included for duration of 1 year.~~

Radioactive decay and subsequent daughters: The effect of radioactive decay with subsequent daughter products during holdup in the containment is included for a duration of 1 year.

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- **Containment Leakage**: The containment leak rate is the design-basis leak rate specified in the Technical Specifications. For the first 24 hours following a LOCA, the leak rate is assumed to be 0.1 vol.%/day of containment volume and the leak rate is assumed to be 0.05 vol.%/day thereafter.

- **Containment Spray**: According to RG 1.183 guidance, the airborne radioactivity in the containment is removed by natural deposition and the containment spray system. Their removal rates are a function of time after accident, which is described in DCD Subsection 15.6.5.5.1.1

- **Atmospheric Dispersion (χ/Q)**: The relative concentration of the plume is given by the following equation (Ref.1):

$$\chi/Q = (U \cdot C \cdot A)^{-1} \qquad \text{(Eq.1)}$$

Where, U = Wind speed (1 m/sec)
C = Building wake factor (=
A = Cross section area of c

The other areas outside the ABCAEES are not considered in this analysis. All components in those areas such as valves in the post-accident sampling system, which are non-safety related, are infrequently operated and have a relatively small leakage.

- **ABCAEES Envelope Areas**: Following the LOCA, the engineered safety feature actuation signal (ESFAS) actuates the auxiliary building controlled area emergency exhaust system (ABCAEES) .The radioactive source leaked from ESF system (i.e., SI/CS systems) recirculation loop flashes to SI/CS component rooms. The source terms in the ESF systems areas are released to the environment through the ABCAEES, which ventilates the auxiliary building controlled areas I and II including the SC/CS heat exchanger room, component cooling water (CCW) pump room, SI pump room, SC/CS pump and mini-flow heat exchanger room, mechanical penetration room, charging pump room, and auxiliary charging pump room.

  For simplification of the RUNT-G model, the auxiliary building controlled areas I and II are assumed to be one area having a volume of 4.97×10^5 ft^3 (1.40×10^{10} cm^3), which consists of 2.50×10^5 ft^3 (7.08×10^{10} cm^3) and 2.47×10^5 ft^3 (6.99×10^9 cm^3) for auxiliary building controlled areas I and II, respectively. For conservative TIDs calculation of the HVAC system components, these areas are assumed to be ventilated by one of the two air cleanup units (ACUs) in each HVAC line of the ABCAEES. The flow rates through such ACUs are summed to be 6,000 cfm (1.02×10^4 m^3/hr).

  The filter efficiency of the ABCAEES for all species of radioactive nuclides except for noble gases is assumed to be 100% according to the guidance of RG1.89, Appendix I.

- **IRWST Source term**: The initial source term in the IRWST consists of 40% of halogens in the core inventory, 30% of alkali metal, and small fractions of other fission products.

- **ESF Components Leakage**: ~~The~~ maximum anticipated leakage rate through all ESF system components containing the IRWST water source term (i.e., SI/SC/CS components) is calculated to be 0.285 ft^3/hr (8.07×10^3 cm^3/hr). In accordance with this RG 1.183 guidance, the ESF leakage of 0.285 ft^3/hr is doubled to the modeled value of 0.57 ft^3/hr (1.61×10^4 cm^3/hr).

Based on the applicable anticipated leakage rates from each valve and pump in the SI/SC/CS systems and the number of the corresponding components (i.e., valves, and pumps), the

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- **Partition Coefficient**: ~~When radionuclides leaked from SI/SC/CS equipment are entered into the ABCAEES, the partition coefficient of each isotope group is as follows, based on the RG 1.183 guidance:~~

When a portion of radionuclides in the leakage from SI/SC/CS equipment becomes airborne in the auxiliary building and subsequently enters into the ABCAEES, the assumed partition coefficient for each isotope group is as follows (based on the RG 1.183 guidance):

  - Noble gas: 1.0
  - Halogen: 0.1
  - Others nuclides: 0.01 (assumed in order to be conservative)

- **IRWST Volume**: The minimum volume of water sources in containment is 8.6×10^4 ft^3 (2.44×10^9 cm^3).

- **Concrete Structure and Geometry**:

  - The inside radius of the containment building is 75 ft (2.29×10^3 cm).
  - The containment wall is surrounded with ¼ in (6.35×10^{-1} cm) steel liner and 4.5 ft (1.37×10^2 cm) concrete.
  - The containment free volume is assumed to be 3.13×10^6 ft^3 (8.86E+10 cm^3).
  - All structures and equipment inside containment are ignored as shielding materials.
  - The containment steel liner is modeled as iron with a density of 7.86 g/cm^3.
  - The density of the concrete wall is 2.242 g/cm^3.
  - For the direct dose calculation from airborne radioactivity in containment, the containment is modeled as a right cylinder which has the same internal radius and volume as the assumed free volume as illustrated in Figure 3.

- **SI/SC/CS Piping geometry**: ~~Schedule 40S steel pipe and nominal pipe size of 16-inch are assumed for conservatism.~~

All the actual pipe sizes in each cubicle containing the SI/SC/CS components where schedule 40S steel pipe is assumed are taken into account.

B

3B

~~3A~~.2.2. Calculation Method and Model

The integrated RUNT-G and ISOSHLD computer model is used to determine the individual doses that contribute to the overall TIDs.

~~3A~~.2.2.1 Airborne Activity inside the Auxiliary Building and Filter Loading Dose due to Containment Leakage

B-4

As shown in Figure ~~4~~ below, the RUNT-G model is developed to simulate the time-dependent leakage from containment, the dispersion in the atmosphere, and the intake to the auxiliary building.

This model is subdivided into three types as gaseous (includes noble gas & organic halogen), elemental and particulate halogen. For the first 24 hrs following a LOCA, the leak rate is assumed to be 0.1 %/day of containment volume and the leak rate is assumed to be 0.05 %/day thereafter (i.e., paths of L_{23} and L_{34}). After the onset of the LOCA event, the wash-out phenomena by containment spray and natural deposition are taken into consideration as the leakage from BARRIER 1 to the sump (i.e., path of L_{28}). These phenomena are only applicable to non-noble gases. The atmospheric dispersion and determination of the intake activity of the ABCAEES are calculated by using the fraction factor on FILTER 1, which is equal to the multiplication product (1.90×10^{-3}) of (χ/Q (relative concentration of the plume, 6.725×10^{-4}

A

m^3/hr

sec/m^3) × W (ABCAEES intake flow rate, 1.02 × 10^4 ~~m^{3}hr~~) and a conversion factor of 1 hour to 3600 seconds). The leak rate from containment is multiplied by the fraction factor to produce radioactivity taken into the auxiliary building via the ABCAEES. Therefore, BARRIER 2 means airborne radioactivity inside the auxiliary building. The release rate to the atmosphere through the auxiliary building (i.e., paths of L_{45} and L_{55}) depends on the cubicle volume and HVAC flow rate.

B-2 and B-3

B-5 and B-6

TIDs for airborne activity in the auxiliary building (i.e., BARRIER 2) and the filter loading of ABCAEES (i.e., FILTER 2) are determined by using the ISOSHLD code which is incorporated in the RUNT-G computer code. Tables ~~2 and 3~~ indicate the main input parameters of the ISOSHLD code, and the ISOSHLD models are shown in Figures ~~5 and 6~~.

