generator tube rupture event (SGTRN). Due to the limited inventory of the SFP, continuous makeup to the RWST as a recovery action requires that the RCS fluid loss be minimized prior to making up to the primary system via spent fuel pumps.

- Modeling of the requirement to depressurize before crediting continuous makeup to the RWST after loss of sump recirculation mode during SLOCA and transient induced LOCA scenarios
- A higher consequential LOSP probability used in the PRA
- New LOSP initiators were defined for this revision of the PRA model that separate offsite power losses into four categories (Grid, Plant, Switchyard and Severe Weather related). The overall effect of these changes to the initiators and to the electric power recovery factors was a decrease in the contribution of LOSP to CDF.
- Addition of common cause failure of DC buses and batteries into the model as evident from the increased contribution to CDF from general transient initiators such as Reactor trip, Turbine Trip, Partial Loss of Feedwater, etc.
- Longer duration assumed for the Emergency Diesel Generator (EDG) maintenance windows as part of the EDG LAR submittal (Reference 47).

The increase in the internal LERF can be attributed to the following changes:

- Requirement to depressurize the RCS to terminate the loss of primary coolant to the secondary side and the initiation of closed loop RHR cooling in the event of an SGTRN event which, as stated above, has increased SGTRN contribution to CDF. Since all SGTRN events resulting in CDF are directly considered to be LERF contributors, the increase in SGTRN CDF has directly resulted in an increase in LERF
- Replacement of the simplified LERF model with revised detailed Level 2 model

The increase in the seismic LERF can be attributed to the above requirement to depressurize the RCS for SGTRN event which has the effect of causing an increase in the Internal Events LERF. Since seismic LERF was quantified with the same simplified LERF model as before, the percentage increase of seismic LERF was less than the percentage increase of Internal Events LERF.

Fire-induced LERF was not quantified in the DC01 model. The new Level 2 model does not account for the effects of fire on containment response.

F.2.1.9 MODEL DC02

The main objectives of the model update are (1) routine maintenance update and (2) to include the upgrade of the Internal Floods model. The changes are documented in PRA Calc C.9 Revs 11, 11a and 12.

- Incorporate any changes to the system models due to design changes, etc.
- Incorporate any changes to the system and plant models as documented in the PRA Tracking database. The changes include findings from gap assessments, peer reviews, internal reviews, etc.
- Using the updated component failure/unavailability database, the updated human error probability values, Electric Power Recovery Model and the updated Internal Events frequency value
- The internal flooding analysis previously performed in response to USNRC Generic Letter 88-20 (Reference 32) was updated using the methodology prescribed in the EPRI report 1019194 (Reference 68) to the requirements of the ASME/ANS RA-Sa-2009 (Reference 80), as endorsed by Regulatory Guide 1.200 Revision 2 (Reference 81).

The major changes to the Internal Events model include;

- Crediting the 3rd charging pump
- Addition of the CCW heat exchangers in the loss of the CCW initiating event Fault Tree
- Updating CFCU model
- Removal of the CST water volume limitation during a seismic event.

The DC02 PRA model was quantified and the results of the quantification are provided below:

Contributor	Mean Core Damage Frequency (per year)	Mean Large Early Release Frequency (per year)
Internal Events	1.35E-05	1.59E-06
Unit 1 Internal Flooding	1.42E-06	8.85E-08
Unit 2 Internal Flooding	6.79E-07	4.38E-08
Seismic Events	2.74E-05	1.58E-06
Fire Events	1.70E-05	-
Unit 1 Total	**5.93E-05**	**3.26E-06** (1)
Unit 2 Total	**5.86E-05**	**3.21E-06** (1)

Note:
(1) Total LERF does not include contribution from fire initiators

The increase in CDF can be primarily attributed to the following change to the model:

- An increase (by a factor of about 3) in the updated frequency of Medium LOCA.

The other reasons for the increase in the CDF can be attributed to the following:

- An increase in the frequency of Loss of 480V Switchgear Room ventilation system and the HEP for the recovery action to provide alternate cooling to the switchgear rooms.
- An increase in the frequency of Loss of Component Cooling Water system (due to the inclusion of CCW heat exchangers and piping failures in the model) and the increase in the frequency of the updated Small LOCA event.

However there are also changes in the model that decreases the overall CDF contribution such as:

- A decrease in the frequency of general transient events.
- Inclusion of the 3rd charging pump which reduces the CDF from loss of RCP seal cooling events.

The overall decrease in the internal LERF can be attributed primarily to the decrease in the frequency of the main contributor to LERF which is an isolated Steam Generator tube rupture. There is also an increase in the contribution to LERF from Medium LOCA due to the increase in its frequency.

F.2.1.10 MODEL DC03 (INTERIM)

This model includes the updates to the Internal Events, Internal Flooding, Seismic, and Internal Fire events incorporating the following changes;

- Resolution of the 2012 Internal Events and Internal Flooding Peer Review F&Os
- Model refinement as part of implementing Risk Informed Tech Specification 4.b

The original seismic model developed as part of the Long Term Seismic Program (Reference 30) and later updated for the Individual Plant Examination of External

Events (Reference 4) is upgraded to satisfy the requirements of the 2009 ASME/ANS PRA standard (Reference 80) as well as incorporating the following changes:

- Updated probabilistic seismic hazard curves incorporating the Shoreline Fault, as well as updates for other regional faults (using values from the 2011 Shoreline Fault Report [Reference 54])
- Updated HEPs for the Internal Event Model and the impact of seismic events on these HEPs

At the time of the 2013 Seismic Peer Review, the Turbine Building Shear Wall fragility had been updated based on the latest hazard spectral information, which was updated with the Shoreline Fault. The fragility analysis for other SSCs was based on the LTSP. The LTSP fragility curves are acceptable for use in DC03 because no scaling is necessary for use with the updated hazard spectral information. The LTSP fragility curves are the same shape (+/-10%) in the period of interest (3-8.5Hz) and there are no components in the PRA model in the 1-3Hz range. In addition, the 2014 Central Coastal California Seismic Imaging Project Report (Reference 53) was a deterministic analysis that did not change the inputs to the seismic PRA. As a result, the interim results and insights from the DC03 model are reasonable for the purposes of a SAMA analysis and therefore are used in this report. For comparison, a summary of the changes in initiating event frequencies used in recent versions of the seismic PRA are summarized below. Currently, an update of the seismic hazard will be submitted in March 2015 to the NRC in response to NRC letter dated March 12, 2012 regarding 10 CFR 50.54(f) request for information pursuant to the post-Fukushima Near-Term Task Force Recommendation 2.1 seismic hazards reevaluation. The impacts of the 2015 seismic hazard results on the SAMA analysis will be addressed following submittal of the 10 CFR 50.54(f) response.

Seismic Initiating Event Frequency Update Summary

Initiating Event	DC01A	DC01B	2010 RAI Response[2]	DC03
SEIS1 (g levels 2.0E-1 to 1.25)	1.72E-02	7.40E-03	1.16E-02	1.30E-02
SEIS2 (g levels 1.25 to 1.75)	8.69E-04	3.50E-04	3.53E-04	3.18E-04

SEIS3 (g levels 1.75 to 2.00)	1.56E-04	7.00E-05	7.32E-05	5.15E-05
SEIS4 (g levels 2.00 to 2.50)	1.24E-04	6.70E-05	6.92E-05	4.14E-05
SEIS5 (g levels 2.5 to 3.00)	2.94E-05	2.60E-05	2.64E-05	1.28E-05
SEIS6 (g levels 3.00 to 3.99)	7.64E-06	1.74E-05	1.45E-05	6.15E-06
New (g levels greater than 4.0)	Note 1	Note 1	2.40E-06	1.16E-06

Note 1: The DC01A and DC01B PRA models did not consider acceleration levels greater than 4.0 g based upon low frequency of occurrence. The 2010 seismic hazard data has a contribution at acceleration levels greater than 4.0 g and is included here.

Note 2: Response to RAI 3.c in PG&E Letter DCL-10-106, Enclosure 1, dated August 27, 2010, provided the 2010 Shoreline Fault Report values.

As can be seen in the table, the most current frequencies, based on the hazard results in the 2011 Shoreline Fault Report (Reference 54), are bounded by the DC01A, DC01B, and 2010 frequencies, except for the lowest level (0.2 g to 1.25 g), which is bounded by the DC01A model, and the highest level (>4.0 g), for which there was no frequency in the DC01A or DC01B models, but is bounded by the 2010 frequency.

The Internal Fire model (i.e., FPRA) is being developed in support of transitioning the DCPP fire protection program to NFPA 805 (Reference 82), mainly based on the guidance provided in NUREG/CR-6850 (Reference 83). The FPRA has been peer reviewed and the peer review F&Os are being resolved. The FPRA includes the following plant modifications yet-to-be implemented, committed as part of the NFPA 805 LAR submittal (Reference 84).

