Diablo Canyon Power Plant Unit 2

10 CFR 50.55a Request for Approval of Alternative - RI-ISI-2

Diablo Canyon Power Plant Unit 2
10 CFR 50.55a Request Number RI-ISI-2

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(1)

-Alternative Provides Acceptable Level of Quality and Safety-

I. ASME Code Components Affected

Code Class 1 and 2 piping welds previously subject to the requirements of American Society of Mechanical Engineers (ASME) Section XI, Table IWB-2500-1, Examination Categories B-F* and B-J, and Table IWC-2500-1, Examination Categories C-F-1 and C-F-2, are affected.

II. Applicable Code Edition and Addenda

The Diablo Canyon Power Plant (DCPP) Unit 2 Inservice Inspection (ISI) program for the fourth ISI interval is based on the 2007 Edition of ASME Section XI through the 2008 Addenda.

III. Applicable Code Requirement

The selection of Code Class 1 and Code Class 2 pipe welds to be examined in the fourth inspection interval is required to be prescriptively determined in accordance with Table IWB-2500-1, Examination Categories B-F* and B-J, and Table IWC-2500-1, Examination Categories C-F-1 and C-F-2.

IV. Reason For Request

The continued use of a risk-informed process as an alternative for the selection of Class 1 and Class 2 piping welds for examination is requested for the fourth ISI Interval of Unit 2. Use of the risk-informed selection process has been shown to reduce the core damage frequency and large early release frequency when compared to the prescriptive deterministic selection method.

** Note that although Examination Category B-F welds are included in the RI-ISI program for other damage mechanisms, Alloy 600/82/182 examinations in the Third Interval were conducted per Code Cases N-722-1 and N-770-1. In the fourth interval, these examinations will be performed in accordance with the versions of the applicable Code Cases that are referenced in the published version of 10 CFR 50.55a.*

V. Proposed Alternative and Basis for Use

As an alternative to the Code Requirement, a risk-Informed process will continue to be used for selection of Class 1 and Class 2 piping welds for examination.

The DCPP Unit 2 ISI program for examination of Class 1 and Class 2 piping welds is currently in accordance with a risk-informed process developed and based on EPRI TR-112657, Revision B-A, with identified differences and with additional guidance taken from ASME Code Case N-578. In 2001, DCPP submitted a request for alternative in PG&E letter DCL-01-015, "Relief Request for Application of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 and 2 Piping Welds," dated February 16, 2001 (Examination Categories B-F, B-J, C-F-1, and C-F-2) inservice inspections to implement a risk-informed inservice inspection (RI-ISI) program. The NRC published a safety evaluation authorizing the use of the RI-ISI program for the second 10-year ISI interval for DCPP Units 1 and 2. Both the original RI-ISI submittal and the resultant NRC Safety Evaluation call for a periodic review and update of the program. An update was performed for the end of the third period of the second interval. Based on that update, another request for alternative for the third ISI interval was submitted in PG&E Letter DCL-12-007, "Request for Approval of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 and 2 Piping Welds," dated January 20, 2012. DCL-12-007 was supplemented by PG&E Letter DCL-12-084, "Response to NRC Request for Additional Information Regarding Request for Approval of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 and 2 Piping Welds," dated September 6, 2012. This request was approved for the entire third interval. The resultant program was implemented for the third interval, and was reviewed and updated after the first, second and third periods of the third interval.

In accordance with NEI 04-05 (April 2004), the following aspects were considered during the reviews:

- Plant Examination Results
- Piping Failures
  -Plant Specific Failures
  -Industry Failures
- PRA Updates
- Plant Design Changes
  -Physical Changes
  -Programmatic Changes
  -Procedural Changes
- Changes in Postulated Conditions
  -Physical Conditions
  -Programmatic Conditions

The updated program resulting from these reviews is the subject of this proposed alternative.

In accordance with the guidance provided by NEI 04-05, a table is provided as Attachment 1 identifying the number of welds added to and deleted from the previously approved RI-ISI program. The changes from the previous program are attributable to the specific issues(s) identified in each review:

During the review after the first period of the third ISI interval, the following issues were identified:

1. In the chemical and volume control system (CVCS), valves CVCS-2-8372A, B, C, and CVCS-2-8367 A, B, C, and CVCS-2-8479-A, B were replaced. In the reactor coolant system (RCS), pressurizer nozzle safe end welds received weld overlays. Multiple welds were deleted, added, or renamed as a result of steam generator (SG) replacement, centrifugal charging pump replacement, and positive displacement pump replacement. As a result, there were multiple changes to the weld population.

2. Based on a change to ASME Section XI Code criteria, the 4-inch nominal pipe size (NPS) Class 2 auxiliary feedwater (AFW) lines from the level control valves to their respective connections to the four main feedwater lines were added to the RI-ISI Program.

3. Six weld overlays were installed as a result of implementation of the RCS Alloy 600 Program. Due to the proximity of adjacent welds, these overlays actually overlaid 12 welds.

During the review after the second period of the third ISI interval, the following issues were identified:

1. The DCPP probabilistic risk assessment (PRA) model used to evaluate the consequences of pipe rupture for the previous RI-ISI update was Model DC01 dated June 2006. Model DC01 was still the model of record during the period under evaluation. As such, there was no change required to any consequence analysis or to the upper bound conditional core damage probability (CCDP) or large early release probability (LERP). However, the model of record (MOR) changed to DC02 in November of 2012. PG&E decided to proactively reflect this change as part of the Interval 3, Period 2 evaluation. For this model the core damage frequency (CDF) is 6.91E-05/yr and large early release frequency (LERF) is 3.17E-06/yr. Maximum CCDP used as the upper bound in the risk impact analysis is 3.98E-02 associated with Consequence Cases CVCS-1,

RCS-1, and SI-3. The update to the PRA model resulted in the following changes in consequence rankings:

Consequence ID	DC01 Rank	DC02 Rank	Change in Consequence Rank
ACC02A	M	H	Medium to High
ACC02B	M	H	Medium to High
ACC02C	M	H	Medium to High
ACC02D	M	H	Medium to High
CS01	M	H	Medium to High
CS02	M	H	Medium to High
CS03A	M	H	Medium to High
CS04A	M	H	Medium to High
CS03B	M	H	Medium to High
CS04B	M	H	Medium to High
CVCS05B	M	L	Medium to Low
CVCS07	M	L	Medium to Low
CVCS08	M	L	Medium to Low
CVCS09	M	L	Medium to Low

2. During the first period of the third ISI interval, the RI-ISI Program was subjected to an extensive review and verification. During the second period of the third ISI interval, the updated risk ranking, summary, and matrix were used to reflect the resulting findings and reconciliations.

3. During the element selection process, it was noted that the four welds in CVCS Risk Category 5a and subject to thermal stratification, cycling, and striping (TASCS) were all single-sided welds and none could be properly examined. Since only one weld was required to be inspected, a weld in the same system with the same degradation mechanism, but a higher Risk Category, was selected as a substitute. In Unit 2, S6-50-3-WIB-186 was selected.

During the review of the third period of the third ISI interval, the following issues were identified:

1. During the Unit 2 Nineteenth Refueling Outage, an indication was found in Class 1 Weld WIB-245. Several crack growth and degradation mechanisms were investigated as part of that evaluation since the cracking did not conform to any known industry operating experience. The specific mechanisms considered include thermal shock from cyclic swirl penetration and cyclic thermal stratification of the unisolable horizontal pipe section and stress corrosion cracking. Temperature monitoring was conducted to further refine the analysis. Vibration was ruled out as a possible cause based on inspections performed by

PG&E, which concluded that physical evidence indicative of excessive vibration was not present.

Because the evaluation could neither rule out nor specify the specific flaw growth mechanism, a conservative analysis was performed combining the effects of two different flaw growth mechanisms comprised of fatigue crack growth (FCG) and stress corrosion cracking (SCC), for justification of continued operation for an additional cycle. PG&E is not attributing SCC with respect to risk-informed ISI weld inspections due to uncertainty regarding the flaw growth mechanism.

