Facility: Diablo Canyon Power Plant

Form ES-401-3

Exam Date: 10/21/2002 **Exam Level:** SRO

Tier	Group	K/A Category Points											Point Total
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	
1. Emergency & Abnormal Plant Evolutions	1	4	4	4				4	4			4	24
	2	3	3	2				2	3			3	16
	3	1	0	0				0	1			1	3
	Tier Totals	8	7	6				6	8			8	43
2. Plant Systems	1	1	2	2	2	2	1	2	2	2	1	2	19
	2	1	2	2	2	1	1	1	2	2	1	2	17
	3	0	1	0	0	0	1	0	0	1	0	1	4
	Tier Totals	2	5	4	4	3	3	3	4	5	2	5	40

3. Generic Knowledge And Abilities	Cat 1	Cat 2	Cat 3	Cat 4	
	5	4	4	4	17

Note: 1. Ensure that at least two topics from every K/A category are sampled within each teir (i.e., the "Tier Totals" in each

2. Actual point totals must match those specified in the table.
3. Select topics from many systems; avoid selecting more than two or three K/A topics from a given system unless
4. Systems/evolutions within each group are identified on the associated outline.
5. The shaded areas are not applicable to the category/tier.
6. The generic K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be
7. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings for the RO license level, and the point totals for each system and category. K/As below 2.5 should be

PWR SRO Examination Outline

Printed: 06/10/2002

Facility: Diablo Canyon Power Plant

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-3**

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
001	Continuous Rod Withdrawal / 1	AA1.04	Operating switch for emergency boration motor-operated valve	Q1
005	Inoperable/Stuck Control Rod / 1	AA1.05	RPI	Q6
017	Reactor Coolant Pump (RCP) Malfunctions (Loss of RC Flow) / 4	AA1.02	RCP oil reservoir level and alarm indicators	Q19
024	Emergency Boration / 1	2.2.30	Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.	Q104
024	Emergency Boration / 1	AK3.02	Actions contained in EOP for emergency boration	Q23
026	Loss of Component Cooling Water (CCW) / 8	AA1.02	Loads on the CCWS in the control room	Q24
029	Anticipated Transient Without Scram (ATWS) / 1	EK2.06	Breakers, relays, and disconnects	Q27
040	Steam Line Rupture / 4	AA2.03	Difference between steam line rupture and LOCA	Q107
051	Loss of Condenser Vacuum / 4	AA2.01	Cause for low vacuum condition	Q108
051	Loss of Condenser Vacuum / 4	2.2.12	Knowledge of surveillance procedures.	Q109

Facility: Diablo Canyon Power Plant

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ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-3**

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
057	Loss of Vital AC Electrical Instrument Bus / 6	AA2.20	Interlocks in effect on loss of ac vital electrical instrument bus that must be bypassed to restore normal equipment operation	Q111
059	Accidental Liquid Radwaste Release / 9	AK1.05	The calculation of offsite doses due to a release from the power plant	Q41
059	Accidental Liquid Radwaste Release / 9	AK3.02	Implementation of E-plan	Q42
062	Loss of Nuclear Service Water / 4	AK3.03	Guidance actions contained in EOP for Loss of nuclear service water	Q47
067	Plant Fire on Site / 9	AK1.01	Fire classifications, by type	Q50
068	Control Room Evacuation / 8	AA2.01	S/G level	Q113
068	Control Room Evacuation / 8	AK3.02	System response to turbine trip	Q51
074	Inadequate Core Cooling / 4	EK2.09	Controllers and positioners	Q57
076	High Reactor Coolant Activity / 9	AK2.01	Process radiation monitors	Q58
E01	Rediagnosis / 3	EK2.1	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	Q60

PWR SRO Examination Outline

Facility: Diablo Canyon Power Plant

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-3**

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
E02	SI Termination / 3	EK1.2	Normal, abnormal and emergency operating procedures associated with SI Termination	Q61
E07	Saturated Core Cooling / 4	2.4.18	Knowledge of the specific bases for EOPs.	Q117
E08	Pressurized Thermal Shock / 4	EK1.2	Normal, abnormal and emergency operating procedures associated with Pressurized Thermal Shock	Q64
E12	Uncontrolled Depressurization of all Steam Generators / 4	2.2.34	Knowledge of the process for determining the internal and external effects on core reactivity.	Q118

PWR SRO Examination Outline

Printed: 06/10/2002

Facility: Diablo Canyon Power Plant

ES - 401 | **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2** | **Form ES-401-3**

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
008	Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) / 3	AA2.05	PORV isolation (block valve switches and indicators)	Q102
008	Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) / 3	AA1.03	Turbine bypass in manual control to maintain header pressure	Q8
009	Small Break LOCA / 3	EK2.03	S/Gs	Q10
027	Pressurizer Pressure Control (PZR PCS) Malfunction / 3	AK1.01	Definition of saturation temperature	Q25
027	Pressurizer Pressure Control (PZR PCS) Malfunction / 3	AK2.03	Controllers and positioners	Q26
038	Steam Generator Tube Rupture (SGTR) / 3	2.4.15	Knowledge of communications procedures associated with EOP implementation.	Q105
054	Loss of Main Feedwater (MFW) / 4	AK1.01	MFW line break depressurizes the S/G (similar to a steam line break)	Q36
054	Loss of Main Feedwater (MFW) / 4	AK3.03	Manual control of AFW flow control valves	Q37
058	Loss of DC Power / 6	AK1.01	Battery charger equipment and instrumentation	Q40
061	Area Radiation Monitoring (ARM) System Alarms / 7	AA1.01	Automatic actuation	Q45
065	Loss of Instrument Air / 8	AA2.07	Whether backup nitrogen supply is controlling valve position	Q112

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
E03	LOCA Cooldown and Depressurization / 4	EK3.2	Normal, abnormal and emergency operating procedures associated with LOCA Cooldown and Depressurization	Q62
E05	Loss of Secondary Heat Sink / 4	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control; 2. Core cooling and heat removal; 3. Reactor coolant system integrity; 4. Containment conditions; 5. Radioactivity release control.	Q63
E05	Loss of Secondary Heat Sink / 4	2.1.25	Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.	Q116
E11	Loss of Emergency Coolant Recirculation / 4	EK2.2	Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	Q65
E11	Loss of Emergency Coolant Recirculation / 4	EA2.1	Facility conditions and selection of appropriate procedures during abnormal and emergency operations	Q66

PWR SRO Examination Outline

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Facility: Diablo Canyon Power Plant