- Westinghouse Low Leakage Shutdown Seal (i.e., GEN III)
- Upgrade of the Hot Shutdown Panels
- Installation of Incipient Detection system in Cable Spreading and SSPS rooms
- Various Electrical Raceway Fire Barrier System (ERFBS) installation

The DC03 PRA model was quantified and the results of the quantification are provided below:

Contributor	**Mean Core Damage Frequency (per year)**	**Mean Large Early Release Frequency (per year)**
Internal Events	1.12E-05	1.30E-06

Unit 1 Internal Flooding	6.43E-06	3.82E-07
Unit 2 Internal Flooding	5.47E-06	3.49E-07
Seismic Events	1.45E-05	5.54E-06
Unit 1 Fire	5.43E-05	4.77E-06
Unit 2 Fire	5.07E-05	4.42E-06
Unit 1 Total	**8.64E-05**	**1.20E-05**
Unit 2 Total	**8.19E-05**	**1.16E-05**

Figures F.2-1 and F.2-8 depict internal, seismic, fire, and internal flooding contribution to CDF and LERF associated with the DC03 PRA model. The relative contribution from Unit 2 fire areas to fire and internal flooding CDF and LERF is similar to those of Unit 1.

F.2.2 DESCRIPTION OF LEVEL 1 TO LEVEL 2 MAPPING

The foundation for the DCPP PRA Level 2 model is the work done for the individual plant examination (IPE) (Reference 33), or Model DCPRA-1991. It was in this model that the containment event tree was developed, containment failure probability assigned, and endstate binning logic built to analyze the core damage effects and assign Level bins. There has been no major overhaul of that logic since then, except for the changes made in the DC01 model to allow a more realistic assessment of the Large Early Release Frequency figure of merit (Reference 51).

Each accident sequence from the Level 1 event tree quantification has a unique "signature" due to the particular combination of top event successes and failures. Ideally, each accident sequence that results in core damage should be evaluated explicitly in terms of accident progression and the release of radioactive materials, if any, to the environment. However, since there can be millions of these sequences, it is impractical to perform such analyses for each accident sequence. Therefore, the sequences must be grouped into plant damage state (PDS) bins. Each of these PDS bins collects all of those sequences for which progression of core damage, release of fission products from the fuel, and the potential for mitigating source terms are similar. Detailed analyses are then focused on specific sequences selected to represent each of these bins.

F.2.2.1 DEFINITION OF PLANT DAMAGE STATES

The PDS bins become the entry states (similar to Initiating Events for the Level 1 event trees) to the containment event trees (CETs), and are characterized by the thermodynamic conditions in the reactor coolant system and in the containment at the time of severe core damage, and the availability of both passive and active plant features that can terminate the accident or mitigate the release of radioactive materials into the environment.

The plant damage states must be defined such that within a plant damage state, the sequence-to-sequence variability in the containment response is small in comparison with the uncertainties in the outcome of the CET top events. In this manner, the issue of containment response uncertainty, and the evaluation of split fractions for the CET top events can be limited to the assessment of containment response uncertainty.

Before PDSs are defined, plant conditions, systems, and features that can have a significant impact on the potential course of an accident must be identified. Once these are identified, a table is constructed to display all of the potential combinations of the PDS characteristics that are physically possible, and to assign an identifier to each of these combinations. The table that results from this process is referred to as a PDS matrix. The definition of the PDS matrix is based on the anticipated response of the plant and containment to severe accidents. Requirements of Level 1 and 2 analyses must both be considered when constructing the matrix.

The containment event tree (CET) could, in principle, be quantified separately for each PDS as the values of the CET top event split fractions can be dependent on the PDS being analyzed. In practice, however, PDSs can be grouped (or binned) together to minimize the amount of CET quantification. Generally, PDSs of lower frequency are grouped with similar PDSs of higher frequency and potentially higher consequences (referred to as "conservative condensation") into key plant damage states (KPDS).

A representative accident sequence is selected for each KPDS. These representative sequences are analyzed in detail with appropriate thermal-hydraulic and fission product codes such as the Modular Accident Analysis Program (MAAP) to characterize the timing of important events (such as the onset of severe core damage and reactor vessel melt-through), as well as the nature of the core damage and fission product release.

All of the plant model information on the operability status of active systems that is important to the timing and magnitude of the release of radioactive materials must be passed into the CET via the definition of the KPDS. This requires that, in addition to representing the systems and functions that are important to keeping the core cooled, the plant model event trees must also address active systems and functions important to containment isolation, containment heat removal, and the removal of radioactivity from the containment atmosphere. The containment spray system is one example of such systems.

The boundary between the Level 1 event trees and the containment event tree was selected for the following reasons:

- All active systems, including the containment engineered safeguards, are included in the Level 1 Event Trees because their dependencies on support systems such as electrical power and component cooling water can be determined more easily in the Level 1 Event Trees,
- The boundary separates the phenomenological model (the CET) from the Level 1 Event Trees that deal only with active systems and operator actions,
- The boundary separates analyses which can be characterized by likelihood (as measured by frequency, i.e. Level 1 Event Trees) with analyses characterized by uncertainty (as measured by probability, i.e. CET).

Separation of the Level 1 Event Trees and the Containment Event Trees allows flexibility in the modeling process. The Level 1 model can be modified without changing the CET, as long as the PDSs are not changed.

The Containment Event Tree considers the effects of physical and chemical processes on the integrity of the containment and on the release of fission products once core damage has occurred. Considerations that influence progression of core damage, the time and mode of containment failure, and the release of radioactive materials to the environment fall into two categories:

- The physical conditions in the reactor coolant system and containment at the time of vessel melt-through, and
- The status and availability of containment systems.

Physical conditions in the RCS and containment that are defined in the PDS matrix are as follows:

- The pressure inside the reactor vessel at the onset of core damage (the pressure at vessel breach will be assessed in the containment event tree),
- The availability of cooling on the secondary side of the steam generators, and
- Whether or not the reactor cavity is flooded at the time of vessel melt-through.

Thus, the PDS matrix defines the following parameters at the onset of core damage:

- RCS pressure at onset of core damage
- Secondary Side Steam Generator Cooling availability
- Whether the RWST is injected into containment at onset of core damage
- Status of Containment Spray and Containment Heat Removal
- Containment Integrity at time of vessel melt-through

Table F.2-1 presents the five parameters discussed above, the rationale for the category selection, the code used in the PDS matrix, when the code is applicable, binning logic, and any comments. This table fully documents the definition of the plant damage state matrix.

F.2.2.2 ORGANIZATION OF PLANT DAMAGE STATES

To help organize and categorize each unique PDS, Table F.2-2 was developed based on the definitions in Table F.2-1. Each cell in the matrix of Table F.2-2 represents a unique plant damage state. Each PDS is identified by a five character identifier using the codes from Table F.2-1. The five character identifier is named as follows:

- The first character represents RCS pressure at onset of core damage. Valid codes are L, I, H, and S, as described in Table F.2-1.
- The second character represents the availability of secondary side steam generator cooling. Valid codes are N, A, and X as described in Table F.2-1.
- The third character indicates whether water from the RWST had been injected into containment prior to vessel breach. Valid codes are N and Y, as discussed in Table F.2-1.
- The fourth character indicates whether containment spray and containment heat removal are available. Valid codes are A, B, C, D, E, F, G, and N, as discussed in Table F.2-1.
- The fifth character indicates containment isolation and bypass status. Valid codes are I, S, L, B, and V, as discussed in Table F.2-1.

The matrix shown in Table F.2-2 includes 12 rows (representing unique combinations of RCS pressure, status of steam generator cooling, and status of RWST injection into containment) and 32 columns denoting the status of the containment isolation and the availability of containment spray and containment heat removal systems.

The complete PDS matrix shown in Table F.2-2 contains a total of 384 plant damage states. However, the number of PDSs identified in Table F.2-2 can be substantially reduced because some are precluded by DCPP design features, by the type of initiating event, modeling assumptions, or any combination of these reasons. Any PDS that is precluded is shaded and footnoted in Table F.2-2.

F.2.2.3 DISCUSSION OF KEY PLANT DAMAGE STATES

As discussed above, in order to simplify CET quantification, the PDSs are grouped into key plant damage states (KPDS). In general, the following numbered guidelines are used to subsume the PDSs into KPDSs such that the KPDS is bounding from a Level 2 analysis perspective for the PDS grouping.

1. Higher RCS pressures prior to vessel breach cause greater containment pressure rise and an early challenge to containment integrity. Therefore, a lower pressure PDS could be binned to a higher pressure PDS (indicated by the first letter in the PDS identifier).
2. The unavailability of steam generator cooling increases vessel pressure at breach with the same consequences as discussed in Guideline 1. Therefore, a PDS with steam generator cooling available

could be binned to a PDS with steam generator cooling not available. The status of steam generator cooling is indicated by second letter in the PDS identifier.