3B

~~3A~~.2.2.2 Airborne Activity and Filter Loading Dose due to SI/SC/CS Leakage

B-7

As shown in Figure ~~7~~, the RUNT-G model is developed to simulate the time-dependent leakage from SI/SC/CS equipment, which are located at elevation 55'-0" in the auxiliary building, and the atmospheric dispersion to the environment through the auxiliary building cubicles.

It is assumed that all the ESF leakages of 0.57 ft^3/hr (1.61 × 10^4 cm^3/hr) are retained on the floor of the corresponding compartments in the auxiliary building (i.e. BARRIER 1), and some of the iodines are flashed and become airborne in the auxiliary building. Then, the airborne iodine radioactivity in cubicles of the auxiliary building is released to the environment via the ABCAEES filter (i.e., paths of L_{23} and L_{38}). The release rate to the environment through the auxiliary building depends on cubicle volume and HVAC flow rate.

B

B-2 and B-3

B-5 and B-6

TIDs for airborne activity in the auxiliary building (i.e., BARRIER 1) and the filter loading of ABCAEES (i.e., FILTER 1) are determined by using the ISOSHLD code which is incorporated in the RUNT-G computer code. The main input parameters of the ISOSHLD code and ISOSHLD model are the same as those in Tables ~~2 and 3~~, and in Figures ~~5 and 6~~, respectively.

3B

~~3A~~.2.2.3 Post-LOCA Direct Dose from the Airborne Source in Containment

B-8

As shown in Figure ~~8~~, the RUNT-G model is developed to calculate the time-dependent activity in the containment, which is divided into three types as gaseous (includes noble gas & organic halogen), elemental and particulate halogen.

After the onset of the LOCA event, leakage from the core to the atmosphere of the containment is modeled as the leakage from the SOURCE to BARRIER 1 (i.e., path L_{12}). The wash-out phenomena by containment spray and natural deposition are described as the leakage from BARRIER 1 to the sump (i.e., path L_{24}), which are only applicable to non-noble gases.

B-4

TIDs from airborne activity in containment (i.e., BARRIER 1) are determined by using the ISOSHLD code which is incorporated in the RUNT-G computer code. Table ~~4~~ indicates the main input parameters of the ISOSHLD code, and the ISOSHLD model is shown in Figure ~~9~~.

B-9

3B

~~3A~~.2.2.4 Post-LOCA Direct Dose from SI/SC/CS Piping

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B-10

As shown in Figure ~~10~~, the RUNT-G model is developed to calculate the time-dependent IRWST source activity in containment. The radioactivity concentration in the IRWST water after the initiation of the LOCA event is used as the source term in SI/SC/CS circulating water.

B-5

TIDs from direct radiation in the IRWST source term (i.e., BARRIER 1) are determined by using the ISOSHLD code which is incorporated in the RUNT-G computer code. Table ~~5~~ indicates the main input parameters of the ISOSHLD code, and the ISOSHLD model is the same as that in Figure ~~9~~.

B-9

~~3B~~

~~3A~~.2.2.5 Post-LOCA Direct Dose from SI/SC/CS Components

TIDs for SI/SC/CS equipment are calculated using the result of the TID calculation for the SI/SC/CS pipes.

For pumps which have the same diameter as the pipe, the TIDs for the pumps are expected to be lower than the TIDs for pipes having the same diameter because of the shield effect by the enclosing steel casing. Therefore, TIDs for the SI/SC/CS piping during the LOCA condition can be conservatively used for pumps in the SI/SC/CS systems. ~~In the case of heat exchangers, because of the shielding effect by the internal steel and the cooling water~~, TIDs for pipes having the same diameter with the effective diameter of the heat exchanger can be expected to yield conservative TID values. The effective diameter of the heat exchanger can be calculated as follows:

B

Replace with D

2a

$$D = (2N)^{1/2} \times d \qquad \text{for U-tube Type Heat Exchanger} \qquad (\text{Eq.}\cancel{2})$$

$$D = (N)^{1/2} \times d \qquad \text{for one-through Type Heat Exchanger} \qquad (\text{Eq.}\cancel{2})$$

2b

Where, N = Number of tubes in heat exchanger
d = Diameter of tube
D= Effective diameter of heat exchanger

~~3B~~

~~3A~~.3. TIDs in fuel handling area in auxiliary building (outside the containment)

TIDs for components in the fuel handling area are calculated based on the post-FHA (fuel handling accident) environment. The input parameters, assumptions, and evaluation model for running the RUNT-G code are described below.

~~3B~~

~~3A~~.3.1. Input Parameters and Assumptions

- **Source Term**: For the purpose of conducting a conservative analysis that bounds most cases, all of the fuel rods in a fuel assembly are assumed to be damaged and all the gap activity in the damaged rods is assumed to be instantaneously released into the spent fuel pool, where total gap activities are 10% of Kr-85, 8% of I-131, 5% of other iodines and noble gases, and 12% of alkali metals in fuel rods. The retention of noble gases in the pool is negligible and the iodine above the pool consists of 57% of elemental iodine and 43% of organic iodine, considering the overall effective decontamination factor of 200 for iodine. The source term for the FHA event is described in detail in DCD Subsection 15.7.4.2 and Table 15.7.4-1.

D

(New Paragraph) The heat exchangers can be also modeled with a single pipe with an effective diameter which is derived from the equation 2a or 2b below. For the tube region, where the heat exchanger tubes and shell are located, the shielding effect of internal steel and cooling water in the shell side are not taken into account. For the plenum region, where the tube side inlet and outlet are located, the same wall thickness and diameter with the tube region are applied; i.e., the wall thickness and volume are decreased. The impact of decreased radioactivity due to the decreased volume in that region is negligible since the relative impact of the decreased wall thickness is larger due to the simplification. Therefore,

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- **Fuel Handling Area**: After the fuel handling accident, airborne radioactive materials in the fuel handling area are vented to the environment via fuel handling area ACUs over a two hour time period. This HVAC system emergency exhaust flow rate is 5,000 cfm/ACU. The total free volume covered by this HVAC system is 8.77×10^5 ft^3. The removal efficiency of the carbon absorbers and HEPA filter is assumed to be 100%.

- **HEPA filter and Charcoal Densities**: The HEPA filter media and charcoal densities are both assumed to be 0.48 g/cm^3 for conservatism.

- **Radioactive decay and subsequent daughters**: The effect of radioactive decay with subsequent daughter products during holdup in the containment is included for a duration of 1 year.

B

~~3A~~3B.3.2. Calculation Method and Model

As shown in Figure ~~11~~ B-11, the RUNT-G model is developed to simulate time dependent activity in the fuel handling area after the onset of the FHA event.

Leakage from the fuel assembly to the atmosphere of the fuel handling area is modeled as the leakage from the SOURCE to BARRIER 1 (i.e. path of L_{12}). Airborne activity in the fuel handling area is released to the environment through the fuel handling area emergency HVAC system (i.e., paths of L_{23} and L_{38}). The release rate to the environment through the fuel handling area depends on cubicle volume and HVAC flow rate. The reduction of the amount of radioactivity by deposition and/or plate-out on structure surfaces is not considered for the reason of conservatism.