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 3** **Form ES-401-3**

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
056	Loss of Offsite Power / 6	AA2.42	Occurrence of a reactor trip	Q110
E13	Steam Generator Overpressure / 4	2.4.35	Knowledge of local auxiliary operator tasks during emergency operations including system geography and system implications.	Q119
E15	Containment Flooding / 5	EK1.3	Annunciators and conditions indicating signals, and remedial actions associated with the Containment Flooding	Q68

Facility: Diablo Canyon Power Plant

ES - 401 **Plant Systems - Tier 2 / Group 1** Form ES-401-3

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
001	Control Rod Drive System / 1	K5.02	Definitions of differential rod worth and integral rod worth; their applications	Q2
004	Chemical and Volume Control System (CVCS) / 1	K2.06	Control instrumentation	Q4
004	Chemical and Volume Control System (CVCS) / 1	K3.08	RCP seal injection	Q5
013	Engineered Safety Features Actuation System (ESFAS) / 2	K2.01	ESFAS/safeguards equipment control	Q15
013	Engineered Safety Features Actuation System (ESFAS) / 2	A3.02	Operation of actuated equipment	Q16
014	Rod Position Indication System (RPIS) / 1	A4.04	Re-zeroing of rod position prior to startup	Q17
017	In-Core Temperature Monitor (ITM) System / 7	K6.01	Sensors and detectors	Q20
022	Containment Cooling System (CCS) / 5	A1.02	Containment pressure	Q21
022	Containment Cooling System (CCS) / 5	A2.05	Major leak in CCS	Q22
056	Condensate System / 4	A2.04	Loss of condensate pumps	Q39
059	Main Feedwater (MFW) System / 4	K3.04	RCS	Q43
059	Main Feedwater (MFW) System / 4	A1.03	Power level restrictions for operation of MFW pumps and valves	Q44

Facility: Diablo Canyon Power Plant

ES - 401 **Plant Systems - Tier 2 / Group 1** **Form ES-401-3**

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
061	Auxiliary / Emergency Feedwater (AFW) System / 4	A3.01	AFW startup and flows	Q46
063	D.C. Electrical Distribution System / 6	K1.03	Battery charger and battery	Q49
068	Liquid Radwaste System (LRS) / 9	2.2.24	Ability to analyze the affect of maintenance activities on LCO status.	Q114
068	Liquid Radwaste System (LRS) / 9	K4.01	Safety and environmental precautions for handling hot, acidic, and radioactive liquids	Q52
071	Waste Gas Disposal System (WGDS) / 9	2.4.33	Knowledge of the process used track inoperable alarms.	Q115
072	Area Radiation Monitoring (ARM) System / 7	K4.02	Fuel building isolation	Q53
072	Area Radiation Monitoring (ARM) System / 7	K5.02	Radiation intensity changes with source distance	Q54

PWR SRO Examination Outline

Facility: Diablo Canyon Power Plant

ES - 401 | **Plant Systems - Tier 2 / Group 2** | **Form ES-401-3**

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
006	Emergency Core Cooling System (ECCS) / 2	K3.02	Fuel	Q7
010	Pressurizer Pressure Control System (PZR PCS) / 3	K5.01	Determination of condition of fluid in PZR, using steam tables	Q11
011	Pressurizer Level Control System (PZR LCS) / 2	K6.04	Operation of PZR level controllers	Q12
011	Pressurizer Level Control System (PZR LCS) / 2	A2.03	Loss of PZR level	Q13
012	Reactor Protection System / 7	2.2.8	Knowledge of the process for determining if the proposed change, test, or experiment involves an unreviewed safety question.	Q103
012	Reactor Protection System / 7	K2.01	RPS channels, components, and interconnections	Q14
016	Non-Nuclear Instrumentation System (NNIS) / 7	A2.02	Loss of power supply	Q18
029	Containment Purge System (CPS) / 8	A3.01	CPS isolation	Q28
034	Fuel Handling Equipment System (FHES) / 8	A4.02	Neutron levels	Q30
035	Steam Generator System (S/GS) / 4	K4.06	S/G pressure	Q31
035	Steam Generator System (S/GS) / 4	A3.01	S/G water level control	Q32
039	Main and Reheat Steam System (MRSS) / 4	2.4.11	Knowledge of abnormal condition procedures.	Q106

Facility: Diablo Canyon Power Plant

ES - 401 **Plant Systems - Tier 2 / Group 2** **Form ES-401-3**

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
055	Condenser Air Removal System (CARS) / 4	K1.06	PRM system	Q38
062	A.C. Electrical Distribution System / 6	K2.01	Major system loads	Q48
073	Process Radiation Monitoring (PRM) System / 7	A1.01	Radiation levels	Q56
073	Process Radiation Monitoring (PRM) System / 7	K4.01	Release termination when radiation exceeds setpoint	Q55
103	Containment System / 5	K3.01	Loss of containment integrity under shutdown conditions	Q59

PWR SRO Examination Outline

Facility: Diablo Canyon Power Plant

ES - 401 **Plant Systems - Tier 2 / Group 3** **Form ES-401-3**

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
007	Pressurizer Relief Tank/Quench Tank System (PRTS) / 5	2.4.28	Knowledge of procedures relating to emergency response to sabotage.	Q101
008	Component Cooling Water System (CCWS) / 8	K2.02	CCW pump, including emergency backup	Q9
041	Steam Dump System (SDS) and Turbine Bypass Control / 4	K6.03	Controller and positioners, including ICS, S/G, CRDS	Q33
041	Steam Dump System (SDS) and Turbine Bypass Control / 4	A3.02	RCS pressure, RCS temperature, and reactor power	Q34

Generic Knowledge and Abilities Outline (Tier 3)

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PWR SRO Examination Outline

Form ES-401-5

Facility: Diablo Canyon Power Plant

Generic Category	KA	KA Topic	Comment
Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements.	Q70
	2.1.3	Knowledge of shift turnover practices.	Q71
	2.1.13	Knowledge of facility requirements for controlling vital / controlled access.	Q120
	2.1.32	Ability to explain and apply all system limits and precautions.	Q69
	2.1.33	Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.	Q121

Category Total: 5

Generic Category	KA	KA Topic	Comment
Equipment Control	2.2.4	(multi-unit) Ability to explain the variations in control board layouts, systems, instrumentation and procedural actions between units at a facility.	Q3
	2.2.14	Knowledge of the process for making configuration changes.	Q123
	2.2.31	Knowledge of procedures and limitations involved in initial core loading.	Q122
	2.2.32	Knowledge of the effects of alterations on core configuration.	Q124

Category Total: 4

Generic Category	KA	KA Topic	Comment
Radiation Control	2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements.	Q35
	2.3.3	Knowledge of SRO responsibilities for auxiliary systems that are outside the control room (e.g., waste disposal and handling systems).	Q126
	2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	Q29
	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	Q125

