History DCPP PRA Model Development and Update

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The current DCPP PRA model is based on the original 1988 Diablo Canyon PRA (DCPRA-1988) model [Reference 1] that was performed as part of the Long-Term Seismic Program (LTSP) [Reference 2]. The DCPRA-1988 was a full-scope Level 1 PRA that evaluated internal and external events. The NRC reviewed the LTSP and issued Supplement No. 34 to NUREG-0675 [Reference 3] in June 1991, accepting the DCPRA-1988. Brookhaven National Laboratory (BNL) performed the primary review of the DCPRA-1988 for the NRC; their review is documented in NUREG/CR-5726 [Reference 4].

The original design of the NSSS and BOP systems of Unit 2 is identical to that of Unit 1. The consistency in design and operation of both units has been maintained. The difference between two units in terms of their design, operation, equipment reliability and availability, was minor and did not warrant a development of a separate PRA model for each unit. The results and insights of Unit 1 PRA model should directly applicable to Unit 2 for most applications.

The DCPRA-1988 was subsequently updated to support the Individual Plant Examination (IPE) in 1991 and the Individual Plant Examination for External Events (IPEEE) in 1993. Since 1993, several other updates have been made to incorporate plant and procedure changes, update plant-specific reliability and unavailability data, improve the fidelity of the model, incorporate Westinghouse Owners Group (WOG) Peer Review comments [Reference 5], and support other applications, such as On-line Maintenance, Risk-Informed In-Service Inspection (RI-ISI), Emergency Diesel Generator Completion Time Extension (EDG CTE), and Mitigating System Performance Index (MSPI).

The DCPRA model updates and the quantification of the model since the original DCPRA-1998 are described in the various revisions of the Calculation File C.9. The vintage of the PRA model is designated by the year in which the update was last completed. It should be noted that updates and re-quantification of the model may have also been performed in the previous year(s) prior to the establishment of the model vintage. For example, the PRA model designated DCPRA-1996 was completed in that year but the update was performed in 1995 and 1996. In recent more recent updates, the updated PRA models are designated by a revision number. For example, the latest Revision 1 DCPRA model has been designated DC01.

The sections below describe the DCPRA model development from the original DCPRA-1988 model to the current DCPRA model, and the revision of the Calculation File C.9 that describes the updates performed for in the PRA model.

Long Term Seismic Program - DCPRA-1988

The objective of the "Long Term Seismic Program" was to satisfy the conditions for issuing the full-power operating license for Unit 1 and 2 by the USNRC. One of the conditions involves the development of and evaluation using a Probabilistic Risk Analysis. The LTSP plan was developed and submitted to the USNRC in early 1985 and was approved by the USNRC in July 1985. The LTSP evaluation was completed in 1988 and a final report (Reference 2) was submitted to the USNRC for review in July 1988.

The review of the LTSP–PRA was performed by the USNRC staff and with the assistance of the Brookhaven National Laboratory (BNL) from 1988 through 1990. BNL was selected by the USNRC to be the technical lead for the review. The USNRC issued Supplement No. 34 to the Safety Evaluation Report NUREG-0675 (SSER 34) in June 1991 (Reference 3), concluding that PG&E has met the probabilistic risk analysis part of the license condition.

A summary of the PRA results is shown in the table below:

Contributor	Mean Core Damage Frequency (per year)
Seismic Events	3.7E-05
Internal Events	1.3E-04
Other External Events	3.9E-05
Total	2.0E-04

The five internal initiating events that have substantial contribution to the Internal Events CDF were:

- Loss of Offsite Power (32.5%)
- Reactor Trip (12.5%)
- Turbine Trip (11.2%)
- Partial Loss of Main Feedwater (8.4%)
- Loss of 1 DC Bus (7.3%)

The remaining 28 percent is distributed among many other events.

The contributions to the "Other External Events" category came primarily from the fire and flood scenarios.

Individual Plant Examination (IPE) – DCPRA-1991

The Diablo Canyon IPE was submitted to the NRC by a letter dated April 14, 1992 in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities – 10CFR 50.54(f)." The NRC issued its staff evaluation of the Diablo Canyon IPE and accepted the study by letter dated June 30, 1993 (Reference 6).

To fulfill the requirements of the IPE, the original PRA model DCPRA_1988 was updated to:

- Reflect then current plant design and operation, which included the use of updated design information through June 1990, and operational data through December 1989.

