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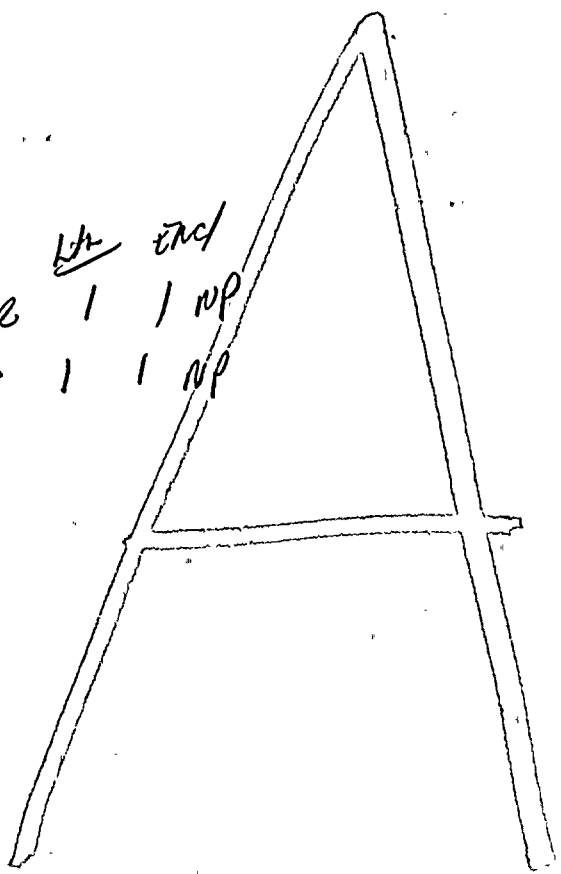
SUBJECT: Forwards proprietary WCAP-11723 & nonproprietary WCAP-11724, "LOFTTR2 Analysis for Steam Generator Tube Rupture...."

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James D. Shiffer
Vice President
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April 29, 1988

PG&E Letter No. DCL-88-114



U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Steam Generator Tube Rupture (SGTR) Analysis

Gentlemen:

PG&E hereby submits a revised steam generator tube rupture (SGTR) analysis for Diablo Canyon Power Plant (DCPP) Units 1 and 2 in accordance with license condition 2.C.(9) of Amendment 12 to Facility Operating License No. DPR-82, which requires:

By April 1988, PG&E shall submit for NRC review and approval an analysis which demonstrates that the steam generator tube rupture (SGTR) analysis presented in the FSAR is the most severe case with respect to the release of fission products and calculated doses. Consistent with the analytical assumptions, PG&E shall propose all necessary changes to the Technical Specifications (Appendix A) to this license.

Enclosure 1 provides PG&E's responses to the plant-specific information requested by the NRC Staff in their evaluation of the Westinghouse Owners Group (WOG) SGTR subgroup methodology (NRC letter to SGTR subgroup dated March 30, 1987, Charles E. Rossi to Alan E. Ladieu).

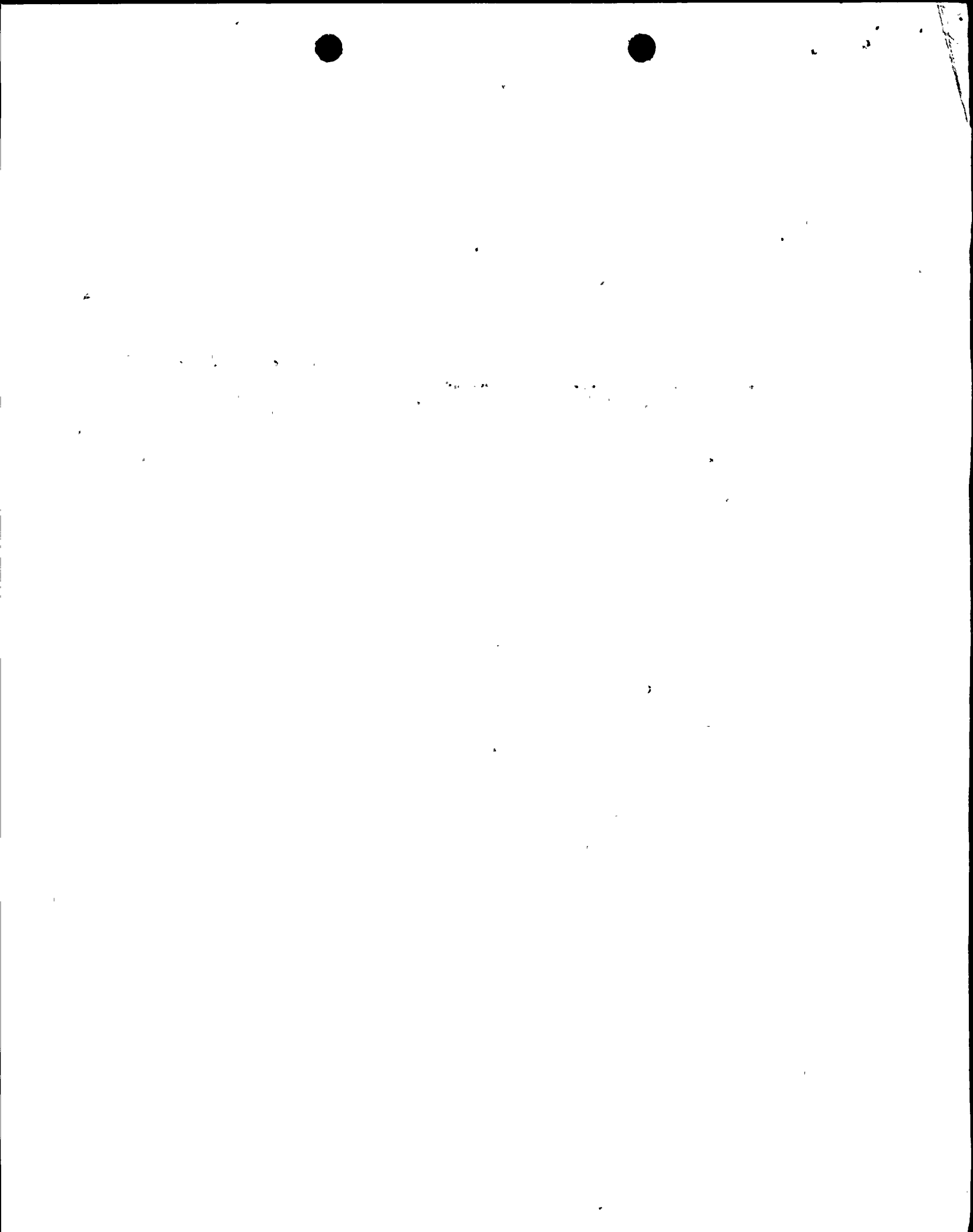
In support of PG&E's responses, Enclosure 2 provides a copy of WCAP-11723 (proprietary) and WCAP-11724 (non-proprietary), "LOFTTR2 Analysis for A Steam Generator Tube Rupture Event for the Diablo Canyon Power Plant Units 1 and 2." The DCPP-specific SGTR analysis provided in these WCAPs is based on the WOG SGTR subgroup program, whose methodology was approved by the NRC Staff in their March 30, 1987 letter. The analysis addresses the NRC Staff's generic concerns noted in DCPP SSER 31, Section 4.25, regarding SGTR events which occurred at other operating plants. In addition, the analysis addresses a recent Westinghouse generic concern regarding potential uncovering of the steam generator tubes after a reactor trip.

PG&E has reviewed the analytical assumptions used in the SGTR analysis and concludes that no changes to the DCPP Technical Specifications are necessary. Pending NRC approval, the SGTR analysis methodology and results will be included in Chapter 15 of the next appropriate DCPP FSAR Update.

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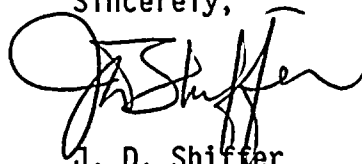


April 29, 1988

As WCAP-11723 contains information proprietary to Westinghouse Electric Corporation, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. It is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations. Accordingly, included in Enclosure 2 is a Westinghouse authorization letter (CAW-88-015), proprietary information notice, and accompanying affidavit. Correspondence with respect to the proprietary aspects of the Application for Withholding or the supporting Westinghouse affidavit should reference CAW-88-015 and should be addressed to R. A. Wiesemann, Manager Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Sincerely,

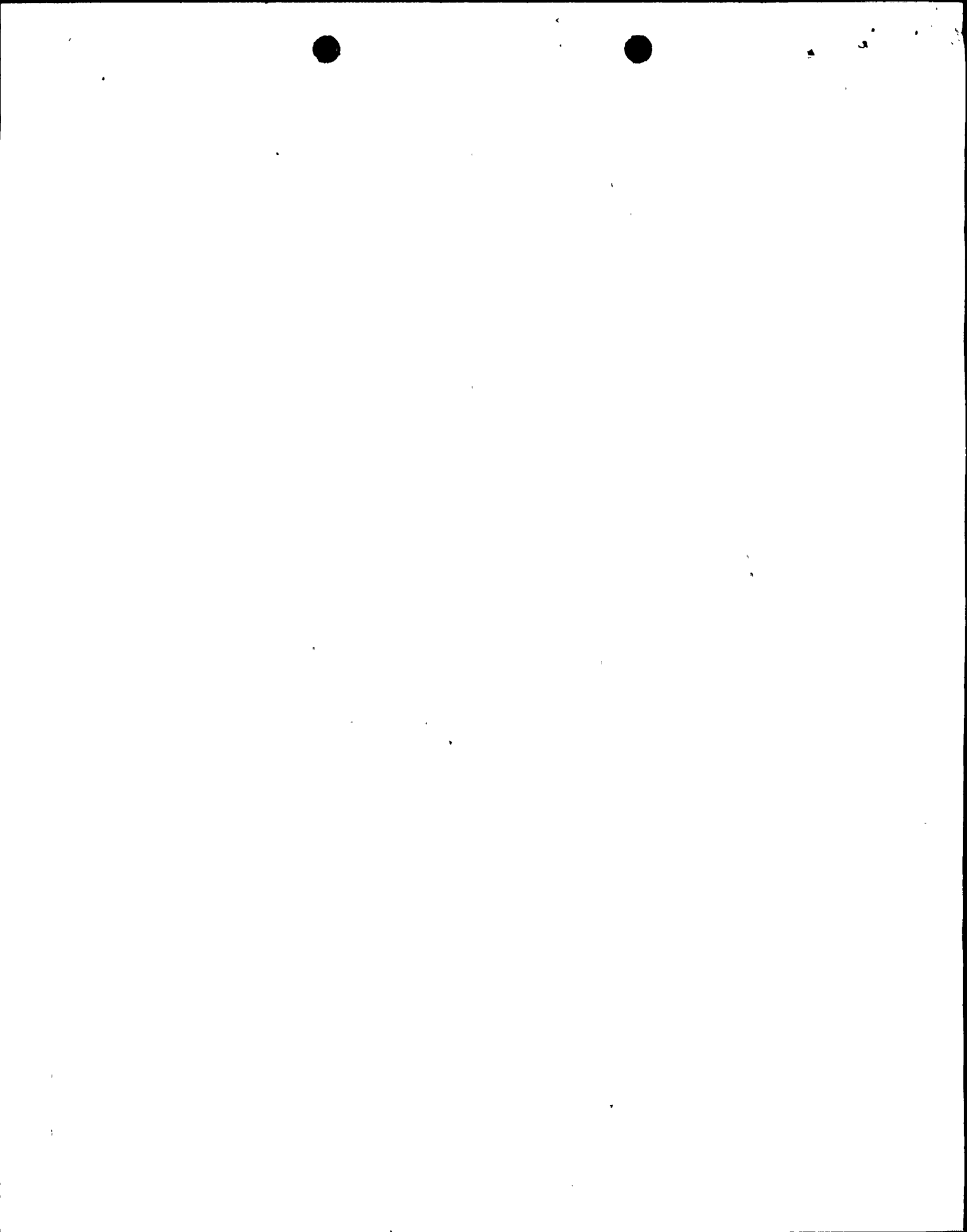


J. D. Shiffer

cc: J. B. Martin	(WCAP-11723 only)
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Enclosures

1977S/0057K/JHA/893



ENCLOSURE 1

PG&E's Response to the NRC Staff's Request
for Plant-Specific Information Regarding
Steam Generator Tube Rupture Analysis

In an NRC letter dated March 30, 1987, from Charles E. Rossi to Alan E. Ladieu, chairman of the Westinghouse Owners Group (WOG) steam generator tube rupture (SGTR) subgroup, the NRC Staff provided an evaluation of WOG SGTR reports WCAP-10698 and WCAP-11002. In this evaluation, the NRC Staff required each member of the SGTR subgroup to submit plant-specific information for the NRC Staff to complete plant-specific safety evaluation reports.