2. The DCPP PRA was updated to Model DC03 in July 2015. In Model DC03, the total CDF is 5.52E-05/yr and the total LERF is 5.73E-06/yr. The maximum CCDP used as upper bound in the risk Impact analysis is 1.74E-02 and the maximum conditional large early release probability (CLERP) is 7.03E-03, both associated with Consequence Cases CVCS-1, RCS-1, and SI-3. The update in PRA model resulted in the following changes in consequence rankings:

Consequence ID	DC02 Rank	DC03 Rank	Change in Consequence Rank
ACC02A	H	M	High to Medium
ACC02B	H	M	High to Medium
ACC02C	H	M	High to Medium
ACC02D	H	M	High to Medium
CS01	H	M	High to Medium
CS02	H	M	High to Medium
CS03A	H	M	High to Medium
CS04A	H	M	High to Medium
CS03B	H	M	High to Medium
CS04B	H	M	High to Medium
CVCS01B	M	L	Medium to Low
CVCS02B	M	L	Medium to Low
RHR01	L	M	Low to Medium
RWST02A-PEN	M	H	Medium to High
RWST02B-PEN	M	H	Medium to High
RWST03A	M	H	Medium to High
RWST03B	M	H	Medium to High
SI01	M	H	Medium to High
SI02	M	H	Medium to High
SI03A	M	H	Medium to High
SI03B	M	H	Medium to High

All issues identified in the periodic reviews have been incorporated into the risk ranking, summary, and matrix. Limits are imposed by the EPRI methodology to ensure that the change in risk of implementing the RI-ISI program meets the requirements of

Regulatory Guides 1.174 and 1.178. The EPRI criterion requires that the cumulative change in CDF and LERF be less than 1E-07 and 1E-08 per year per system, respectively. A new risk impact analysis was performed, and the revised program continues to represent a risk reduction when compared to the last deterministic Section XI inspection program. The revised program represents an overall reduction of plant risk of 4.96E-08 inCDF and 2.00E-08 in LERF.

As indicated in the following table, this evaluation has demonstrated that unacceptable risk impacts will not occur for any system from implementation of the RI-ISI program regardless of whether the enhanced probability of detection (POD) is credited for the RI-ISI examinations.

Unit 2 Risk Impact Results

System	$\Delta Risk_{CDF}$		$\Delta Risk_{LERF}$	
	w/ POD	w/o POD	w/ POD	w/o POD
RCS	-1.66E-08	1.48E-09	-6.71E-09	5.98E-10
CVCS	-7.13E-09	-4.35E-09	-2.88E-09	-1.76E-09
SIS	-1.59E-08	-8.94E-09	-6.43E-09	-3.62E-09
RHRS	-9.65E-09	-4.78E-09	-3.90E-09	-1.93E-09
CSS	2.51E-12	2.51E-12	2.51E-13	2.51E-13
RWST	-2.61E-10	-2.61E-10	-1.05E-10	-1.05E-10
CCW	0.00E+00	0.00E+00	0.00E+00	0.00E+00
FWS	3.80E-13	6.20E-13	3.80E-14	6.20E-14
MSS	7.50E-14	7.50E-14	7.50E-15	7.50E-15
AFW	1.65E-13	2.45E-13	1.65E-14	2.45E-14
Total	**-4.96E-08**	**-1.68E-08**	**-2.00E-08**	**-6.82E-09**

The following augmented inspection programs were considered during the RI-ISI application:

- The augmented examination program for flow accelerated corrosion (FAC) per NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," dated May 2, 1989, is relied upon to manage this damage mechanism but is not otherwise affected or changed by the RI-ISI program.
- The augmented examinations for thermal fatigue in non-isolable reactor coolant system branch lines are performed in accordance with EPRI Materials Reliability Program document, MRP-146, which is relied upon to manage this damage mechanism but is not otherwise affected or changed by the RI-ISI program.
- The augmented visual examinations for pressure retaining welds in Class 1 components fabricated with Alloy 600/82/182 materials are performed in accordance with Code Case N-722-1, which is relied upon to manage the damage mechanism of primary water stress corrosion cracking (PWSCC) but is not otherwise affected or changed by the RI-ISI program.

- The augmented examinations and acceptance standards for Class 1 piping and vessel nozzle butt welds fabricated with UNS N06082 or UNS W86182 weld filler metal are performed in accordance with Code Case N-770-1 which is relied upon to manage the damage mechanism of PWSCC but is not otherwise affected or changed by the RI-ISI program. Note that welds selected for examination in accordance with Code Case N-770-1 are considered as part of the RI-ISI population such that they are evaluated for other potential degradation mechanisms. However, they are excluded from selection under the RI-ISI Program. In the fourth interval these examinations will be performed in accordance with the version of Code Case N-770 that is referenced in the published version of 10 CFR 50.55a. This is expected to be Code Case N-770-2 per the Notice of Proposed Rulemaking dated September 18, 2015.

The RI-ISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. As a minimum, risk ranking of piping segments will be reviewed and adjusted on an ASME period basis. In addition, significant changes may require more frequent adjustment as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant specific feedback.

The risk-informed process continues to provide an adequate level of quality and safety for selection of the Class 1 and Class 2 piping welds for examination. Therefore, pursuant to 10 CFR 50.55a(z)(1), PG&E requests that the proposed alternative be authorized.

VI. PRA Quality

The PRA Quality Assessment is provided in Attachment 2.

VII. Duration of Proposed Alternative

The alternative will be used for DCPP Unit 2 until the end of that unit's fourth 10-year ISI Program inspection interval, subject to the review and update guidance of NEI 04-05. The fourth inspection interval is currently scheduled to end on March 13, 2026.

DCPP Unit 2 - Inspection Location Selection Comparison Between Previously Approved and Revised RI-ISI Program by Risk Category												
System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Previously Approved (Third Interval)			Updated (Fourth Interval)		
	Category	Rank		DMs	Rank		Weld Count	RI-ISI	Other[2]	Weld Count	RI-ISI	Other[2]
RCS	2	High	High	TASCS, TT	Medium	B-J	9[3]	6[3]		9[3]	5[3]	
RCS	2	High	High	TASCS	Medium	B-J	10	4		10	4	
RCS	2 (2)	High (High)	High	TT (PWSCC)	Medium (Medium)	B-F	1[4]	0[4]		1[4]	0[4]	
RCS	2	High	High	TT	Medium	B-J	13	0		13	0	
CVCS	2	High	High	TASCS, TT	Medium	B-J	5	3		5	2	
CVCS	2	High	High	TT	Medium	B-J	3	1		3	0	
SIS	2	High	High	TT	Medium	B-J	18	6		18	5	
RHR	2	High	High	TASCS	Medium	C-F-1	11	3		11	3	
RCS	4 (2)	Medium (High)	High	None (PWSCC)	Low (Medium)	B-F	0[4]	0[4]		13[4]	0[4]	
RCS	4	Medium	High	None	Low	B-F	21	2		8	0	
						B-J	277	34		285	31	
CVCS	4	Medium	High	None	Low	B-J	92	11		92	10	
CVCS	4	Medium	High	None	Low	C-F-1	21	2		22	0	
SIS	4	Medium	High	None	Low	B-J	30	4		30	11	
SIS	4	Medium	High	None	Low	C-F-1	68	7		131	6	
RHR	4	Medium	High	None	Low	C-F-1	175	18		175	18	
RWST	4	Medium	High	None	Low	C-F-1	45	5		117	12	
CCW	4	Medium	High	None	Low	C-F-2	12	2		12	2	

DCPP Unit 2 - Inspection Location Selection Comparison Between Previously Approved and Revised RI-ISI Program by Risk Category												
System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Previously Approved (Third Interval)			Updated (Fourth Interval)		
	Category	Rank		DMs	Rank		Weld Count	RI-ISI	Other[2]	Weld Count	RI-ISI	Other[2]
CVCS	5a	Medium	Medium	TASCS, TT	Medium	B-J	2	1		0	0	
CVCS	5a	Medium	Medium	TT	Medium	B-J	2	0		0	0	
SIS	5a	Medium	Medium	IGSCC	Medium	B-J	13	2		13	2	
SIS	5a	Medium	Medium	TASCS	Medium	C-F-1	4	0		4	1	
RCS	6a	Low	Medium	None	Low	B-J	3	0		3	0	
CVCS	6a	Low	Medium	None	Low	B-J	8	0		8	0	
CVCS	6a	Low	Medium	None	Low	C-F-1	673	0		0	0	
SIS	6a	Low	Medium	None	Low	B-J	134	0		134	0	
SIS	6a	Low	Medium	None	Low	C-F-1	160	0		68	0	
RHR	6a	Low	Medium	None	Low	B-J	20	0		20	0	
RHR	6a	Low	Medium	None	Low	C-F-1	85	0		85	0	
CSS	6a	Low	Medium	None	Low	C-F-1	72	0		72	0	
RWST	6a	Low	Medium	None	Low	C-F-1	72	0		0	0	
CVCS	6b	Low	Low	TASCS, TT	Medium	B-J	0	0		2	0	
CVCS	6b	Low	Low	TT	Medium	B-J	52	0		54	0	
SIS	6b	Low	Low	IGSCC	Medium	B-J	7	0		7	0	
AFW	6b	Low	Low	TT	Medium	C-F-2	15	0		15	0	
FWS	6b (5b)	Low (Medium)	Low	TASCS (FAC)	Medium (High)	C-F-2	28	0		32	0	