3. Similar to Guideline 2, a PDS with steam generator cooling not applicable (e.g., large LOCA sequence) could be binned to a PDS with steam generator cooling not available.
4. The failure to inject the contents of the RWST reduces the heat sinks in the containment, eliminates a means of fission product scrubbing, and eliminates a means of cooling the core debris. These factors increase the likelihood of containment failure or increase the source term. A PDS in which the RWST inventory has been injected could be binned to a PDS in which the RWST inventory has not been injected. The status of the RWST is signified by the third letter in the PDS identifier.
5. In terms of containment heat removal, CFCUs have the greatest capacity and should be effective for the duration of the accident. Recirculation spray is also effective for the duration of the accident, while injection spray is effective for only a few hours. Any PDS that represents greater containment heat removal capacity could be binned to a PDS with less heat removal capacity. The status of containment heat removal is indicated by the fourth letter in the PDS designator.
6. States in which containment integrity is lost will release fission products sooner than states in which containment integrity is maintained. Therefore, a PDS in which containment integrity is maintained could be binned to a PDS without containment integrity. The status of containment integrity is indicated by the fifth letter of the PDS designator.
7. Large containment leaks release fission products more rapidly than small leaks. Therefore, a PDS exhibiting a small leak could be binned to a PDS representing a large leak.
8. Large containment bypasses release fission products more rapidly than small bypasses. Of the accidents analyzed, large containment bypasses have the highest source term and the earliest release of fission products. The large bypass represents the most severe KPDS scenario. Therefore, as a conservative measure, any non-large bypass PDS could be binned to a PDS with a large bypass.

Table F.2-3 presents the list of sixteen key plant damage states (KPDSs) selected based on the CDF values of Model DCPRA-1991 results and the above grouping guidelines. The relative ranking of the plant damage states and therefore the definition of the KPDSs is not expected to change significantly from Model DCPPRA-1991 to DC03.

F.2.2.4 DESCRIPTION OF CONTAINMENT EVENT TREE END STATES MAPPED TO RELEASE CATEGORIES

For large, dry containment PWRs such as DCPP, many factors can potentially affect radionuclide releases into the environment following a severe accident. The most important of these factors become the principal issues requiring consideration in the development of release categories. Release categories are groups of containment event tree end states that can be represented by similar source terms; i.e., the variations in source terms for accident sequences within a given release category are smaller than variations from one release category to another. Table F.2-4 shows the principal issues affecting the definition of release categories for large, dry containment PWRs.

F.2.2.4.1 Containment Event Tree Discussion

The containment event tree is shown in Figure F.2-9 with the top events described in Figure F.2-10. All of the quantified sequences which lead to containment failure or containment bypass were binned to the release categories shown in Table F.2-5. The binning logic for the release category assignment is documented in Reference 35. These release categories address the issues presented in Table F.2-4 and are explained below:

For the containment bypass sequences, separate release categories were created for the interfacing system LOCA and SGTR.

RCS pressure at the time of vessel failure was reduced to three categories, High, Medium, and Low. However, to simplify the RCS pressure category, the medium pressure sequences will be conservatively binned with the high or low pressure sequences as follows:

- For late containment failures where early high retention in the RCS can become a liability, the medium pressure sequences are binned with the high pressure sequences.
- For early-large containment failures, the medium pressure sequences are binned with the low pressure sequences because less retention in the RCS becomes a liability.

- For early-small containment failures, the medium pressure sequences are binned with the high pressure sequences because containment pressures may be greater later in the event and high RCS retention then becomes a liability.

Containment failure time was reduced to either Early (at the time of vessel failure due to DCH and/or hydrogen burning) or Late (> 4 hours after vessel failure due to hydrogen burning and/or long term overpressurization). Additionally, separate categories were created for no containment failure sequences (i.e. long-term intact) and basemat melt-through sequences.

Containment failure size was reduced to either Large (> 3 inch diameter failure where rapid containment depressurization is expected) or Small (< 3 inch diameter failure where gradual containment depressurization is expected).

Containment spray system (in recirculation spray mode) was assumed to be either On or Off for each of the "in-containment" release categories.

Similarly, the core debris was assumed to be either cooled or non-cooled after vessel failure and core relocation for each of the "in-containment" release categories. This assumption is particularly of note to DCPP since a dry cavity is postulated for all non-high pressure melt ejection sequences. Note that the CET considers a small, but non-zero probability for non-coolable core slump situations during HPME sequences, which implies that some high pressure sequences will result in CCI. Non-coolable release categories are designated with the letter "U" placed after the name of the similar coolable release category (e.g. RC03 and RC03U).

In summary, Table F.2-5 has a total of 32 release categories (RC01 and RC01U through RC16 and RC16U) that address the "in-containment" release sequences discussed above. Additionally, there are two containment bypass release categories for SGTR and interfacing system LOCA (RC17 and RC18), two containment intact categories for long term containment intact sequences and basemat melt-through sequences (RC20 and RC21), and a category for non-severe core damage sequences with an intact containment (RC19).

F.2.2.4.2 Release Category Source Terms

These 37 release categories (32 "in-containment" categories, 2 containment bypass categories, 2 containment intact categories, and one basemat melt-through sequence) are originally grouped into fire release category groups as discussed in Section 4.8.2 of the individual plant examination (IPE) submittal (Reference 33). The original release category grouping is presented in Table F.2-6. For this SAMA analysis, the release category RC18, interfacing system loss of coolant accident (ISLOCA), is separated from the containment bypass category group and assigned as its own release category group. Release category RC17 is split between the Containment Bypass group and the ISLOCA group. The Containment Bypass group is characterized by having AFW available prior to core damage. However the non-isolated SGTR (SGTRN) initiated sequences with AFW success can also end up in RC17 which are better characterized as ISLOCA with regard to Level 2 offsite release. Approximately 50% are initiated by SGTRN (non-isolated SGTR) and are therefore allocated to the ISLOCA group. The ISLOCA group is then comprised of 50% of the frequency of RC17 plus RC18. RC18 was reviewed to see if there are any sequences with AFW successful that should be allocated to the Containment Bypass group. None were found.

The ISLOCA group now also includes frequency from seismic initiators > 4g. Currently the DC03 model does not model seismic initiators at this elevated “g” level. Due to the severity of the “g” level the entire frequency is allocated to the ISLOCA group, i.e. the CCDP from this “g” level is assumed to be 1.0. Table F.2-6 also shows the naming convention for each release category groups (e.g., ST1 for Large, Early containment failures).

The information in Table F.2-6 is expanded in Table F.2-7 to include the frequency results of each release category (RC) from DC03 model and the KPDS-RC mapping information from the IPE submittal (Table 4.7-4).

A representative release category from each release category group is selected based on their relative contribution to the total frequency of the release category group and consideration of the consequences of the release category. The characteristics of the accident sequences associated with the selected release categories are then used to calculate release category source terms using MAAP 4.0.7.

The source term for a given release category consists of the release fractions for the core radionuclide groups (expressed as fractions of initial core inventory) and the timing of the release.

The following provides additional discussion of the representative MAAP cases.

ST1, Large Early – The scenario includes a loss of all injection, loss of all feedwater, loss of containment sprays and with cooldown starting at 15 minutes into the event. The containment is assumed to be failed with a large area (7 ft^2) at the time of vessel breach. This scenario tends to represent the largest potential consequences, since other variations involve operation of containment sprays with and without successful debris coolability.

ST2, Small Early – The case includes loss of all injection with a pre-existing containment breach and without containment sprays. Other cases within this release category group would either have medium to low RCS pressure and could include water covering the debris outside the vessel. The case selected would tend to represent the highest consequence conditions.

ST3, Late Failure – This representative case includes a seal loss-of-coolant (LOCA) with successful operation of auxiliary feedwater (AFW). Core damage occurs at 3.8 hours followed by vessel breach at 6.1 hours. Containment failure occurs at about 38 hours due to prolonged core-concrete interaction and corresponding pressurization. The general characteristics of a late release include substantial time available for passive removal of fission products within the containment and therefore a significantly reduces source term.

ST4, Containment Bypass – A steam generator turb rupture (SGTR) with successful operation of AFW is selected to represent this release category group.

ST5, Interfacing System LOCA – This is a lower frequency case but with higher consequences. A residual scenario that involves an SGTR without successful AFW and its frequency is added to the ISLOCA frequency.

ST6, Containment Intact – This represents scenarios involving core damage and vessel breach, but without containment breach due to operation of the containment sprays. The source term calculated is limited to that associated with normal design leakage.

For use in the SAMA analysis, the Diablo Canyon DC03 PRA Level 2 release categories were binned into logical groups (release category groups). Table F.2-6 summarizes the results of this binning process.

F.2.2.4.3 Source Term Results

Source term release fractions and release timings are discussed below. Only 30 out of the original 37 release categories have a quantified frequency greater than 1.0E-14 /year.

F.2.2.4.3.1 Source Term Release Fractions

Table F.3-12 presents the source term release fractions for each release category group.

F.2.2.4.3.2 Source Term Release Timings

Table F.3-11 provides for each release category group the summary of key event timings, representative MAAP case, and its description.

F.2.3 PRA MODEL TECHNICAL ADEQUACY FOR SAMA

Pacific Gas & Electric Company (PG&E) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA model for the operating PG&E nuclear generation plant. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Diablo Canyon Power Plant (DCPP) PRA.

F.2.3.1 PRA MAINTENANCE AND UPDATE

The PG&E risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the PG&E risk management program, which consists of a departmental administrative procedure, TS3.NR1 (Reference 94) and an administrative work procedure, AWP E-028

(Reference 95). These procedures delineate the responsibilities and guidelines for updating the full power internal events PRA models at the Diablo Canyon Power Plant. The overall PG&E risk management program defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New procedures and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Equipment unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities for equipment that can have a significant impact on the PRA model are updated approximately every 6 years. The last update was completed in November 2012.

