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October 30, 2014

Mr. Rajender Auluck
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Policy and Support Branch
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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Transmittal of White Paper HCVS-WP-02, *Sequences for HCVS Design and Method for Determining Radiological Dose from HCVS Piping*, Revision 0, October 23, 2014

Project Number: 689

Dear Mr. Auluck:

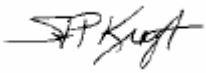
The Nuclear Energy Institute (NEI),[1] on behalf of the nuclear industry, is pleased to submit to the U.S. Nuclear Regulatory Commission (NRC) NEI HCVS-WP-03 – *Sequences for HCVS Design and Method for Determining Radiological Dose from HCVS Piping*, Revision 0, October 23, 2014. The information contained in this white paper will be used, in part, by NRC licensees to implement the requirements of NRC Order EA-13-109, *Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions*, June 6, 2013. We request that the NRC review and endorse NEI HCVS-WP-02 Revision 0.

Earlier drafts of this white paper were presented and discussed at several NRC public meetings. We believe that Revision 0 responds to NRC staff comments. If the NRC staff wishes to hold additional meetings to further discuss this or other topics related to licensee implementation of EA-13-109, please let us know.

[1] The Nuclear Energy Institute (NEI) is the organization responsible for establishing unified industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all entities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel cycle facilities, nuclear materials licensees, and other organizations and entities involved in the nuclear energy industry.

If you have any questions or require additional information, please do not hesitate to contact me.

Sincerely,



Steven P. Kraft

Attachment

HCVS-WP-02: Sequences for HCVS Design and Method for Determining Radiological Dose from HCVS Piping

Revision 0, October 23, 2014

PURPOSE:

To provide a generic, simple, and consistent approach that can be used by the BWR fleet to determine when the vent will initially need to function, number of vent cycles to be assumed in the design, and the radiological impact of HCVS operation.

DESIGN PARAMETER DISCUSSION:

The design of the Hardened Containment Vent System (HCVS) in response to Order EA-13-109 needs to accommodate severe accident and the non-severe accident sequences expected under EA-12-049. The three design parameters to be developed include:

1. Initial time that HCVS needs to be functional.
2. Number of open/closed cycles expected for HCVS operation during initial 24 hour period.
3. Radiation environment in the vicinity of HCVS components and associated operator exposure.

In order to facilitate the development of these design parameters, it is important to understand the types of accident scenarios that would require operation of the HCVS. Figure 1 has been developed to provide information from both strategies to prevent core damage as described in the response to Order EA-12-049 along with consideration of accidents that could result in core damage. Figure 1 represents a type of event tree showing how mitigation actions to prevent core damage may fail and the general timeline for each accident progression. The upper pathway (Case 1) represents successful implementation of the FLEX strategy to prevent core damage. This timeline is obtained from a plant's overall integrated plan in response to Order EA-12-049. Note that anticipatory venting occurs at 5 hours in the example provided.

Case 2 involves successful implementation of the mitigation strategy, however, at some time later in the event, injection is assumed to become unavailable. This type of scenario was modeled in SECY-12-0157 and represented a loss of DC power at 16 hours and subsequent loss of RCIC at about 18 hours due to water entering the RCIC steam supply line. Without injection, this scenario leads to level dropping to below top of active fuel (TAF) at about 23 hours. In SECY-12-0157, pressure in the drywell exceeded 60 psig at about the same time (23 hours) and the wetwell vent was opened as described in the plant procedures.

Case 3 represents a potential scenario where RCIC does not inject. This scenario progresses to level dropping to below TAF in about 30 minutes with the onset of core damage at about 1 hour. This particular scenario can be seen as a sensitivity case for the short term station blackout event in the State-of-the-Art Reactor Consequence Analysis (SOARCA). Although both STSBO scenarios were considered unlikely, the sensitivity case with RCIC failure to inject was less likely than the base case STSBO scenario. The SOARCA sensitivity results indicate that vessel breach occurs at about 8 hours into the event. The SOARCA analysis did not model wetwell venting, however, from the drywell pressure data provided, the drywell pressure exceeds 60 psig at the time of vessel breach.

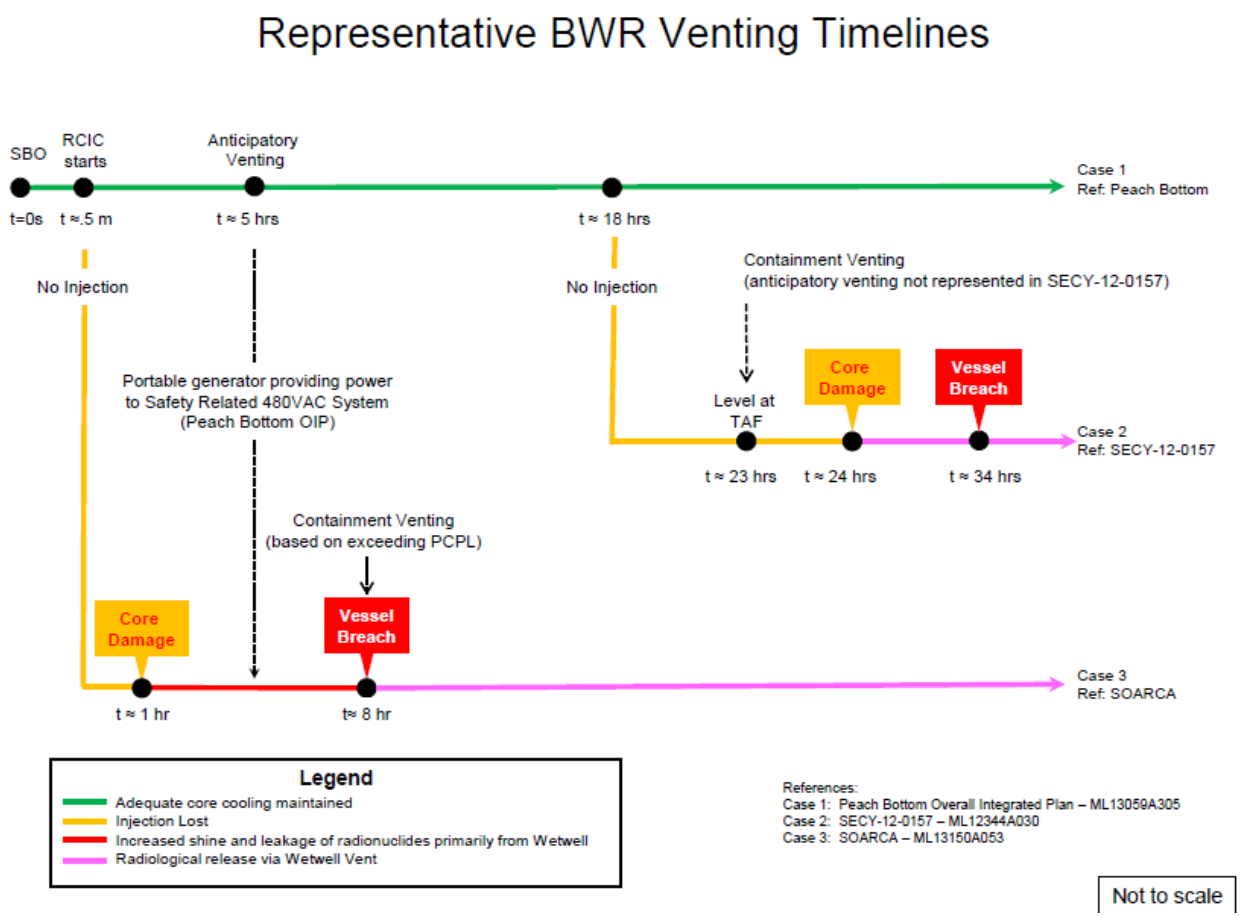


Figure 1 – Representative BWR venting Timelines

These sequences are based on NRC analysis of severe accidents and are suitable for estimates for venting of the containment during a severe accident. These sequences provide a bounding set of sequences and thus can be used to develop the HCVS design parameters addressed below.