Category Total: 4

Generic Knowledge and Abilities Outline (Tier 3)

Printed: 06/10/2002

PWR SRO Examination Outline

Form ES-401-5

Facility: Diablo Canyon Power Plant

Generic Category	KA	KA Topic	Comment
Emergency Procedures/Plan	2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	Q127
	2.4.27	Knowledge of fire in the plant procedure.	Q129
	2.4.32	Knowledge of operator response to loss of all annunciators.	Q128
	2.4.45	Ability to prioritize and interpret the significance of each annunciator or alarm.	Q67

Category Total: 4

Generic Total: 17

Facility: Diablo Canyon Power Plant Form ES-401-4

Exam Date: 10/21/2002 **Exam Level:** RO

Tier	Group	K/A Category Points											Point Total
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	
1. Emergency & Abnormal Plant Evolutions	1	3	3	3				3	2			2	16
	2	4	4	3				3	2			1	17
	3	1	1	1				0	0			0	3
	Totals Tier	8	8	7				6	4			3	36
2. Plant Systems	1	3	2	2	2	2	2	2	2	2	2	2	23
	2	2	2	2	2	1	2	2	2	2	2	1	20
	3	1	1	1	1	0	1	0	0	1	2	0	8
	Tier Totals	6	5	5	5	3	5	4	4	5	6	3	51
3. Generic Knowledge And Abilities					Cat 1		Cat 2		Cat 3		Cat 4		
					3		3		3		4		13

Note: 1. Ensure that at least two topics from every K/A category are sampled within each teir (i.e., the "Tier Totals" in each
2. Actual point totals must match those specified in the table.
3. Select topics from many systems; avoid selecting more than two or three K/A topics from a given system unless
4. Systems/evolutions within each group are identified on the associated outline.
5. The shaded areas are not applicable to the category /tier.
6. The generic K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be
7. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings for
the RO license level, and the point totals for each system and category. K/As below 2.5 should be

PWR RO Examination Outline

Facility: Diablo Canyon Power Plant

Printed: 06/10/2002

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-4**

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
005	Inoperable/Stuck Control Rod / 1	AA1.05	RPI	Q6
015	Reactor Coolant Pump (RCP) Malfunctions / 4	AA2.01	Cause of RCP failure	Q80
017	Reactor Coolant Pump (RCP) Malfunctions (Loss of RC Flow) / 4	AA1.02	RCP oil reservoir level and alarm indicators	Q19
024	Emergency Boration / 1	AK3.02	Actions contained in EOP for emergency boration	Q23
024	Emergency Boration / 1	2.1.25	Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.	Q83
026	Loss of Component Cooling Water (CCW) / 8	AA1.02	Loads on the CCWS in the control room	Q24
027	Pressurizer Pressure Control (PZR PCS) Malfunction / 3	AK1.01	Definition of saturation temperature	Q25
027	Pressurizer Pressure Control (PZR PCS) Malfunction / 3	AK2.03	Controllers and positioners	Q26
055	Loss of Offsite and Onsite Power (Station Blackout) / 6	2.2.27	Knowledge of the refueling process.	Q85
062	Loss of Nuclear Service Water / 4	AK3.03	Guidance actions contained in EOP for Loss of nuclear service water	Q47
067	Plant Fire on Site / 9	AK1.01	Fire classifications, by type	Q50

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PWR RO Examination Outline

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ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** Form ES-401-4

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
068	Control Room Evacuation / 8	AK3.02	System response to turbine trip	Q51
074	Inadequate Core Cooling / 4	EK2.09	Controllers and positioners	Q57
076	High Reactor Coolant Activity / 9	AK2.01	Process radiation monitors	Q58
E07	Saturated Core Cooling / 4	EA2.2	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	Q93
E08	Pressurized Thermal Shock / 4	EK1.2	Normal, abnormal and emergency operating procedures associated with Pressurized Thermal Shock	Q64

PWR RO Examination Outline

Facility: Diablo Canyon Power Plant

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ES - 401 | Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2 | Form ES-401-4

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
001	Continuous Rod Withdrawal / 1	AA1.04	Operating switch for emergency boration motor-operated valve	Q1
008	Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) / 3	AA1.03	Turbine bypass in manual control to maintain header pressure	Q8
009	Small Break LOCA / 3	EK2.03	S/Gs	Q10
029	Anticipated Transient Without Scram (ATWS) / 1	EK2.06	Breakers, relays, and disconnects	Q27
054	Loss of Main Feedwater (MFW) / 4	AK1.01	MFW line break depressurizes the S/G (similar to a steam line break)	Q36
054	Loss of Main Feedwater (MFW) / 4	AK3.03	Manual control of AFW flow control valves	Q37
058	Loss of DC Power / 6	AK1.01	Battery charger equipment and instrumentation	Q40
059	Accidental Liquid Radwaste Release / 9	AK1.05	The calculation of offsite doses due to a release from the power plant	Q41
059	Accidental Liquid Radwaste Release / 9	AK3.02	Implementation of E-plan	Q42
061	Area Radiation Monitoring (ARM) System Alarms / 7	AA1.01	Automatic actuation	Q45

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
E01	Rediagnosis / 3	EK2.1	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	Q60
E02	SI Termination / 3	EK1.2	Normal, abnormal and emergency operating procedures associated with SI Termination	Q61
E03	LOCA Cooldown and Depressurization / 4	EK3.2	Normal, abnormal and emergency operating procedures associated with LOCA Cooldown and Depressurization	Q62
E04	LOCA Outside Containment / 3	EA2.2	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	Q92
E05	Loss of Secondary Heat Sink / 4	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control; 2. Core cooling and heat removal; 3. Reactor coolant system integrity; 4. Containment conditions; 5. Radioactivity release control.	Q63
E11	Loss of Emergency Coolant Recirculation / 4	EK2.2	Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	Q65
E11	Loss of Emergency Coolant Recirculation / 4	EA2.1	Facility conditions and selection of appropriate procedures during abnormal and emergency operations	Q66

PWR RO Examination Outline

Printed: 06/10/2002

Facility: Diablo Canyon Power Plant

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 3** **Form ES-401-4**

E/APE #	E/APE Name / Safety Function	KA	KA Topic	Comment
E13	Steam Generator Overpressure / 4	EK3.4	RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated	Q75
E15	Containment Flooding / 5	EK1.3	Annunciators and conditions indicating signals, and remedial actions associated with the Containment Flooding	Q68
E15	Containment Flooding / 5	EK2.1	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	Q94

PWR RO Examination Outline

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Facility: Diablo Canyon Power Plant