The following provides PG&E's responses to the five items requiring plant-specific information for Diablo Canyon Power Plant (DCPP) Units 1 and 2.

NRC Request (1)

Each utility in the SGTR subgroup must confirm that they have in place simulators and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design basis analysis. Demonstration runs should be performed to show that the accident can be mitigated within a period of time compatible with overfill prevention, using design basis assumptions regarding available equipment, and to demonstrate that the operator action times assumed in the analysis are realistic.

PG&E Response (1)

PG&E has a training simulator on the DCPP site. A licensed operator training program is in place to train the DCPP operators for response to a SGTR event in a manner consistent with the operator action times assumed in the WCAP-10698 design basis analysis.

In the event of a SGTR, the operator is required to take action to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in DCPP Emergency Procedure E-3. To assure that the operator actions and times are realistic and consistent with those assumed in the WCAP-10698 design basis analysis, PG&E observed six sessions of the licensed operators training program held at the DCPP training simulator. The operators were unaware that they were being observed and they had no foreknowledge that a SGTR event would be part of the training session. The following four operator action times were observed and recorded.



- The time it takes to identify and isolate the ruptured steam generator.
- The time it takes to initiate cool down of the reactor coolant system (RCS) using steam dumps after the ruptured steam generator has been isolated.
- The time it takes to initiate depressurization of the RCS after the cooldown is complete.
- The time it takes to terminate safety injection (SI) after the RCS has been depressurized.

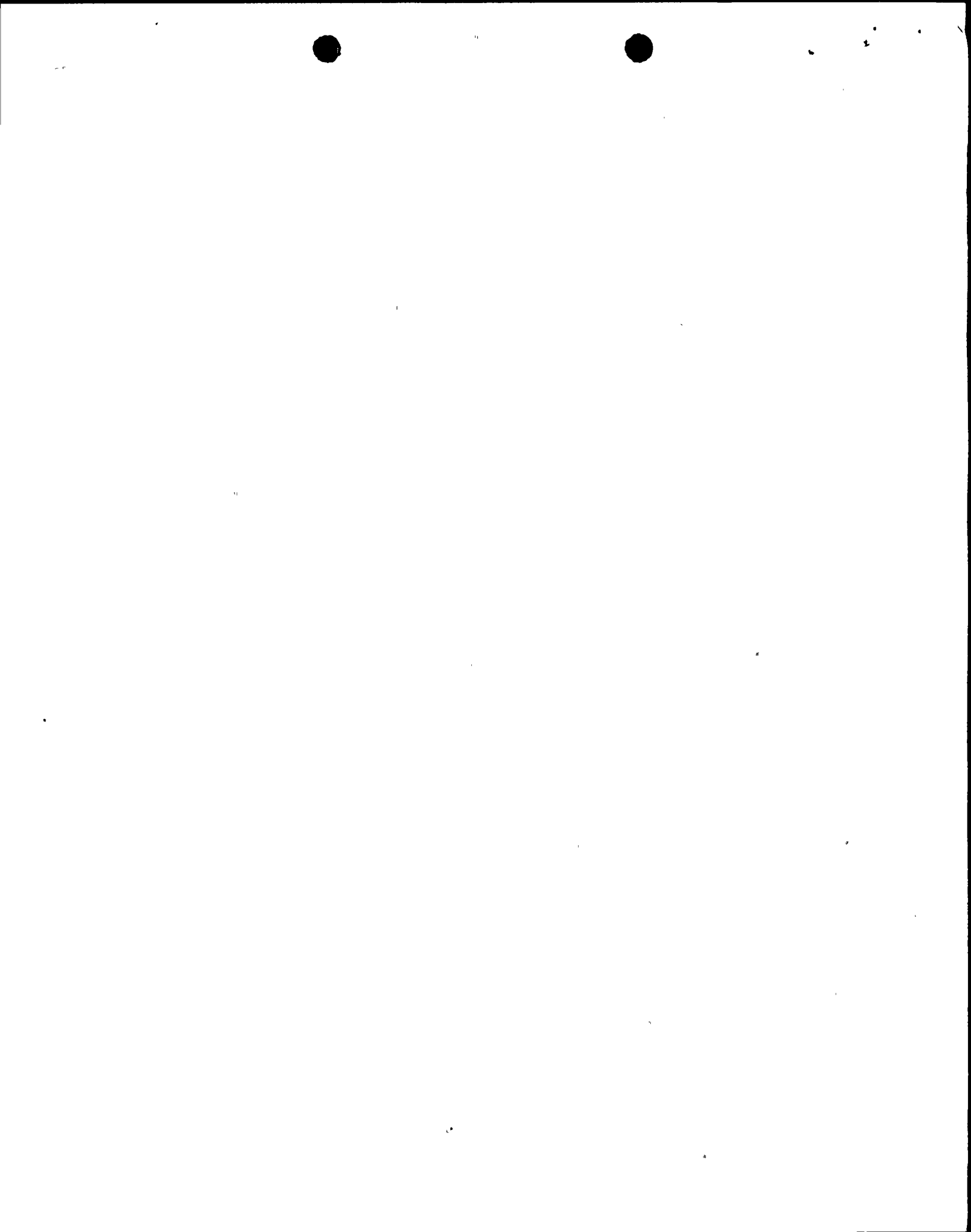
Observations for the four operator action times were statistically averaged to obtain realistic times for the DCP-PP-specific SGTR analysis provided in WCAP-11723/WCAP-11724 (see Enclosure 2). The table below summarizes the average times observed during the six training sessions, the times assumed in the DCP-PP-specific analysis (WCAP-11723/WCAP-11724) based on the observed times, and the times assumed in the WCAP-10698 design basis analysis.

<u>Operator Actions</u>	<u>Operator Action Time (minutes)</u>		
	<u>Observed (average)</u>	<u>WCAP-11723/ WCAP-11724</u>	<u>WCAP-10698</u>
Identify and isolate ruptured SG	9.6	10	10
Initiate cooldown after isolation	6.1	5	5
Initiate depressurization after cooldown	3.3	4	2
Terminate SI after depressurization	0.7	1	1

Based on the observed times, the times assumed in WCAP-11723/WCAP-11724 are realistic and consistent with those assumed in WCAP-10698. As analyzed in Section II of WCAP-11723/WCAP-11724, usage of these realistic operator action times, together with design basis assumptions regarding available equipment, demonstrates that an SGTR event at DCP-PP can be mitigated within a time compatible with steam generator overfill prevention.

NRC Request (2)

A site specific SGTR radiation offsite consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis should be performed using the methodology in SRP Section 15.6.3, as supplemented by the guidance in the NRC letter from H. Berkow to Alan Ladieu, dated December 17, 1985.



PG&E Response (2)

A DCPP-specific SGTR radiation offsite consequences analysis has been performed to demonstrate acceptable consequences for a design basis SGTR event for DCPP. The analysis and results are described in Section III of WCAP-11723/WCAP-11724 (see Enclosure 2). The analysis assumes the most limiting single failure based on the information in WCAP-10698, Supplement 1. The analysis is consistent with the methodology in WCAP-10698, WCAP-10698 Supplement 1, and SRP 15.6.3, as supplemented by the guidance in the NRC letter from H. Berkow to A. Ladieu dated December 17, 1985. The results of the analysis indicate that the radiation doses are within the guidelines of 10 CFR 100 and SRP 15.6.3. Pending NRC approval, the analysis methodology and results will be included in Chapter 15 of the next appropriate DCPP FSAR Update.

In addition to the analysis of the offsite doses addressed in WCAP-11723/WCAP-11724, an analysis of the dose to control room operators from a postulated SGTR event was performed to demonstrate that the potential dose consequences are within the guidelines of 10 CFR 50 General Design Criteria (GDC) 19. The control room doses were calculated using the atmospheric releases provided in WCAP-11723/WCAP-11724 and the atmospheric dispersion factors calculated previously for the LOCA (DCPP FSAR Update Table 15.5-6).

The resultant control room operator doses from atmospheric releases for the duration of the postulated SGTR accident are summarized below:

	<u>Pre-Accident Spike (rem)</u>	<u>Accident Initiated Spike (rem)</u>	<u>GDC 19 Guideline (rem)</u>
Thyroid	1.59	0.24	30
Whole Body	0.030	0.029	5
Beta Skin	0.027	0.027	30

