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<tr><td colspan="9">Facility: SONGS 2 & 3 NRC</td><td colspan="10">Date of Exam: 10/31/11</td></tr>
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<tr><td rowspan="3">1. Emergency & Abnormal Plant Evolutions</td><td>1</td><td>2</td><td>2</td><td>3</td><td></td><td></td><td></td><td>3</td><td>4</td><td></td><td></td><td>4</td><td>18</td><td colspan="2">3</td><td colspan="2">3</td><td>6</td></tr>
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<tr><td>Tier Totals</td><td>4</td><td>4</td><td>3</td><td></td><td></td><td></td><td>4</td><td>7</td><td></td><td></td><td>5</td><td>27</td><td colspan="2">5</td><td colspan="2">5</td><td>10</td></tr>
<tr><td rowspan="3">2. Plant Systems</td><td>1</td><td>3</td><td>2</td><td>2</td><td>4</td><td>2</td><td>3</td><td>1</td><td>3</td><td>3</td><td>3</td><td>2</td><td>28</td><td colspan="2">2</td><td colspan="2">3</td><td>5</td></tr>
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<tr><td colspan="5" rowspan="2">3. Generic Knowledge and Abilities Categories</td><td colspan="2">1</td><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">4</td><td rowspan="2">10</td><td>1</td><td>2</td><td>3</td><td>4</td><td rowspan="2">7</td></tr>
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Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the “Tier Totals” in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems / evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems / evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics’ importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

SONGS 2 & 3
NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

E/APE # / Name Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
E05 / Excess Steam Demand / 4					X		EA2.1	Ability to determine and interpret the following as they apply to the Excess Steam Demand: Facility conditions and selection of appropriate procedures during abnormal and emergency conditions	4.0	76
011 / Large Break LOCA / 3						X	2.2.37	Equipment Control: Ability to determine operability and/or availability of safety related equipment	4.6	77
015/017 / RCP Malfunctions / 4					X		AA2.01	Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Cause of RCP failure	3.5	78
027 / Pressurizer Pressure Control Malfunction / 3						X	2.4.50	Emergency Procedures/Plan: Ability to verify system alarms setpoints and operate controls identified in the alarm response manual	4.0	79
029 / ATWS / 1					X		EA2.01	Ability to determine and interpret the following as they apply to the ATWS: Reactor nuclear instrumentation	4.7	80
058 / Loss of DC Power / 6						X	2.4.4	Emergency Procedures/Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures	4.7	81
007 / Reactor Trip - Stabilization - Recovery / 1						X	2.1.7	Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	4.4	39
009 / Small Break LOCA / 3		X					EK2.03	Knowledge of the interrelations between small break LOCA and the following: SGs	3.0	40
015 / 17 / RCP Malfunctions / 4				X			AA1.03	Ability to operate and/or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Reactor trip alarms, switches, and indicators	3.7	41
022 / Loss of Reactor Coolant Makeup / 2			X				AK3.06	Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: RCP thermal barrier cooling	3.2	42

SONGS 2 & 3
NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

E/APE # / Name Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
025 / Loss of RHR System / 4					X		AA2.02	Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Leakage of reactor coolant from RHR into closed cooling water system or into reactor building atmosphere	3.4	43
026 / Loss of Component Cooling Water / 8					X		AA2.06	Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The length of time after the loss of CCW flow to a component before that component may be damaged	2.8	44
027 / Pressurizer Pressure Control System Malfunction / 3				X			AA1.03	Ability to operate and/or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressure control when on a steam bubble	3.6	45
029 / ATWS / 1		X					EK2.06	Knowledge of the interrelations between ATWS and the following: Breakers, relays, and disconnects	2.9	46
038 / Steam Generator Tube Rupture / 3					X		EA2.01	Ability to determine or interpret the following as they apply to a SGTR: When to isolate one or more SGs	4.1	47
040 / Steam Line Rupture - Excessive Heat Transfer / 4	X						AK1.01	Knowledge of the operational implications of the following concepts as they apply to the Steam Line Rupture: Consequences of PTS	4.1	48
011 / Large Break LOCA / 3						X	2.4.49	Emergency Procedures/Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls	4.6	49
055 / Station Blackout / 6	X						EK1.02	Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling	4.1	50
056 / Loss of Offsite Power / 6			X				AK3.02	Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Actions contained in EOP for loss of offsite power	4.4	51
057 / Loss of Vital AC Instrument Bus / 6					X		AA2.12	Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: PZR level controller, instrumentation, and heater indications	3.5	52

SONGS 2 & 3
NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

E/APE # / Name Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
058 / Loss of DC Power / 6			X				AK3.02	Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of DC power	4.0	53
062 / Loss of Nuclear Service Water / 4						X	2.4.35	Emergency Procedures/Plan: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects	3.8	54
065 / Loss of Instrument Air / 8				X			AA1.04	Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air: Emergency air compressor	3.5	55
077 / Generator Voltage and Electric Grid Disturbances / 6						X	2.1.23	Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation	4.3	56
K/A Category Point Totals:	2	2	3	3	4 / 3	4 / 3	Group Point Total:			**18 / 6**

SONGS 2 & 3
NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 2

E/APE # / Name Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
032 / Loss of Source Range NI / 7					X		AA2.09	Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Effect of improper HV setting	2.9	82
051 / Loss of Condenser Vacuum / 4						X	2.4.18	Emergency Procedures/Plan: Knowledge of specific bases for EOPs	4.0	83
061 / Area Radiation Monitoring System Alarms / 7					X		AA2.06	Ability to determine and interpret the following as they apply to the Area Radiation Monitoring System Alarms: Required actions if the alarm channel is out of service	4.1	84
CE / A16 / Excess RCS Leakage / 2						X	2.4.21	Emergency Procedures/Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.6	85
001 / Continuous Rod Withdrawal / 1				X			AA1.02	Ability to operate and/or monitor the following as they apply to the Continuous Rod Withdrawal: Rod in-hold-out switch	3.6	57
003 / Dropped Control Rod / 1					X		AA2.01	Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod position indication to actual rod position	3.7	58
024 / Emergency Boration / 1	X						AK1.02	Knowledge of the operational implications of the following concepts as they apply to Emergency Boration: Relationship between boron addition and reactor power	3.6	59
060 / Accidental Gaseous Radwaste Release / 9					X		AA2.06	Ability to determine and interpret the following as they apply to the Accidental Gaseous Release: Valve lineup for release of radioactive gases	3.6	60
061 / Area Radiation Monitoring System Alarms / 7						X	2.4.6	Emergency Procedures/Plan: Knowledge of EOP mitigation strategies	3.7	61
067 / Plant Fire on Site / 8	X						AK1.02	Knowledge of the operational implications of the following concepts as they apply to Plant Fire on Site: Fire fighting	3.1	62
069 / Loss of Containment Integrity / 5		X					AK2.03	Knowledge of the interrelations between the Loss of Containment Integrity and the following: Personnel access hatch and emergency access hatch	2.8	63

SONGS 2 & 3
NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 2

E/APE # / Name Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
076 / High Reactor Coolant Activity / 9					X		AA2.01	Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Location or process point that is causing an alarm	2.7	64
CE / A11 / RCS Overcooling - PTS / 4		X					AK2.2	Knowledge of the interrelations between RCS Overcooling and the following: 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	3.2	65
K/A Category Point Totals:	2	2	0	1	3 / 2	1 / 2	Group Point Total:			**9 / 4**

SONGS 2 & 3
NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
006 / Emergency Core Cooling											X	2.2.37	Equipment Control: Ability to determine operability and / or availability of safety-related equipment	4.6	86
062 / AC Electrical Distribution											X	2.4.21	Emergency Procedures/Plan: knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.6	87
013 / Engineered Safety Features Actuation								X				A2.06	Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent ESFAS actuation	4.0	88
061 / Auxiliary/Emergency Feedwater								X				A2.07	Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air or MOV failure	3.5	89
012 / Reactor Protection											X	2.2.22	Emergency Procedures/Plan: Knowledge of limiting conditions for operations and safety limits	4.7	90
003 / Reactor Coolant Pump								X				A2 .03	Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems	2.7	1
003 / Reactor Coolant Pump					X							K5.03	Knowledge of the operational implications of the following concepts as they apply to the RCPs: Effect of RCP shutdown on Tave, including the reason for unreliability of Tave in the shutdown loop	3.1	2

