**VERMONT YANKEE**
**NUCLEAR POWER CORPORATION**

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December 28, 1999
BVY 99-162

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

References:

(a) Letter, VYNPC to USNRC, "Submittal of the Vermont Yankee Individual Plant Examination for External Events (IPEEE) Report – Response to Generic Letter 88-20, Supplement 4," BVY 98-91, dated June 30, 1998.

(b) Letter, USNRC to VYNPC, "Request for Additional Information on Vermont Yankee Nuclear Power Station Individual Plant Examination Of External Events (IPEEE) Submittal (TAC No. M83689)," NVY 99-23, dated February 26, 1999.

(c) Letter, VYNPC to USNRC, "Response to Request For Additional Information Concerning VY-IPEEE," BVY 99-56, dated April 16, 1999.

**Subject:** **Vermont Yankee Nuclear Power Station**
**License No. DPR-28 (Docket No. 50-271)**
**Response to Request For Additional Information Concerning VY-IPEEE**

In Reference (b), Vermont Yankee (VY) was requested to respond to questions concerning our IPEEE submittal (Reference a) or provide a schedule for responding within 60 days. In Reference (c), VY proposed to provide the requested information by December 31, 1999.

The Attachment A to this letter provides the requested information.

It should be noted that most of the identified improvement opportunities that were under evaluation or not completed when the original IPEEE was submitted in June 1998, are now complete. Three items that are not yet complete, are listed on the attached commitment summary form. A brief description and schedule for these items are listed with a more detailed discussion in the RAI response (Attachment A).

A011

If you have any questions concerning this transmittal, please contact Mr. Jeffrey T. Meyer at (802) 258-4105.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Don M. Leach
Vice President, Engineering

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector – VYNPS
USNRC Project Manager – VYNPS
Vermont Department of Public Service

# SUMMARY OF VERMONT YANKEE COMMITMENTS

BVY NO.: BVY 99-162

The following table identifies commitments made in this document by Vermont Yankee. Any other actions discussed in the submittal represent intended or planned actions by Vermont Yankee. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager of any questions regarding this document or any associated commitments.

| COMMITMENT | COMMITTED DATE OR "OUTAGE" |
|---|---|
| Seismic RAI 2 (Item 3) – Modify diesel fire pump fuel oil supply line tubing to alleviate the potential for crimping. | RFO-22 |
| Seismic RAI 2 (Item 5) – Enhance the support of the fire system northwest standpipe in the Reactor Building to improve ruggedness. | RFO-22 |
| Flooding RAI 2 (Item 7) – Enhance procedures and training to improve the mitigation response/strategy of potential flooding event. | 9/1/2000 |
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# Attachment A

Vermont Yankee Nuclear Power Station

Individual Plant Examination External Events (IPEEE)

Response to NRC Request for Additional Information

(December 1999)

**FIRE RAI 1:**

From the submittal the bases for crediting fire detection and suppression systems are not clear. In particular, it appears that carbon dioxide (cardox) systems have been credited in several of the risk dominant areas. However, cardox systems are especially vulnerable to inconsistent performance if not installed and maintained in accordance with accepted national codes and standards. Generic system reliability values were apparently applied to these systems. These values assume fully code compliant systems.

Please confirm whether or not the fire detection and suppression systems including the cardox suppression systems at Vermont Yankee have been installed and maintained in accordance with accepted national standards. If any of the installed systems do not meet these standards, assess the impact on core damage frequency (CDF) if the non-compliant system is not credited.

**FIRE RAI 1 RESPONSE:**

The fire detection and suppression systems, including the cardox systems, credited in the VY Fire IPEEE have been designed, installed, and maintained consistent with the requirements of industry standards, as applicable to nuclear facilities. Surveillances have been performed on these systems according to the VY Technical Requirements Manual, insurance standards, and general industry guidelines. The details of the fire detection and suppression system have been reviewed and approved by the NRC as evidenced in the Fire Protection SER dated January 13, 1978 and later SER revisions.

The Vermont Yankee fire detection and suppression systems have undergone a complete NFPA code compliance review (including walkdowns) in 1997 and 1998. Hardware modifications or evaluations have been performed to address non-compliant issues. There are currently no open items that effect the operability of fire protection systems credited in the VY Fire IPEEE.

Given the above, it is appropriate to apply the fire suppression generic failure probabilities provided in the FIVE methodology.

**FIRE RAI 2:**

In the assessment of compartment-specific fire frequencies, both fire severity and fire data correction factors were used to modify the fire ignition frequencies for many postulated fire sources. Further, the assessment has also applied fire frequency weighting factors for some fire sources that are not consistent with the Electric Power Research Institute (EPRI) fire-induced vulnerability evaluation (FIVE) methodology.

The first point of concern is that the "correction factors" that have been independently applied may have inappropriately reduced compartment fire frequencies. In particular, the licensee has applied a correction factor based on the fraction of fires in the database that occurred while a plant was "at-power". The fire frequencies cited in the FIVE study include consideration of this fact, and to apply an additional correction is contrary to the FIVE methodology. Typically, fire events are considered relevant if they could have occurred at power even if the event actually occurred during some other mode of operation. This would include, for example, events occurring during pre-operational testing, during start-up operations, and during certain shutdown events where the shutdown mode of operation is not identified as a specific contributing factor in the event. Events that are typically excluded would include events such as fires caused by welding or cutting in areas where such activities are not permitted during power operation, or events attributed to construction activities that are no longer present. Again, these factors have already been incorporated into the generic fire frequencies cited in procedural documents such as the EPRI FIVE methodology.

The second point of concern relates to the licensee's application of the FIVE methodology with respect to ignition source weighting factors. This impacts the switchgear rooms, battery rooms and intake/discharge structures. The licensee applied the analysis of "Method B", which is dividing the number of ignition sources in the fire compartment by the number in the selected location, rather than the FIVE approach of "Method A", which does not use an ignition source weighting factor. This may have artificially reduced fire frequencies. In particular, it is important that the overall plant-wide frequency of fire events be preserved, and it is not clear that the licensee IPEEE analysis has done so. Since the licensee used the FIVE methodology as its basis for its IPEEE fire analysis, the licensee has not provided adequate justification for its deviations from the FIVE's prescription of methods of applying weighting factors.

The third point of concern relates to the application of fire severity factors. The licensee cites the EPRI Fire Probabilistic Risk Assessment (PRA) Implementation Guide as the source of these values. The severity factors were used to adjust the basic ignition frequencies of the associated components for those areas surviving screening. These adjusted ignition frequencies were apparently used in scenarios where fire suppression was credited. Since the success of fire suppression would reduce the potential for a large fire, there appears to be a significant possibility that the use of a severity factor, when fire suppression is modeled, double counts for suppression efforts.

Considering the points above, please provide the following:

(a) For the compartments that were quantitatively screened, please reassess the screening analysis using defensible fire ignition frequencies. For each compartment, identify the fire sources, the baseline fire frequencies, and any correction of severity factors used to develop the final compartment frequency. Provide justification for any modifying factors used including consideration of the concerns discussed above. Discuss any changes to the original screening results.

(b) For the compartments which survive the revised screening analysis from (a) above, please analyze or re-analyze the detailed fire scenarios. For the source/target sets analyzed in each compartment, discuss the partitioning used to develop the scenario fire event frequency. As part of the re-analysis, identify any cases where both a severity factor and independent credit for fire detection and suppression were applied in the original analysis. For those cases, either eliminate the severity factor or eliminate the detection suppression credit. Based on the above approach. Please recalculate the CDF contribution for each compartment and provide the results.

(c) Given the above results, identify those compartments/scenarios that dominate the fire CDF. Re-assess the implications of the analysis with regard to the identification of plant vulnerabilities, modifications, and improvements.

**FIRE RAI 2 RESPONSE:**

Ignition Source Weighting Factors:

Calculation of the fire initiating event frequency for the switchgear rooms, battery rooms, and intake/discharge structures uses FIVE [Reference 3] Method A of the Ignition Source Weighting Factor ($WF_{LS}$) for the "location-specific" ignition sources, i.e., electrical cabinets, batteries, and pumps. The VY IPEEE Submittal inadvertently referred to use of Ignition Source Weighting Factor Method B for these "location-specific" ignition sources.

Revised Fire Severity Factors:

The VY IPEEE used the EPRI database in NSAC-178L [Reference 1] as the starting point for developing area-specific fire initiating event frequencies. The scope of the IPEEE is limited to initiating events that occur at-power, yet many of the fires in the EPRI database (and included in the EPRI fire frequency calculations) occurred while plants were in a shutdown mode (during refueling). Because of the significant differences in plant configuration during at-power versus refueling conditions, and the significant increase in the scope of maintenance and construction activities during outages compared to when a plant is at-power, the VY IPEEE internal fires baseline assessment (presented in the VY IPEEE Submittal) did not include the shutdown fire event data in the compartment fire initiating event frequency calculations.

Table Fire-2-1 identifies the revised IPEEE fire severity factors with increased consideration given to the shutdown fire events. Note that some of the fire incidents occurring during shutdown were minor fires that did not have the potential to affect multiple plant components, e.g., fires that were "self-extinguishing". Consistent with the VY calculation of fire severity factors and with NSAC/178L, minor fires that occurred during shutdown and at power were not included in the revised severity factor calculation. This was done to ensure that the resulting initiating event frequencies remain representative of fires of the severity level, i.e., fully developed, modeled by the FIVE fire hazard method.

Revised Screening Assessment:

New "sensitivity" fire initiating event frequencies were calculated based on the revised fire severity factors of Table Fire-2-1 performed to consider the shutdown fire events data. As with much of the FIVE methodology it is important to be aware that there are substantial conservatisms in the data. Therefore, while the sensitivity study performed to include shutdown events can produce relatively large increases in initiating event frequency and CDF, there remain substantial conservatisms in the data. For example, the EPRI frequencies include fire events that occurred during pre-operational testing or in the first 1 to 2 years of plant operation i.e., "infancy" period data. Such data is not generally appropriate to mature plant operation (such as VY). This "infancy" period data are significant contributors to many of the EPRI calculated initiating event frequencies.

Based on the revised fire initiating event frequencies, we re-reviewed the FIVE Phase 2 initial screening of critical fire areas. Table Fire-2-2 compares the original critical fire area screen performed in the VY IPEEE Submittal with the revised screening to support this RAI. Based on this screening review, no new areas were identified as potentially needing detailed fire modeling except for the Turbine Building. However, the Turbine Building area is judged not to require detailed fire modeling based on: (a) the initial screening value of 1.1E-06/yr is very close to the 1E-06/yr screening criteria, (b) the Turbine Building area conservatively lumps together a number of internal areas and connecting buildings that should otherwise be analyzed separately, and (c) the EPRI initiating event frequencies are conservative in that they include pre-operational events and "infancy" events (in the case of Turbine Building-Other Pumps, the percentage contribution to the ignition frequency from such inappropriate events is 50%).

Detailed Fire Scenario Re-Assessment:

Table Fire-2-3 summarizes the detailed fire scenario re-assessment based on the sensitivity fire initiating event frequencies performed for this RAI. In general, this sensitivity study shows that the fire area/compartment CDFs increased when considering the shutdown fire events data. Fire scenario FRB3MC (a fire conservatively assumed to involve MCCs 8E, 9D and 89A) is now above the 1E-06/yr screening threshold. As shown in Table FIRE-2-4, this sensitivity assessment not surprisingly shows that the list of dominant fire compartments and the list of dominant fire scenarios remain essentially the same. No new fire insights were generated as a result of this sensitivity study.

Severity Factor and Fire Suppression Dependence:

The NRC cites the potential for double counting fire suppression benefits in VY fire scenarios, which also incorporate fire severity factors. VY could not verify the existence of any significant double counting of suppression effort in the VY IPEEE fire scenarios. With respect to the issue in question, two methods of potential double counting can be identified by VY (refer to the response to RAI Fire #4 regarding a separate dependency issue, manual fire fighting and sprinkler systems):

1. Severe Control Room fire scenarios with credit for suppression.

2. Use of EPRI fire severity factors, the calculation of which may have involved discounting fire events with suppression actuation as "not severe".

With respect to the first case, double counting of suppression efforts in severe Control Room fires can be postulated if the analyst employs the EPRI severity factor for Control Room electrical panel fires (as documented in Appendix D of the EPRI Fire PRA Implementation Guide) and a Control Room fire suppression failure probability based on Figure M-1 from Appendix M of The Guide. The reason for this postulated double counting is that these two specific parameters are calculated by EPRI using the same small number of fire events and, thus, may not necessarily be independent. However, this potential double counting issue does not apply to the VY IPEEE Control Room fire assessment. The VY assessment does not use the EPRI Figure M-1 suppression failure curve, but instead employs a conservative 0.1 failure probability for Control Room fire suppression (effectively precluding any double counting of suppression efforts in such scenarios).

With respect to the second case, one can postulate a double counting of suppression effort when employing EPRI severity factors because the EPRI data analysis of severity factors is based on interpretation of documented industry events and a particular fire event may be categorized as "not severe" (based on consideration of eight characteristics) even though the event involves suppression. This issue is also judged not to introduce significant double counting of suppression efforts in the VY IPEEE fire scenarios, for the following reasons:

- VY calculated severity factors (based on review of the EPRI Fire Events Database) for those fire sources in which EPRI does not provide a severity factor. All severity factor calculations performed by VY conservatively categorize every fire event that caused actuation of a fire suppression system and/or required the fire brigade to respond as a "severe" fire. This approach effectively precludes the systematic introduction of any significant dependency between VY calculated severity factors and fire suppression credit.

- The postulated dependency resulting from the use of EPRI severity factors is not a systematic issue. The fact that certain industry fire events involving suppression may be categorized by EPRI as "not severe" is an issue of raw data interpretation and analysis; it is not a statement that fire events that involved suppression which limited the effects of the fire were systematically categorized by EPRI as "not severe" fires. The EPRI data evaluation process merely has included a step to determine or judge fire severity. This determination is dictated by suppression system actuation.

**Table Fire-2-1: Vermont Yankee IPEEE Revised Fire Severity Factors (9/24/99)**

(does not include correction to capture only "at-power" events)



| Location | Fire Source | Total Number of Fires | Vermont Yankee RAI Review: No. of Fires Not Self-Extinguishing | Vermont Yankee RAI Review: No. of Fires Counted/Total No. of Fires | Vermont Yankee RAI Review: Revised VY Severity Factor (Note 1) | EPRI Severity Factor | Revised VY IPEEE Severity Factor (Notes 2 & 3) |
|---|---|---|---|---|---|---|---|
| Reactor Building (BWR) | Electrical Cabinets | 24 | 7 | 0.292 | 0.29 | (Note 5) | 0.29 |
| | Pumps | 12 | 8 | 0.667 | 0.67 | 0.20 | 0.20 |
| | | | | | | | |
| Diesel Generator Rm | Diesel Generator | 65 | 22 | 0.338 | 0.34 | 0.40 | 0.40 |
| | Electrical Cabinets | 6 | 1 | 0.167 | 0.17 | (Note 5) | 0.17 |
| | | | | | | | |
| Switchgear Rm | Electrical Cabinets | 19 | 8 | 0.421 | 0.42 | 0.12 | 0.12 |
| | | | | | | | |
| Battery Room | Batteries | 4 | 0 | 0.000 | 0.10 | (Note 5) | 0.10 |
| | | | | | | | |
| Control Room | Electrical Cabinets | 12 | 3 | 0.250 | 0.25 | 0.20 | 0.20 |
| | | | | | | | |
| Cable Sprd'g Room | Electrical Cabinets | 4 | 2 | 0.500 | 0.50 | 0.15 (Note 4) | 0.15 |
| | | | | | | | |
| Intake Structure | Electrical Cabinets | 3 | 1 | 0.333 | 0.33 | (Note 5) | 0.33 |
| | Fire Pumps | 5 | 1 | 0.200 | 0.20 | 0.20 | 0.20 |
| | Other Pumps | 4 | 1 | 0.250 | 0.25 | 0.20 | 0.20 |
| | | | | | | | |
| Turbine Building | T/G Excitor | 5 | 5 | 1.000 | 1.00 | (Note 5) | 1.00 |
| | T/G Oil | 17 | 12 | 0.706 | 0.71 | (Note 5) | 0.71 |
| | T/G Hydrogen | 7 | 6 | 0.857 | 0.86 | (Note 5) | 0.86 |
| | Electrical Cabinets | 16 | 6 | 0.375 | 0.38 | (Note 5) | 0.38 |
| | Other Pumps | 8 | 4 | 0.500 | 0.50 | 0.20 | 0.20 |
| | Main Feedwater Pumps | 10 | 8 | 0.800 | 0.80 | (Note 5) | 0.80 |
| | Boiler | 2 | 1 | 0.500 | 0.50 | (Note 5) | 0.50 |
| | | | | | | | |
| Radwaste Area | Misc. Components | 11 | 5 | 0.455 | 0.45 | (Note 5) | 0.45 |

**Table Fire-2-1: Vermont Yankee IPEEE Revised Fire Severity Factors (9/24/99)**

(no correction for "at-power" events)

Page 2 of 3

| Location | Fire Source | Total Number of Fires | Vermont Yankee RAI Review | | | EPRI Severity Factor | Revised VY IPEEE Severity Factor (Notes 2 & 3) |
|---|---|---|---|---|---|---|---|
| | | | No. of Fires Not Self-Extinguishing | No. of Fires Counted/Total No. of Fires | Revised VY Severity Factor (Note 1) | | |
| Yard Transformer | Yard Transformer with propagation to TB | 5 | 5 | 1.000 | 1.00 | (Note 5) | 1.00 |
| | Yard Transformer with LOSP | 2 | 2 | 1.000 | 1.00 | (Note 5) | 1.00 |
| | Yard Transformers | 19 | 17 | 0.895 | 0.89 | (Note 5) | 0.89 |
| | | | | | | | |
| Plant-Wide Components | Fire Protection Panels | 3 | 1 | 0.333 | 0.33 | (Note 5) | 0.33 |
| | RPS MG-Sets | 7 | 2 | 0.286 | 0.29 | 0.14 | 0.14 |
| | Non-Qualified Cable Runs | 8 | 4 | 0.500 | 0.50 | (Note 5) | 0.50 |
| | Non-Qualified Junction Box | 2 | 0 | 0.000 | 0.10 | (Note 5) | 0.10 |
| | Transformers | 10 | 4 | 0.400 | 0.40 | 0.10 | 0.10 |
| | Battery Chargers | 5 | 1 | 0.200 | 0.20 | (Note 5) | 0.20 |
| | Off-gas/H2 Recombiners | 41 | 16 | 0.390 | 0.39 | (Note 5) | 0.39 |
| | H2 Tanks | 4 | 2 | 0.500 | 0.50 | (Note 5) | 0.50 |
| | Misc. H2 Fires | 4 | 2 | 0.500 | 0.50 | (Note 5) | 0.50 |
| | Gas Turbines | 4 | 2 | 0.500 | 0.50 | (Note 5) | 0.50 |
| | Air Compressors | 6 | 2 | 0.333 | 0.33 | (Note 5) | 0.33 |
| | Ventilation Subsystems | 12 | 3 | 0.250 | 0.25 | 0.08 | 0.08 |
| | Elevator Motors | 8 | 2 | 0.250 | 0.25 | (Note 5) | 0.25 |
| | Dryers | 11 | 4 | 0.364 | 0.36 | (Note 5) | 0.36 |
| | Transients | 13 | 5 | 0.385 | 0.38 | (Note 5) | 0.38 |
| | Cable Fires (Welding) | 4 | 3 | 0.750 | 0.75 | (Note 5) | 0.75 |
| | Transients (Welding) | 24 | 3 | 0.125 | 0.13 | (Note 5) | 0.13 |

**Table Fire-2-1: Vermont Yankee IPEEE Revised Fire Severity Factors (9/24/99)**
(No correction for "at-power" events)


Notes to Table Fire-2-1.

