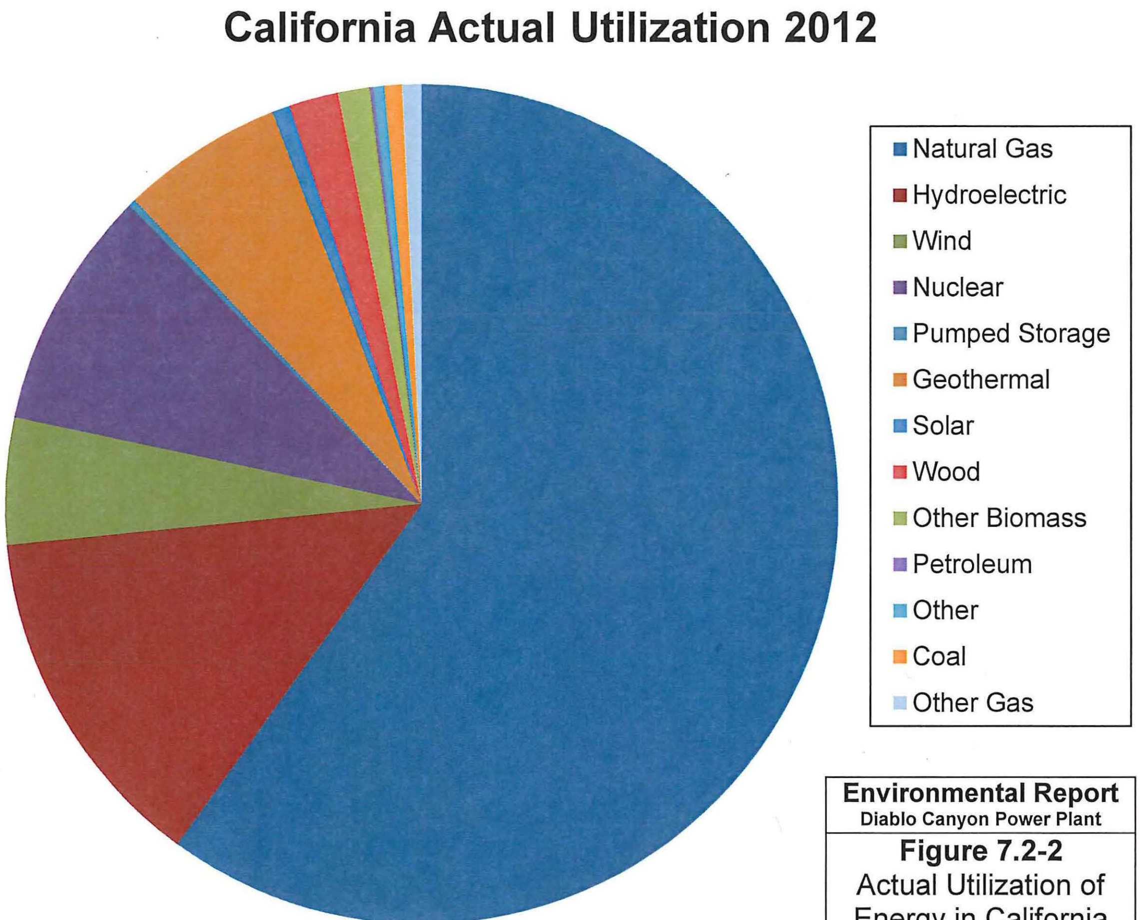
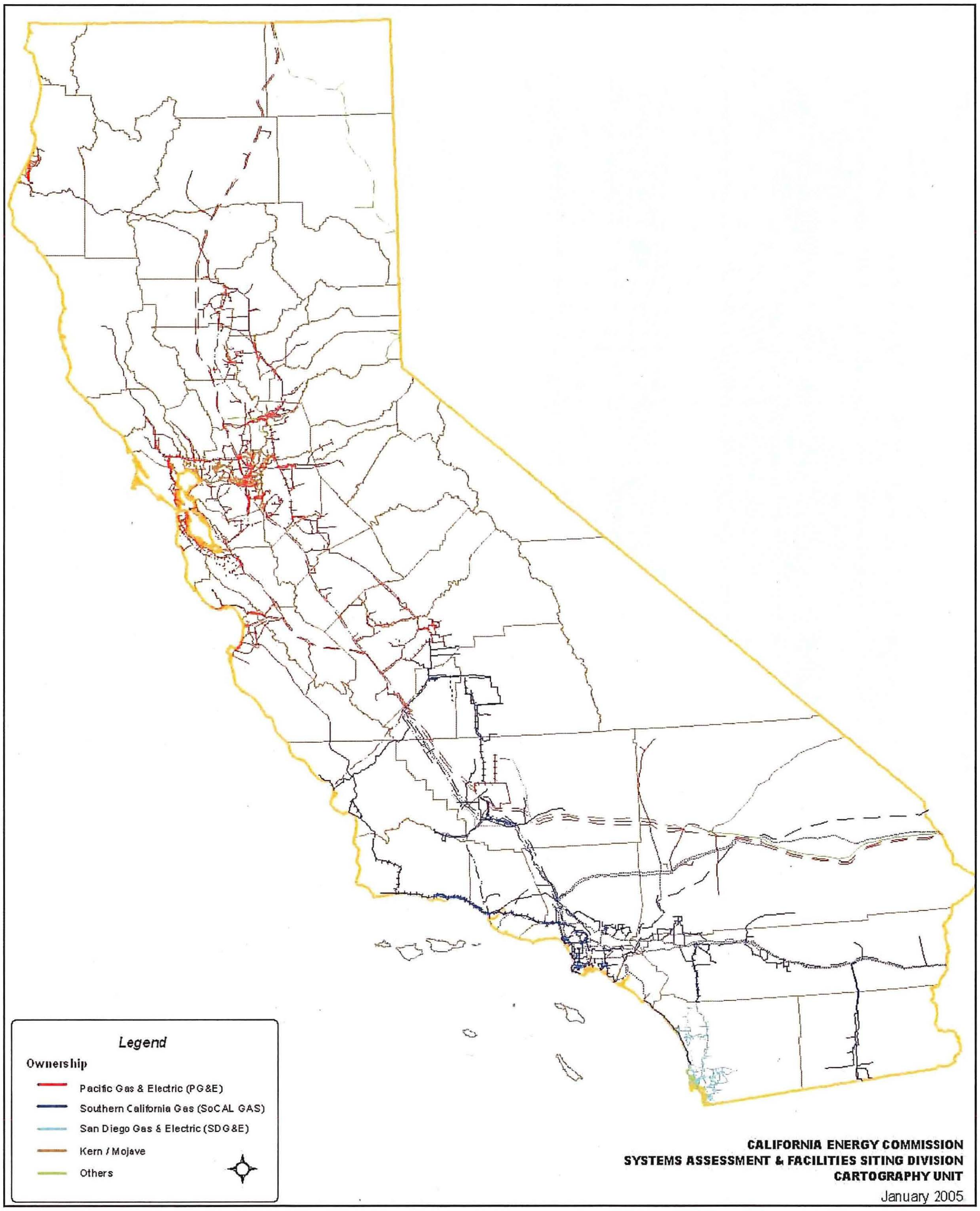