DCPP Unit 2 - Inspection Location Selection Comparison Between Previously Approved and Revised RI-ISI Program by Risk Category												
System(1)	Risk		Consequence Rank	Failure Potential		Code Category	Previously Approved (Third Interval)			Updated (Fourth Interval)		
	Category	Rank		DMs	Rank		Weld Count	RI-ISI	Other(2)	Weld Count	RI-ISI	Other(2)
RCS	7a	Low	Low	None	Low	B-J	13	0		13	0	
CVCS	7a	Low	Low	None	Low	C-F-1	0	0		748	0	
SIS	7a	Low	Low	None	Low	B-J	216	0		216	0	
SIS	7a	Low	Low	None	Low	C-F-1	9	0		34	0	
CSS	7a	Low	Low	None	Low	C-F-1	12	0		12	0	
MSS	7a	Low	Low	None	Low	C-F-2	118	0		122	0	
AFW	7a	Low	Low	None	Low	C-F-2	130	0		130	0	
FWS	7a (5b)	Low (Medium)	Low	None (FAC)	Low (High)	C-F-2	37	0		37	0	

Notes

1. Systems were described in Table 3.1-2 of the original submittal (PG&E Letter DCL-01-015, dated February 16, 2001), with the exception of AFW. This ASME Code Class 2 system consists of 145 elements.
2. The column labeled "Other" is generally used to identify augmented inspection program locations that are credited beyond those locations selected per the RI-ISI process, as addressed in Section 3.6.5 of EPRI TR-112657, Rev. B-A. This option was not applicable for the DCPP RI-ISI application. The "Other" column has been retained in this table solely for uniformity purposes with other RI-ISI application template submittals.
3. One of the elements selected for RI-ISI is the surge line elbow and is not counted as part of the weld count.
4. The examinations for these welds are performed in accordance with Code Case N-770-1, which is relied upon to manage the damage mechanism of PWSCC but is not otherwise affected or changed by the RI-ISI program. Note that welds selected for examination in accordance with Code Case N-770-1 are considered as part of the RI-ISI population such that they are evaluated for other potential degradation mechanisms. However, they are excluded from selection under the RI-ISI program. In the fourth interval, these examinations will be performed in accordance with the version of Code Case N-770 that is referenced in the published version of 10 CFR 50.55a. For the fourth interval, these welds have been re-categorized in the RI-ISI application for ease of identification.

Attachment 2
PRA Technical Adequacy for RI-ISI Application

As discussed in the NRC safety evaluation of EPRI TR 1021467 and PG&E's response to RAI Question #7 for approval of the RI-ISI third interval as documented in PG&E Letter DCL-12-84, the impact of the external event PRAs do not significantly impact the RI-ISI application. Therefore the following DCPP PRA development history and technical adequacy is focused on the Internal Events and Internal Flooding PRAs.

A.1 History of DCPP PRA Model Development

The current DCPP PRA model is based on the original 1988 Diablo Canyon PRA (DCPRA -1988) model, developed as part of the Long-Term Seismic Program (LTSP). 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 in June 1991, accepting the DCPRA-1988. Brookhaven National Laboratory performed the primary review of the DCPRA-1988 for the NRC; their review is documented in NUREG/CR-5726.

The DCPRA-1988 was subsequently updated to support the Individual Plant Examination in 1991 and the Individual Plant Examination for External Events in 1993. Since 1993, several other updates have been made to incorporate plant and procedure changes, update plant-specific reliability and unavailability data, and to improve the fidelity of the model.

At the time the fourth RI-ISI consequence case ranking evaluation process started, the MOR was DC03. DC03 incorporated the resolution of 2012 Internal Events and Internal Flooding Peer Review facts and observations (F&Os) along with a routine data update. The fourth RI-ISI consequence case ranking is based on quantitative risk insights from MOR DC03. The latest MOR is DC03A which was updated in 2016 and incorporates Westinghouse safe shutdown reactor coolant pump (RCP) seal modeling into the internal events model. The DC03A update is not expected to impact the results of the consequence case ranking because the changes made in DC03A did not significantly influence the CCDPs, and initiating event frequencies used in the RI-ISI evaluation.

A.2 Internal Events and Internal Flooding PRA Peer Review

The DCPP Internal Events and Internal Flooding PRA had a full scope peer review in accordance with NEI guidance. This review was conducted in December, 2012. The peer review was done in accordance with Capability Category II requirements of the ASME/ANS RA-Sa-2009 Standard as endorsed by RG 1.200, Revision 2, with the full consideration of NRC regulatory positions described in Appendix A, B, and C.

The peer review found the Internal Events PRA and Internal Flooding model to be technically adequate. The results of this peer review, including (F&O) resolutions and impact on this RI-ISI alternative request submittal, are summarized in Table A-1 for Interval Events and Table A-2 for Flooding.

A.3 Review of Modeling Uncertainties

Table A4-2 in PRA Calculation C.10 Revision 7, "PRA Technical Adequacy," dated March 2016, provides a list potential modeling uncertainties and their characterization. The review of this table identified no key modeling uncertainty that could impact either the consequence analysis or risk ranking requiring changes to the model or sensitivity analysis.

A.4 PRA Maintenance and Upgrade

The PG&E risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the DCPP risk management program and associated procedures. These procedures delineate the responsibilities and guidelines for maintaining the PRA models at DCPP.

A.5 Conclusion

DCPP Internal Events and Internal Flooding PRAs have been developed, refined, and maintained to reflect the as-built/as-operated condition of the plant per applicable industry guidance documents and PG&E administrative procedures. The Internal Events and Flooding PRAs have been peer reviewed to the latest PRA standard as endorsed by RG 1.200, Revision 2. All F&Os from the peer reviews were satisfactorily resolved and there is no open issue that could impact the results of this analysis.