TIDs for airborne activity in the fuel handling area (i.e., BARRIER 1) and the filter loading of the fuel handling area emergency HVAC system (i.e., FILTER 1) are determined by using the ISOSHLD code which is incorporated in the RUNT-G computer code. The main input parameters of the ISOSHLD code for airborne activity in the fuel handling area is given in Table ~~6~~ B-6, and the other main parameters and ISOSHLD model are the same as those in Table ~~3~~ B-3, and Figures ~~5 and 6~~ B-5 and B-6, respectively, presented above.

~~3A~~3B.4. TIDs in the main steam valve house inside the auxiliary building (outside the containment)

TIDs for components in the main steam valve house (MSVH) are calculated based on the post-MSLB (main steam line break) environment. The input parameters, assumptions, and evaluation model for running the RUNT-G code are described below.

~~3A~~3B.4.1. Input Parameters and Assumptions

- **Source Term**: Per RG 1.183, Appendix E, Section 2, for the main steam line break accident, the release from the breached fuel is based on the estimate of the number of fuel rods assumed to have experienced Departure from Nucleate Boiling (DNB) and the assumption that 5% of the core inventory of the noble gases and iodines is in the fuel gap, except for Kr-85 at 10% and I-131 at 8%.

  The expected number of fuel rods in DNB is assumed to be 1% of the core where the failed fuel is modeled with a radial peaking factor of 1.8. There is no fuel melt expected during the MSLB. To determine the activity in the steam generator resulting from primary-to-secondary leakage, a primary coolant of 2.744×10^5 kg and primary to secondary (P-T-S) leakage of 0.6 gpm are used.

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The initial secondary coolant source term is assumed to be at the Technical Specification limit of 3.7×10^3 Bq/g (0.1μCi/g) Dose Equivalent (DE) I-131 and is given in DCD Table 15A-9. The source term for the MSLB event is described in detail as specified in DCD Subsection 15.1.5.3.3 and Table 15.1.5-12.

- **Main Steam Valve House**: The volume of each MSVH (i.e., room number of 137-A31C/D) is 123,955 ft^3.

Radioactive decay and subsequent daughters: The effect of radioactive decay with subsequent daughter products during holdup in the containment is included for a duration of 1 year.

3B

~~3A~~.4.2. Calculation Method and Model

As shown in Figure ~~12~~, the RUNT-G model is developed to simulate the time-dependent activity in the MSVH.

B-12

After the onset of the MSLB event, the radioactivity leaked from the broken steam piping to the MSVH, which includes secondary coolant and RCS coolant activities, is modeled as the leakage from the SOURCE to BARRIER 1 (i.e., path of L_{12}). The reduction of the amount of radioactivity by deposition and/or plate-out on the steam piping or structure surfaces is not considered.

B

B-7

TIDs from airborne activity in the MSVH (i.e., BARRIER 1) ~~are~~ determined by using the ISOSHLD code which is incorporated in the RUNT-G computer code. Table ~~7~~ indicates the main input parameters of the ISOSHLD code, and the ISOSHLD model is the same as that shown in Figure ~~5~~.

B-5

~~3B~~.6

~~3A.5.~~ References

1. USAEC, "Meteorology and Atomic Energy", 1968.

B.5

~~3B.5~~ ~~B.1.1.~~ Summary of Accident TID Calculation

~~Following~~ the requirements in RG 1.89, the TIDs are adjusted by another 10% EQ safety margin for uncertainty.

In accordance with

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Table ~~3A-~~1 Main ISOSHLD Input Parameters for Airborne Activity in Containment (1 of 3)

Geometry		ISOSHLD Parameter	Values
1) TID from Airborne Activity in Containment			
Source Dimension	Shape	IGEOM	Cylindrical Source (Immersion Dose Model)
	Height[1)]	SLTH	5.40×10^3 cm
	Radius	T(1)	2.29×10^3 cm
	Volume[2)]	N/A	4.81×10^9 cm^3
Source Characteristic	Material	N/A	Air
	Density	N/A	1.29×10^{-3} g/cm^3
Dose Point X1	X	X (=SLTH/2)	2.70×10^3 cm
	Y	N/A	0.0 cm
	Z	N/A	0.0 cm
Dose Point X2	X	N/A	2.70×10^3 cm
	Y	DELR	2.70×10^3 cm
	Z	N/A	0.0 cm
Dose Point X3	X	X (=SLTH/2)	0.0 cm
	Y	N/A	0.0 cm
	Z	N/A	0.0 cm

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Table ~~3A-~~1 Main ISOSHLD Input Parameters for Airborne Activity in Containment (2 of 3)

Geometry		ISOSHLD Parameter	Values
2) TID from Radioactivity in IRWST Sump: Above IRWST Sump			
Source Dimension	Shape	IGEOM	Cylindrical Source & Slab Shield on Cylinder End
	Height[1)]	T(1)	1.49×10^2 cm
	Radius	SLTH	2.29×10^3 cm
	Volume[2)]	N/A	2.44×10^9 cm^3
Source Characteristic	Material	N/A	Water
	Density	N/A	1.0 g/cm^3
Air in Containment	Thickness	T(2)	1.37×10^2 cm
	Material	N/A	Air
	Density	N/A	1.29×10^{-3} g/cm^3
Dose Point X1 & X2[1)]	X	X (=T(1)+T(2))	2.85×10^3 cm
	Y	N/A	0.0 cm
	Z	N/A	0.0 cm
Dose Point X3	X	X (=T(1)+2.54cm))	1.51×10^2 cm
	Y	N/A	0.0 cm
	Z	N/A	0.0 cm
3) TID from Radioactivity in IRWST Sump: Within IRWST Sump			
Source Dimension	Shape	IGEOM	Cylindrical Source (Immersion Dose Model)
	Height[1)]	SLTH	1.49×10^2 cm
	Radius	T(1)	2.29×10^3 cm
	Volume[2)]	N/A	4.81×10^9 cm^3
Source Characteristic	Material	N/A	Water
	Density	N/A	1.00 g/cm^3
Dose Point X4	X	X (=SLTH/2)	74.3 cm
	Y	N/A	0.0 cm
	Z	N/A	0.0 cm

B-
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~~3B-~~

Table ~~3A-~~1 Main ISOSHLD Input Parameters for Airborne Activity in Containment (3 of 3)

Geometry		ISOSHLD Parameter	Values
4) TID by Plate-out Radioactivity on Containment Wall: Center of Containment			
Source Dimension	Shape	IGEOM	Point Source
	Radius	T(1)	2.29×10^{3} cm
Source Characteristic	Material	N/A	Air
	Density	N/A	1.29×10^{-3} g/cm^{3}
Dose Point X1 & X3[2)]	X	X (=T(1))	2.29×10^{3} cm
	Y	N/A	0.0 cm
	Z	N/A	0.0 cm
5) TID by Plate-out Radioactivity on Containment Wall: Wall of Containment			
Source Dimension	Shape	IGEOM	Disk Source
	Thickness	T(1)	0.0 cm
	Radius	SLTH	2.70×10^{3} cm
Source Characteristic	Material	N/A	Air
	Density	N/A	1.29×10^{-3} g/cm^{3}
Dose Point X2[3)]	X	X	1.00 cm
	Y	N/A	0.0 cm
	Z	N/A	0.0 cm

1) The dose rate at containment wall surface is assumed to be the same as that at center of containment atmosphere (i.e., dose point of X1) from the source in IRWST sump water for conservatism. Therefore, every parameter has the same value with the input parameter for dose point X1.
2) The dose rate at bottom of containment is assumed to be the same as that at center of containment atmosphere (i.e., dose point of X1) from the source deposited on the surface is used as. Therefore, every parameter has the same value with the input parameter for dose point X1
3) The dose point is set 1 cm away from the wall to avoid the singularity at X = 0.

