<table>
<tr><td colspan="14">Facility: Cooper Nuclear Station</td><td colspan="5">Date of Exam: July 28, 2014</td></tr>
<tr><td rowspan="2">Tier</td><td rowspan="2">Group</td><td colspan="12">RO K/A Category Points</td><td colspan="5">SRO-Only Points</td></tr>
<tr><td>K 1</td><td>K 2</td><td>K 3</td><td>K 4</td><td>K 5</td><td>K 6</td><td>A 1</td><td>A 2</td><td>A 3</td><td>A 4</td><td>G *</td><td>Total</td><td colspan="2">A2</td><td colspan="2">G*</td><td>Total</td></tr>
<tr><td rowspan="3">1.
Emergency & Abnormal Plant Evolutions</td><td>1</td><td>3</td><td>5</td><td>4</td><td colspan="3" rowspan="3">N/A</td><td>3</td><td>2</td><td colspan="2" rowspan="3">N/A</td><td>3</td><td>20</td><td colspan="2">3</td><td colspan="2">4</td><td>7</td></tr>
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<tr><td>Tier Totals</td><td>4</td><td>7</td><td>5</td><td>4</td><td>3</td><td>4</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>2</td><td>2</td><td>3</td><td>3</td><td>2</td><td>2</td><td>2</td><td>3</td><td>26</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.

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO)	Form ES-401-1

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	Q#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4		X					**Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following:** (CFR: 41.7 / 45.8) AK2.06 Reactor power	3.8	1
295003 Partial or Complete Loss of AC / 6			X				**Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER :** (CFR: 41.5 / 45.6) AK3.03 Load shedding	3.5	2
295004 Partial or Total Loss of DC Pwr / 6					X		**Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER :** (CFR: 41.10 / 43.5 / 45.13) AA2.02 Extent of partial or complete loss of D.C. power	3.5	3
295005 Main Turbine Generator Trip / 3				X			**AA1. Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP :** (CFR: 41.7 / 45.6) ~~AA1.03 Reactor manual control/Rod control and information system~~ AA1.01 Recirculation system: Plant Specific	~~2.7~~ 3.1	4
295006 SCRAM / 1	X						**Knowledge of the operational implications of the following concepts as they apply to SCRAM :** (CFR: 41.8 to 41.10) AK1.01 Decay heat generation and removal	3.7	5
295016 Control Room Abandonment / 7						X	**AK1. Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM ABANDONMENT :** (CFR: 41.8 to 41.10) 2.2.44 **Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.** | (CFR: 41.5 / 43.5 / 45.12)	4.2	6
295018 Partial or Total Loss of CCW / 8			X				**AK3. Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER :** (CFR: 41.5 / 45.6) ~~AK3.05 Placing standby heat exchanger in service~~ AK3.02 Reactor power reduction	~~3.2~~ 3.3	7

295019 Partial or Total Loss of Inst. Air / 8				X			**AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR :** (CFR: 41.7 / 45.6) AA1.04 Service air isolations valves: Plant-Specific	3.3	8
295021 Loss of Shutdown Cooling / 4		X					**AK2. Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following:** (CFR: 41.7 / 45.8) AK2.02 Reactor water cleanup	3.6	9
295023 Refueling Acc / 8				X			**AA1. Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS :** (CFR: 41.7 / 45.6) AA1.06 Neutron monitoring	3.3	10
295024 High Drywell Pressure / 5						X	**High Drywell Pressure** ~~2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions. (CFR: 41.12 / 45.10)~~ 2.4.31 **Knowledge of annunciator alarms, indications, or response procedures**. (CFR: 41.10 / 45.3)	3.5 4.2	11
295025 High Reactor Pressure / 3		X					**EK2. Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:** (CFR: 41.7 / 45.8) EK2.09 Reactor power	3.9	12
295026 Suppression Pool High Water Temp. / 5					X		**EA2. Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:** (CFR: 41.10 / 43.5 / 45.13) EA2.02 Suppression pool level	3.8	13
295027 High Containment Temperature / 5							NA for Cooper (Mark III Only)		
295028 High Drywell Temperature / 5			X				**EK3. Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE :** (CFR: 41.5 / 45.6) EK3.01 Emergency depressurization	3.6	14
295030 Low Suppression Pool Wtr Lvl / 5		X					**EK2. Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following:** (CFR: 41.7 / 45.8) EK2.09 SPDS/ERIS/CRIDS/GDS: Plant-Specific	2.5	15
295031 Reactor Low Water Level / 2	X						**EK1. Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL :** (CFR: 41.8 to 41.10) EK1.03 Water level effects on reactor power	3.7	16