1. The revised severity factor values are calculated in response to the NRC RAI. Calculated severity factors are based on a detailed review of the EPRI fire events database to determine the number of significant fires occurring in each area. The severity factors represent the ratio of the number of significant fires to the total number of fires. For cases where there were 0 significant fires, i.e., fraction of 0/x, the severity factor is calculated as follows: If x<10, the severity factor was conservatively set at 1.0E-01. If x>10, the severity factor was conservatively set at 1/x.

2. The revised severity factors (SF) are applied in the calculation of the compartment specific fire initiating event frequency for the corresponding fire ignition source.

3. When available, the EPRI severity factors are used in the calculation of the compartment specific fire initiating event frequency for the corresponding fire ignition source.

4. Appendix D of the EPRI Fire PRA Implementation Guide does not provide a severity factor for electrical cabinets located in the Cable Spreading Room. Because the data base contains little data for such fires (only four events which occurred at-power), the EPRI severity factors for Switchgear Room and Control Room electrical cabinets (which are based on a total of 27 events) are used to develop a severity factor for electrical cabinets in the Cable Spreading Room. The EPRI severity factor of 0.2 for Control Room electrical cabinets is based on 10 fire events, and the severity factor of 0.12 for Switchgear Room electrical cabinets is based on 17 fire events. The weighted average [(0.2)*(10) +(0.12)*(17)]/(27) = 0.15 is used as the severity factor for electrical cabinets in the Cable Spreading Room.

5. No EPRI severity factor documented in EPRI TR-105928, Fire PRA Implementation Guide.

| Table FIRE-2-2<br>Revised FIVE Phase II Initial Screening Summary | | | | | | | |
|---|---|---|---|---|---|---|---|
| FIVE Phase I Critical Fire Areas | | FIVE Initial Screening Quantification | | | | | |
| | | VY IPEEE Submittal | | | Sensitivity in Support of RAI | | |
| ID | Description & CCDP | Fire IE | Screening CDF (Note 1) | Screen ? | Revised Fire IE | Revised Screening CDF (Note 1) | Screen ? |
| RB345 | RB, El. 345', CCDP is 2.63E-06. | 3.3E-04 | 8.7E-10 | Y | 5.8E-04 | 1.5E-09 | Y |
| RB318 | RB, El. 318', CCDP is 1.16E-05. | 9.5E-04 | 1.1E-08 | Y | 1.6E-03 | 1.9E-08 | Y |
| RB303 | RB, El. 303', CCDP is 3.05E-04. | 9.5E-04 | 2.9E-07 | Y | 1.6E-03 | 4.9E-07 | Y |
| RB-5 | RB, El. 280', General Area North, CCDP is 1.30E-04. | 2.0E-03 | 2.6E-07 | Y | 5.6E-03 | 7.3E-07 | Y |
| RB-6 | RB, El. 280', General Area South, CCDP is 6.32E-05. | 1.9E-03 | 1.2E-07 | Y | 5.6E-03 | 3.5E-07 | Y |
| RB280M | RB, El. 280 MG-Set Area, CCDP-See Note 2. | 4.3E-05 | Note (2) | N | 4.3E-05 | Note (2) | N |
| RB-3 | RB, El. 252', North, CCDP is 8.82E-02, CSZ-See Note 3. | 1.7E-03 | 1.5E-04 | N | 3.8E-03 | 3.4E-04 | N |
| RB-4 | RB, El. 252', South, Steam Tunnel & SW Corner Room, CCDP is 4.69E-03, CSZ-See Note 3. | 1.3E-03 | 6.1E-06 | N | 2.3E-03 | 1.1E-05 | N |
| RBSEC | RB, SE Corner Room, CCDP is 1.09E-05. | 6.7E-04 | 7.3E-09 | Y | 9.5E-04 | 1.0E-08 | Y |
| RBNEC | RB, NE Corner Room, CCDP is 4.03E-06. | 6.7E-04 | 2.7E-09 | Y | 9.5E-04 | 3.8E-09 | Y |
| RBRCL | RB, El.213', NW Corner Room, CCDP is 5.14E-05. | 7.0E-04 | 3.6E-08 | Y | 1.3E-03 | 6.7E-08 | Y |
| RBRCU | RB, El. 232', NW Corner Room, CCDP is 7.85E-05. | 4.2E-04 | 3.3E-08 | Y | 5.7E-04 | 4.5E-08 | Y |
| RBHP | RB, El. 213', HPCI Room, CCDP is 6.91E-06. | 6.8E-04 | 4.7E-09 | Y | 1.3E-03 | 9.0E-09 | Y |
| RB-1 | RB, El. 213', Torus Area - North, CCDP is 1.75E-04. | 5.6E-04 | 9.8E-08 | Y | 7.2E-04 | 1.3E-07 | Y |
| RB-2 | RB, El. 213' Torus Area - South, CCDP is 1.96E-03. | 5.6E-04 | 1.1E-06 | N | 7.2E-04 | 1.4E-06 | N |
| CR | CB, El. 272', Control Room, CCDP-See Note 2. | 1.6E-03 | Note (2) | N | 2.1E-03 | Note (2) | N |
| CV | CB, El. 262', Cable Vault, CCDP-See Note 2. | 1.9E-03 | Note (2) | N | 2.2E-03 | Note (2) | N |
| CV-BT | CB, El. 262', Battery Room, CCDP-See Note 2. | 3.3E-04 | Note (2) | N | 3.4E-04 | Note (2) | N |
| SGE | CB, El. 248', East Switchgear Room, CCDP is 1.60E-01. | 1.0E-03 | 1.6E-04 | N | 1.4E-03 | 2.2E-04 | N |
| SGW | CB, El. 248', West Switchgear Room, CCDP is 2.00E-01. | 1.0E-03 | 2.0E-04 | N | 1.4E-03 | 2.8E-04 | N |

| Table FIRE-2-2 Revised FIVE Phase II Initial Screening Summary | | | | | | | |
|---|---|---|---|---|---|---|---|
| **FIVE Phase I Critical Fire Areas** | | **FIVE Initial Screening Quantification** | | | | | |
| | | **VY IPEEE Submittal** | | | **Sensitivity in Support of RAI** | | |
| **ID** | **Description & CCDP** | **Fire IE** | **Screening CDF (Note 1)** | **Screen ?** | **Revised Fire IE** | **Revised Screening CDF (Note 1)** | **Screen ?** |
| DGA | TB, El. 252', Diesel Generator Room A, CCDP is 4.13E-05. | 7.5E-03 | 3.1E-07 | Y | 1.1E-02 | 4.5E-07 | Y |
| DGB | TB, El. 252', Diesel Generator Room B, CCDP is 4.19E-05. | 7.4E-03 | 3.1E-07 | Y | 1.1E-02 | 4.6E-07 | Y |
| TURB | TB, Remaining Areas(e.g., FOB, Warehouse, Machine Shop, CCDP is 2.93E-05. | 3.1E-02 | 9.1E-07 | Y | 3.9E-02 | 1.1E-06 | N |
| INTCW | Intake Structure, Circ. Water Pump Room, CCDP is 3.42E-06. | 3.8E-04 | 1.3E-09 | Y | 4.6E-04 | 1.6E-09 | Y |
| INTSW | Intake Structure. Service Water Pump Room, CCDP is 1.33E-04. | 9.0E-04 | 1.2E-07 | Y | 2.3E-03 | 3.1E-07 | Y |
| RADW | Radwaste Building & Corridor, CCDP is 1.22E-05. | 1.8E-03 | 2.2E-08 | Y | 4.3E-03 | 5.2E-08 | Y |
| DISCH | Discharge Structure, CCDP is 2.41E-06. | 2.9E-04 | 7.0E-10 | Y | 3.9E-04 | 9.4E-10 | Y |
| DGOP | Fuel Oil Storage Tank and Transfer Pump House, CCDP is 4.88E-06. | 1.7E-03 | 8.3E-09 | Y | 2.4E-03 | 1.2E-08 | Y |
| AOG | Advanced Off Gas Building, CCDP is 2.59E-06. | 3.7E-02 | 9.6E-08 | Y | 5.3E-02 | 1.4E-07 | Y |
| RHOUSE | Switchyard Relay House, CCDP is 1.68E-04. | 1.6E-03 | 2.7E-07 | Y | 2.4E-03 | 4.0E-07 | Y |
| MTFMR | Main and Auxiliary Transformers, CCDP is 1.70E-05. | 4.0E-03 | 6.8E-08 | Y | 4.0E-03 | 6.8E-08 | Y |
| STFMR | Start-Up Transformers, CCDP is 1.64E-04. | 1.7E-03 | 2.8E-07 | Y | 1.7E-03 | 2.8E-07 | Y |

Table Notes:

(1): Screening CDF = (Fire IE) * (CCDP)

(2): No explicit quantification performed in the original VY IPEEE Submittal; judgement used in determining initial screening.

(3): Reactor Building Cable Separation Zone adjacent to RB-3 and RB-4 was not screened in the original VY IPEEE Submittal and was evaluated with detailed fire modeling.

## Table FIRE-2-3

### Summary of Detailed Fire Scenario Re-Assessment

| Fire Areas Not Screened in FIVE Phase II | | | | Comparison of Detailed Fire Scenario Results | | | |
|---|---|---|---|---|---|---|---|
| | | | | VY IPEEE Submittal | | Sensitivity in Support of RAI | |
| **Area ID** | **Scenario Initiator** | **Description & CCDP** | **Non-Suppress Prob.** | **Fire IE** | **CDF (Note 1)** | **Revised Fire IE** | **Revised CDF (Note 1)** |
| RB280M | F280MG | Recirc. MG-Set Fire, El. 280, auto suppression credited, CCDP is 1.58E-01. | 0.05 | 4.3E-05 | 3.4E-07 | 4.3E-05 | 3.4E-07 |
| RB-3 | FRB3CL | RB3 Self-Ignited Cable Fire, no suppression credited, CCDP is 4.00E-03. | 1.0 | 3.5E-04 | 1.4E-06 | 4.7E-04 | 1.9E-06 |
| | FRB3MC | RB3 In-situ MCC Panel Fire, no suppression credited, CCDP is 1.42E-03. | 1.0 | 4.3E-04 | 6.1E-07 | 1.5E-03 | 2.1E-06 |
| | FRB3OL | RB3 Transient Oil Spill Fire, no suppression credited, CCDP is 4.14E-03. | 1.0 | 2.9E-06 | 1.2E-08 | 3.0E-06 | 1.2E-08 |
| | FRB3TR | RB3 Transient/In-situ Class A Trash Fire, no suppression credited, CCDP is 3.93E-03. | 1.0 | 2.8E-04 | 1.1E-06 | 2.9E-04 | 1.1E-06 |
| RB-4 | FRB4CL | RB4 Self-Ignited Cable Fire, no suppression credited, CCDP is 1.68E-03. | 1.0 | 2.8E-04 | 4.7E-07 | 3.9E-04 | 6.6E-07 |
| | FRB4CR | RB4 CRD Repair Room Fire (conservatively includes SWC room frequencies), no suppression credited, CCDP is 3.26E-03. | 1.0 | 4.6E-04 | 1.5E-06 | 6.8E-04 | 2.2E-06 |
| | FRB4MC | RB4 In-situ MCC Panel Fire, no suppression credited, CCDP is 1.50E-05. | 1.0 | 1.2E-04 | 1.8E-09 | 4.2E-04 | 6.3E-09 |
| | FRB4OL | RB4 Transient Oil Spill Fire, no suppression credited, CCDP is 2.70E-03. | 1.0 | 2.9E-06 | 7.8E-09 | 3.0E-06 | 8.1E-09 |
| | FRB4TR | RB4 Transient/In-situ Class A Trash Fire, no suppression credited, CCDP is 1.61E-03. | 1.0 | 2.8E-04 | 4.5E-07 | 2.9E-04 | 4.7E-07 |
| RB-CSZ | FRBSZ1 | Cable Separation Zone Division S1 Fire Affecting Division S2 Trays, auto & manual suppression credited, CCDP-See Note 2. | 0.05 & 0.1 | 1.2E-04 | <6.0E-07 (Note 2) | 1.5E-04 | <7.5E-07 (Note 2) |
| | FRBSZ2 | Cable Separation Zone Division S2 Fire Affecting Division S1 Trays, auto & manual suppression credited, CCDP-See Note 2. | 0.05 & 0.1 | 8.0E-05 | <4.0E-07 (Note 2) | 1.0E-04 | <5.0E-07 (Note 2) |
| RB-2 | FRB2X1 | Torus Room, El. 213' South, No propagation of in-situ pump fire, pumps eliminated from sources, no suppression credited, CCDP is 1.89E-03. | 1.0 | 3.6E-04 | 6.8E-07 | 3.9E-04 | 7.4E-07 |
| CR | FCR | Control Room Fire, El. 272', manual suppression credited, CCDP is 2.70E-02. | 0.1 | 1.3E-03 | 3.5E-06 | 2.1E-03 | 5.7E-06 |

## Table FIRE-2-3

## Summary of Detailed Fire Scenario Re-Assessment

| Fire Areas Not Screened in FIVE Phase II | | | | Comparison of Detailed Fire Scenario Results | | | |
|---|---|---|---|---|---|---|---|
| | | | | VY IPEEE Submittal | | Sensitivity in Support of RAI | |
| Area ID | Scenario Initiator | Description & CCDP | Non-Suppress Prob. | Fire IE | CDF (Note 1) | Revised Fire IE | Revised CDF (Note 1) |
| CV | FCVS1F | CV Division S1 Panel Fire Affecting Division S2 Trays, auto suppression credited, CCDP is 2.70E-02. | 0.04 | 1.3E-04 | 1.4E-07 | 3.0E-04 | 3.2E-07 |
| | FCVS2F | CV Division S2 Panel Fire Affecting Division S1 Trays, no suppression credited, CCDP is 2.70E-02. | 1.0 | 3.4E-04 | 9.2E-06 | 4.9E-04 | 1.3E-05 |
| | FCVCL | CV Self-Ignited Cable Tray Fire, auto suppression credited, CCDP is 2.70E-02. | 0.04 | 9.7E-04 | 1.1E-06 | 1.3E-03 | 1.4E-06 |
| CVBT | FCVBAT | CV Battery Room Fire, no suppression credited, CCDP is 2.70E-02. | 1.0 | 1.2E-04 | 3.3E-06 | 1.2E-04 | 3.2E-06 |
| SGE | FSG24 | East Switchgear Room Fire at Bus 2/9, auto suppression credited, CCDP is 2.26E-01. | 0.04 | 3.2E-04 | 2.9E-06 | 4.9E-04 | 4.4E-06 |
| | FSGE4 | East Switchgear Room Fire at Bus 4, auto suppression credited, CCDP is 2.12E-01. | 0.04 | 6.0E-05 | 5.1E-07 | 9.0E-05 | 7.6E-07 |
| | FSGET | East Switchgear Room Fire at Transformer T-9, no suppression credited, CCDP is 1.64E-01. | 1.0 | 7.3E-06 | 1.2E-06 | 1.1E-05 | 1.8E-06 |
| SGW | FSGW1 | West Switchgear Room Fire at Bus 1/8, auto suppression credited, CCDP is 2.72E-01. | 0.04 | 2.3E-04 | 2.5E-06 | 3.4E-04 | 3.7E-06 |
| | FSGW3 | West Switchgear Room Fire at Bus 3, auto suppression credited, CCDP is 3.60E-01. | 0.04 | 1.4E-04 | 2.0E-06 | 2.2E-04 | 3.2E-06 |
| | FSGWT | West Switchgear Room Fire at Transformer T-8, no suppression credited, CCDP is 1.92E-01. | 1.0 | 7.3E-06 | 1.4E-06 | 1.1E-05 | 2.1E-06 |

Table Notes:

(1): CDF = (non-suppression probability) * (CCDP) * (Fire IE)

(2): Refer to the response to Fire RAI Question No. 4. A detailed quantification is not performed.

