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August 30, 2006

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293 License No. DPR-35

License Renewal Application Amendment 7

REFERENCE: Entergy letter, License Renewal Application, dated January 25, 2006 (2.06.003)

LETTER NUMBER: 2.06.079

Dear Sir or Madam:

In the referenced letter, Entergy Nuclear Operations, Inc. applied for renewal of the Pilgrim Station operating license. NRC TAC NO. MC9669 was assigned to the application.

This License Renewal Application (LRA) amendment consists of five attachments. Attachment A contains the response to the RAIs on LRA Sections 2.2 (Plant Level Scoping Results) and 2.3 (Scoping and Screening Results: Mechanical) conveyed in NRC letter dated July 31, 2006. Attachment B contains the response to the RAIs on scoping and screening aspects of LRA Sections 2.3.1 (Reactor Coolant System), 2.3.2 (Engineered Safety Features), and 2.3.3 (Auxiliary Systems) conveyed in NRC letter dated July 31, 2006. Attachment C contains the response to the RAIs on time-limited aging analysis aspects of LRA Section 4.2.1 (Reactor Vessel Fluence) and LRA Appendix B (Aging Management Programs and Activities) Sections B.1.3 (Control Rod Drive Return Nozzle) and B.1.26 (Reactor Vessel Surveillance Program). Attachment D contains clarification of the response to RAIs on severe accident mitigation alternatives provided in LRA Amendment 4 dated July 5, 2006. Attachment E contains changes to the LRA (Section 2.3.3.12 Primary Containment Atmosphere Control).

This letter contains no new or revised commitments.

Please contact Mr. Bryan Ford, (508) 830-8403, if you have any questions regarding this subject.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 30, 2006.

Sincerely,

(original signed by S. Bethay)

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NRC Resident Inspector
Pilgrim Nuclear Power Station

ATTACHMENT D to Letter 2.06.079
(14 pages)

Clarification of the Response to RAIs on Severe Accident Mitigation Alternatives
Provided in LRA Amendment 4 dated July 5, 2006

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Background

On January 25, 2006, Entergy Nuclear Operations, Inc. requested the renewal of the operating license for the Pilgrim Nuclear Power Station (PNPS), to extend the terms of their operating license an additional 20 years beyond the current expiration date. Appendix E of the License Renewal Application (LRA) consisted of the Applicant's Environmental Report – Operating License Renewal Stage (i.e., the ER). Attachment E of the ER contained the evaluation of Severe Accident Mitigation Alternatives (SAMAs). By letter dated May 22, 2006, the NRC provided a request for additional information (RAI) concerning the analysis of SAMAs performed in support of the PNPS LRA. By letter dated July 5, 2006 PNPS provided the additional information requested. By email dated August 7, 2006, the NRC provided a request for clarification of Entergy's responses to some of the SAMA RAIs for telecon discussions. The response to this clarification request follows.

NRC Request for Clarification to RAI 1.e

With regard to the independent team of consultants that reviewed the PRA revision, identify the criteria and scope of the review.

Response to Request for Clarification to RAI 1.e

The independent team of consultants was comprised of three prominent outside experts from Scientech:

- Mr. Robert Bertucio

Mr. Bertucio reviewed the entire plant model, including the accident sequence event trees; the system fault tree models and their associated system notebooks.

- Mr. Jeff Julius

Mr. Julius reviewed the Human Reliability Analysis. His review included both the human reliability analysis approach and results.

- Mr. P. J. Fulford

Mr. Fulford reviewed the Level II Containment Performance Analysis, including the following.

  - Bins and Plant Damage States
  - Containment Failure Characterization
  - Containment Event Tree
  - Radionuclide Release Characterization
  - Containment Event Tree Quantification.

The review concentrated on the technical adequacy and accuracy of the PNPS PSA model. The review criteria are the criteria embodied in the Owners Group certification and NEI peer review processes.

NRC Request for Clarification to RAI 2.b

From the response, it is not clear for what the CET release categories (see Table RAI.2-4) are used. It appears that they have no function. Please discuss.

Response to Request for Clarification to RAI 2.b

The CET release categories have no direct function in determining the final collapsed accident progression bins used in the MACCS2 evaluation. The CET release categories presented in Table RAI.2-4 are used as a method to present the Level 2 results.

NRC Request for Clarification to RAI 2.c

1. Please describe the terms in source term algorithm equation in more detail. For example, how do the in-vessel releases get to the environment, what if there is no core concrete interaction, and what about revaporization/resuspension from containment surfaces? Please discuss.