DCPP Internal Events and Internal Flooding PRAs are technically adequate to support the RI-ISI alternative request.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
IE-A5	IE-A5-01 (Systematic review of each system) IE-A5 not met	F	Closed	There is no evidence in the documentation of a systematic evaluation of every system to assess the possibility of an initiating event occurring due to failure of the system.	This F&O has been resolved by additional reviews; no new or changed initiating events were identified. Each system was screened for potential initiating events. If a system did not screen, it was then reviewed to confirm that a bounding or representative initiating event is already modeled in the PRA. An interview with an Operations representative was conducted to confirm the system screening and to discuss low power or non-power operations for each system. This supporting requirement (SR) is judged to now be met at capability category II, based on the use of a structured approach for evaluation of each system for initiating event potential.
IE-A7	IE-A7-01 (Events which occurred other than at-power) IE-A7 not met Associated SRs: IE-A8 met at capability category I IE-A9 met at capability category I	F	Closed	The identification of initiating events does not include consideration of events occurring during low-power or shutdown conditions, and events which result in a controlled shutdown leading to a scram prior to reaching low-power conditions as specified in the standard. A review of historical events, plant operating history, and interviews with plant personnel are also required by the standard.	This F&O has been resolved by additional reviews; no new or changed initiating events were identified. A re-review of plant information in the Twice-Daily Shift Manager Turnover Reports, On-line/Off-line Daily Log, and Outage History was conducted to identify potential initiating events. Low power and non-power operation events were discussed as part of the system screening performed to resolve F&O Internal Event (IE)-A5, discussed above. This SR is judged to now be met, based on consideration of shutdown and low power events and unplanned shutdowns. Associated SRs IE-A8 and IE-A9 are also judged to be met at capability category II based on interviews having been conducted, and on review of operating history for precursor events.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
IE-C5	IE-C5-01 (Initiating event frequency based on a reactor year basis) IE-C5 not met	F	Closed	Initiating event frequencies are converted to events per calendar year by multiplying by the site critical hours per calendar year factor calculated from site operating experience, instead of a unit-specific factor as required by the standard. This distinguishes differences in the plant units' operating experience.	An assessment was performed to determine whether use of unit specific initiating event frequencies would have an impact on applications. The conclusion of this assessment was that the difference in CDF and LERF are negligible and would not impact the results of any risk-informed applications.
IE-C10	IE-C10-01 (Combination of component failure with the unavailability of other components) IE-C10 met	F	Closed	Use of plant specific information, including common cause failure (CCF) treatment, plant-specific data, repair times, and the applicability of mitigating function success criteria in the initiating event fault tree was not evident.	This F&O was resolved by additional review and model update if required. A summary review of the initiating event fault trees indicates that plant-specific information, including CCF treatment, plant-specific data, repair times, and the applicability of the mitigating function success criteria are currently used in the PRA model. A detailed review was performed and documented to confirm that all the required plant-specific information is included in the initiating event fault trees.
IE-C14	IE-C14-01 (Interfacing systems loss-of-coolant accident (ISLOCA) frequency) IE-C14 not met	F	Closed	There is no documented systematic review of all containment penetrations for potential ISLOCAs, including identification of screened penetrations and the basis for screening, and relevant surveillance test procedures and their impact on the potential for an ISLOCA.	A table listing the containment penetrations and disposition regarding their potential as an ISLOCA pathway was developed. A set of screening criteria was developed consistent with the SR requirement. These criteria were used explicitly to screen each potential ISLOCA pathway. The unscreened ISLOCA flow paths are consistent with what is modeled in RISKMAN. Also, impact of surveillance testing was added to the documentation.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
IE-C15	IE-C15-01 (Uncertainty associated with initiating events) IE-C15 not met Associated SR: IE-C1 met	F	Closed	No discussion of uncertainty parameters for initiating event fault trees was identified.	Parametric uncertainty for IE frequencies is given in the DCPP PRA documentation as Range Factors (Error Factors) for loss of cooling accident (LOCA) IEs and alpha/beta values for gamma distributions.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
IE-D1	IE-D1-01 (Documentation) ID-D1 not met Associated SRs: IE-D2 not met IE-D3 not met AS-C1 not met SY-C1 not met DA-E1 not met QU-F1 not met LE-G1 not met IFPP-B1 not met IFSO-B1 not met IFSN-A5 met IFSN-B1 not met IFQU-B1 met	F	Closed	The documentation is not written in a manner that facilitates PRA applications, upgrades, and peer review. The peer review team identified that the existing documentation heavily references the original DCPP PRA documents, especially PLG-0637. This makes it difficult to understand details of the model, difficult to confirm that the model addresses PRA requirements, and difficult to update and use it for PRA applications. This finding applies to other elements of the standard besides IE.	References to PLG-0637 as the basis have been taken out and information has been included in the new calculation revisions for system notebooks, initiating event notebooks, event tree notebooks, and other PRA development documentation.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
IE-D2	IE-D2-01 (Documentation) IE-D2 not met Associated SRs: IE-A3 met IE-A10 met IE-B3 met at capability category II IE-C2 met IE-C3 met IE-C4 met IE-C8 met IE-C9 met IE-C10 met IE-C12 met IE-D1 not met	F	Closed	The peer review team identified specific examples of deficiencies in the documentation of initiating events which need to be addressed, including specific references missing, addressing dual unit loss of instrument air as an initiating event, identification of "freeze dates," identification of credited operator recovery actions, details of uncertainty parameters and Bayesian updating of data, details of initiating event fault trees (see IE-C10-01), and comparison to generic data sources.	All identified initiating event documentation deficiencies were addressed in the most recent model update.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
AS-A11	AS-A11-01 (Transfer between event trees and preserving dependencies) AS-A11 met	S	Closed	This F&O is a suggestion that the event tree transfers would be more easily followed if they were explicitly given in the event trees.	The transfers between event trees is statically set by the initiators in RISKMAN. By looking at the initiator, it is clear how the event trees are link and the order that they transfer.
AS-B3	AS-B3-01 (Phenomenological conditions created by accident progressions) AS-B3 not met Associated SRs: AS-B3 not met SY-A18 met SY-A21met SY-A23 met SY-B14 (met)	F	Closed	There does not appear to be a review of phenomenological conditions created by each accident sequence; thus, there may be non-safety related components that are affected by an accident sequence that were not reviewed for the accident impact on the functionality of the component.	A review of phenomenological conditions was performed for all of the initiating events in the DCPP PRA. This review was documented in Calculation I.1. As a result of this review several changes were made to the DCPP PRA model to correctly account for the phenomenological conditions.
AS-B7	AS-B7-01 (Time-phased dependencies) AS-B7 met	F	Closed	Time-phased dependencies were found to be modeled in the accident sequences (e.g., AC power recovery and DC battery depletion.) However, the documentation has inconsistencies that need to be resolved.	Documentation was reviewed and inconsistencies were identified and corrected.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
AS-C2	AS-C2-01 (Documenting processes used to develop accident sequences) AS-C2 not met	F	Closed	The processes used to develop accident sequences are not sufficiently documented, as noted in F&Os AS-A11-01 and AS-B7-01, which identify issues related to the documentation of the accident sequence analyses.	See AS-A11-01 and AS-B7-01 for resolutions.
SC-A1	SC-A1-01 (Definition of core damage) SC-A1 not met Associated SR: SC-A2 not met	F	Closed	Two definitions of core damage are used in the documentation. The first definition, peak node temperature >1800°F, is a valid success criterion, and meets the definition in Section 1-2 of the standard. However, the second criterion of "the time until the water level is collapsed below the top of active fuel" is not a valid definition since the definition of core damage as written in Section 1-2 requires the consideration of uncovery and heat-up, and this definition does not consider heat-up.	The definition of core damage dependent on collapsed water level was removed from the documentation. Modular Accident Analysis Program (MAAP) runs were updated using the core damage definition of > 1800°F peak fuel temperature.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
SC-A4	SC-A4-01 (Shared systems between units) SC-A4 not met	F	Closed	The identification of shared systems between the units and how they are credited is not documented. For example, no discussion on the diesel fuel oil (DFO) transfer system is provided, although it is a known shared system. This is significant to ensure that a shared system is not inadvertently credited for both units simultaneously if the system does not have that capacity.	This F&O has been resolved by further evaluation; no PRA model changes were required. A review of the mitigating systems credited in the PRA model for dual-unit initiators identified only the DFO transfer system as a shared mitigation system not specifically evaluated. Other shared systems were identified correctly. The model correctly credits the DFO system with consideration made that both units are impacted. With this F&O resolved, SR SC-A4 is met.
SC-A5	SC-A5-01 (Mission times) SC-A5 not met	F	Closed	No discussion could be found that verified that each accident sequence actually reached a safe stable state at the minimum specified mission time of 24 hours.	MAAP Calculations were reviewed and run past 24 hours to verify that a safe stable state was achieved. Residual heat removal (RHR) entry conditions were also reviewed and verified for the applicable accident sequences. With this F&O resolved, along with additional F&O SC-A5-02 (see below), SR SC-A5 is met at capability category II/III.
SC-A5	SC-A5-02 (Mission times) SC-A5 not met	F	Closed	Several accident sequences were identified where RHR entry conditions were met prior to 24 hours, but RHR was not required for success in the accident sequence. If RHR is not questioned, then the end state may not be stable since heat removal via the SGs will be diminished as decay heat lowers, and RHR will be required to maintain temperatures long term.	See response to F&O SC-A5-01 (above).