In addition to these activities, PG&E risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal and external events PRA models for the Diablo Canyon Power Plant.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. PRA model updates were also performed in shorter intervals in the past to incorporate design changes, procedure changes and/or newly developed industry data and models that may have a significant impact on the plant risk results. The most recent update of the DCPP PRA model (designated as DC02) was completed in November 2012.

F.2.3.2 PRA SELF ASSESSMENT AND PEER REVIEW

The Diablo Canyon Power Plant PRA models including the Internal Events, Internal Floods, Internal Fires, and Seismic Events, were peer reviewed between 2008 and 2013 against the latest PRA Standard (Reference 80) as endorsed by R.G. 1.200 (Reference 81);

Internal Fire model - December 2010 and January 2008

Internal Events model – December 2012

Internal Floods model – December 2012

Seismic Events model – January 2013

All Findings and Observations (F&Os) identified during the peer reviews of the Internal Fire, Internal Event, and Internal Floods have been closed out, except for two Internal Events PRA related F&Os (HR-E4-01 and HR-G7-01) and one Fire PRA related F&O (F&O SF-A5-01).

The Internal Events PRA F&O HR-E4-01 involves the use of either simulator observations or talk-throughs with operation personnel to verify the PRA plant response models. Simulator observations were performed on March 27, 2014 to address this F&O. This is documented in draft PRA Calc G.2, Revision 7 (Reference 96). This calculation is being reviewed and once it is approved this F&O will be closed-out. This is a documentation issue F&O and has no impact on Model DC03.

The Internal Events PRA F&O HR-G7-01 involves assessing the degree of dependence among potential multiple human actions in the same accident sequence. A detailed Internal Events model human (operators) action dependency was performed and documented in draft PRA Calc G.3, Revision 0 (Reference 97). Changes resulted from this assessment were incorporated to Model DC03. Once this draft calculation is reviewed and approved, this F&O will be closed out.

The Fire PRA F&O AF-A5-01 identified a need for fire brigade training to cope with seismically-induced fires. The resolution of this F&O is being tracked by the DCPP corrective action program to revise a fire brigade training procedure. This is a documentation only finding and has no impact on the Fire PRA model or this analysis.

The update of the seismic fragilities of PRA SSCs is currently in progress. Once the update is completed in 2015, the seismic F&Os will be resolved followed by the update of the Seismic model. As discussed in Section F.2.1.10, the Seismic Events model in DC03 should be considered as preliminary for this analysis.

PG&E also tracks issues/problem/errors associated with the PRA model and plant design and procedure changes that affect the PRA model that were identified by the PRA engineers. These issues were recorded in the PRA Action Tracking Database. Depending on the safety significance of these issues, some of them were either addressed by updating the affected portion(s) of the PRA model immediately or disposed of during the periodic update of the PRA model. Open items that will be addressed in future model updates are listed as Addendum 1 to this report. The potential impacts of these open items on the DCPP SAMA analysis are discussed in Section F.2.3.4.

F.2.3.3 GENERAL CONCLUSION REGARDING PRA CAPABILITY

The DCPP PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in the risk-informed portion of the SAMA licensing actions. The PRA has been updated to reflect design changes, model enhancements, and recent probabilistic seismic hazard curves. As specific risk-informed PRA applications are performed, remaining gaps to specific requirements in the PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

F.2.3.4 ASSESSMENT OF PRA CAPABILITY NEEDED FOR SAMA IDENTIFICATION AND EVALUATION

All F&Os identified through the latest peer reviews of the Internal Events model, Internal Floods model, and Internal Fire Events model, except two Internal Events F&Os and one Fire Events F&O as noted in Section F.2.3.2 were resolved. Any changes to the PRA models and documentation identified as part of the resolution of those F&Os were incorporated into Model DC03. Also the remaining three open F&Os were resolved from technical and/or PRA modeling perspective and reflected in Model DC03. These F&Os will be considered "open" until associated PRA documentation is signed off. As the closure of these F&Os requires only signing-off of draft calculations, it is considered as a documentation issue and has no impact on the PRA capability needed for the SAMA analysis.

The evaluation of the impact of the open items/issues on the above systems, operator actions, etc., are provided in Addendum 1. If the open item/issue is involved with a system or operator action for which a SAMA has been identified, then the open item/issue will have no impact on the conclusion of the SAMA identification process. If an open item/issue does not have a significant impact or has a conservative impact on the risk results (CDF/LERF), then it will have an insignificant impact on the RRW values of the above modeled system, operator actions, etc. Since the RRW values were used to determine the importance of the model events/split fractions with respect to the identification of potential SAMA, the open item/issue will also have an insignificant impact on the SAMA identification process and the conclusion of the process.

F.2.3.5 CONCLUSION REGARDING PRA CAPABILITY FOR SAMA IDENTIFICATION AND EVALUATION

The DCPP PRA model DC03 results are suitable for use as a resource in the SAMA identification process. This conclusion is based on:

- The PRA maintenance and update processes in place
- The PRA technical capability evaluations that have been performed and are being planned
- The SAMA identification process uses the RRW values of PRA model events/split fractions that are associated with system, operator action, etc.

Although the "open items/issues" listed in Addendum 1 will be resolved in future model updates, they have insignificant impact on the conclusion of this process.

F.3 LEVEL 3 RISK ANALYSIS

This section addresses the key input parameters and analysis of the Level 3 portion of the risk assessment. In addition, Section F.7.3 summarizes a series of sensitivity evaluations to potentially critical parameters.

F.3.1 ANALYSIS

The MACCS2 code (Reference 22) version 1.13.1 was used to perform the Level 3 probabilistic risk assessment (PRA) for Diablo Canyon Power Plant (DCPP). The MACCS2 code was developed to support probabilistic risk assessments (Reference 22) and is the standard code used to calculate off-site population dose and economic costs in support of a SAMA analysis, as recognized by the industry guidance document NEI 05-01 (Reference 13) endorsed by the NRC (Reference 26).

For the DCPP analysis, the input parameter values used in NUREG-1150 (Reference 19), as detailed in NUREG/CR-4551 (Reference 20) and reflected in the MACCS2 "Sample Problem A" (Reference 22) formed the initial basis for the present analysis. Where applicable, these generic values were replaced with updated values specific to DCPP and the surrounding area. Site-specific data included population distribution, economic parameters, and meteorological data. Standardized economic parameters from the NUREG-1150 study for the costs of evacuation, relocation and decontamination were escalated from the time of their formulation (1986) to more recent (July 2014) costs. Plant-specific release data included release frequencies and the time-dependent distribution of nuclide releases from six (6) accident sequences at DCPP. The behavior of the population during a release (evacuation parameters) was based on plant and site-specific set points (i.e., declaration of a General Emergency) and evacuation time estimates (Reference 67). These data were used in combination with site specific meteorology to calculate risk impacts (exposure and economic) to the surrounding (within 50 miles) population.

The NRC sponsored the development of the MACCS code as a successor to the CRAC2 code for the performance of commercial nuclear industry probabilistic safety assessments (PSAs). The MACCS code was used in the NUREG-1150 PSA study (Reference 19) in the early 1990's. Prior to being released to the public, MACCS was independently verified by Idaho National Engineering and Environmental Laboratory (Reference 6). The use of the MACCS2 code is consistent with NEI 05-01 (Reference 13), as endorsed by the NRC in LR-ISG-2006-03.4 (Reference 26). The MACCS2 methodology has been employed in numerous applications, including in WASH-1400 (NUREG-75/014, Reactor Safety Study (1975)) (Reference 17) and NUREG-1150 (Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants), for assessing impacts of postulated severe accidents for nuclear power plants. NUREG-1150 (Reference 19) is a seminal work in PRA performed by the NRC and the national laboratories that includes a Level 3 PRA. It was subjected to extensive peer review and has been accepted by the NRC as a standard reference for MACCS2 inputs for SAMA analyses. The MACCS2 code is being used in a slightly updated fashion to support the current state-of-the-art reactor consequence analysis (SOARCA) being performed by the NRC. Moreover, the Gaussian plume model employed in the DCPP MACCS2 analysis is the standard atmospheric plume model used for nuclear safety and environmental evaluations for numerous regulatory applications. It is the underlying radiological dispersion and consequence model underpinning NRC Regulatory Guide 1.194 (Reference 24), Regulatory Guide 1.145 (NUREG/CR-2260) (Reference 18), and DOE-STD-3009-94, Appendix A (Reference 7) for NRC and DOE nuclear safety analyses.

The MACCS2 Gaussian plume model has been shown to provide results that are in good agreement, and generally conservative, when compared with more sophisticated models that address variable meteorological and terrain effects.

Mollenkamp et al. (Reference 12) compared several codes for recorded data in a terrain changing location in the Midwest. This study compared MACCS2 to a fully three-dimensional (3-D) code (which possesses the ability to take into account terrain changes and spatial variability of weather), at a series of one-mile wide arcs at various distances downwind over a distance of 100 miles. The results showed reasonably good agreement obtained with MACCS2 compared to the three-dimensional Lagrangian operational dispersion integrator (LODI) model. The study concluded that, compared to LODI, MACCS2 predicts a more rapid decrease of exposure with distance and that this should be considered for estimating consequences at distances greater than 200 miles. The analysis distance of 50 miles used for the SAMA analysis is well within this range of acceptability based on this code comparison, and in general should provide nominally conservative results (compared to LODI) due to the greater deposition closer to the source.