2. The CAPB definitions appear to combine in one CAPB sequences which are the same except that the fission product releases are either mitigated in the drywell, mitigated in the reactor building or are not mitigated. The three, presumably significantly different release fractions, are then frequency weighted to produce a release fraction for the CAPB. The validity of this process is not clear. The usual purpose of having several source term categories is to do separate consequence analysis for groups with different release characteristics. Please discuss.

3. The last portion of the response to 2b discussed release timings for the CAPBs while the last portion of 2c, part ii of the response indicates that release times were frequency weighted. Please clarify. What is considered a late release as far as the CAPBs are concerned?

Response to Request for Clarification to RAI 2.c.1

All sources of release are considered, including in-vessel releases, ex-vessel releases due to core-concrete interaction (CCI), and in-vessel revolatilization releases from the primary coolant system after vessel breach.

The magnitude of the source term release resulting from an accident progression was estimated using a source term algorithm. This algorithm is a set of algebraic expressions that calculate release of each radionuclide group to the environment based on the release from fuel debris and removal mechanisms active in the severe accident progression.

Several terms must be defined to understand the algorithm.

R (Release to Environment). The release of fission products to the environment that is attributable to a distinct source. These sources are:

- In-vessel releases (R_{IV}). Releases to the environment due to core melt in-vessel.
- Ex-vessel releases due to core-concrete interaction (CCI) inside containment (R_{CCI})
- In-vessel revolatilization releases from the primary coolant system after vessel breach (I, CS and TE only) (R_{REV}).

RF (Release Fraction). Release fraction is the fraction of material in a given fission product group that evolves from the core debris and becomes available for release to the environment. Deposition mechanisms act on this material to limit its ultimate release to the environment. Release fractions were defined for each source:

$RF_{IV\ (i)}$ In-vessel release fraction for each fission product (I).

$RF_{CCI\ (i)}$ Core-concrete interaction release fraction.

$RF_{REV\ (i)}$ Revolatilization release fraction.

DF (Decontamination Factor). The decontamination factor accounts for the reduction in airborne mass of fission products by the deposition mechanism. Mathematically, the DF is the ratio of fission product mass entering (or initially present in) a volume to the mass leaving. The inverse of the decontamination factor is the transmission factor.

Decontamination factors were defined for each product group for primary coolant system and vessel deposition, torus scrubbing and drywell sprays decontamination:

DF_{VSL} Primary coolant system and vessel decontamination factor.

DF_{ESPY} Drywell sprays decontamination factor for in-vessel releases.

DF_{ECONT} Containment natural deposition decontamination factor for in-vessel releases.

DF_{CCI} Core-concrete interactions overlying pool scrubbing decontamination factor.

DF_{LCONT} Containment natural deposition decontamination factor for ex-vessel and revolatilization releases.

DF_{LSPY} Drywell sprays decontamination factor for ex-vessel and revolatilization releases.

DF_{TORUS} Decontamination factor for aerosol species flowing from the vessel to the torus.

$DF_{DW\text{-}TORUS}$ Decontamination factor for aerosol species flowing from the drywell to the torus.

DF_{RB} Decontamination factor for aerosol species from the reactor building to the environment.

Making use of these terms, we can calculate the total release to the environment as:

$$R_{env(i)} = R_{IV(i)} + R_{CCI(i)} + R_{REV(i)}$$

Where the release terms are defined as follows:

<u>In-vessel releases:</u>

$$R_{IV(i)} = \frac{RF_{IV(i)}}{DF_{IV}}$$

<u>Ex-vessel releases:</u>

$$R_{CCI(i)} = \left[1 - RF_{IV(i)}\right] * \frac{RF_{CCI(i)}}{DF_{CCI}}$$

<u>In-vessel revolatilization release:</u>

$$R_{REV(i)} = \left[RF_{IV(i)} * \left(1 - \frac{1}{DF_{VSL}} \right) \right] * \frac{RF_{REV(i)}}{DF_{REV}}$$

Where:

DF_{IV} = decontamination factor for in-vessel releases

$$DF_{IV} = DF_{ECONT} * DF_{ESPY} * DF_{TORUS} * DF_{RB}$$

DF_{CCI} = decontamination factor for core-concrete interaction releases

$$DF_{CCI} = DF_{LCONT} * DF_{LSPY} * DF_{POOL} * DF_{DW-TORUS} * DF_{RB}$$

DF_{REV} = decontamination factor for revolatilization releases

$$DF_{REV} = DF_{LCONT} * DF_{LSPY} * DF_{RB}$$

Release fractions (RFs) and decontamination factors (DFs) are calculated in a manner consistent with their definitions. RFs are calculated by dividing the release by the initial mass; DFs are calculated by dividing the mass entering a volume by the mass leaving that volume (e.g., DFVSL is the ratio of released mass in-vessel to the mass in the containment at vessel failure). This method for calculating RFs and DFs from the MAAP output data is illustrated below for a few source term algorithm parameters.