ES - 401 **Plant Systems - Tier 2 / Group 1** **Form ES-401-4**

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
001	Control Rod Drive System / 1	K5.02	Definitions of differential rod worth and integral rod worth; their applications	Q2
001	Control Rod Drive System / 1	A4.13	Stopping other changes in plant, e.g., turbine, S/G, SDBCS, boration, before adjusting rods	Q72
003	Reactor Coolant Pump System (RCPS) / 4	2.4.47	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	Q77
003	Reactor Coolant Pump System (RCPS) / 4	K6.04	Containment isolation valves affecting RCP operation	Q78
004	Chemical and Volume Control System (CVCS) / 1	K2.06	Control instrumentation	Q4
004	Chemical and Volume Control System (CVCS) / 1	K3.08	RCP seal injection	Q5
013	Engineered Safety Features Actuation System (ESFAS) / 2	K2.01	ESFAS/safeguards equipment control	Q15
013	Engineered Safety Features Actuation System (ESFAS) / 2	A3.02	Operation of actuated equipment	Q16
015	Nuclear Instrumentation System / 7	A4.02	NIS indicators	Q81
017	In-Core Temperature Monitor (ITM) System / 7	K6.01	Sensors and detectors	Q20
017	In-Core Temperature Monitor (ITM) System / 7	K1.02	RCS	Q82
022	Containment Cooling System (CCS) / 5	A1.02	Containment pressure	Q21

Facility: Diablo Canyon Power Plant

ES - 401 **Plant Systems - Tier 2 / Group 1** **Form ES-401-4**

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
022	Containment Cooling System (CCS) / 5	A2.05	Major leak in CCS	Q22
056	Condensate System / 4	A2.04	Loss of condensate pumps	Q39
056	Condensate System / 4	K1.03	MFW	Q86
059	Main Feedwater (MFW) System / 4	K3.04	RCS	Q43
059	Main Feedwater (MFW) System / 4	A1.03	Power level restrictions for operation of MFW pumps and valves	Q44
061	Auxiliary / Emergency Feedwater (AFW) System / 4	K1.04	RCS	Q87
061	Auxiliary / Emergency Feedwater (AFW) System / 4	A3.01	AFW startup and flows	Q46
068	Liquid Radwaste System (LRS) / 9	K4.01	Safety and environmental precautions for handling hot, acidic, and radioactive liquids	Q52
068	Liquid Radwaste System (LRS) / 9	2.1.32	Ability to explain and apply all system limits and precautions.	Q89
072	Area Radiation Monitoring (ARM) System / 7	K4.02	Fuel building isolation	Q53
072	Area Radiation Monitoring (ARM) System / 7	K5.02	Radiation intensity changes with source distance	Q54

PWR RO Examination Outline

Printed: 06/10/2002

Facility: Diablo Canyon Power Plant

ES - 401 **Plant Systems - Tier 2 / Group 2** **Form ES-401-4**

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
002	Reactor Coolant System (RCS) / 2	A1.11	Relative level indications in the RWST, the refueling cavity, the PZR and the reactor vessel during preparation for refueling	Q76
006	Emergency Core Cooling System (ECCS) / 2	K3.02	Fuel	Q7
010	Pressurizer Pressure Control System (PZR PCS) / 3	K5.01	Determination of condition of fluid in PZR, using steam tables	Q11
011	Pressurizer Level Control System (PZR LCS) / 2	K6.04	Operation of PZR level controllers	Q12
011	Pressurizer Level Control System (PZR LCS) / 2	A2.03	Loss of PZR level	Q13
012	Reactor Protection System / 7	K2.01	RPS channels, components, and interconnections	Q14
012	Reactor Protection System / 7	K3.02	T/G	Q79
014	Rod Position Indication System (RPIS) / 1	A4.04	Re-zeroing of rod position prior to startup	Q17
016	Non-Nuclear Instrumentation System (NNIS) / 7	A2.02	Loss of power supply	Q18
029	Containment Purge System (CPS) / 8	A3.01	CPS isolation	Q28
029	Containment Purge System (CPS) / 8	A4.01	Containment purge flow rate	Q73
035	Steam Generator System (S/GS) / 4	K4.06	S/G pressure	Q31

Facility: Diablo Canyon Power Plant

ES - 401 **Plant Systems - Tier 2 / Group 2** **Form ES-401-4**

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
035	Steam Generator System (S/GS) / 4	A3.01	S/G water level control	Q32
055	Condenser Air Removal System (CARS) / 4	K1.06	PRM system	Q38
062	A.C. Electrical Distribution System / 6	K2.01	Major system loads	Q48
063	D.C. Electrical Distribution System / 6	K1.03	Battery charger and battery	Q49
064	Emergency Diesel Generator (ED/G) System / 6	2.1.32	Ability to explain and apply all system limits and precautions.	Q88
073	Process Radiation Monitoring (PRM) System / 7	A1.01	Radiation levels	Q56
073	Process Radiation Monitoring (PRM) System / 7	K4.01	Release termination when radiation exceeds setpoint	Q55
086	Fire Protection System (FPS) / 8	K6.04	Fire, smoke, and heat detectors	Q91

PWR RO Examination Outline

Printed: 06/10/2002

Facility: Diablo Canyon Power Plant

ES - 401 **Plant Systems - Tier 2 / Group 3** **Form ES-401-4**

Sys/Ev #	System / Evolution Name	KA	KA Topic	Comment
008	Component Cooling Water System (CCWS) / 8	K2.02	CCW pump, including emergency backup	Q9
034	Fuel Handling Equipment System (FHES) / 8	A4.02	Neutron levels	Q30
041	Steam Dump System (SDS) and Turbine Bypass Control / 4	K6.03	Controller and positioners, including ICS, S/G, CRDS	Q33
041	Steam Dump System (SDS) and Turbine Bypass Control / 4	A3.02	RCS pressure, RCS temperature, and reactor power	Q34
045	Main Turbine Generator (MT/G) System / 4	A4.02	T/G controls, including breakers	Q74
045	Main Turbine Generator (MT/G) System / 4	K1.06	RCS, during steam valve test	Q84
076	Service Water System (SWS) / 4	K4.03	Automatic opening features associated with SWS isolation valves to CCW heat exchangers	Q90
103	Containment System / 5	K3.01	Loss of containment integrity under shutdown conditions	Q59

Generic Knowledge and Abilities Outline (Tier 3)

Printed: 06/10/2002

PWR RO Examination Outline

Form ES-401-5

Facility: Diablo Canyon Power Plant

Generic Category	KA	KA Topic	Comment
Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements.	Q70
	2.1.3	Knowledge of shift turnover practices.	Q71
	2.1.32	Ability to explain and apply all system limits and precautions.	Q69