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
SC-B3	SC-B3-01 (LOCA break sizes) SC-B3 not met Associated SRs: SC-B1 met at capability category II IE-B4 met IE-C1 met IE-C13 met at capability category I/II	F	Closed	The current success criterion for LOCAs is based on plant capabilities and system responses. The specific break sizes associated with the transitions between the LOCA definitions have not been adequately justified by specific thermal-hydraulic evaluations.	This F&O was resolved by an update to initiating event frequencies. Additional analyses have been performed and break sizes have been identified. The medium LOCA transition size was updated, and the frequencies of LOCAs adjusted. Upon resolution of this F&O, and additional F&O SC-B3-02 (below), the SR SC-B3 will be met.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
SC-B3	SC-B3-02 (ISLOCA sizes) SC-B3 not met Associated SR: SC-B1 met at capability category II	F	Closed	The thermal-hydraulic analysis for ISLOCA referenced for the success criteria validation is based on an 8-inch break size, and not on a 2-inch break size. The use of an 8-inch break size is inappropriate because the required equipment and timing associated with responding to a 2-inch break would be significantly different than the required equipment and timing associated with an 8-inch break. In addition, the RHR pumps are assumed to be unavailable based on conservative assumptions related to the effects of the ISLOCA; more realistic assumptions should be applied.	This F&O was resolved by conducting additional analyses to validate or revise the current ISLOCA break sizes and corresponding success criteria and plant impacts. Documentation was updated to properly identify and validate assumptions on impacts to the RHR pumps. Upon resolution of this F&O, and additional F&O SC-B3-01 (above), the SR SC-B3 will be met.
SC-B4	SC-B4-01 (Define large break LOCAs) SC-B4 met	F	Closed	The analysis code used to establish success criteria has known limitations with respect to its modeling of large LOCAs. The limitations of the code are not summarized anywhere in the analyses, so it is not clear that the limitations of the code were considered when developing the success criteria.	This F&O has been resolved by additional reviews; no model updates were required. The success criteria from the design basis analysis are consistent with the PRA success criteria for large LOCAs.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
SC-B4	SC-B4-02 (Anticipated Transient Without Trip (ATWT) definition) SC-B4 met	F	Closed	The discussions associated with the ATWT scenarios and the success criteria for ATWT are not consistent in the documentation with regards to parameters relevant to ATWT events. The actual criteria for plant-specific ATWT conditions needs to be defined, justified, and evaluated for system response required to mitigate the ATWT.	Documentation was updated to be consistent with the model.
SC-B5	SC-B5-01 (Crediting PORVs for depressurization when AFW not available) SC-B5 met	F	Closed	In the documentation of the comparison of success criteria to similar plants, one outlier was noted in the success criteria for a small LOCA without AFW available. This is assumed to result in core damage, but the use of power-operated relief valves (PORVs) to depressurize and cooldown is credited at similar plants. The basis for not crediting the use or PORVs is not documented, and discussions with plant PRA personnel did not identify any reason that the PORVs could not be credited at DCPP.	The impact of not crediting feed and bleed for small LOCA scenarios was determined to be approximately 1E-8/yr CDF. Although the risk benefit for this credit is not significant, it could contribute some risk benefit in certain configurations, such as an AFW pump being inoperable. Therefore, the DCPP PRA model has been updated to ensure that small LOCA scenarios correctly credit the use of feed and bleed when appropriate.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
SC-C2	SC-C2-01 (Unclear process of developing success criteria) SC-C2 not met	F	Closed	The process followed for developing the success criteria for each accident scenario is not clearly documented. For example, there are two definitions of core damage used, the basis for the timing of human actions is not clear (two criteria used - but nothing showing why both are acceptable), the limitations of the software used for the success criteria is not documented, etc.	Removed the collapsed water level definition of core damage and now use peak node temperature of greater than 1800°F. Limitations of computer codes addressed in SC-B4-01. Impact of ATWT success criteria addressed in SC-B4-02.
SC-C3	SC-C3-01(Documenting sources of uncertainty) SC-C3 not met Associated SRs: IE-D3 not met SY-C3 not met	F	Closed	A review of many of the PRA elements identified that there was not summarization of the sources of uncertainty or assumptions associated with the individual PRA element.	This F&O has been resolved by a documentation update. Each PRA element calculation has been reviewed and the assumptions and sources of uncertainty have been documented. With this F&O resolved, SC-C3 is met.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
SY-A4	SY-A4-01 (Walkdowns and interviews) SY-A4 not met	F	Closed	Neither plant walkdowns nor interviews with knowledgeable plant personnel were performed to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	This F&O has been resolved by providing additional evidence that confirms the system analyses were correctly developed, refined and maintained to reflect the as-built and as-operated plant. Based on the maturity of the system models and their ongoing application at the plant, it is judged unlikely that additional walkdowns or interviews would identify significant deficiencies requiring model updates, and that the current system models reasonably reflect the as-built/as-operated plant condition and configuration. Therefore, resolution of this F&O would not impact the calculations of risk changes for the RI-ISI Program.
SY-A11	SY-A11-01 (Failures to run in first hour)	S	Closed	Failures to run in first hour (rather than over the entire 24-hour mission time) were not addressed by creating a new basic event. This could lead to model update issues.	Failure to run during first hour is considered in the model. These failure probabilities are incorporated into the basic event for failure to run and adequately account for the impact on component reliability. The F&O addresses the ease of model update given that only one basic event exists for two failure modes.
SY-A16	SY-A16-01 (Modeling of pre-initiators) SY-A16 not met) Associated SR: HR-A1not met	F	Closed	No pre-initiator human failure events (HFEs) are modeled in the AFW system model. Since AFW is a standby system, at least one pre-initiator HFE (e.g., failure to restore pump after maintenance or testing) is expected to be in the model.	Pre-initiators review was performed and pre-initiator HFEs were identified in G.1 Revision 2. Several miscalibration and misposition HFEs were added to the PRA model.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
SY-A20	SY-A20-01 (Simultaneous unavailability of redundant SSCs) SY-A20 not met	F	Closed	Simultaneous unavailability of redundant safety-related equipment due to a planned activity is excluded from consideration, consistent with Technical Specification (TS) 3.0.3 restrictions. This assumption is probably not appropriate for nonsafety-related equipment, whose unavailability is not restricted by a TS. An example of this is multiple instrument air compressors concurrently out of service.	This F&O has been resolved by examination of the maintenance schedules and update of documentation.
SY-A23	SY-A23-01 (Consistent system model nomenclature) SY-A23 met	F	Closed	Consistent system/component failure mode nomenclature is used in all system notebooks, except the AFW notebook.	Changed basic event naming convention for all AFW top events
SY-B3	SY-B3-01 (CCF groups) SY-B3 not met	F	Closed	No documentation was found for the CCF group definition for the safety injection (SI) top event. For other systems, CCF groups appear to generally be defined inside of RISKMAN files but not in the documentation.	Documentation was revised for all systems to specifically list the common cause failures that are modeled.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
SY-B8	SY-B8-01 (Spatial and environmental hazards impacting multiple SSCs) SY-B8 not met Associated SR: SY-B14 met	F	Closed	No discussion of spatial and environmental dependencies, or room heatup and dependence on heating, ventilation and air conditioning (HVAC) could be found in the sampled system notebooks. The peer review team subsequently identified additional documentation that was available to potentially address these gaps.	This F&O has been closed with no action taken. Documentation of the effects of room heatup is available and references plant specific room heatup calculations. These results are not reiterated within the individual system notebooks but system modeling is consistent with the room heatup calculations.
SY-B10	SY-B10-01 (Modeling of permissive and interlocks) SY-B10 not met	F	Closed	The treatment of permissives and interlocks could not be located in the system notebooks.	PG&E performed a systematic evaluation of modeling of permissives and interlocks in the Internal Events PRA (IEPRA) and the Fire PRA (FPRA) and documented in PRA Calculation 14-01, Revision 1. The evaluation includes identification and modeling of (1) those systems that are required for initiation and actuation of a system, (2) the conditions needed for automatic actuation (e.g., low vessel water level), and (3) control features (e.g., protection and control permissive, lock-out signals, and component interlocks that are required to complete actuation logic, as required in the SR of Section 2 of AMSE/ANS RA-SA-2009 Standard. Based on the results of the review, permissive and interlocks of the following structures, systems, and components (SSCs) are included in the Internal Events model: 8701/8702, 8982A/B, and 9003A/B, 8804A/B.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
SY-B15	SY-B15-01 (Inter-system operator dependencies) SY-B15 not met	F	Closed	Human actions that had the potential to impact multiple trains of a given system (miscalibration) and actions from one system that could impact the function of another system are not addressed.	To address this F&O, the DCPP procedures were reviewed to identify realignment and calibration activities for all systems and components including any dependencies between activities and components. As a result of this review, numerous pre-initiator HFEs were identified in standby systems and were quantified using the EPRI Human Reliability (HRA) Calculator THERP module. Although pre-initiator dependency across trains was identified due to misposition and included in the DCPP HFEs, none of the HFEs involved miscalibration across systems or trains.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
SY-C2	SY-C2-01 (Documentation) SY-C2 not met Associated SRs: SY-A22 met at capability category II SY-B1 met SY-B3 not met SY-B6 met SY-B7 met at capability category II SY-B9 met SY-B11 met	F	Closed	The peer review team identified specific examples of deficiencies in the documentation of system models which need to be addressed, including documenting assumptions, references, HVAC dependencies, success criteria and timing, and discussion of available inventories of air, power, and cooling to support the mission time.	This F&O has no impact on the RI-ISI Program. Updating the documentation to address specific examples of missing information would not impact the calculations of risk changes for the RI-ISI Program. However, all identified documentation issues were resolved during the latest DCPP PRA model update.
HR-A1	HR-A1-01 (Pre-initiator events) HR-A1 not met Associated SRs: HR-A2 not met SY-A16 not met	F	Closed	The identification of pre-initiator HFEs based on whether the procedure or practice involves realignment or calibration should be performed before screening processes are applied.	To address this F&O, DCPP procedures were reviewed to identify realignment and calibration activities. This review was performed in order to be consistent with the ANS/ASME Standard supporting requirements HR-A1 and HR-A2. As a result of this review, additional pre-initiator HFEs were identified for inclusion into the PRA model and were quantified using the EPRI HRA Calculator THERP module. These new HFEs were incorporated into the PRA model.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
HR-A3	HR-A3-01(Pre-initiator events) HR-A3 met	F	Closed	Pre-initiator HRA screening criteria could remove restoration errors prematurely. If a system or train is automatically actuated following an event, then a restoration error of manual valves in the flow path could be missed. Examples include mispositioning of a valve in the standby component cooling water (CCW) pump train if it receives an automatic start signal on low header pressure and misposition of a valve in SI pump train if the valve does not automatically open on an engineered safety features actuation system (ESFAS) signal.	To address this F&O, all of the screening criteria were reviewed and revised as necessary to ensure that the criteria applied specifically to the component being operated/calibrated. The DCPP procedures were then reviewed against the new criteria to identify realignment and calibration activities.
HR-C3	HR-C3-01 (Consideration of miscalibration) HR-C3 not met	F	Closed	The pre-initiator HRA documentation discusses the reasons for not including common miscalibration, but the standard requires inclusion of such miscalibration events.	To address this F&O, DCPP procedures were reviewed to identify realignment and calibration activities. This review was performed in order to be consistent with the ANS/ASME Standard supporting requirements HR-A1 and HR-A2. As a result of this review, additional pre-initiator HFEs were identified for inclusion into the PRA model and were quantified using the EPRI HRA Calculator THERP module. These new HFEs were incorporated into the PRA model.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