In-vessel Release Fraction (RF_{IV})

$$RF_{IV} = \frac{\text{Inv Rel}_f}{\text{Initial}_f}$$

Where,

Inv Rel_f = the final isotope mass in-vessel release,
$Initial_f$ = the initial isotope.

Decontamination Factor for the RPV (DF_{VSL})

$$DF_{VSL} = \frac{\text{Inv Rel}_v}{\text{Cont}_v}$$

Where,

Inv Rel_v = the isotope mass in-vessel release at vessel failure,
$Cont_v$ = the isotope mass in the containment at vessel failure.

Ex-vessel Release Fraction (RF_{CCI})

$$RF_{CCI} = \frac{\text{Exv Rel}_f}{\text{Initial}_f - \text{Inv Rel}_v}$$

Where,

Exv Rel_f = the final isotope mass ex-vessel release.

Revolatilization Release Fraction (RF_{lV})

$$RF_{REV} = \frac{Ps_v - Ps_f}{Ps_v}$$

Where,

Ps_f = the final isotope mass in the primary system,
Ps_v = the isotope mass in the primary system at vessel failure

The remaining decontamination factors are present in Table RAI.2-5.

The above equations were transported into an EXCEL spreadsheet and calculations were performed for all releases to the environment for the release endstates represented in the CET.

Response to Request for Clarification to RAI 2.c.2

The CAPBs used in the consequence analysis are distinct source terms groups with different source term characteristics. The initial process of generating specific source terms was accomplished by grouping releases into release categories that represent all postulated accident scenarios that produce a similar fission product source term. The criteria used to characterize the release are the estimated magnitude of total release and the timing of the first significant release of radionuclides. This process resulted in the generation of hundreds of source terms. Since it was not feasible to perform a calculation with the MACCS2 consequence model for each of the source terms, another interface was developed between the source term analysis and the consequence analysis. This interface involved grouping the large number of source terms into a much smaller number of source terms.

These source term groups were defined in terms of similar accident progressions properties and were frequency weighted for each group. The properties are: the occurrence of core damage, the occurrence of vessel breach, primary system pressure at vessel breach, the location of containment failure, the timing of containment failure and the occurrence of core-concrete interactions.

The source terms used in the consequence analysis were determined as follows:

1. The appropriate source terms based on the source term algorithm (described in Response to RAI 2.c.1) were selected and assigned to a particular CET accident progression endstate.
2. Based on the source terms from Step 1, the source term for each plant damage state CET accident progression endstate was determined.
3. The mean frequency of each release category was determined by summing the individual plant damage state CET accident progression endstates contained in the particular release category (i.e., no containment failure, early high release, etc.).
4. The release category individual fractional contributions for each CET accident progression were determined by dividing the result from Step 3 by the individual PDS frequency.
5. Each PDS accident progression CET endpoint source term, release timing, release energy and release elevation was multiplied by the value determined in Step 4.
6. The individual results of Step 5 were summed to arrive at the final values used in the MACCS2 analysis.

The validity of this binning process was examined by calculating the population dose risk (PDR) and offsite economic cost risk (OECR) for each the of the eleven CET accident progressions bins that are binned into CAPB-14. The results, presented in Table RAI.2-6, show only minor variation in the PDR and OECR values for the two approaches. Therefore, the binning process is reasonable.

Response to Request for Clarification to RAI 2.c.3

The release timings for the CAPBs used in the MACCS2 analysis are based on frequency weighted values. The initial values of the CAPB release timings (prior to performing the frequency weighted calculations) are described in the response to RAI 2b.

CAPB releases timings greater than 7.5 hours are considered a late release.