- Incorporate comments from the lead consultant for the DCPRA-1988 model, and NRC/BNL comments on the model into the updated PRA model

- Expand the DCPRA-1988 model to include the Level 2 containment performance analysis

The following summarized the plant modifications/ improvements incorporated into the PRA model:

1. Diesel Generator Fuel-oil Transfer System. Recirculation lines were added to the system to allow the system to operate continuously once started. This eliminates multiple start demands of the system and hence increasing the reliability of the system.

   In addition, manual operation of the system level control valves on the diesel generator day tanks was provided and to allow a portable engine-driven pump to be connected to the system.

2. Charging Pump Backup Cooling. Provisions were made to allow the use of fire water to cool one of the centrifugal charging pumps in the event of a total loss of component cooling water. This allows reactor coolant pump seal injection and therefore maintains RCP seal cooling in the event of a complete loss of component cooling water.

The core damage frequency from the IPE is 8.8E-05 per year. The CDF is lower that of the original DCPRA-1988 model due to the implementation of above improvements and the incorporation of the improvements into PRA model. The dominant initiating event category contributors to this CDF are given below:

- Loss of Offsite Power (41%)
- General Transients (Reactor Trip, Turbine Trip, etc.) (26%)
- LOCAs (Excessive, Large, Medium, or Small) (9.3%)
- Loss of One DC Bus (F, G, or H) (8.2%)
- Loss of ASW or CCW (6.2%)
- Floods (3.6%)

The Level 2 results were provided in Release Category Groups and the annual contributions from these groups are presented in the Table below:

Release Category Group	Frequency (per year)	Percentage
Small, Early Containment Failure	7.61E-06	8.7
Large, Early Containment Failure	2.45E-06	2.9
Late Containment Failure	3.97E-05	45.2
Containment Bypass	1.62E-06	1.8
Long Term Containment Intact	3.64E-05	41.4

The large early containment failure release group is dominated by those HPME direct containment heating sequences (58%) that are predicted to occur at vessel breach and are predicted to cause large containment failures. The second most likely cause of early containment failure is hydrogen burn (26%).

Individual Plant Examination for External Events (IPEEE) – DCPRA-1993

The Diablo Canyon IPEEE report was submitted to the NRC by a letter dated June, 1994 in response to Generic Letter 88-20, Supplement 4 (Reference 7) which requested each utility to perform an Individual Plant Examination of External Events for severe accident vulnerabilities. The results of the IPEEE showed that no vulnerabilities to severe accidents at the plant due to external events were identified. In addition, no containment performance vulnerabilities were identified in this study. The Diablo Canyon IPEEE was accepted by the NRC via Reference 8.

To fulfill the requirements of the NRC GL 88-20, Supplement 4, the original PRA model DCPRA_1988 was updated to:

- Reflect then current plant design and operation, which included the use of updated design information through March 1993, operational data through December 1991, and human action failure rates and internal events updated through June, 1993.

- Perform a containment performance assessment for the seismic, fire and “other” external events PRA

The following summarized the plant modifications/ improvements incorporated into the PRA model:

1. Dedicated Sixth Emergency Diesel Generator. This plant modification has a significant impact on the plant safety as it increases the availability of the

backup power for the Vital AC Bus F. This has reduced the contribution of loss of offsite power events to the overall core damage frequency.

2. Revision of the 230kV Switchyard Fragility. After the Loma Prieta earthquake, the NRC requested that PG&E reevaluate the fragility of the 230kV switchyard base on the Loma Prieta earthquake experience. This reevaluation resulted in the change in the fragility of the switchyard which was used in the IPEEE.

The results of the IPEEE indicate that the core damage frequency due to seismic events is 4.0E-05 per year and that due to fire events is 2.7E-05 per year. It was determined that each of the “other” external events evaluated contributed less than 1.0E-06 per year to core damage and was screened out as a result. These results do not differ significantly from those previously determined from the LTSP evaluation.

The most important seismic sequences were the seismic-induced station blackout with the following characteristics:

- Seismic event that fails 500kV and 230kV power as well as a primary turbine building shear wall, causing the loss of all vital AC power.

- Seismic event that fails 500kV and 230kV power with the random failure of all diesel generators

The fire risks were dominated by fires in the control room and the cable spreading rooms.