1. <u>Timing of Initial Vent Opening</u>

   From Figure 1, the most limiting time when the HCVS needs to be operational is associated with the mitigation strategy in response to Order EA-12-049. This time, 5 hours, would represent the shortest time in which the HCVS would need to be operational. Each plant should modify this time based on plant specific analysis and compare the initial venting time with those from Case 2 and Case 3. The shortest time should be selected.

2. <u>Number of Vent Cycles</u>

   The next design parameter for the HCVS is the number of open/closed cycles expected during the first 24 hours of operation. Each of the 3 cases can be reviewed to determine the number of expected cycles. Where it is expected that plants may implement

venting control strategies differently, the information provided here is meant to guide the process and provide some generic results to inform the assessment.

- For Case 1, the result could be as easy as a single cycle requirement. For some of the plant integrated plans for responding to EA-12-049, anticipatory venting would involve opening the wetwell vent and leaving it open. Closing of the vent would be expected once containment heat removal was established, no longer requiring the vent.

  The NRC SOARCA analysis included an investigation into vent cycling for a mitigated long term station blackout event. The scenario included reactor vessel depressurization at 1 hour and transfer to a portable injection pump at 4 hours when station batteries were assumed unavailable and RCIC was lost. The case shows that reactor water level was maintained above TAF. In this analysis, the vent was cycled by opening the vent at 45 psig and closing it at 25 psig. As can be seen in Figure 2, the vent is only cycled once within the first 24 hours and only 4 times within a 48 hour period.

  Should the plant's strategy for EA-12-049 involve vent cycling, their plant-specific analysis could be used to identify the number of cycles estimated during the first 24 hours.

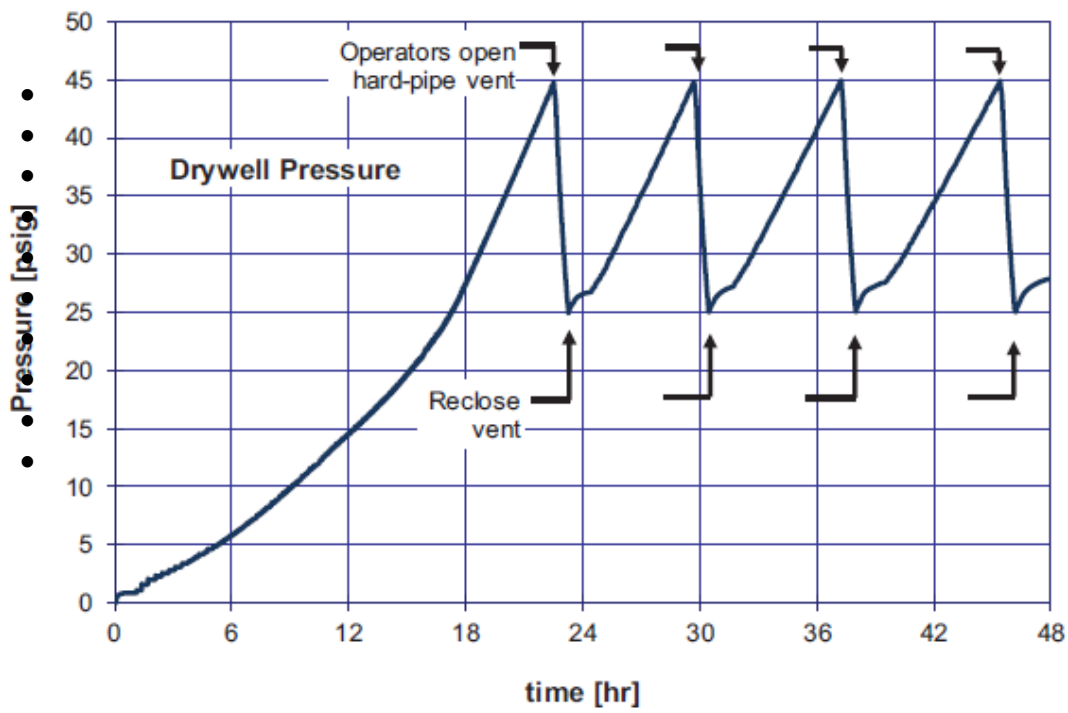


Figure 2 – SOARCA Vent Cycling Results

- Case 2 was analyzed in SECY-12-0157 and the sequences identified included several with vent cycling. However, information on the number of cycles was not included in the SECY documentation. The EPRI report, "Investigation of Strategies for Mitigating Radiological Releases in Severe Accidents", Report No. 1026539, included MAAP analysis for strategies that involved cycling of the containment vent. The scenarios investigated included two severe accident scenarios with cycling of the vent between 60-40 psig. The results are summarized here:
  1. Drywell flooding and vent cycling – 12 cycles within 24 hours
  2. Drywell sprays and vent cycling – 8 cycles within 24 hours

  Vent cycling was initially identified in the EPRI Technical Report 1026539 as a possible strategy to improve the deposition of radionuclides within the containment and reduce the amount of fission product release from the vent path. There is currently an analysis being prepared by EPRI to support the technical basis for rulemaking on filtering strategies. In addition, the NRC is performing similar calculations to investigate possible mitigation strategies which include vent cycling. Where the results of the rulemaking analysis are not available in time to inform phase 1 or 2 of Order EA-13-109, information from EPRI Technical Report 1026539 can be used to estimate the approximate number of vent cycles required.

- Case 3 is similar to Case 2 in that vent cycling could be implemented during a severe accident to reduce the overall fission product release from the vent path. In fact, the scenarios discussed above from EPRI Technical Report 1026539 were for cases without injection early, consistent with Case 3 from Figure 1.

Based on the above, the following summarizes the available results on vent cycling within the initial 24 hour period:

1. Anticipatory venting in a non-core damage scenario – 1 cycle
2. SOARCA Long Term SBO in a non-core damage scenario – 1 cycle
3. EPRI Technical Report 1026539 in a severe accident – 12 cycles

Vent cycling is not a requirement of EA-13-109 and the actual benefits have not yet been fully established as part of the related rulemaking.