Environmental Report
Diablo Canyon Power Plant

Figure 7.2-2
Actual Utilization of Energy in California

Source: Reference 19



Environmental Report
Diablo Canyon Power Plant

Figure 7.2-3
Natural Gas Pipeline Layout in California

TABLE 8-1
IMPACTS COMPARISON SUMMARY

			Alternatives			
Impact Category	**Proposed Action (License Renewal)**	**Decommissioning**	**Natural Gas-Fired Generation**	**Purchased Power**	***Combination Alternative***	***Demand Side Management (DSM) and Energy Efficiency (EE)***
Land Use	SMALL	SMALL	SMALL	MODERATE	*MODERATE to LARGE*	*SMALL*
Water Quality	SMALL	SMALL	SMALL	SMALL to MODERATE	*SMALL*	*SMALL*
Air Quality	SMALL	SMALL	MODERATE	SMALL to MODERATE	*SMALL to MODERATE*	*SMALL*
Ecological Resources	SMALL	SMALL	SMALL to MODERATE	SMALL to MODERATE	*SMALL to MODERATE*	*SMALL*
Threatened or Endangered Species	SMALL	SMALL	SMALL to MODERATE	SMALL	*SMALL to MODERATE*	*SMALL*
Human Health	SMALL	SMALL	SMALL	SMALL to MODERATE	*SMALL to MODERATE*	*SMALL*
Socioeconomics	SMALL	SMALL	SMALL to MODERATE	SMALL to MODERATE	*SMALL to MODERATE*	*SMALL to MODERATE*
Waste Management	SMALL	SMALL	SMALL	SMALL to MODERATE	*SMALL to MODERATE*	*SMALL*
Aesthetics	SMALL	SMALL	SMALL	SMALL to MODERATE	*SMALL to MODERATE*	*SMALL*
Cultural Resources	SMALL	SMALL	SMALL to MODERATE	SMALL	*SMALL to MODERATE*	*SMALL*

SMALL - Environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource.
MODERATE - Environmental effects are sufficient to alter noticeably, but not to destabilize, any important attribute of the resource. 10 CFR 51, Subpart A, Appendix B, Table B-1, Footnote 3.

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Alternatives: Base (Decommissioning)	Alternatives: Natural Gas-Fired Generation	Alternatives: Purchased Power	Alternatives: *Combination Alternative*	Alternatives: *DSM and EE*
Alternative Descriptions					
DCPP license renewal for 20 years, followed by decommissioning	Decommissioning following expiration of current DCPP licenses. Adopting the GEIS description by reference (Reference 1) as comparable to DCPP decommissioning.	New construction at the DCPP site.	Would involve construction of new generation capacity in the region. Adopting by reference GEIS description of alternate technologies (Section 7.2.1.1)	*New construction of natural gas combine-cycle (NGCC) at the DCPP site. Construction of new wind energy, concentrated solar power (CSP) solar photovoltaic (PV), and geothermal somewhere in California within the PG&E service area.*	*SMALL – Adopting by reference Supplemental GEISs 33, 37, and 38 descriptions of impacts from conservation programs (Section 7.2.2.4)*
		Assuming PG&E can use existing Diablo-Gates transmission line rights-of-way and connect to the gas pipeline for the Morro Bay Power Plant, approximately 15 miles would need to be constructed[2]		*Assuming PG&E can use existing Diablo-Gates transmission line rights-of-way and connect to the gas pipeline for the Morro Bay Power Plant, approximately 15 miles would need to be constructed[3]. Transmission lines would also need to be constructed for the offsite alternatives; length of lines depends on the location of each alternative.*	

[2] Connection to the existing pipeline is feasible, assuming the pipeline has the capacity to support the 4 combined-cycle units at DCPP.
[3] Connection to the existing pipeline is feasible, assuming the pipeline has the capacity to support the 2 combined-cycle units at DCPP.

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Base (Decommissioning)	Alternatives: Natural Gas-Fired Generation	Alternatives: Purchased Power	Alternatives: *Combination Alternative*	Alternatives: *DSM and EE*
		Use existing switchyard and transmission lines.	Construct transmission lines from available power sources located within the State or Pacific Northwest Region.	*Use existing switchyard and transmission lines for the NGCC power plant. For the remainder of alternatives, construct transmission lines from available power sources located within California*	
		Four 562.5-MW of net power (Combined-cycle turbines to be used); capacity factor 0.90		*<u>NGCC:</u> 1105 MW generated; Two 562.5-MW of net power (Combined-cycle turbines to be used); capacity factor 0.90* *<u>Wind:</u> 290MW generated; 830 MW capacity; capacity factor 0.35* *<u>CSP:</u> One facility with 400MWe capacity* *<u>PV:</u> 290MW generated; 1160 MW capacity; 0.25 capacity factor* *<u>Geothermal:</u> 100MW*	
		New mechanical-draft cooling towers would need to be constructed to support the closed cycle cooling		*New mechanical-draft cooling towers would need to be constructed for NGCC and CSP facilities to support the closed cycle cooling systems.*	

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Base (Decommissioning)	Alternatives: Natural Gas-Fired Generation	Purchased Power	*Combination Alternative*	*DSM and EE*
		systems.			
		Natural gas, 1,015 Btu/ft^3; 6,600 Btu/kWh; 0.0034 lb SO_x/MMBtu; 0.0109 lb NO_x/MMBtu; 115,347,192,118 ft^3 gas/yr		*Natural gas, 1,015 Btu/ft^3; 6,600 Btu/kWh; 0.0034 lb SO_x/MMBtu; 0.0109 lb NO_x/MMBtu; 57,673,596,059 ft^3 gas/yr*	
		Selective catalytic reduction with steam/water injection		*Selective catalytic reduction with steam/water injection for NGCC*	
~~1,350~~*1,440* permanent employees		31 workers per plant (Section 7.2.2.1)		*31 workers per NGCC plant (Section 7.2.2.1). Jobs would be generated on a temporary basis by the construction of the NGCC plant and the offsite alternatives.*	
Land Use Impacts					
Small – Adopting by reference Category 1 issue findings (Attachment A, Table A-1, Issues 52, 53)	SMALL – Not an impact evaluated by GEIS (Reference 1)	SMALL – 25 to 30 acres per facility at DCPP location; pipeline could be routed along existing transmission line corridors and could	MODERATE – In part, most transmission facilities could be constructed along existing transmission	*MODERATE to LARGE – NGCC: 25 to 30 acres per facility at DCPP location; pipeline could be routed along existing transmission line corridors and could require additional 90 to 100*	*SMALL – Adopting by reference Supplemental GEISs 33, 37, and 38 descriptions of*