HR-D3	HR-D3-01 (Pre-initiator HFEs) HR-D3 met at capability category I	F	Closed	The detailed discussion of pre-initiator HFEs does not discuss the quality of procedures, administrative controls, or man-machine interface (MMI) requirements in performing the assessments.	A new section dealing with procedure and human-machine interface quality has been added to the DCPP pre-initiator HRA documentation.
HR-E1	HR-E1-01 (Crediting manual verification steps when automatic actuation failed) HR-E1 met Associated SR: SY-A17met	F	Closed	Operator actions associated with starting pumps or aligning valves are not credited even when the emergency operating procedures (EOP) specifically states "Verify" pump started or "Verify" valve open/closed. In the event the automatic signal fails to start the pump or align the valve, credit should be taken for the operator backing up the automatic signal.	A review was performed to verify that no manual recovery for failure of an automatic signal that could be credited was missed. In order to avoid unnecessary complexity in the PRA model, the scope of the review was limited to risk significant basic events. The risk significant basic events were reviewed in conjunction with the EOPs to determine whether any additional manual recoveries of automatic signal failures could be found. No additional operator actions were identified that could mitigate the failure of an automatic signal for risk significant components. Therefore, no change to the DCPP PRA model is required.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
HR-E3	HR-E3-01 (Consistent interpretation of procedures) HR-E3 met at capability category I	F	Closed	There is no discussion in the HRA documentation on how the specific scenarios discussed in operator talk-throughs were selected, the questions posed to the operators, the entire sequence of procedures followed in the response to the accident sequence, etc. Actual operator interview sheets are not included; only a summary of the discussion is provided. Without having the basis for why the scenarios discussed were selected, it is not possible to ensure that the most risk-significant or important operator actions were discussed. Additionally, without the operator Interview sheets it is not possible to verify what the operators/trainers said, and that the responses were taken in context.	Operator interviews were re-performed and documented for each applicable operator action.
HR-E4	HR-E4-01 (Confirming response models via simulator observations or talk-throughs) HR-E4 met at capability category I)	F	Closed	Talk-throughs performed with Operations and Training personnel do not address confirming that the response models (i.e. thermal-hydraulic analysis codes) used to support the PRA are realistic. Additionally, no documentation of the use of simulator observations to confirm the response models can be found.	Simulator observations were performed to validate response models.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
HR-G5	HR-G5-01 (Verification of the time estimates in HRA via observation of simulator or walk-throughs) HR-G5 met at capability category II Associated SRs: HR-E3 met at capability category I HR-E4 met at capability category I	F	Closed	For some HFEs, no basis for the required time to perform the action is provided.	Operator interviews were re-performed and documented in for each applicable operator action. Response times were verified via interviews.
HR-G6	HR-G6-01 (Combining identical HFEs) HR-G6 met	F	Closed	Two HFEs appear to be essentially identical with the same human error probability (HEP). These two should be combined into one HFE, since the use of both could adversely affect the HRA dependence analysis and the impact of the state of knowledge correlation in the quantified results.	The two HEPs never appear in the same cutsets because of the mutually exclusive house event impacts used in the top event split fractions. Because they do not appear in the same cutsets, the dependency between two HFEs is immaterial. The current model is adequate and no model changes are needed. Documentation changes were made to clarify the diesel fuel oil modeling. RISKMAN data descriptions were also updated to avoid confusion.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
HR-G7	HR-G7-01 (HFE dependencies) HR-G7 not met	F	Closed	The HFE dependency documentation does not list a set of operator actions that were evaluated or how the dependence between actions is determined. The process to identify and evaluate HFE dependencies does not seem to provide a thorough means for identifying and accounting for dependent human actions.	The HRA dependency analysis was updated. The updated documentation clearly describes the operator actions evaluated and how the dependencies were evaluated.
HR-H2	HR-H2-01 (Staffing level assumed in HRA) HR-H2 met	F	Closed	The staffing levels credited in the HRA include personnel not on-site 24 hours, 7 days a week, but are available via call-in – so they should not be credited for shorter term responses. Additionally, minimum Operations staffing levels should be used when evaluating the post-initiator recovery actions.	This F&O has been resolved by a documentation update; no model changes were required. All HFEs were reviewed and updated to reflect actual on-site staffing levels. There were no impacts to the probabilities of existing HFEs.
HR-I2	HR-I2-01 (Documentation)	S	Closed	The peer review team identified specific examples of deficiencies in the documentation of HRA that need to be addressed, including normal vs. minimum staffing levels, use of multiple procedures, editorial corrections, and significant digits in the HEPs.	This F&O has been resolved by a documentation update.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
HR-I2	HR-I2-02 (Estimation of HEPs)	S	Closed	A screening value is used for post-initiator event ZHEAS6 (Failure to close header cross tie valves, FCV-495 and FCV-496.) This HFE is used in many accident sequences, including ISLOCA accident scenarios. The number of these scenarios and their use in ISLOCAs indicate that they are relatively significant events which should not use a screening value.	This F&O has been resolved by additional review; no model changes were required. A review confirmed that event ZHEAS6 is not a significant HFE from a risk importance standpoint and use of a screening value is therefore consistent with the standard.
DA-C1	DA-C1-01 (Use of the latest industry documentation for SSC failure rate, CCF, and offsite power recovery) DA-C1 not met	F	Closed	It is not evident that recognized sources are utilized for CCF and off-site power recovery data.	This F&O has been resolved by additional review; no model changes were required. The generic source of CCF data was not clearly identified in the documentation, but a review determined that all CCF data are from NUREG/CR-6928 which is a current recognized source. Offsite power recovery data comes from NUREG/CR (INEEL/EXT-04-02326).
DA-C4	DA-C4-01 (Basis for identification of an event as a failure) DA-C4 not met Associated SR: DC-C3 not met	F	Closed	A clear basis for the identification of events as failures has not been developed. Also, no evidence was found that degraded states were distinguished as being applicable (or not) as failures.	Detailed documentation of the basis for component failure identification was added to the DCPP Data Analysis Notebook.
DA-C5	DA-C5-01 (Documenting evaluation of failure events) DA-C5 not met	F	Closed	Documentation is inadequate to confirm whether component failures occurring close in time are separately counted.	This finding is related to the documented evaluation of failures occurring close in time when compiling plant reliability data. The documentation was updated to include reference to the Maintenance Rule methodology. A single example of such failures was identified and corrected.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
DA-C6	DA-C6-01 (Removing post-maintenance events from demand counts) DA-C6 met	F	Closed	Some post-maintenance tests have been included in the accounting of demands and operating hours for plant-specific data, which conflicts with the standard.	Data analysis was reviewed and post-maintenance testing demands were removed from the counts. Updates to the impacted failure probabilities in the model were made.
DA-C10	DA-C10-01 (Planned coincident unavailability) DA-C10 not met	F	Closed	There was no discussion regarding counting of successful demands when components are decomposed into sub elements.	The documentation for plant-specific data was updated to account for any component sub elements which may have unique demand counts.
DA-C14	DA-C14-01 (Planned coincident unavailability) DA-C14 not met Associated SR: SY-A20 not met	F	Closed	No assessment of routine planned maintenance activities for multiple component unavailabilities, or documentation that Maintenance Rule practices do not allow for routine instances of multiple trains or equipment being unavailable, were identified in the documentation.	Examined the 12-week rolling maintenance outage window matrix at DCPP and did not identify any planned, repetitive activity which would cause coincident unavailability due to maintenance for redundant equipment (both intra-system and intersystem). Calculation or modeling of coincident maintenance unavailability was therefore unnecessary.
DA-C16	DA-C16-01 (Disposition of plant-specific loss-of-offsite power (LOOP) events) DA-C16 met	F	Closed	Plant specific LOOP events are not identified in the documentation.	The finding is related to gaps in documentation of the disposition of plant-specific LOOP events used in determining the initiating event frequency. A review of the LOOP initiator frequency determined that plant-specific LOOP events are properly considered in the determination of initiating event frequency.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
DA-D4	DA-D4-01 (Tests and check of data updates) DA-D4 met at capability category II/III Associated SR: DA-E1 not met	F	Closed	The peer review team identified specific examples of deficiencies in the documentation of data which need to be addressed, related to Bayesian update data checks.	The Bayesian updating is done using the RISKMAN Data Module. Throughout the process, RISKMAN shows the analyst a plot of the prior distribution, and a plot of the prior distribution together with the posterior distribution. RISKMAN also shows various stats for these distributions such as the mean, median, and range factor. This process helps the analyst determine if the update and the distributions are valid and make sense. The Bayesian update checks for all failure rates and all initiating events were added as an attachment to the PRA Data Update Documentation. All distributions, including priors and posteriors, with their plots and statistics are stored in the RISKMAN files.
DA-D6	DA-D6-01 (Documenting method and references in data calculation) DA-D6 met at capability category III	F	Closed	NUREG/CR-5485 was used for CCF methodology; however this is not listed as a reference or in discussions in the calculation.	This F&O has been resolved by a documentation update to include the applicable reference to NUREG/CR-5485 for the generic data source for CCF.
DA-D8	DA-D8-01 (Documenting evaluation of design changes on impact on data) DA-D8 not met	F	Closed	No documentation of analysis done on impact on data of design changes (such as recirculation sump screen design change notices (DCNs), or new charging pump DCNs) could be found in the data calculation.	This F&O has been resolved with no action taken. The evaluation of the potential impact to PRA data due to DCNs are made as part of the design change process and documented during the design change process. On a routine basis as part of model maintenance, all design changes since the last model update are re-reviewed for impacts on the model. Based on the documented evaluation of DCNs, SR DA-D8 is judged to be met at capability category II since plant data are used for significant basic events.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
DA-E2	DA-E2-01 (Documentation) DA-E2 not met Associated SR: DA-D5 met at capability category III	F	Closed	Documentation does not facilitate review. Additional uncontrolled backup materials such as spreadsheets are required for a traceable basis for plant data.	The information provided in the backup documents was accurate and review of these documents did not result in a finding that would impact the PRA model. All PRA data analysis documentation was updated to include all information in a single calculation file without external attachments or spreadsheets, including data calculation files.
QU-B1	QU-B1-01 (RISKMAN code limitations)	S	Closed	The peer review team recommended that the quantification document include a specific section that discusses RISKMAN code limitations.	This suggestion F&O has been resolved by a documentation update to include the RISKMAN code limitations. The limitations of the RISKMAN code do not adversely impact its use in the RI-ISI Program.
QU-C2	QU-C2-01 (HFE dependency) QU-C2 not met	F	Closed	Human action dependencies are not evaluated with a minimum default value of the HEP to prevent underestimating risk.	Refer to F&O HR-G7-01. There is no requirement in the standard to use any minimum HEP for dependent actions, only to account for such dependencies.
QU-D4	QU-D4-01 (Comparison to other similar plants) QU-D4 met at capability category I	F	Closed	The documentation includes a comparison of results to other similar plants, but causes of significant differences are not identified.	Resolved and documented by performing a more in-depth comparison with other Westinghouse 4-loop plants.
QU-E1	QU-E1-01 (Uncertainty)	S	Closed	A review of generic sources of uncertainty was performed; however, this analysis would be improved by a review of plant-specific sources of uncertainty.	This suggestion F&O has been resolved by a documentation update. The assumptions and uncertainties associated with each technical element of different hazard groups are identified in the documentation. As suggested in this F&O, these documents have been updated by systematically reviewing PRA development documents (e.g., system notebooks, success criteria notebook, event-tree notebooks, etc.).