In a simplistic view, the number of cycles could be established by the following:

- Anticipatory venting prior to core damage – 1 cycle
- Wetwell venting during a severe accident as defined by severe accident guidelines – 1 cycle
- Drywell venting during a severe accident as a result of isolation of Wetwell vent on high suppression pool level during a severe accident as defined by severe accident guidelines – 1 cycle

Based on the above, a design minimum number of wetwell vent cycles would be 3 in the first 24 hour period unless the plant's strategy for EA-12-049 involves vent cycling; then their plant-specific analysis would be used to identify the number of cycles estimated during the first 24 hours.

3. Radiation Exposure due to Venting in a Severe Accident

Figure 1 includes an identification of when it is expected that elevated radiation environments could exist in the reactor building and at the site boundary. In a severe accident, it is expected that high containment radiation doses would be seen following the onset of core damage. Operator actions for portable equipment can be evaluated based upon MAAP and MELCOR-informed analysis driven primarily from containment shine and the location of the HCVS piping

For Case 1, there is no core damage so this section of the white paper does not apply. In Case 2, since the vent may have already been opened prior to core damage, elevated shine dose coming from the vent piping would be expected at 23 hours. In addition, local doses at the discharge of the vent path would be expected. For Case 3, there may be a period of time following the onset of core damage when the vent path had not yet been activated. During this time period, fission products would be primarily transported into the wetwell via the open safety relief valves. This could create a shine dose and the potential for leakage in the area external to the wetwell within the reactor building. Access to those areas at that time could be limited due to elevated exposure. As with Case 2, once the vent path is open, shine from the vent pipe and local doses at the vent discharge would be expected.

If a utility would want to consider vent cycling to reduce the radiological release, as demonstrated in EPRI Technical Report 1026539, then an estimate of 12 cycles over a 24 hour period could be selected as described in the report.

Refer to Appendix A for a description of a method that can be used to determine dose rates/source term from the HCVS and estimate the dose in the vicinity of the vent piping to establish the elements and magnitude released at critical time periods of a severe accident. The reason that the parameters of the base case were selected was that the resultant values can be shown to be both reasonable and bounding for a wide range of scenarios.

To apply the results of the base case to another plant's HCVS, three primary variables (scaling factors) are utilized to provide a scaled value for dose. These variables are:

- Core rated thermal power
- HCVS pipe inner diameter
- Drywell volume

Optional Site Analysis utilizing NUREG-1465 and site characteristics can be performed if use of the generic base case with scaling factors is not used for a particular unit. The Content of Appendix A can provide guidance for site specific analysis content.

CONCLUSIONS:

- Timing of Initial Vent Opening
  - 5 hours would represent the shortest time in which the HCVS would need to be operational. Each plant should modify this time based on plant specific analysis.
- HCVS Cycling Evaluation
  - Based on the above, a conservative generic design minimum number of wetwell vent cycles (from EPRI report referenced on Page 4 of this white paper) would be as shown below, in the first 24 hour period unless the plant's strategy for EA-12-049 involves vent cycling; then their plant-specific analysis would be used to identify the number of cycles estimated during the first 24 hours.
  - A generic number of 8 Wetwell vent cycles within the first 24 hours is reasonable
  - A generic number of 12 Drywell and Wetwell vent cycles within the first 24 is reasonable
  - The number of cycles is very dependent on the strategy and scenario selected for the evaluation and Multiple Vent cycling is not a requirement in response to EA-13-109 and the actual benefits have not yet been fully established as part of the related rulemaking.
- Generic Radiological Analysis
  - A Scaled approach is acceptable that uses:
    - Vent line dose curves (generated based on NUREG-1465 release fractions and timing) for "base case"
    - Scaling factors applied to "base case" for applicability to a particular unit

APPENDIX A – Method for Determining Radiological Dose From HCVS Piping

1.0 Purpose

The purpose of this appendix is to present a simplified method for determining radiological impacts resulting from use of the Hardened Containment Venting System (HCVS) during a severe accident:

- Determine total integrated dose to use for environmental qualification for that equipment required for operation of the HCVS
- Identify potential radiological conditions that could impact or impede operator action associated with event response

Determination of radiological consequences addresses the following elements of NRC Order EA-13-109:

- A.1.2.10 - Design for severe accident & dynamic conditions
- A.1.1.3 – Account for radiological conditions that would impede event response
- A.1.1.4 – Accessible controls and indication

2.0 Approach

The general approach for creation of the method and associated dose rates followed this sequence:

1. Development of base case for reference plant and determination of dose rates 3' from the Wetwell (WW) vent pipe (both related to gases and aerosols).
2. Development of sensitivity factors and validation that base case was both reasonable and bounding for a wide range of scenarios/cases
3. Development of scaling factors to allow the results of the base case to be applied to other plants via these simple factors in lieu of repeating detailed plant-specific analysis

This approach was utilized to account for the wide variety of scenarios/conditions that could develop during a severe accident scenario, to prevent "stylized" evaluations of radiological consequences, and to allow plants to greatly simplify their assessments.

3.0 Base Case

The following base case was developed using airborne concentration levels from the Drywell (DW) to create a bounding value for WW venting that accounts for the sensitivity factors described in Section 4. It is assumed that plants will preferentially utilize the WW vent.

The time dependent airborne radioactivity in containment is based on the severe accident methodology presented in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 contains a bounding severe accident scenario consisting of a time history of releases from the reactor core to the containment. It addresses different phases of the accident, including the initial gap release phase, the in-vessel releases caused by fuel melt and ex-vessel releases following breach of the pressure vessel.

The effect of operation of the HCVS is demonstrated by estimating the maximum dose rate from the vent line as a function of time following reactor shutdown. The maximum dose rate is calculated by assuming the activity concentration in the vent line is the same as the activity concentration in the containment.

The following key assumptions are applied:

- Extended Loss of AC Power (ELAP) at reactor shutdown
- The release fractions and timing are based on Table 3.12 of NUREG-1465 (Table 1, below) with the gap release phase beginning when ELAP occurs. However, because of the relatively short decay times of noble gases and earliest anticipated time that HCVS will be operated following the ELAP, dose rates associated with noble gases released during the gap release phase are not included in the dose/time history curve. Separate consideration for dose associated with gap release is made for evaluation of operator actions prior to venting.
- All particulate activity is directed into the Drywell.
- The radioactive inventory is the design basis LOCA core inventory with credit for radiodecay and daughter product generation following reactor shutdown.
- No depletion of the airborne activity in the containment due to leakage or venting from containment. This maximizes the activity in containment and therefore maximizes the dose rate on the vent line.
- The earliest time that the HCVS will be operated is two hours following ELAP. (**NOTE:** This assumption is included for the base case to ensure that the peak dose rate is captured for the model. Most plants would not vent for ***at least*** 6 hours following ELAP)
- No fission product removal mechanisms credited other than radiodecay.
- Activity concentration in the vent line is equal to that in the containment (Drywell)

Table 3.12 BWR Releases Into Containment*

	Gap Release***	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.5	1.5	3.0	10.0
Noble Gases**	0.05	0.95	0	0
Halogens	0.05	0.25	0.30	0.01
Alkali Metals	0.05	0.20	0.35	0.01
Tellurium group	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble Metals	0	0.0025	0.0025	0
Cerium group	0	0.0005	0.005	0
Lanthanides	0	0.0002	0.005	0

* Values shown are fractions of core inventory.
** See Table 3.8 for a listing of the elements in each group
*** Gap release is 3 percent if long-term fuel cooling is maintained.