SAMA analyses require running a significant number of weather trials to obtain robust statistical results. The MACCS2 code is a flat-earth Gaussian plume models that can meet the computational demands of calculating many kinds of consequence results, with the appropriate level of statistical sampling. In contrast, computer codes that can accommodate multiple-station data so as to be able to model spatial variation of wind speed and direction (e.g., mesoscale models) and thus provide regional consequences would be impractical for analyzing the large number of weather sequences needed for SAMA analyses.

MACCS2 is generally unique among currently supported dispersion analysis modeling codes for its ability to address three required elements for SAMA analysis, that being (1) atmospheric dispersion of releases, (2) prompt and long term radiological health impacts (e.g., population dose), and (3) economic impacts. Various dispersion codes are available for analyzing chemical releases, such as CALPUFF & AERMOD which are preferred codes for EPA purposes, but these codes are not able to address the second and third elements of radiological health impacts and economic impacts needed to support a SAMA evaluation. Similarly, a variety of other dispersion codes exist for use for emergency planning purposes in the nuclear field, but these codes do not address long term radiological health impacts or prompt or long term costs. Only MACCS2 addresses the three required elements to support the regional analysis for cost-benefit considerations. For these reasons, MACCS2 is the only code specifically named as an acceptable methodology for estimating environmental consequences of severe accident analysis in the NRC Environmental Standard Review Plan, NUREG-1555 (Reference 27).

Regarding terrain variability, it is noted that the DCPP site is surrounded on the land portion by complex mountainous type terrain. The DCPP FSAR (Reference 52) Section 2.3.6 summarizes the investigation of the impacts of the local geography on airborne releases as follows:

> Despite the prevalence of the marine inversion and the northwesterly wind flow gradient along the California coast in the dry season, the long-term accumulation of plant emissions, released routinely or accidentally, in any particular geographical area downwind from the plant is virtually impossible. Pollutants injected into the marine inversion layer of the coastal wind regime are transported and dispersed by a complex array of land-sea breeze regimes that exist all along the coast wherever canyons or valleys indent the coastal range. Because of the complexities of the wind circulation in these regimes and their fundamental diurnal nature, the net result is a very effective and wide daily dispersal of any pollutants that are present in the marine coastal air.

The MACCS2 analysis performed for SAMA seeks to evaluate the average (i.e., mean) consequences associated with a radiological release. While the uncertainty band associated with the consequence results for any single release could be postulated to be greater due to the complex terrain (e.g., a mountain valley channeling a release to a population center), the effect on the mean consequence results is expected to be minor.

Additionally, it is noted that the complex mountainous terrain would be expected to increase the amount of deposition close to the site due to impaction and potentially reduce the radiological material reaching population centers located further from the site.

Based on the above considerations, the use of the Gaussian model implemented in MACCS2 is judged acceptable for the development of dose risk and cost risk inputs into SAMA.

F.3.2 POPULATION

The population surrounding the DCPP site is estimated for the year 2045, the last year of projected operation for Unit 2 given a 20 year license extension.

The population distribution projection was based on year 2010 census data available via SECPOP 4.2 (Reference 23). The baseline resident year 2010 population from SECPOP 4.2 was determined for each of 160 grid elements of a polar coordinate grid consisting of sixteen directions (i.e., N, NNE, NE,...NNW) for each of ten concentric distance rings with outer radii at 1, 2, 3, 4, 5, 10, 20, 30, 40 and 50 miles surrounding the site. Transient population data (including employee population) from the DCPP Evacuation Time Estimate (ETE) study (Reference 67) for the approximate 20 mile radial area around the site was added to the SECPOP permanent population, on a grid element basis. In addition to the ETE category of transient population, special facilities populations were also included in the initial year 2010 population estimate. Including transient population from 20-50 miles would be overly conservative for this analysis since transients are treated as permanent residents in the MACCS2 model. These additional transients, if impacted by a postulated release, would accrue costs such as per-diem housing that would be inappropriate since these transients would return to their residences outside the impacted area. Inclusion of transients in the wider region could also lead to double counting if the transient permanently resides elsewhere in the 50-mile region.

To estimate growth rates for the permanent and transient (and special facilities) populations, California county population projection data was obtained from the California Department of Finance (Reference 92). Table F.3-1 presents the county growth rates for the years 2010 to 2045. Individual growth rates were calculated for each grid element based on the county growth rate and the proportion of land in each grid element associated with the applicable counties. The combined resident and transient data (including special facilities) were projected to the year 2045.

Table F.3-2 presents the year 2010 transient and special facility population within 20 miles of DCPP. Table F.3-3 presents the year 2010 residential population within 50 miles of DCPP from SECPOP 4.2.

The total year 2045 population for the 160 grid elements in the region is estimated at 707,417. The distribution of the population is given for the 20-mile radius and the 50-mile radius from DCPP in Tables F.3-4 and F.3-5, respectively.

F.3.3 ECONOMY

MACCS2 requires certain agricultural and land based economic data (fraction of land devoted to farming, annual farm sales, fraction of farm sales resulting from dairy production, and property value of farm and non-farm land) for each of the 160 grid elements. This data can be generated by SECPOP 4.2 (Reference 23), but due to outdated economic information contained in the code (i.e., the 2007 Census of Agriculture), the code was not utilized to develop the county specific economic values for the DCPP analysis. Instead, the economic values were developed manually following the SECPOP calculation approach documented in NUREG/CR-6525 (Reference 23) using data from the 2012 National Census of Agriculture (Reference 63) and 2012 data (for consistency with the census of agricultural data) from the Bureau of Economic Analysis (Reference 2) for each of the four (4) counties surrounding the plant, to a distance of 50 miles. Economic values were updated to July 2014 using the consumer price index (CPI) from the Bureau of Labor Statistics (Reference 64). The values used for each of the 160 grid elements were the data from each of the surrounding counties multiplied by the fraction of that county's area that lies within that sector. Region-wide wealth data (i.e., farm wealth and non-farm wealth) were based on county-weighted averages for the region within 50-miles of the site using the same economic data sources noted above along with additional data from the U.S. Census Bureau (Reference 88 and 86), the U.S. National Resources Conservation Service (Reference 87), and other sources (Reference 91). Spatial elements within the same county have the same index value. Spatial elements involving multiple counties have unique index values. The portion of each county within 50-miles of the site was accounted for in the calculation. The fraction of each spatial element that is land (as opposed to water) was visually estimated using maps and images of the regions surrounding DCPP and was also taken into consideration. Region index values were assigned based on application of the county level data to a 50-mile radius grid surrounding each site. Data from the 2012 Census of Agriculture (Reference 63) was used to determine the farmland fraction for each of the counties surrounding DCPP. County specific land use and related economic parameter values are summarized in Table F.3-6.

In addition, generic economic data that is applied to the region as a whole were revised from the NUREG-1150 based data in order to account for cost escalation since 1986, the year that input was first specified. A factor of 2.17, representing cost escalation from 1986 (CPI index of 109.6) to July 2014 (CPI index of 238.3) was applied to parameter values describing cost of evacuating and relocating people and decontamination activities.

MACCS2 generic economic parameter values utilized in the DCPP analysis are summarized in Table F.3-7.

F.3.4 FOOD AND AGRICULTURE

Food ingestion is modeled using the new MACCS2 ingestion pathway model COMIDA2, consistent with MACCS2 User's Guide (Reference 22). The COMIDA2 model utilizes national based food production parameters derived from the annual food consumption of an average individual such that site specific food production values are not utilized. Annual dose limits trigger crop or milk disposal, as appropriate. Values are chosen consistent with the most recent guidance of FDA 63 FR-43402 (Reference 89). These parameters and their values used in the DCPP analysis are presented in Table F.3-8. The fraction of population dose due to food ingestion is typically small compared to other population dose sources. For DCPP, approximately 1% of the total population dose is due to food ingestion.

All spatial elements are designated as river systems or ocean. Per NUREG/CR-4551 the designation of lake is only used for very large bodies of water, such as Lake Michigan, which may serve as drinking water sources. The lakes around the Diablo Canyon site are smaller and are expected to behave like river systems.

F.3.5 NUCLIDE RELEASE

The core inventory at the time of the accident is based on a plant specific evaluation and corresponds to 611 effective full power days (end-of-cycle) for DCPP operating at 3411 MWt, the current licensed value. Table F.3-9 summarizes the estimated DCPP core inventory (Reference 59) used in the MACCS2 analysis. This core inventory calculation reflects the current and anticipated Diablo Canyon fuel management / burnup approach.

DCPP nuclide release categories, as determined by the MAAP computer code, are related to the MACCS2 categories as shown in Table F.3-10. Releases were modeled as occurring at the top of the reactor building (67 meters). Note that minor adjustment to the containment height based on site grade changes has negligible impact on results. The thermal content of each of the releases was assumed to be the same as ambient, i.e., buoyant plume rise was not modeled. Each of these assumptions was considered in sensitivity analyses, presented in Section F.7.3.