**Table FIRE-2-4**

**Comparison of VY Dominant Fire Areas and Scenarios**

| VY IPEEE Submittal | | VY Response to RAI | |
|---|---|---|---|
| **Fire Compartment** | **CDF** | **Fire Compartment** | **CDF** |
| CV | 1.0E-05 | CV | 1.5E-05 |
| SGW | 5.9E-06 | SGW | 9.0E-06 |
| SGE | 4.6E-06 | SGE | 6.9E-06 |
| CR | 3.5E-06 | RB3 | 5.1E-06 |
| CVBT | 3.3E-06 | CR | 5.7E-06 |
| RB3 | 3.1E-06 | CVBT | 3.2E-06 |
| RB4 | 2.4E-06 | RB4 | 3.3E-06 |
| | | TURB | 1.1E-06 |
| **Fire Scenario** | **CDF** | **Fire Scenario** | **CDF** |
| FCVS2F | 9.2E-06 | FCVS2F | 1.3E-05 |
| FCR | 3.5E-06 | FCR | 5.7E-06 |
| FCVBAT | 3.3E-06 | FSG24 | 4.4E-06 |
| FSG24 | 2.9E-06 | FSGW1 | 3.7E-06 |
| FSGW1 | 2.5E-06 | FCVBAT | 3.2E-06 |
| FSGW3 | 2.0E-06 | FSGW3 | 3.2E-06 |
| FRB4CR | 1.5E-06 | FRB4CR | 2.2E-06 |
| FRB3CL | 1.4E-06 | FRB3MC | 2.1E-06 |
| FSGWT | 1.4E-06 | FSGWT | 2.1E-06 |
| FSGET | 1.2E-06 | FRB3CL | 1.9E-06 |
| FRB3TR | 1.1E-06 | FSGET | 1.8E-06 |
| FCVCL | 1.1E-06 | FCVCL | 1.4E-06 |
| | | FRB3TR | 1.1E-06 |
| | | FTURB | 1.1E-06 |

**FIRE RAI 3:**

The heat loss factor (HLF) is defined as the fraction of energy released by the fire that is transferred to the enclosure boundaries. This is a key parameter in the prediction of component damage, as it determines the amount of heat available to the hot gas layer. A larger HLF means that a larger amount of heat (due to a more severe fire, a longer burning time, or both) is needed to cause a given temperature rise. It can be seen that if the value assumed for the HLF is unrealistically high, fire scenarios can be improperly screened out. Figure R.1 (see NRC RAI letter) provides a representative example of how hot gas layer temperature predictions change assuming different HLFs. Note that: 1) the curves are computed for a 1000 kW fire in a 10m x 5m x 4m compartment with a forced ventilation rate of 1130 cfm; 2) the FIVE-recommended damage temperature for qualified cable is 700F for qualified cable and 425F for unqualified cable; and 3) the Society for Fire Protection Engineers (SFPE) curve in the figure is generated from a correlation provided in the SFPE Handbook.

Based on evidence provided by the 1982 paper by Cooper et al., the EPRI Fire PRA Implementation Guide recommends a HLF of 0.94 for fires with durations greater than five minutes and 0.85 for "exposure fires away from a wall and quickly developing gas layers." However, as a general statement, this appears to be a misinterpretation of the results.

Reference R.3 (Cooper et al. paper), which documents the results of multi-compartment fire experiments, states that the higher HLFs are associated with the movement of the hot gas layer from the burning compartment to adjacent, cooler compartments. Earlier in the experiments, where the hot gas layer is limited to the burning compartment, Reference R.3 reports much lower HLFs (on the order of 0.51 to 0.74). These lower HLFs are more appropriate when analyzing a single compartment fire.

In summary, (a) hot gas layer predictions are very sensitive to the assumed value of the HLF; and (b) large HLFs cannot be justified for single-room scenarios based on the information referenced in the EPRI Fire PRA Implementation Guide.

For each scenario where the hot gas layer temperature was calculated, please specify the heat loss factor value used in the analysis. In light of the preceding discussion, please either: a) justify the value used and discuss its effects on the identification of fire vulnerabilities, or b) repeat the analysis using a more justifiable value and provide the resulting change in scenario contribution to CDF.

**FIRE RAI 3 RESPONSE:**

The VY IPEEE internal fires risk analysis was performed based on pertinent industry guidance documents available and current at the time of the Submittal development. The following key points were considered in the development of the heat loss factor (HLF) application approach of the VY IPEEE internal fires assessment:

1. EPRI TR-100370, Fire-Induced Vulnerability Evaluation (FIVE): [Reference 3]

*This document describes the implementation of the basic fire screening methodology and the transient analysis methodology. . . . A separate document, entitled "Simplified Methods of Quantitative Fire Hazard Analysis," describes the technical bases for the methodologies.* (p. 10.4-5)

Box 20 - Heat loss factor . . . *A value of 0.7 should normally represent a conservative value for this parameter. The smaller the value used, the more conservative the analysis will be.* (p. 10.4-21)

*The fire screening methodology described here is intended to provide a preliminary analysis of the fire hazard represented by a specified enclosure fire scenario. This preliminary screening methodology uses conservative estimates of fire parameters to provide conservative estimates of fire hazard conditions. Some of these conservative input estimates include:*

*Exposure fires instantly attain their peak intensities;*

*Fires burn with unit heat release rates associated with fully involved conditions;*
*Only 70 percent of energy released is lost to boundaries;*
*Heat loss by convection in ventilated room fires is neglected;*
*Plume and hot gas layer temperature effects are superimposed.* (p. 10.4-31)

2. EPRI TR-105928, Fire PRA Implementation Guide: [Reference 2]

***Heat Loss Factor.*** *FIVE suggests 0.70 as a conservative value and 0.85 as a realistic value. The reference used in FIVE provides test data which report values of 0.74 to 0.93 at 2.5 minutes (depending on the size of the fire source and enclosure) and 0.93 to 0.99 at 5 minutes. . . . COMPBRN calculations indicate higher values for sample (0.98 at 5 minutes). The sample case was benchmarked to 20′ separation experiment. We expect a better value from COMPBRN because of concrete heat sinks (versus gypsum wall board in tests).*

*Another reference indicates potential dependency on room aspect ratio:*

*0.6 - ceiling span divided by height is high, smooth ceilings, and fire far from walls*
*0.9 - fire close to walls*

*The EPRI Fire PRA recommends using at least 0.94 for times ≥ 5 minutes where the whole compartment is filled with HGL. However, smaller values (0.85) should be considered appropriate for exposure fire scenarios away from a wall and quickly developing hot gas layers (e.g., flammable liquid pool fires).* (pp. 4-28 - 4-29)

3. EPRI TR-100443, Methods of Quantitative Fire Hazard Analysis: [Reference 4]

*The heat loss factor is a complex function of the fire history, the boundary thermal properties and the effective heat transfer surface area. . . . For screening purpose, the heat loss factor is treated as a constant. Heat loss factors typically range between 70 to 95 percent of the total energy released in enclosure fires. Since the lower value produces a more conservative temperature rise estimate, a heat loss fraction of 0.7 is suggested for screening purposes.* (p. 24)

***7.2.4. Layer Descent Analysis.*** *. . . A comparison of calculated temperature histories for an example case shows the average temperature rise in a space 12m by 18m by 6m high when subjected to a steady 1000kW fire source located at floor level. . . . For these calculations, the heat loss factor was set to 0.85, a value deemed closer to reality for most scenarios than the value of 0.70 suggested for screening purposes.* (p. 27)

As stated in Section 4.4.4 of the VY IPEEE Submittal, a HLF of 0.70 was generally used as a screening value. A HLF of 0.85 was employed in selected cases when screening assessments indicated less conservative assessments were warranted. This approach is judged appropriate and is consistent with the available industry guidance regarding internal fire hazard analysis.

In response to this RAI, VY reviewed the documented deterministic fire models performed in support of the IPEEE Submittal, and the experimental studies referenced in this RAI (1982 Cooper Test Results and Society of Fire Protection Engineering (SFPE) Handbook) and in the above guidance documents. Based on our review of this information, the following statements regarding the application of HLFs in IPEEE internal fire assessments are judged appropriate:

1. Instantaneous HLFs range from approximately 0.60 to 0.95 and vary with, among other factors, time (increasing trend with time), burn rate, and enclosure characteristics. As the IPEEE fire models employ a constant HLF for the duration of the modeled fire, selection of the HLF needs to consider the duration of the fire modeled and the range of instantaneous HLFs indicated from experiments.

2. The 1982 Cooper experimental findings result from tests that differ in two significant areas from the IPEEE internal fires applications:

(a) The Cooper tests results benefit from heat transfer to the entire enclosure volume as they are performed with the fire placed effectively on the floor and instrumentation placed throughout the enclosure from 0.15 m above the floor to 0.07 m below the ceiling. In contrast, the fire scenarios modeled in the VY IPEEE, in almost all cases, involve fire sources placed many feet above the floor and the deterministic fire models only credit the enclosure volume above the fire. This factor would generally preclude the lower range of HLFs as a reasonable value for use in IPEEE non-screening assessments in which the fire is located above the floor.

(b) The Cooper tests are performed in gypsum board lined enclosures devoid of any major contents. In contrast, the areas modeled in the VY IPEEE are built of concrete and steel and contain large masses of steel components within their boundaries. The thermal conductivity for concrete is approximately 2-3 times that of gypsum. If the tests were performed in enclosures constructed of concrete walls and ceilings and with large mass equipment located in the rooms, the average gas temperatures would be lower. This factor would generally increase the range of HLFs calculated in the referenced experiments.

3. Instantaneous HLF values range from approximately 0.65 to 0.75 for short duration fires (i.e., less than or equal to approximately 1-2 minutes). As such, a HLF value of 0.70 is conservative in almost all cases when applied as a constant value for the entire duration of the fire. This is consistent with the guidance documents referenced above.

4. Application of a HLF value greater than approximately 0.70 to 0.75 is not appropriate for very short duration fires (i.e., less than or equal to approximately 1-2 minutes).

5. Application of a constant HLF value of 0.85 for the entire duration of a modeled fire is an appropriate approach when performing reasonable, non-screening assessments for fire durations greater than or equal to five minutes. This is consistent with the 0.85 "realistic" value cited in the above referenced guidance documents and with the Friedman experimental results presented in the SFPE Handbook. The 0.85 is acceptable for freely connected multi-room and single-room (given item #2 above) enclosures.

6. Application of a constant HLF value in the range of 0.90 to 0.95 may be appropriate when performing reasonable, non-screening assessments for fires greater than 5 minutes in duration and occurring in freely connected multi-room enclosures, but is not appropriate for general application given the lack of fire modeling insights (i.e., the purpose of PSAs in general and the IPEEE in specific) that may result. The following factors should exist when applying such a HLF value: freely connected multi-room compartment, low aspect room geometry, irregular ceiling surface, fire duration significantly greater than 5 minutes, fire located close (within 1 room height) to a wall.

VY IPEEE Deterministic Fire Hazard Models

Deterministic fire growth and propagation analyses were performed in support of the VY IPEEE to provide a bases for fire compartment boundaries, critical damage distances in specific and generic fire scenarios, and time to damage in specific fire scenarios.

A summary of the VY IPEEE internal fires deterministic modeling cases incorporating a HLF is provided in Table Fire-3-1.

Over two hundred (212) deterministic fire modeling cases are documented in calculational packets as Tier 2 documentation (consistent with NRC guidance regarding IPE and IPEEE documentation approach) to the VY IPEEE internal fires assessment. Heat Loss Factors are not incorporated into many of these cases due to a variety of appropriate reasons, such as: the case is a radiant heat calculation; the case is a transient thermal response case; target damage indicated, without consideration of the average room temperature rise contribution.

Twenty-six of the documented cases explicitly incorporate a HLF into the calculation. Of these 26 cases, 21 cases use the conservative value of 0.70 and 5 cases use the value of 0.85. These 5 cases are:

RB3SZ.01H: Determine ceiling jet and hot gas layer effects on divisional cables from postulated RWP clothing fire associated with temporary radiation protection change-out area, RB Cable Separation Zone (CSZ).

RB3SZ.02H: Determine ceiling jet and hot gas layer effects on divisional cables from postulated Class A trash fire associated with temporary radiation protection change-out area, RB Cable Separation Zone (CSZ).

SZ.DET Pocket: Determine time for hot gas layer in the cable separation zone to attain an average temperature of 425F for the S1 cable beam pocket from a postulated self-ignited cable fire in Division S2, RB Cable Separation Zone (CSZ).

WSWGR.02aHGL: Determine time to cable damage from a Bus 1 or 3 fire in the West Switchgear Room.

ESWGR.02aHGL: Determine time to cable damage from a Bus 2 or 4 fire in the East Switchgear Room.

The compartment and fire scenario characteristics for these cases are summarized in Table Fire-3-2. The application of the 0.85 HLF value for these cases is judged appropriate given the guidance discussed above and knowledge from the reviewed experimental results. All cases have the following characteristics:

- low aspect compartment geometry
- concrete compartment construction
- irregular ceiling surface
- fire located above the floor
- fire located within 1 room height from walls
- fire duration $\geq$ 5 minutes

Cases RB3SZ.01H, RB3SZ.02H and SZ.DET Pocket model fire aspects in the RB Cable Separation Zone (CSZ). These cases have an additional characteristic in that the separation zone has multi-room features. The CSZ is located in the northwest corner of Reactor Building Elevation 252'-6". The CSZ is comprised of two distinct large areas freely connected by a short corridor type area approximately 10' wide. The ceiling is characterized by soffit type pockets bounded by ceiling beams allowing the hot gas behavior to be similar to the Cooper tests. The fire gases travel upward into ceiling areas bounded by ceiling beams. Once the gas interface elevation drops to the bottom of the ceiling beams, smoke and gases spill over into the adjacent ceiling areas and down the corridor area. Pressure differentials caused by differential heights of the lower density gas layer between adjacent areas drive cooler air from the adjacent areas into the area of the fire. Such scenarios can be argued to warrant HLF values greater than 0.85.

**Table Fire-3-1**

**Summary of Heat Loss Factor (HLF) Application in VY IPEEE Fire Modeling**

| Fire Model Case | Fire Model Description | HLF Applied to Fire Modeling Case (Note 1) | | Basis for Application of HLF |
|---|---|---|---|---|
| | | 0.85 | 0.70 | |
| RB1.01P (in-plume) | An in-situ fire in RB1, at the sump pumps on the north wall. Target is 15′-2″ above the source (pumps). | | x | (Note 2) |
| RB1.02P (in-plume) | A transient fire in RB1, against the wall. The source is RWP clothing, and the target is 15′-2″ above the source. | | x | (Note 2) |
| RB1.03P (in-plume) | A transient fire in RB1, against the wall. The source is Class A Trash (2 bags), and the target is 15′-2″ above the source. | | x | (Note 2) |
| RB2.01P (in-plume) | An in-situ fire in RB2, at the sump pumps in the SW area. Target is 19′ above the source (pumps). | | x | (Note 2) |
| RB2.02P (in-plume) | A transient fire in RB2, against the wall. The source is RWP clothing, and the target is 19′ above the source. | | x | (Note 2) |
| RB2.03P (in-plume) | A transient fire in RB2, against the wall. The source is Class A Trash (2 bags), and the target is 19′ above the source. | | x | (Note 2) |
| RB3.01H (Ceiling Jet /HGL) | A hypothetical target is located at the ceiling and 1′ longitudinal distance from the fire is modeled to determine if damage due to ceiling jet or HGL can occur (RB3) | | x | (Note 2) |
| RB3.02H (Ceiling Jet / HGL) | No specific target is modeled. This model was run to define the critical distance for damage due to the ceiling jet/HGL. (RB3) | | x | (Note 2) |
| RB4.02H (Ceiling Jet / HGL) | No specific target is modeled. This model was run to define the critical distance for damage due to the ceiling jet / HGL. (RB4) | | x | (Note 2) |
| RB4.07H (ceiling jet / HGL) | This scenario targets the SRV conduits which run directly above the starter box in the CRD repair room (RB4). | | x | (Note 2) |

**Table Fire-3-1**

**Summary of Heat Loss Factor (HLF) Application in VY IPEEE Fire Modeling**

| Fire Model Case | Fire Model Description | HLF Applied to Fire Modeling Case (Note 1) | | Basis for Application of HLF |
|---|---|---|---|---|
| | | 0.85 | 0.70 | |
| RB MG SET A ROOF | This scenario is modeled to determine if there is damage to the structural steel roof as the hot gases pass up through the equipment access hatch. | | x | (Note 2) |
| RB MG SET B ROOF | This scenario is modeled to determine if there is damage to the structural steel roof as the hot gases pass up through the equipment access hatch. | | x | (Note 2) |
| RB3SZ.01H (Ceiling Jet / HGL) | This scenario was run to determine the hot gas layer effects of 2 step-off pads next to a wall. | x | | (Note 3) |
| RB3SZ.02H (Ceiling Jet / HGL) | This scenario was run to determine the hot gas layer effects of 1 trash container next to a wall. | x | | (Note 3) |
| SZ.DET Pocket | This scenario was run to determine the time for the HGL in the SZ (from self-ignited cables) to attain an average temperature of 425°F for the SZ SI ceiling pocket. | x | | (Note 3) |
| RB TR #1.2 (Ceiling Jet / HGL) | Two trash bags are placed at the edge of the Torus Room (RCIC side) of the CFZ. This was done to determine if Torus Room CFZ are adequate for separation of RB 232 (Torus Room). | | x | (Note 2) |
| RB TR #3.2 (Ceiling Jet / HGL) | Two trash bags are placed at the edge of the Torus Room (East side) of the CFZ. This was done to determine if Torus Room CFZ are adequate for separation of RB 232 (Torus Room). | | x | (Note 2) |
| RB NEU #1.2 (Ceiling Jet / HGL) | An oil spill of bearing oil from one of two pumps is modeled. | | x | (Note 2) |
| RB NEU #2.2 (Ceiling Jet / HGL) | This scenario is the same as RB NEU #1.2 except it is assumed that the oil fire is the size of the stairwell opening and the spill occurs on El. 252'. (HGL Effects) | | x | (Note 2) |

**Table Fire-3-1**

**Summary of Heat Loss Factor (HLF) Application in VY IPEEE Fire Modeling**

| Fire Model Case | Fire Model Description | HLF Applied to Fire Modeling Case (Note 1) | | Basis for Application of HLF |
|---|---|---|---|---|
| | | 0.85 | 0.70 | |
| RB 252 #1.1 (In-Plume) | This model is done to determine if the Torus Room can be separated from the El. 252'. The pathways of concern are two manways in the El. 252' floor into the ceiling of the Torus Room. | | x | (Note 2) |
| RB 252 #2.2 (Ceiling Jet / HGL) | The model is done determine if a fire in MCC 89A or MCC 89B is great enough ignite targets in RB4 or RB3, respectively, across the CFZ. (Ceiling Jet / HGL) | | x | (Note 2) |
| RB 252 #3.2 (Ceiling Jet / HGL) | This scenario models a transient floor based fire located directly in between the two sets of cable trays in the RB3B Separation Zone (SZ). The suppr. system is not credited for this screening analysis. (CJ/HGL) | | x | (Note 2) |
| WSWGR_FL.01a (Timing) | This model was run to determine the burn time and the time to damage for the cables in the West SWGR room, using 1 quart of EPA 2000. | | x | (Note 2) |
| WSWGR.02aHGL (Timing) | This scenario was run to determine the time to cable damage from a Bus 1 or 3 fire in the West switchgear room. | x | | (Note 3) |
| ESWGR_FL.01a (Timing) | This model was run to determine the burn time and the time to damage for the cables in the East SWGR room, using 1 quart of EPA 2000. | | x | (Note 2) |
| ESWGR.02aHGL (Timing) | This scenario was run to determine the time to cable damage from a Bus 2 or 4 fire in the East switchgear room. | x | | (Note 3) |

**NOTES TO TABLE FIRE-3-1:**

(1) Of the 212 deterministic fire damage modeling cases, 182 cases did not apply HLFs (e.g., radiant case, damage already indicated without consideration of average room temperature rise). Of the remaining 26 cases which employed a HLF, 21 cases used the conservative value of 0.70 and 5 cases used the more realistic value of 0.85.