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Table ~~3A-~~2 Main ISOSHLD Input Parameters for Airborne Activity in Auxiliary Building

Geometry		ISOSHLD Parameter	Values
Source Dimension	Shape	IGEOM	Cylindrical Source (Immersion Dose Model)
	Height[1)]	SLTH	6.10×10^{2} cm
	Radius	T(1)	1.58×10^{3}cm
	Volume[2)]	N/A	4.81×10^{9} cm^{3}
Source Characteristic	Material	N/A	Air
	Density	N/A	1.29×10^{-3} g/cm^{3}
Dose Point	X	X (=SLTH/2)	3.05×10^{2} cm
	Y	N/A	0.0 cm
	Z	N/A	0.0 cm

(1) Height of rooms in auxiliary building is assumed to be 610cm (=20ft)

(2) Volume of rooms or cubicles, which contains components/equipment in the ESF systems, ranges from 2.21×10^{8} cm^{3} (=7.82×10^{3} ft^{3}) to 9.15×10^{8} cm^{3} (=3.23×10^{4} ft^{3}), but it is conservatively assumed to be 4.81×10^{9} cm^{3} (= 1.70×10^{5} ft^{3}) as a bounding volume, thus leading to maximization of the potentially expected TIDs for the corresponding components/equipment.

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Table ~~3A-~~3 Main ISOSHLD Input Parameters for ABCAEES Filter

Geometry		ISOSHLD Parameter	Values
Source Dimension	Shape	IGEOM	Rectangular Source
	Width (X)	T(1)	6.10×10^{1} cm
	Length (Y)	Y	6.10×10^{1} cm
	Height (Z)	SLTH	6.10×10^{1} cm
	Volume	N/A	2.27×10^{5} cm^{3}
Source Characteristic	Material[1)]	N/A	Carbon
	Density	N/A	4.80×10^{-1} g/cm^{3}
Dose Point[1)]	X	X (=T(1) + T(2))	3.30×10^{1} cm
	Y	YP	3.05×10^{1} cm
	Z	SP	3.05×10^{1} cm

(1) Be assumed to be 2.54cm away from HVAC ACU source term

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Table ~~3A-4~~ Main ISOSHLD Input Parameters for Direct Dose from Containment

Geometry		ISOSHLD Parameter	Values
Source Dimension	Shape	IGEOM	Cylindrical Source & Cylindrical Shield
	Height[1)]	SLTH	5.40×10^3 cm
	Radius	T(1)	2.29×10^3 cm
	Volume	N/A	8.86×10^{10} cm^3
Source Characteristic	Material	N/A	Air
	Density	N/A	1.29×10^{-3} g/cm^3
Containment Concrete Shield Wall[2)]	Thickness	T(2)	1.37×10^2 cm
	Material	N/A	Concrete
	Density	N/A	2.242 g/cm^3
Dose Point	X	X	2.42×10^3 cm
	Y	Y (=SLTH/2)	2.70×10^3 cm
	Z	N/A	0.0 cm

(1) Calculated based on the containment plane area of 1.64E $\times$ 10^{7}cm^2 (=1.77 $\times$ 10^{4}ft^2)

(2) The shielding effect of the 137cm (=4.5ft) containment cylindrical concrete wall is only considered. The additional shielding effect due to structures in the auxiliary building is not considered for conservatism.

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Table ~~3A-~~5 Main ISOSHLD Input Parameters for Direct Dose from SI/SC/CS Piping

Geometry		ISOSHLD Parameter	Values
Source Dimension	Shape	IGEOM	Cylindrical Source & Cylindrical Shield
	Height1)	SLTH	6.10×10^{2} cm
	Radius	T(1)	1.90×10^{1} cm
	Volume	N/A	6.95×10^{5} cm^{3}
Source Characteristic	Material	N/A	Water
	Density	N/A	1.00 g/cm^{3}
Pipe Wall2)	Radius	T(2)	1.27 cm
	Material	N/A	Steel
	Density	N/A	7.86 g/cm^{3}
Air	Radius	T(3)	3.05×10^{1} cm
	Material	N/A	Air
	Density	N/A	1.29×10^{-3} g/cm^{3}
Concrete Shield Wall	Radius	T(4)	Wall Thickness (T) of Adjacent Room
	Material	N/A	Concrete
	Density	N/A	2.242 g/cm^{3}
Dose Point	X	X	5.08E+01 cm w/o concrete wall 3) (5.33×10^{1} + T) cm w/ concrete wall
	Y	Y (=SLTH/2)	3.05×10^{2} cm
	Z	N/A	0.0 cm

(1) Piping length is assumed to be 20ft.

(2) Only shielding effect of the pipe wall of 1.27cm is considered for conservatism

(3) Assumed to be 1ft away from the SI/SC/CS piping

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Figure ~~3A-~~1 A Simplified Flowchart for Determination of Accident TIDs