Release frequencies, nuclide release fractions (of the core inventory), shown in Table F.3-12, and the time distribution of the release were analyzed to determine the sum of the exposure (50-mile dose) and economic (50-mile economic costs) risks from six (6) accident sequences. Each accident sequence was chosen to represent a set of similar accidents. Representative MAAP cases for each of the release categories were chosen based on a review of the Level 2 model cutsets and the dominant types of scenarios that contributed to the results. A brief description of each of those MAAP cases is provided in Table F.3-11, and a summary of the release magnitude and timing for those cases is provided in Table F.3-12. Multiple release duration periods (i.e., plume segments) were defined which represent the time distribution of each category's releases.

A dry deposition velocity of 0.01 m/sec is used for the MACCS2 analysis, consistent with NRC recommendation as documented in the MACCS2 Sample Problem A (Reference 22). The dry deposition velocity is evaluated in the sensitivity analysis, presented in Section F.7.3.

F.3.6 EVACUATION

Reactor trip for each sequence was taken as time zero relative to the core containment response times. A General Emergency (GE) is declared when plant conditions degrade to the point where it is judged that there is a credible risk to the public. For the DCPP analysis the time of the GE declaration was estimated based on the DCPP emergency action levels (Reference 60). The declaration times are presented in Table F.3-12.

The MACCS2 User's Guide input parameters of 95% of the population within 10 miles of the plant evacuating and 5% not evacuating were employed. These values are conservative relative to the NUREG-1150 study (Reference 19), which assumed evacuation of 99.5% of the population within the Emergency Planning Zone (EPZ).

The time to begin evacuation and the base speed are derived from the DCPP Evacuation Time Estimate (ETE) study (Reference 67). The evacuees are assumed to begin evacuation 100 minutes after a general emergency has been declared at a base evacuation radial speed of 0.76 m/sec. A time of approximately 7.3 hours is used to model evacuation of the 10+ mile EPZ (Region R11 of the ETE), based on weighting the ETE times accounting for season (i.e., winter vs. summer), time of the week (i.e., midweek vs. weekend), time of day (i.e., daytime vs. nighttime), and weather conditions (i.e., good vs. rain). Special events (i.e., 4th of July fireworks show) have been considered for the Diablo Canyon ETE and are included in the time estimate. An additional 15 minutes is added to the ETE evacuation times to account for processing time by offsite officials.

The ETE study evacuation times range from 6 hrs 45 minutes (winter evening good conditions) to 13 hrs 20 minutes (for 4th of July fireworks show) for a 100% evacuation of the 10+ mile EPZ (Region R11). These ETE times include "shadow evacuation" of 20% of the residential population outside the 10+ mile EPZ, to a distance of more than 20 miles. The ETE study evacuation times are based on year 2010 census data. Transportation infrastructure improvements are assumed to generally keep pace with population growth over the SAMA projection period (i.e., to 2045). The evacuation parameters were considered further in the sensitivity analyses presented in Section F.7.3. The sensitivity analyses include evaluation of a slower evacuation speed and no evacuation to provide reasonably bounding results.

Shielding and exposure factors are chosen consistent with those developed and used in the NUREG-1150 (Reference 19) and NUREG/CR-4551 (Reference 20) study for the Surry site which represent mid-range values.

F.3.7 METEOROLOGY

Annual hourly meteorology DCPP data sets from 2002 through 2006 were investigated for use in the MACCS2 analysis. Of the hourly data of interest (10-meter wind speed, 10-meter wind direction, multi-level temperatures used to calculate stability class, and precipitation), year 2003 and 2005 data had a significant number of data voids compared to the other years of data and were therefore not finalized. Traditionally, up to 10% of missing data is considered acceptable. MACCS2 requires complete sequential hourly data, therefore missing data must be estimated. Data gaps were filled by (in order of preference): interpolation (if the data gap was less than 6 hours), or using data from the same hour and a nearby day (substitution technique). The 10-meter wind speed and direction were combined with precipitation and atmospheric stability (derived from the vertical temperature gradient) to create the hourly data file for use by MACCS2. Precipitation data was derived from the California Irrigation Management Information System (CIMIS). Site 52 located at California Polytechnic State University in San Luis Obispo (35.31N, -120.66W) about 13 miles from DCPP served as the primary precipitation data site. Site 160 located (35.34N, -120.73W) about 11 miles from DCPP served as a secondary precipitation data site to fill in missing data.

The 2002 meteorological data set was found to result (see Section F.7.3 for discussion of sensitivity analysis) in the largest economic cost risk and dose risk compared to the 2004 and 2006 data sets, and the initial 2002 data set had an acceptable amount of data voids (about 5%). Therefore, the 2002 hourly meteorology was selected as the base case.

Atmospheric mixing heights were specified for AM and PM hours for each season of the year based on data from Holzworth (Reference 8). These values ranged from 500 meters to 900 meters.

F.3.8 MACCS2 RESULTS

Table F.3-13 shows the mean off-site doses and economic impacts to the region within 50 miles of DCPP for each of six (6) release categories calculated using MACCS2. The mean off-site dose impacts are multiplied by the annual frequency for each release category and then summed to obtain the dose-risk and offsite economic cost-risk (OECR) for each unit. Table F.3-13 provides these results.

F.4 BASELINE RISK MONETIZATION

This section explains how DCPP calculated the monetized value of the status quo (i.e., accident consequences without SAMA implementation). DCPP also used this analysis to establish the maximum benefit that could be achieved if all on-line DCPP risk were eliminated, which is referred to as the Maximum Averted Cost-Risk (MACR). Per the site PRA model (designated DC03), the CDF for internal events (including internal flooding), fire, and seismic contributors of 8.64E-05/year was used for the calculations in the following sections. Non-fire/non-seismic external events risk is addressed in Section F.4.6.2.

The Unit 1 results, which are larger than the Unit 2 results, are considered to be representative of the risk for each unit (see Sections F.2.1 and F.2.1.10) and they are used in the SAMA cost benefit calculations.

F.4.1 OFF-SITE EXPOSURE COST

The baseline annual off-site exposure risk was converted to dollars using the NRC's conversion factor of $2,000 per person-rem, and discounted to present value using NRC standard formula (Reference 21):

W_{pha} = $C \times Z_{pha}$

Where:

W_{pha} = monetary value of public health accident risk after discounting

C = $[1-\exp(-rt_f)]/r$

t_f = years remaining until end of facility life = 20 years

r = real discount rate (as fraction) = 0.03 per year

Z_{pha} = monetary value of public health (accident) risk per year before discounting ($ per year)

The Level 3 analysis showed an annual off-site population dose risk of 98.89 person-rem. The calculated value for C using 20 years and a 3 percent discount rate is approximately 15.04. Therefore, calculating the discounted monetary equivalent of accident dose-risk involves multiplying the dose (person-rem per year) by $2,000 and by the C value (15.04). The calculated off-site exposure cost is $2,974,534.

F.4.2 OFF-SITE ECONOMIC COST RISK

The Level 3 analysis showed an annual off-site economic risk of $246,912. Calculated values for off-site economic costs caused by severe accidents must be discounted to present value as well. This is performed in the same manner as for public health risks and uses the same C value. The resulting value is $3,713,461.

F.4.3 ON-SITE EXPOSURE COST RISK

Occupational health was evaluated using the NRC recommended methodology that involves separately evaluating immediate and long-term doses (Reference 21).

For immediate dose, the NRC recommends using the following equation:

Equation 1:

$$W_{IO} = R\{(FD_{IO})_S - (FD_{IO})_A\}\{[1 - \exp(-rt_f)]/r\}$$

Where:

W_{IO} = monetary value of accident risk avoided due to immediate doses, after discounting

R = monetary equivalent of unit dose ($2,000 per person-rem)

F = accident frequency (events per year) (8.64E-05 (internal events, fire, and seismic CDF))

D_{IO} = immediate occupational dose [3,300 person-rem per accident (NRC estimate)]

$_S$ = subscript denoting status quo (current conditions)

$_A$ = subscript denoting after implementation of proposed action

r = real discount rate (0.03 per year)

t_f = years remaining until end of facility life (20 years).