(2) Instantaneous Heat Loss Factors typically range from approximately 0.60 to 0.95 and vary with, among other factors, time (increasing trend with time), burn rate, and enclosure characteristics. HLF values in the range of 0.65 to 0.75 are generally applicable for short duration fires (i.e., less than or equal to approximately 1-2 minutes). As such, a HLF value of 0.70 is conservative in almost all cases when applied as a constant value for the entire duration of the fire. This is consistent with the following industry guidance documents which reference 0.70 as a suitably conservative value for screening purposes:

- EPRI TR-100443, "Methods of Quantitative Fire Hazard Analysis"
- EPRI TR-105928, "Fire PRA Implementation Guide"
- EPRI TR-100370s, "Fire-Induced Vulnerability Evaluation (FIVE)"

(3) Application of a constant HLF value of 0.85 for the entire duration of a modeled fire is an appropriate approach when performing reasonable, non-screening assessments for fire durations greater than or equal to five minutes. This is consistent with the 0.85 "realistic" value cited in the previously referenced guidance documents and with the Friedman experimental results (SFPE Handbook). The 0.85 HLF value is acceptable for freely connected multi-room and single-room enclosures.

**Table Fire-3-2**

**VY IPEEE - Summary of Cases Using Non-Screening HLF Values**

| Case | Case Description | Compartment | | | | | Fire | | | | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Multi-Room | Ceiling Surface | Height (ft.) | Area (ft.2) | Aspect Ratio | Fuel Source | Height Above Floor (ft.) | Distance From Walls (ft.) | Duration (min) | |
| RB3SZ.01H (Ceiling Jet / HGL) | This scenario was run to determine the hot gas layer effects of RP clothing and change-out area next to a wall in the Reactor Building CSZ. | Yes | Irregular | 27 | 3500 | Low (<0.5) | Class A (214 BTU/s) | 3 | <1 Room Height | 30 | HLF of 0.85 is appropriate; higher values can be argued as realistic. Base case shows no damage to cable targets.<br><br>Use of a 0.70 HLF instead of 0.85 also shows no target damage occurs; no model changes would result. |
| RB3SZ.02H (Ceiling Jet / HGL) | This scenario was run to determine the hot gas layer effects of Class A trash fire next to a wall in the Reactor Building CSZ. | Yes | Irregular | 27 | 3500 | Low (<0.5) | Class A (333 BTU/s) | 3 | <1 Room Height | 5 | Base case HLF of 0.85 is appropriate; higher values can be argued as realistic. Base case shows no damage to cable targets.<br><br>Use of a 0.70 HLF instead of 0.85 also shows no target damage occurs; no model changes would result. |

**Table Fire-3-2**

**VY IPEEE - Summary of Cases Using Non-Screening HLF Values**

| Case | Case Description | Compartment | | | | | Fire | | | | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Multi-Room | Ceiling Surface | Height (ft.) | Area (ft.2) | Aspect Ratio | Fuel Source | Height Above Floor (ft.) | Distance From Walls (ft.) | Duration (min) | |
| SZ.DET Pocket | This scenario was run to determine the time for the HGL in the SZ (from self-ignited cables) to attain an average temperature of 425F in the Reactor Building Separation Zone area ceiling pockets. This model supports initiator FRBSZ2. | Yes | Irregular | 27 | 3500 | Low (<0.5) | Cable (HRR varies with time) | >17 | <1 Room Height | 30 | Base case HLF of 0.85 is appropriate and has been justified; higher values can be argued as realistic. Base case shows time to target damage of 12.5 minutes.<br><br>Use of a 0.70 HLF instead of 0.85 would show a time to target damage of approx. 8 minutes and indicate an increase in the estimated fire brigade failure probability. If no manual suppression were credited for the FRBSZ2 event, the conservative CDF (assuming all sys. fail) would increase from the IPEEE value of <5.2E-07/yr to approx. <5.2E-06/yr. No new insights are identified.<br><br>Use of a 0.90 HLF instead of 0.85 would show a time to target damage of approx. 15 minutes; no modeling changes would result. |

**Table Fire-3-2**

**VY IPEEE - Summary of Cases Using Non-Screening HLF Values**

| | | Compartment | | | | | Fire | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Case | Case Description | Multi-Room | Ceiling Surface | Height (ft.) | Area (ft.2) | Aspect Ratio | Fuel Source | Height Above Floor (ft.) | Distance From Walls (ft.) | Duration (min) | Comment |
| WSWGR.02a HGL (Timing) | This scenario was run to determine the time to cable damage from a Bus 1 or 3 fire in the West switchgear room. | No | Irregular | 13 | 1650 | Low (<0.5) | Bus Cubicle (515 BTU/s) | 7 | <1 Room Height | 20 | Base case HLF of 0.85 is appropriate for this case. Base case shows that cables located in a radius of approx. 6 feet from fire source would be damaged at 5.7 min. if suppression did not actuate (suppression actuation of current SWGR CO2 sys. occurs at approx. 85 sec.).<br><br>Use of a 0.70 HLF instead of 0.85 would show damage to cables located in a radius of approx. 6 feet in 2.7 minutes; this would not change the conclusions of the base case FSGW1 & FSGW3 evaluations, given the location of credited equipment & cables when suppression is success. |

**Table Fire-3-2**

**VY IPEEE - Summary of Cases Using Non-Screening HLF Values**

| Case | Case Description | Compartment | | | | | Fire | | | | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Multi-Room | Ceiling Surface | Height (ft.) | Area (ft.2) | Aspect Ratio | Fuel Source | Height Above Floor (ft.) | Distance From Walls (ft.) | Duration (min) | |
| ESWGR.02a HGL (Timing) | This scenario was run to determine the time to cable damage from a Bus 2 or 4 fire in the East switchgear room. | No | Irregular | 13 | 1650 | Low (<0.5) | Bus Cubicle (515 BTU/s) | 7 | <1 Room Height | 20 | Base case HLF of 0.85 is appropriate for this case. Base case shows that cables located in a radius of approx. 6 feet from fire source would be damaged at 6.88 min. if suppression did not actuate (suppression actuation of current SWGR CO2 sys. occurs at approx. 85 sec.).<br><br>Use of a 0.70 HLF instead of 0.85 would show damage to cables located in a radius of approx. 6 feet in 3.3 minutes; this would not change the conclusions of the base case FSG24 & FSGE4 evaluations, given the location of credited equipment & cables when suppression is success. |

**FIRE RAI 4:**

There are a number of points related to the analysis of the Reactor Building Cable Separation Zone (CSZ) that require clarification. First, the treatment of manual suppression appears optimistic. A 90% manual suppression reliability appears to have been assumed (a manual non-suppression probability of 0.1 is cited). A fire brigade initial response time of ten minutes was estimated for this zone; however, the initial brigade response time should not be equated to the fire suppression time. Suppression time must include the time needed to detect the fire, verify the fire, assemble and equip the fire fighting team, assess the fire situation, and actively suppress the fire. Further, manual suppression reliability should consider the time to fire damage as compared to the time required for suppression. In the study a damage time of 12.5 minutes is cited for damage to cable tray Division S1 and Division S2. Given an initial brigade response time of 10 minutes (after detection), it would appear highly optimistic to assume a 90% manual suppression reliability before damage occurs. Secondly, the bases of the fire ignition frequencies, including any severity or correction factors, used for this zone were not described. This aspect of the analysis may also be impacted by the concerns discussed in RAI number 2 above. Third, the dependence between failure of the automatic system and failure of manual fire fighting support systems (e.g., hose stream standpipes) was not discussed. For example, if the fire water supply to the sprinkler system fails, then it would appear likely that the water supply to manual hose stream might also fail. Fourth, the compartment is cited as being covered by a partial pre-action sprinkler system. It is not clear what coverage this system provides, how this system was credited in the analysis, or how its effectiveness against the postulated fire scenarios was assessed. Finally, it is not clear what systems are potentially impacted by fire in this area.

For the Reactor Building CSZ, please discuss in more detail the fire protection features of this area including a description of the fire detection system, fixed fire suppression system, and systems needed to support manual fire fighting activities. Describe how the effectiveness of the fixed suppression system was assessed and credited in the analysis. Discuss the time required for fire detection and verification, initial fire brigade response, assembling and equipping the brigade, fire situation assessment, and actual fire suppression for the postulated fires in this zone. Identify the plant systems and equipment located in this area and discuss their importance to plant safety. Describe the fire scenarios postulated for this area, including the bases for the assumed fire ignition frequencies associated with each fire scenario and consideration of concerns identified in RAI number 2 above. Also discuss the conditional core damage probability (CCDP) for loss of all equipment in the room for each of the postulated fire scenarios. Discuss the estimated time(s) to critical component damage that have been assumed in the analysis and their bases. Provide an explicit justification for the assumed manual suppression reliability estimate used in the analysis or repeat the analysis for the CSZ using more realistic manual non-suppression probabilities. Assess the potential for dependence between failure of automatic suppression and failure of manual fire fighting support systems. Given the above factors, reassess the CDF contribution of this compartment.

**FIRE RAI 4 RESPONSE:**

The Cable Separation Zone (CSZ) is located in the northwest corner of Reactor Building Elevation 252'. Fire areas adjacent to the CSZ include RB3 and RB4, with the CSZ being freely connected to RB3 via a short corridor type area. Elevation 252', including the CSZ is a large area with a high ceiling. Building construction materials include steel and concrete. (Refer to the response to RAI Fire #3 for further discussion of this area.

<u>Key Equipment Located in CSZ</u>

The major plant systems that are degraded/failed by a divisional cable tray fire are summarized below:

<u>Division S1:</u>
ECCS Logic Division B Instrumentation
HPCI
RCIC (RCIC failure not guaranteed, may isolate if hot short occurs)

EDG-1B (remote control from control room degraded)
SRVs 72-1A and 1B (fail to open for depressurization, no hot short potential)
Inboard MSIV power
CRD pump B
RBCCW
Power to RPS Cabinet 5A
LPCI/RHR Division B
Core Spray B
Divisional containment isolation valves
Divisional isolation of Service Water to balance of plant equipment (normally open SW valves MOV-19A/B, may isolate if hot short occurs)

<u>Division S2:</u>
ECCS Logic Division A Instrumentation
HPCI (HPCI failure not guaranteed, may isolate if hot short occurs)
RCIC
EDG-1A (remote control from control room degraded)
SRVs 72-1C and 1D (fail to open for depressurization, no hot short potential)
Outboard MSIV power
CRD pump A
RBCCW
Power to RPS Cabinet 6A
LPCI/RHR Division A
Core Spray A
Divisional containment isolation valves
Divisional isolation of Service Water to balance of plant equipment (normally open SW valve MOV-20, may isolate if hot short occurs)

The majority of the electrical cables supplying Reactor Building equipment are routed from the Cable Vault through the CSZ. These cables are divided into Division S1 and Division S2 cable tray systems. Within the cable separation zone, the divisional tray systems are located in the overhead and are separated horizontally from each other by >17-1/2'. Division S1 cables are routed through the west side of the CSZ to area RB4. These cables supply loads located on the south side of the Reactor building (Division S1 equipment). Division S2 cables are routed through the east side of the CSZ to area RB3. These cables supply Division S2 equipment that is located on the north side of the Reactor Building.

<u>Fire Detection and Suppression Equipment</u>

As discussed in Sections 4.8 and 4.9.8 of the VY IPEEE Submittal, ionization detectors installed on the ceiling provide automatic fire detection within the CSZ. Ionization detectors are reliable, highly sensitive to products of combustion (smoke), and therefore, provide effective detection of fire events. Nine (9) detectors are located on the CSZ ceiling, and a total of 96 detectors are located on the ceiling throughout Reactor Building El. 252' (RB3, RB4 and CSZ). The Reactor Building fire protection panel is located in the Radwaste Corridor outside of the Reactor Building. This panel is supplied by AC power and has automatic DC power backup. Thus, the fire panel and automatic detection circuitry are very reliable. Consistent with the EPRI Fire PRA Implementation Guide, the fire panel and automatic detection circuitry are judged to be negligible contributors to the total failure probability of the suppression system.

The CSZ is protected with a pre-action sprinkler system. Sprinkler heads are located above <u>and</u> below the cable trays in the CSZ and coverage within the zone is considered to be 100%. The pre-action sprinkler piping is normally pressurized with air as a means of monitoring the integrity of the piping system. Actuation of a single ionization detector in the cable separation zone will alarm locally and at the main fire panel in the control room and will automatically open a single water control valve (DV-76-301), which permits water flow into the piping system. Heat from the fire subsequently opens the fused-link sprinkler heads allowing delivery of water to suppress the fire.

Manual fire fighting hose stations are available throughout the Reactor Building on each major floor elevation. Two hose stations are located on El. 252'. The north hose station is located adjacent to the cable separation zone, is supplied from the northwest riser, and has sufficient hose length to manually suppress a fire in the cable separation zone. The other hose station is located on the south side of floor El. 252' and is supplied from the southeast riser. Use of the south hose station to suppress a fire at the cable separation zone would require connecting an additional length of hose.

The pre-action suppression system and the manual hose stations are fed from the main fire loop. The design of the main fire loop is reliable in that branch connections are fed from the loop in both directions. There is no single manual isolation valve that is common to both the pre-action system and either of the manual hose stations on El. 252. Thus, the hose stations are independent of the pre-action system, except that both depend on the main fire loop piping (as stated above) and the fire pumps.

The fire loop is supplied by two, 100% capacity pumps each having a diverse driver (diesel-driven fire pump and a motor driven fire pump). Separate controls automatically start the pumps on decreasing fire header pressure. Both pumps take suction directly from the Connecticut River.

The diesel-driven pump is self-contained with a dedicated fuel oil tank and redundant, dedicated batteries for starting the engine. The motor driven pump receives power from 480VAC Switchgear Bus 9, which is backed by emergency diesel generator EDG-1A or the dedicated Vernon Tie Line if normal power is lost. The capability to cross-tie the emergency buses is also available. Note that a fire in a divisional cable tray could degrade control cables to the associated EDG. However, these fire events do not cause a loss of normal power, nor do they effect the reliability of the Vernon Tie Line. Therefore, normal offsite power and the independent Vernon Tie Line must both fail during this event in order to fail the AC power supply to the motor-driven fire pump. Hence, AC power failure to the motor-driven pump is judged to be a negligible contributor to the total failure probability of the motor-driven fire pump. Therefore, it is reasonable and appropriate to credit the failure probability of both fire pumps for CSZ fire scenarios. Note that the fire pumps are not credited for supplying water to the RPV in the CDF calculations for any fire scenario; thus, the only function of the fire pumps in the fire scenarios is to accomplish suppression.

Suppression Credit and Dependence

Analysis of the cable separation zone in the VY IPEEE internal fires assessment probabilistically credits both the automatic fire protection system in the area and the fire brigade manual suppression. The FIVE generic failure probability of 5.0E-02 is used for the pre-action system. This non-suppression probability value is appropriate - refer to RAI Fire #1 response. The manual non-suppression probability is 0.1, which is consistent with FIVE. The basis for this manual non-suppression probability is discussed further below.

Using pump failure probabilities consistent with the VY IPE, and conservatively accounting for the potential for common-cause failure of the pumps, the total failure probability of suppression (Ps) at the CSZ is estimated as follows:

Ps = Pp + (Pa * Pm)

Where:
Ps = failure probability of suppression at CSZ.
Pp = failure probability of fire pumps to supply fire loop (2.3E-04).
Pa = failure probability of pre-action system (5.0E-02 from FIVE).
Pm = failure probability of manual suppression (1.0E-01 from FIVE).