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Base (Decommissioning)	Natural Gas-Fired Generation	Purchased Power	Combination Alternative	DSM and EE
		require additional 90 to 100 acres for easements (Section 7.2.2.1)	corridors (Section 7.2.2.2) Adopting by reference GEIS description of land use impacts from alternate technologies (Reference 1)	*acres for easements (Section 7.2.2.1); approximately 3,655 acres would be needed for natural gas wells and collection stations (Section 7.2.2.2) CSP: 2,000 acres for 400 MW generation in a maximum solar exposure area (Section 7.2.2.2) PV: 143,132 acres for 1,160 MW generation in a maximum solar exposure area (Section 7.2.2.2) Wind: 283,860 to 341,130 acres for 830 MW of generation; land may be dual use (Section 7.2.2.2). Geothermal: 20 acres for 100MW generation.*	*impacts from conservation programs (Section 7.2.2.4)*
Water Quality Impacts					
SMALL – Adopting by reference Category 1 issue findings (Attachment A, Table A-1, Issues 3, 4, 6-12, 32, and 37). Five	SMALL – Adopting by reference Category 1 issue finding (Attachment A, Table A-1, Issue 89)	SMALL – Reduced cooling water demands, inherent in combined-cycle design (Section 7.2.2.1)	SMALL to MODERATE – Adopting by reference GEIS description of water quality impacts from alternate technologies	*SMALL to MODERATE – Reduced cooling water demands, inherent in natural gas combined-cycle design (Section 7.2.2.1). Adopting by reference GEIS description of water quality impacts from alternate*	*SMALL – Adopting by reference Supplemental GEISs 33, 37, and 38 descriptions of impacts from conservation*

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Alternatives: Base (Decommissioning)	Natural Gas-Fired Generation	Purchased Power	*Combination Alternative*	*DSM and EE*
Category 2 groundwater issues not applicable (Section 4.1, Issue 13; Section 4.5, Issue 33; Section 4.6, Issue 34; Section 4.7, Issue 35; and Section 4.8, Issue 39)			(Reference 1)	*technologies (Reference 1).*	*programs (Section 7.2.2.4)*
Air Quality Impacts					
SMALL – Adopting by reference Category 1 issue findings (Attachment A, Table A-1, Issue 51). One Category 2 issue not applicable (Section 4.11, Issue 50).	SMALL – Adopting by reference Category 1 issue finding (Attachment A, Table A-1, Issue 88)	MODERATE – 199 tons SO_x/yr 638 tons NO_x/yr 134 tons CO/yr 111 tons PM_{10}/yr[a] 8,780,805 tons CO_2/yr (Section 7.2.2.1)	SMALL to MODERATE – Adopting by reference GEIS description of air quality impacts from alternate technologies (Reference 1)	*SMALL to MODERATE – 96 tons SO_x/yr 309 tons NO_x/yr 65 tons CO/yr 54 tons PM_{10}/yr[a] 4,246,297 tons CO_2/yr (Section 7.2.2.2). Adopting by reference GEIS description of air quality impacts from alternate technologies for CSP, PV, wind, and geothermal facilities (Reference 1).*	*SMALL – Adopting by reference Supplemental GEISs 33, 37, and 38 descriptions of impacts from conservation programs (Section 7.2.2.4)*

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Alternatives: Base (Decommissioning)	Natural Gas-Fired Generation	Purchased Power	*Combination Alternative*	*DSM and EE*
Ecological Resource Impacts					
SMALL – Adopting by reference Category 1 issue findings (Attachment A, Table A-1, Issues 15-24, 45-48). One Category 2 issue not applicable (Section 4.9, Issue 40). DCPP holds a current NPDES Permit, which constitutes compliance with Clean Water Act Section 316(b) (Section 4.2, Issue 25; Section 4.3, Issue 26; and Section 4.4, Issue 27).	SMALL – Adopting by reference Category 1 issue finding (Attachment A, Table A-1, Issue 90)	SMALL to MODERATE – Construction of the pipeline could alter habitat. (Section 7.2.2.1)	SMALL to MODERATE – Adopting by reference GEIS description of ecological resource impacts from alternate technologies (Reference 1)	*SMALL to MODERATE – Construction of the NGCC pipeline could alter habitat. (Section 7.2.2.1). Adopting by reference GEIS description of ecological resource impacts from alternate technologies for CSP, PV, wind, and geothermal facilities (Reference 1).*	*SMALL – Adopting by reference Supplemental GEISs 33, 37, and 38 descriptions of impacts from conservation programs (Section 7.2.2.4)*
Threatened or Endangered Species Impacts					
SMALL – Several federally-listed threatened, endangered, or	SMALL – Not an impact evaluated by GEIS (Reference 1)	SMALL to MODERATE – Federal and state laws prohibit	SMALL – Federal and state laws prohibit destroying or	*SMALL to MODERATE – Federal and state laws prohibit destroying or adversely affecting*	*SMALL – Adopting by reference Supplemental*

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Base (Decommissioning)	Alternatives: Natural Gas-Fired Generation	Purchased Power	*Combination Alternative*	*DSM and EE*
candidate species are known to occur in the vicinity of the DCPP site or along the transmission corridors. PG&E is currently unaware of any adverse issues that involve threatened or endangered species associated with the operation and/or maintenance of DCPP, including the existing transmission lines, towers, and access roads (Section 4.10, Issue 49).		destroying or adversely affecting protected species and their habitats. However, routing of the proposed natural gas pipeline could potentially affect those species in the Morro Bay Estuary.	adversely affecting protected species and their habitats	*protected species and their habitats. However, routing of the proposed NGCC natural gas pipeline could potentially affect those species in the Morro Bay Estuary.*	*GEISs 33, 37, and 38 descriptions of impacts from conservation programs (Section 7.2.2.4)*
Human Health Impacts					
SMALL – Adopting by reference	SMALL – Adopting by reference Category 1 issue	SMALL – Adopting by reference GEIS conclusion that	SMALL to MODERATE – Adopting by	*SMALL to MODERATE – Adopting by reference GEIS conclusion that some*	*SMALL – Adopting by reference*