Table 1 – Source Terms from NUREG-1465

Key inputs:

- Drywell volume is 306200 ft^3
- Reactor power level is 4067MWt

- HCVS vent pipe inner diameter of 19.25” (nominal 20” pipe diameter)

Source terms are calculated using a RADTRAD model of the reference plant. Resulting activity concentrations are then applied uniformly through the vent pipe. A MicroShield model is then utilized to generate maximum dose rates from the vent pipe. Resultant dose rates are shown in Table 2 and graphically represented in Figure 1 below.

Unshielded Dose Rates (Rem/hr)

Distance	2 Hr	4 Hr	8 Hr	10 hr	12 Hr	18 Hr	24 Hr	48 Hr	72 Hr	168 Hr
1 ft	1.184E+04	1.664E+04	1.593E+04	1.464E+04	1.366E+04	1.216E+04	1.099E+04	8.690E+03	7.615E+03	5.350E+03
3 ft	5.199E+03	7.313E+03	7.013E+03	6.447E+03	6.018E+03	5.363E+03	4.848E+03	3.832E+03	3.355E+03	2.354E+03
10 ft	1.212E+03	1.708E+03	1.642E+03	1.510E+03	1.411E+03	1.258E+03	1.138E+03	8.988E+02	7.864E+02	5.508E+02
20 ft	3.810E+02	5.370E+02	5.162E+02	4.750E+02	4.437E+02	3.959E+02	3.580E+02	2.827E+02	2.473E+02	1.732E+02

Table 2 - Base case dose rates

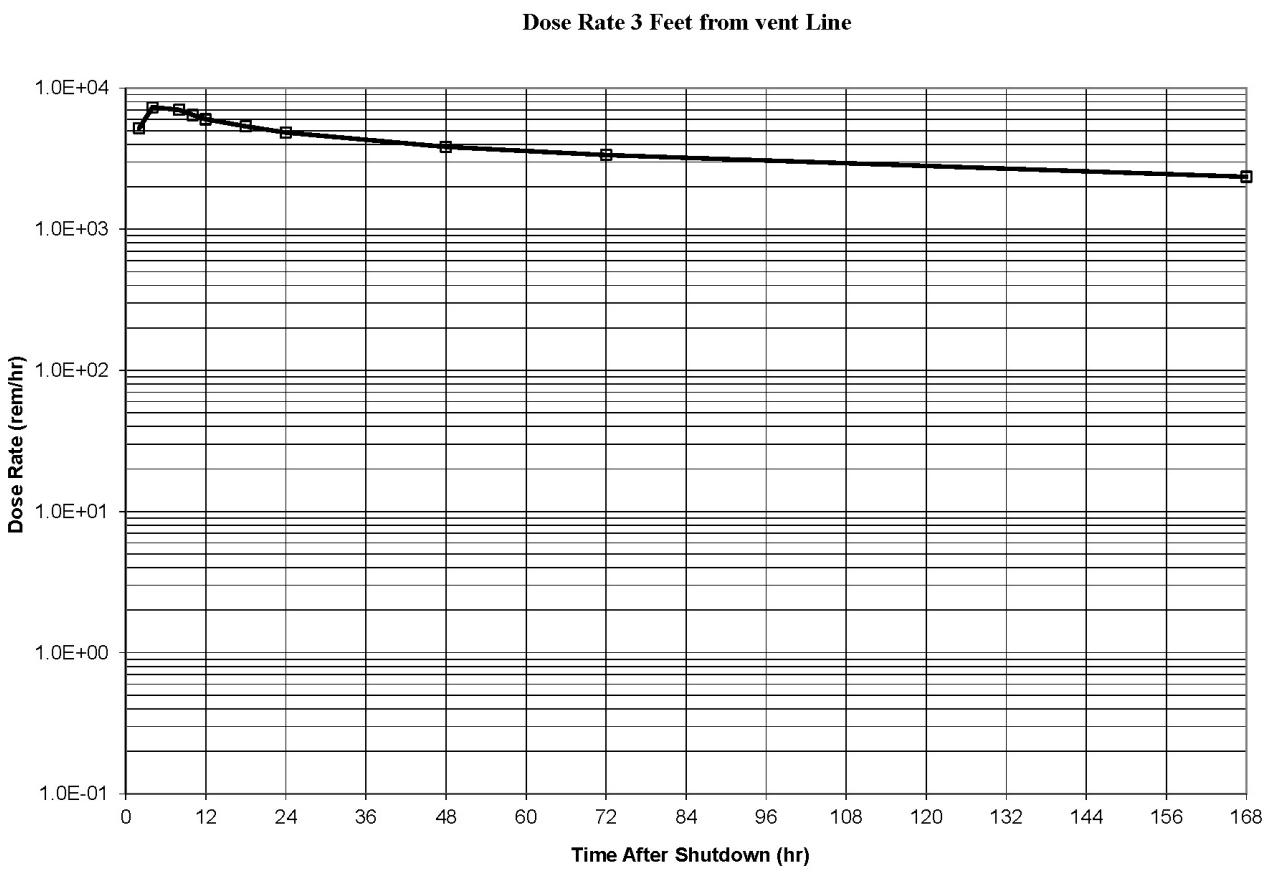


Figure 1 - Base case dose rates

4.0 Sensitivity factors

The reason that the parameters of the base case were selected was that the resultant values can be shown to be both reasonable and bounding for a wide range of scenarios. A number of mechanisms can impact the concentration of radionuclides being vented and associated dose rates from the vent pipe. Primary factors, their qualitative impact on dose rates, and a brief description of the basis for impact is provided below. The net result, as described and as shown in Figures 2 through 9, is that the base case estimate represents an upper bound of the radiation dose in the vicinity of the pipe.