The external events impact on containment performance was also assessed which included the evaluation of the containment structure, penetrations, hatches, and isolation valves, and the containment heat removal capability. These SSCs have high seismic capabilities. Containment performance for fire initiators was conservatively evaluated and it was determine that sequences are similar to those of the internal events. The conclusion was that external events do not pose any unique threat to containment performance, and it is not significantly different that that identified in the IPE.

DCPRA-1995 Model

The update and revision of the DCPRA-1995 model was completed in May 1996. The important changes to the model are documented in Revision 5 of Calculation File C.9 and they are summarized below:

- Addition of the two backup battery chargers 121 and 131 in the model to reduce unnecessary conservatism.

- AFW pump surveillance frequencies were changed from monthly to quarterly.

- An alignment was added to the DFO system (top event FO) to model unavailability during STP P-12B (1 and 2).

- The initial power alignments (i.e., Normal vs. Backup) were switched for the DFO pumps modeled in top event FO.

- The testing frequency for valves 8821A/B in the SI system model (top event SI) was changed from refueling to quarterly.

- The entire instrument AC system model (top events I1, I2, I3, and I4) was modified to reflect the replacement of the old instrument inverters with new uninterruptible power supplies (UPS units).

- The probability distributions of RCP seal leakage leading to core uncovery as a function of time, used in the electric power recovery model (top event RE) were replaced with new distributions which are based on calculations performed for the qualified O-ring material.

  Additionally, the electric power recovery model was revised to always select the distributions for core uncovery time (from RCP seal LOCAs) for scenarios with no depressurization/cooldown.

- The SSPS system model was modified to (1) incorporate the Eagle 21 modification which included the deletion of the High Steam Differential Pressure, High Steam Flow, and the Low-Low Tavg input signals; and (2) the design modifications and testing frequency changes made to reduce the CVCS letdown and charging valves testing frequency.

- The ASW system model was modified to (1) create a new split fraction, ASG, for LOSP and all support available, (2) remove demusseling from a number of alignments, (3) use the unavailability variable ZMVU2F/D for the unit-to-unit crosstie valve (this also effected Top Event AI), and (4) the ASC split fraction was train separated. A review of the quantification indicated that split fractions AS4 and AS7 were not being properly selected, so the event tree split fraction rules were modified accordingly.

The operational data from 01/ 01/92 through 12/31/94 were used in the update of the initiating event frequency, component failure rate, equipment maintenance unavailability and common cause failure probability. The common cause failure probabilities were calculated based on the updated component failure rates. No updates were done on the alpha factors for common cause failure probability.

The core damage frequency in the updated DCPRA-1995 model for internal events (including flooding events) is 4.52E-05 per year. The important initiating event contributors and their percentage contributions to the total internal events CDF are shown below:

- Loss of Offsite Power (18.4 %)
- Loss of Auxiliary Saltwater (12.0%)
- Medium LOCA (10.0%)
- Reactor Trip (8.1%)
- Turbine Trip (6.8%)
- Flooding Scenario FL1 (5.5%)
- Large LOCA (4.6%)
- Loss of DC Bus (G) (4.3%)
- Partial Loss of MFW (4.0%)
- Loss of DC Bus (F) (3.4%)

The decrease in the internal events CDF when compared to that for the IPE is attributable to the changes in the PRA model described above.

DCPRA-1997 Model

The update and revision of the DCPRA-1997 model was completed in January 1999. The major changes to the model are documented in Revision 6 of Calculation File C.9 and they are summarized below:

- The fail on demand for the DC batteries was removed from the vital DC top events since this failure mode was not considered applicable. Instead, a longer mission time (interval between tests) was assumed for the batteries.

- The surveillance test frequency for SSPS slave relays (part of top events SA and SB) was reduced due to a change in the technical specification.

- Similar electric power recovery factors were added to transient-induced loss of offsite power, as is applied to loss of offsite power initiating events.

- The recovery rules applied when the dedicated fuel oil transfer pumps fail (top event FO fails) were revised to allow recovery of some sequences that are recoverable.

- The ASW success criterion (for top event AS and initiating event LOSW) was modified. For unit to unit ASW crosstie to be available, FCV-601 and both pumps from the opposite unit must be available, consistent with the loss of ASW abnormal operating procedure.

- For the AFW system model, the raw water reservoir was added as a backup source of water to the condensate storage tank (CST).

- The PTS analysis was modified, so it assumed reactor vessel conditions as of 2005, instead of end of life (i.e. 2020). Using end of life vessel conditions was overly conservative.