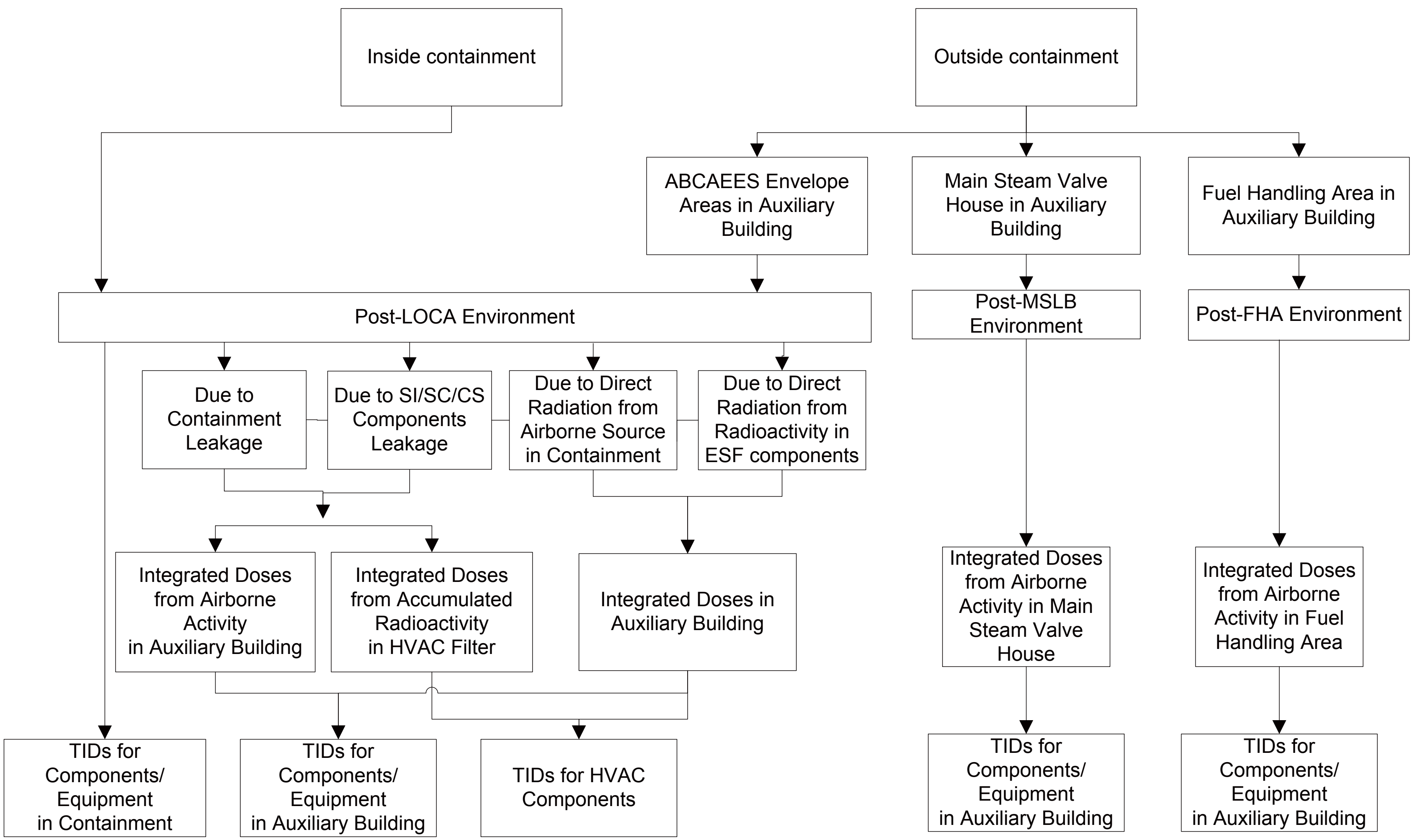

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Figure ~~3A-~~2 RUNT-G Model to Calculate radioactivity in containment, IRWST sump, and walls

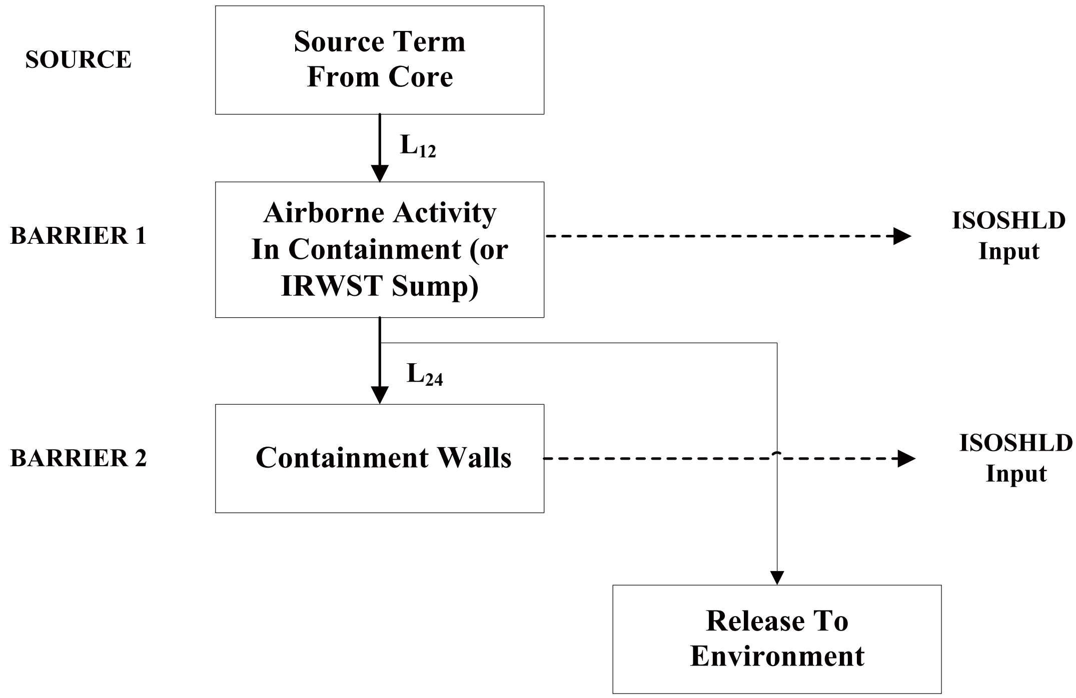

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Figure ~~3A-~~3 ISOSHLD Model for Dose Evaluation Inside Containment