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Alternatives: Base (Decommissioning)	Natural Gas-Fired Generation	Purchased Power	*Combination Alternative*	*DSM and EE*
Category 1 issue findings (Attachment A, Table A-1, Issue 56, 58, 61, 62). The issue of microbiological organisms (Section 4.12, Issue 57) does not apply. Risk due to transmission line-induced currents are minimal due to conformance with consensus code (Section 4.13, Issue 59).	finding (Attachment A, Table A-1, Issue 86)	some risk of cancer and emphysema exists from emissions (Reference 1)	reference GEIS description of human health impacts from alternate technologies (Reference 1)	*risk of cancer and emphysema exists from NGCC emissions. Adopting by reference GEIS description of human health impacts from alternate technologies for CSP, PV, wind, and geothermal facilities (Reference 1).*	*Supplemental GEISs 33, 37, and 38 descriptions of impacts from conservation programs (Section 7.2.2.4)*
Socioeconomic Impacts					
SMALL – Adopting by reference Category 1 issue findings (Attachment A, Table A-1, Issues 64, 67). Two Category 2 issues are not	SMALL – Adopting by reference Category 1 issue finding (Attachment A, Table A-1, Issue 91)	SMALL to MODERATE – Reduction in permanent work force at DCPP could affect surrounding counties (Section 7.2.2.1)	SMALL to MODERATE – Adopting by reference GEIS description of socioeconomic impacts from alternate technologies (Reference 1)	*SMALL to MODERATE – Reduction in permanent work force at DCPP could affect surrounding counties. Adopting by reference GEIS description of socioeconomic impacts from alternate technologies for CSP, PV, wind, and geothermal facilities*	*SMALL – Adopting by reference Supplemental GEISs 33, 37, and 38 descriptions of impacts from conservation programs*

TABLE 8-2
IMPACTS COMPARISON DETAIL

	Alternatives				
Proposed Action (License Renewal)	**Base (Decommissioning)**	**Natural Gas-Fired Generation**	**Purchased Power**	***Combination Alternative***	***DSM and EE***
applicable (Section 4.16, Issue 66 and Section 4.17.1, Issue 68). Location in medium population area with no growth controls minimizes potential for housing impacts (Section 4.14, Issue 63). Plant property tax payment represents 6 percent of county's total tax revenues (Section 4.17.2, Issue 69). Capacity of public water supply and transportation infrastructure minimizes potential for related impacts (Section 4.15,				*(Reference 1).*	*(Section 7.2.2.4)*

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Base (Decommissioning)	Natural Gas-Fired Generation	Purchased Power	*Combination Alternative*	*DSM and EE*
			Alternatives		
Issue 65 and Section 4.18, Issue 70).					
Waste Management Impacts					
SMALL – Adopting by reference Category 1 issue findings (Attachment A, Table A-1, Issues 77-85)	SMALL – Adopting by reference Category 1 issue finding (Attachment A, Table A-1, Issue 87)	SMALL – Almost no waste generation (Section 7.2.2.1)	SMALL to MODERATE – Adopting by reference GEIS description of waste management impacts from alternate technologies (Reference 1)	*SMALL to MODERATE – Almost no waste generation is associated with NGCC. PV panel manufacturing generates hazardous wastes.*	*SMALL – Adopting by reference Supplemental GEISs 33, 37, and 38 descriptions of impacts from conservation programs (Section 7.2.2.4)*
Aesthetic Impacts					
SMALL – Adopting by reference Category 1 issue findings (Attachment A, Table A-1, Issues 73, 74)	SMALL – Not an impact evaluated by GEIS (Reference 1)	SMALL – Steam turbines and stacks would create visual impacts comparable to those from existing DCPP facilities (Section 7.2.2.1)	SMALL to MODERATE – Adopting by reference GEIS description of aesthetic impacts from alternate technologies (Reference 1)	*SMALL to MODERATE – Impacts from NGCC-related steam turbines and stacks would create visual impacts comparable to those from existing DCPP facilities. Wind farm turbines and turbine blades, and ground-mounted PV systems would create negative visual impacts.*	*SMALL – Adopting by reference Supplemental GEISs 33, 37, and 38 descriptions of impacts from conservation programs (Section 7.2.2.4)*
Cultural Resource Impacts					
SMALL – SHPO	SMALL – Not an	SMALL to	SMALL –	*SMALL to MODERATE –*	*SMALL –*

TABLE 8-2
IMPACTS COMPARISON DETAIL

Proposed Action (License Renewal)	Alternatives				
	Base (Decommissioning)	Natural Gas-Fired Generation	Purchased Power	*Combination Alternative*	*DSM and EE*
consultation minimizes potential for impact (Section 4.19, Issue 71)	impact evaluated by GEIS (Reference 1)	MODERATE – Impacts to cultural resources would be likely due to undeveloped nature of the proposed natural gas pipeline connection (Section 7.2.2.1)	Adopting by reference GEIS description of cultural resource impacts from alternate technologies (Reference 1)	*Impacts to cultural resources would be likely due to undeveloped nature of the proposed natural gas pipeline connection (Section 7.2.2.1) and construction of new CSP, PV, geothermal, and wind facilities.*	*Adopting by reference Supplemental GEISs 33, 37, and 38 descriptions of impacts from conservation programs (Section 7.2.2.4)*

SMALL - Environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource.
MODERATE - Environmental effects are sufficient to alter noticeably, but not to destabilize, any important attribute of the resource. 10 CFR 51, Subpart A, Appendix B, Table B-1, Footnote 3.

Btu	= British thermal unit	MM	= million
CO	= carbon monoxide	MW	= megawatt
CO_2	= carbon dioxide	NO_x	= nitrogen oxide
ft^3	= cubic foot	PM_{10}	= particulates having diameter less than 10 microns
gal	= gallon	SHPO	= State Historic Preservation Officer
GEIS	= Generic Environmental Impact Statement (NRC 1996)	SO_x	= oxides of sulfur
kW-h	= kilowatt-hour	TSP	= total suspended particulates
lb	= pound	yr	= year

a. All TSP for gas-fired alternative is PM_{10}

9.2 ALTERNATIVES

NRC

"...The discussion of alternatives in the report shall include a discussion of whether alternatives will comply with such applicable environmental quality standards and requirements." 10 CFR 51.45(d) as adopted by 10 CFR 51.53(c)(2)

The *natural* gas*, energy efficiency, combination,* and purchased power alternatives discussed in Chapter 7 could potentially be constructed and operated to comply with all applicable environmental quality standards and requirements. PG&E notes that increasingly stringent air quality protection requirements could make the construction of a large fossil-fueled power plant*, such as that associated with the natural gas and combination alternatives,* infeasible in many locations. PG&E also notes that the U.S. Environmental Protection Agency has new requirements for the design and operation of cooling water intake structures at new and existing facilities (40 CFR 125 Subparts I and J). The requirements would necessitate construction of cooling towers for the gas-fired *or combination* alternative if surface waters could no longer be used for once-through cooling.