Table RAI.2-5 Remaining Pilgrim Decontamination Factors

Decontamination Factors	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba	Reference
Drywell sprays decontamination factor for in-vessel releases (DF_{ESPY})	1	11	11	11	11	11	11	11	11	NUREG/CR-4551, Vol.4, Rev.1, Part 2, page B.2.3
Containment Decontamination Factor for In-vessel Releases (early rupture failure-saturated pool) (DF_{ECONT})	1	1.32	1.32	1.35	1.35	1.35	1.35	1.35	1.35	Table 5.1 page 5.5-10 NUREG/CR-4551 Vol. 2 Part 4 Rev.1
Containment Decontamination Factor for Ex-vessel Releases (early rupture failure-saturated pool) (DF_{ECONT})	1	1.32	1.32	1.41	141	1.41	1.41	1.41	1.41	Table 5.1 page 5.5 -22 NUREG/CR-4551 Vol. 2 Part 4 Rev.1
Containment Decontamination Factor for Ex-vessel Releases (late rupture failure-saturated pool) (DF_{LCONT})	1	14.93	14.93	10.53	10.53	10.53	11.77	11.77	11.77	Table 5.1 page 5.5-26 NUREG/cr-4551 Vol. 2 Part 4 Rev.1
Drywell sprays decontamination factor for ex-vessel and revolatilization releases (DF_{LSPY})	1	11	11	11	11	11	11	11	11	NUREG/CR-4551, Vol.4, Rev.1, Part 2, page B.2.3
Decontamination factor for aerosol species flowing from the vessel to the torus (DF_{TORUS})	1	81	81	81	81	81	81	81	81	NUREG/CR-4551, Vol.4, Rev.1, Part 2, page B.2.3
Decontamination factor for aerosol species flowing from the drywell to the torus ($DF_{DW\text{-}TORUS}$)	1	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	NUREG/CR-4551, Vol.4, Rev.1, Part 2, page B.2.3
Decontamination Factor for the Overlying Pool (DF_{Pool})	1	4.4	4.4	4.4	4.4	4.4	4.4	4.4	4.4	NUREG/CR-4551, Vol.4, Rev.1, Part 2, page B.2.4
Decontamination factor for aerosol species from the reactor building to the environment (DF_{RB})	1	2.49	2.49	2.59	2.59	2.59	2.59	2.59	2.59	NUREG/CR-4551, Vol.4, Rev.1, Part 2, page B.2.3 (these values represents the average values)

Table RAI.2-6 Comparison of PDR and OECR for each APB Release Mode for CAPB-14

Release Mode	Frequency (/yr)	Population Dose Risk (PDR) (person-rem/yr)	Off-site Economic Cost Risk (OECR) ($/yr)
APB-3	5.73E-07	1.89E-01	9.34E+01
APB-4	3.04E-08	1.00E-02	4.96E+00
APB-7	3.18E-08	6.91E-02	2.42E+02
APB-8	2.37E-08	8.07E-02	2.98E+02
APB-9	2.37E-08	1.32E-01	5.05E+02
APB-31	2.88E-08	4.93E-02	1.69E+02
APB-32	2.37E-08	7.67E-02	2.91E+02
APB-33	2.37E-08	1.29E-01	4.91E+02
APB-55	1.44E-06	2.46E+00	8.45E+03
APB-56	2.71E-08	8.78E-02	3.33E+02
APB-57	4.73E-08	2.57E-01	9.80E+02
APB Totals	**2.27E-06**	**3.54E+00**	**1.19E+04**
CAPB-14 Totals	**2.26E-06**	**3.82E+00**	**1.12E+04**

NRC Request for Clarification to RAI 3.c

The response to this RAI includes a revision to the analysis of SAMAs 2 and 19. The basis for this revision is described in Note 1 to Table RAI.3-2.

1. Note 1 indicates that successful venting PDSs result in CAPBs 12, 13, 14 and 15. Two of these CAPBs (14 and 15) are stated in Table E.1-9 to involve drywell failure due to core-concrete interaction. It is not clear how, with successful venting the drywell would fail due to CCI or with drywell failure the filtered vent would impact the releases. Please clarify the assignment of containment venting to accident progression bins (CET end states) and CAPBs.

2. Note 1 provides the change in release fractions for the 4 CAPBs for the assumed factor of 2 reduction in source term due to the filtered vent. While several source terms are not reduced at all or are reduced by only a few percent, the Cs and I release fractions are reduced by approximately 50% (i.e. a factor of 2) and the Te source term is reduced by more than a factor of 10. Please discuss these source term reductions.