Category Total: 3

Generic Category	KA	KA Topic	Comment
Equipment Control	2.2.4	(multi-unit) Ability to explain the variations in control board layouts, systems, instrumentation and procedural actions between units at a facility.	Q3
	2.2.13	Knowledge of tagging and clearance procedures.	Q95
	2.2.33	Knowledge of control rod programming.	Q96

Category Total: 3

Generic Category	KA	KA Topic	Comment
Radiation Control	2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements.	Q35
	2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	Q29
	2.3.11	Ability to control radiation releases.	Q97

Category Total: 3

Generic Category	KA	KA Topic	Comment
Emergency Procedures/Plan	2.4.14	Knowledge of general guidelines for EOP flowchart use.	Q99
	2.4.29	Knowledge of the emergency plan.	Q98
	2.4.45	Ability to prioritize and interpret the significance of each annunciator or alarm.	Q67
	2.4.48	Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.	Q100

Category Total: 4

Facility: Diablo Canyon Power Plant

Generic Category	KA	KA Topic	Comment

Generic Total: 13

Tier / Group	Randomly Selected K/A	Reason for Rejection
RO 1/1	APE:055.2.1.29 (Q85)	Generic K/A was inappropriate for particular system or procedure.
RO 1/2; SRO 1/2	APE:E05.2.2.24 (Q63)	Generic K/A was inappropriate for particular system or procedure.
SRO 1/1	APE:051.2.4.33 (Q109)	Generic K/A was inappropriate for particular system or procedure.
SRO 1/1	APE:E12.2.2.29 (Q118)	Generic K/A was inappropriate for particular system or procedure.
SRO 1/2	APE:038:2.2.26 (Q105)	Generic K/A was inappropriate for particular system or procedure.
SRO 1/2	APE:E05.2.2.20 (Q116)	Generic K/A was inappropriate for particular system or procedure.
SRO 2/2	SYS:039.2.1.11 (Q106)	Generic K/A was inappropriate for particular system or procedure.
RO 2/1 ; SRO 2/1	SYS: 061.A3.04 (Q46)	Auxiliary feedwater does not automatically isolate at this plant.
RO 1/1 ; SRO 1/1	APE: 062.AK3.04 (Q47)	Nuclear Service Water does not exist at this plant. Even when Auxiliary Salt Water is substituted, the K/A is not applicable.
RO 2/2 ; SRO 2/2	SYS: 073.K4.02 (Q55)	Letdown does not isolate due to a process radiation monitor high radiation at this plant.
SRO 1/1	APE: E07.2.2.25 (Q117)	Saturated Core Cooling does not have knowledge of bases in T.S. for LCOs and safety limits at this plant.

Facility: DCPP Date of Examination: 10/21/2002
Examination Level (circle one): RO / SRO Operating Test Number: 1

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Mode Requirements	ADMNRC – 01, Perform Sealed Valve Checklist (JPM) RO/SRO
	Plant Parameters	ADMNRC – 12SRO, Verify AFD is within Tech Spec Limits (JPM)
		ADMNRC – 2RO, Perform QPTR (JPM)
A.2	Temporary Mods	ADMNRC – 3RO, Prepare Main Annunciator Problem Evaluation (JPM)
		ADMNRC – 3SRO, Review Main Annunciator Problem Evaluation (JPM)
A.3	Radiation Control	ADMNRC – 4, SCA Frisk (JPM) RO/SRO
A.4	Emergency Plan	Question RO: Responsibilities of Emergency Liaison Coordinator Question RO: Emergency Exposure Limits
		ADMNRC – 5SRO, Perform offsite Dose Assessment (JPM)

Facility: DCPP Date of Examination: 10/21/2002
Examination Level (circle one): RO / SRO Operating Test Number: 2

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Plant Parameters	ADMNRC – 6RO, Calculate SDM (JPM)
		ADMNRC – 6SRO, Verify SDM (JPM)
	Fuel Handling	ADMNRC – 7RO, Determine SFP Heat Load (JPM)
		ADMNRC – 7SRO, Verify SFP Heat Load (JPM)
A.2	Tagging	ADMNRC – 8RO, Perform Clearance Review (JPM)
	Maintenance	ADMNRC – 9SRO, Perform Risk Assessment (JPM)
A.3	Radiation Control	ADMNRC – 10, High Radiation Area Entry (JPM) RO/SRO
A.4	Emergency Plan	Question RO: Notification Times Question RO: OSC Activation and Location
		ADMNRC – 11SRO, Perform offsite Dose Assessment (JPM)

PART B EXAM, TEST 1

Facility: __DCPP__ Date of Examination: ____10/28/2002__
Exam Level (circle one): RO / SRO(I) / SRO(U) Operating Test No.: ___1___

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
TAB 1 004 – CVCS RO/SROI/SROU Makeup to RWST – **NRCLJC – 9**	D,S,L	I
TAB 2 074 – Inadequate Core Cooling RO/SROI Establish Feed from Condensate System – **NRCLJC – 12**	D,A,S,L	IVA
TAB 3 006 – ECCS RO/SROI Align RHR to Containment Spray – **NRCLJC – 3**	D,A,S,L	II
TAB 4 062 – AC Distribution RO/SROI Crosstie Vital Bus G to H – **NRCLJC – 4**	D,S,L	VI
TAB 5 068 – Control Room Evacuation RO/SROI/SROU Control Room Actions Prior to Evacuation – **NRCLJC – 5**	D,S	VIII
TAB 6 008 – CCW RO/SROI Respond to High Ultimate Heat Sink Temp – **NRCLJC – 6**	D,A,S	VIII
TAB 7 010 – PZR Pressure Control RO/SROI/SROU Initiate Auxiliary Spray – **NRCLJC – 14**	N,A,S,L	III

B.2 Facility Walk-Through

TAB 8 064 – Emergency Diesel Generators RO/SROI/SROU Local Start of a Diesel Generator – **NRCLJP – 15**	D,A	VI
TAB 9 040 – Steam Line Rupture RO/SROI/SROU Locally Close an MSIV – **NRCLJP – 16**	M,R,L	IVB
TAB 10 061 – Auxiliary Feedwater RO/SROI Align Alternate AFW from Fire Water – **NRCLJP – 21**	D,R,L	IVB

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA

PART B EXAM, TEST 2

Facility: __DCPP____________________ Date of Examination: ____10/28/2002__
Exam Level (circle one): RO / SRO(I) / SRO(U) Operating Test No.: ___2___