Mechanism	Effect on dose rate	Description
Radionuclide Decay	↓	**Noble gases:** From the time of shutdown, there is rapid reduction in dose rates associated with noble gas. Within 12 hours of reactor shutdown, dose rates are dominated by radioactive aerosols. Refer to Figure 2 **Radioactive aerosols:** Due to the relatively long half-lives of the various aerosols, radiodecay has only a moderate effect (factor of 3 reduction over the 7 day window). The decay rate lowers over time as short-lived nuclides decay away, leaving long-live nuclides such as Cs remaining. Radiodecay of aerosols is included in the base case.
Natural Deposition	↓	There is a significant reduction in airborne concentration due to natural deposition. Refer to Figure 3 for the effects on DW airborne concentration/dose. Natural deposition rates were derived using the Powers model for natural deposition as described in NUREG/CR-6604. Insights: Event scenarios where there is early failure of core injection and early core damage typically see the biggest benefit from natural deposition due to the longer period between the onset of core damage and venting.
Suppression Pool Scrubbing	↓↓↓(↓)	Figure 4 represents a simplified view of the effects of a pool decontamination factor (DF) of 80. The DF was determined using the correlation from NUREG/CR-6153 and assuming the smaller of the two possible submergence depths (downcomer in lieu of T-quenchers), which results in a conservative value for DF. Even conservatively stated, the impact of pool scrubbing is very large. For events without pool bypass, the effect includes the arrow in parentheses. Mark II containments with the potential for suppression pool bypass retain the bulk of the pool scrubbing benefits

		(first three arrows). This was determined by first examining the amount of radionuclides expected to be discharged through the SRVs into the pool prior to vessel breach. Figure 5 is from NUREG/CR-7110 (Figure 5-44) and identifies the fraction of core inventory that would enter the DW atmosphere after vessel breach for a station blackout with RCIC failure. After establishing the quantity of radionuclides likely to be in the DW atmosphere, the next step is to determine what percentage is then transferred into the WW. Multiple MAAP cases were performed to evaluate the forcing function from DW to WW. Figure 6 shows the relative airborne concentrations when RCIC fails at 4 hours and vessel breach occurs at 16 hours. Figure 7 shows the same case but forcing MAAP to retain more core debris on the floor and in-vessel to create more CCI/ablation/heat in the DW to create more pressure. With the dominant energy source in the wetwell (i.e. debris in the suppression pool), wetwell pressure remains elevated above the drywell limiting flow from DW-WW. In both cases, the DW:WW aerosol mass ratio remains almost 100:1. This demonstrates the significant conservatism in using the drywell aerosol concentrations for vent line dose assessment (i.e. actual WW concentrations significantly lower).
Deposition in vent pipe	↑↑↑	There is not a lot of conclusive experimental data for deposition rates of aerosol sizes for radionuclides at flow rates anticipated for venting evolutions (500-1000+ ft/sec). However, sensitivity cases were evaluated to consider potential impact. Figure 8 represents a late RCIC failure case (most conservative) where venting is initiated at the same time as core damage, scrubbing DF was minimized, and deposition rates of 1% and 2% were used and restricted to the first 100’ of pipe to drive up dose rates.