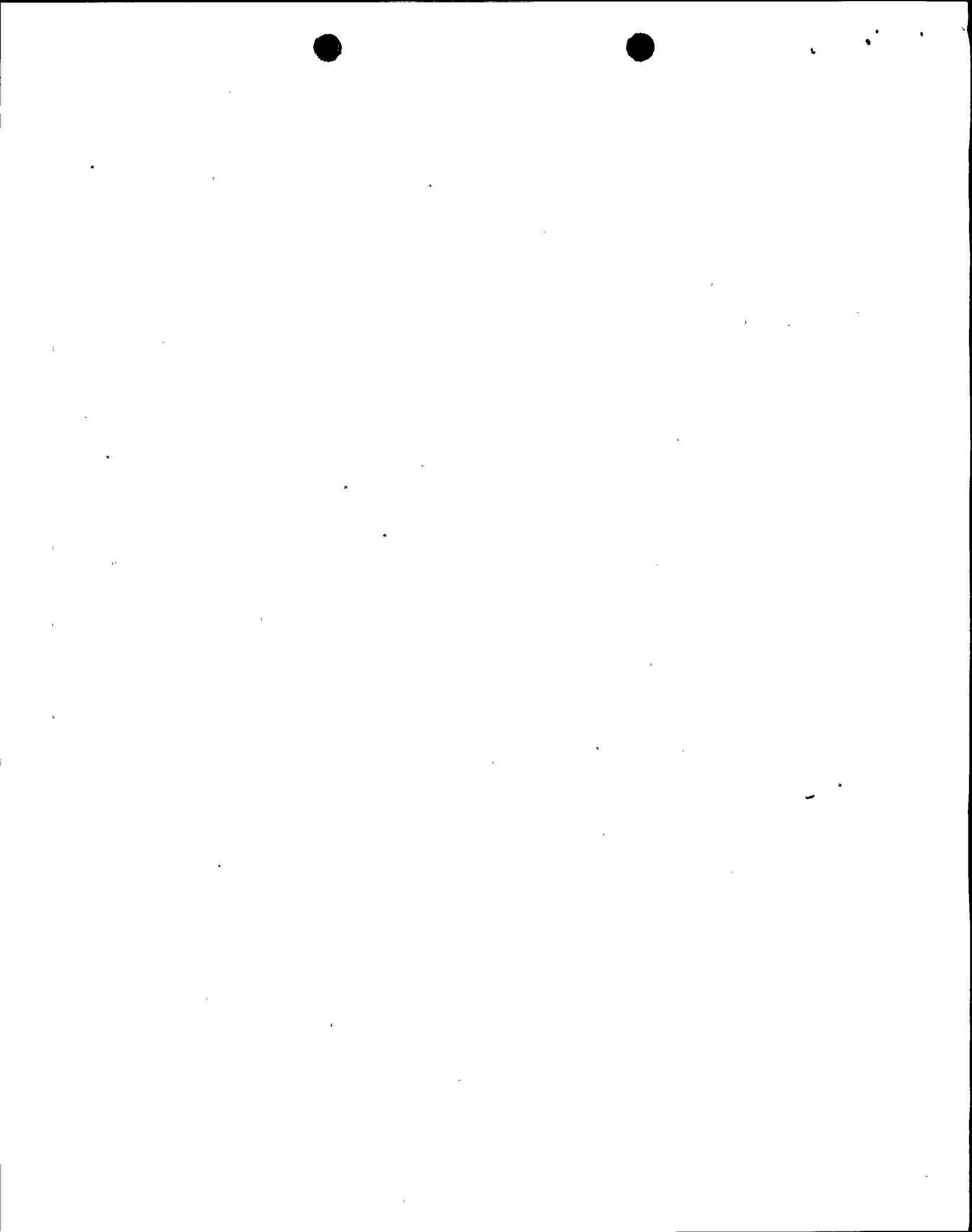
As shown above, the resultant doses from a postulated SGTR are well below the guidelines of GDC 19. Pending NRC approval, the results of this analysis will be included in Chapter 15 of the next appropriate DCPP FSAR Update.

NRC Request (3)

An evaluation of the structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overflow.

PG&E Response (3)

The DCPP-specific SGTR analysis provided in WCAP-11723/WCAP-11724 demonstrates that the steam generators will not overflow and cause water to accumulate in the main steam lines. However, as required by the NRC, stress analyses have



been performed on the main steam lines to confirm their structural adequacy under water-filled conditions.

Analyses assumed all DCPD steam lines from the steam generators to the main steam isolation valves (MSIVs) were completely filled with water. The piping was evaluated for sustained loading, including the dead weight of piping and fluid. The supporting system variable spring hangers were modeled with their actual spring constants, and their hot settings were accounted for as upward forces.

Since the SGTR event is considered an accident case, the piping and pipe supports are required to be qualified under faulted condition allowables. It is postulated that a SGTR and an earthquake do not occur at the same time.

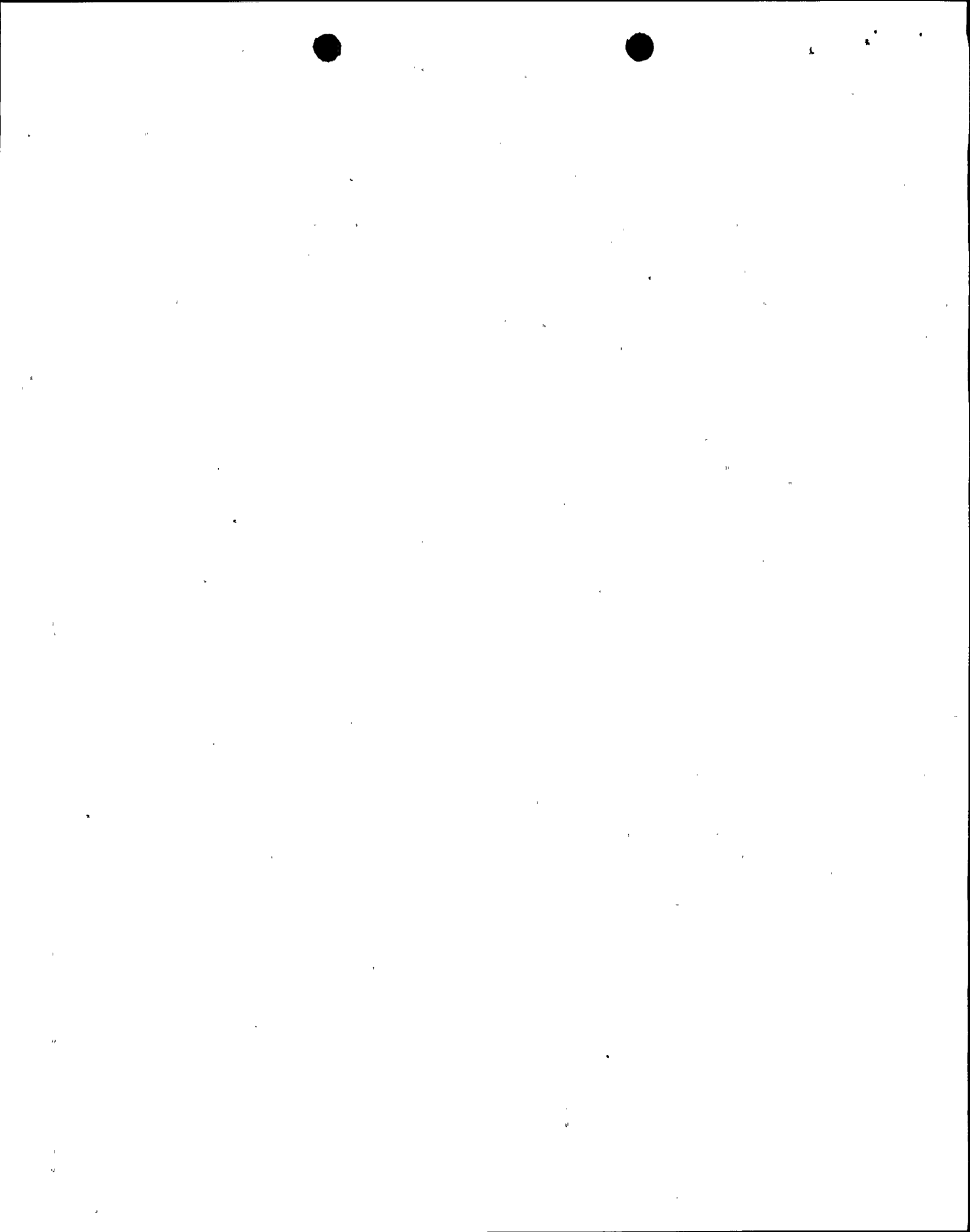
The results of the analyses are summarized below:

- Piping stresses are within the code allowables.
- Piping dead load stresses combined with thermal stresses are within the code allowables.
- Because rigid pipe supports are designed for MSIV slam and other higher loads, only spring hangers and snubbers were reviewed for a change in displacements. The results indicate that all spring hangers and snubbers have enough travel capability for any significant change in pipe displacements due to flooding of the main steam piping.

The analyses' results confirm that the piping and pipe supports are qualified for main steam piping flooding due to SGTR.

NRC Request (4)

A list of systems, components, and instrumentation which are credited for accident mitigation in the plant specific SGTR EOP(s). Specify whether each system and component specified is safety grade. For primary and secondary PORVs and control valves specify the valve motive power and state whether the motive power and valve controls are safety grade. For non-safety grade systems and components state whether safety grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overflow or justify that non-safety grade components can be utilized for the design basis event. Provide a list of all radiation monitors that could be utilized for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible. If the EOPs specify steam generator sampling as a means of ruptured SG



identification, provide the expected time period for obtaining the sample results and discuss the effect on the duration of the accident.

PG&E Response (4)

The attached table provides a list of all systems, components, and instrumentation which are required to carry out each of the steps in DCPD Emergency Procedure (EP) E-3, "Steam Generator Tube Rupture." The safety grade status of each item is indicated. For primary and secondary PORVs and control valves, the valve motive power is listed and the safety grade status of motive power and instrumentation is indicated. In those cases where non-safety grade systems or components are identified, safety grade backups which can be expected to function are listed.

The following list specifies the quality and reliability of radiation monitors that could be used for identification of the accident and the ruptured steam generator.

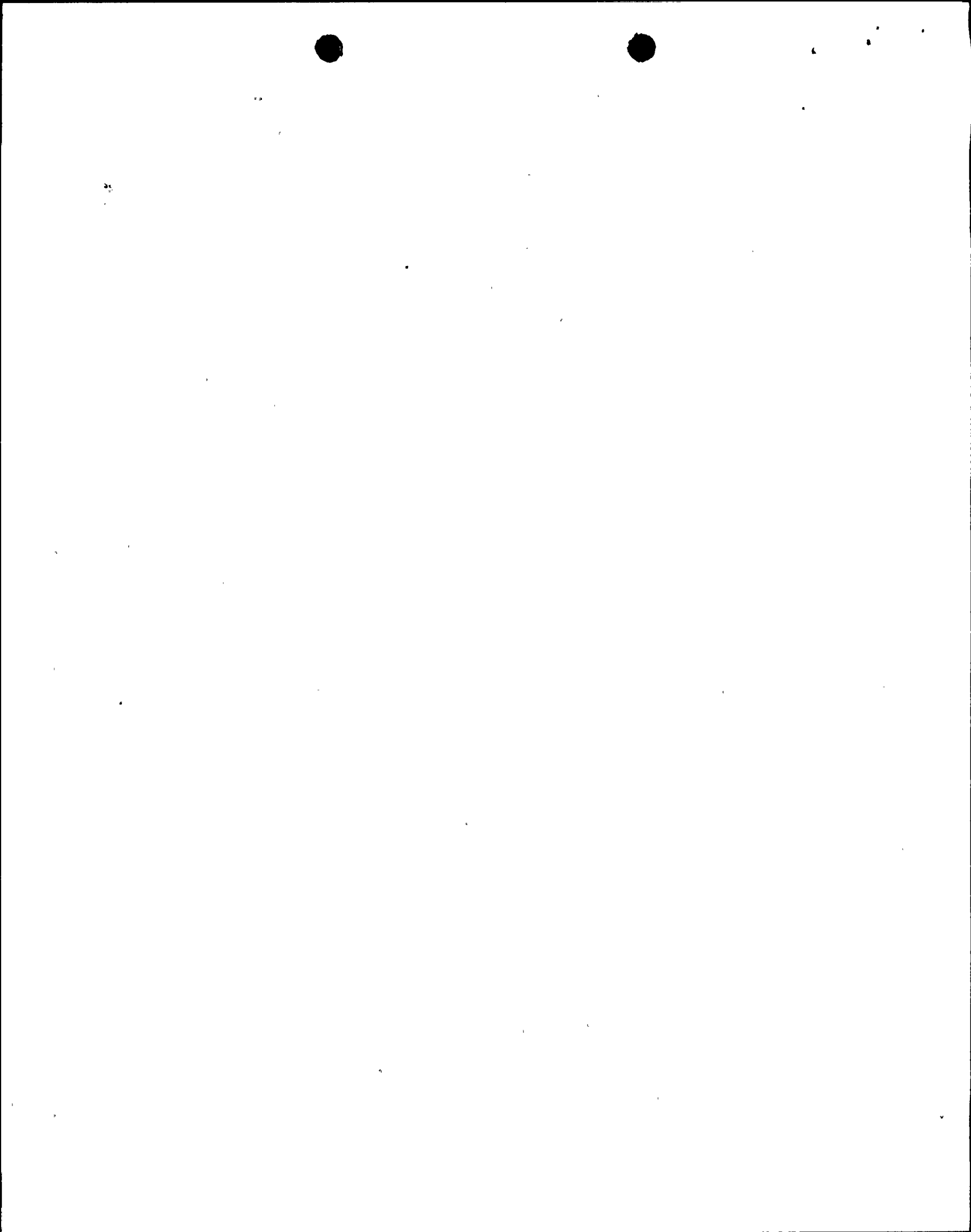
<u>Radiation Monitor</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Power Supply</u>	<u>Instrument Class</u>
Main steam line	RE-71	Yes	1E	IB/E/Cat 2
	RE-72	Yes	1E	IB/E/Cat 2
	RE-73	Yes	1E	IB/E/Cat 2
	RE-74	Yes	1E	IB/E/Cat 2
Steam jet air ejector	RE-15	No	non-1E	IB/C/Cat 3
SG blowdown header	RE-19	No	non-1E	II
SG blowdown discharge	RE-23	No	1E	IB/E/Cat 3
SG blowdown vent	RE-27	No	1E	IB/E/Cat 3