Attachment 2 – Environmental Report, Amendment 2

Section 4.20
Appendix F

4.20 SEVERE ACCIDENT MITIGATION ALTERNATIVES

NRC

The environmental report must contain a consideration of alternatives to mitigate severe accidents "...if the staff has not previously considered severe accident mitigation alternatives for the applicant's plant in an environmental impact statement or related supplement or in an environment assessment..." 10 CFR 51.53(c)(3)(ii)(L)

"...The probability weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to ground water, and societal and economic impacts from severe accidents are small for all plants. However, alternatives to mitigate severe accidents must be considered for all plants that have not considered such alternatives...." 10 CFR 51, Subpart A, Appendix B, Table B-1, Issue 76

This section summarizes the PG&E analysis of alternative ways to mitigate the impacts of severe accidents. Attachment F provides a detailed description of the severe accident mitigation alternatives (SAMA) analysis.

The term "accident" refers to any unintentional event (i.e., outside the normal or expected plant operation envelope) that results in the release or a potential for release of radioactive material to the environment. NRC categorizes accidents as "design basis" or "severe." Design basis accidents are those for which the risk is great enough that NRC requires plant design and construction to prevent unacceptable accident consequences. Severe accidents are those that NRC considers too unlikely to warrant design controls.

NRC concluded in its license renewal rulemaking that the unmitigated environmental impacts from severe accidents met its Category 1 criteria. However, NRC made consideration of mitigation alternatives a Category 2 issue because not all plants had completed ongoing regulatory programs related to mitigation (e.g., individual plant examinations and accident management). Site-specific information to be presented in the license renewal environmental report includes: (1) potential SAMAs; (2) benefits, costs, and net value of implementing potential SAMAs; and (3) sensitivity of analysis to changes in key underlying assumptions.

PG&E maintains a probabilistic risk assessment (PRA) model to use in evaluating the most significant risks of radiological release from DCPP fuel assemblies and escape from the reactor coolant system into the containment structure.

Original SAMA Analysis

As discussed in PG&E Letter DCL-09-079, *dated November 23, 2009, PG&E completed a SAMA analysis.* For th*is*~~e~~ *original* SAMA analysis, PG&E used the PRA model ~~output~~ *insights* as input to an NRC-approved model that calculates economic costs and dose to the public from hypothesized releases from the containment structure into the environment (*PG&E Letter DCL-09-079, Enclosure 2,* Attachment F). Then, using NRC regulatory analysis techniques, PG&E calculated the monetary value of the unmitigated DCPP severe accident risk. The result represents the monetary value of the base risk of dose to the public and workers, offsite and onsite economic impacts, and replacement power. This value became a cost/benefit-screening tool for potential SAMAs; a SAMA whose cost of implementation exceeded the base risk value could be rejected as being not cost-beneficial.

DCPP used industry and DCPP-specific information to create a list of 25 SAMAs for consideration. PG&E analyzed this list and screened out SAMAs that would not apply to the DCPP design or that were deemed not cost beneficial based on their implementation costs and perceived dose benefits. PG&E prepared cost estimates for the remaining SAMAs and used the base risk value compared with estimated risk benefits via PRA modeling techniques to screen out SAMAs that would not be cost-beneficial.

PG&E calculated the risk reduction that would be attributable to each remaining candidate SAMA (assuming SAMA implementation) and re-quantified the risk value. The difference between the base risk value and the SAMA-reduced risk value became the averted risk, or the value of implementing the SAMA. PG&E used this information in conjunction with the cost estimates for implementing each SAMA to perform a detailed cost/benefit comparison.

PG&E performed additional analyses to evaluate how the SAMA analysis would change if certain key parameters were changed, including re-assessing the cost benefit calculations using the 95th percentile level of the failure probability distributions. The results of the uncertainty analysis are discussed in *PG&E Letter DCL-09-079, Enclosure 2,* Attachment F, Section F.7.

Based on the results of this SAMA analysis, none of the SAMAs ~~have~~ *had* a positive net value.

However, when the 95th percentile probabilistic risk analysis results ~~are~~ *were* considered, SAMAs 12, 13, 24, and 25 ~~are~~ *were* potentially cost beneficial.

- SAMA 12: Improve Fire Barriers for auxiliary saltwater and component cooling water Equipment in the Cable Spreading Room
- SAMA 13: Improve Cable Wrap for the power operated relief valves in the Cable Spreading Room

- SAMA 24: Prevent Clearing of reactor coolant system Cold Leg Water Seals
- SAMA 25: Fill or Maintain Filled The Steam Generators to Scrub Fission Products

<u>Updated SAMA Analysis</u>

By a letter dated May 2, 2014, the NRC staff advised PG&E that it would need to update the information contained in the environmental report submitted in November 2009. In response, PG&E performed an updated SAMA analysis using an updated PRA model. The updated PRA model incorporated plant design changes, an upgrade to the internal flooding analysis, and an updated fire model. In addition, the updated SAMA analysis incorporated more recent population, economic, and evacuation information and updated seismic hazard curves that considered the Shoreline fault and other regional faults. PG&E Letter DCL-15-027, Enclosure 2 amended the DCPP Environmental Report to provide the updated SAMA analysis. 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.

DCPP used industry and DCPP-specific information to create a list of 23 SAMAs for consideration. PG&E analyzed this list and screened out SAMAs that would not apply to the DCPP design or that were deemed not cost beneficial based on their implementation costs and perceived dose benefits. In addition, some SAMAs are addressed by elements of the DCPP FLEX strategy. PG&E prepared cost estimates for the remaining SAMAs and used the base risk value compared with estimated risk benefits via PRA modeling techniques to screen out SAMAs that would not be cost-beneficial.

PG&E calculated the risk reduction that would be attributable to each remaining candidate SAMA (assuming SAMA implementation) and re-quantified the risk value. The difference between the base risk value and the SAMA-reduced risk value became the averted risk, or the value of implementing the SAMA. PG&E used this information in conjunction with the cost estimates for implementing each SAMA to perform a detailed cost/benefit comparison.