The operational data from 01/01/95 through 11/30/96 were used in the update of the initiating event frequency, and operational data from 01/01/95 through 09/30/96 were used to update component failure rate, equipment maintenance unavailability and common cause failure probability. The common cause failure probabilities were calculated based on the updated component failure rates.

The core damage frequency in the updated DCPRA-1997 model for internal events (including flooding events) is 3.32E-05 per year. The important initiating event contributors and their percentage contributions to the total internal events CDF are shown below:

- Loss of Offsite Power (18.1 %)
- Medium LOCA (12.0%)
- Loss of DC Bus (G) (9.4%)
- Loss of DC Bus (F) (9.2%)
- Low Auxiliary Saltwater (8.1%)
- Flooding Scenario FL1 (7.1%)
- Large LOCA (6.1%)
- Reactor Trip (3.6%)
- Turbine Trip (3.3%)

The changes made to DCPRA-1997 model has the effect of lowering the contributions from initiating events Loss of Auxiliary Seawater and general transients such as Reactor Trip and Turbine Trip. However, some conservatism in the modeling of the impact on ASW system initiated by the Loss of DC Bus F or G has caused these initiating events to increase in its importance with respect to CDF contribution. This conservative modeling was removed in the next PRA model revision.

DC00 Model

The update and revision of the DC00 model was completed in June 2000. This update was done to support the DCPP Risk-Informed In-service Inspection (RI-

ISI) submittal to the NRC. The update and revision was done in two stages: (1) the incorporation of updated component database, system and event tree model changes into the PRA model, and (2) the integration of internal events model, seismic events model, and the fire events model into a single combined PRA model. The major changes to the PRA model are documented in Revisions 7 and 8 of Calculation File C.9, and they are summarized below:

- Auxiliary Salt Water System. Success criteria were changed to be consistent with thermal-hydraulic basis from the "Station Blackout Submittal" (Reference 10) and generic letters on Service Cooling Water Systems. Demusseling valves and associated flow paths were included in the system model (Top Events AS and AI), and system alignment changes were also made to be consistent with current operational practice.

- RCS Pressure Relief System. Added the third PORV (474) in Top Event PR and include a new Top Event (PRX) in the Electric Power Support System Event Tree ELECPWR for questioning RCS pressure relief for a specified set of initiators.

- Event Trees - Changes were made to the General Transient and Support Systems Event Trees stemming from changes to RCS pressure relief (Top Event PR and new Top Event PRX) and Auxiliary Seawater System (Top Event AS), and the related dependencies.

- Balance of Plant (BOP) Systems. Defined a new event tree model BOPSUPP that questions the availability of BOP Systems such as Feedwater, Condensate, Circulating Water/Service Water, Non-Vital Power, and Instrument Air.

- Large Early Release Frequency (LERF). Quantification of LERF was included in the model so that it can be easily juxtaposed with the commonly used figure of merit, Core damage Frequency (CDF).

The first revision of Alpha factors for the calculation of common cause failure probability was performed for this update. New common cause groups were defined for the following components:

- RHR MOVs (Reference 11)
- DC Battery Chargers (Reference 12)
- DC Batteries (Reference 12)

Alpha factors for were updated for the following components based on then more recent common cause failure databases:

- Diesel Generators (Reference 11)

- Residual Heat Removal Pumps (Reference 11)
- Auxiliary Feedwater Pumps (Reference 11)
- Auxiliary Saltwater Pumps (Reference 11)
- Reactor Trip Breakers (Reference 13)
- RT Breaker UV Coils (Reference 13)
- RT Breaker Shunt Trip Coils (Reference 13)

The alpha factors used in the PRA were updated with DCPP plant specific data from November 1984 through September 1996.

Several new initiating events were added (Intake Internal Flooding – FLLOSW, Load Rejection – LREJU, Loss of Instrument Air – LOIA, Feedwater Line Break Outside Containment – FWLBO, Loss of Non-Vital Electric Bus – LNVEL, Loss of Turbine Building Service Cooling Water – LSCW, and Catastrophic RCP Seal Failure – SELOCA) and the MSRV Stuck Open initiator one was deleted as a result of a review of the NRC Initiating Event Database (NUREG/CR-5750 – Reference 9). New generic priors were generated based on NUREG/CR-5750 and used in this revision, which included an update of DCPP data from 12/31/96 through 11/30/99.