Assuming F_A is zero, the best estimate of the immediate dose cost is:

W_{IO} = $R\,(FD_{IO})_S\,\{[1 - \exp(-rt_f)]/r\}$

= $2{,}000*8.64E\text{-}05*3{,}300*\{[1 - \exp(-0.03*20)]/0.03\}$

= $8,576

For long-term dose, the NRC recommends using the following equation:

Equation 2:

W_{LTO} = $R\{(FD_{LTO})_S - (FD_{LTO})_A\}\,\{[1 - \exp(-rt_f)]/r\}\{[1 - \exp(-rm)]/rm\}$

Where:

W_{LTO} = monetary value of accident risk avoided long-term doses, after discounting, $

D_{LTO} = long-term dose [20,000 person-rem per accident (NRC estimate)]

m = years over which long-term doses accrue (as long as 10 years)

Using values defined for immediate dose and assuming F_A is zero, the best estimate of the long-term dose is:

W_{LTO} = $R\,(FD_{LTO})_S\,\{[1 - \exp(-rt_f)]/r\}\,\{[1 - \exp(-rm)]/rm\}$

= $2{,}000*8.64E\text{-}05*20{,}000*\{[1 - \exp(-0.03*20)]/0.03\}\,\{[1 - \exp(-0.03*10)]/(0.03*10)\}$

= $44,905

The total occupational exposure is then calculated by combining Equations 1 and 2 above. The total accident related on-site (occupational) exposure risk (W_O) is:

W_O = $W_{IO} + W_{LTO}$ = ($8,576+$44,905) = $53,481

F.4.4 ON-SITE CLEANUP AND DECONTAMINATION COST

The total undiscounted cost of a single event in constant year dollars (C_{CD}) that NRC provides for cleanup and decontamination is \$1.5 billion (Reference 21). The net present value of a single event is calculated as follows. NRC uses the following equation to integrate the net present value over the average number of remaining service years:

PV_{CD} = $[C_{CD}/mr][1-\exp(-rm)]$

Where:

PV_{CD} = net present value of a single event

C_{CD} = total undiscounted cost for a single accident in constant dollar years (\$1.5B)

r = real discount rate (0.03)

m = years required to return site to a pre-accident state (10 years)

The resulting net present value of a single event is \$1.3E+09. The NRC uses the following equation to integrate the net present value over the average number of remaining service years:

U_{CD} = $[PV_{CD}/r][1-\exp(-rt_f)]$

Where:

PV_{CD} = net present value of a single event (\$1.3E+09)

r = real discount rate (0.03)

t_f = 20 years (license renewal period)

The resulting net present value of cleanup integrated over the license renewal term, \$1.95E+10, must be multiplied by the CDF (8.64E-05) to determine the expected value of cleanup and decontamination costs. The resulting monetary equivalent is \$1,683,933.

F.4.5 REPLACEMENT POWER COST

Long-term replacement power costs were determined following the NRC methodology in NUREG/BR-0184 (Reference 21). The net present value of replacement power for a single event, PV_{RP}, was determined using the following equation:

$$PV_{RP} = [\$1.2\times10^8/r] * [1 - \exp(-rt_f)]^2$$

Where:

PV_{RP} = net present value of replacement power for a single event, ($814M)

r = 0.03

t_f = 20 years (license renewal period)

To attain a summation of the single-event costs over the entire license renewal period, the following equation is used:

$$U_{RP} = [PV_{RP} /r] * [1 - \exp(-rt_f)]^2$$

Where:

U_{RP} = net present value of replacement power over life of facility (5.53B $-year)

After applying a correction factor to account for Diablo Canyon's size relative to the "generic" reactor described in NUREG/BR-0184 (i.e., 1180 megawatt electric / 910 megawatt electric), the replacement power costs are determined to be 7.16E+09 ($-year). Multiplying 7.16E+09 ($-year) by the CDF (8.64E-05) results in a replacement power cost of $619,048.

F.4.6 MAXIMUM AVERTED COST-RISK

The DCPP MACR is the total averted cost-risk if all internal and external events risk associated with on-line operation were eliminated. This is calculated by summing the following components:

- Maximum Internal Event (with Internal Flooding), Fire, and Seismic Averted Cost-Risk
- Maximum Non-Fire/Non-Seismic External Events Averted Cost-Risk

As described in Section F.5.1 the MACR is used in the Phase I analysis as a means of screening SAMAs. The following subsections provide a description of how each of these components is calculated and used together to obtain the DCPP MACR.

F.4.6.1 INTERNAL EVENTS, FIRE, AND SEISMIC MAXIMUM AVERTED COST-RISK

Because the DCPP PRA in an integrated model that includes the internal event (with internal flooding), fire, and seismic contributors, the PRA results can be used to directly account for these event types in the MACR calculation. The maximum internal events, fire, and seismic averted cost-risk is the sum of the contributors calculated in Sections F.4.1 through F.4.5:

Maximum Averted Internal Events Cost-Risk

Off-site exposure cost	$2,974,534
Off-site economic cost	$3,713,461
On-site exposure cost	$53,481
On-site cleanup cost	$1,683,933
Replacement power cost	$619,048
Total cost (per unit)	$9,044,457

This total represents the per unit monetary equivalent of the risk that could be eliminated if all risk associated with on-line internal events, internal flooding, fire, and seismic hazards could be eliminated for DCPP.

F.4.6.2 NON-FIRE/NON-SEISMIC EXTERNAL EVENTS MAXIMUM AVERTED COST-RISK

The maximum averted cost-risk for the non-fire/non-seismic external events must be quantified for the cost benefit calculations; however, this cost-risk must be estimated based on information in the IPEEE given that complete, current, quantifiable external events models are not available for these types of events. As a result, an alternate method of accounting for the external events contributions must be established.

The method chosen to account for non-fire/non-seismic external events contributions in the SAMA analysis is to use a multiplier on the PRA results (which include internal events, fire events, seismic events, and internal flooding events). In previous SAMA analyses, it has been assumed that the risk posed by external events and internal events is approximately equal. This assumption is not unreasonable unless available analyses indicate that there are external events contributors that present a disproportionate risk to the site. Because the DCPP PRA model includes fire and seismic results, the external events contributors that typically comprise the large majority of external events at nuclear power plants are not required to be addressed by a multiplier. In this case, the non-fire/non-seismic external events are relatively small contributors to risk such that doubling the internal events risk may misrepresent these contributors. Hence, development of a multiplier representative of the non-fire/non-seismic external risk was deemed necessary.

This is the ratio of total CDF (including internal and all external contributors) to only the internal, fire, and seismic CDF. This ratio is called the non-fire/non-seismic External Events (EE) multiplier and its value is calculated as follows:

EE Multiplier = $(8.64^{E}\text{-}05+2.56^{E}\text{-}06) / (8.64^{E}\text{-}05) = 1.03$

The contributions of the non-fire/non-seismic external events initiators are summarized in the following table:

Non-Fire/Non-Seismic IPEEE Contributor Summary External Event Initiator Group CDF

High Winds	3.20E-07
Transportation & Nearby Facility – ship impact	1.90E-08
Transportation & Nearby Facility – accidental aircraft impact	7.00E-07
External Flooding	7.20E-07
Chemical Release	8.00E-07
Total EE CDF	**2.56E-06**

F.4.6.3 DCPP MAXIMUM AVERTED COST-RISK

The total DCPP MACR can be calculated by multiplying the MACR for internal events, internal flooding, fire, and seismic contributors from Section F.4.6.1 by the EE multiplier established in Section F.4.6.2:

DCPP MACR = 9,044,457 * 1.03 = 9,315,791

The MACR and implementation costs are considered on a per-unit scale for consistency (unless otherwise noted). Any "economy of scale" that may exist in the implementation costs have been accounted for given that the implementation costs were originally developed on a site basis and then divided by 2 for use in the net value calculations.

F.5 PHASE 1 SAMA ANALYSIS

The Phase 1 SAMA analysis, as discussed in Section F.1, includes the development of the initial SAMA list and a coarse screening process. This screening process eliminated those candidates that are not applicable to the plant's design or are too expensive to be cost beneficial even if the risk of on-line operations were completely eliminated. The following subsections provide additional details of the Phase 1 process.

F.5.1 SAMA IDENTIFICATION

The initial list of SAMA candidates for DCPP was developed from a combination of resources. These include the following:

- DCPP Unit 1 PRA results and PRA Group Insights (representative of Unit 2 results)
- Industry Phase 2 SAMAs (review of potentially cost effective Phase 2 SAMAs from selected plants)
- DCPP Individual Plant Examination IPE (Reference 33)
- DCPP IPEEE (Reference 37)

These resources provide a list of potential plant changes that are most likely to reduce risk in a cost-effective manner for DCPP.

In addition to the "Industry Phase 2 SAMA" review identified above, an industry based SAMA list was used in a different way to aid in the development of the DCPP plant specific SAMA list. While the industry Phase 2 SAMA review cited above was used to identify potential SAMAs that might have been overlooked in the development of the DCPP SAMA list due to PRA modeling issues, a generic SAMA list was used to help identify the types of changes that could be used to address the areas of concern identified through the DCPP importance list review. For example, if Instrument Air availability was determined to be an important issue for DCPP, the industry list would be reviewed to determine if a plant enhancement had already been conceived that would address DCPP's needs. If an appropriate SAMA was found to exist, it would be used in the DCPP list to address the Instrument Air issue; otherwise, a new SAMA would be developed that would meet the site's needs. This generic list was compiled as part of the development of multiple industry SAMA analyses and is available in NEI 05-01 (Reference 13).

It should be noted that the process used to identify DCPP SAMA candidates focuses on plant specific characteristics and is intended to address only those issues important to the site. In this case, the existing capabilities of the plant preclude the need to include many of the potential SAMAs that have been identified for other PWRs. As a result, because of the effectiveness of past plant reviews and subsequent modifications, the types of changes that might be cost effective for DCPP are reduced and the SAMA list is relatively short.

F.5.1.1 LEVEL 1 DCPP IMPORTANCE LIST REVIEW

The DCPP PRA was used to generate a list of model events (split fractions of the PRA model) sorted according to their risk reduction worth (RRW) values with respect to CDF. The events with the largest RRW values in this list are those events that would provide the greatest reduction in the CDF if the failure probability were set to zero. Because a PRA's importance list can be extensive, it is desirable to limit the review to only those contributors that could yield potentially cost beneficial results. One method that can be used to limit the scope of the importance list review is to correlate the RRW value threshold to the lowest expected cost of implementation for a SAMA. Usually, operator actions in the form of procedure changes are among the least expensive enhancements that can be made at a site, so they are often used as the representative "lowest cost SAMA". However, because the cost of performing a procedure change can vary by orders of magnitude depending on the scope of the change and the procedure that is being changed, this does not provide a clear basis for a review threshold. In addition, the use of this type of a threshold can lead to a review process that is beyond the scope of what is described in NEI 05-01.