PG&E performed additional analyses to evaluate how the SAMA analysis would change if certain key parameters were changed, including re-assessing the cost benefit calculations using the 95th percentile level of the failure probability distributions. The results of the uncertainty analysis are discussed in amended Attachment F, Section F.7.

Based on the results of the updated SAMA analysis, two of the SAMAs have a positive net value and are potentially cost-beneficial:

- *SAMA 3: Change procedures to explicitly address vulnerability of auto safety injection (SI)*

- *SAMA 21: Change fire procedures to include fire area specific guidance on containment isolation valves*

When the 95th percentile probabilistic risk analysis results are considered, SAMAs 8 and 16 are also potentially cost beneficial:

- *SAMA 8: Protect RHR cables in fire areas 6-A-2 and 6-A-3*

- *SAMA 16: Change procedures to caution about spurious SI signals in specific fire areas*

None of the potentially cost-beneficial SAMAs from the original SAMA analysis were found to be potentially cost-beneficial in the updated SAMA analysis. This is due to the significant changes to the PRA model that incorporated plant design changes, an upgrade to the internal flooding analysis, and an updated fire model (see Attachment F, Sections F2.1.9 and F2.1.10). Specifically, in the updated SAMA, the fire risk is dominant (numerically), while in the original SAMA, the seismic risk was dominant. In the original SAMA, the risk ranking was based on separate hazard groups (e.g., fire, seismic, internal), while in the updated SAMA, the risk ranking is based on the "combined" model (that is, a single quantification model including all hazard groups considered for the SAMA analysis). A synergistic effect of these two items is that the fire events are driving the SAMA results as shown in the final list of potentially cost-beneficial SAMAs.

While these results are believed to accurately reflect potential areas for improvement at DCPP, PG&E notes that this analysis should not necessarily be considered a formal disposition of these proposed changes, as other engineering reviews are necessary to determine the ultimate resolution. PG&E will consider ~~the four~~*new* SAMAs *3, 8, 16, and 21* using *existing action-tracking and design change processes*~~the appropriate DCPP design process~~. These SAMAs do not relate to the management of aging during the period of extended operation, and are therefore unrelated to any of the technical matters that must be addressed pursuant to 10 CFR 54.

ATTACHMENT F – SEVERE ACCIDENT MITIGATION ALTERNATIVES

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Acronyms Used in Attachment F

ADV	atmospheric dump valve
AFW	auxiliary feedwater
AOT	allowable outage time
AMSAC	anticipated transient without scram mitigating system actuation circuitry
ASME	American Society of Mechanical Engineers
ASW	auxiliary saltwater
ATWS	anticipated transient without scram
ATWT	anticipated transient without trip
B/F	bleed and feed
BNL	Brookhaven National Laboratory
BOP	balance of plant
BWR	boiling water reactor
CCP	centrifugal charging pump
CCW	component cooling water
CDF	core damage frequency
CF	containment failure
CIMIS	California Irrigation Management Information System
CRD	control rod drive
CS	containment spray
CSR	cable spreading room
CST	condensate storage tank
CT	completion time
CTE	completion time extension
CVCS	Chemical and Volume Control System
DCPP	Diablo Canyon Power Plant
DFO	diesel fuel oil
DOE	Department of Energy
ECCS	emergency core cooling system
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ESAM	estimated (equivalent) seismic action multiplier
F&O	fact and observation
FWST	fire water storage tank
GE	general emergency
HEP	human error probability
HPME	high pressure melt ejection
HRA	human reliability analysis
HVAC	heating ventilation and air-conditioning
IA	instrument air
IE	initiating event
IPE	individual plant examination
IPEEE	individual plant examination – external events
ISGTR	induced steam generator tube rupture
ISLOCA	interfacing system LOCA
LAR	license amendment request
LCV	level control valve

Acronyms Used in Attachment F

LERF	large early release frequency
LLOCA	large loss of coolant accident
LOCA	loss of coolant accident
LODI	Lagrangian operational dispersion integrator
LOOP	loss of off-site power
LTSP	long-term seismic program
MAAP	modular accident analysis program
MACCS2	MELCOR accident consequences code system, version 2
MACR	maximum averted cost-risk
MCR	main control room
MDP	motor-driven auxiliary feedwater pump
MFW	main feedwater
MLOCA	medium loss of coolant accident
MMACR	modified maximum averted cost-risk
MOV	motor operated valve
MSIV	main steam isolation valve
MSPI	mitigating systems performance index
N2	nitrogen
NCP	normal charging pump
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
OECR	off-site economic cost risk
PACR	potential averted cost-risk
PDP	positive displacement pump
PG&E	Pacific Gas & Electric
PORV	power operated relief valve
PRA	probabilistic risk analysis
PSA	probabilistic safety assessment
PTS	pressurized thermal shock
PWR	pressurized water reactor
RCP	reactor coolant pump
RCS	reactor coolant system
RDR	real discount rate
RHR	residual heat removal
RI-ISI	risk-informed in-service inspection
RITSTF	risk-informed technical specification test frequency
RM	risk management
RPV	reactor pressure vessel
RRW	risk reduction worth
RWR	raw water reservoir
RWST	refueling water storage tank
SAMA	severe accident mitigation alternative
SBO	station blackout
SSC	system, structure, component
SEIS	seismic
SER	safety evaluation report

Acronyms Used in Attachment F

SF	split fraction
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SLB	steam line break
SLOCA	small loss of coolant accident
SR	supporting requirement
SRV	safety relief valve
SSPS	solid state protection system
SWGR	switchgear
TAF	top of active fuel
TD	turbine-driven
UPS	uninterruptible power supply
VCT	volume control tank
VSLOCA	very small loss of coolant accident
WOG	Westinghouse Owners Group

SEVERE ACCIDENT MITIGATION ALTERNATIVES

The severe accident mitigation alternatives (SAMA) analysis discussed in Section 4.20 is presented below.

F.1 METHODOLOGY

The methodology selected for this analysis is contained in NEI 05-01 (Reference 13), Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document (Reference 13), which has been reviewed and endorsed by the NRC. It involves identifying SAMA candidates that have potential for reducing plant risk and determining whether or not the implementation of those candidates is beneficial on a cost-risk reduction basis. The metrics chosen to represent plant risk include the core damage frequency (CDF), the dose-risk, and the offsite economic cost-risk. These values provide a measure of both the likelihood and consequences of a core damage event.