No credit is taken for the radiation monitors in the DCPD SGTR analysis, which is consistent with the methodology in WCAP-10698.

Steam generator sampling is also available as a means of ruptured steam generator identification. Sample results would be available within three hours. As sampling of the steam generator fluid is not the primary method for determining a SGTR event, it is expected that the sampling time span would not impact accident duration.

NRC Request (5)

A survey of plant primary and "balance-of-plant" systems design to determine the compatibility with the bounding



plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the difference on the margin of overfill should be provided.

PG&E Response (5)

A DCPD-specific analysis was performed to determine the margin to overfill for a design basis SGTR event. Section II of WCAP-11723/WCAP-11724 describes this analysis (see Enclosure 2). The analysis was performed using the methodology developed in WCAP-10698. A review of the DCPD plant design was conducted to determine the appropriate design basis conditions and parameters to be used for the SGTR analysis. Design differences between the bounding plant analysis in WCAP-10698 and the DCPD analysis are noted in WCAP-11723/WCAP-11724. Initial conditions and assumptions for the analysis were conservative with respect to overfill. The worst case single failure relative to overfill was identified and simulated in the analysis.



ATTACHMENT TO ENCLOSURE 1

List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>
(1) Check If RCPs Should Remain In Service: Note - There is no equipment credited for accident mitigation in this Step							
(2) Identify Ruptured SG(s):	Primary	AFW supply valves	LCV-106	Yes	Motor	1E	IA
	Primary		LCV-107	Yes	Motor	1E	IA
	Primary		LCV-108	Yes	Motor	1E	IA
	Primary		LCV-109	Yes	Motor	1E	IA
	Primary		LCV-110	Yes	EH	1E	IA
	Primary		LCV-111	Yes	EH	1E	IA
	Primary		LCV-113	Yes	EH	1E	IA
	Primary		LCV-115	Yes	EH	1E	IA
	Backup	AFW pumps	AFWP1-1	Yes	Steam Turbine	*	N/A
	Backup		AFWP1-2	Yes	Motor	1E	N/A
	Backup		AFWP1-3	Yes	Motor	1E	N/A
	Backup	AFWP discharge pressure transducer	PT 433	Yes	N/A	1E	IA
	Backup		PT 434	Yes	N/A	1E	IA
	Backup	AFWP discharge pressure controller	PC 86	Yes	N/A	1E	IA
	Backup		PC 87	Yes	N/A	1E	IA
	Backup		PC 88	Yes	N/A	1E	IA
	Backup		PC 89	Yes	N/A	1E	IA
	Backup	AFWP discharge level summator	LM-86A	Yes	N/A	1E	IA
	Backup		LM-87A	Yes	N/A	1E	IA
	Backup		LM-88A	Yes	N/A	1E	IA
	Backup		LM-89A	Yes	N/A	1E	IA

*Steam inlet valve FCV 95 is 1E powered
N/A - Not Applicable.



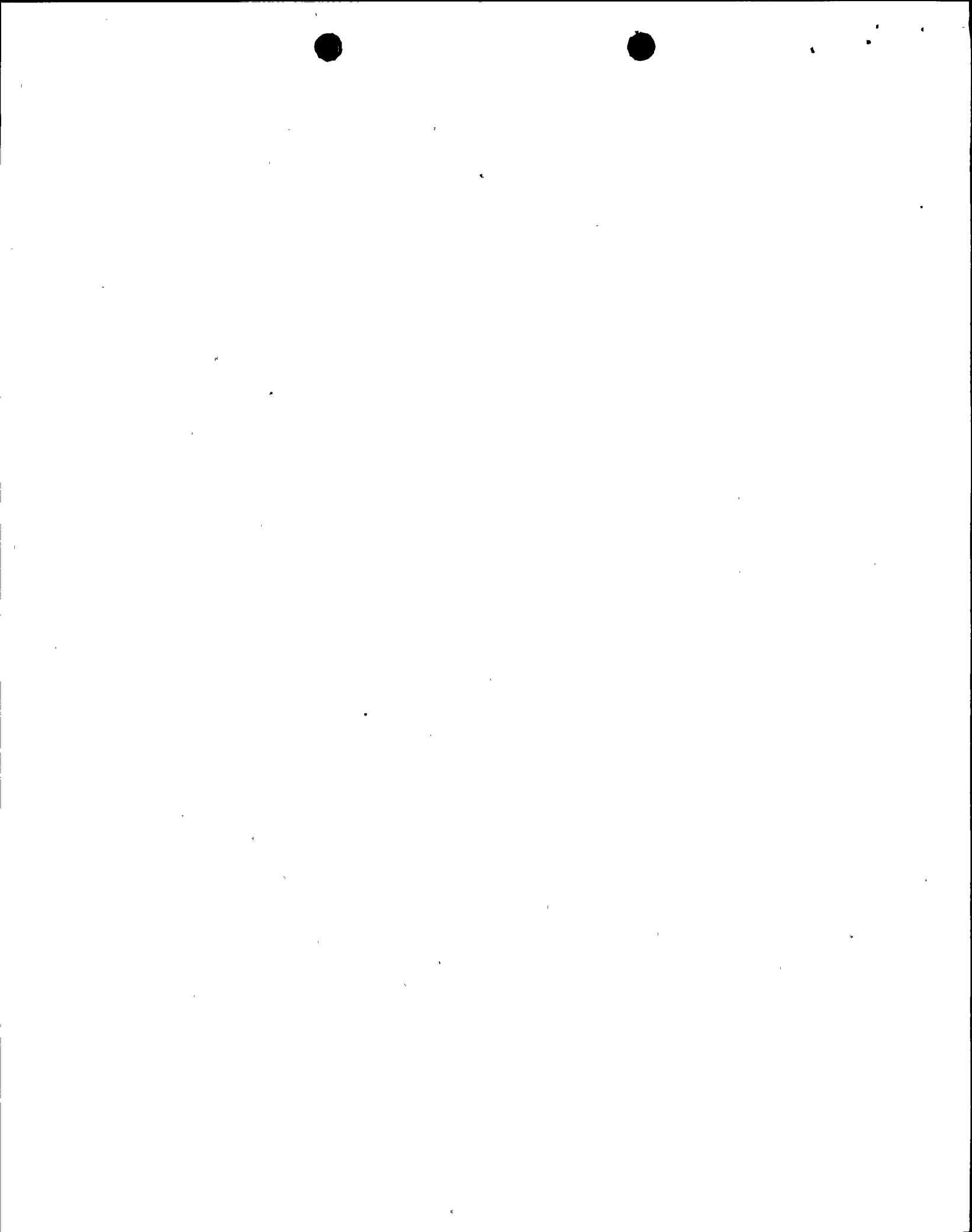
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ATTACHMENT TO ENCLOSURE 1

List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>
(2)	Primary	AFWP SG level controller	LC-86	Yes	N/A	1E	IA
	Primary		LC-87	Yes	N/A	1E	IA
	Primary	SG AFW supply hand controller	HC-86	Yes	N/A	1E	IA
	Primary		HC-87	Yes	N/A	1E	IA
	Primary		HC-88	Yes	N/A	1E	IA
	Primary		HC-89	Yes	N/A	1E	IA
	Primary	SG AFW supply indicating hand controller	HIC-70	Yes	N/A	1E	IA
	Primary		HIC-71	Yes	N/A	1E	IA
	Primary		HIC-72	Yes	N/A	1E	IA
	Primary		HIC-73	Yes	N/A	1E	IA
	Primary	SG AFW LCV position modulator	POM-110	Yes	N/A	1E	IA
	Primary		POM-111	Yes	N/A	1E	IA
	Primary		POM-113	Yes	N/A	1E	IA
	Primary		POM-115	Yes	N/A	1E	IA
	Primary	SG AFW LCV position transmitter	POT-110	Yes	N/A	1E	IA
	Primary		POT-111	Yes	N/A	1E	IA
	Primary		POT-113	Yes	N/A	1E	IA
	Primary		POT-115	Yes	N/A	1E	IA
	Backup	AFWP discharge pressure indicator	PI-52A/B	Yes	N/A	1E	IA
	Backup		PI-53A/B	Yes	N/A	1E	IA
Primary	SG level	LM-519A	Yes	N/A	1E	IA	
Primary		LM-529A	Yes	N/A	1E	IA	
Primary		LM-539A	Yes	N/A	1E	IA	
Primary		LM-549A	Yes	N/A	1E	IA	

N/A - Not Applicable.



ATTACHMENT TO ENCLOSURE 1

List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>
(2)	Primary	SG level	LT-519	Yes	N/A	1E	IA
	Primary		LT-529	Yes	N/A	1E	IA
	Primary		LT-539	Yes	N/A	1E	IA
	Primary		LT-549	Yes	N/A	1E	IA
	Primary	SG level	LQ-519	Yes	N/A	1E	IA
	Primary		LQ-529	Yes	N/A	1E	IA
	Primary		LQ-539	Yes	N/A	1E	IA
	Primary		LQ-549	Yes	N/A	1E	IA
	Primary	SG level	LM-519	Yes	N/A	1E	IA
	Primary		LM-529	Yes	N/A	1E	IA
	Primary		LM-539	Yes	N/A	1E	IA
	Primary		LM-549	Yes	N/A	1E	IA
	Primary	SG level	LC-519A/B	Yes	N/A	1E	IA
	Primary		LC-529A/C	Yes	N/A	1E	IA
	Primary		LC-539A/C	Yes	N/A	1E	IA
	Primary		LC-549A/C	Yes	N/A	1E	IA
	Primary	Main steam radiation monitor	RE-71	Yes	N/A	1E	IB/E/Cat 2
	Primary		RE-72	Yes	N/A	1E	IB/E/Cat 2
	Primary		RE-73	Yes	N/A	1E	IB/E/Cat 2
	Primary		RE-74	Yes	N/A	1E	IB/E/Cat 2

N/A - Not Applicable.