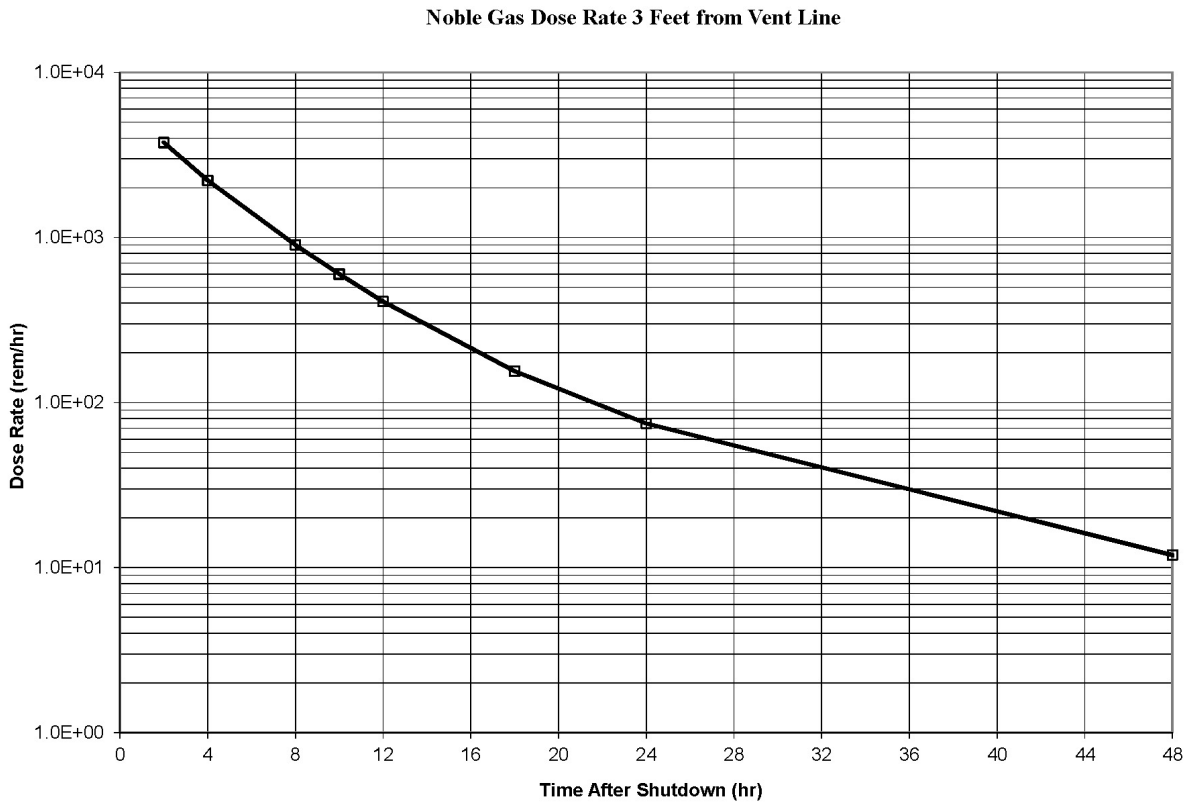


Figure 2 – Noble Gas Decay

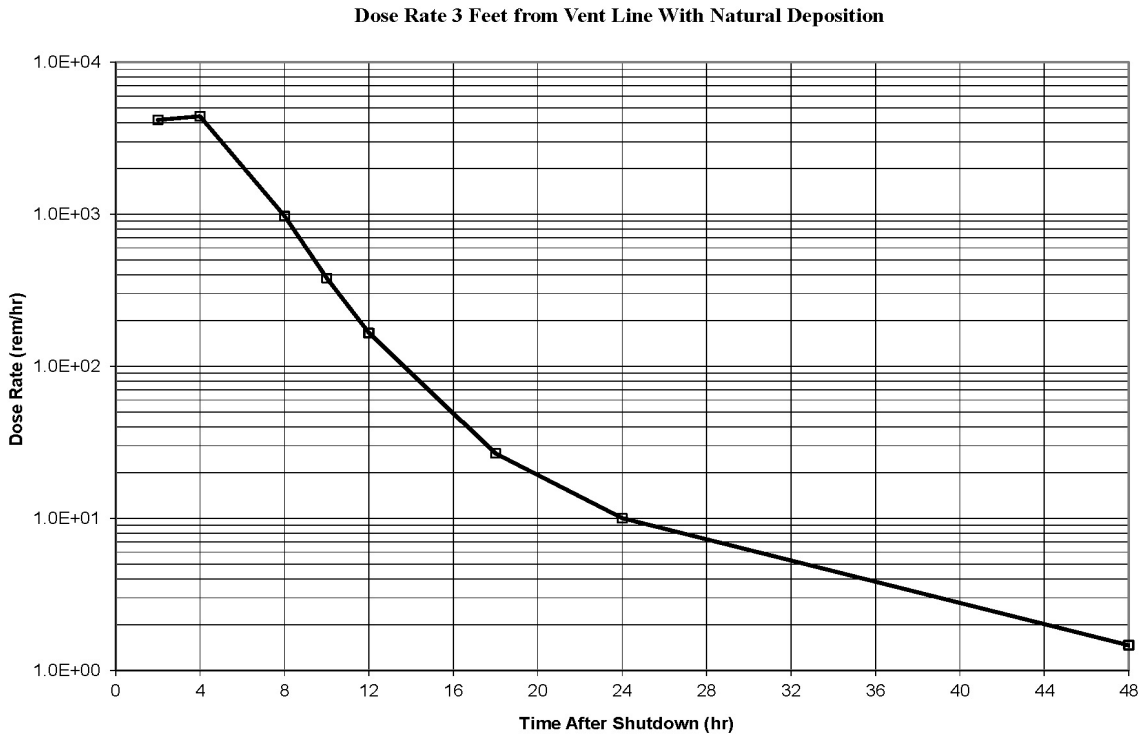


Figure 3 – Effects of Natural Deposition of Radioactive Aerosols

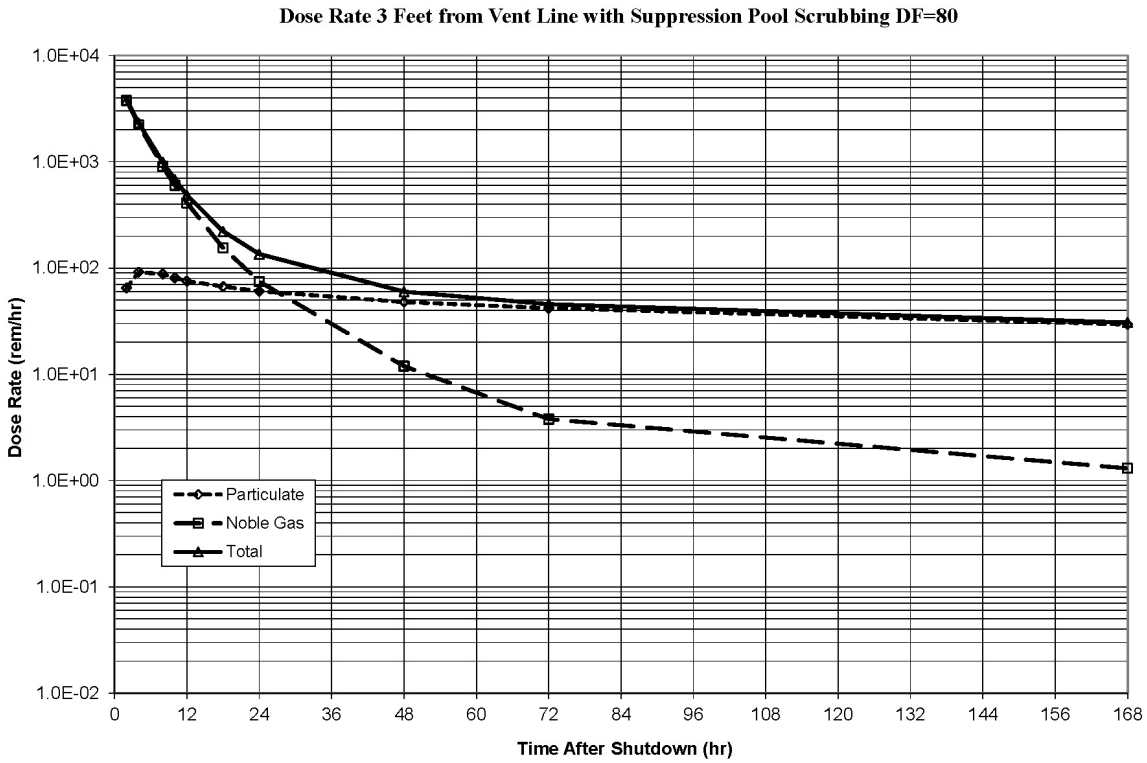


Figure 4 – Effects of Suppression Pool Scrubbing

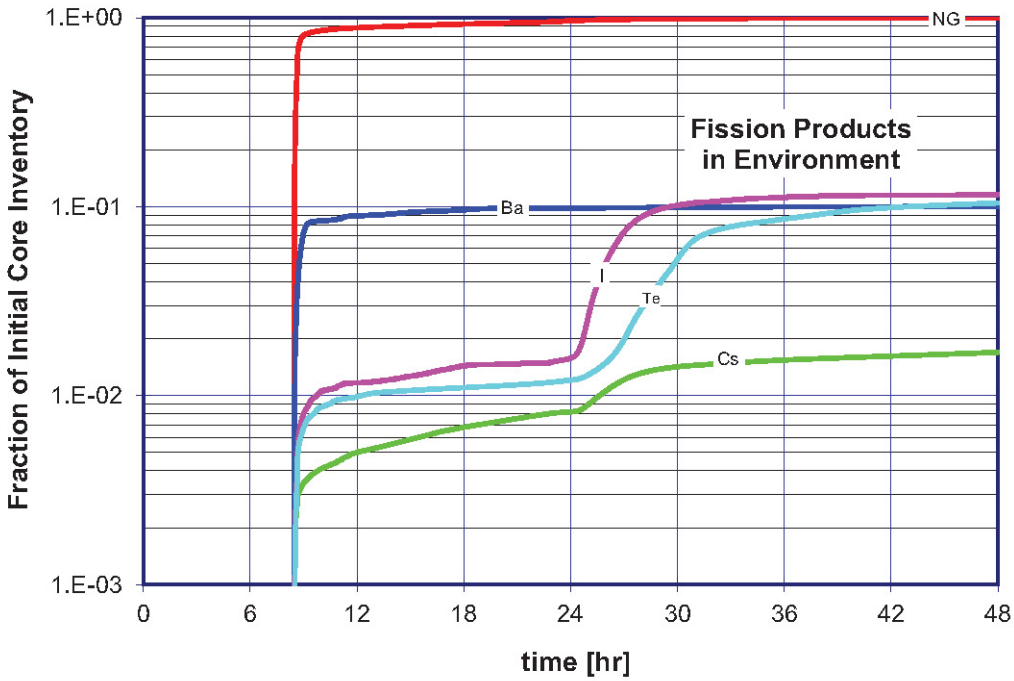