The contributions to the total core damage frequency and large early release frequency from Internal Events, Seismic Events and Fire Events are shown in the Table below:

Contributor	Mean Core Damage Frequency (per year)	Mean Large Early Release Frequency (per year)
Internal Events	1.41E-05	5.54E-07
Seismic Events	3.36E-05	1.25E-06
Fire Events	1.50E-05	6.42E-09
Total	6.26E-05	1.81E-06

The important internal initiating event contributors (including flooding events) and their percentage contributions to the total internal events CDF are shown below:

- Flooding Scenario Failing CCW - FL1 (16.6%)
- Loss of Offsite Power (16.3%)
- Loss of Auxiliary Saltwater (12.3%)
- Steam Line Break Inside Containment (10.8%)
- Loss of Component Cooling Water (4.5%)
- Loss of Switchgear Room Ventilation (3.8%)
- Reactor Trip (3.3%)
- Catastrophic RCP Seal Failure (3.0%)

The CDF contribution from Internal Events from the DC00 PRA model is lower than the previous version of the PRA model. This is due primarily to the changes in the system and event tree models and revised database as indicated above. The contributions to CDF from the LOCAs, in particular the Medium and Large LOCA were reduced due primarily to the new initiating event frequencies from NUREG/CR-5750 (Reference 9). Revision in the modeling of impact on the ASW system for loss of DC Bus F and G initiating events had also reduce the contributions of these initiating events to total internal event CDF.

There is no change in the modeling of the seismic initiating events. The seismic-induced CDF is also slight lower than that from the IPEEE and is due primarily to the updated system models and the revised database used in the PRA.

There is also no change in the modeling of the fire initiating events, Similarly, The fire-induced CDF is also slight lower than that from the IPEEE and is due primarily to the updated system models and the revised database used in the PRA.

DCC0 Model

The update and revision of the DCC0 model was completed in March 2001 based on the changes made to the DC00 PRA model since June of 2000 – that is, over a period of several months. The major changes to the PRA model are documented in Revision 9 of Calculation File C.9 and they are summarized below:

- AMSAC System. This system was credit to actuate the AFW system and turbine trip. The system model (Top Events AMA and AMB) developed was incorporated into the Mechanical Support Systems event tree MECHSP. The other event tree models were impacted by the implementation of the AMSAC system: General Transient, SGTR, ATWT, and the Interfacing System LOCA event tree model.

- Backfeeding from the 500kV switchyard. The operator action for backfeeding from the 500kV was implemented via a new Top Event OGR which was added to the Electric Power support system event tree model ELECPWR. New component failure rates/unavailability for equipment associated with the 230kV and 500kV switchgear were developed and used in the system model for the offsite power source.

- Cross-tying of Vital Buses – that is, one diesel generator feeds loads of two vital buses. This recovery action was incorporated into the Electric Power System event tree model ELECPWR.

- Included the aligning of the Raw Water Reservoir (RWR) to the suction of the AFW pumps in Top Event AW.

- Credit was taken for makeup to the RWST (Top Event MU) given loss of Low Head pump trains. Dependency of operator actions between failure to initiate sump recirculation (Top event RF) and the operator actions to makeup to the RWST was considered and incorporated in the model update.

- Electric Power Recovery: The latest HEPs were used in Tope Event RE and the battery lifetime was revised from 12 hours to 7 hours.

- Evaluation of Pre-Initiating Event Human Actions. Several such human actions were evaluated and incorporated in the various system models: failure to restore fuel oil system (top Event FO), failure to restore diesel fuel oil LCV control switch, and failure to restore battery charger operability.

- The following HEPs were either newly created or HEPs that were revised/re-evaluated: ZHECC2, ZHEAS5, ZHEFL1, ZHEFL2, ZHEAS4, ZHEBC1, ZHERE8, ZHERE9, ZHEREA, ZHEREB, ZHESV3, ZHEPR1, ZHEAW2, ZHEAW5, ZHEAW6, ZHEMU2, and ZHEHU3. These updated/newly created HEPs were incorporated into the DCC0 PRA model as described above.

The component databases were not updated in this revision of the PRA model. The seismic analysis was updated to allow the use of the safety injection pumps for a Very Small LOCA (VSLOCA) event after the RCS has been sufficiently depressurized.