SONGS 2 & 3
NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
004 / Chemical and Volume Control		X										K2.06	Knowledge of bus power supplies to the following: Control instrumentation	2.6	3
004 / Chemical and Volume Control				X								K4.12	Knowledge of the CVCS design feature(s) and/or interlock(s) which provide for the following: Minimum level of VCT	3.1	4
005 / Residual Heat Removal								X				A2.02	Ability to (a) predict the impacts of the following malfunctions or operations on the RHR system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown	3.5	5
005 / Residual Heat Removal	X											K1.01	Knowledge of the physical connections and/or cause-effect relationships between the RHRS and the following systems: CCWS	3.2	6
006 / Emergency Core Cooling						X						K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the ECCS: BIT / borated water sources	3.4	7
007 / Pressurizer Relief / Quench Tank											X	2.4.11	Emergency Procedures/Plan: Knowledge of abnormal condition procedures	4.0	8
008 / Component Cooling Water				X								K4.07	Knowledge of the CCWS design feature(s) and/or interlock(s) that provide for the following: Operation of the CCW swing-bus power supply and its associated breakers and controls	2.6	9
010 / Pressurizer Pressure Control						X						K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems	2.7	10
012 / Reactor Protection						X						K6.09	Knowledge of the effect that a loss or malfunction of the following will have on the RPS: CEAC	3.6	11
012 / Reactor Protection			X									K3.01	Knowledge of the effect that a loss or malfunction of the RPS will have on the following: CRDS	3.9	12
013 / Engineered Safety Features Actuation											X	2.1.28	Conduct of Operations: Knowledge of the purpose and function of major system components and controls	4.1	13

SONGS 2 & 3
NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
013 / Engineered Safety Features Actuation										X		A4.02	Ability to manually operate and/or monitor in the control room: Reset of ESFAS channels	4.3	14
022 / Containment Cooling									X			A3.01	Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation	4.1	15
026 / Containment Spray	X											K1.02	Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: Cooling water	4.1	16
026 / Containment Spray										X		A4.01	Ability to manually operate and/or monitor in the control room: CSS controls	4.5	17
039 / Main and Reheat Steam									X			A3.02	Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS	3.1	18
059 / Main Feedwater	X											K1.03	Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: SGs	3.1	19
061 / Auxiliary/Emergency Feedwater					X							K5.03	Knowledge of the operational implications of the following concepts as they apply to the AFW: Pump head effects when control valve is shut	2.6	20
062 / AC Electrical Distribution			X									K3.02	Knowledge of the effect that a loss or malfunction of the AC distribution system will have on the following: EDG	4.1	21
063 / DC Electrical Distribution									X			A3.01	Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights	2.7	22
064 / Emergency Diesel Generator							X					A1.08	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the EDG system controls including: Maintaining minimum load on EDG (to prevent reverse power)	3.1	23
064 / Emergency Diesel Generator		X										K2.01	Knowledge of bus power supplies to the following: Air compressor	2.7	24

SONGS 2 & 3
NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
073 / Process Radiation Monitoring				X								K4.01	Knowledge of the PRM system design feature(s) and/or interlock(s) that provide for the following: Release termination when radiation exceeds setpoint	4.0	25
076 / Service Water										X		A4.04	Ability to manually operate and/or monitor in the control room: Emergency heat loads	3.5	26
078 / Instrument Air				X								K4.03	Knowledge of the IAS system design feature(s) and/or interlock(s) that provide for the following: Securing of SAS upon loss of cooling water	3.1	27
103 / Containment								X				A2.05	Ability to (a) predict the impacts of the following malfunctions or operations on the Containment system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency containment entry	2.9	28
K/A Category Point Totals:	3	2	2	4	2	3	1	3 / 2	3	3	2 / 3	Group Point Total:			**28 / 5**

SONGS 2 & 3
NRC Written Examination Outline
Plant Systems – Tier 2 Group 2

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
015 / Nuclear Instrumentation								X				A2.03	Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Xenon oscillations	3.5	91
045 / Main Turbine Generator								X				A2.08	Ability to (a) predict the impacts of the following malfunctions or operations on the MTG System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam dumps are not cycling properly at low load, or stick open at higher load (isolate and use atmospheric reliefs when necessary)	3.1	92
035 / Steam Generator											X	2.4.47	Emergency Procedures/Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material	4.2	93
002 / Reactor Coolant									X			A3.03	Ability to monitor automatic operation of the RCS, including: Pressure, temperature, and flows	4.4	29
011 / Pressurizer Level Control			X									K3.01	Knowledge of the effect that a loss or malfunction of the PZR LCS will have on the following: CVCS	3.2	30
015 / Nuclear Instrumentation					X							K5.17	Knowledge of the operational implications of the following concepts as they apply to the NIS: DNB and DNBR definition and effects	3.5	31
016 / Non-Nuclear Instrumentation	X											K1.12	Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following systems: S/G	3.5	32
033 / Spent Fuel Pool Cooling							X					A1.01	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Spent Fuel Pool Cooling System controls including: Spent fuel pool water level	2.7	33

SONGS 2 & 3
NRC Written Examination Outline
Plant Systems – Tier 2 Group 2

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
034 / Fuel Handling Equipment				X								K4.02	Knowledge of design feature(s) and/or interlock(s) that provide for the following: Fuel movement	2.6	34
056 / Condensate								X				A2.04	Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of condensate pumps	2.6	35
068 / Liquid Radwaste						X						K6.10	Knowledge of the effect that a loss or malfunction of the following will have on the Liquid Radwaste System: Radiation monitors	2.5	36
071 / Waste Gas Disposal											X	2.1.27	Conduct of Operations: Knowledge of system purpose and/or function	3.9	37
072 / Area Radiation Monitoring											X	2.4.31	Emergency Procedures/Plan: Knowledge of annunciator alarms, indications, or response procedures	4.2	38
K/A Category Point Totals:	1	0	1	1	1	1	1	1 / 2	1	0	2 / 1	Group Point Total:			**10 / 3**

Facility: SONGS 2 & 3 NRC Date of Exam: 10/31/11

Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.5	Ability to use procedures related to shift staffing, such as minimum crew complement, over time limitations, etc.			3.9	94
	2.1.34	Knowledge of primary and secondary plant chemistry limits			3.5	95
	2.1.19	Ability to use plant computers to evaluate system or component status	3.9	66		
	2.1.32	Ability to explain and apply system limits and precautions	3.8	67		
	Subtotal			2		2
2. Equipment Control	2.2.17	Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator			3.8	96
	2.2.40	Ability to apply Technical Specifications for a system			4.7	97
	2.2.12	Knowledge of surveillance procedures	3.7	68		
	2.2.14	Knowledge of the process for controlling equipment configuration or status	3.9	69		
	2.2.35	Ability to determine Technical Specification Mode of Operation	3.6	70		
	Subtotal			3		2
3. Radiation Control	2.3.11	Ability to control radiation releases			4.3	98
	2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions	3.5	71		
	2.3.11	Ability to control radiation releases	3.8	72		
	Subtotal			2		1
4. Emergency Procedures / Plan	2.4.27	Knowledge of "fire in the plant" procedures			3.9	99
	2.4.45	Ability to prioritize and interpret the significance of each annunciator or alarm			4.3	100
	2.4.2	Knowledge of systems setpoints, interlocks and automatic actions associated with EOP entry conditions	4.5	73		
	2.4.26	Knowledge of facility protection requirements, including fire brigade and portable firefighting equipment usage	3.1	74		
	2.4.32	Knowledge of operator response to loss of all annunciators	3.6	75		
	Subtotal			3		2
Tier 3 Point Total				**10**		**7**