Figure 5 – Fission products in DW atmosphere after vessel breach

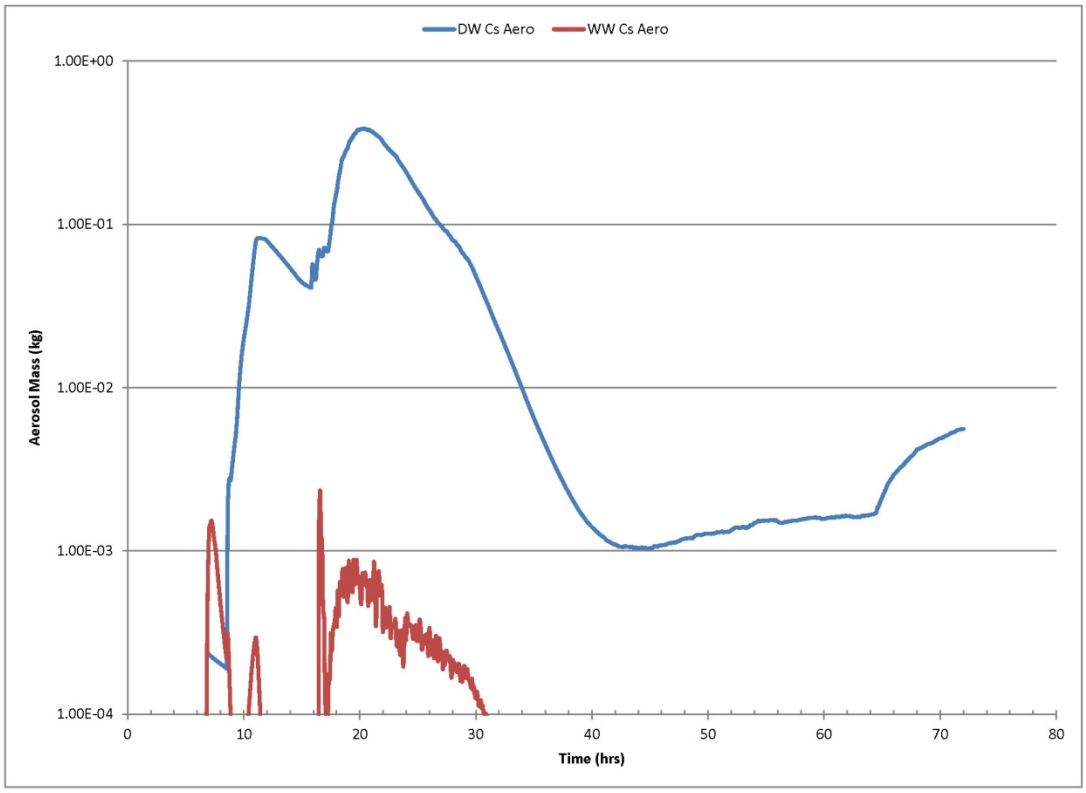


Figure 6 – Drywell vs WW airborne concentration for vessel breach

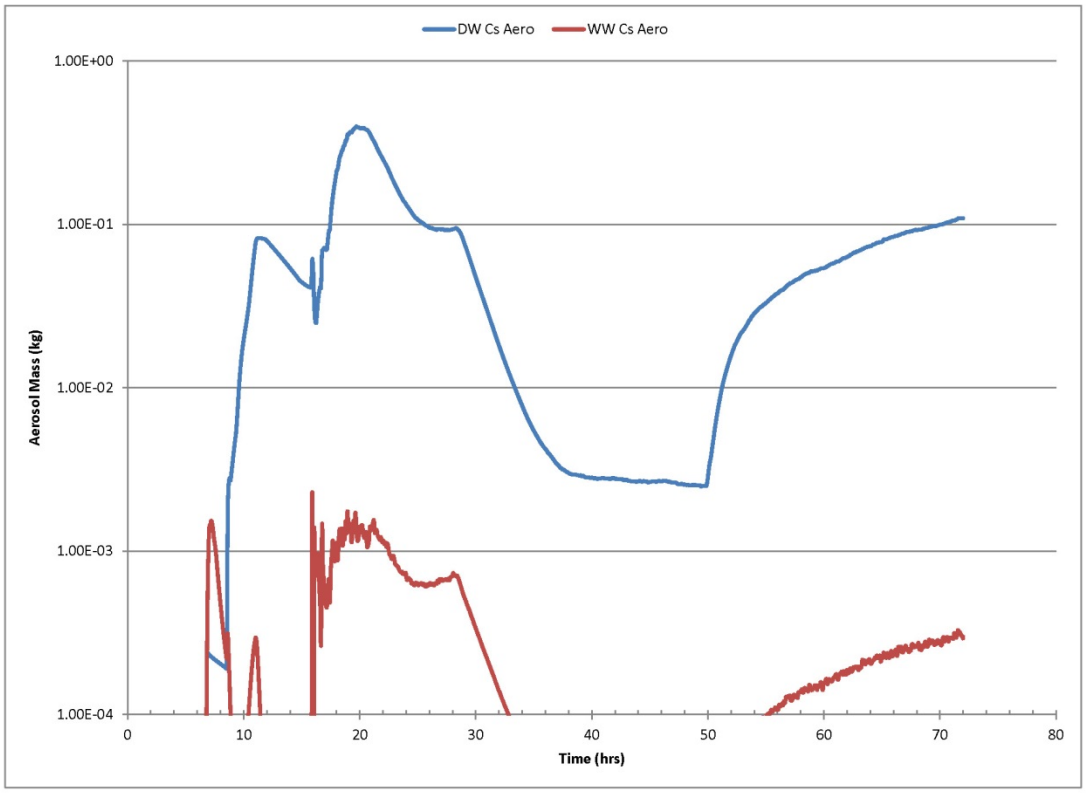


Figure 7 – Drywell vs WW airborne concentration at vessel breach – forced into WW