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
QU-F2	QU-F2-01 (Documentation) QU-F2 not met Associated SR: QU-F1not met	F	Closed	The peer review team identified specific examples of deficiencies in the documentation of quantification which need to be addressed as specified in the standard.	Quantification documentation updated to include items listed in the Supporting Requirement.
QU-F6	QU-F6-01 (Documenting definition of significant) QU-F6 not met	F	Closed	There was no definition for significant basic event located in the documentation.	Definition of significant sequences and basic event importance added to the quantification documentation.
LE-C1	LE-C1-01 (Plant-specific level 2 model) LE-C1 met at capability category I	S	Closed	Containment challenges in high level requirement LE-B must be compared to the containment structural capability analysis described in high level requirement LE-D.	This suggestion F&O was closed with no action taken. The containment structural capability has been assessed and documented adequately.
LE-C2	LE-C2-01 (Modeling of operator actions following the onset of core damage) LE-C2 not met	F	Closed	The LERF analysis states that there are no post-core damage operator actions available or credited. However, a review of plant procedures identified that there are several severe accident mitigation guidelines (SAMG) procedures available that do include post-core damage actions that need to be reviewed and credited as applicable.	All SAMG procedures were reviewed. No additional human actions were identified either because they were already credited as part of core damage mitigation or because the non-prescriptive nature of SAMG procedures did not lend themselves to HRA techniques.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
LE-C3	LE-C3-01 (Crediting repair of SSCs in significant LERF sequences) LE-C3 met at capability category I	S	Closed	No repair of equipment, other than the potential restoration of AC power following a loss of station power (LOSP) event, is credited in the LERF analysis. The recovery of offsite power is only credited pre-core damage, but could be considered for post-core damage scenarios.	The impact of not including repair of equipment is conservative in that no credit is taken. Furthermore, the larger uncertainty involved in estimating equipment repair likelihood, especially post-core damage, could skew the existing LERF results. Therefore, the impact of not meeting capability category II is conservative. The conservative treatment of not crediting repair or recovery of equipment does not reduce the risk importance of the system screened-in for RI-ISI program.
LE-C4	LE-C4-01 (Feasibility of scrubbing) LE-C4 met at capability category I	S	Closed	The LERF model does not credit mitigating actions (e.g., isolate the ruptured steam generator after core damage, depressurize the RCS and terminate the leak, recover containment integrity.) Additional fission product scrubbing provided by the containment sprays is not credited. Because it is assumed that all early releases are large, it is implied that all SG tube rupture (SGTR) and ISLOCA core damage sequences remain un-scrubbed.	Excluding mitigating actions from the PRA results in a conservative calculation of LERF. This conservative treatment is acceptable for systems in the scope of the RI-ISI Program, except for the containment spray (CS) system. For CS, not crediting scrubbing mitigation could underestimate the change in LERF. However, the frequency of core damage sequences that would still have the CS system available is not significant in typical pressurized water reactor PRAs, and the operation of CS therefore has limited impact on LERF.
LE-C9	LE-C9-01 (Equipment survivability or human action under adverse environments) LE-C9 met at capability category I	S	Closed	No credit is taken for any equipment survivability or human actions under adverse conditions or after containment failure.	This suggestion F&O does not adversely impact the RI-ISI Program. Excluding mitigating actions or equipment from the PRA results in a conservative calculation of LERF. This conservative treatment is acceptable for the RI-ISI Program.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
LE-C13	LE-C13-01 (Realistic containment bypass analysis) LE-C13 met at capability category I	S	Closed	All core damage events involving either a spontaneous SGTR, pressure induced SGTR, or a thermally induced SGTR event, as well as ISLOCA, were conservatively assumed to lead to a large early release. In addition, fission product scrubbing provided by the CSs is not credited.	This suggestion F&O does not adversely impact the RI-ISI Program. Credit for scrubbing of fission products is addressed by F&O LE-C4-01 (above.) Conservative treatment of ISLOCA and induced SGTR impacts results in a conservative estimate of LERF. This conservative treatment is acceptable for the RI-ISI Program.
LE-D7	LE-D7-01 (Realistic containment isolation analysis) LE-D7 met at capability category II	F	Closed	There is no traceable basis for the list of containment isolation (CI) valves that are present in the model and the systematic disposition of all of the containment penetrations that are not in the model.	A systematic evaluation of containment penetrations was performed and documented in PRA Calculation E.8 Revision 8 and in a separate spreadsheet. A set of screening criteria was developed consistent with the requirement of this SR, and consistent with large early release definition. Each containment penetration is dispositioned explicitly using this set of screening criteria. This F&O is closed and has no impact in RI-ISI application.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
LE-E2	LE-E2-01 (Documentation)	Best Practice	N/A	The discussion in PRA documentation associated with the plant damage state (PDS) and containment event tree (CET) descriptions are very detailed, easy to follow, and address many more potential damage states than typically evaluated in a LERF analysis. There is sufficient information in the tables and write-ups to understand when equipment is failed due to post core melt and/or post containment failure environments. Additionally, environmental/spatial impacts are addressed and the basis for equipment nonsurvivability is clearly delineated.	No disposition is required for this best practice F&O.
LE-E2	LE-E2-02 (Definition of LERF with 3-inch opening) LE-E2 met	F	Closed	No actual calculation verifying the 3-inch containment break size which constitutes a large release exists.	As documented in response to F&O LE-D7-01, CI analysis was re-performed based on greater than 2-inch definition of the large early release path.
LE-F2	LE-F2-01 (Review of LERF sequences for reasonableness) LE-F2 met	F	Closed	The LERF results documentation does not reflect the latest LERF cutsets. Additionally, the results include an out-of-date assumption on RCP seal LOCA sizes which needs to be deleted and actual detailed results presented.	The seal LOCA split fractions were confirmed to not have changed since the level 2 analysis was performed, so there are no model updates required to address this issue. The latest update to the quantification documentation includes LERF cutsets and insights.