Tier / Group	Randomly Selected K/A	Reason for Rejection
1 / 1	015/017 AA2.11	Q #78 – Jogging of RCPs is not performed at SONGS during Inadequate Core Cooling. Randomly reselected 015/017 AA2.01.
2 / 2	016 K1.07	Q #32 – There is no interface between the Non-Nuclear Instrumentation and Emergency Core Cooling Systems at SONGS. Randomly reselected 016 K1.12.
2 / 2	033 A1.02	Q #33 – There is no interface between the Spent Fuel Pool Cooling System and Radiation Monitoring System at SONGS. Randomly reselected 033 A1.01.
2 / 2	034 A4.01	Q #34 – There is no interface between the Fuel Handling Equipment System and Radiation Monitoring System (radiation levels) at SONGS. Reselected 034 K4.02.
2 / 2	071 G 2.4.4	Q #37 – There are no Abnormal or Emergency Operating Procedures associated with the Waste Gas Disposal System. Reselected 071 G 2.1.27.
1 / 2	024 G 2.2.22	Q #59 – Unable to develop a psychometrically sound question that discriminates at the RO license level. Reselected 024 AK1.02.
2 / 1	063 A1.01	Q #22 – This K/A was determined operationally invalid during exam validation. Question was left on Worksheet for CE review. Reselected 063 A3.01.
1 / 2	060 AK3.03	Q #60 – Unable to develop a psychometrically sound question that discriminates at the RO license level. Reselected 060 AA2.06.
2 / 1	006 K5.11	Q #07 – This K/A identified as generic fundamentals. Reselected 006 K6.01.
2 / 1	013 A4.01	Q #14 – This K/A adequately covered during Scenario events. Randomly reselected 013 A4.02.
1 / 1	057 AA2.11	Q #52 – Unable to develop a psychometrically sound question that discriminates at the RO license level. Randomly reselected 057 AA2.12.
1 / 1	038 EA1.08	Q #47 – Unable to develop a psychometrically sound question that discriminates at the RO license level. Reselected 038 EA2.01.
1 / 1	061 G 2.1.25	Q #54 – This K/A is addressed by Admin JPM RA1. Reselected 061 G 2.4.35.
1 / 2	003 AK1.04	Q #58 – This K/A identified as generic fundamentals. Reselected 003 AA2.01.
1 / 1	040 AK1.04	Q #48 – Unable to develop a psychometrically sound question that discriminates at the RO license level. Randomly reselected 040 AK1.01.
1 / 1	007 G 2.2.25	Q #39 – Unable to develop a psychometrically sound question that discriminates at the RO license level. Randomly reselected 007 G 2.1.7.
1 / 1	054 G 2.2.36	Q #49 – Replaced K/A to test concept not addressed during Simulator Scenario development (Safety Injection Throttle / Stop). Reselected 011 G 2.4.49.
1 / 1	027 G 2.4.3	Q #79 – Unable to develop a psychometrically sound question that discriminates at the SRO license level. Reselected 027 G 2.4.50.

Tier / Group	Randomly Selected K/A	Reason for Rejection
1 / 1	057 AA2.05	Q #81 – Unable to develop a psychometrically sound question that discriminates at the SRO license level. Reselected Abnormal Operating Instruction not addressed during Simulator Scenario development (Loss of DC Power) 058 G 2.4.4.
2 / 1	010 G 2.2.37	Q #86 – Pressurizer Pressure Control System adequately addressed by Simulator Scenarios. Reselected 006 G 2.2.37.
2 / 1	062 G 2.1.30	Q #87 – Unable to develop a psychometrically sound question that discriminates at the SRO license level. Reselected 062 G 2.4.21.
2 / 1	013 A2.04	Q #88 – Unable to develop a psychometrically sound question that discriminates at the SRO license level. Reselected 013 A2.06.
2 / 1	063 G 2.4.8	Q #90 – Unable to develop a psychometrically sound question that discriminates at the SRO license level. Reselected 012 G 2.2.22.
2 / 2	068 G 2.4.49	Q #91 – Unable to develop a psychometrically sound question that discriminates at the SRO license level. Reselected 015 A2.03.
2 / 2	072 A2.02	Q #92 – Unable to develop a psychometrically sound question that discriminates at the SRO license level. Reselected 045 A2.08.
1 / 1	015/17 AK3.07	Q #41 – K/A replaced as it provided information required for Q#51. Unable to develop a psychometrically sound question that discriminates at the RO license level for K/A AK3.07. Reselected 015/17 AA1.03.
2 / 1	003 K2.02	Q #02 – Unable to develop a psychometrically sound question that discriminates at the RO license level. Power supply concepts are adequately addressed in JPM S-4. Reselected 003 K5.03.
3 / 3	G 2.3.6	Q #98 – Multiple questions about releases already appear on exam (Q #36 & #72). Reselected G 2.3.11.
2 / 2	072 K4.02	Q #38 – Changed K/A to better address question. There is no interlock associated with RE-7850 in the Fuel Handling Building other than an alarm. Reselected 072 G 2.4.31.
2 / 1	005 K1.08	Q #06 – Changed K/A to better address question because there is no direct interface between the Shutdown Cooling System and Saltwater Cooling System at SONGS. Reselected 005 K1.01.
1 / 1	009 EA2.14	Q #76 – Changed K/A to better address question. Reselected E05 EA2.1.

Facility: SONGS Units 2 and 3		Date of Examination:	10/31/11
Examination Level	RO ☐	Operating Test Number:	NRC

Administrative Topic (see Note)	Type Code*	Describe Activity to be Performed
Conduct of Operations (RA1)	M, R	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (4.3) JPM: Manually Calculate Saltwater Cooling Flow (J295A).
Conduct of Operations (RA2)	M, R	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. (3.9) JPM: Determine Time until Shutdown Cooling Required (J053A2).
Equipment Control (RA3)	M, R	2.2.12 Knowledge of surveillance procedures. (3.7) JPM: Perform Reactor Coolant System Flow Rate Determination (J162A2).
Radiation Control	-	-
Emergency Plan (RA4)	D, S	2.4.39 Knowledge of RO responsibilities in emergency plan implementation. (3.9) JPM: Activate Emergency Response Data System (J158A2).

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.

*Type Codes & Criteria:

(C)ontrol room, (S)imulator, or Class(R)oom

(D)irect from bank (≤ 3 for ROs; ≤ for 4 for SROs & RO retakes)

(N)ew or (M)odified from bank (≥ 1)

(P)revious 2 exams (≤ 1; randomly selected)

RA1 The applicant will manually calculate Unit 3 Saltwater Cooling (SWC) Pump 3P-112 flow to support OPERABILITY evaluation per SO23-2-8, Saltwater Cooling System Operation, Attachment 6, SWC Flow Calculation. The critical steps include calculating differential pressures and flow based on provided readings. This is a modified bank JPM.

RA2 The applicant will complete SO23-12-11, EOI Supporting Attachments, Attachment 16, Determine Time Until Shutdown Cooling Required for Unit 3. The critical steps include determining time remaining until Shutdown Cooling is required for decay heat removal and the minimum cooldown rate required to establish Shutdown Cooling entry conditions before feedwater source inventory is depleted. This is a modified bank JPM.

RA3 The applicant will perform SO23-3-3.3, RCS Flow Rate Determination, Attachment 2, RCP ΔP Flow Calculation on Unit 2 without access to the Plant Computer System. The critical steps include determining if Acceptance Criteria is met for Reactor Coolant System flow. This is a modified bank JPM.

RA4 The applicant will activate the Emergency Response Data System (ERDS) per SO23-VIII-30, Units 2/3 Operations Leader Duties and SO23-3-2.32, Critical Functions Monitoring System. The critical steps include accessing the Plant Computer System and demonstrating both methods for connecting to the Nuclear Regulatory Commission during an emergency. This is a bank JPM.

Facility:	SONGS Units 2 and 3	Date of Examination:	10/31/11
Examination Level	SRO ☐	Operating Test Number:	NRC

Administrative Topic (see Note)	Type Code*	Describe Activity to be Performed
Conduct of Operations (SA1)	M, R	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (4.4) JPM: Manually Calculate Saltwater Cooling Flow and Determine OPERABILITY (J295A).
Conduct of Operations (SA2)	M, R	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. (4.2) JPM: Determine Time until Shutdown Cooling Required and Event Reportability (J053A2).
Equipment Control (SA3)	M, R	2.2.12 Knowledge of surveillance procedures. (4.1) JPM: Perform Reactor Coolant System Flow Rate Determination and Evaluate Technical Specifications (J162A2).
Radiation Control (SA4)	N, R	2.3.11 Ability to control radiation releases (4.3). JPM: Calculate Dispersion Factor for Gaseous Release (New).
Emergency Plan (SA5)	N, R	2.4.41 Knowledge of the emergency action level thresholds and classifications (4.6). JPM: Classify an Emergency Plan Event (New).

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.