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
TAB 1 006 – ECCS RO/SROI Perform Actions for Trip with SI – **NRCLJC – 8**	M,A,S	III
TAB 2 004 – CVCS RO/SROI Establish Emergency Boration – **NRCLJC – 1**	D,A,S,L	I
TAB 3 022 – Containment Cooling RO/SROI Place CFCU Drain Collection In Service – **NRCLJC – 10**	N,S	V
TAB 4 002 – RCS RO/SROI/SROU Initiate Bleed and Feed for Loss of Heat Sink – **NRCLJC – 22**	D,A,S,L	IVA
TAB 5 015 – Nuclear Instrumentation RO/SROI/SROU Remove PR Channel 42 From Service – **NRCLJC – 23**	D,S	VII
TAB 6 074 – Inadequate Core Cooling RO/SROI/SROU Actions during FR-C.1 – **NRCLJC – 13**	N,A,S,L	IVA
TAB 7 064 – Emergency Diesel Generators RO/SROI Manual Start DG 12 from Control Room – **NRCLJC – 18**	D,S	VI

B.2 Facility Walk-Through

System / JPM Title	Type Code*	Safety Function
TAB 8 068 – Control Room Evacuation RO/SROI/SROU Align 480V Buses from HSP – **NRCLJP – 19**	D	VIII
TAB 9 061 – Auxiliary Feedwater RO/SROI Reset TDAFWP – **NRCLJP – 20**	D,R,L	IVB
TAB 10 068 – Liquid Radwaste RO/SROI/SROU Isolate Ruptured LHUT – **NRCLJP – 17**	D,R	IX

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA

Facility: DCPP Units 1 & 2 Scenario No.: 1 Op-Test No.: 1

Examiners: Operators:

Objectives: Evaluate the crew's ability to swap condensate booster pump sets
Evaluate the crew's ability to diagnose and respond to a VCT level control channel failure
Evaluate the crew's ability to diagnose and respond to a MFW pump controller problem
Evaluate the crew's ability to diagnose and respond to a Turbine Control failure in Auto
Evaluate the crew in using EOPs during an ATWS
Evaluate the crew in using the EOPs during a loss of 230kV event
Evaluate the crew's ability to diagnose and respond to a loss of TDAFW and MDAFW pumps
Evaluate the crew in using EOPs during an FRH condition

Initial Conditions: 100% power, equilibrium Xe, 1150 ppm, BOL (IC-1) MDAFW pump 1-2 OOS last 12 hours for bearing inspection, back in service in 8 hours. PRA OK.

Turnover: Start Standby Condensate Booster Pump set, place set 1-1 in standby.

Time min	Event No.	Malf. No.	Event Type*	Event Description
3	1		N, BOP	Swap Condensate Booster Pump sets
10	2		N,R, ALL	Commence power reduction to 70% (NO report Htr 2 DP oil leak)
20	3	mal tur4, 3	C, BOP	Turbine control failure requiring manual ramp
30	4	Xmt cvc19	I, RO	VCT Level channel 112 fail high
40	5	Ovr cc3049e Ovr cc3049h	C, ALL	MFW Pump master controller failure requiring manual control
On MFWP trip	6	mal ppl5	I, ALL	ATWS
Cond on 13D/E open	7	mal syd2	C, ALL	Loss of 230kV
Cond on 13D/E open	8	pmp afw2 mal afw1	M, ALL	Loss of All Feedwater (MDAFW and TDAFW Pump failure)

* (N)ormal (R)eactivity (I)nstrument (C)omponent (M)ajor

The Crew will swap condensate booster pump sets, referencing OP C7A:I

The Turbine Building NO will report an oil leak on Heater 2 Drip Pump, requiring a power reduction to 70% in preparation for tripping the pump. OP L-4 will be used for the power reduction, providing guidance on boration and setup of the turbine controls. A boration will commence and a controlled power reduction follows.

The Turbine controls will then shift to manual following a fault in the auto circuitry. This produces no alarms, but indications on the turbine control panel will indicate the change as well as the changes in plant parameters when the power reduction stops with boron injection underway. The crew will have to choose between stopping the ramp, or ramping manually to prepare for tripping the Heater 2 Drip Pump.

VCT Level channel 112 fails, giving a high VCT level alarm and diverting letdown to the hold up tanks. The crew should recognize the channel failure and respond per AP-19. Letdown should be restored to the VCT. The ramp may be stopped, but should be recommenced after the crew determines the failure does not impact the ramp.

The Master Feedwater Pump controller fails, requiring the crew to take manual control of both Main feedwater Pumps. The operator may not be able to analyze the problem and take corrective actions quick enough, which will then result in a Reactor Trip signal from low SG levels. If the operator does react and take control of the pumps manually, the crew will be forced to make a decision on continuing a manual ramp with manual feedwater, or trip the unit.

The unit will not trip on an auto trip signal or a manual trip initiation. The crew will be forced to use the RNO of E-0 and open breakers 13D and 13E. This will cause the rods to fall into the core. The crew will continue with E-0 actions.

Upon opening 13D/E, a loss of 230kV will occur. Plant response will lead to a Safety Injection during the implementation of E-0.

Upon opening 13D/E, the TDAFW pump and remaining MDAFW pump will trip and not restart. The crew should recognize a RED path on Heat Sink, and following transition from E-0 to E-1, enter FR-H.1. With the loss of 230kV, Condensate Booster pumps and MFW pumps are not available, leaving only Bleed and Feed as the method to cool the core. The scenario will end when Bleed and Feed is established.

Facility: DCPP Units 1 & 2 Scenario No.: 1 Op-Test No.: 2

Examiners: Operators:

Objectives: Evaluate the crew's ability to increase Accumulator Pressure
Evaluate the crew's ability to reduce power
Evaluate the crew's ability to diagnose and respond to failure in RMUW system
Evaluate the crew's ability to diagnose and respond to a PT 505 failure
Evaluate the crew's ability to diagnose and respond to a failed PZR spray valve controller
Evaluate the crew in using EOPs during a Steam Space LOCA
Evaluate the crew's ability to diagnose and respond to of failure of the SI signal

Initial Conditions: 100% power, equilibrium Xe, BOL 1150 ppm, (IC-1). MDAFW pump 1-2 OOS last 12 hours for bearing inspection, back in service in 8 hours. PRA good.

Turnover: Increase Accumulator 1-1 pressure per OP B-3B:I.