Response to Request for Clarification to RAI 3.c.1

The CET classifies containment venting from either the drywell or torus as containment failure, hence, a CET sequence that involves 'successful venting' with core damage occurring post vent, and the occurrence of core concrete interactions (CCI) is valid. The occurrence of CCI in the CAPB description does not necessarily imply drywell failure due to CCI-induced over-pressure failure. Containment failure in CAPBs 12, 13, 14 and 15 could result from any of the severe accident phenomena (i.e., high temperatures drywell seal failure, containment overpressurization due to CCI, etc.). The occurrence of CCI is used an accident progression characteristic that influences the release.

Response to Request for Clarification to RAI 3.c.2

Note 1 to Table RAI.3-2 compares the base case CAPB source terms and the revised CAPB source terms associated with 'TW' plant damage states.

As stated in Note 1, the assumed factor of reduction of 2 in source terms due to the filtered vent was applied to the source terms associated with core damage accident sequences that are binned into plant damages states 1, 5, 12, 18, 40 and 43. These plant damages represent successful venting with core damage occurring after containment venting. The source term reduction of these plant damage states influences the final source terms for CAPBs 12, 13, 14, 15 as presented in Note 1 of RAI response 3.c.

NRC Request for Clarification to RAI 5.a

While, as stated in the response, relocation of equipment to eliminate or reduce the correlated failure of the vulnerable equipment would be cost prohibitive, relocation is not the only way of reducing the risk due to the identified pieces of equipment. Simple, inexpensive fixes might be possible depending on the failure mode. For example, if the mode of failure of the electrical panels is a simple structural failure such as tipping due to lack of a top support or failure of an anchor bolt, then a fix might be cost effective. Please discuss the individual failure modes in assessing the potential for cost effective SAMAs.

Response to Request for Clarification to RAI 5.a

IPEEE Table 3-15 lists the important seismic faults that dominate seismic risk.

The seismically correlated events were developed to account for failure of *like-equipment* located on the same elevation. *Like equipment* is defined as equipment of the same manufacturer, same model, and same anchorage capacity. Major groups of identical components located on the same elevation were represented by a seismic "*Common Cause*" event that fails all components in that group. The conditional failure probability for a single component was assigned to the common cause event (i.e., complete correlation).

However, the individual equipment which make up the correlated failure combinations have high seismic capacity and do not show up as important seismic faults that dominate seismic risk. The correlated events are relatively important because they fail all components in the group. Thus, simple inexpensive fixes that would reduce the conditional failure probability for a single component would not significantly reduce the relative importance of the correlated event. Therefore, only reorientation and relocation, not individual failure modes, were considered in assessing the potential for cost effective SAMAs.

NRC Request for Clarification to RAI 5.e

Apparently, the modifications included in SAMA 27 were revised along with the evaluation of the benefit from these modifications. That is, the benefit was increased by including the reduction in initiator frequency but the cost went up by including a new DC power source in the modification. Would the original modifications considered impact the initiating event frequency and if so what would be the benefit?

Response to Request for Clarification to RAI 5.e

The original modification considered in SAMA 27 would impact the initiating event frequency. However, a more refined cost estimate was performed on this original modification. The revised cost estimate for this SAMA to mitigate the loss of DC-bus initiator by improving DC bus reliability is $1,953,682 and its revised baseline benefit with uncertainty is $1,341,800. Therefore, this SAMA is not cost effective for PNPS.

NRC Request for Clarification to RAI 5.g

The cost of a redundant diesel fire pump is given as over $5.5 Million. This seems excessive. Please justify.

Response to Request for Clarification to RAI 5.g

Cost estimates for SAMA consideration followed Entergy's standard process for development of project estimates. The process is applied to establish conceptual (+/- 25% to 50% accuracy), preliminary (+/- 15% to 30% accuracy), and definitive (+/- 10% to 20% accuracy) estimates during the study, design, and implementation phases of a design project. This procedure replaced and enhanced the completeness and accuracy of estimates previously developed.

The SAMA cost estimates capture all anticipated expenses by identifying all parts of the organization that must support the proposed SAMA modification from the conceptual perspective. Typical expenses associated with project cost estimating include calculations,

drawing updates, specification updates, bid evaluations, contract issuance, design package preparation, walkdowns, planning and scheduling, estimating, procurement, configuration management, ALARA, QC/QA, training, simulator, IT, design basis update, construction, multi-discipline and independent review of design concepts and calculations, 50.59 review, FSAR update, cost control, contingency, security, procedures, post work testing, and project management and close-out. In addition, the project cost estimates include corporate indirect charges.

In summary, the cost estimates for the subject SAMAs followed Entergy's standard process for development of project estimates. Therefore, these cost estimates are reasonable conceptual level estimates.