*Type Codes & Criteria:
(C)ontrol room, (S)imulator, or Class(R)oom
(D)irect from bank (≤ 3 for ROs; ≤ for 4 for SROs & RO retakes)
(N)ew or (M)odified from bank (≥ 1)
(P)revious 2 exams (≤ 1; randomly selected)

SA1 The applicant will manually calculate Unit 3 Saltwater Cooling (SWC) Pump 3P-112 flow to support OPERABILITY evaluation per SO23-2-8, Saltwater Cooling System Operation, Attachment 6, SWC Flow Calculation. The critical steps include calculating differential pressures and flow based on provided readings, and determining OPERABILITY of the Saltwater Cooling System per SO23-2-8, Saltwater Cooling System Operation, Attachment 4, Saltwater Injection Temperature vs. Minimum Saltwater Flow. This is a modified bank JPM.

SA2 The applicant will complete SO23-12-11, EOI Supporting Attachments, Attachment 16, Determine Time Until Shutdown Cooling Required for Unit 3. The critical steps include determining time remaining until Shutdown Cooling is required for decay heat removal and the minimum cooldown rate required to establish Shutdown Cooling entry conditions before feedwater source inventory is depleted. The applicant will then assess notification requirements for the event in progress per SO123-0-A7, Notification and Reporting of Significant Events. This is a modified bank JPM.

SA3 The applicant will perform SO23-3-3.3, RCS Flow Rate Determination, Attachment 2, RCP ΔP Flow Calculation on Unit 3 without access to the Plant Computer System. The critical steps include determining if Acceptance Criteria is met for Reactor Coolant System flow. When calculations are complete, the applicant will evaluate surveillance results to determine if Technical Specification Limiting Conditions for Operations have been met. This is a modified bank JPM.

SA4 The applicant will calculate the dispersion factor for a Gaseous Release and determine if release is desirable per SO23-8-15, Radwaste Gas Discharge, Attachment 4, Determination of Current Weather Conditions. The critical steps include calculating a chi over Q (χ/Q) and evaluating current weather conditions prior to approving the release. This is a new JPM.

SA5 The applicant will determine the Recognition Category, Emergency Class, and Emergency Action Level per SO123-VIII-1, Recognition and Classification of Emergencies. The critical steps include determining the Recognition Category, Emergency Class, and Emergency Action Level using the newly formatted Hot and Cold Emergency Action Level Classification Charts. This is a new JPM.

Facility:	SONGS Units 2 and 3	Date of Examination:	10/31/11
Exam Level:	RO ☐ SRO(I) ☐ **SRO (U)** ☐	Operating Test No.:	NRC

Control Room Systems@ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U, including 1 ESF)			
	System / JPM Title	Type Code*	Safety Function
S-1	004 – Chemical and Volume Control System (New) Perform an Emergency Boration Restoration	N, S	1
S-2	**013 – Engineered Safety Feature Actuation System (J268)** **Reset a Control Room Isolation Signal**	**EN, M, S**	**2**
S-3	006 – Emergency Core Cooling System (J086S2) Pressurize a Safety Injection Tank (RO Only)	D, S	3
S-4	**003 – Reactor Coolant Pump System (J027FS)** **Start a Reactor Coolant Pump**	**A, L, M, S**	**4-P**
S-5	059 – Main Feedwater System (J221FS) Reset Valid Reactor Trip Override	A, M, S	4-S
S-6	**022 – Containment Cooling System (J260FS)** **Place Containment Emergency Cooling in Service**	**M, S**	**5**
S-7	062 – AC Electrical Distribution System (J266FS) Perform a Drop and Pick up Transfer of Bus 2A06	A, D, S	6
S-8	073 – Process Radiation Monitoring System (J302S) Bypass Containment Purge Isolation Radiation Monitor	D, S	7
In-Plant Systems@ (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)			
P-1	**015 – Nuclear Instrumentation System (J003)** **Place the EPPM in Service**	**A, E, M, R**	**7**
P-2	002 – Reactor Coolant System (J253) Align Remote Shutdown Panel During Security Event	D, E	2
P-3	**063 – DC Electrical Distribution System (New)** **Place the Swing Battery Charger in Operation**	**E, N**	**6**

@	All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path	4-6 / 4-6 / 2-3
(C)ontrol room	
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / ≥ 1
(EN)gineered safety feature	- / - / ≥ 1 (control room system)
(L)ow Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

NRC JPM Examination
Summary Description

S-1 The applicant will perform an Emergency Boration restoration per SO23-13-11, Emergency Boration of the RCS / Inadvertent Dilution or Boration, Step 4, Emergency Boration Restoration. This is a new JPM under the Chemical and Volume Control System - Reactivity Control Safety Function. (K/A 004.A2.14 - IR 3.8 / 3.9)

S-2 The applicant will perform a reset of Control Room Isolation Signal (CRIS) per SO23-3-2.22, Engineered Safety Features Actuation System Operation, Attachment 18, CRIS/TGIS Reset and Restoration. This is a modified bank JPM under the Engineered Safety Features Actuation System – Reactor Coolant System Inventory Control Safety Function. (K/A 013.A4.02 - IR 4.3 / 4.4)

S-3 The applicant will pressurize a Safety Injection Tank per SO23-3-2.7.1, Safety Injection Tank Operation. This is a bank JPM under the Emergency Core Cooling System - Reactor Pressure Control Safety Function. (K/A 006.A1.13 - IR 3.5 / 3.7)

S-4 The applicant will start the fourth Reactor Coolant Pump (RCP) during a Plant Startup per SO23-3-1.7, Reactor Coolant Pump Operation. The alternate path occurs when a thrust bearing high temperature alarm is received. Annunciator 56C05 – RCP P003 THRUST BRG TEMP HI, requires an RCP trip which fails at CR-56 but is successful via the 6900 Volt Electrical Distribution System at CR-63. This is a modified bank JPM under the Reactor Coolant Pump System - Heat Removal from Reactor Core Safety Function. (K/A 003.A4.06 - IR 2.9 / 2.9)

S-5 The applicant will reset a valid Reactor Trip Override (RTO) per SO23-9-6, Feedwater Control System Operation, Section 6.5, Feedwater Control System Operation During a Valid RTO. The alternate path occurs when Steam Generator level starts to rise due to a

setpoint failure after the RTO is reset. This is a bank JPM under the Main Feedwater System – Secondary System Heat Removal from Reactor Core Safety Function. This is a PRA significant action. (K/A 059.A2.11 - IR 3.0 / 3.3)

S-6 The applicant will place the Containment Emergency Cooling System in service following trip of the Containment Normal Chillers per SO23-1-4.1, Containment Emergency Cooling, Section 6.1, Placing the Containment Emergency Cooling System in Service. When the Containment Cooling Actuation Signal fails to actuate, Containment Emergency Cooling is then placed in service per Section 6.5, Placing Containment Emergency Cooling System in Service on a Component Basis. This is a modified bank JPM under the Containment Cooling System - Containment Integrity Safety Function. (K/A 022.A4.01 - IR 3.6 / 3.6)

S-7 The applicant will perform a Drop and Pick up Transfer of Bus 2A06 per SO23-6-2, Transferring 4 kV Buses, Section 6.11, Drop and Pick up Transfer of 1E 4 KV Buses. On failure of the opposite unit breaker to transfer, the Train B Emergency Diesel Generator (EDG) is selected to repower the bus. The alternate path occurs when the EDG Breaker fails to close upon Bus transfer due to a low-voltage interlock not being met. This is a bank JPM under the AC Electrical Distribution System - Electrical Safety Function. (K/A 062.A4.07 - IR 3.1 / 3.1)

S-8 The applicant will bypass the Containment Purge Isolation System function for a Containment Purge Isolation Radiation Monitor to allow Chemistry to change filters per SO23-3-2.24.11, Containment Radiation Monitor System Operation and SO23-3-2.36, Radiation Monitor Data Acquisition System. This is a bank JPM under the Process Radiation Monitoring System - Instrumentation Safety Function. (K/A 073.A4.02 - IR 3.7 / 3.7)

P-1 The applicant will place the Essential Plant Parameters Monitoring (EPPM) Panel in service following a Control Room Evacuation per SO23-13-2, Shutdown From Outside the Control Room, Attachment 4 (5), 21 (31) Duties. The alternate path occurs when local control is taken and the instrument indications start to oscillate. This is a modified bank JPM under the Nuclear Instrumentation System - Instrumentation Safety Function. This is a PRA significant action. (K/A 068.AA1.12 - IR 4.4 / 4.4)

P-2 The applicant will align the Remote Shutdown Panel during a Security Event per SO23-13-25, Operator Actions During Security Events, Attachment 3, Remote Shutdown Panel Actions For Security Event. This is a bank JPM under the Reactor Coolant System - Primary System Heat Removal from Reactor Core Safety Function. (K/A 2.4.28 - IR 3.2 / 4.1)

P-3 The applicant will energize Vital DC Bus D1 with B021, Swing Battery Charger per SO23-6-15, Operation of 125 VDC Systems, Attachment 16, B021, Swing Battery Charger, Operations. This is a new JPM under the DC Electrical Distribution System - Electrical Safety Function. This is a PRA significant action. (K/A 058.AA1.01 - IR 3.4 / 3.5)

Facility:	SONGS 2 & 3	Scenario No.:	1	Op Test No.:	October 2011 NRC
Examiners:		Operators:			

Initial Conditions: 100% power MOL - RCS Boron is 980 ppm (via sample).