Time min	Event No.	Malf. No.	Event Type*	Event Description
3	1		N, RO	Increase Accumulator Pressure
10	2		R, All	Commence Power Decrease (EPOS request fast ramp to 850 MW)
On Boration	3	Ovr cc2010c	C, RO	43/MU fail to auto borate, manual boration required
20	4	xmt TUR2	I, BOP	PT 505 failure low
30	5	cnh pzr3	I, ALL	PRZ spray valve controller fails open in Auto
40	6	Mal pzr1	M, ALL	PZR steam space LOCA
On SI	7	ppl3a ppl3b	I, ALL	Failure of SI to actuate (manual alignment necessary)

* (N)ormal (R)eactivity (I)nstrument (C)omponent (M)ajor

Following a tailboard, the crew will increase pressure in Accumulator 1-1 to normal using OP B-3B:I.

After the Accumulator pressure increase, a call from EPOS will request a fast ramp to < 850 MW. The crew will tailboard the ramp and reactivity needs. A boration will start and a ramp commenced.

The boration will fail, the Makeup deviation alarm will alarm. 43/MU will not work in borate mode and must be used in the manual mode. The crew will use PK5-11 and AP-19 to determine the problem and use the alternate method to continue the ramp as requested.

After the crew commences manual boration and the ramp is started again, PT-505 will fail low, causing rods to drive in. The RO must recognize an instrument failure and take the rods to manual. Discussion on tripping bistables in 6 hour per ITS 3.3.1-1 should take place.

The PZR spray controller will fail in auto mode next, requiring the RO to take manual control of the spray valves to control pressure. The SFM will use PK5-17 and AP-13 to guide the crews response.

A PZR steam space LOCA takes place over 10 minutes to a final value of 850 gpm. This will require the crew to diagnose the pressure reduction with minimal PZR level change, and to quantify the leak.

After the leak size is sufficient, an SI will be required. The crew should SI before the low pressure setpoint, however an Over Power reactor trip may cause a reactor trip before the crew can respond. The SI signal will fail, requiring a manual SI signal initiation and manually aligning the valves and pumps for injection.

The scenario will terminate after transition to E-1.2 is completed.

Facility: DCPP Units 1 & 2 Scenario No.: 2 Op-Test No.: 1

Examiners: ____________ Operators: ____________

Objectives:
Evaluate the crew's ability to swap CCW heat exchangers
Evaluate the crew's ability to decrease reactor power
Evaluate the crew's ability to diagnose and respond to a Tc instrument drift
Evaluate the crew's ability to diagnose and respond to a loss of non-vital 120 VAC
Evaluate the crew's ability to diagnose and respond to an LDTV failure
Evaluate the crew in using EOPs during a Seismic event
Evaluate the crew's ability to diagnose and respond to a failure of Train A ECCS Equipment
Evaluate the crew in using EOPs during a Main Feedline Break
Evaluate the crew in using EOPs during a LBLOCA

Initial Conditions: 100% power, equilibrium Xe, 1150 ppm BOL (IC-1) DEG 1-1 OOS for fuel pump replacement. OOS 12 hours, expected return in 8 hours. PRA OK.

Turnover: Swap CCW Heat Exchangers.

Time min	Event No.	Malf. No.	Event Type*	Event Description
3	1		N, RO	Swap CCW heat exchangers
10	2		N, R, ALL	Power decrease to 80% (EPOS: Fire at Midway)
20	3	Xmt rcs138	I, RO	RCS Tc (TE-441) fail high
30	4	Mal eps2a	C, RO	Loss of non-vital 120 VAC (PY-15)
40	5	Xmt tur22	C, ALL	Turbine Governor Valve failure (FCV-142)
50	6	Mal sei1		Seismic event
Cond on seismic	7	Mal mfw5d	M, ALL	SG 4 MFL Break IC
0	8	Mal ppl3b (3)	C, ALL	Failure of Tr A ECCS
Seismic + 1 min	9	Mal rcs1(1)	M, ALL	RCS Loop 1 25% DBA

* (N)ormal (R)eactivity (I)nstrument (C)omponent (M)ajor

The crew will tailboard swapping the CCW heat exchanger for run time. This will entail swapping the running ASW train, and aligning the CCW heat exchanger per OP E-5:IV. The crew will start ASW pump 1-2, make alignments, and secure ASW pump 1-1 and make associated valve lineups.

EPOS will call requesting a decrease to 900 MW due to a fire at the Midway station. The crew will tailboard the ramp and reactivity change. A crew will borate and start a ramp per OP L-4.

During the ramp, Loop 4 Tc (TE-441) will fail high, causing rods to step in on a false high Tave. The crew should recognize the failed instrument and place rods in manual. The SFM stop the ramp, and reference OP AP-5 to ensure the plant is stable and for Tech Spec requirements on tripping bistables in 6 hours and deselecting that channel from Tave recording and control. Rods should be placed back to auto.

Before bistables are tripped, PY-15, Non-Vital 120 VAC will fail, causing many unrelated alarms. The crew should let rods control Tave to Tref because MSRs have been lost. The SFM will reference AP-4, and should request PY-15 be placed on backup power, which will restore the bus and clear the most of the alarms associated with the failure. The ramp should be reinstated if stopped.

Following the restoration of PY-15, the Turbine Governor valve, FCV-142, will fail causing a load rejection. The SFM will enter AP-25 and stabilize the plant. The Asset Team will be contacted for repair.

A Seismic event will take place, causing a Main Feed Line Break on SG 1-4 inside containment and a LBLOCA on Loop 1. The MSL Break will mask the LOCA initially. Train A SI will also fail to initiate and will require manual alignment of valves and pumps. The crew will isolate SG 1-4 using E-2, identify the LOCA and transition to E-1 where they will meet conditions to trip the RCPs. The scenario will continue until transition to E-1.3.

Facility: DCPP Units 1 & 2 Scenario No.: 2 Op-Test No.: 2

Examiners: Operators:

Objectives: Evaluate the crew's ability to increase reactor power
Evaluate the crew's ability to diagnose and respond to a PZR level channel failure low
Evaluate the crew's ability to restore letdown
Evaluate the crew's ability to respond to a SGTL
Evaluate the crew's ability to diagnose and respond to an SG pressure channel failure
Evaluate the crew's ability to diagnose and respond to a vacuum leak
Evaluate the crew's ability to diagnose and respond to a unit trip
Evaluate the crew in using EOPs during an Faulted/Ruptured SG
Evaluate the crew's ability to diagnose and respond to a failure of Phase A

Initial Conditions: 30% power, 235 ppm EOL. (IC-42) MDAFW pump 1-2 OOS last 12 hours for bearing inspection, back in service in 8 hours. PRA good.

Turnover: Increase power per OP L-4 to 100%.