ATTACHMENT TO ENCLOSURE 1

List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>
(3) Isolate Flow From Ruptured SG(s):	Primary	SG PORV (10% dump)	PCV-19	Yes	Air (Back up N ₂)	1E	IA
	Primary		PCV-20	Yes	Air (Back up N ₂)	1E	IA
	Primary		PCV-21	Yes	Air (Back up N ₂)	1E	IA
	Primary		PCV-22	Yes	Air (Back up N ₂)	1E	IA
	Backup	SG PORV block valves	1015	Yes	Manual	N/A	N/A
	Backup		2015	Yes	Manual	N/A	N/A
	Backup		3015	Yes	Manual	N/A	N/A
	Backup		4015	Yes	Manual	N/A	N/A
	Primary	MSIV	FCV 41	Yes	Air (fail closed)	1E	IA
	Primary		FCV 42	Yes	Air (fail closed)	1E	IA
	Primary		FCV 43	Yes	Air (fail closed)	1E	IA
	Primary		FCV 44	Yes	Air (fail closed)	1E	IA
	Primary	MSIV bypass valves	FCV 22	Yes	Air (fail closed)	1E	IA
	Primary		FCV 23	Yes	Air (fail closed)	1E	IA
	Primary		FCV 24	Yes	Air (fail closed)	1E	IA
	Primary		FCV 25	Yes	Air (fail closed)	1E	IA
Primary	T-D AFWP steam supply isolation valves	FCV 37	Yes	Motor	1E	IA	
Primary		FCV 38	Yes	Motor	1E	IA	
Backup	T-D AFWP steam supply valve	FCV 95	Yes	Motor	1E	IA	

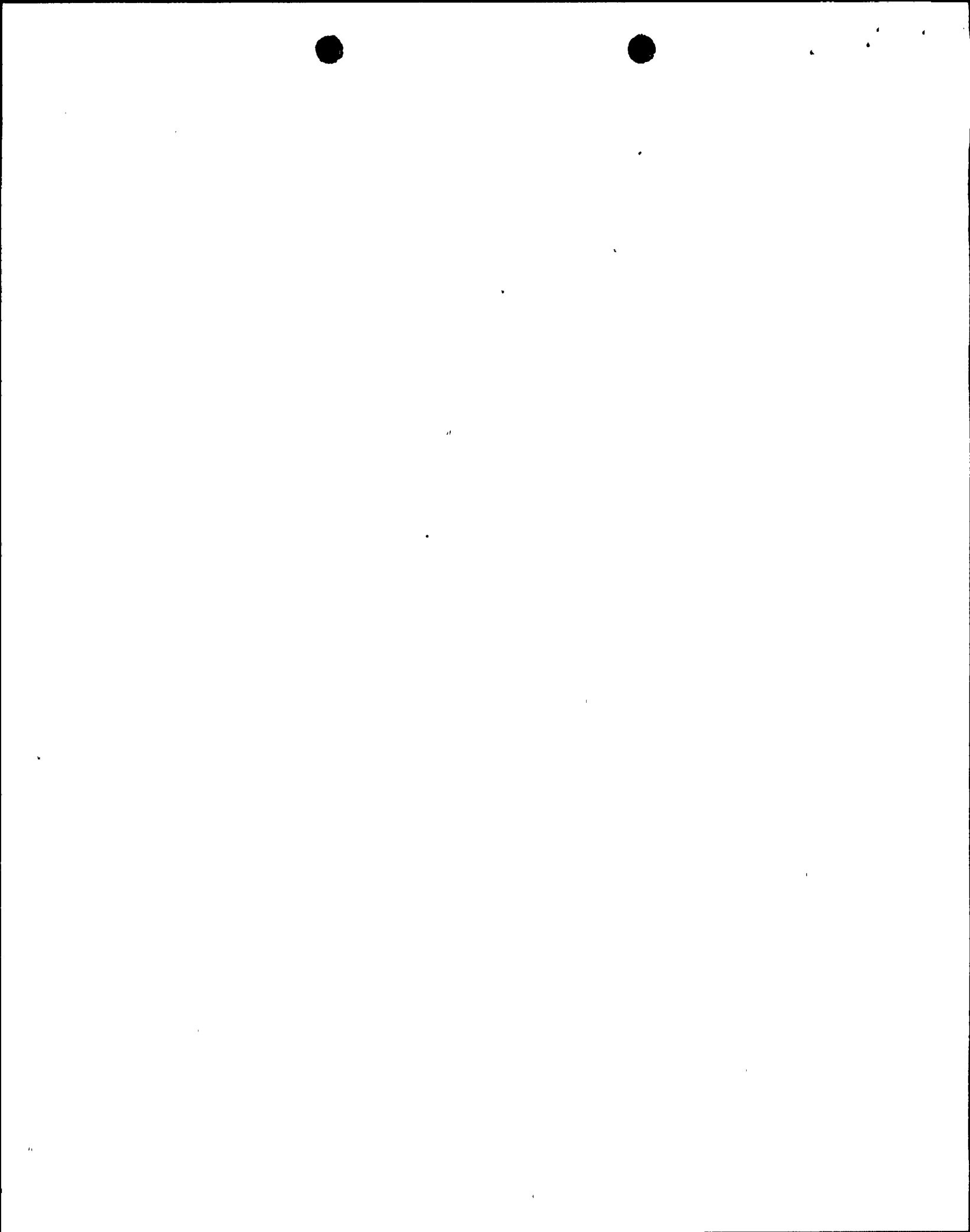
N/A - Not Applicable.



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List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3

<u>EP E-3</u> <u>Reference</u> <u>Step</u>	<u>Primary</u> <u>or</u> <u>Backup</u>	<u>Equipment</u> <u>Name</u>	<u>Equipment</u> <u>Number</u>	<u>Safety</u> <u>Grade</u>	<u>Motive</u> <u>Power</u>	<u>Power</u> <u>Supply</u>	<u>Instrument</u> <u>Class</u>
(3)	Primary	SG blowdown isolation valves	FCV 760	Yes	Air(fail closed)	1E	IA
	Primary		FCV 761	Yes	Air(fail closed)	1E	IA
	Primary		FCV 762	Yes	Air(fail closed)	1E	IA
	Primary		FCV 763	Yes	Air(fail closed)	1E	IA
	Primary	SG blowdown isolation valves	FCV 151	Yes	Air(fail closed)	1E	IA
	Primary		FCV 154	Yes	Air(fail closed)	1E	IA
	Primary		FCV 157	Yes	Air(fail closed)	1E	IA
	Primary		FCV 160	Yes	Air(fail closed)	1E	IA
	Primary	SG blowdown sample isolation valves	FCV 244	Yes	Air(fail closed)	1E	IA
	Primary		FCV 246	Yes	Air(fail closed)	1E	IA
	Primary		FCV 248	Yes	Air(fail closed)	1E	IA
	Primary		FCV 250	Yes	Air(fail closed)	1E	IA

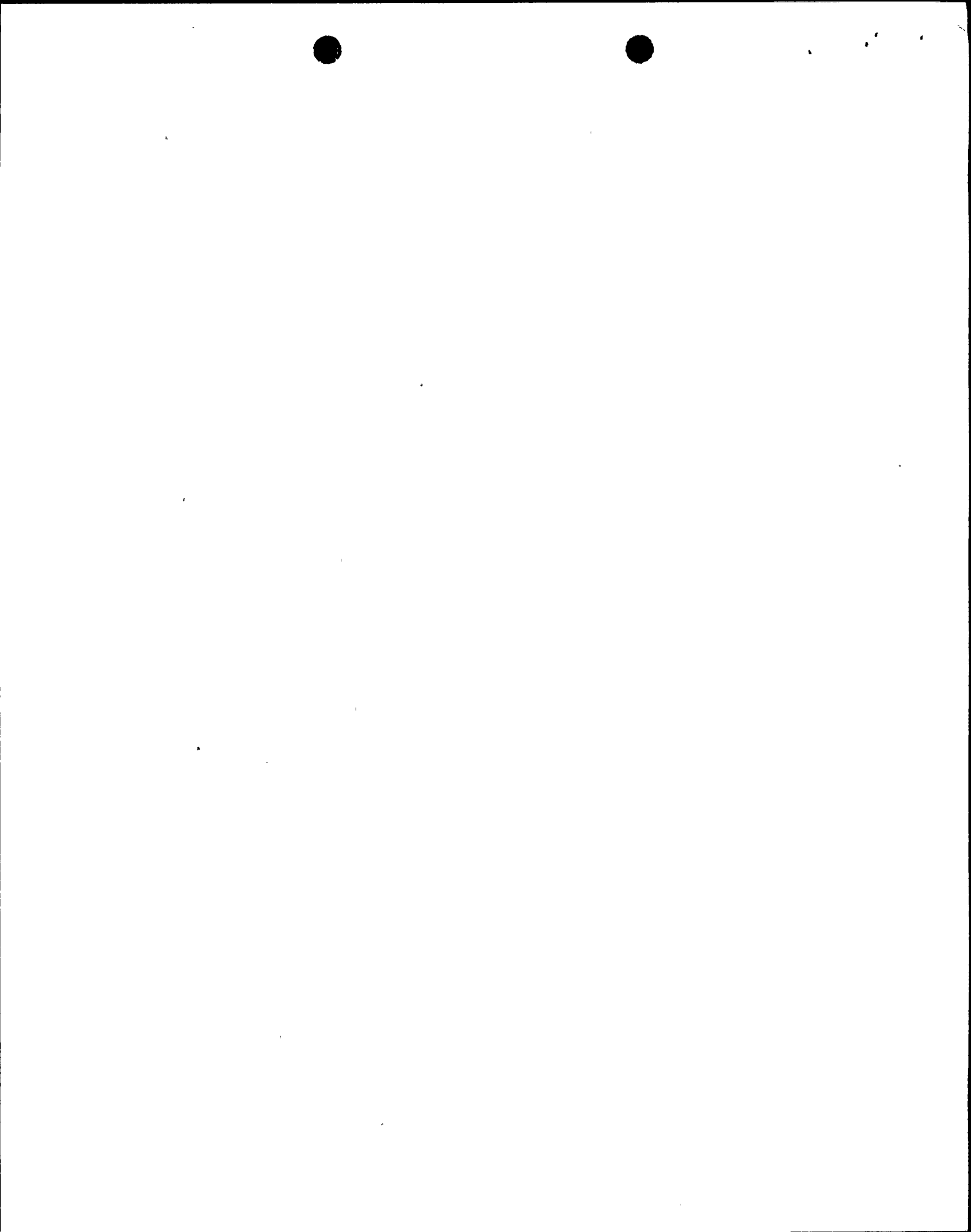


ATTACHMENT TO ENCLOSURE 1

List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>
(3)	Primary	SG safety valves	RV 3	Yes	Mechanical	N/A	N/A
	Primary		RV 4	Yes	Mechanical	N/A	N/A
	Primary		RV 5	Yes	Mechanical	N/A	N/A
	Primary		RV 6	Yes	Mechanical	N/A	N/A
	Primary		RV 7	Yes	Mechanical	N/A	N/A
	Primary		RV 8	Yes	Mechanical	N/A	N/A
	Primary		RV 9	Yes	Mechanical	N/A	N/A
	Primary		RV 10	Yes	Mechanical	N/A	N/A
	Primary		RV 11	Yes	Mechanical	N/A	N/A
	Primary		RV 12	Yes	Mechanical	N/A	N/A
	Primary		RV 13	Yes	Mechanical	N/A	N/A
	Primary		RV 14	Yes	Mechanical	N/A	N/A
	Primary		RV 58	Yes	Mechanical	N/A	N/A
	Primary		RV 59	Yes	Mechanical	N/A	N/A
	Primary		RV 60	Yes	Mechanical	N/A	N/A
	Primary		RV 61	Yes	Mechanical	N/A	N/A
	Primary		RV 222	Yes	Mechanical	N/A	N/A
Primary	RV 223	Yes	Mechanical	N/A	N/A		
Primary	RV 224	Yes	Mechanical	N/A	N/A		
Primary	RV 225	Yes	Mechanical	N/A	N/A		
(4) Check Ruptured SG Level:	Primary	AFW supply valves	See (2) above.				
	Backup	AFW pumps	See (2) above.				
	Primary	Main FW supply valves	FCV 510	Yes	Air(fail closed)	1E	IA
	Primary		FCV 520	Yes	Air(fail closed)	1E	IA
	Primary		FCV 530	Yes	Air(fail closed)	1E	IA
	Primary		FCV 540	Yes	Air(fail closed)	1E	IA

N/A - Not Applicable.