2.3E-04 + (5.0E-02 * 1.0E-01) = 5.23E-03

This total non-suppression probability represents an increase of 2.3E-04 (<5%) to the 5.0E-03 failure probability used in the baseline IPEEE fire evaluation. This increase is judged to be small and well within the uncertainty of the fire analysis. Even when considering this increase, the estimated screening core damage frequency for the CSZ fire scenarios remains below 1E-06. These scenarios are discussed below under the subsection Cable Separation Zone Scenario Quantification.

Basis for Non-Suppression Probability of Manual Suppression

The key aspects of VY manual fire fighting effectiveness are extensively discussed in Section 4.12.3 of the VY IPEEE Submittal. The following are key points from that section:

(a) A fire brigade of five trained staff is maintained at all times in accordance with the VY Technical Requirements Manual.

(b) The brigade leader and at least two other members are trained in plant systems and operations.

(c) The brigade is equipped with complete turnout gear, SCBA, flashlights, and other equipment.

(d) Formal comprehensive training is repeated every two years and covers the VY Fire Protection Plan, pre-fire strategies, fire fighting tactics, equipment location, access/egress routes, communications, lighting, ventilation, and use/maintenance of SCBA.

(e) Live fire schooling is repeated every year. The brigade is challenged by live fires, smoky conditions, and the need to operate various fire suppression equipment. All fire activities are conducted in full gear and SCBA.

(f) Each brigade shift receives a minimum of 4 practice drills per year. At least one occurs on the backshift and one is unannounced.

(g) VY has pre-fire strategy plans for all safety-related areas (including the Separation Zone) and for several non-safety areas.

(h) Inspection of fire brigade equipment is controlled by plant procedures.

The above characteristics provide confidence in the maintained readiness and effectiveness of the VY fire brigade in responding to a fire event in a timely manner and in minimizing the potential of suppression-induced failure of adjacent otherwise functional equipment.

Fire brigade failure is defined as failure to prevent fire spread beyond the initial fuel source such that equipment damage occurs beyond the initial fuel source. The failure probability for manual suppression is assumed to be 0.1, which is consistent with the FIVE methodology. Interviews with plant fire protection and training personnel suggest that a reasonable response time is approximately 10 minutes for the Cable Separation Zone. VY believes that a manual suppression failure probability of 0.1 is reasonable based on the following discussion.

First, the FIVE Methodology approach calculates the manual suppression failure probability based on plant-specific brigade drill results:

$$P_{ms} = 1 - [(\#\text{ of brigade drills performed in } t_d + t_r < t_{crit}) / (\text{total \# of brigade drills})]$$

where:

$P_{ms}$ = manual suppression failure probability

$t_d$ = time to detect fire

$t_r$ = manual suppression response time

$t_{crit}$ = time for target to reach damage temperature

The FIVE approach requires that plant-specific drill results exist with sufficient specificity to complete the above calculation. Historical fire brigade response time data is very limited. Consistent with the conservative screening nature of FIVE, FIVE directs that a failure probability less than 0.1 not be used.

As discussed in Section 4.8 of the VY IPEEE Submittal, review of the EPRI fire events database results in a manual suppression failure probability of approximately 0.03 - 0.04. In this second approach the industry generic experience is used to assess the effectiveness of the fire brigade function (analogous to calculating offsite power recovery failure based on industry events). Review of the EPRI Fire Events Database (FEDB) first identified those events involving brigade response, then the data fields for the ignition source and for components effected by the fire were compared. Depending upon various data interpretations (e.g., Was the fire already extinguished by automatic suppression systems prior to fire brigade arrival?; Was adjacent equipment indeed impacted?), the failure probability is estimated between 0.03 and 0.04.

The FEDB review approach inherently takes into account the realistic nature and the spectrum of fires, such as, they involve various combustion efficiencies and various heat release rate profiles. Using 3.5E-2 as the mean value and applying a lognormal distribution (refer to Figure FIRE-3-1) shows that the brigade failure probability is in the 2E-2 to 3E-1 range. Detrimental shaping factors (e.g., brigade nominal response time equal to or greater than the estimated time to target damage) would place the value at the higher end of the range; non-detrimental shaping factors (e.g., brigade nominal response time much less than the estimated time to target damage) would place the value at the middle to lower end of the range.

Based on the above, 0.1 is considered reasonable for times to damage of approximately 10 minutes or greater.

<u>Cable Separation Zone Fire Scenario</u>

The fire scenario of interest in the CSZ is a self-ignited cable fire in a Division S2 cable tray which could eventually damage Division S1 cable trays. According to the fire hazards model, the target (Division S1 cables) is subject to damage from radiant exposure and hot gas layering. Hot gas layering was specifically evaluated because many Division S1 cables (but not all cables) are routed vertically near the ceiling (beam pocket) before they enter the RB4 steam tunnel. The fire hazards model for this scenario (exposure to Division S1 cables) has the following results:

| | |
|---|---|
| Time for temperature to reach 425F: | 12.5 minutes (hot gas exposure) |
| | 30 minutes (radiant exposure) |
| Time to detect: | 15 seconds |
| Fused-link Sprinkler Head Opening: | 2 minutes |

The 12.5 minute damage time is a conservative (low) estimate based the FIVE fire hazard methodology. According to the FIVE methodology, manual suppression can be credited given our estimate of 10 minutes for fire brigade response. VY's determination to credit manual suppression within 12.5 minutes was based on consideration of both the conservative nature of the FIVE hazards methodology and other simplifying modeling assumptions (summarized below). These conservatisms substantiate that there is adequate time for fire brigade response and that the 0.1 non-suppression probability from FIVE is appropriate for use in the VY evaluation.

1. The initial heat release rate (HRR) and fire growth used in this scenario is conservative given the configuration of the cable trays. The fire source cable trays are positioned in a 2x6 stacked cable tray arrangement. All trays in one stack contain top and bottom metal covers. The other stack of trays consists of 5 trays, each with a top metal cover and open bottom ladder configuration. These metal covers will significantly reduce the rate of vertical fire propagation. The tray at the bottom of this stack consist of two side by side 12" trays, both containing top and bottom covers. The fire hazard scenario assumes that the fire originates in the lowest open bottom tray (origin tray and trays above the origin tray have metal top covers). Although these metal covers will significantly reduce the rate of vertical fire propagation, no reduction factor in propagation is considered in the analysis. Therefore, contrary to FIVE, the actual growth rate of a cable fire in this tray configuration is expected to be very low and the time to damage opposite divisional cable targets significantly delayed.

However, given the sensitivity of the smoke detectors, fire detection is expected to occur early in the event.

2. A HLF of 0.85 is used in the analysis of the Cable Separation Zone, when a higher heat loss factor may be more appropriate based on the geometry of the area, postulated movement of hot gases, and fire duration (refer to earlier response to Fire RAI No. 3). Given the geometry of the CSZ, its ceiling characteristics, and the large volume of El. 252', hot gases are likely to migrate to the general volume/area thereby reducing the overall gas temperature.

3. The time to target damage does not include consideration of the target cable(s) thermal response parameter (TRP). This is conservative because incorporation of the cable TRP into the time to damage calculation would delay the time for actual damage by approximately 1 to 2 additional minutes.

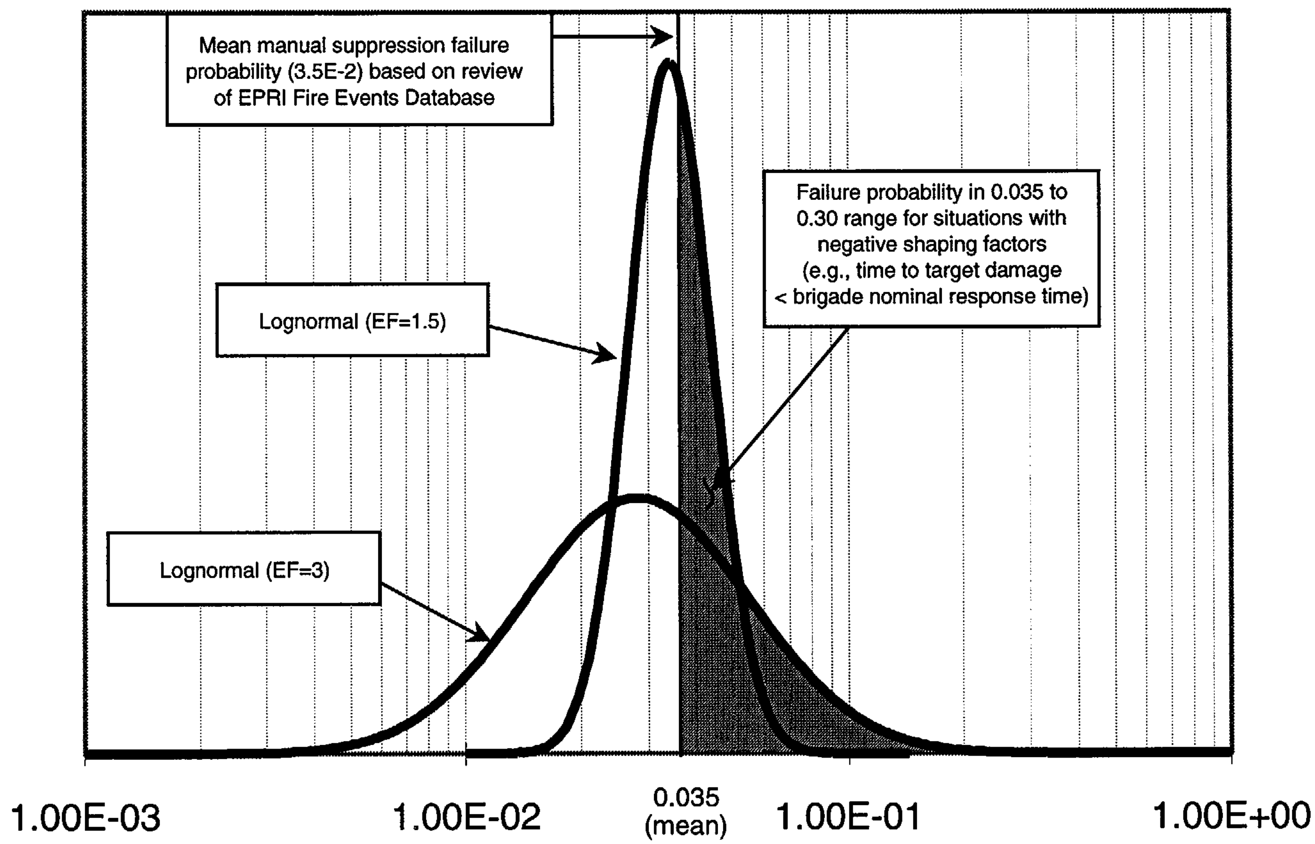
Cable Separation Zone Scenario Quantification

The Reactor Building CSZ fire scenarios were evaluated via two initiators, FRBSZ1 and FRBSZ2 (refer to Tables Fire-2-2 and Fire-2-3 for initiators FRBSZ1 and FRBSZ2). These initiators were not propagated through the VY support and frontline event trees to calculate Core Damage Frequency (CDF) because the initiating event frequency of each event ANDed with the non-suppression probability (per-action and manual) is below the FIVE screening threshold of 1E-06. If these initiators were quantified in detail, credit for equipment that is not guaranteed to fail as a result of the fire would further reduce the FRBSZ1 and FRBSZ2 initiators below the FIVE screening threshold as discussed below.

1. Several systems/components are not guaranteed to fail given the cable damage associated with the CSZ fire scenarios. This is because certain MOVs are not required to change position during system initiation. Thus, a hot short must also occur to cause a spurious valve transfer and system failure. For example, the steam isolation valve for RCIC (MOV-15) is normally open and must remain open during RCIC operation. A hot short in the valve's control cable (Division S1 cable tray) must occur to cause isolation of RCIC. The probability of a hot short is estimated to be in the range of 0.068 [Reference 5] and 0.015 [Reference 6] according to NRC and industry information. Crediting the probability of a hot short for potential failure where appropriate, (e.g., RCIC, HPCI, Service Water cooling to Turbine Building equipment) would further reduce the FIVE screening CDF for the FRBSZ1 and FRBSZ2 fire scenarios.

2. The potential to recover failed systems is also not credited; this is conservative. For example, feedwater/condensate (which is assumed failed because of postulated loss of SW to the Turbine Building Closed Cooling Water System) could be operated intermittently for RPV inventory control. After operators re-establish SW cooling to the turbine building (estimated to take less than 1 hour), feedwater/condensate could be operated indefinitely and the main condenser could be restored. Additionally, the hard-piped torus vent (also not credited) remains available for containment heat removal if main condenser recovery efforts are not successful.

## Figure FIRE-3-1 MANUAL SUPPRESSION FAILURE PROBABILITY DISTRIBUTION



**FIRE RAI 5:**

The detailed analysis of Reactor Building compartment RB3 resulted in a CDF of 3.1E-06/yr. The quantitative screening analysis for this compartment shows a CDF of 1.5E-4/yr based on an ignition frequency of 1.7E-03/yr. This implies a CCDP of about 0.1, presumably assuming loss of all equipment in the area. However, it does not appear that this area is protected by automatic suppression, and the analysis states that manual suppression was not credited in the analysis of this compartment. Hence, it is unclear how the screening CDF of 1.5E-04 was reduced to the final estimated CDF of 3.1E-06/yr.

Please provide a detailed discussion of how this compartment was analyzed, starting with ignition frequency and concluding with CDF. Explain the factors (ignition frequencies, severity factors, partitioning factors, weighting factors, etc.) that were used to estimate fire ignition frequencies and include in this discussion consideration of the concerns raised in RAI number 2 above. Describe the plant equipment and systems that are located in this room. Describe the fire source/target sets considered in the detailed quantification scenarios and the basis for their selection. Include a discussion of the CCDP for each scenario analyzed.

**FIRE RAI 5 RESPONSE:**

Assessment of discrete fire scenarios rather than conservatively failing all equipment in the RB3 area is the principle reason that the CCDP drops. Factors that contributed to the reduction in the initial screening CDF for RB3, based on the detailed evaluation, are: (1) elimination of some fire sources (panels/cabinets) that were determined not to propagate fire, (2) fire damage modeling that shows that not all targets are damaged by a fire, and (3) application of the probability of a hot short for loss of service water cooling to turbine building equipment (rather than assuming guaranteed failure) for all but one of the RB3 fire scenarios.

The fire initiating event frequency (1.7E-03/yr) used in the initial quantitative screening analysis of compartment RB3 was based on a conservative count of "fire ignition sources", i.e., electrical cabinets, panels, and equipment. In addition to the conservative estimate of fire sources, all equipment and associated cables within RB3 were assumed failed in the initial screening analysis irrespective of their location relative to these fire ignition sources. These assumptions ensured a conservative application of the screening method (note that RB3 is a relatively large compartment with a high ceiling and considerable distance exists between many of the fire ignition sources and equipment/cables).

AS presented in Section 4.7.21 of the VY IPEEE Submittal, RB3 has automatic fire detection with no automatic suppression. This area has a relatively low combustible loading and is a relatively large area. A summary of the equipment cables located in various cable trays or conduits in RB3 includes Division A of the ECCS initiation logic, Division A of LPCI/RHR-torus cooling and Division A Core Spray and SRVs 71C/D. Control cables to LPCI UPS-1B are located in this area and subject to open-circuit or short-circuit damage. HPCI, RCIC and CRD pump-A, RBCCW pump A/B cables are also located in various portions of RB3 as are control/power cables for service water supply valves to the Turbine Building Closed Cooling Water System (TBCCW).

The target-set evaluations and the quantification of the RB3 scenarios (including assumed failed equipment) are summarized in Sections 4.9.6 and 4.10.9 of the IPEEE Report. The CCDPs for these initiators are shown in Table Fire-2-3 for Fire RAI No. 2.

The detailed evaluation of the RB3 compartment included plant walkdowns and internal inspections of specific electrical cabinets/panels. The purpose of the walkdowns was to identify characteristics of discrete fire scenarios. The purpose of the cabinet inspections was to collect cabinet attributes for determining which cabinets do not propagate a fire. Guidance from EPRI TR-105928, Fire PRA Implementation Guide, was used to determine the cabinet attributes. EPRI TR-105928 states that electrical cabinets that are not vented do not propagate fire.

Vents are described as louvers on the front, back and/or sides; grilles on the front, back, sides and/or top; open top; open top with shields; fans (typical on solid state equipment). Cabinet penetrations are defined as air drop with flange and water seal, air drop with open conduit, air drop with rated fire seal, sealed conduit. EPRI TR-105928 does not consider these types of penetrations to be vents and, in the absence of other ventilation, these penetrations will not allow sufficient air exchange to replace oxygen being consumed by the fire, and an incipient fire will self-extinguish when there is no longer enough oxygen to support combustion. Therefore, non-vented panels and cabinets of low voltage (<480V) were judged not to propagate fire (non-fire hazard) and were eliminated from the detailed target-set evaluations.

Fire significant panels that were evaluated in detail include 480V MCC-8E, 480V MCC-9D, 480V MCC-89A and CRD Accumulator Indicator Panel. The 480V MCCs are located in proximity to each other and, therefore, were conservatively grouped together as a single fire source with impact on nearby cables (Initiator FRB3MC in the VY IPEEE Submittal). Other fire events evaluated in RB3 include a self-ignited cable tray (Initiator FRB3CL), transient trash fire (Initiator FRB3TR) and transient oil fire (Initiator FRB3OL). Due to its location, the CRD Accumulator Panel was evaluated with the self-ignited cable tray fire (FRB3CL). Tables 4.10.1 and 4.14.1 of the IPEEE Report summarize the target-set scenarios, initiating event frequencies and CDF results for these initiators. As shown in Table 4.10.1, the probability of a hot short causing inadvertent isolation of service water cooling to non-essential turbine building equipment was applied for initiators FRB3CL, FRB3TR and FRB3OL. This hot short probability was conservatively assumed to be 1.0 for initiator FRB3MC, thus guaranteeing isolation of service water non-essential equipment. This was done for FRB3MC because the subject MCCs house the power supply breakers for the SW non-essential load motor operated valves.