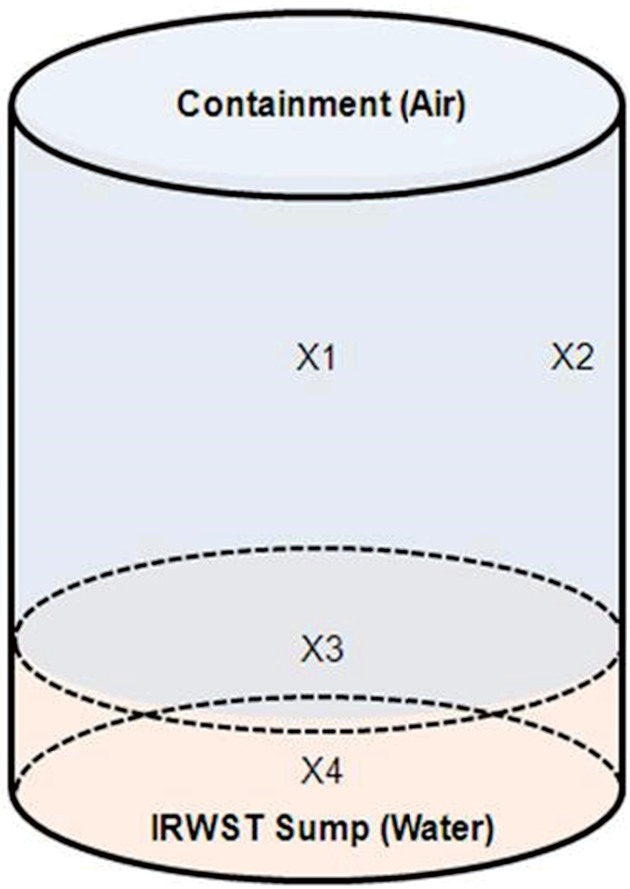

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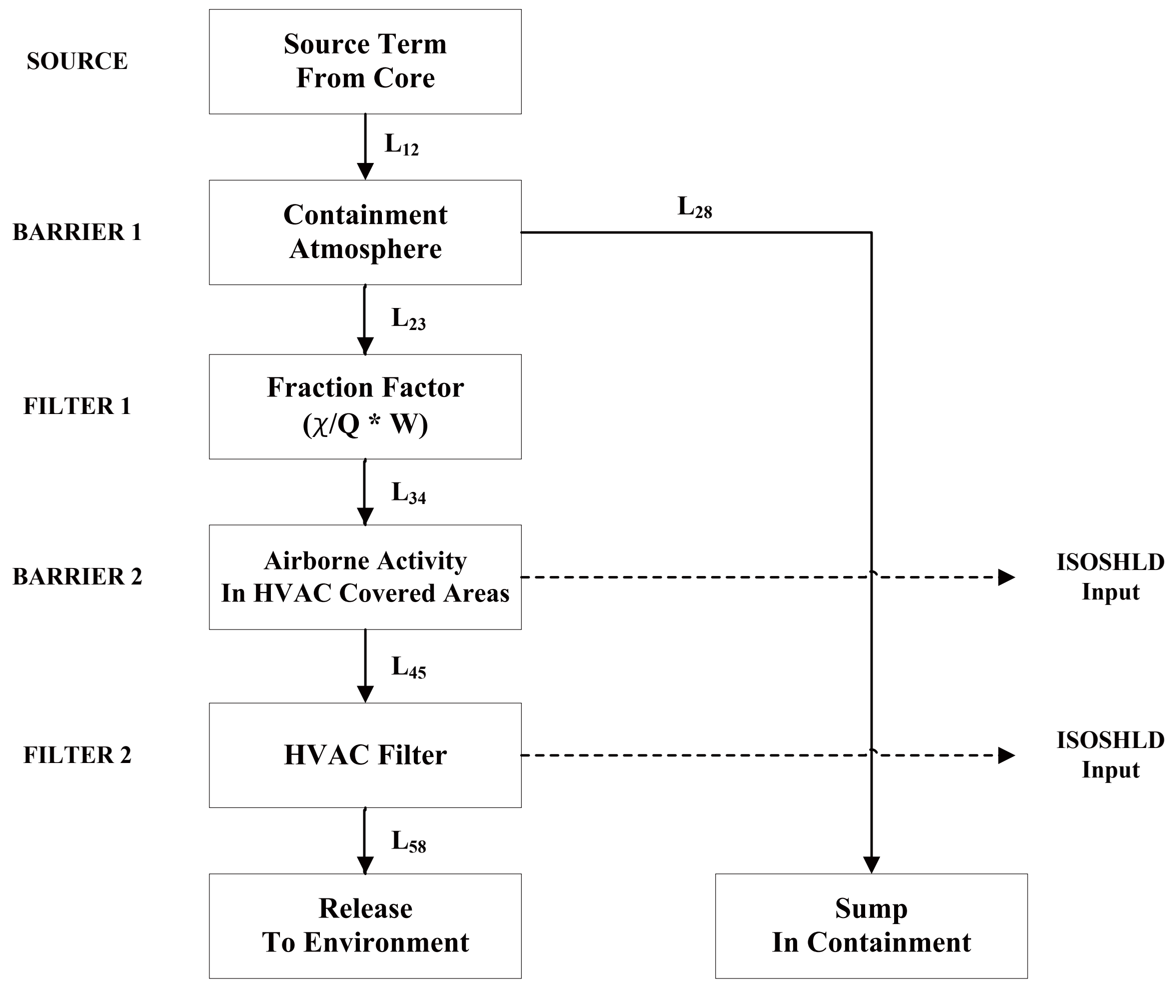
Figure ~~3A-~~4 RUNT-G Model for Containment Leakage



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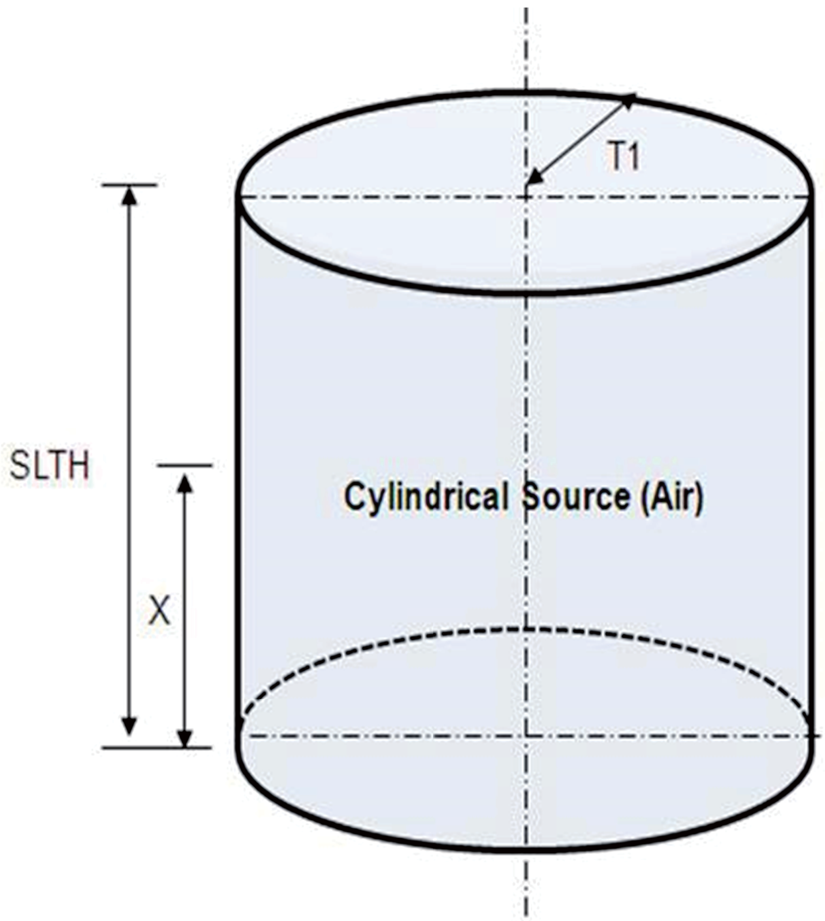
B- 3B