ATTACHMENT TO ENCLOSURE 1

**List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3**

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>	
(4)	Primary	Main FW supply bypass valves	FCV 1510	Yes	Air(fail closed)	1E	IA	
	Primary		FCV 1520	Yes	Air(fail closed)	1E	IA	
	Primary		FCV 1530	Yes	Air(fail closed)	1E	IA	
	Primary		FCV 1540	Yes	Air(fail closed)	1E	IA	
	Backup	Main FW isolation valves	FCV 438	Yes	Motor	1E	IA	
	Backup		FCV 439	Yes	Motor	1E	IA	
	Backup		FCV 440	Yes	Motor	1E	IA	
	Backup		FCV 441	Yes	Motor	1E	IA	
	(5) Check PZR PORVs And Stop Valves:	Primary	PRZR PORVs	PCV 455C	Yes	Air (Back up N ₂)	1E	IA
		Primary		PCV 456	Yes	Air (Back up N ₂)	1E	IA
Primary		PCV 474		No	Air (fail closed)	1E	II	
Backup		PRZR PORV isolation	8000A	Yes	Motor	1E	IA	
Backup			8000B	Yes	Motor	1E	IA	
Backup			8000C	Yes	Motor	1E	IA	
(7) Check Intact SG Levels:		Primary	AFW flow control valves	See (2) above.				
	Backup	AFW pumps	See (2) above.					

(6) Check If SGs Are Not Faulted: Note - There is no equipment credited for accident mitigation in this step.



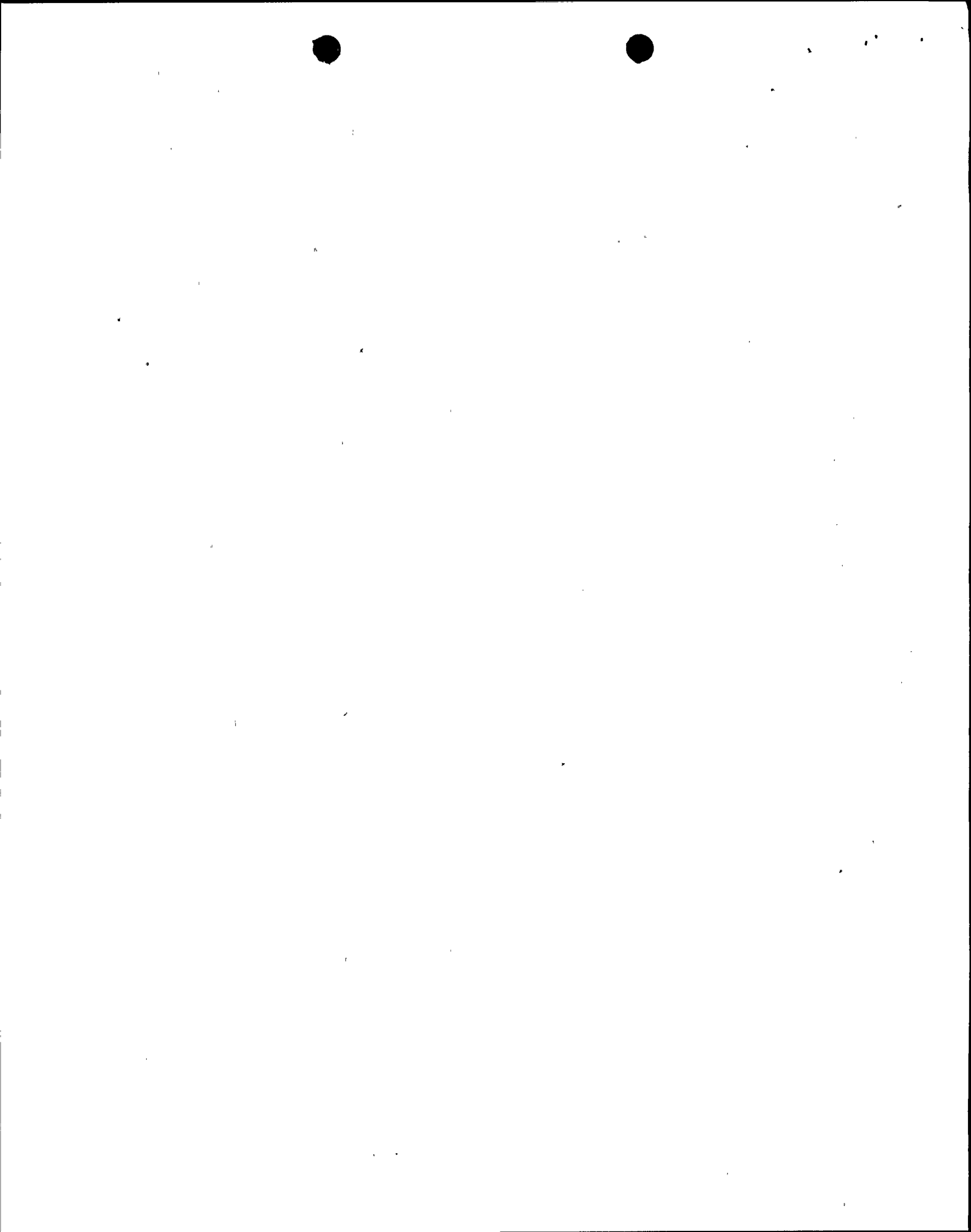
ATTACHMENT TO ENCLOSURE 1

**List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3**

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>	
(8) Reset SI:	Primary	SI reset device	SSPS "K" relay	Yes	N/A	1E	N/A	
(9) Reset Both Trains Of Containment Isolation Phase A And Phase B:	Primary	Cont. phase A reset device	SSPS "K" relay	Yes	N/A	1E	N/A	
(10) Establish Instrument Air To Containment:	Primary Primary Primary	Instrument air compressor Instrument air isolation valve	01 02 FCV 584	*No *No Yes	N/A N/A Air (fail closed)	Non 1E Non 1E 1E	N/A N/A IA	
(11) Verify All AC Buses:	Primary Primary Primary Primary Primary Primary Primary	12 Kv Bus 12 Kv Bus 4 Kv Bus 4 Kv Bus 4 Kv Bus 4 Kv Bus 4 Kv Bus	E D E D F G H	No No No No Yes Yes Yes	N/A N/A N/A N/A N/A N/A N/A	<u>Power</u> Non 1E Non 1E Non 1E Non 1E 1E 1E 1E	<u>Control</u> 1E 1E 1E 1E 1E 1E 1E	N/A N/A N/A N/A N/A N/A N/A

N/A - Not Applicable.

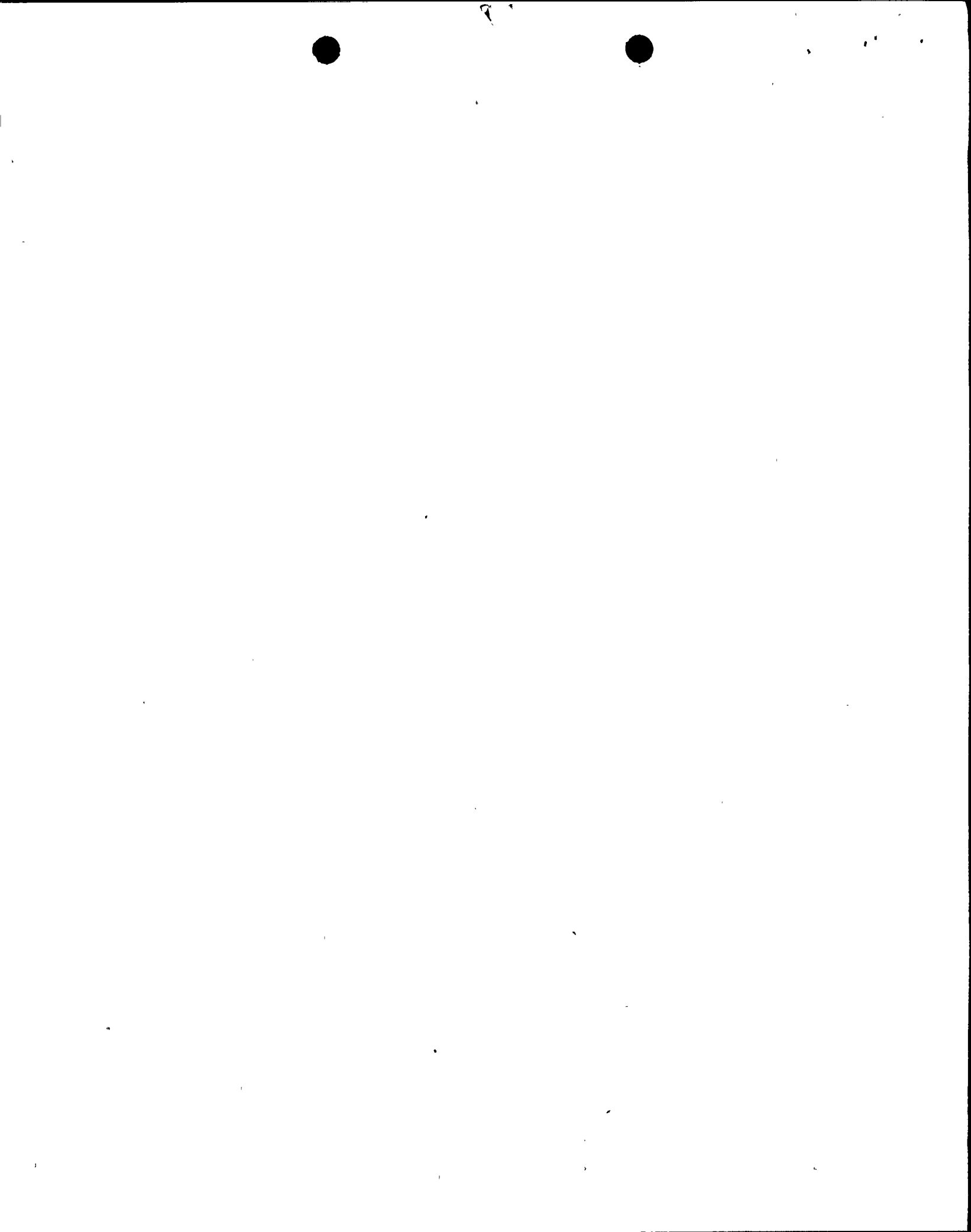
* - Design Class I actuators use local N₂ tanks, or the component fails safe.