**SEISMIC RAI 1:**

With the exception of flood due to actuation or failure of the fire suppression systems, seismic induced internal floods are not discussed in the Submittal. This is not consistent with NUREG-1407. It is stated in NUREG-1407 that "The scope of the evaluation of seismically induced floods, in addition to that of the external sources of water (e.g. tanks, upstream dams), should include the evaluation of some internal flooding consistent with discussion in Appendix I of EPRI NP-6041." According to EPRI NP-6041, "the effects of possible ruptured vessels or piping systems that could flood or cascade into essential equipment should be considered." The consideration is not limited to the flooding due to actuation and failure of the fire suppression systems. Please describe how seismically induced internal flood sources were identified and evaluated in the IPEEE.

**SEISMIC RAI 1 RESPONSE:**

The IPEEE internal flooding evaluation addressed both seismic and non-seismic internal flooding concerns. The evaluation included an assessment of possible spray, flood, and cascading of flood water on essential equipment due to postulated failure of seismic and non-seismic pipes, e.g., Service Water, Fire Water, Circulating Water piping, including piping connected to seismic and non-seismic tanks. Section 5.4.2 of the VY IPEEE Submittal lists the plant walkdown and inspection attributes used to evaluate all potential seismic and non-seismic flood sources. These attributes meet the considerations identified in EPRI NP-6041, Appendix F, "Spray and Flooding", for considering the effects of possible ruptured vessels or piping systems that could spray, flood or cascade onto essential equipment. (VY assumes that the NRC is referring to EPRI NP-6041 Appendix F and not Appendix I in the above RAI question.) Therefore, no quantification or additional flood analysis is required for the seismic assessment.

**SEISMIC RAI 2:**

Please provide the results of the evaluation of the seismic improvement opportunities described in Section 7.2.2 of the Submittal. Please provide a schedule for those items selected for implementation.

**SEISMIC RAI 2 RESPONSE:**

1. A-46/IPEEE Outliers: (Complete, Not Credited in CDF) VY IPEEE Submittal Section 3.4 identified that A-46/IPEEE outliers would be resolved in accordance with A-46 criteria and also be demonstrated that they have a HCLPF > .3g. Table 3.4.2 of the submittal provided a listing of outliers to be resolved. Included in that table is the reason that each component had been designated as an outlier (e.g., spatial interaction, anchorage outlier, equipment type not represented in the SQUG GIP equipment classes, equipment features not in compliance with required caveats, etc.). To date, all outliers have been evaluated and addressed within the context of the reason that they were originally designated an outlier. All planned actions requiring physical modifications to plant structures, systems and components (SSCs) have been implemented. All remaining outliers have been evaluated for resolution and received an independent review of that evaluation. Based on the above, all A-46/IPEEE outliers are complete.

2. Seismic/PRA for CST: (Complete, Not Credited in CDF) The limiting HCLPF value in the seismic margins analysis for the plant is the Condensate Storage Tank (CST) at a value of .25g. The analysis has been reviewed and VY concludes that the tank shell stresses at the juncture with the chair rail support anchorage is the limiting feature in defining the HCLPF analysis results. Additional scoping investigation has been performed to define any potential modifications that can be implemented to raise the calculated HCLPF value. Based on this investigation, no simple cost effective enhancements have been identified that will significantly improve the as-defined HCLPF value of .25g. As this value of .25g is significantly above the site design value of .14g, VY concludes that no structural modifications to this tank are required.

3. Diesel Fire Pump Fuel Tank: (Completion scheduled for prior to startup from RFO-22, Not Credited in CDF). One of the two concerns identified with this tank was the potential for unrestrained motion of the tank at the tank shell-to-saddle support interface to cause crimping of the fuel line to the pumps. To mitigate this concern, modification to locally reroute the fuel oil supply line tubing will be implemented. This modification will alleviate the potential for tube crimping and thereby eliminate the concern of relative motion between the tank shell and its' saddle support. The second concern related to this tank dealt with a masonry wall comprising part of the tank enclosure. Upon further review, VY concluded that the subject masonry wall HCLPF is enveloped by the analytical sampling performed and documented in Section 3.2.4 of the VY IPEEE Submittal. Therefore, with the exception of the tubing reroute discussed above, no further consideration for structural modifications to the fuel oil tank or masonry enclosure wall are required.

4. Bus 1/2 Anchorage: (Complete, Not Credited in CDF) These buses are classified non-nuclear safety (NNS) and are not electrically relied on under scenarios that include a loss of normal power (LNP) or seismic event. The findings from the walkdown of this equipment have been further evaluated and are summarized as follows: (1) As noted in Reference 4-29 of the VY IPEEE submittal report (see Section 4.12.1.1(3)), the successful performance of unanchored switchgear has been documented at earthquake levels in excess of the VY RLE; this conclusion of successful performance inherently includes consideration of the equipment being a fire ignition source, (2) these buses are normally energized through the switchyard from offsite sources and this path is deemed to be a more seismically vulnerable path than the buses themselves, (3) these buses are located in rooms that are protected by fire suppression systems that were evaluated and it was concluded that there were no seismic vulnerabilities attributed with the suppression systems. Therefore, further consideration for modifications to Bus 1 and Bus 2 is not necessary.

5. Fire System Standpipe in RB: (Completion scheduled for prior to startup from RFO-22, Not Credited in CDF). The proposed improvement is to enhance the support of the fire system northwest standpipe in the Reactor Building. This initiative is being considered as a beneficial improvement. The current plan is to complete design development and implementation during the current operating cycle.

6. $H_2$ Piping: (Complete, Not Credited in CDF) The subject piping and associated equipment is located in the Turbine Building which is an NNS, non-seismic structure. Based on further review, it is concluded that significant piping system reroute/re-support and interconnected equipment anchorage modifications would be required in order to provide any substantive reduction in vulnerability from the effects of a RLE. Therefore, further modifications to this system will not be pursued.

7. Control Room Ventilation: (Complete, Not Credited in CDF) The proposed improvement was to perform a detailed assessment of the need for Control Room ventilation and its capability. The detailed review has concluded that control room isolation is not required following a seismic event and that current operating procedures, training and indications are adequate for operators to initiate alternate means of control room ventilation if needed during an A-46/SQUG or IPEEE Seismic Margins scenario. Therefore, the Control Room ventilation system has been removed from the scope of A-46/SQUG and the IPEEE Seismic Margins program.

**INTERNAL FLOODING RAI 1:**

The Time Reliability Correlation method from NUREG/CR-1278 was used for estimating human error probabilities (HEPs) for internal flooding events. The Submittal states that "for the relatively routine operator actions needed for the isolation of a pipe," failure to diagnose a flooding event was judged to be small, and optimum stress levels were judged to be reasonable; therefore, nominal HEP values were used. The submittal also states that because "the typical action(s) necessary to mitigate a flooding event are not complex," the HEP values associated with action execution ("manipulative error") were judged to be small (on the order of 1E-4/yr) and hence adequately accounted in the HEP values related to diagnostic error; as a result only HEP's for failing to diagnose an event were derived.

However, the submittal neither provided a clear description of the human actions involved in the accident sequences of each flooding initiating event nor demonstrated how these assumptions were applied in the quantification of each particular action under different accident conditions. The use of these assumptions is of particular interest for the initiator "RBTRF2: un-isolable service water return line break in the reactor building" the mitigation of which depends on several human actions, some of which do not appear to be "routine operator actions". Please address the following:

(a) Provide a detailed description of the accident sequences (and pertinent failure probabilities) associated with the initiator RBTRF2, depicting how this event leads to core damage.

(b) For each human action credited in RBTRF2, please describe how associated HEP's were derived on the basis of these assumptions. That is, for each particular action <u>and</u> each particular accident sequence in which the action is modeled, explain: (i) why the probability to fail to recognize the need for the action is small, (ii) why stress would be optimum, (iii) how much time is available (from the moment it is recognized the action is needed), (iv) how much time is needed to accomplish the action, (v) are there procedures available, (vi) would it be "routine action" under the specific action conditions.

(c) Describe how the times needed to perform an action were estimated. For example, were they estimated by walkthroughs?

(d) Describe how the dependencies among human actions in this particular initiator were treated.

**INTERNAL FLOODING RAI 1 RESPONSE:**

The HEP evaluation for the baseline RBTRF2 initiator is based on the Vermont Yankee panel alarm response sheets, off-normal procedures and EOPs, which provide guidance for detecting and mitigating flooding events. As stated in Section 7.2.3 of the VY IPEEE submittal, the RBTRF2 initiator was identified as an improvement opportunity for evaluation of procedural enhancements, hardware changes and Service Water restoration actions to improve mitigation of, and recovery from, this flooding event. Operator training and procedural enhancements are currently under evaluation as stated in VY's response to Internal Flooding RAI 2(a). When implemented, these training and procedural enhancements will have a positive influence on the baseline RBTRF2 HEP and CDF results. These proposed enhancements have not (as yet) been reflected in the RBTRF2 baseline model or the HEP discussions provided below.

<u>Summary of RBTRF2 Baseline Model/Sequences:</u>

Initiator RBTRF2 evaluates a postulated major pipe break in the Service Water system discharge header located in the Reactor Building. The discharge header at this location is common to SW headers A and B in the Reactor Building. SW discharge flow from the Turbine Building also connects to this discharge pipe located in the Torus room (refer to Figure Flood-2-1).

As stated in the VY IPEEE submittal, SW return piping is located on El. 303', 280', 252' and 232' of the reactor building. The maximum break flow occurs if the break is postulated to occur at a low elevation (Torus room). Flow rates from higher elevation breaks will be reduced, and the water discharging at the higher elevations will accumulate in the Torus Room, El. 213' (RB basement) via floor openings, grating, stairwell, etc. Permanent berms around the ECCS corner room stairwells and floor openings at El. 252' prevent this water from propagating to the ECCS corner rooms at El. 252'. The height of these berms was recently increased as a result of the internal flooding evaluation. Accordingly, this berm modification has benefited this RBTRF2 initiator by guiding postulated flood water away from the corner room stairwells. Because a major pipe break is postulated, SW flow must be throttled back or eventually shutoff to limit the extent of Reactor Building flooding and to allow for implementation of recovery actions to restore the SW system. However, the baseline RBTRF2 model conservatively assumes no credit for any recovery of service water. The RBTRF2 initiating event frequency is estimated at 1.6E-04/yr with a baseline CDF of approximately 5.9E-06/yr. The sequence of events as related to the mitigating human actions is provided below. Again, as stated above, enhancements are currently under evaluation, which will have a positive influence on these sequences as modeled in the initial internal flooding evaluation (VY IPEEE Submittal).

1. SW pipe break occurs at a specific location in Reactor Building. This break location also affects a portion of the piping boundary used for alignment of SW Alternate Cooling Mode. Thus, both SW and Alternate Cooling are assumed to be failed and non-recoverable. The maximum flow rate out the break is estimated to be 10,000 gpm. At this flow rate, torus room (EL. 213') sump high level alarms (there are 4 sumps in the torus room) will initiate in the control room almost immediately. The control room alarm response is to dispatch an operator to observe the local alarm conditions.

2. The flooding event tree model assumes failure of systems/equipment that depend directly on SW. This includes EDGs A and B, Main Condenser, and RHR A and B Heat Exchangers (containment heat removal). Other systems that are available short term but assumed unavailable for long term due to loss of cooling include Feedwater/Condensate and CRD pumps (RPV injection). These components receive cooling from TBCCW/RBCCW, which provide short-term heat removal without SW.

3. HPCI and RCIC are assumed to fail as a result of flood water propagation via communication between the torus room and HPCI/RCIC rooms at El. 213'. At the maximum flow rate, HPCI and RCIC pumps will begin to flood in approximately 10 minutes. Initiation of flood mitigative actions is assumed to take greater than 10 minutes for the RBTRF2 event, therefore, the PRA model assumes HPCI and RCIC are disabled (set to guaranteed failure).

4. The ECCS corner rooms have watertight doors at El. 213' protecting RHR and Core Spray equipment from torus room flooding. These doors are alarmed to ensure that they are not inadvertently left opened. The diagonal corner room walls are flood designed up to El. 229'. Therefore, flood water does not begin to propagate to the ECCS corner rooms until the adjacent torus room water elevation exceeds El. 229'. At the maximum break flow of 10,000 gpm, the design flood height of the corner rooms would be exceeded in approximately 2 hours. Thus, the model assumes that 2 hours are available to identify the event and take actions to reduce/limit break flow before water begins to propagate to the ECCS corner rooms. Not included in the 2 hour time frame is credit for any reduction in break flow rate caused by closure of MOV-19A/B or MOV-20 (SW flow to Turbine Building). These valves can be closed remotely from the control room.

5. RPV inventory makeup is initially provided via intermittent operation of a motor driven feedwater pump, CRD pumps, Condensate pump, or via the low pressure ECCS (Core Spray/LPCI) pumps.

6. The "initial action" (IA) models termination of the flood event prior to the water level exceeding the design flood height of the ECCS corner rooms. The operator actions include local observation/confirmation of the alarmed condition, break location, communication of observations to the control room. Depending on the specific break location and severity, actions include throttle/stop required service water pumps and open/close SW header manual valves as necessary to control service water. Although the equipment in the corner rooms is not credited for long term operation in this RBTRF2

scenario, preventing flooding of the corner rooms would allow use of this equipment at a later time when service water or alternate cooling is recovered. Given the alarm conditions and local observation/confirmation of this initiator and the available 2-hour time window (which is based on a conservative continuous flow rate), the initial action HEP is estimated at 5.0E-4.

7. If the initial action (IA) is successful, (i.e., flood terminated before torus room water level exceeds El. 229'), we assume that the ECCS corner room equipment, LPCI/RHR pumps and Core Spray pumps, are not flooded. However, without torus cooling, (torus cooling assumed failed because SW and Alternate Cooling are assumed failed due to the SW break), the model assumes that LPCI and CS cannot perform a long term injection function due to eventual heatup of the suppression pool. Thus, Condensate Transfer (CT) and hard-piped torus vent are credited for RPV inventory makeup and decay heat removal.

8. If the initial operator action fails, water is assumed to begin to propagate to the ECCS corner rooms. The "ultimate action" (UA) is modeled as a recovery action to the initial action (IA) to terminate the flooding event before the water level exceeds the elevation of the torus catwalk, El. 240'. In the current model, it is assumed that water level above this elevation may impact the ability to align condensate transfer (CT) via the LPCI/RHR system piping for long term RPV inventory makeup (if not done earlier). It should be noted that CT can also be aligned to the Core Spray system piping using valves that are not impacted by the flood. This alignment is also proceduralized in existing plant procedures and will be considered along with other training and procedural enhancements being evaluated for Reactor Building flooding events. Operators have 40 to 50 minutes to prevent the water level from exceeding the catwalk once flooding begins to exceed the flood design height of the ECCS corner rooms. The operator actions are similar to that described for the initial action (IA). Given the available time window, the ultimate action HEP is estimated at 5.0E-2.

9. If the ultimate action (UA) is success, flooding of the torus catwalk is prevented, and Condensate Transfer system manual valves remain available for alignment of long term RPV inventory makeup. RPV inventory makeup (via Condensate Transfer System) and decay heat removal (via the hard-piped Torus Vent system) are evaluated probabilistically in the event tree. Random failure of either system is assumed to cause core damage.

10. If the ultimate action (UA) fails, we assume that the torus catwalk is flooded and prevents the use of Condensate Transfer (conservatively assumes that CT manual valves were not aligned earlier or elsewhere via Core Spray) as the RPV inventory makeup source. Hence, all makeup is assumed failed and core damage occurs.

The initial action (IA) and ultimate action (UA) are similar in that they model cognitive and manipulative HEP elements. In the case of initiator RBTRF2, there is approximately 2 hours for operators to diagnose the flood event and take initial action to control flooding before level exceeds the corner room design flood height. The ultimate action is modeled as a recovery action to the failed initial action, given the additional time (40 to 50 minutes) before other equipment may be affected. These operator action times do not credit any reduction in SW flow, which can be quickly realized by reducing the number of operating pumps or by throttling flow. Current EOPs and off-normal procedures already direct operators to perform these actions. Training and enhanced procedures are being evaluated to further improve the existing guidance.

<u>Characterization of the HEP:</u>

The characterization of the HEP models mainly considers that there is substantial time available to the crew to take the initial actions.

1. Identification can be performed within several minutes based on control room indication coupled with local observation of the break.

   (a) Several sump high level alarms will quickly annunciate in the control room. This will result in an operator being dispatched to the alarm location. The walk from the control room to the torus area is a routinely traveled route, and will take only a few minutes. It is also likely for the Reactor Building Auxiliary Operator (AO) to be in the vicinity and quickly observe the event conditions. Any significant accumulation of water would be obvious to the observer, and misdiagnosis is judged to be highly unlikely. Also, low service water pressure would likely result in an alarm condition and cause isolation of Turbine Building SW flow. This would point to the service water system as the likely source of the flood.

(b) For any flooding event, particularly for the RBTRF2 event, multiple personnel, i.e., the entire operating crew (including shift engineer and auxiliary operators) will be involved in the diagnosis and decision making.

2. Diagnosis of the actions needed to be taken is clear based on existing EOPs, off-normal procedures and panel alarm response sheets. Isolation of the break can be performed over a 2-hour period and the time required to perform the actions is short compared to the 2 hours. As stated above, enhancements are currently under evaluation, refer to Flooding RAI 2(a).