Time min	Event No.	Malf. No.	Event Type*	Event Description
3	1		N, R, ALL	Increase power to 50%
10	2	xmt pzr40	I, RO	PZR level channel failure low
15	3		N, ALL	Restore Letdown
25	4	xmt mss58	C, BOP	SG 1 pressure channel PT- 516 fail hi (manually close PCV – 19)
35	5	Mal rcs4a	C, ALL	SGTL on SG 1-1 (approx. 5 gpm)
45	6	loa cnd1	C, ALL	Vacuum leak / power reduction
50	7	Mal sei1		Seismic Event (below Rx Trip Setpoint)
Cond on Seismic	8	Mal gen1	C, ALL	Main Generator lockout / unit trip
Cond on Seismic	9	Mal mss6a	M, ALL	SG 1-1 MSL fault
Manually Seismic + 5 min	10	Mal rcs4a	M, ALL	SGTR 1-1 (increase SGTL to 1215 gpm over 5 minutes)
0	11	Mal ppl1b	I, RO	Failure of Train B Phase A

* (N)ormal (R)eactivity (I)nstrument (C)omponent (M)ajor

The scenario starts at 30% during a startup. The crew will tailboard and commence a ramp to 50% per OP L-4 and dilute as necessary.

During the ramp, PZR Level Channel LT-459 will fail low, giving PZR level and Charging mismatch alarms. The RO will take manual control or charging and maintain seal injection and PZR level in band. The SFM will enter AP-5 and direct LT-459 be removed from input to control and determining per ITS 3.3.1 that bistables must be tripped in 6 hours.

Letdown will then be reestablished per OP B-1A:XII, allowing normal charging and letdown functions in automatic to resume.

SG 1-1 Pressure Channel PT-516 will fail high, causing the atmospheric, PCV-19 to open. There will be no alarms, and only the sound of steam and the indication of a PCV open light will indicate the problem. The BOPCO will have to take manual control of PCV-19 and close the valve. The SFM will respond per AP-5 and ITS 3.3.2 and determine bistables must be tripped in 6 hours.

A small SGTL will develop on SG 1-1 of approximately 5 gpm. The SJAE Rad alarm (PK11-06) will alarm. The BOPCO will also notice RM-15 counts increasing on the chart recorded on VB-1. The SFM will enter AP-3 and direct the RO to determine the leak rate. He will also determine ITS 3.4.13.d limits of 150 gpd has been exceeded and must start planning for a shutdown.

As the shutdown is planned, a small vacuum leak is initiated. Condensate DO2 and conductivity alarms (PK12-04/05) will alarm. The crew will notice vacuum slowly decreasing. The SFM will direct the RO to start a load decrease while entering AP-7. The BOPCO will be directing leak diagnostics outside the control room.

A seismic event will cause the turbine to trip on a lockout, but because the reactor is below P-9, the reactor will stay on line. The SFM must determine that this condition is acceptable and direct the crew to verify normal plant response.

A MSL Break occurs (SG 1-1 safety fails open) following the seismic event, causing a cooldown and SI to occur. The SFM will enter E-0 and E-2 and direct the BOPCO to isolate SG 1-1. The BOPCO will also determine that Phase A train B did NOT occur, and utilizing Attachmnent E, align Phase A manually.

Shortly after the MSL break, a SGTR will develop on SG 1-1. The level increase will be masked from the cooldown and rapid level increases from all AFW pumps running. No rad alarms will occur since these are power dependant on N-16. Once RCS pressure is determined to be too low and SG level response is diagnosed as a SGTR, the SFM will transition to E-3, and direct response from there. He will then transition to ECA-1.3.

The scenario terminates at the transition to ECA-1.3

Facility: DCPP Units 1 & 2 Scenario No.: 3 Op-Test No.: 2

Examiners: ______ Operators: ______

Objectives: Evaluate the crew's ability to diagnose and respond to a loss of Data A on DRPI
Evaluate the crew's ability to diagnose and respond to a Load Transient Bypass Valve failure
Evaluate the crew's ability to diagnose and respond to a RCP seal failure
Evaluate the crew's ability to shutdown the unit
Evaluate the crew's ability to diagnose and respond to a Loss of RWST
Evaluate the crew in using EOPs during a SBLOCA
Evaluate the crew's ability to diagnose and respond to a loss of Charging Pumps
Evaluate the crew in using EOPs during a loss of emergency coolant recirculation

Initial Conditions: 100% power, equilibrium xenon, EOL (IC-35). DEG 1-1 OOS for fuel pump replacement. OOS 12 hours, expected return in 8 hours. CSP 1-1 OOS 20 hours for scheduled motor work, expected return 20 hours. PRA OK.

Turnover: Maintain Power.

Time min	Event No.	Malf. No.	Event Type*	Event Description
3	1	Mal rod8a	I, RO	Loss of Data A on DRPI
10	2	Mal cnd1	C, ALL	LTB (FCV-230) fail open
Cond LTBV	3		R, ALL	Stabilize Power
20	4	Mal rcp2a	C, RO	RCP Seal 2 failure
25	5		R,N, ALL	Controlled Shutdown
30	6	Mal sei		Seismic event
Cond on sei	7	Mal rcp2a	C, RO	RCP Seal 1 failure
Cond on sei	8	Loa sis1	C, ALL	Loss of RWST
Cond on sei	9	pmp cvc1 pmp cvc2	C, ALL	Loss of CCP 1 and 2
Cond on sei	10	Mal rcs3	M, ALL	SBLOCA

* (N)ormal (R)eactivity (I)nstrument (C)omponent (M)ajor

A Data A failure on DRPI will occur, alarming PK03-21. The SFM will direct DRPI be selected to B train.

The LTB valve, FCV-230, will fail open increasing reactor power above 100% and alarming PK10-07. The crew will have to shed load to maintain power below 100%. Rods will step and boration will be required. OPdT runback may occur. The SFM will enter AP-25 and direct the control room in stabilizing the plant.

RCP 1-1 #2 Seal will fail, causing seal leakoff to #1 to decrease and #2 to increase. PK05-01 will alarm and the SFM will direct the RO/BOPCO to start investigating, including Aux Board RCDT trends while monitoring temperature trends and RCP vibration. The crew should prepare for an orderly shutdown.

A seismic event will cause an RCP 1 seal 1 leak at 10 gpm requiring a pump trip and closure of the seal leakoff valve, a loss of both CCPs, a SBLOCA of 3000 gpm, and a Loss of RWST. No water will be available for injection. The crew will proceed through E-0, E-1 and transition to ECA-1.1 when Cold Leg Recirculation capability cannot be confirmed. The crew will be challenged to NOT trip the RCPs with no SI pumps available and no subcooling. They will proceed until cooldown is established with dumping steam and a 100°F/hr cooldown rate is established.