Turnover: Maintain steady-state power conditions.

Critical Tasks:

- Restore Flow to CCW Non-Critical Loop Prior to Exceeding Reactor Coolant Pump Operating Limits (Thrust Bearing Temperatures ≥ 225°F) per SO23-13-7, Loss of CCW / SWC.
- Manually Trip the Reactor Using Reactor Trip Pushbuttons Following Multiple CEA Drops per SO23-13-13, Misaligned or Immovable Control Element Assembly
- Manually Trip the Reactor by Deenergizing CEDM Motor Generators Following Reactor Protection System Failure per SO23-12-1, Standard Post Trip Actions.

Event No.	Malf. No.	Event Type*	Event Description
1 +10 min	RC24B	I (RO, SRO)	Pressurizer Spray Valve (PV-0100B) Fails 25% Open.
2 +20 min	SC01A	C (BOP, SRO) TS (SRO)	Salt Water Cooling Pump (P-112) Shaft Seizure.
3 +60 min	RD5603	R (RO) N (BOP, SRO) TS (SRO)	Control Element Assembly (CEA #56) Drops into Core. Power Reduction for Dropped CEA.
4 +65 min	RD0103	C (RO, SRO)	2nd Control Element Assembly (#01) Drops into Core. Reactor Trip Required.
5 +65 min	RP22A-H	C (RO/BOP)	Reactor Trip Breakers Fail to Open Upon Manual Reactor Trip.
6 +65 min	RCP LP	M (RO, BOP, SRO)	Loss of Reactor Coolant Pump Buses 2A01 and 2A02. Loss of Forced Circulation.
7 +65 min	TC02A TC02H	C (BOP)	High Pressure Turbine Stop Valves (HV-2200A & HV-2200H) Fail to Close.
8 +70 min	FW23	C (BOP)	Loss of Condenser Vacuum at 100% Severity.
9 +70 min	RP01O RP01P	C (RO)	Auxiliary Feedwater Pumps (P-141 & P-504) Fail to Start on Emergency Feedwater Actuation Signal.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	Target Quantitative Attributes
9	Total malfunctions (5-8)
3	Malfunctions after EOP entry (1-2)
4	Abnormal events (2-4)
1	Major transients (1-2)
2	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

SCENARIO SUMMARY NRC #1

The crew will assume the watch at 100% power with no scheduled activities per Operating Instruction (OI) SO23-5-1.7, Power Operations.

The first event is a Pressurizer Spray Valve that fails 25% open. The crew will respond per Annunciator Response Procedures (ARPs) and Abnormal Operating Instruction (AOI) SO23-13-27, Pressurizer Pressure and Level Malfunction. If pressure drops below 2025 PSIA, the SRO will refer to Technical Specifications.

When Pressurizer pressure is normal, the Train A Salt Water Cooling Pump will seize. This will require a transfer to Train B Component Cooling Water System per AOI SO23-13-7, Loss of Component Cooling Water (CCW) / Salt Water Cooling (SWC). The SRO will refer to Technical Specifications.

When CCW and SWC are restored, Control Element Assembly (CEA) #56 will drop into the core. Crew actions are per AOI SO23-13-13, Misaligned or Immovable Control Element Assembly, and include a power reduction as required per procedure. The crew will restore Reactor Coolant System Cold Leg temperature and then continue with a power reduction using AOI SO23-13-28, Rapid Power Reduction. The SRO will refer to Technical Specifications.

When power has been lowered an additional 3% to 5%, a second CEA (#01) will drop into the core necessitating a manual Reactor Trip. An automatic trip is not initiated as this is not a Targeted CEA.

When the Reactor Trip pushbuttons are depressed, the Reactor Trip Circuit Breakers (RTCBs) will fail to open. This condition creates an Anticipated Transient Without Scram and is remedied by deenergizing 480 Volt Buses B15 and B16 on CR-63 which open the Control Element Drive Mechanism Motor Generator output contactors. The Reactor Trip is complicated by a loss of both Reactor Coolant Pump (RCP) Buses 2A01 and 2A02, High Pressure Turbine Stop Valves that fail to close, loss of Condenser Vacuum, and Motor Driven Auxiliary Feedwater Pumps that fail to start on an Emergency Feedwater Actuation Signal. The RTCBs will eventually open when a bona fide Reactor Protection System signal is received when RCPs are lost.

The crew will perform Emergency Operating Instruction (EOI) SO23-12-1, Standard Post Trip Actions (SPTAs) and then transition to EOI SO23-12-7, Loss of Forced Circulation / Loss of Offsite Power. The scenario is terminated when Natural Circulation is verified per EOI SO23-12-11, EOI Supporting Attachments, Floating Step 3 (FS-3), Monitor Natural Circulation Established.

Risk Significance:

• Failure of risk important system prior to trip:	Pressurizer Spray Valve Failure
	Loss of Saltwater Cooling Pump
• Risk significant core damage sequence:	Anticipated Transient Without Scram
• Risk significant operator actions:	Trip Reactor Due to 2nd Dropped Rod
	Deenergize Buses B15 and B16
	Manually Trip Turbine
	Start MDAFW Pumps Following EFAS

Facility:	SONGS 2 & 3	Scenario No.:	2	Op Test No.:	October 2011 NRC
Examiners:		Operators:			

Initial Conditions: 100% power MOL - RCS Boron is 980 ppm (via sample).

Turnover: Maintain steady-state power conditions.

Critical Tasks:

- Restore Flow to CCW Non-Critical Loop Within 10 minutes and Prior to Exceeding Reactor Coolant Pump Operating Limits (Thrust Bearing Temperatures ≥ 225°F) per SO23-13-26, Loss of Power to an AC Bus.
- Restore Feedwater Flow to At Least One Steam Generator Prior to Reaching 0% Wide Range Level in Both Steam Generators per SO23-12-1, Standard Post Trip Actions.
- Restore Power to a 1E 4160 Volt Bus per SO23-12-11, EOI Supporting Attachments, Attachment 24, Supplying 1E 4 kV Bus with Opposite Unit Diesel.

Event No.	Malf. No.	Event Type*	Event Description
1 +10 min	RC15B	I (RO, SRO)	Pressurizer Pressure Control Channel Y (PT-0100Y) Fails High.
2 +20 min	SG03C	I (BOP, SRO) TS (SRO)	Steam Generator (E-088) Pressure Transmitter (PT-1023-3) Fails Low.
3 +25 min	CV19 CVCS LP	I (RO, SRO)	Letdown Temperature Control Valve Transmitter (TT-0223) Fails Low. Boronometer Valve (TV-0224A) Fails to Reposition.
4 +45 min	ED03A	C (RO, BOP, SRO) TS (SRO)	Overcurrent Trip of 1E 4160 Volt Bus 2A04.
5 +50 min	PG24	M (RO, BOP, SRO)	Loss of Offsite Power.
6 +50 min	RD5002 RD6402 RD7402	C (RO)	Three (3) Stuck Control Element Assemblies. Loss of Reactivity Control, Emergency Boration Required.
7 +52 min	EG08B	C (BOP)	Train B Emergency Diesel Generator (G-003) Start Failure. Station Blackout.
8 +55 min	AFW LP	C (BOP)	Auxiliary Feedwater Pump (P-140) Fails to Start on Emergency Feedwater Actuation Signal (EFAS). Loss of Feedwater Flow.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	Target Quantitative Attributes
8	Total malfunctions (5-8)
3	Malfunctions after EOP entry (1-2)
3	Abnormal events (2-4)
1	Major transients (1-2)
2	EOPs entered/requiring substantive actions (1-2)
1	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

SCENARIO SUMMARY NRC #2

The crew will assume the watch at 100% power with no scheduled activities per Operating Instruction (OI) SO23-5-1.7, Power Operations. When the Shift Turnover is complete, a Pressurizer Pressure Channel fails high. Actions are per Abnormal Operating Instruction (AOI) SO23-13-27, Pressurizer Pressure and Level Malfunction. The alternate controlling channel will be placed in service and Pressurizer Heaters will be restored to operation. If pressure drops below 2025 PSIA, the SRO will refer to Technical Specifications.