The NEI 05-01 guidance describes the SAMA identification process in Section 5.1 as a process to "identify plant-specific SAMA candidates by reviewing dominant risk contributors (to both CDF and population dose) in the Level 1 and Level 2 Probabilistic Safety Assessment (PSA) models." Section 5.1 indicates that the definition of the dominant contributors is open to interpretation, but the guidance does not imply that the identification process should represent an exhaustive search for all plant enhancements that could be cost-beneficial. For example, some minor plant procedure changes could be very inexpensive, but the SAMA identification process should not be defined as one that requires a review all events that could yield averted cost-risks that are greater than the cost of such a procedure change.

Because there is not a universal definition for “dominant risk contributors”, an attempt has been made in this analysis to characterize “dominant contributors” and to establish a review threshold that can reasonably be considered to address them.

The ASME/ANS PRA Standard (Reference 79) includes a definition of “significant” contributors to risk, but it is described in quantitative terms related to the percentages of risk represented, and the guidance does not provide many qualitative insights about the nature of “significant contributors”. In general, the term “dominant” suggests something that is ruling, governing, or in a commanding position, which does not appear to be consistent with a “risk significant” basic event or accident sequence. For example, a risk significant basic event is one with a Fussell-Vesely (FV) value of 0.005 or greater, which corresponds to an event that would reduce the CDF by 0.5% if it were made completely reliable. Events contributing only 0.5% to the CDF could not reasonably be described as “governing” or “ruling” the risk profile.

For the SAMA analysis, the threshold of a dominant basic event is considered to be a factor of 10 larger than for a risk significant event. Similarly, the threshold for a dominant individual accident sequence is considered to be an order of magnitude larger than the value of 1% defined in the ASME/ANS PRA Standard for risk significant accident sequences. The definitions of the “dominant” basic events and accident sequences are assumed to be:

- Dominant Basic Events are those events with FV values greater than or equal to 0.05 (or Risk Reduction Worth values of about 1.05 or greater) for the relevant figure of merit (e.g., CDF).
- Dominant Individual Accident Sequences are those which contribute 10 percent or more to the relevant figure of merit (e.g., CDF).

In the DCPP PRA model, there are only five (5) split fractions with RRW values greater than 1.05, and these are considered to represent the dominant contributors. However, split fractions with RRW values of 1.01 or greater were reviewed as part of the analysis and the results have been included to make the review more robust. Table F.5-1 documents the disposition of each split fraction in the Level 1 PRA model with an RRW value of 1.01 or greater. When the impacts of the non-fire/non-seismic external events are considered, this corresponds to an event that would reduce the cost-risk by about $100,000 if it were made completely reliable. Viewed from another perspective, a RRW value of 1.01 corresponds to a CDF reduction of about 1% assuming the split fraction failure probability were set to zero. For a total CDF of 8.90E-05 /yr (internal events and external events), this corresponds to a potential CDF reduction of about 9.0E-07 /yr. Such a change in CDF is below the widely accepted threshold in Region III of Figure 4 in Regulatory Guide 1.174 (Reference 93) of what constitutes a "very small change" (less than 1E-6 /yr).

The Unit 1 importance list review was performed to identify the failure scenarios important to DCPP risk and to develop methods to mitigate those scenarios. For each event on the importance list, the reasons for the event's importance are determined through sequence and systems analysis. Strategies to mitigate the relevant failures are developed based on accident sequence review, plant knowledge, and industry insights. For DCPP, the PRA model was used to develop a Level 1 importance list that accounts for fire, seismic, and internal events. Note that the review of each split fraction involves an evaluation of the sequences including the split fraction to identify the factors that make the split fraction important.

F.5.1.2 LEVEL 2 DCPP IMPORTANCE LIST REVIEW

A similar review was performed on the importance listings from the Level 2 results. In this case, the development of a composite importance file based on the following release categories was considered to identify potential SAMAs:

- ST1 (Large Early)
- ST2 (Small Early)
- ST5 (ISLOCA)

This method was chosen to prevent high frequency-low consequence events (e.g., from the "Late" release category, ST3) from biasing the importance listing. These three release categories included in the review account for about 97 percent of the dose-risk while accounting for only about 20 percent of the Level 2 frequency. Exclusion of the other results from the Level 2 review allows the contributors that are most important to dose-risk and cost-risk to rise to the top of the importance list.

Further grouping of the release categories was required given that the consequences of the ST2 release category are low relative to those for ST1 and ST5. A separate importance list was developed for ST2 to ensure that its contributors could be reviewed without masking the important events in the ST1 and ST5 release categories.

The Level 2 split fractions were also reviewed down to the 1.01 level.

Tables F.5-2a and F.5-2b document the disposition of each split fraction in the Level 2 RRW lists with RRW values greater than 1.01.

It should be noted that the DCPP Severe Accident Mitigation Guidelines provide further actions to mitigate and recover from severe accidents. The types of actions proceduralized include spraying and/or flooding the containment breakpoint to reduce airborne releases, using a fire truck to provide a pumping source for steam generator makeup (or for spraying containment), starting the EDGs without a DC power source, flooding containment to provide core debris cooling/release scrubbing, etc. These types of strategies are not included as SAMAs because they are already implemented at the site.

F.5.1.3 Industry SAMA Review

The SAMA identification process for DCPP is primarily based on the PRA importance listings, the IPE, and the IPEEE. In addition to these plant-specific sources, selected industry SAMA submittals were reviewed to identify any Phase II SAMAs that were determined to be potentially cost beneficial at other plants. These SAMAs were further analyzed and included in the DCPP SAMA list if they were considered to address potential risks not identified by the DCPP importance list review.

While many of the industry SAMAs reviewed are ultimately shown not to be cost beneficial, some are close contenders and a small number have been estimated to be cost beneficial at other plants. Use of the DCPP importance ranking should identify the types of changes that would most likely be cost beneficial for DCPP, but review of selected industry Phase II SAMAs may capture potentially important changes not identified for DCPP due to PRA modeling differences or SAMAs that represent alternate methods of addressing risk. Given this potential, it was considered prudent to include a review of selected industry Phase II SAMAs in the DCPP SAMA identification process.

Phase II SAMAs from the following United States nuclear power sites have been reviewed:

- Susquehanna (Reference 62, Reference 69)
- Shearon Harris (Reference 5, Reference 70)
- H.B. Robinson (Reference 4, Reference 71)
- Point Beach (Reference 15, Reference 72)
- Prairie Island (Reference 16, Reference 73)
- Wolf Creek (Reference 65, Reference 74)
- Grand Gulf (Reference 76, Reference 77)
- Seabrook (Reference 78, Reference 79)

Two General Electric BWR and six Westinghouse PWR sites were chosen from available documentation to serve as the potential Phase 2 SAMA sources. Many of the industry Phase 2 SAMAs were already represented by other SAMAs in the DCPP list, were known not to impact important plant systems or be relevant to the DCPP design, or were judged not to have the potential to be close contenders for DCPP. As a result, they were not added to the DCPP SAMA list. If there were any unique SAMAs that were considered to have the potential to be cost effective for DCPP, they were added to the list. The potentially cost effective SAMAs for each of the sites identified above are reviewed in the following subsections.

F.5.1.3.1 Susquehanna Steam Electric Station

Review of SSES Cost Beneficial SAMAs

Industry Site SAMA ID	SAMA Description	Discussion for DCPP	Disposition for DCPP SAMA List
2a	Improve Cross-Tie Capability Between 4KV AC Emergency Buses (A-D, B-C)	SSES did not credit cross-tie between EDG trains and relied on the swing EDG to mitigate EDG failures. DCPP hardware and procedures provide the capability to cross-tie any of the vital 4KV buses, including the vital buses from the opposite unit. The PRA model conservatively does not credit the inter-unit cross-tie capability.	Not included – already implemented.
6	Procure Spare 480V AC Portable Station Generator	This SAMA was developed to address the hardware failure contribution from their existing portable 480V generator. A form of the portable generator SAMA is included on the DCPP list (SAMA 12), but the SAMA is expanded to meet the site specific needs for SBO mitigation.	Already included.
2b	Improve Cross-Tie Capability Between 4KV AC Emergency Buses (A-BC-D)	This SAMA is an enhancement over SSES SAMA 2a and allows cross-tie between any EDG division. See explanation provided above for SAMA 2a.	Not included – already implemented.
3	Proceduralize Staggered RPV Depressurization When Fire Protection System Injection is the Only Available Makeup Source	This SAMA is specific to the SSES site and is based on the need to split flow from a single injection system between units. It is not applicable to the DCPP design.	Not included – not applicable to DCPP.
5	Auto Align 480V AC Portable Station Generator	This SAMA was designed for a plant that already had a portable generator. For DCPP, the generator would support the 125V DC battery chargers, but because the battery life is estimated to be 12 hours, ample time would be available to align the system and the incremental benefit associated with auto alignment is considered to be minimal.	Not included – No significant risk benefit.