The Fire Initiating Event FS5 was revised to correctly model its impact on the ASW system, that is, the fire scenario fails only the two Unit 1 ASW pumps instead of all four ASW pumps.

The DCC0 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	1.04E-05	4.94E-07
Seismic Events	3.12E-05	1.28E-06
Fire Events	1.33E-05	6.31E-09
Total	5.38E-05	1.78E-06

The important internal initiating event contributors (including flooding events) and their percentage contributions to the total internal events CDF are shown below:

- Flooding Scenario Failing CCW - FL1 (22.5%)
- Loss of Offsite Power (17.8%)
- Loss of Auxiliary Saltwater (17.4%)
- Loss of Common Cooling Water (6.1%)
- Catastrophic RCP Seal Failure (6.0%)
- Reactor Trip (4.2%)
- Medium LOCA (3.2%)

The majority of the reduction in Internal Events CDF when compared to the CDF value of the previous DC00 model is attributable to the following changes to the model:

- The addition of AMSAC to actuate the AFW system and trip the turbine resulted in a reduction in frequency of all the ATWT sequences. It also provides a redundant AFW pump start signal when SSPS fails.
- The steamline break initiators (SLBI and SLBO) now credit manual SSPS actuation.
- The ability to backfeed from the 500kV switchyard and crosstie the vital buses in accordance with the EOPs was fully implemented.
- Pre-initiators and post-initiators HEPs were updated.
- Unit 2 outage bus durations were changed to reflect more realistic out of service times.

The majority of the reduction in seismic CDF is attributable to the change to the seismic analysis incorporating use of the safety injection pumps (and depressurization) for a very small LOCA (VSLOCA) event.

The reduction in fire CDF is attributable to a correction made to the impact of Fire Initiator FS5 on the ASW system in the PRA model. The reduction in the contributions to CDF by the fire initiating events can also be attributed to the improvement in the internal events portion of the PRA model as described above.

DC01

The update and revision of the DC01 model was initiated in 2004 and it was completed in June 2006. Plant design changes for the period 1/1/200 through 12/31/2004 (Reference 14) were reviewed and plant procedure revisions (then current as of 2/04/2005) were also reviewed (Reference 15). Any plant design and/or procedure changes that have an impact on the PRA model were incorporated into the model. The component database (failure rates, maintenance unavailability, and certain electric power component unavailability) was updated using plant-specific operation data from 10/01/96 through 09/30/01 (Calculation File H.1.5, revision 6). In addition, the updates and revisions of the PRA model leading to the DC01 were done in support of the following DCPP programs: 14 days Diesel Generator AOT LAR submittal, MSPI and Safety Monitor implementation. Note that, many of the changes to the PRA model were done to facilitate the implementation of the above programs and did not have significant impact on the CDF and LERF results. Other model changes had an impact on the results of the PRA model.

The major changes to the PRA model are briefly described in Revision 10 of Calculation File C.9 and they are summarized below:

- Separating the 480V buses from the then existing Vital AC Power top events and model the 480V buses in separate top events.

  Separating the batteries from the then existing 125V DC Power top events and model the batteries under separate top events. The batteries are required to provide 125V DC power on demand whereas the battery chargers would provide long tern DC power supply.

  The above model changes allow the modeling of the DC-AC power system interface more accurately and allow the more accurate modeling of the impact of loss of 480V and/or 4kV buses on safety/accident mitigating equipment modeled in the PRA.

  The impacted support system and frontline system event tree models due to the above modeling changes were revised accordingly.

- In most of the then existing system model fault trees, the basic events defined in these fault trees were for “super-components” which contain more than one component and component failure modes. As required by the MSPI program, major equipment failure modes must be modeled explicitly as basic events. Changes were made to many of the mitigating system models to meet this MSPI requirement. These changes do not have any significant impact on the system unavailability and hence plant risk.

- The loss of offsite power initiating event was revised to conform to the information/model in Reference 16. The total loss of offsite power frequency is divided into 5 different types of causes and a separate initiating event frequency is then developed for each type. New generic prior distributions were generated using the NRC Initiating Events Database (Reference 16) as a source. The experience data of this data source covers the period between 1986 and 2003, with the Diablo Canyon specific operating records through 9/31/2005. The “new” loss of offsite power initiating events were then updated with the plant specific data.