When Technical Specifications have been referenced, a Steam Generator Pressure Channel fails low. The crew will respond per AOI SO23-13-18, Reactor Protection System Failure, and OI SO23-3-2.38, Digital Control System Operation. Steam Generator Pressure trips will be bypassed in the Reactor Protection System and the Feedwater Control System. The SRO will refer to Technical Specifications.

The next event is a low failure of the Letdown Temperature Control Valve (TCV) Transmitter. Letdown temperature quickly rises, causing a high temperature alarm, with a failure of automatic actions to isolate the Boronometer. The crew will respond per the Annunciator Response Procedures (ARPs) to manually satisfy the automatic actions and restore Letdown temperature to normal. The TCV will remain in MANUAL.

When Letdown temperature is stable, a loss of Train A 1E Bus 2A04 will occur due to an overcurrent trip and lockout. The crew will enter AOI SO23-13-26, Loss of Power to an AC Bus. Crew actions include placing a Charging Pump in service, transferring to the Train B Component Cooling Water System, initiating Train B Toxic Gas Isolation System, and starting a Containment Dome Air Circulating Fan. The SRO will refer to Technical Specifications.

When conditions are stable, a Loss of Offsite Power will occur. The crew will enter Emergency Operating Instruction (EOI) SO23-12-1, Standard Post Trip Actions (SPTAs), and perform actions to stabilize the plant. During the trip, three Control Element Assemblies will be stuck, Train B Emergency Diesel Generator (EDG) will fail to start, and the Turbine Driven Auxiliary Feedwater (TDAFW) Pump will fail to start on the Emergency Feedwater Actuation Signal (EFAS). The SRO will recognize a Loss of Reactivity Control, Station Blackout, and Loss of Feedwater requiring entry into EOI SO23-12-9, Functional Recovery.

The Loss of Feedwater event is remedied by manually starting P-140, TDAFW Pump. The Station Blackout cannot be remedied until power is available, therefore, reenergizing the 1E 4 kV Bus is a priority and is accomplished by cross connecting with the Unit 3 Train B Emergency Diesel Generator. The Loss of Reactivity Control is remedied by Reactor power level lowering below $1x10^{-4}$% power. The crew may opt to initiate Emergency Boration when 1E Bus power is restored. The scenario is terminated when power is restored to Bus 2A06 and a Charging Pump and Component Cooling Water Train are returned to service.

Risk Significance:

- Failure of risk important system prior to trip: Loss of 1E 4160 Volt Bus 2A04
- Risk significant core damage sequence: Loss of Reactivity Control

  Station Blackout / Loss of Feedwater Flow
- Risk significant operator actions: Restore Flow to Non-Critical Loop

  Restore Power to 4160 Volt Bus 2A06

Facility:	SONGS 2 & 3	Scenario No.:	3	Op Test No.:	October 2011 NRC
Examiners:	____________		Operators:	____________	
	____________			____________	
	____________			____________	

Initial Conditions: 100% power MOL - RCS Boron is 980 ppm (via sample).

Turnover: Maintain steady-state power conditions.

Critical Tasks:
- Restore Flow to CCW Non-Critical Loop Within 10 minutes and Prior to Exceeding Reactor Coolant Pump Operating Limits (Thrust Bearing Temperatures ≥ 225°F) per SO23-13-7, Loss of CCW / SWC.
- Establish Stable Reactor Coolant System Temperature per SO23-12-11, EOI Supporting Attachments, FS-30, Establish Stable RCS Temperature during ESDE.
- Identify and Isolate the Most Affected Steam Generator Prior to Exiting SO23-12-5, Excess Steam Demand Event.

Event No.	Malf. No.	Event Type*	Event Description
1 +10 min	CC06B CCW LP	C (BOP, SRO) TS (SRO)	Component Cooling Water Pump (P-025) Overcurrent Trip. Component Cooling Water Pump (P-024) Start Failure.
2 +20 min	NI08C	I (RO, SRO) TS (SRO)	Nuclear Instrument Linear Power Channel (JI-0002C) Low Failure.
3 +45 min	FW23	R (RO) N (BOP, SRO)	Partial Loss of Condenser Vacuum @ 3% Severity. Perform a Turbine Load Reduction.
4 +55 min	FC05B	I (BOP, SRO)	Steam Generator (E-088) Main Feedwater Master Controller Setpoint Fails to 45% Level on 240 Second Ramp.
5 +57 min	MS03A	M (RO, BOP, SRO)	Steam Generator (E-088) Main Steam Line Break Inside Containment @ 0.5% Severity. ESDE Inside Containment.
6 +60 min	K403B	I (RO)	Train B Safety Injection Actuation Signal Relay Failure.
7 +65 min	RP01M	C (RO)	Containment Spray Pump (P-012) Start Failure.
8 +65 min	MSIS LP	C (BOP)	Main Steam Isolation Signal Fails to Actuate.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	**Target Quantitative Attributes**
8	Total malfunctions (5-8)
3	Malfunctions after EOP entry (1-2)
4	Abnormal events (2-4)
1	Major transients (1-2)
2	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

SCENARIO SUMMARY NRC #3

The crew will assume the watch at 100% power with no scheduled activities per Operating Instruction (OI) SO23-5-1.7, Power Operations.

The first event is a trip of the running Component Cooling Water (CCW) Pump P-025. The crew will attempt to start the standby Train A CCW Pump P-024, however, this pump also fails to start. Operator actions are per Abnormal Operating Instruction (AOI) SO23-13-7, Loss of Component Cooling Water/Salt Water Cooling. The crew will transfer CCW to Train B along with the Non-Critical Loop and Letdown Heat Exchanger. The SRO will refer to Technical Specifications.

When the plant systems are stable, Linear Power Channel C will fail low. The crew will refer to AOI SO23-13-18, Reactor Protection System Failure / Loss of Vital Bus. The RO will determine the affected instrument by operating the Channel C Core Protection Calculator ROM Station. Once identified, the affected Channel C trips will be bypassed. The SRO will refer to Technical Specifications.

When the channel is bypassed, a partial loss of Condenser vacuum will occur. The crew will respond per the Annunciator Response Procedures (ARPs) and AOI SO23-13-10, Loss of Condenser Vacuum and lower power level until the Turbine Vacuum Limit is in the Area of Unrestricted Operation. Once power level is reduced, the source of the vacuum leak will be located and Condenser vacuum will be restored.

When plant parameters are stable, Steam Generator E-088 Master Controller Setpoint slowly fails to 45%. Entry into SO23-13-24, Feedwater Control System Malfunction is required. Steam Generator level control is restored by placing the Master Controller in MANUAL and will remain in this position until the Main Steam Line break occurs.

When control of level is established, a Main Steam Line break will occur inside Containment on Steam Generator E-088. The crew will enter Emergency Operating Instruction (EOI) SO23-12-1, Standard Post Trip Actions (SPTAs) and then transition to EOI SO23-12-5, Excess Steam Demand Event (ESDE). Procedure entries include EOI SO23-12-11, EOI Supporting Attachments, Floating Step 30 (FS-30), Establish Stable RCS Temperature during ESDE, which is required to stabilize Reactor Coolant System (RCS) temperature when E-088 dryout is reached or pressure lowers below 200 PSIG.

This scenario is complicated by a failure of Train B Safety Injection Actuation System valves to open, a Train A Containment Spray Pump start failure, and a Main Steam Isolation Signal that fails to actuate. This scenario is terminated when RCS Cold Leg temperature has been stabilized per FS-30 and the affected Steam Generator isolated per SO23-12-5, Excess Steam Demand Event.