3. The manual actions to respond to the SW line break include stopping unneeded SW pumps and re-aligning SW valves (based on specific break location and pre-event seasonal alignment of SW). These actions are considered easily performed. The HEP associated with optimum stress is considered appropriate because the time available for action is large, the tasks are not complicated, and performance of the tasks can be accomplished well within the calculated action time. Given the amount of time available, a detailed time study was not performed during the HEP quantification process.

The operator action failure cutset is not currently the controlling cutset for quantification. Regardless of the assessed HEP or resulting CDF, VY is pursuing training and procedural enhancements. Requantification would not change this conclusion.

**INTERNAL FLOODING RAI 2(a):**

The CDF from internal flooding of 9E-06/yr takes credit for several improvements most of which are stated in the submittal as being "under evaluation". (This CDF is 200% higher than the total CDF of 4.3E-06/yr from all internal events calculated in the IPE. If the internal flooding CDF is included, the total internal event CDF becomes 1.3E-05/yr, and the flooding contributes about 70% of the total internal event CDF.) Furthermore, the actual internal flooding CDF may be even higher if improvements credited will not be implemented. The submittal does not provide a CDF estimate without crediting the improvements which have not been implemented. Please provide: (a) a schedule for those items selected for implementation, and (b) the CDF estimate without crediting any of the improvements (described in Section 7.2.3 of the submittal) that are not planned to be implemented.

**INTERNAL FLOODING RAI 2(a) RESPONSE:**

The status of all VY IPEEE internal flooding improvements is listed below. All items that are credited in the CDF are either complete or a schedule is provided for implementation/completion.

<u>Internal Flood Proposed Improvements Listed in VY IPEEE Submittal Section 7.2.3:</u>

1. <u>RB252 Equipment Locker</u>: (Complete, Credited in CDF). The improvement modification raised the equipment storage locker at the east end of the CRD stairway to minimize flow blockage to the CRD stairwell to improve water removal to the Torus Room.

2. <u>RB252 Floor Sleeves</u>: (Complete, Credited in CDF). The improvement modification lowered the sleeve height at El. 252' (30" and 24" diameter sleeves) to improve water removal to the Torus Room.

3. <u>ECCS Corner Room Equipment Hatches</u>: (Complete, Credited in CDF). The improvement modification sealed the hatch lift points and hatch edges to ensure that the hatches are water tight.

4. <u>ECCS Corner Room Flood Berms</u>: (Complete, Credited in CDF). The improvement modification increased the berm height to enhance the flood protection of the ECCS Corner Room stairwells and pipe/electrical chases, which penetrate the ceilings of the ECCS Corner Rooms (El. 252').

5. <u>El 303 Floor Chase Berms</u>: (Complete, Credited in CDF). The proposed improvement was to either increase the berm height at the existing floor chases along the north wall (or seal floor chase opening or the panel) or otherwise ensure that Panel CP82-2 (located below on El. 280') is not adversely affected. Further evaluation has shown that Panel CP-82-2 is sufficiently offset from the cable chase at El. 303' to preclude significant spray onto the panel. Additionally, the conduits entering the top of the panel are threaded and attached with lock nuts. This will disperse water away from the conduits and prevent any significant water entry. The other ends of the conduits are not located near the El. 303' cable chase and there is little potential for water to enter the conduits.

6. <u>Upper RCIC Water Relief</u>: (Complete, Credited in CDF). The proposed improvement is to provide a relief path at El. 232' so water accumulation in upper RCIC (due to random fire pipe failure) will relieve to the lower RCIC area before floor failure occurs. An analysis has been performed to show that existing flood relief will occur prior to floor collapse.

7. <u>RB Unisolable SW Break</u>: (Evaluation in Progress, Completion scheduled for September 1, 2000, Not Credited in CDF). Training and procedural enhancements are being evaluated to improve the mitigation response of this event. Flooding RAI 2(b) provides a description of the concept for the improved strategy.

8. <u>FOB/Switchgear Room Doors</u>: (Complete, Credited in CDF). The improvement modification to reduce the flood interaction between the front office building and the switchgear room included installation of weather stripping

to reduce the gap at the bottom of the single door to the west switchgear room and the double door between the west switchgear room and the turbine building.

9. <u>FOB to Switchgear Room Vestibule Door</u>: (Complete, Credited in CDF). The improvement modification removed the door latch to ensure that the door will open to relieve water accumulating at the switchgear room entrance door.

10. <u>FOB to Turbine Building Door</u>: (Complete, Credited in CDF). The improvement modification consisted of replacing the front office to turbine building double door with a saloon type door, which will relieve any water accumulating in the front office building.

11. <u>FOB Flooding Procedures</u>: (Complete, Credited in CDF). The intent of this improvement was to provide additional mitigative guidance until previously discussed modifications were completed. With FOB modifications 8, 9 and 10 installed, existing procedural guidance is adequate.

12. <u>Diesel Generator Room Independence</u>: (Complete, Not Credited in CDF). The proposed improvement was to evaluate procedural enhancements and hardware changes for mitigating the effects of a SW line break in a Diesel Generator Room. The total CDF for both diesel room SW flooding events is low (approximately 1E-07/yr) with no credit given for operator action to mitigate the event (i.e., operator opens the diesel room doors). Based on the low CDF, VY judges that hardware or procedural changes are not warranted.

13. <u>Torus Integrity</u>: (Complete, Credited in CDF) The proposed improvement is to evaluate the potential for containment failure during a major flood in the Reactor Building basement (Torus Room). Failure to isolate a major break or failure to terminate SW flow would eventually result in filling the torus compartment unless the water removal rate by sump pumps (limited capacity) exceeds the break flow rate. VY has performed a review of the torus capacity relative to uplift and external loading during a postulated extreme flooding condition.

    The capability of the torus structure to remain in place under flooding scenarios, which may cause an uplift load on the torus shell was reviewed. The torus structure incorporates a significant vertical tie-down restraint system that was implemented to satisfy the requirements of the reanalysis performed under the BWR Mark I Long Term Program. Based on review of the design loads imposed on the torus, including combined earthquake and hydrodynamic blow-down loads, VY concludes that these design loads envelope the postulated static buoyant loads resulting from a flooding scenario. Therefore, the torus structure will remain affixed to the Reactor Building reinforced concrete base mat as a result of the flooding scenario.

    The external loading on the torus shell has also been reviewed. Vermont Yankee estimates that buckling of the shell may begin to occur if torus room water level accumulates to greater than the height of the torus catwalk. The torus catwalk is located at El. 239′-7″, approximately 26 feet above the torus room floor, El. 213′-9″. The top of the torus is approximately at El. 244′. Level above the catwalk would require approximately 1.7 million gallons of water to accumulate in the torus room. The time for level to exceed the catwalk elevation depends on the specific break scenario. For initiator RBTRF2 (un-isolable guillotine break in the SW discharge line), the time for level to exceed the catwalk is estimated at approximately 3 hours (assuming no flow reduction which should occur with isolation of SW to non-essential Turbine Building loads). The likelihood of a significant flooding scenario of this magnitude is considered to be very low since the flooding source can be terminated by simply stopping the SW pumps. The likelihood of an external loading scenario will be further minimized by enhanced mitigation procedures currently being assessed for the RBTRF2 un-isolable SW initiator, refer to Flooding RAI 2(b). Irrespective of these mitigation enhancements, postulated failure of the torus shell due to flooding external loads is addressed in Flooding RAI No. 3, Containment Performance.

14. <u>Alternate Cooling Alignment</u>: (Complete, Credited in CDF) The proposed improvement is to evaluate procedural and hardware enhancements for aligning Alternate Cooling Mode during a flood event in the Reactor Building basement (Torus Room). The base case model (VY IPEEE Submittal) credits alignment of alternate cooling if random component failures (pumps) occur in the SW cooling water system after successful break isolation and restoration of SW cooling. However, alternate cooling is not credited in any SW flooding initiator where break isolation fails because failure to isolate the break is assumed to cause guaranteed failure of alternate cooling mode. Alternate cooling is also not credited in fire system flooding scenarios where break isolation fails. Access to the torus room floor El. 213' is needed to align alternate cooling. Current procedures for aligning alternate cooling are judged to be adequate and there are no simple cost effective hardware solutions to allow remote alignment. However, Operations Department personnel have estimated that, if needed, alignment of alternate cooling could be performed with up to approximately 2-1/2 to 3 feet of water on the torus room floor. Because there is a potential to exceed this water level during a major SW flooding event, (even with successful break isolation), a CDF sensitivity study was performed to assess the change in CDF when conservatively assuming no credit for alternate cooling for all major SW flooding scenarios. Based on this sensitivity study, the total flooding CDF increased from 9.03E-06/yr (baseline VY IPEEE Submittal) to approximately 9.37E-06/yr. This is less than a 4% increase in the baseline CDF. This increase in CDF is judged to be small and well within the uncertainty of the flooding evaluation. Therefore, significant water level on the torus room floor during postulated SW break scenarios leading to the inability to align alternate cooling, is judged not to be a significant contributor to plant risk. Based on this result, VY judges that alternate cooling procedural and hardware changes are not warranted.

**INTERNAL FLOODING RAI 2(b):**

The internal flooding associated with "RBTRF2: un-isolable service water return line break in reactor building," yielded a CDF of about 6E-06, which is about 70% of the total internal flooding CDF. Section 7.2.3 of the submittal states that it has been proposed to evaluate hardware and procedural modifications (not credited); however, it is not explained what exactly these improvements are. Since this is the most risk significant initiator, it is reasonable to expect a more focused emphasis in terms of improvements in this particular area. Please explain if any specific improvements have been identified and have been, or are planned to be, implemented.

**INTERNAL FLOODING RAI 2(b) RESPONSE:**

The PRA identified the conceptual ideas for improving the Reactor Building flood mitigation capability. Several of the modifications described in the response to Flooding RAI 2(a), which were credited in the CDF and are now complete, extend the availability of equipment which could be used initially to respond to an un-isolable SW leak. Vermont Yankee engineering has performed preliminary evaluations on various solution concepts ranging from a multi-million dollar backfit of an independent discharge header, to a package consisting of training, enhanced procedural guidance and further PRA analysis. Hardware modifications are not being proposed at this time because further review has identified: (1) a potential success path(s) for SW break isolation and restoration of the SW function to either Reactor Building or Turbine Building equipment, and (2) additional proceduralized success path(s) for RPV injection that will provide additional defense in depth for the reactor inventory control function. VY is currently evaluating these potential success paths, operator training and enhanced procedures. Based on the preliminary evaluation, VY believes that these enhancements will significantly reduce the CDF from this flooding initiator (RBTRF2) and other Reactor Building flooding scenarios without physical hardware modifications. Additional training and procedural enhancements are being considered based on the following concepts: (a) limiting or reducing the flood rate from a SW pipe break with specific flood termination criteria, and (b) initiating actions for restoration of SW system function. The feasibility of these improvements is being assessed by VY Engineering, Operations and PRA engineers. A summary of conceptual improvements to the RBTRF2 initiator and potential PRA impact is given below.

<u>Break Isolation and SW Restoration:</u> The current PRA model of initiator RBTRF2 assumes that a break located in the Reactor Building SW discharge line is guaranteed to fail the SW system heat removal function. The assumption of failing all SW is the reason this event is referred to as un-isolable. Recovery of the SW function is not credited in the current PRA model. Flooding Figure 2-1 provides a simplified piping schematic of the common SW discharge pipe in the Reactor Building. One concept being investigated for training and enhanced procedures includes the possible isolation and restoration of SW depending on the break location as described below. Crediting restoration of the SW function upon isolation of the break will reduce the CDF of the RBTRF2 event. However, detailed investigations may identify other solutions that are feasible from a PRA perspective.

<u>Break located to the right of SW-18 (T):</u> With a break located to the right of SW-18, the break is isolated by closing SW-18 (torus catwalk) and by closing the SW supply valves (SW-MOV-20 or SW-MOV-19A/B) (torus catwalk) to the Turbine Building. SW function to Reactor Building and safeguards equipment can be restored by ensuring that the SW discharge is aligned to the west cooling tower deep-basin (the normal cold season alignment).

<u>Break located to the left of SW-18 (R):</u> With a break located to the left of SW-18, the break is isolated by closing SW-18 (torus catwalk) and by closing the SW supply or outlet valves local to the individual Reactor Building heat loads. SW function to Turbine Building equipment can be restored by ensuring that the SW discharge is aligned to the SW discharge block (the normal warm season alignment).

Based on the above, SW function restoration to either RB or TB loads could be accomplished in only the time necessary to close/align the valves (estimated at approximately 1 to 2 hours. If not already performed, the SW pumps can be shutoff during the valve alignment to terminate the water flow into the building.

VY estimates that crediting SW restoration upon isolation of the break will significantly reduce the CDF of the RBTRF2 event. This is because of the following:

<u>Break Location at (T):</u> Successful restoration of the SW function from a break located at (T) will allow continued use of both divisions of torus cooling via LPCI/RHR heat exchangers A and B and other safeguards loads including the EDGs. With success of torus cooling, LPCI and CS injection can perform long term core inventory control.

<u>Break Location at (R):</u> Successful restoration of the SW function from a break located at (R) will allow long term use of Turbine Building equipment. This includes long term use of the feedwater/condensate pumps for inventory control and the main condenser for core decay heat removal. Use of the main condenser will reduce the challenge to primary containment.

<u>Additional RPV Inventory Success Paths:</u> Review of this scenario to determine improvement strategies has identified several additional success paths for delivering water to the RPV. These success paths are being evaluated from a PRA perspective along with potential training and procedural enhancements to reduce the calculated CDF.

The current PRA model only credits use of the Condensate Transfer System (CTS) for RPV inventory control after RPV depressurization only if isolation of the flooding source is successful. This is conservative because, even with failure to isolate, the CTS could be aligned to the RPV through the Core Spray injection lines via manual valves that are located outside of the torus area (Appendix N of VY EOPs).

Another option that is available but not credited in the PRA, is to supply RPV inventory after depressurization via the RHRSW system cross-tie to RHR through MOV-183 and MOV-184. These MOVs are located at El. 243' in the northeast ECCS corner room and can be remotely operated from the Control Room. This source of water would be available during the RBTRF2 initiator with or without successful break isolation as these valves would not be affected by flooding for many hours. Use of the RHRSW to RHR cross-tie is included in the Vermont Yankee EOPs, Appendix L. The fire water system is also available as an RPV injection source through MOVs 183 and 184 (Appendix M of VY EOPs). Use of the RHRSW to RHR cross-tie using the SW and RHRSW pumps or the fire water pumps is being investigated as part of the overall improvement strategy for this and other flooding initiators.

<u>Characterization of Pipe Failure Probability:</u> The internal operating pressure of the Service Water discharge pipe is relatively low. Low operating pressures may allow a reduction in the calculated pipe failure probability used for this event. Typical operating pressure is between 20 to 30 PSIG with the SW discharge aligned to the west cooling tower deep basin (winter). In summer months, the operating pressure is in the range of 10 PSIG with the system aligned to the discharge block. Potential reduction in the initiating event frequency will be reviewed further as part of the overall enhancement package.

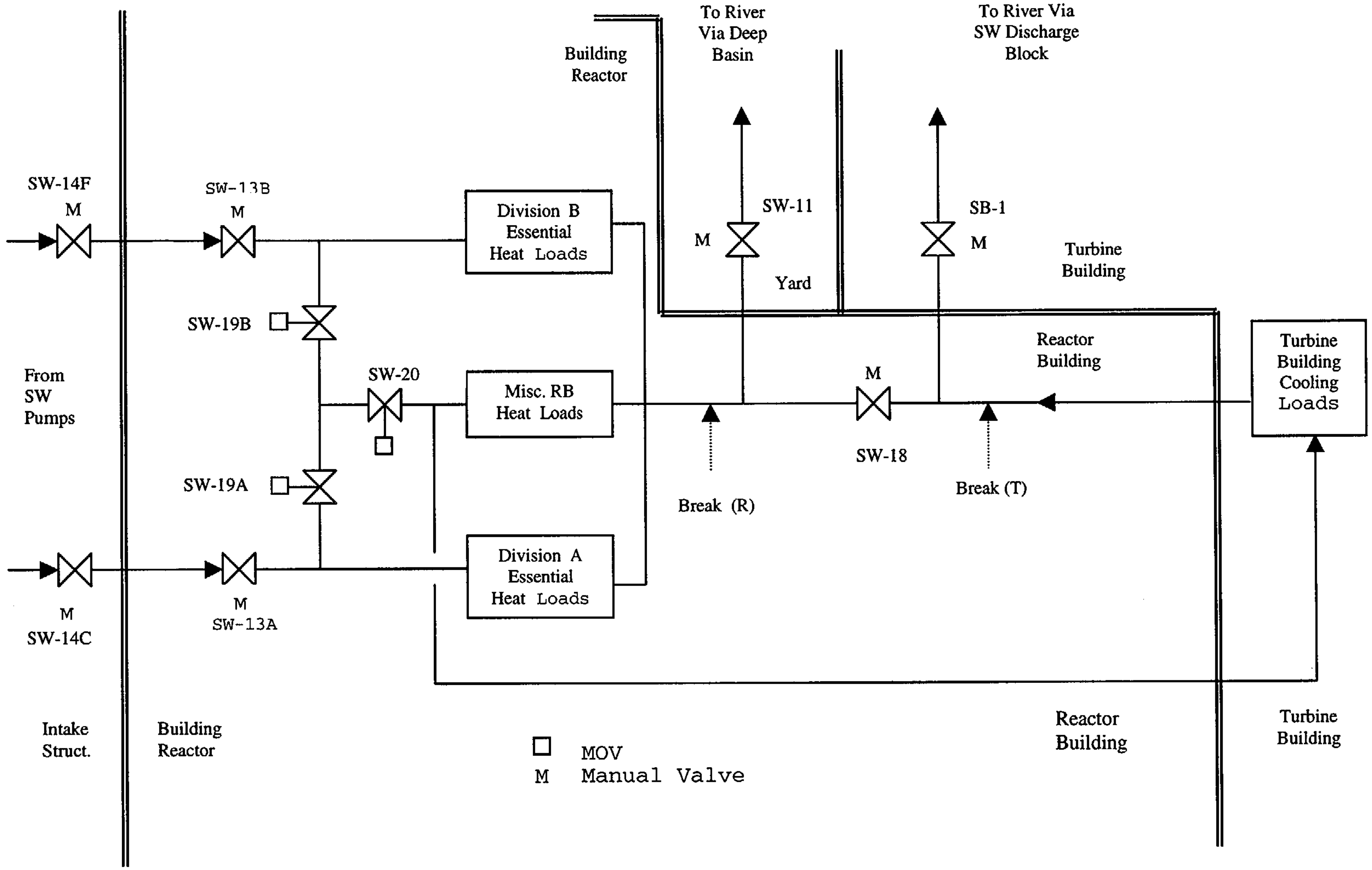



**Figure: Flood - 2 - 1**

**Simplified Flow Diagram - SW Isolation**

**INTERNAL FLOODING RAI 3:**

It is stated in the submittal that "Based on the comprehensive internal flooding evaluation performed, internal flooding events are judged to have an insignificant influence on the reliability of containment performance as analyzed in the IPE".