Figure ~~3A-~~5 ISOSHLD Model for Immersion Dose Evaluation



RAI 176-8089 - Question 03.11-11

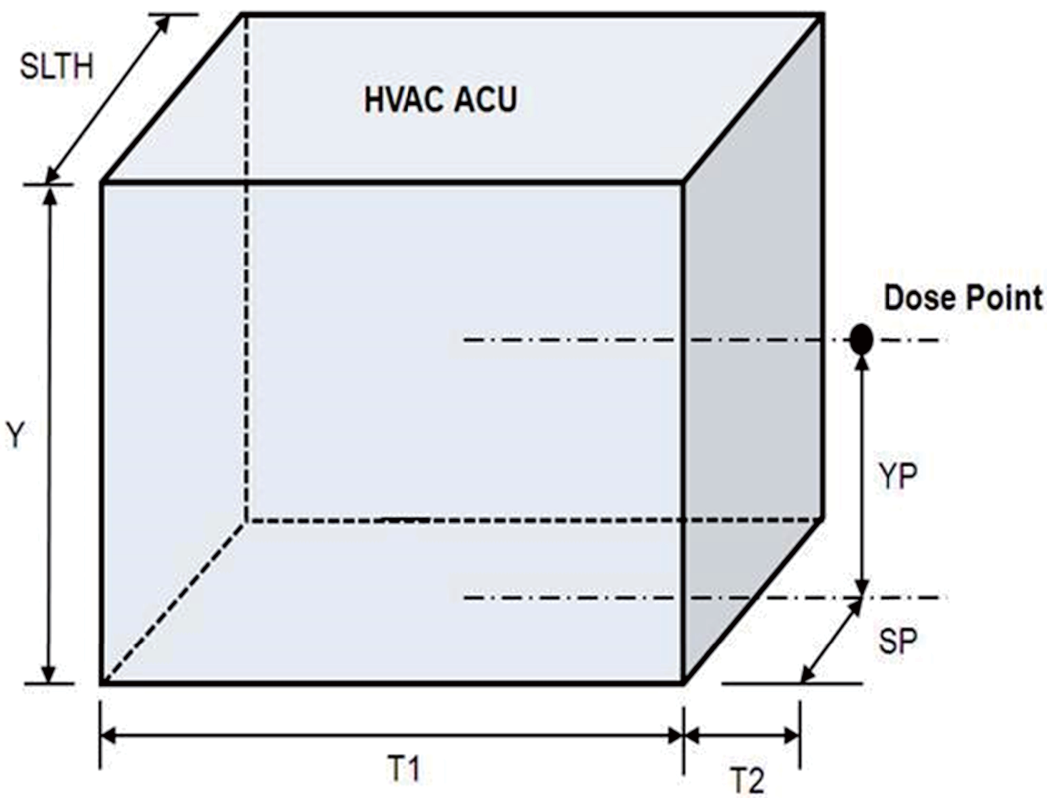
RAI 176-8089 - Question 03.11-9

RAI 176-8089 - Question 03.11-11 Rev.1

A

B- 3B-

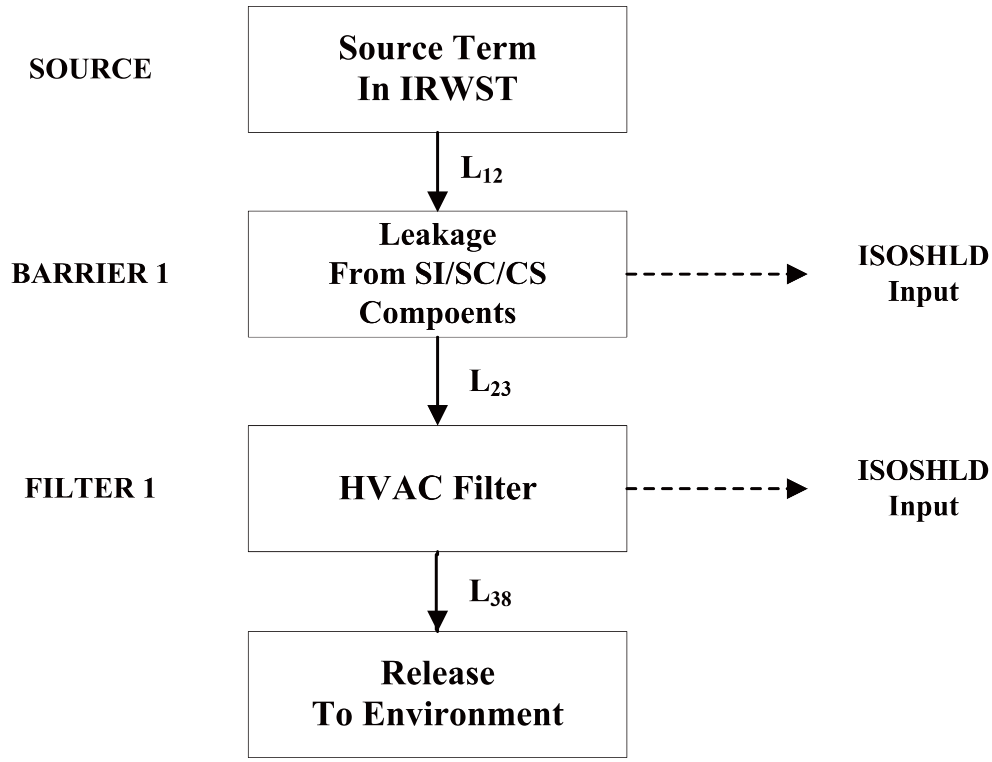
Figure ~~3A-~~6 ISOSHLD Model for ABCAEES Filter



A

B- ~~3B-~~

Figure ~~3A-~~7 RUNT-G Model for SI/SC/CS Components Leakage



RAI 176-8089 - Question 03.11-11

RAI 176-8089 - Question 03.11-9

RAI 176-8089 - Question 03.11-11 Rev.1

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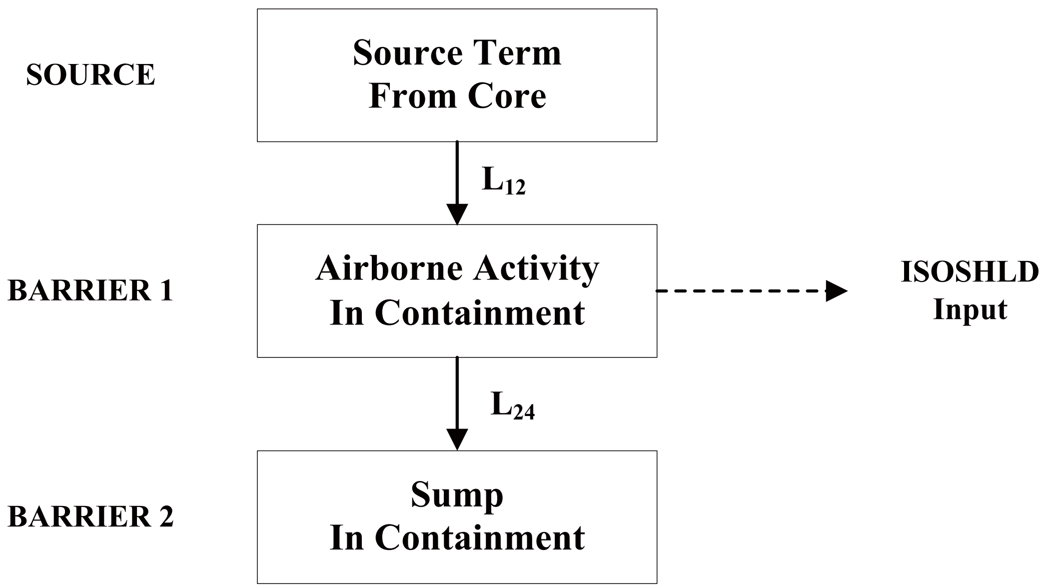
B- 3B-