NRC Request for Clarification to RAI 5.h

The cost of changing two valves to fail open on loss of air or power is given as $3.2 million. This seems excessive. Please justify.

Response to Request for Clarification to RAI 5.h

See response to NRC Request for Clarification to RAI 5.g.

NRC Request for Clarification to RAI 6.a

1. For SAMAs 6 and 20, it appears that flooding internal to the drywell was evaluated. It would appear that flooding (or sprays) on the outside might serve the same purpose and avoid the necessity for the relocation of the drywell vent. Please discuss.

2. For SAMAs 7 and 21, it would appear that use of existing fire water sprays might be effective in mitigating releases. Please discuss.

3. SAMA 22 is understood to be simply providing a means of flooding the floor of the drywell rather than providing a core retention device (the latter is considered in SAMA 5). Please clarify.

Response to Request for Clarification to RAI 6.a.1

SAMAs 6 and 20 evaluated flooding internal to the drywell to ensure the drywell head seal does not fail due to high temperature. Flooding or sprays on the outside might serve the same purpose, but would still cost more than the estimated benefit for these SAMAs ($0 in Table RAI.3-2).

Response to Request for Clarification to RAI 6.a.2

There are only a few fire protection automatic suppression systems within the reactor building (23' and 51' elevations). As such, they have limited capability in providing fission product scrubbing. The proposed design modification would upgrade the fire protection system to a sufficient capacity to handle postulated loads from severe accidents. The revised baseline with uncertainty value of $94,714 is less than the estimated implementation cost of greater than $2.5 million. Therefore, this SAMA is not considered to be cost effective.

Response to Request for Clarification to RAI 6.a.3

To clarify, SAMA 22 would allow debris to be cooled by providing a means of flooding the floor of the drywell rather than providing a core retention device. In SAMA 5, the proposed design modification involves a core retention device inside the reactor pedestal area.

NRC Request for Clarification to RAI 6.d

The appropriateness of a factor of 3 reduction in operator failure to vent for SAMA 53 is not clear. The benefit of the controlled venting occurs for sequences involving successful venting which are not significantly affected by reducing the operator error to vent. In addition, the evaluation for SAMA 53 leads to only a 3.6% reduction in CDF while the provision of a passive vent considered in response to RAI 5.h (which presumably eliminates all failure to vent sequences) led to a 14.5% reduction. Please provide further support for the evaluation.

Response to Request for Clarification to RAI 6.d

SAMA 53 (Control containment venting within a narrow band of pressure), would establish a narrow pressure control band to prevent rapid containment depressurization when venting is implemented thus avoiding adverse impact on the low pressure ECCS injection systems (LPCI and core spray) taking suction from the torus.

The original response to RAI 6.d used a factor of 3 reduction in the operator failure to vent probability based on the following:

- Current PSA does not model control venting to allow LPCI and Core Spray operation,
- Modeling of control venting requires impact on net positive suction head (NPSH) requirements for LPCI and Core Spray when opening the torus vent path, and
- Examination of the feasibility of re-closing the torus vent valves AO-5042B and AO-5025 against high containment pressures is not available.

In response to this request for clarification, the following alternative means of evaluating the benefits for SAMA 53 is provided.

This evaluation was performed by crediting continued vessel injection from LPCI or Core Spray for those sequences in which torus venting is successful and alternative injection systems fail after torus venting. Specifically, an additional event (LPCI-CS) was added to the PSA results (cutsets) that involve successful torus venting.

Currently there is no detailed engineering analysis that examines the impact of opening the torus vent path on NPSH requirements for LPCI and Core Spray and no examination of the feasibility of re-closing the torus vent valves AO-5042B and AO-5025 against high containment pressure. However, MAAP computer runs for Vermont Yankee Nuclear Power Station (VYNPS) predict that the available NPSH will be below the required NPSH following opening of the torus vent path. Since PNPS and VYNPS have Mark I containments and similar vent size opening (8-inches), the same NPSH difficulties are expected for PNPS. Since the available NPSH is likely to be less that the required NPSH with the vent path open, a failure probability of 0.9 was assigned to event LPCI-CS.

This PSA model change resulted in a 2.54 percent reduction in CDF and a revised baseline benefit of approximately $387,096.

As described in the response to RAI 6.a, the cost of performing an engineering analysis, procedure changes, simulator changes, and training is estimated to be $300,000. Therefore, this SAMA is potentially cost effective for PNPS provided the existing torus vent path, valves, and controls do not require hardware modification.