Table A-1. Diablo Canyon Internal Events PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
LE-G3	LE-G3-01 (Documenting LERF calculations) LE-G3 not met	F	Closed	The relative contribution of contributors is not documented in the LERF calculation, and the information in the quantification calculation does not reflect the latest results, and does not include all the types of contributions discussed in this supporting requirement.	The quantification documentation was updated to include the contribution to LERF from initiating events as well as other requirements from this SR.
LE-G5	LE-G5-01 (Limitations in the LERF analysis) LE-G5 not met	F	Closed	The limitations in the various portions of the LERF analyses that would impact applications are not identified or discussed.	This F&O has no impact on the RI-ISI Program. The DCPP PRA model includes a complete level 2 detailed analysis. There are currently no general limitations in the LERF analysis that would impact applications. The F&O is related to documentation of limitations in the LERF analysis. Documenting the limitations of the LERF analysis would not impact the calculations of risk changes for the RI-ISI Program.

Table A-2. Diablo Canyon Internal Flood PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
IFSO-A1	IFSO-A1-01 (Applicable external sources)	S	Closed	Not all external flooding sources are identified in the documentation, and walkdown information does not identify tank inventories.	The internal flooding PRA was updated to address this F&O. Identification of potential flood sources include in-leakage from other flood areas. Tank inventories were identified.
IFSO-A6	IFSO-A6-01 (Spray protection) IFSO-A6 met	F	Closed	The walkdown reports identify equipment which is protected from the effects of spray; however, the documentation does not discuss what is specifically credited as spray protection and the limitations of that protection. This could result in future plant modifications which alter the plant configuration in a manner which impacts the spray protection without being recognized as an impact to the PRA.	The internal flooding PRA was updated to address this F&O. Discussion of what constitutes spray protection was enhanced.
IFSO-B3	IFSO-B3-01(Uncertainty)	S	Closed	Sources of epistemic uncertainty related to flood sources are not explicitly discussed.	Internal flooding documentation was updated with assessment of uncertainty.
IFSN-A3	IFSN-A3-01(Automatic and/or operator responses) IFSN-A3 not met	F	Closed	Relevant automatic or operator responses to flood events which could terminate or contain flood propagation are not identified in the documentation.	The internal flooding PRA was updated to address this F&O. For infinite flood sources, and large flood sources, auto and/or operator responses to terminate or contain a flood were added.
IFSN-A4	IFSN-A4-01 (Capacity of drains, berm, dikes, etc.) IFSN-A4 not met	S	Closed	Details on the capacity of floor drains and sumps, and the impact of berms, dikes, and curbs are not discussed in the documentation. These features in general are not credited, and a more realistic evaluation could be performed.	Internal flooding documentation was updated. In general, credit for dikes, berms, and curbs is not taken to terminate or contain flood propagation. Curbs are discussed as a means to estimate water height in local area where flood originates.

Table A-2. Diablo Canyon Internal Flood PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
IFSN-A6	IFSN-A6-01 (Spray targets)	S	Closed	No detailed evaluation of potential spray targets based on the distance from the source with consideration of the maximum potential spray elevation and specific propagation paths has been made.	The internal flooding PRA was updated to address this F&O. For spray, see resolution of IFSO-A6-01. The distance criteria for adverse spray impact from pressurized pipe and high-energy flood sources were added to the documentation and were applied for spray scenario development.
IFSN-A7	IFSN-A7-01(Flooding impacts on SSCs)	S	Closed	For flooding effects to SSCs other than submersion, the documentation does not describe the effects in a manner which is easily verifiable.	The internal flooding PRA was updated to address this F&O. For spray impact, spray target component screening and spray scenario development for unscreened components was performed, see resolution of IFSO-A6-01 and IFSN-A6-01. The affected equipment due to submergence (and spray) for unscreened scenarios are listed in the documentation.

Table A-2. Diablo Canyon Internal Flood PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
IFSN-A8	IFSN-A8-01 (Drain line and back flow paths) IFSN-A8-01 met at capability category II	F	Closed	The potential for inter-area propagation through various flowpaths identified in the standard are not identified in the documentation.	The Internal Flooding PRA Report was updated and documents the identification of propagation pathways at DCPP. Due to the open layout design and numerous openings in different elevations of the auxiliary building and turbine buildings (e.g., open stairways and grate-covered floor openings), floods originating in one level are expected to propagate freely to the basement of the building. Other progagation pathways involving unsealed cable trays, conduit and pipe penetrations were also considered and documented in the internal flooding update.
IFSN-A9	IFSN-A9-01 (Flood depth and propagation) IFSN-A9 not met	F	Closed	No calculations determine the flooding rates and the time to equipment damage.	Flood calculations were performed for selected areas where bounding assumptions were too severe and more detailed analysis was required, including flood areas with limited drainage paths and large flood source capacities. The calculations consider flood rates, flood propagation through door gaps, opening between rooms and floor drains. The flooding depth (level rise) timing is evaluated in the updated internal flooding PRA report.
IFSN-A10	IFSN-A10-01 (Size of flood sources) IFSN-A10 met	F	Closed	Evaluations of the flooding scenarios do not include the impact of emptying a source on the flood depth in the areas, or the propagation of infinite water sources without operator action to isolate the flood.	The Internal Flooding PRA was updated to address this F&O. The size of infinite flood sources, circulating water, auxiliary saltwater and firewater from the raw water reservoir, were included in the flood scenario development along with the flood area, source, flood rate, SSC damage, and operator actions.

Table A-2. Diablo Canyon Internal Flood PRA Peer Review F&Os and Disposition

SR	Topic	F&O Level	Status	Finding	Disposition
IFSN-A11	IFSN-A11-01(Multi-unit effects) IFSN-A11 not met	F	Closed	The impact of large flooding sources in areas that could impact both units has not been considered. The potential for a large circulating water or auxiliary saltwater (ASW) flood event on the common turbine building and intake structure resulting in a dual-unit shutdown was identified.	For the turbine building flood scenarios, ASW and circulating water piping failure is assumed to cause a dual unit trip. ASW and circulating water pipe breaks in the intake structure causing dual unit trip are not considered credible scenarios. In response to this F&O, pipe failures in auxiliary building flood areas that are shared between the two units are included in the flood initiator frequency count for both units (see Appendix G of Section 9, Revision 1 of the Internal Flooding PRA Report)

Table A-2. Diablo Canyon Internal Flood PRA Peer Review F&Os and Disposition					
SR	**Topic**	**F&O Level**	**Status**	**Finding**	**Disposition**
IFSN-A12	IFSN-A12-01 (Screening of flood scenarios) IFSN-A12 met	F	Closed	Flooding scenarios are screened or assumed not to propagate based on drains, curbs and barriers between rooms, and the screening implicitly assumes that the leak is smaller than the drain capacity and/or that the operators take action to reduce or stop the flow before water backs up into the room and fails additional equipment or propagates beyond the room. The propagation screening does not look at accumulation on the area where the water is going and whether equipment in that area would be impacted due to flood or whether the flood could propagate beyond the second flood area to another area and damage equipment.	The scenarios in the Internal Flooding PRA Report were reviewed. Additional propagation scenarios previously screened in Revision 0 were identified and scoped in with flood source capacity and propagation paths considered in characterization and quantification of the flood scenarios. In addition, select HFEs were developed to model the flood isolation for large flood sources such as firewater from the raw water reservoir. Failure of these HFEs results in additional PRA equipment damage beyond the original source flood area, such as both RHR pumps being damaged whenever the 54-foot pipe tunnel in the auxiliary building is flooded beyond its capacity volume.
IFPP-A5	IFPP-A5-01 (Walkdown documents) IFPP-A5 met	F	Closed	The walkdown documentation has missing information associated with the flooding sources.	Walkdown documentation was updated to include missing information for all flooding sources.