ATTACHMENT TO ENCLOSURE 1

List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>
(11)	Backup	Emergency	1-1	Yes	Air/fuel	1E	N/A
	Backup	Diesel Generator	1-2	Yes	Air/fuel	1E	N/A
	Backup		1-3	Yes	Air/fuel	1E	N/A
(12) Check If RHR Pumps Should Be Stopped:	Primary	RHR Pumps	1-1	Yes	Motor	1E	N/A
	Primary		1-2	Yes	Motor	1E	N/A
(13) Check Ruptured SGs Pressure: Note - There is no equipment credited for accident mitigation in this step.							
(14) Initiate RCS Cooldown	Primary	10% steam dump valves, to atmosphere	See (3) above.				
(15) Check Ruptured SGs Pressure: Note - There is no equipment credited for accident mitigation in this step.							
(16) Check RCS Subcooling: Note - There is no equipment credited for accident mitigation in this step.							

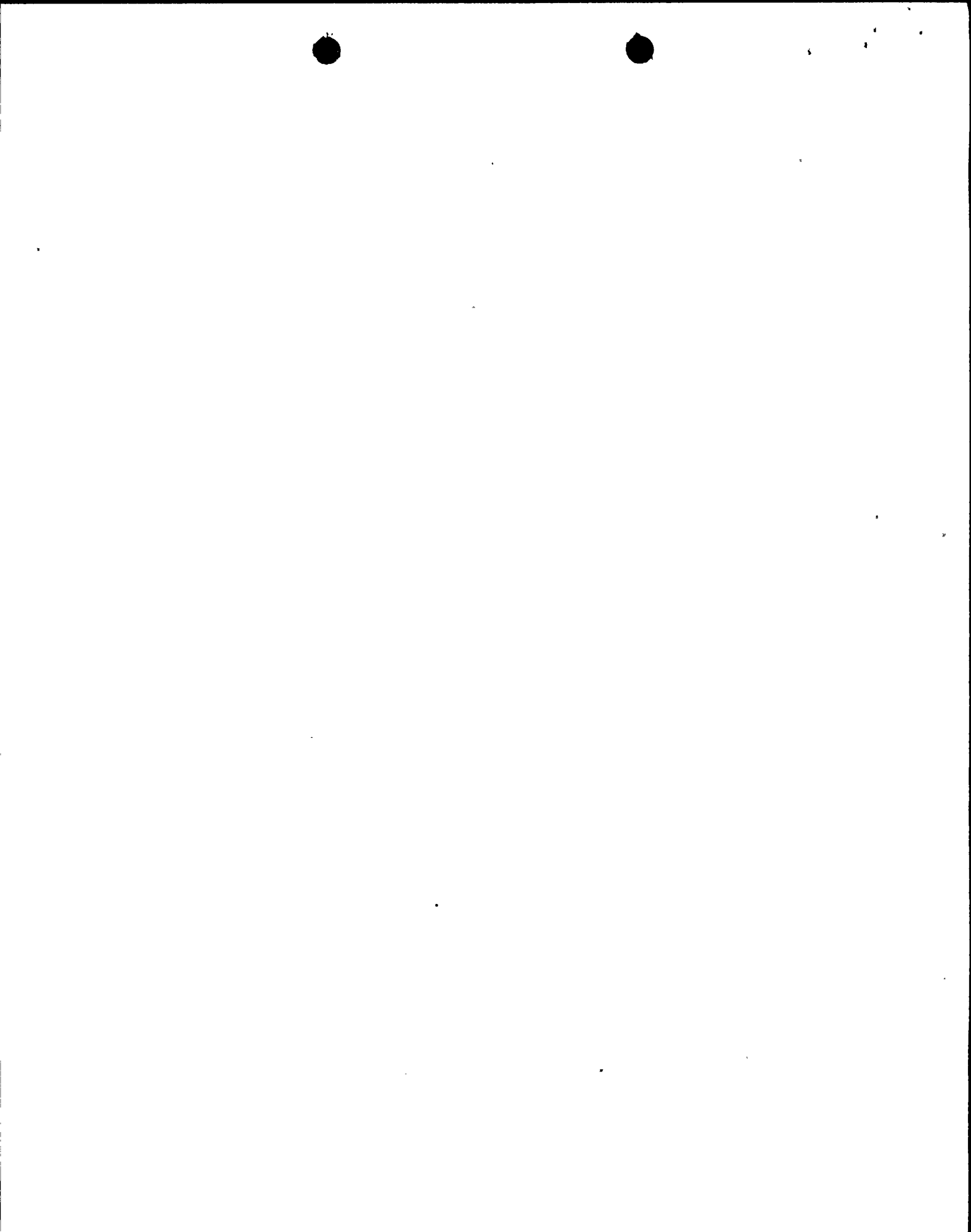


ATTACHMENT TO ENCLOSURE 1

List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>
(17) Depressurize RCS To Minimize Break Flow And Refill PZR:	Primary	PZR spray valves	PCV-455A PCV-455B	No No	Air (Fail Closed) Air (Fail Closed)	1E 1E	II II
	Backup	PZR auxiliary spray valves	8145 8148	Yes Yes	Air (backup air) Air (backup air)	1E 1E	IA IA
(20) Check If ECCS Flow Should Be Terminated:							
(21) Stop ECCS Pumps And Place In Standby:	Primary Primary	SI pump switches	52HF15/CS 52HH15/CS	Yes Yes	N/A N/A	1E 1E	N/A N/A
(22) Establish Charging Flow:	Primary Primary	Charging pumps flow control valves	FCV-128 HCV-142	No Yes	Air (fail open) Air (Back up N ₂)	1E 1E	II IA

N/A - Not Applicable.

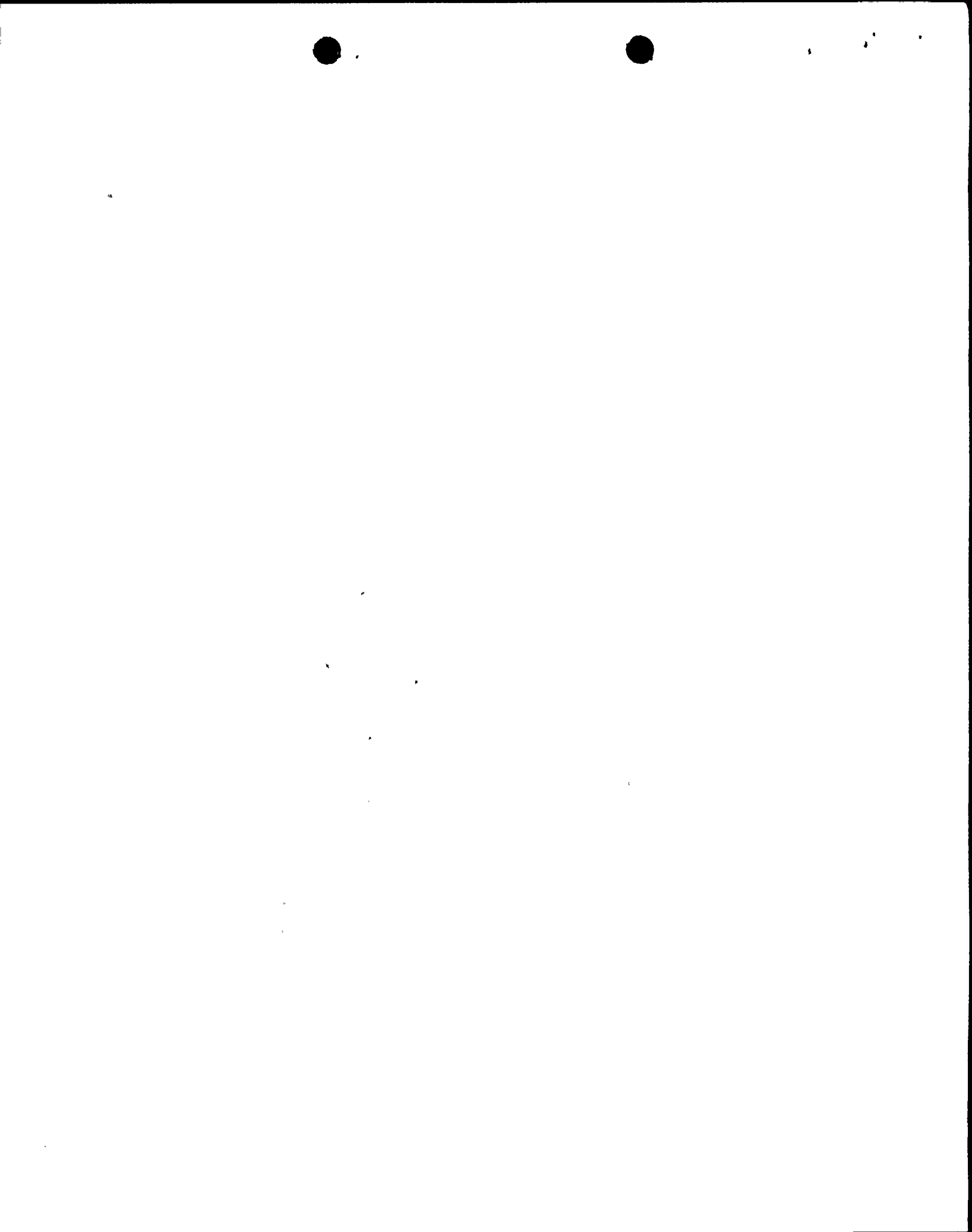


ATTACHMENT TO ENCLOSURE 1

List of Systems, Components, and Instrumentation
Which are Credited for Accident Mitigation in Diablo Canyon SGTR Emergency Operating Procedure EP E-3

<u>EP E-3 Reference Step</u>	<u>Primary or Backup</u>	<u>Equipment Name</u>	<u>Equipment Number</u>	<u>Safety Grade</u>	<u>Motive Power</u>	<u>Power Supply</u>	<u>Instrument Class</u>
(22)	Primary	Charging pumps flow control valves	8105	Yes	Motor	1E	IA
	Primary		8106	Yes	Motor	1E	IA
	Primary	Charging line to cold leg	8146	No	Air (Fail Open)	1E	II
	Primary		8147	No	Air (Fail Open)	1E	II
	Primary	Charging line to cold leg	8107	Yes	Motor	1E	IA
Primary	8108		Yes	Motor	1E	IA	
(23) Isolate BIT:	Backup	SI pumps	SI 1-1	Yes	Motor	1E	N/A
	Backup		SI 1-2	Yes	Motor	1E	N/A
	Backup	BIT isolation valves	See (23) below.				
(23) Isolate BIT:	Primary	BIT isolation valves	8801A	Yes	Motor	1E	IA
			8801B	Yes	Motor	1E	IA
			8803A	Yes	Motor	1E	IA
			8803B	Yes	Motor	1E	IA
	Backup	SI pumps	See (22) above.				

N/A - Not Applicable.



ENCLOSURE 2

Attachment 1 provides a summary of the methodology used in WCAP-11723 and WCAP-11724.

Attachment 2 provides a Westinghouse authorization letter (CAW-88-015), Proprietary Information Notice, and accompanying affidavit.