Risk Significance:

- Failure of risk important system prior to trip: Loss of Component Cooling Water
- Risk significant core damage sequence: ESDE Inside Containment
- Risk significant operator actions: Transfer CCW Non-Critical Loop
  Initiate Main Steam Isolation Signal
  Stabilize RCS Temperature during ESDE
  Isolate Steam Generator E-088

Facility: SONGS 2 & 3	Scenario No.: 4	Op Test No.: October 2011 NRC	
Examiners:		Operators:	

Initial Conditions: 100% power MOL - RCS Boron is 980 ppm (via sample).

Turnover: Maintain steady-state power conditions.

Critical Tasks:

- Restore Component Cooling Water Flow Due to Train A Leakage Prior to Exiting SO23-13-7, Loss of Component Cooling Water / Saltwater Cooling.
- Initiate Emergency Boration for Two (2) Stuck Control Element Assemblies Prior to Exiting SO23-12-1, Standard Post Trip Actions.
- Restore Feedwater Flow to At Least One Steam Generator Prior to Reaching 0% Wide Range Level in Both Steam Generators per SO23-12-6, Loss of Feedwater.

Event No.	Malf. No.	Event Type*	Event Description
1 +10 min	RC15B	I (RO, SRO)	Pressurizer Pressure Control Channel Y (PT-0100Y) Fails Low.
2 +20 min	CC05A	C (BOP, SRO) TS (SRO)	Train A Component Cooling Water Heat Exchanger (E-001) Tube Leak.
3 +25 min	RP18	I (RO, SRO) TS (SRO)	Control Element Assembly Calculator #2 Failure.
4 +45 min	MFW LP	R (RO) N (BOP, SRO)	Main Feedwater Pump (K-005 / P-063) Trip. Initiate Rapid Power Reduction to 70%.
5 +50 min	FW09D FW09E	M (RO, BOP, SRO)	Main Feedwater Pump (K-006 / P-062) High Vibration Trip.
6 +50 min	RD0602 RD4102	C (RO)	Two (2) Stuck Control Element Assemblies (#6 & #41) upon Reactor Trip. Emergency Boration Required.
7 +51 min	2A07 LP	C (BOP)	Non-1E Bus 2A07 Fails to AUTO Transfer Upon Reactor Trip.
8 +55 min	FW02A FW02B FW25	C (BOP)	Motor Driven AFW Pumps (P-141 / P-504) Shaft Seizure (post-trip). Turbine Driven AFW Pump (P-140) Overspeed Trip (post-trip). Loss of all Feedwater.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	**Target Quantitative Attributes**
8	Total malfunctions (5-8)
3	Malfunctions after EOP entry (1-2)
3	Abnormal events (2-4)
1	Major transients (1-2)
2	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

SCENARIO SUMMARY NRC #4

The crew will assume the watch at 100% power with no scheduled activities per Operating Instruction (OI) SO23-5-1.7, Power Operations. When the Shift Turnover is complete, a Pressurizer Pressure Channel fails low. Actions are per the Annunciator Response Procedures (ARPs) and Abnormal Operating Instruction (AOI) SO23-13-27, Pressurizer Pressure and Level Malfunction. The alternate controlling channel will be placed in service and Pressurizer Heaters will be restored to operation. If pressure rises above 2275 PSIA, the SRO will refer to Technical Specifications.

When conditions are stable, a tube leak will develop on the Train A Component Cooling Water (CCW) Heat Exchanger. The crew will respond per AOI SO23-13-7, Loss of Component Cooling Water / Saltwater Cooling. Crew actions include transferring to the Train B Component Cooling Water System as well is attempting to isolate the Train A leakage. The SRO will refer to Technical Specifications.

When CCW actions are complete, a Control Element Assembly Calculator (CEAC) will fail. The crew will perform actions per the ARPs and OI SO23-3-2.13, Core Protection / Control Element Assembly Calculator Operation. The SRO will refer to Technical Specifications.

The next event is a trip of Main Feedwater Pump P-062. The crew will reference AOI SO23-13-28, Rapid Power Reduction. A Rapid Power Reduction is performed to reduce Main Turbine load to 70%. Actions include a Boration to the Charging Pump suction per OI SO23-3-2.2, Makeup Operations and insertion of Control Element Assemblies per OI SO23-3-2.19, Control Element Drive Mechanism Control System Operation.

When power is stable at 70%, a second Main Feedwater Pump will trip requiring a manual Reactor trip. The crew will enter Emergency Operating Instruction (EOI) SO23-12-1, Standard Post Trip Actions (SPTAs), and determine that two Control Element Assemblies have failed to insert and requiring an Emergency Boration.

The scenario is complicated with a failure of Non-1E Bus 2A07 to AUTO transfer on Reactor Trip. The Motor Driven Auxiliary Feedwater Pumps will operate for two minutes prior to tripping and the Turbine Driven AFW Pump will trip after five minutes rendering a total Loss of Feedwater Flow.

The crew will transition from EOI SO23-12-1, SPTAs, to EOI SO23-12-6, Loss of Feedwater. When the Reactor Coolant Pumps are secured in EOI SO23-12-6, the Turbine Driven Auxiliary Feedwater Pump overspeed trip will be reset per EOI SO23-12-11, EOI Supporting Attachments, FS-11, Reset P-140 Overspeed Trip.

The scenario is terminated when Auxiliary Feedwater System flow is restored to either Steam Generator.

Risk Significance:

- Failure of risk important system prior to trip: Train A CCW Heat Exchanger
- Risk significant core damage sequence: Loss of Feedwater Flow
- Risk significant operator actions: Transfer CCW Non-Critical Loop

  Emergency Borate Due to Stuck CEAs

  Restore Feedwater Flow to any Steam Generator

Facility:	SONGS 2 & 3	Scenario No.:	5	Op Test No.:	October 2011 NRC
Examiners:	______		Operators:	______	
	______			______	
	______			______	

Initial Conditions: 100% power MOL - RCS Boron is 980 ppm (via sample).

Turnover: Maintain steady-state power conditions. Pump the Containment Normal Sump.

Critical Tasks:

- Manually Initiate Reactor Trip Following Reactor Protection System Failure Within One Minute of Entry into SO23-12-1, Standard Post Trip Actions.
- Establish Minimum Safety Injection Flow Prior to Exiting SO23-12-1, Standard Post Trip Actions.
- Establish Stable Reactor Coolant System Temperature per SO23-12-11, EOI Supporting Attachments, FS-30, Establish Stable RCS Temperature during ESDE.
- Identify and Isolate the Most Affected Steam Generator Prior to Exiting SO23-12-11, EOI Supporting Attachments, Attachment 29, Isolation of Steam Generator with ESDE.

Event No.	Malf. No.	Event Type*	Event Description
1 +10 min		N (RO, SRO)	Pump Containment Normal Sump for Return to Service Testing of Containment Sump Pump (P-008).
2 +20 min	SG05F	I (BOP, SRO) TS (SRO)	Steam Generator (E-089) Narrow Range Level (LT-1113-2) Fails Low.
3 +35 min	RC16B PZR LP	I (RO, SRO) TS (SRO)	Pressurizer Level Control Channel Y (LT-0110-2) Fails High. 1E 480 Volt Pressurizer Heater Bank Overcurrent Trip.
4 +40 min	NSW LP	C (BOP, SRO)	Nuclear Service Water Pump (P-139) Overcurrent Trip. Nuclear Service Water Pump (P-138) Auto Start Failure.
5 +41 min	OBE LP		Operating Basis Earthquake (OBE) Without Main Feedwater Pump Trip.
6 +41 min	TU08 RP15 RP24A-D RC19	I (RO)	Inadvertent Turbine Trip. Automatic Reactor Trip Failure. Diverse Scram System / ATWS Trip Failure. Failed Fuel Upon Reactor Trip.
7 +43 min	RC03 MS03B	M (RO, BOP, SRO)	Small Break Loss of Coolant Accident at 300 GPM. Steam Generator (E-089) Steam Line Break Inside Containment.
8 +45 min	RP01H	C (BOP)	Component Cooling Water Pump (P-026) Start Failure on SIAS. Manual Start Required.
9 +45 min	EC08DA RP01C	C (RO)	Train A HPSI Pump (P-018) Overcurrent Trip. Train B HPSI Pump (P-019) Start Failure on SIAS. Manual Start Required.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	Target Quantitative Attributes
9	Total malfunctions (5-8)
4	Malfunctions after EOP entry (1-2)
2	Abnormal events (2-4)
2	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
1	EOP contingencies requiring substantive actions (0-2)
4	Critical tasks (2-3)

SCENARIO SUMMARY NRC #5

The crew will assume the watch at 100% power per Operating Instruction (OI) SO23-5-1.7, Power Operations. Scheduled activities include performance of Return-to-Service testing of Containment Sump Pump P-008 per OI SO23-2-16, Operation of Waste Water Systems.