The SAMA process consists of the following steps:

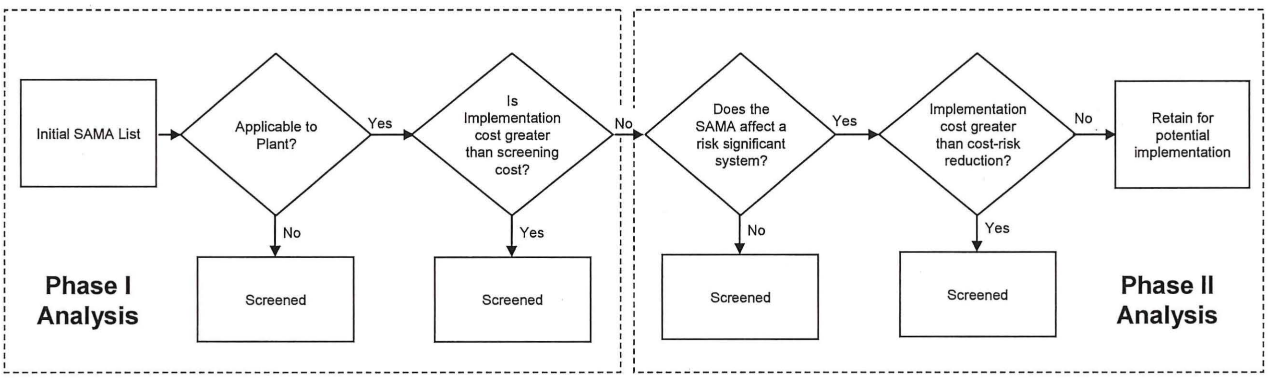
- Diablo Canyon Power Plant (DCPP) Probabilistic Risk Assessment (PRA) Model – Use the DCPP PRA model as the basis for the analysis (Section F.2). Incorporate those External Events contributions not addressed by the current PRA model as described in Section F.4.6.2.
- Level 3 PRA Analysis – Use DCPP Level 1 and 2 Internal Events PRA output and site-specific meteorology, demographic, land use, and emergency response data as input in performing a Level 3 PRA using the MELCOR Accident Consequences Code System Version 2 (MACCS2) (Section F.3). Incorporate those External Events contributions not addressed by the current PRA model as described in Section F.4.6.2.
- Baseline Risk Monetization – Use U.S. Nuclear Regulatory Commission (NRC) regulatory analysis techniques to calculate the monetary value of the unmitigated DCPP severe accident risk. This becomes the maximum averted cost-risk that is possible (Section F.4).
- Phase 1 SAMA Analysis – Identify potential SAMA candidates based on the DCPP Probabilistic Risk Assessment (PRA), Individual Plant Examination – External Events (IPEEE), and documentation from the industry and the NRC. Screen out SAMA candidates that are not applicable to the DCPP design or are of low benefit in pressurized (PWRs) such as DCPP, candidates that have already been implemented at DCPP or whose benefits have been achieved at DCPP

using other means, and candidates whose estimated cost exceeds the maximum possible averted cost-risk (Section F.5).

- Phase 2 SAMA Analysis – Calculate the risk reduction attributable to each of the remaining SAMA candidates and compare to the estimated cost of implementation to identify the net cost-benefit. PRA insights are also used to screen SAMA candidates in this phase (Section F.6).
- Uncertainty Analysis – Evaluate how changes in the SAMA analysis assumptions might affect the cost-benefit evaluation (Section F.7).
- Conclusions – Summarize results and identify conclusions (Section F.8).

The steps outlined above are described in more detail in the subsections of this attachment. The graphic below summarizes the high level steps of the SAMA process.

SAMA SCREENING PROCESS



F.2 DIABLO CANYON PRA MODEL

The SAMA analysis is based upon the 2014 DCPP PRA model (i.e., DC03 model). The original PRA model was submitted in 1988 as part of the Long-Term Seismic Program (LTSP) (Reference 30) and has been subsequently updated a number of times to maintain design fidelity with the operating plant and reflect the latest PRA technology.

The following subsections provide more detailed information related to the evolution of the Diablo Canyon Internal Events PRA model and the current results. These topics include:

- PRA changes since the IPE / IPEEE
- Level 1 model overview
- Level 2 model overview

- PRA model review summary

The CDF values for the models presented in Section F.2.1 are all point estimate values. The evaluation of base case benefits was based on point estimate values.

Sections F.4.6.2 and F.5.1.7 provide a description of the process used to integrate external events contribution into the Diablo Canyon SAMA process.

F.2.1 PRA MODEL BACKGROUND

The DCPRA-1988 model was a full-scope Level 1 PRA that evaluated internal and external events (Reference 29). The NRC reviewed the LTSP and issued Supplement No. 34 to NUREG-0675 (Reference 31) 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 38).

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 the two units in terms of their design, operation, equipment reliability and availability, was minor and did not warrant development of a separate PRA model for each unit. As such, the results and insights of the Unit 1 PRA model should be directly applicable to Unit 2 for most applications.

The Unit 1 PRA model takes credit for cross-tying the ASW system, a shared system. The detailed ASW model includes, in addition to Unit 1 components, Unit 2 pumps, valves, traveling screens, and maintenance on the Unit 2 equipment as well as the cross-tie valves. There are also separate initiating events for loss of ASW due to system faults, and loss of ASW due to flooding that fails ASW for both units. Loss of Unit 2 ASW has no effect on Unit 1 core damage results unless it is needed by Unit 1. The 4KV vital alternating current (AC) buses can also be cross-tied, which is credited in the PRA model. There are separate models for the Unit 2 vital buses and the breakers needed to cross-tie to the Unit 1 vital buses. Loss of Unit 2 vital buses has no impact on Unit 1 core damage unless it is needed by Unit 1.

The DCPRA-1988 was subsequently updated to support the Individual Plant Examination (IPE) in 1991 and the Individual Plant Examination of 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 44), 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-1988 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 year(s) prior to the establishment of the model vintage. For example, PRA model designated DCPRA-1996 was completed in 1996 but the update was performed in 1995 and 1996. In more recent updates, the updated PRA models are designated by a revision number. For example, the latest Revision 3 of the DCPRA model has been designated DC03.

The subsections below describe the DCPRA model development from the original DCPRA-1988 model to the current DCPRA model (DC03), and the revision of the Calculation File C.9 that describes the updates performed for in the PRA model.

F.2.1.1 MODEL DCPRA-1988 (LONG TERM SEISMIC PROGRAM)

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 30) 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 31), 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 percent)
- Reactor Trip (12.5 percent)
- Turbine Trip (11.2 percent)
- Partial Loss of Main Feedwater (8.4 percent)
- Loss of 1 DC Bus (7.3 percent)

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.