A value of a large early release frequency (LERF) less than 1E-06/yr was used in the IPE to screen for vulnerabilities. The IPE concluded that no vulnerability with respect to containment performance exists although a LERF of 9.7E-07/yr was estimated, which is very close to the IPE's criterion for vulnerability.

The internal flooding CDF is a factor of 2 (and the total internal event CDF a factor of 3) higher than the IPE's CDF. The submittal does not include a discussion of accident progression and containment performance due to flooding. For example, it is not clear if all sequences associated with RBTRF2 are long-term sequences. Also, it appears that sequences associated with other initiators have the potential for LERF, especially those leading to loss of switchgear or to the loss of the diesels. The low CDF of these initiators can be due to credit taken for improvements "under evaluation". The submittal did not provide an assessment of the containment performance for Vermont Yankee as-built as-operated. Therefore, it is unclear how it can be concluded that internal flooding has an "insignificant influence on the reliability of containment performance as analyzed in the IPE". Please explain how you have reached this conclusion.

**INTERNAL FLOODING RAI 3 RESPONSE:**

The VY internal flooding containment performance discussion in Section 5.4.6 of the VY IPEEE Submittal addresses the identification of containment performance vulnerabilities and insights not identified in the VY IPE, consistent with the following NRC guidance:

- Generic Letter 88-20, Supplement 4:

  *The evaluation of containment performance for external events should be directed toward a systematic examination of whether there are sequences that involve containment failure modes distinctly different from those found in the IPE internal events evaluation or contribute significantly to the likelihood of functional failure of the containment (i.e., loss of containment barrier independent of core melt).*

- NUREG-1407, Section 4.1.5:

  *Perform containment analysis if containment failure modes differ significantly from those found in the IPE internal events evaluation.*

As stated in Section 5.4.6, no such issues were identified for internal flooding.

The information summarized below supports the conclusion that VY internal flooding events have an insignificant influence on the reliability of containment in that internal flooding core damage sequences, (a) progress in the same manner as the core damage sequences assessed in the VY IPE, (b) do not introduce containment performance insights beyond those already identified in the VY IPE, and (c) contribute to large early releases in a similar manner as sequences already assessed in the VY IPE.

<u>Internal Flooding Core Damage Sequences</u>

As discussed in Section 5.4 of the VY IPEEE Submittal, the internal flooding core damage frequency is calculated at 9.0E-6/yr. (Refer to RAI Response 2(b) for a summary of proposed enhancements, which should result in lowering of this CDF.) As can be seen from the breakdown below (Table Flood-3-1), the internal flooding core damage sequences are primarily loss of coolant injection scenarios. These internal flooding post-core damage accident sequence progressions proceed in the same manner as the like classes assessed in the VY Level 2 IPE.

Table Flood-3-1
Flooding CDF by Accident Class

| Accident Class | % of Int. Flooding CDF |
|---|---|
| IA (loss of injection w/RPV at high pressure) | 4.5% |
| ID (loss of injection w/RPV at low pressure) | 92.0% |
| II (loss of containment heat removal) | 3.5% |

Comparison of VY IPE and VY Internal Flooding Containment Performance

The internal flooding sequences analyzed in the VY IPEEE Submittal do not identify containment performance insights beyond those already identified in the VY IPE. Containment performance relative to internal flooding is summarized below.

Primary Containment Capability - Postulated impacts on primary containment capability from internal flooding include uplifting of the torus and external loading during internal flooding scenarios involving significant flooding of the Torus Room. Refer to Flooding RAI 2(a) response for postulated impact of flooding induced loading on the torus.

If torus failure due to external loading during such flooding scenarios is postulated not to occur, the core damage accident progression and containment performance would progress in a like manner as similar accidents analyzed for the internal events IPE. An estimated 96% of the internal flooding CDF is comprised of sequences in which isolation of the flooding source is successful and external loading is judged not to be significant. Further discussion on these accident progressions is provided below as part of the response to this RAI.

If torus failure due to external loading during flooding scenarios is postulated to occur, the timing of containment failure may be comparatively early but the release would not be a large magnitude due to the scrubbed release pathway. With the torus room flooded above the torus catwalk (point at which buckling may begin to occur), torus water inventory would not be lost if torus shell failure were to occur above or below the normal torus water level. Thus, the down-comers and SRV T-quenchers would remain well below the water level and would not become exposed. Torus shell failure during an extreme, unabated Reactor Building flooding event would allow a filtered, scrubbed release through the torus water inventory similar to the hard-piped torus vent. Therefore, postulated failure of the torus during such scenarios would not impact the key containment performance risk measure (i.e., Large Early Release Frequency, LERF).

Primary Containment Isolation - The scope of the containment isolation pathways considered here is the same as that evaluated in the VY Level 2 IPE and identified in Section 4.11.1 of the VY IPEEE Submittal. This scope includes containment penetration paths larger than 2" in diameter. Closed-loop piping systems such as RBCCW and RWCU are not reviewed, as a release via such a pathway is low likelihood, requiring piping/component failures in addition to valve failures/spurious actuation.

Internal flooding-induced impacts on automatic primary containment isolation valves (PCIVs) may be postulated from spray or flooding. The VY Primary Containment Isolation System is equipped with the following design features, which minimize the likelihood of containment isolation failure.

1. The PCIS is designed to fail to a safe mode given loss of electric power. The PCIS sensor and logic circuitry provides both automatic and remote manual isolation capabilities.

2. The control logic for the closure of the PCIVs is designed to assure that once an isolation signal has been initiated, the valves continue to close until full closure is achieved. Once full closure occurs, the valves will not automatically re-open even if the closure signal ceases.

3. The containment isolation valves are located at or above El. 244. It will take several hours for the flood water level to achieve this elevation; thus termination of the flood event before these valves are impacted is very likely.

4. The valves are solenoid-controlled; air-operated valves that are designed to fail closed on loss of air/power. One exception to this design is the torus vacuum relief AOVs, which fail open on loss of power/pneumatic pressure. However, highly reliable backup check valves provide containment isolation of this penetration.

5. Two in-series (redundant) isolation valves protect all penetration paths.

No insights or vulnerabilities are identified from the internal flooding analysis regarding primary containment isolation; it remains a reliable function, as assessed in the VY IPE.

<u>ISLOCA Containment Bypass</u> - The scope of the containment bypass pathways considered here is the same as that evaluated in the ISLOCA analysis of the VY IPE. This scope includes LPCI and Core Spray injection lines, and the RHR SDC suction line. The LPCI and Core Spray injection lines are protected with a normally closed MOV located outside containment and a passive check valve located inside containment. These components are not subject to internal flooding effects. Both the RHR SDC line inboard motor-operated isolation valve (located inside containment) and the outboard motor-operated isolation valve (located at El. 252' in the drywell personnel access hatch enclosure) are located away from any postulated internal flooding effects.

No insights or vulnerabilities are identified from the internal flooding analysis regarding containment bypass scenarios.

<u>Coolant Injection Capability</u> - Loss of coolant injection capability influences containment performance by impacting the likelihood of maintaining the core in-vessel and providing debris cooling (which impacts shell melt-through and containment over-temperature failure). Internal flooding sequences typically lead to flooding-induced failure or degradation of ECCS equipment and isolation of SW (which also leads to degraded ECCS equipment). However, coolant injection using the condensate transfer system, and containment heat removal (via hard-piped Torus vent), are not necessarily failed by flooding effects. Post-core damage scenarios such as these that progress into the Level 2 analysis with limited low pressure coolant makeup and decay heat removal alternatives are typical of those assessed in the VY IPE. Other than the obvious link between internal flooding and flooding-induced SW isolation and/or equipment failure, the internal flooding analysis does not result in the identification of new insights regarding coolant injection capability beyond those already identified in the VY IPE.

<u>Containment Heat Removal Capability</u> - Loss of containment heat removal influences containment performance by subjecting the primary containment to slowly developing high containment pressure and temperature challenges. Internal flooding sequences typically lead to flooding-induced failure or degradation of ECCS equipment and isolation of SW (which also degrades ECCS equipment). However, coolant injection using the condensate transfer system and the containment heat removal function are not necessarily failed by flooding effects. Post-core damage scenarios such as these that progress into the Level 2 analysis with limited low pressure coolant makeup and decay heat removal alternatives are typical of those assessed in the VY IPE. Other than the link between internal flooding and flooding-induced SW isolation and/or equipment failure, the internal flooding analysis does not result in the identification of new insights regarding containment heat removal capability beyond those already identified in the VY IPE.

<u>Failure to Scram</u> - Failure to scram influences containment performance by subjecting the primary containment to fast developing and severe pressure challenges. The scram system is characterized by fail-safe attributes. No insights or vulnerabilities are identified from the internal flooding analysis regarding the reactor scram system; it remains a reliable function, as assessed in the VY IPE.

Large Early Release Frequency (LERF)

Large early releases are identified in the VY IPE Level 2 analysis with the release category High/Early (H/E). The threshold for the High magnitude attribute is >10% CsI released; the threshold for the Early time frame attribute is release beginning <6 hrs. after accident initiation.

The Class II (loss of containment heat removal) internal flooding core damage sequences do not result in large early releases as the releases occur after the Early time frame criterion. The Class II sequences result in late releases (i.e., more than 24 hours elapse between initiating event and radionuclide release).

The Class IA and ID (loss of RPV injection) internal flooding core damage sequences have the potential of resulting in large early releases. As the internal flood sequences progress in a similar manner as the sequences already analyzed in the VY Level 2 IPE, application to the internal flooding sequences of VY Level 2 IPE information regarding the potential for LERF is appropriate. Based on the VY IPE, the conditional probability of large early release for Class IA and ID sequences is 1.09E-1 and 3.61E-1, respectively. Using this information, the large early release frequency associated with the internal flooding sequences is calculated as shown below in Table Flood-3-2.

Table Flood-3-2
Flooding Accident Class Estimated LERF

| Accident Class | Internal Flooding | | |
|---|---|---|---|
| | CDF (1/yr) | Cond. Prob. of LERF | LERF (1/yr) |
| IA (loss of injection w/RPV at high pressure) | 4.05E-7 | 1.09E-1 | 4.41E-8 |
| ID (loss of injection w/RPV at low pressure) | 8.28E-6 | 3.61E-1 | 2.99E-6 |
| II (loss of containment heat removal) | 3.15E-7 | 0.00 | 0.00 |
| TOTAL (Internal Flooding) | | | 3.03E-6 |

Although a detailed Level 2 evaluation was not performed for internal flooding scenarios, the estimated breakdown of internal flooding LERF by containment failure mode is consistent with the breakdown assessed in the VY Level 2 IPE, and is shown in the following Table Flood-3-3. Drywell Overtemperature and Drywell Shell Melt-Through remain the dominant contributing primary containment failure modes to LERF. The LERF contribution from Drywell Overtemperature failure is higher for internal floods than the IPE results because (1) the internal flood sequences are primarily loss of injection scenarios and (2) the IPE results cover the spectrum of accident types (e.g., Loss of Injection, Station Blackout, LOCA, Loss of Containment Heat Removal, ATWS).

Table Flood-3-3
Containment Failure Mode Estimated Contribution to LERF

| Containment Failure Mode | % Contribution to LERF | |
|---|---|---|
| | VY IPE (all Accident Types) | VY IPEEE (Internal Flood) |
| Drywell Overtemperature Failure | 44% | 80% |
| Drywell Shell Melt-Through | 17% | 10% |
| Post-Core Melt Energetic Phenomena | 12% | 5% |
| Wetwell Overpressure Failure | 11% | 3% |
| Drywell Vent | 9% | 1% |
| Containment Bypass | 6% | 0% |
| Other | 1% | 1% |

**HIGH WIND, FLOOD, AND OTHER EXTERNAL EVENTS RAI 1:**

Regarding gas pipeline rupture accidents, the IPEEE Submittal did not state whether there are any gas pipelines near the plant site or provide any information relating to gas pipeline rupture accidents. Please provide an assessment of gas pipeline rupture accidents at Vermont Yankee.

**HIGH WIND, FLOOD, AND OTHER EXTERNAL EVENTS RAI 1 RESPONSE:**

In response to this RAI, Vermont Yankee has conducted an assessment of gas pipeline rupture accidents to determine if the plant and facilities design meet the intent of the 1975 Standard Review Plan (SRP) criteria [Reference 7]. The assessment followed the review process described in 1975 SRP, Sections 2.2.1 and 2.2.2, "Locations and Routes, Descriptions," which identify potential external hazards from industrial, military, and transportation facilities and routes. If a hazard was identified, it would then be evaluated according to 1975 SRP Section 2.2.3, "Evaluation of Potential Accidents," to determine if there are any design basis events for the plant. Based on this VY assessment, evaluation criteria in 1975 SRP, Sections 2.2.1 and 2.2.2 are met, and there are no plant vulnerabilities identified from a postulated gas pipeline rupture accident. A summary of this assessment is provided below.

<u>Gas Pipeline Distribution</u>

Gas transmission and distribution systems in Vermont, New Hampshire, and Massachusetts were examined to ensure that the closest approach to the Vermont Yankee plant was determined. New England currently has over 33,000 miles of natural gas pipeline. This includes 1,785 miles of main transmission pipeline and 31,332 miles of distribution pipeline [Reference 8]. Within the New England gas pipeline network, there are five transmission systems and twenty-five local distribution companies (LDCs).

The closest transmission system to Vermont Yankee is the Tennessee Gas Pipeline Company, which extends to within roughly 30 miles of the plant [Reference 9]. The nearest in-state distribution system is the Vermont Gas System, Inc., which services the northwestern Vermont region, more than 100 miles from Vermont Yankee. The closest LDC is the Berkshire Gas Company, whose natural gas service area extends to within approximately eight miles of Vermont Yankee [Reference 10].

Several infrastructure enhancements are underway to increase New England's gas capacity and projected demand growth. However, only one project, a LDC expansion, affects the southern Vermont region. The Southern Vermont Gas Company proposes to bring distribution service to the area from Bennington to Rutland in southwestern Vermont. The service would be fueled by a new pipeline interconnection from the Iroquois Gas Transmission System originating near Albany, New York. This project will be approximately 40 miles from Vermont Yankee.

<u>Results</u>

The 1975 SRP, Sections 2.2.1 and 2.2.2 refer to Regulatory Guides 1.78 and 1.91 to determine if a potentially hazardous situation exists with regard to chemical releases, including problems of a pipeline rupture. Both regulatory guides depend on distance as either a screening criterion or a means to calculate safe distances from explosions. Regulatory Guide 1.78, for instance, explains that chemicals stored or situated at distances greater than five miles from a facility need not be considered. This criterion is based on the position that if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the postulated plume to such a degree that there should be sufficient time for the plant to take appropriate action. In addition, the probability of such a plume remaining in a given area long enough to be a hazard is quite small [Reference 11].

Regulatory Guide 1.91 describes a method to determine if explosions are a concern to critical plant structures. The method calculates a "safe distance," which is the point beyond which any in-place explosion that might occur is not likely to have any adverse effect on plant operation. The largest probable quantity of explosive material (about 10,000,000 pounds) illustrated in Regulatory Guide 1.91 has a safe distance of nearly two miles [Reference 12].

Based on the above information, the closest gas transmission pipeline or LDC is about eight miles from Vermont Yankee. This distance is beyond the Regulatory 1.78 screening criterion of five miles. It also is well beyond the Regulatory Guide 1.91 safe distance of two miles for the largest probable quantity of explosive material.

**VERMONT YANKEE IPEEE RAI - REFERENCES**


1. NSAC 178L, "Fire Events Database for U.S. Nuclear Power Plants", June 1992.

2. EPRI TR-105928, "Fire PRA Implementation Guide", December 1995.

3. EPRI TR-100370, "Fire-Induced Vulnerability Evaluation (FIVE)", April 1992.

4. EPRI TR-100443, "Methods of Quantitative Fire Hazard Analysis", May 1992.

5. NUREG/CR-2258, "Fire Risk Analysis for Nuclear Power Plants", September 1981.

6. "PSA Evaluation of Potential for Loss of Remote Shutdown Capabilities", Robert F. Kirchner, Niagara Mohawk Power Corporation, 1995 ANS Conference, Seattle, Washington.

7. USNRC NUREG-75/087, "Standard Review Plan," November 1975.

8. The New England Gas Association, "Update on New England's Natural Gas Market," September 1999.

9. Personal Communication, T. Dunn, Vermont Department of Public Service, March 3, 1999.

10. The New England Gas Association, Needham Heights, MA 02494-2800, New England Natural Gas Service Area Map, September 1999.

11. USNRC, Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," June 1974.

12. USNRC, Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," February 1978.