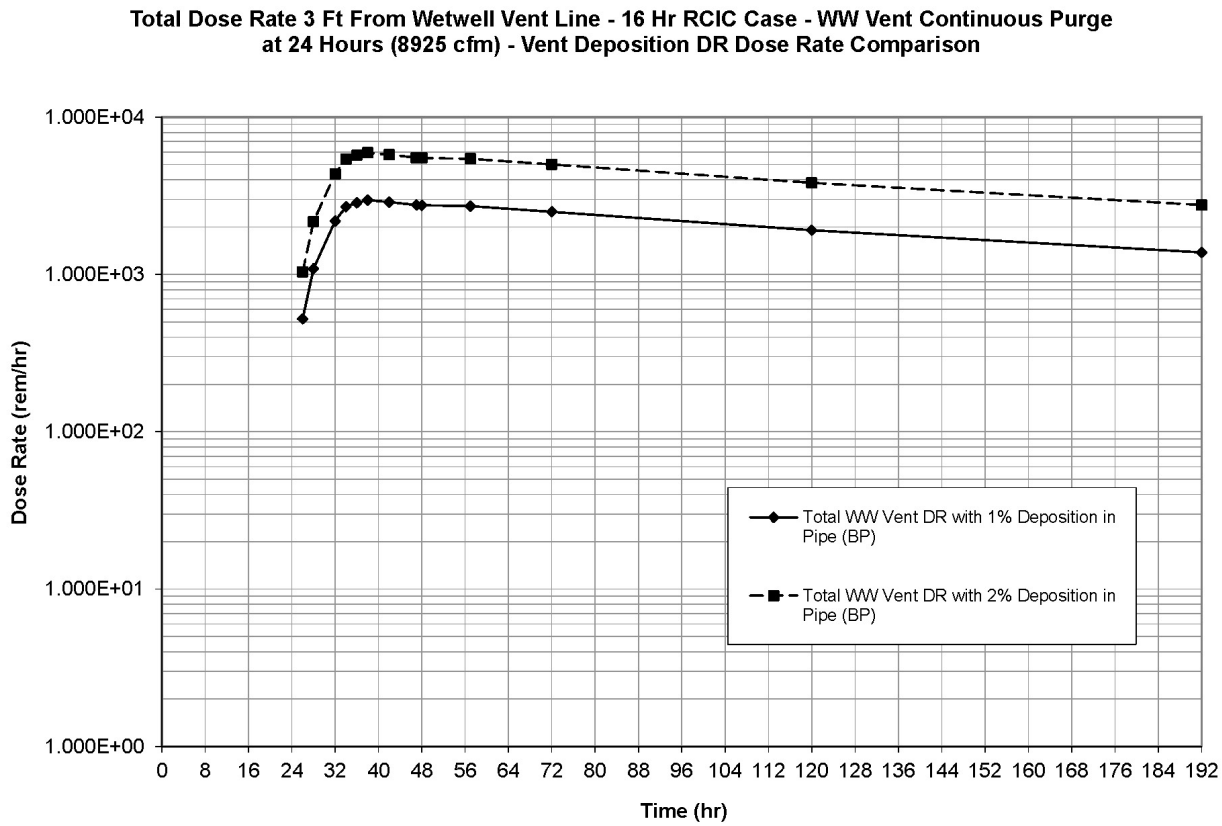


Figure 8 – Sensitivity effects of pipe deposition

Notes on Figure 8:

- A conservative DF of 80 was applied for suppression pool scrubbing. As shown in Figure 9, pool scrubbing can reasonably be shown to be MUCH higher than that (500+), particularly when discharging into the pool via SRV discharge and associated T-quenchers.
- No credit was assumed for natural deposition inside either the Drywell or Suppression Chamber prior to initiation of venting.
- 100% of the core inventory of Noble Gas, Halogens, Alkali Metals, and Tellurium Group was assumed to be released either to the Suppression Chamber or Drywell.
- Benefits related to Severe Accident Water Addition (SAWA) effects on in-vessel retention are not credited in this curve

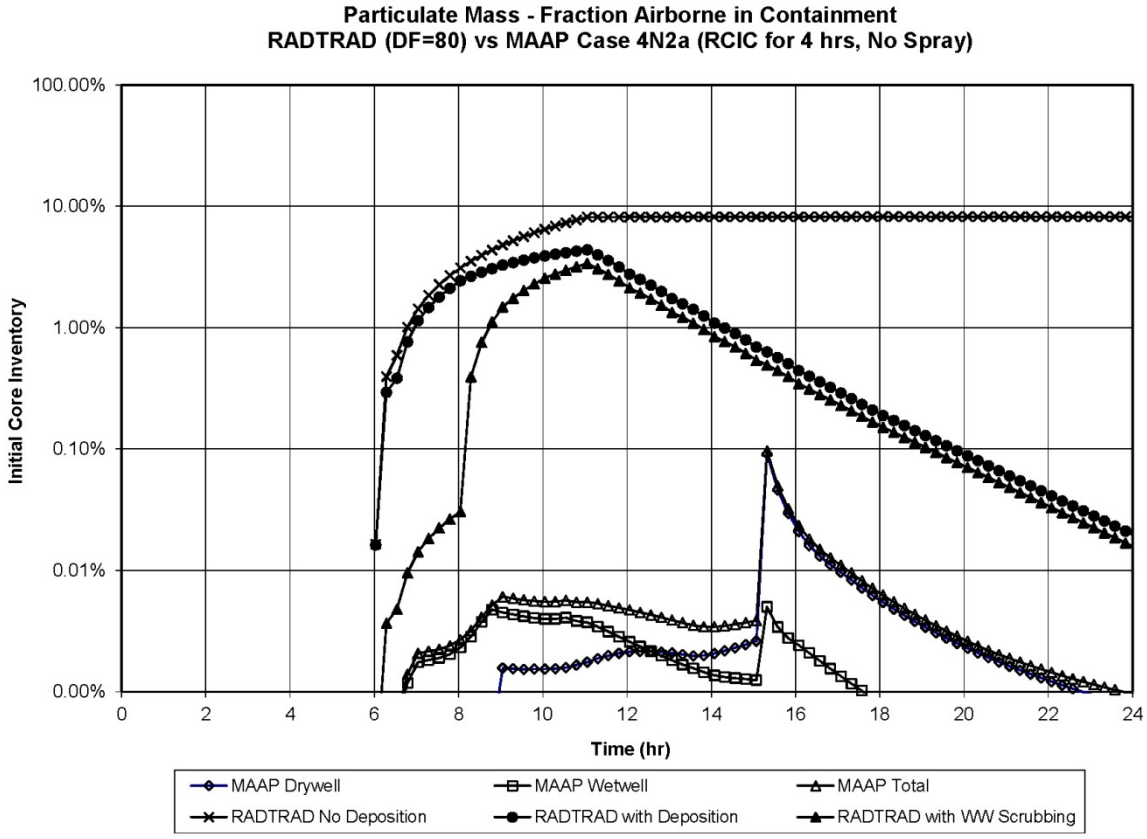


Figure 9 – MAAP vs RADTRAD/1465 model
This figure demonstrates the relative conservatism of the models for airborne radionuclide concentrations for the base model vs MAAP.

As described above, the conclusion from the assessment of sensitivity factors is that the base case estimate represents an upper bound for radiation dose.

5.0 Scaling

To apply the results of the base case to another plant's HCVS, three primary variables (scaling factors) are utilized to provide a scaled value for dose. These variables are:

- Core rated thermal power
- HCVS pipe inner diameter
- Drywell volume

Establish ratio of thermal power between reference plant and plant being evaluated (Scaling Factor 1). SF_1=X/4067 MWth, where X is the rated thermal power for plant being evaluated.
A larger thermal power results in a higher total quantity of radionuclides

Establish ratio of drywell volumes between reference plant and plant being evaluated (Scaling Factor 2). SF_2= 306200 ft^3/Y ft^3, where Y=the drywell free air volume of the plant being evaluated.

NOTE: Drywell free air volume includes the volume of the wetwell-drywell vent pipes, but NOT the wetwell volume free air space.
A larger volume results in a lower concentration of radionuclides in the Drywell

Establish ratio of nominal cross sectional area of HCVS pipe (Scaling Factor 3).
$SF_3=(Z/2)^2/(19.25/2)^2$, where Z is the nominal inner diameter of the plant being evaluated.

Note: To account for minor scaling errors using this simplified factor equation, an additional adjustment factor should be applied when using pipe that is 16" or smaller nominal diameter. The adjustment factors are as follows:

12-16" pipe (nominal pipe size): Multiply by 1.1. Therefore $SF_3=1.1*((Z/2)^2/(19.25/2)^2)$
8-10" pipe: Multiply by 1.2. Therefore $SF_3=1.2*((Z/2)^2/(19.25/2)^2)$

A larger cross-sectional area of pipe will result in a higher quantity of radionuclides per section of pipe based on mixture being vented from containment.

Using the upper bound peak dose at 4 hours and 3 feet from the source as shown in Table 2, the plant-specific peak dose rate from vent pipe = 7313 Rem/hr* SF_1* SF_2* SF_3