F.2.1.2 MODEL DCPRA-1991 (INDIVIDUAL PLANT EXAMINATION - IPE)

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 36).

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

- Reflect the 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 DCPRA_1988, and continued in DCPRA-1991:

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 than that of the original DCPRA-1988 model due to the implementation of the above improvements and the incorporation of the improvements into the PRA model. The dominant initiating event category contributors to this CDF are given below:

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

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 percent) 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 burns (26 percent).

F.2.1.3 MODEL DCPRA-1993 (INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS - IPEEE)

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 32) 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 a letter dated December 4, 1997 (Reference 40).

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

- Reflect the 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 DCPRA_1988 to make Model DCPRA-1991. These changes were subsumed into the next model revision DCPRA-1993 (IPEEE):

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. [1]
2. Revision of the 230 kV Switchyard Fragility. After the Loma Prieta earthquake, the NRC requested that PG&E reevaluate the fragility of the 230 kV switchyard based 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 were 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 500 kV and 230 kV power as well as a primary turbine building shear wall, causing the loss of all vital AC power.
- Seismic event that fails 500 kV and 230 kV 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, isolation

[1] At the time of the IPEEE submittal, the addition of the sixth EDG was ongoing and not completed. Modeling of the sixth EDG first appeared in the DCPRA-1995 and continues through the current Probabilistic Risk Assessment Model DC03.

valves and the containment heat removal capability. These SSCs have high seismic capabilities. Containment performance for fire initiators was conservatively evaluated and it was determined 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 than that identified in the IPE.

F.2.1.4 MODEL DCPRA-1995

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 incorporate (1) 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 Chemical and Volume Control System (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) reflect the train separation of the ASC split fraction. 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 model changes that had the most impact are: (1) new probability distributions of RCP seal leakage based on the qualified O-ring material, (2) update to the ASW system model, and (3) addition of the two Backup Battery Chargers 121 and 131 in the model.

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

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.

F.2.1.5 MODEL DCPRA-1997

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 model changes that had the most impact include: (1) addition of electric power recovery actors similar to LOOP initiating events to general transients with offsite power failing independently, (2) addition of the raw water storage reservoir as a backup source of water to the condensate storage tank in the AFW system model, and (3) recovery of the dedicated fuel oil transfer pumps failure if feasible.

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

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 regarding the impact on the ASW system initiated by the Loss of DC Bus F or G has caused these initiating events to increase in importance with respect to CDF contribution. This conservative modeling was removed in the next PRA model revision.

F.2.1.6 MODEL DC00

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 34) 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.[2]
- RCS Pressure Relief System. Added the third PORV (474) in Top Event PR and included 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.[2]
- 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 57)
- DC Battery Chargers (Reference 41)
- DC Batteries (Reference 41)

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[2] These DC00 model changes had the most impact to the CDF.

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

- Diesel Generators (Reference 57)
- Residual Heat Removal Pumps (Reference 57)
- Auxiliary Feedwater Pumps (Reference 57)
- Auxiliary Saltwater Pumps (Reference 57)
- Reactor Trip Breakers (Reference 39)
- RT Breaker UV Coils (Reference 39)
- RT Breaker Shunt Trip Coils (Reference 39)

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
- Catastrophic RCP Seal Failure – SELOCA

The MSRV Stuck Open initiator was deleted as a result of a review of the NRC Initiating Event Database (NUREG/CR-5750) (Reference 42). 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 percent)
- Loss of Offsite Power (16.3 percent)
- Loss of Auxiliary Saltwater (12.3 percent)
- Steam Line Break Inside Containment (10.8 percent)
- Loss of Component Cooling Water (4.5 percent)
- Loss of Switchgear Room Ventilation (3.8 percent)
- Reactor Trip (3.3 percent)
- Catastrophic RCP Seal Failure (3.0 percent)

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 LOCAs, in particular the Medium and Large LOCA were reduced due primarily to the new initiating event frequencies from NUREG/CR-5750 (Reference 42). Revision in the modeling of impact on the ASW system for loss of DC Bus F and G initiating events had also reduced 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 slightly 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 slightly lower than that from the IPEEE and is due primarily to the updated system models and the revised database used in the PRA.

F.2.1.7 MODEL DCC0

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 credited 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 500 kV 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 Top 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 model changes that had the most impact include: (1) crediting the Anticipated Transient Without Scram Mitigating System Actuation (AMSAC) to actuate the AFW system and trip the main turbine, (2) credit for manual Solid State Protection System (SSPS) for the steamline break imitators, and (3) the ability to backfeed from the 500kV switchyard and cross-tie the vital buses in accordance with the emergency operating procedure.

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 percent)
- Loss of Offsite Power (17.8 percent)
- Loss of Auxiliary Saltwater (17.4 percent)
- Loss of Common Cooling Water (6.1 percent)
- Catastrophic RCP Seal Failure (6.0 percent)

- Reactor Trip (4.2 percent)
- Medium LOCA (3.2 percent)

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 500 kV switchyard and crosstie the vital buses in accordance with the EOPs was fully implemented.
- Pre-initiator and post-initiator 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.

F.2.1.8 MODEL 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 1/1/2000 through 12/31/2004 (Reference 48) were reviewed and plant procedure revisions (then current as of 2/04/2005) were also reviewed (Reference 50). 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 day 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 more accurate modeling of the DC-AC power system interface and 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 mode. 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 Draft NUREG/CR (INEEL/EXT-04-02326) (Reference 49). 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 49) 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 motor-driven AFW pumps due to pump runout in the event of depressurization of one or more SGs due to steam line break downstream of the MSIVs.
  - Credit was given to the safety valves in the event that the 10 percent 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 11). 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 of merit (Reference 51).

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 percent)
- Flooding Scenario Failing CCW - FL1 (11.6 percent)
- Steam Generator Tube Rupture (11.2 percent)
- Loss of Offsite Power – Grid Related (7.9 percent)
- Reactor Trip (7.8 percent)
- Turbine Trip (5.8 percent)
- Partial Loss of Feedwater (4.7 percent)
- Loss of Switchgear Ventilation (4.2 percent)
- Station Blackout due to LOOP Initiating Events (6 percent)
- Anticipated Transients Without Scram (ATWS) (1.4 percent)
- Station Blackout due to non-LOOP Initiating Events (1.3 percent)

There is an increase in the Internal Events CDF of approximately 4 percent from the previous 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 46). This is 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