When the Containment Sump is pumped, a Steam Generator Level Transmitter will fail low. The crew will determine level instrument failure per Annunciator Response Procedures (ARPs), enter Abnormal Operating Instruction (AOI) SO23-13-18, Reactor Protection System Failure, and be required to bypass the failed signal using SO23-3-2.38, Digital Control System Operation. The SRO will refer to Technical Specifications.

When bypassing is complete, the controlling Pressurizer Level Channel will fail high. Actions are per the ARPs and AOI SO23-13-27, Pressurizer Pressure and Level Malfunction. This event is complicated by a Train B 1E Pressurizer Heater overcurrent trip. The SRO will refer to Technical Specifications.

Once Technical Specifications are addressed, the running Nuclear Service Water (NSW) Low Pressure Pump will trip. The standby NSW High Pressure Pump will fail to AUTO start and require manual actions as outlined in the Annunciator Response Procedures.

When NSW flow is restored, an Operating Basis Earthquake will occur which is immediately followed by an inadvertent Turbine trip. The Reactor will fail to trip upon Turbine trip and require manual actions by the crew. A Small Break Loss of Coolant Accident, failed fuel, and an Excess Steam Demand Event (ESDE) inside Containment are initiated when the Reactor is manually tripped.

The crew will enter Emergency Operating Instruction (EOI) SO23-12-1, Standard Post Trip Actions (SPTAs), and then transition to EOI SO23-12-9, Functional Recovery. Recovery actions include entry into EOI SO23-12-11, EOI Supporting Attachments, Attachment 29, Isolation of Steam Generator with ESDE, and FS-30, Establish Stable RCS Temperature during ESDE.

This scenario is complicated by failure of the Train B Component Cooling Water and High Pressure Safety Injection Pumps to automatically start upon a Safety Injection Actuation Signal (SIAS). Additionally, the Train A High Pressure Safety Injection Pump (P-018, Swing Pump) will overcurrent trip upon SIAS.

The scenario is terminated when Steam Generator E-089 is isolated per SO23-12-11, EOI Supporting Attachments, Attachment 29, Isolation of Steam Generator with ESDE.

Risk Significance:

- Failure of risk important system prior to trip: Loss of Train B 1E Pressurizer Heaters
- Risk significant core damage sequence: Inadvertent Turbine Trip with Reactor Trip Failure; Small Break LOCA with ESDE
- Risk significant operator actions: Initiate Manual Reactor Trip; Start Train B Component Cooling Water Pump; Start Train B High Pressure Safety Injection Pump; Isolate Steam Generator E-089

Facility: SONGS 2 & 3	Scenario No.: 6	Op Test No.: October 2011 NRC	
Examiners:		Operators:	

Initial Conditions: ~4% power MOL - RCS Boron is 1450 ppm (via sample).

Turnover: Place Auxiliary Feedwater System in Standby and raise Reactor power from 4% to 18%.

Critical Tasks:
- Reduce Reactor Coolant System T_{HOT} to less than 530°F Prior to Exiting SO23-12-4, Steam Generator Tube Rupture.
- Manually Actuate Safety Injection and Containment Cooling Actuation Signals Prior to Exiting SO23-12-4, Steam Generator Tube Rupture.
- Isolate the Ruptured Steam Generator Prior to Exiting SO23-12-4, Steam Generator Tube Rupture.

Event No.	Malf. No.	Event Type*	Event Description
1 +10 min		N (BOP, SRO)	Place Auxiliary Feedwater System in Standby.
2 +30 min		R (RO) N (BOP, SRO)	Raise Reactor Power from 4% to 18% in Preparation for Turbine Startup.
3 +40 min	RC11A	I (RO, SRO)	Reactor Coolant System Loop 1 T_{HOT} (TT-0111X1) Fails High.
4 +45 min	TP02B TP08A	C (BOP, SRO)	Turbine Plant Cooling Water Pump (TPCW) P-120 Trip. TPCW Pump P-119 Auto Start Failure. Manual Start Required.
5 +50 min	CS05A	TS (SRO)	Refueling Water Storage Tank Level Transmitter (LT-0305-1) Fails Low.
6 +55 min	SG06B	C (RO, SRO) TS (SRO)	Steam Generator Tube Leak (E-089) at ~10 GPM.
7 +60 min	SG06B	M (RO, BOP, SRO)	Steam Generator Tube Rupture (E-089) at ~300 GPM.
8 +60 min	ED06R	C (BOP)	1E 480 Volt Buses 2B06 and 2B26 Feeder Breaker Ground Overcurrent.
9 +65 min	SIAS LP CCAS LP	I (RO)	Safety Injection (SIAS) and Containment Cooling Actuation Signals (CCAS) Fail To Automatically Actuate. Manual Actuation Required.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications

Actual	Target Quantitative Attributes
7	Total malfunctions (5-8)
2	Malfunctions after EOP entry (1-2)
2	Abnormal events (2-4)
1	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

SCENARIO SUMMARY NRC #6

The crew will assume the watch with Reactor power at 4% per Operating Instruction (OI) SO23-5-1.3.1, Plant Startup from Hot Standby to Minimum Load. The Steam Bypass Control System is in operation controlling Reactor Coolant System temperature.

When Shift Turnover is complete, the Auxiliary Feedwater System will be placed in Standby per OI SO23-2-4, Auxiliary Feedwater System Operation. When the Auxiliary Feedwater Pumps are secured, a power increase to 18% using CEAs and Reactor Coolant System Dilution with entry into MODE 1 will be performed.

When power has been raised 3% to 5%, a Reactor Coolant System (RCS) Loop 1 T_{HOT} Channel will fail high. Actions are per Abnormal Operating Instruction (AOI) SO23-13-27, Pressurizer Pressure and Level Malfunction. Actions include transferring to an OPERABLE channel, changing the input to the T_{AVE} program, and restoring Pressurizer level.

When Pressurizer level is restored, the running Turbine Plant Cooling Water (TPCW) Pump will trip. The crew will respond per the Annunciator Response Procedures (ARPs) and start the standby TPCW Pump.

When plant conditions are stable, a Refueling Water Storage Tank Level Transmitter fails low. The crew will reference AOI SO23-13-18, Reactor Protection System Failure, and place the failed unit in BYPASS. The SRO will refer to Technical Specifications.

The next event is a Steam Generator E-089 Tube Leak. Entry into AOI SO23-13-4, Reactor Coolant Leak, will direct the crew to identify the source and quantity of leakage. The SRO will refer to Technical Specifications and based on leakage indications, will direct a Reactor Trip and entry into Emergency Operating Instruction (EOI) SO23-12-1, Standard Post Trip Actions (SPTAs).

When the Reactor is tripped, a Steam Generator Tube Rupture will occur. The event is complicated with a loss of 1E 480 Volt Buses 2B06 and 2B26 and a failure of the Safety Injection Actuation (SIAS) and Containment Cooling Actuation Signals (CCAS). Both signals must be manually initiated from the Control Room and the Train B Emergency Diesel Generator placed in Maintenance Lockout.

Actions to cooldown and isolate the Steam Generator are performed per EOI SO23-12-4, Steam Generator Tube Rupture. During the cooldown, all Reactor Coolant Pumps must be secured and Natural Circulation verified per EOI SO23-12-11, EOI Supporting Attachments, FS-3, Monitor Natural Circulation Established. The scenario is terminated when Reactor Coolant System T_{HOT} is lowered below 530°F, the ruptured Steam Generator is isolated, and Natural Circulation is verified.

Risk Significance:

- Failure of risk important system prior to trip: Steam Generator Tube Leak
- Risk significant core damage sequence: Steam Generator Tube Rupture
- Risk significant operator actions: Loss of 1E 480 Volt Buses 2B06 & 2B26

  Manually Actuate SIAS & CCAS

  Lower RCS T_{HOT} below 530°F

  Isolate Ruptured Steam Generator