Figure ~~3A-~~8 RUNT-G Model for Direct Shine from Containment



RAI 176-8089 - Question 03.11-11

RAI 176-8089 - Question 03.11-9

RAI 176-8089 - Question 03.11-11 Rev.1

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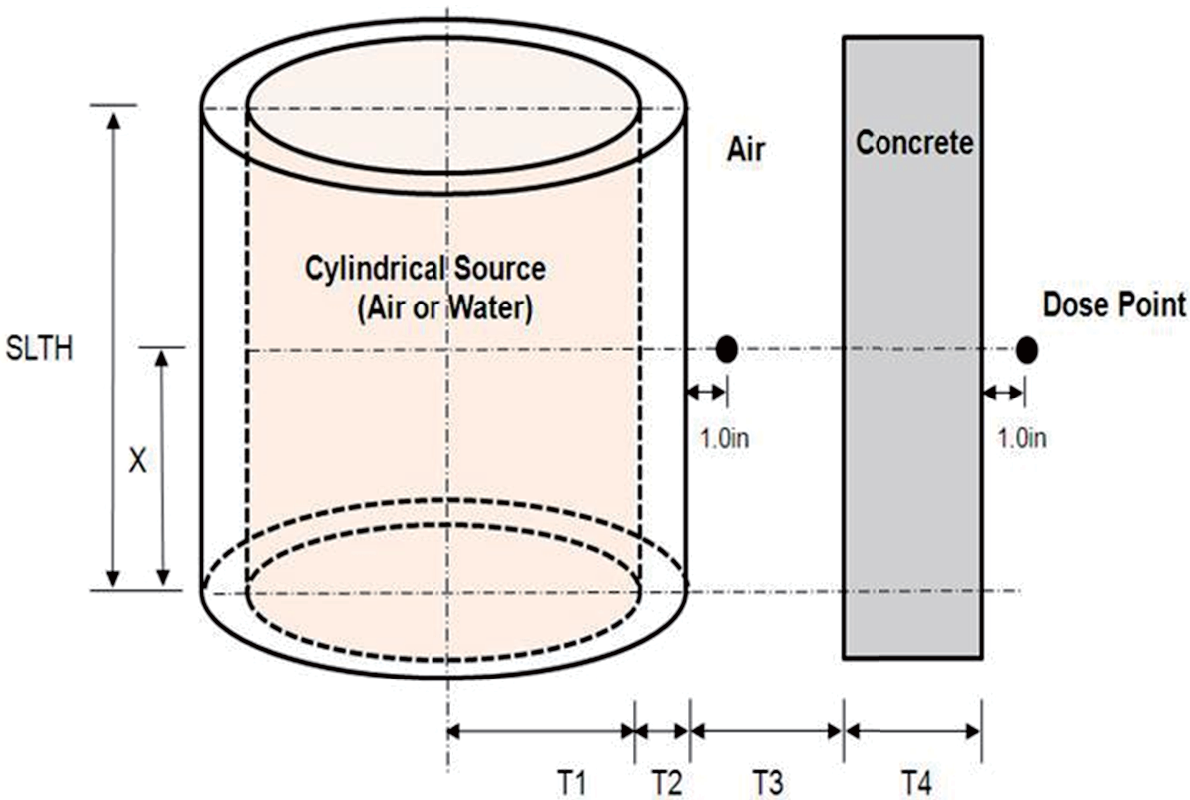
B- ~~3B-~~

Figure ~~3A-~~9 ISOSHLD Model for Direct Shine from Containment or SI/SC/CS Piping

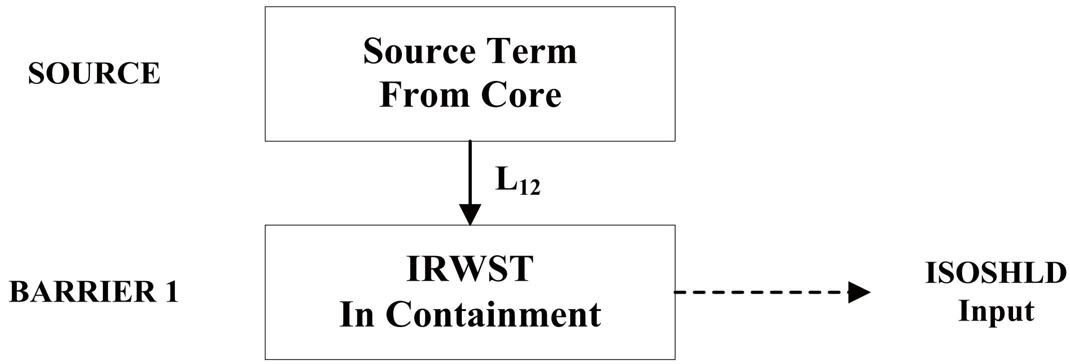


A

B-

~~3B-~~

Figure ~~3A-~~10 RUNT-G Model for Direct Shine from ESF Components



RAI 176-8089 - Question 03.11-11

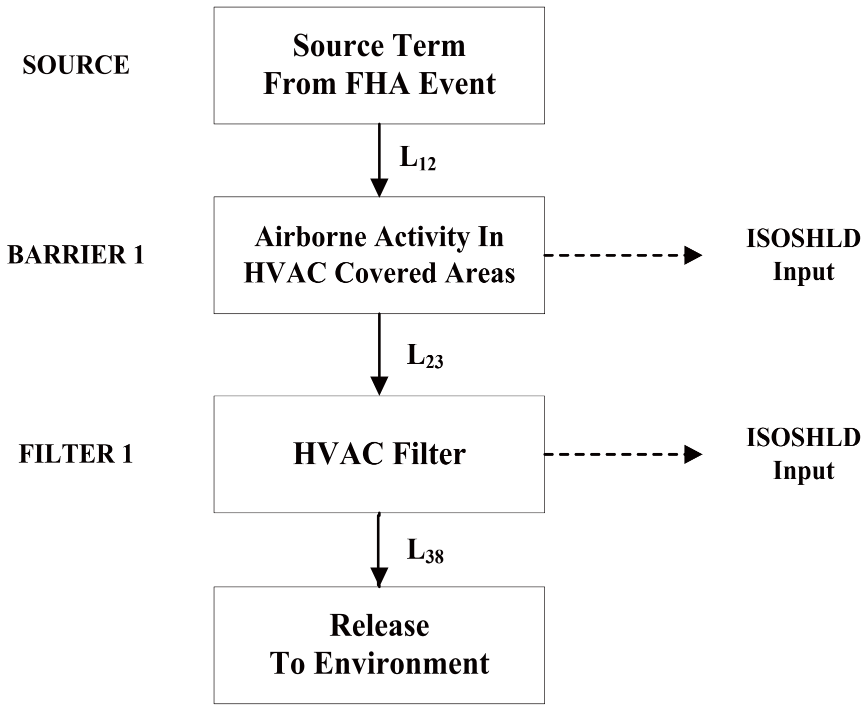
RAI 176-8089 - Question 03.11-9

RAI 176-8089 - Question 03.11-11 Rev.1

A

B- ~~3B-~~

Figure ~~3A-~~11 RUNT-G Model for Fuel Assembly Leakage



RAI 176-8089 - Question 03.11-11

RAI 176-8089 - Question 03.11-9

RAI 176-8089 - Question 03.11-11 Rev.1

A

B-

~~3B-~~

Figure ~~3A-~~12 RUNT-G Model for Radioactivity Release from MSLB Event