- The offsite electric power recovery model was updated to reflect the new loss of offsite power durations corresponding to the new set of loss of offsite power initiating events as briefly described above. The offsite power non-recovery curves corresponding to this new set of initiating events were used in the evaluation of the offsite power non-recovery factors.

- Incorporation of the Rhodes RCP Seal LOCA Model for station blackout scenarios. This was done in conjunction with the updated electric power (offsite and onsite) recovery model.

- Extensive revision to the Auxiliary Feedwater System was done for this version of the PRA model. A summary of the system model changes is provided below:

    - Included the Fire Water Storage Tank (FWST) as a supplemental water supply to the CST. Note that, the FWST does have sufficient volume to be considered a full backup source in the PRA model.
    - Added new system top events to handle different sets of boundary conditions and corresponding SGs and AFW Pumps Success Criteria
    - The RUNOUT protection function for MDP1-2 was added to the system model, while assuming that the pump runout events would not adversely impact MDP 1-3. Note that in the previous model, it was conservatively assumed the guaranteed failure of the moto-driven AFW pumps due to pump runout in the events of depressurization of one or more SG due to steam line break downstream the MSIVs.
    - Credit was given to the safety valves in the event that the 10% ADV were not available.

- Depressurization of the RCS was added to the event sequence model via the new Top Event OR instead of being embedded in Top Event MU which previously also included the modeling of the depressurization of RCS for closed loop RHR cooling.

- New probability for the consequential loss of offsite power (LOOPCN) after a plant trip was developed and used in the Top Event OG model which questions the availability of the offsite grid after a plant trip

- The HRA was updated using the EPRI HRA Calculator (Reference 17). This was completed in November of 2002 and the updated HEPs were used in this revision of the PRA model.

- Update to the Level 2 PRA model to allow a more realistic assessment of the Large Early Release Frequency figure or merit (Reference 18).

The DC01 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	1.08E-05	1.60E-06
Seismic Events	3.77E-05	1.89E-06
Fire Events	1.70E-05	-
Total	6.55E-05	3.49E-06 (1)
Note: (1) Total LERF does not include contribution from fire initiators		

The important internal initiating event contributors (including flooding events) and their percentage contributions to the total internal events CDF are shown below:

- Medium LOCA (12.2%)
- Flooding Scenario Failing CCW - FL1 (11.6%)
- Steam Generator Tube Rupture (11.2%)
- Loss of Offsite Power – Grid Related (7.9%)
- Reactor Trip (7.8%)
- Turbine Trip (5.8%)
- Partial Loss of Feedwater (4.7%)
- Loss of Switchgear Ventilation (4.2%)

There is an increase in the Internal Events CDF of approximately 4% from the last quantification (DCC0). Some changes in the model have the effect of increasing the CDF and others have the opposite effect. The resulting increase in Internal Events CDF and the characteristics of the important initiating event contributors are attributable to the following changes to the model:

- An increase in the HEP value following HRA update (Calculation File G.2, Revision 5 – Reference 19). This is event from the increase in the risk importance in the Medium and Large LOCA initiator due to the increase in the HEP value for operation actions to switch to sump recirculation mode of operation.

- Modeling of the 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 un-isolated steam 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 sump recirculation mode of operation during SLOCA, and transient induced LOCA scenarios
- A higher consequential LOSP probability used in the PRA
- New LOSP initiators were defined for this revision 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 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 20).

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 that the effect of causing an increased 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.

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6. U.S. Nuclear regulatory Commission, “Staff Evaluation of the Diablo Canyon Power Plant (DCPP) Units 1 and 2, Individual Plant Examination (IPE) – Internal Events Submittal,” June 30, 1993.

7. U.S. Nuclear regulatory Commission, “Individual Plant Examination of External Events for severe Accident Vulnerabilities,” Generic Letter 88-20, Supplement 4, June 28, 1991

8. U.S. Nuclear Regulatory Commission, “Review of Diablo Canyon Individual Plant Examination of External Events (IPEEE) Submittal, December 4, 1997

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16. Draft NUREG/CR (INEEL/EXT-04-02326), “ Evaluation of Loss of Offsite Power Events at Nuclear Plants: 1986-2003 (Draft)”, October, 2004 by S.A.Eide, C.D. Gentillon, and T.E Wierman of INEEL and D.M. Rasmuson of USNRC

17. EPRI Calculator

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19. Calculation File G.2, Revision 5

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21. E.11, Revision 10, Miscellaneous Systems – PRA System Analysis