Attachment 3 provides a copy of WCAP-11723 and WCAP-11724.



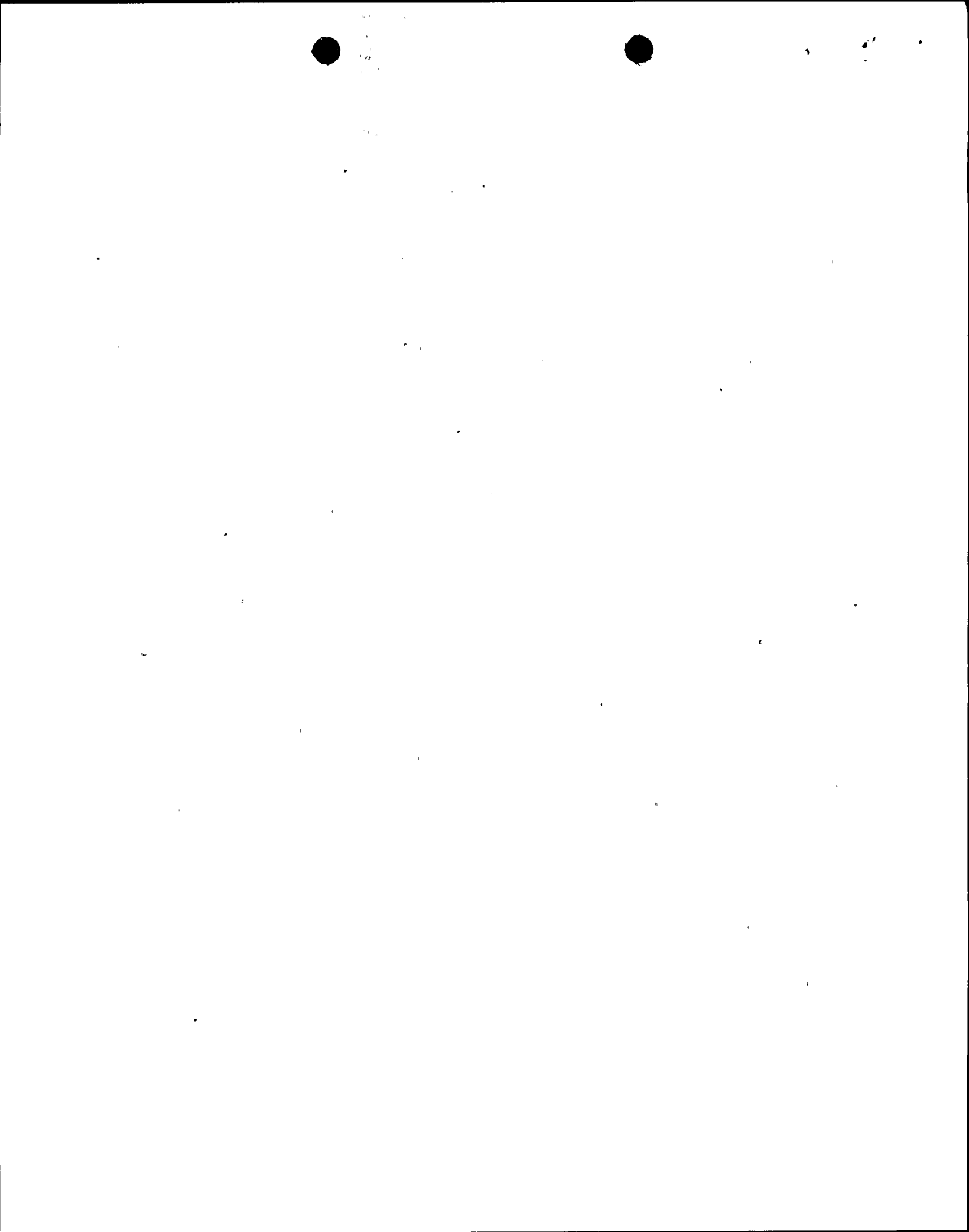
ATTACHMENT 1. TO ENCLOSURE 2

Methodology of WCAP-11723 and WCAP-11724

The analysis provided in WCAP-11723 and WCAP-11724 addresses both the margin to steam generator overfill and the calculated offsite radiation doses. The analysis was performed using the methodology developed in WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," and Supplement 1 to WCAP-10698, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident." This methodology was developed by the WOG SGTR Subgroup and was approved by the NRC in their evaluations dated December 17, 1985 and March 30, 1987. The results of the analysis indicate that margin to overfill is demonstrated for DCPD and the calculated offsite radiation doses are within the guidelines of 10CFR100 and SRP 15.6.3. Therefore it is concluded that the consequences of a design basis steam generator tube rupture at DCPD are acceptable.

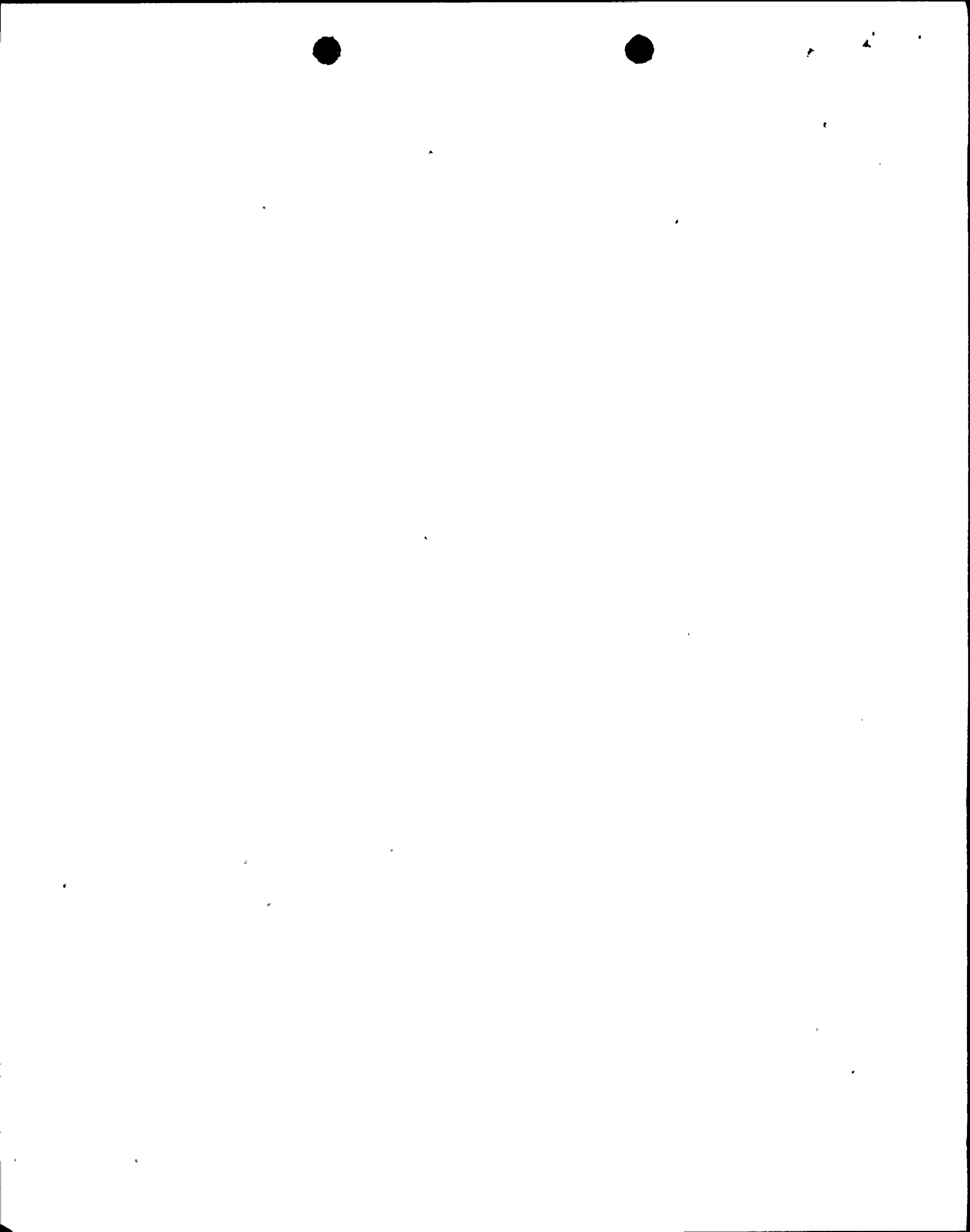
The LOFTTR2 program, an updated version of the LOFTTR1 program, was used to perform the SGTR analysis for DCPD provided in WCAP-11723/WCAP-11724. The LOFTTR1 program was developed as part of the revised SGTR analysis methodology and was used for the SGTR evaluations in WCAP-10698 and Supplement 1 to WCAP-10698. However, the LOFTTR1 program was subsequently modified to accommodate steam generator overfill and the revised program, designated as LOFTTR2, was used for the evaluation of the consequences of overfill in WCAP-11002, "Evaluation of Steam Generator Overfill Due to a Steam Generator Tube Rupture Accident." The LOFTTR2 program is identical to the LOFTTR1 program, with the exception that the LOFTTR2 program has the additional capability to represent the transition from two regions (steam and water) on the secondary side to a single water region if overfill occurs, and the transition back to two regions again depending upon the calculated secondary conditions. Since the LOFTTR2 program has been validated against the LOFTTR1 program, the LOFTTR2 program is also appropriate for performing licensing basis SGTR analyses.

Westinghouse has recently notified all affected utilities of the potential for an increase in the calculated radioactivity release to the environment following an SGTR event and certain other accidents due to the potential uncovering of the steam generator tubes after a reactor trip. The notification to PG&E for DCPD was provided in Westinghouse letter PGE-87-164, dated December 28, 1987. This potential concern has been addressed in the analysis of the DCPD offsite radiation doses for an SGTR in WCAP-11723/WCAP-11724. The offsite dose analysis methodology includes a calculation of the water level relative to the top of the tubes for the ruptured and intact steam generators and the effect of the water level on the iodine transport. For the DCPD analysis, the calculated water level in both the ruptured and intact steam generators drops below the top of the tubes soon after reactor trip, but then begins to increase and recovers the top of the tubes a short time later. When the rupture location is greater than approximately 12 inches below the secondary water level, the iodine



transferred by the unflashed portion of the rupture flow is assumed to mix with the secondary water and partition between the water and steam. The rupture location is assumed to be at the intersection of the outer tube row and the upper anti-vibration bar, which is approximately 4 inches below the apex of the tube bundle. Consistent with other SGTR analyses performed by Westinghouse using the WOG SGTR Subgroup methodology, it is assumed that all of the break flow activity is released to the environment if the water level above the rupture location is less than approximately 12 inches.

The SGTR analysis methodology in WCAP-10698 is based on the use of the maximum attainable safety injection (SI) flow rate since this produces conservative results for the margin to overfill and offsite radiation dose analyses. Realistic estimates of the maximum attainable SI flow rates were developed for DCPD based on the actual SI system design and operating performance for use in the margin to overfill analysis. The SI flow rates used conservatively represent the maximum attainable flow rates for DCPD. Therefore, the analysis is consistent with the methodology in WCAP-10698.



ATTACHMENT 2 TO ENCLOSURE 2

Westinghouse authorization letter CAW-88-015, proprietary information notice,
and accompanying affidavit AW-76-31.