295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1		X					**EK2. Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following:** (CFR: 41.7 / 45.8) EK2.12 Rod control and information system: Plant-Specific	3.6	17
295038 High Off-site Release Rate / 9						X	**295038 High Off-Site Release Rate** 2.1.30 **Ability to locate and operate components, including local controls**. | (CFR: 41.7 / 45.7)	4.4	18
600000 Plant Fire On Site / 8			X				**AK3 Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE:** AK3.04 Actions contained in the abnormal procedure for plant fire on site	2.8	19
700000 Generator Voltage and Electric Grid Disturbances / 6	X						**AK1. Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: |** (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8) AK1.02 Over-excitation	3.3	20
K/A Category Totals:	3	5	4	3	2	3	Group Point Total:		20

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO)		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	Q#
295002 Loss of Main Condenser Vac / 3									
295007 High Reactor Pressure / 3									
295008 High Reactor Water Level / 2		X					**AK2. Knowledge of the interrelations between HIGH REACTOR WATER LEVEL and the following:** (CFR: 41.7 / 45.8) AK2.06 RCIC: Plant-Specific	3.4	21
295009 Low Reactor Water Level / 2					X		**AK2. Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following:** (CFR: 41.7 / 45.8) AK2.01 Reactor water level indication	3.9	22
295010 High Drywell Pressure / 5									
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5				X			**AA1. Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE :** (CFR: 41.7 / 45.6) AA1.01 Suppression pool cooling	3.9	23
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1									
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1	X						**Knowledge of the operational implications of the following concepts as they apply to LOSS OF CRD PUMPS:** (CFR: 41.8 to 41.10) AK1.02 Reactivity control	3.6	24
295029 High Suppression Pool Wtr Lvl / 5									
295032 High Secondary Containment Area Temperature / 5									
295033 High Secondary Containment Area Radiation Levels / 9			X				**EK3. Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS :** (CFR: 41.5 / 45.6) EK3.01 Emergency depressurization	3.3	25
295034 Secondary Containment Ventilation High Radiation / 9									

295035 Secondary Containment High Differential Pressure / 5		X					**EK2. Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE and the following:** (CFR: 41.7 / 45.8) EK2.01 Secondary containment ventilation	3.6	26
295036 Secondary Containment High Sump/Area Water Level / 5						X	**295036 Secondary Containment High Sump/Area Water Level** 2.4.18 **Knowledge of the specific bases for EOPs**. | (CFR: 41.10 / 43.1 / 45.13)	3.3	27
500000 High CTMT Hydrogen Conc. / 5									
K/A Category Point Totals:							Group Point Total:		7

ES-401 BWR Examination Outline Form ES-401-1

Plant Systems - Tier 2/Group 1 (RO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	Q#
203000 RHR/LPCI: Injection Mode			X									**K3. Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following:** (CFR: 41.7 / 45.4) K3.03 Automatic depressurization logic	4.2	28
205000 Shutdown Cooling					X							**K5. Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) :** (CFR: 41.5 / 45.3) K5.03 Heat removal mechanisms	2.8	29
206000 HPCI							X					**A1. Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM controls including:** (CFR: 41.5 / 45.5) A1.08 System lineup: BWR-2,3,4	4.1	30
207000 Isolation (Emergency) Condenser												Not Applicable to Cooper		
209001 LPCS									X			**A3. Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including:** (CFR: 41.7 / 45.7) A3.02 Pump start	3.8	31
209002 HPCS												Not Applicable to Cooper		
211000 SLC		X						X				**K2. Knowledge of electrical power supplies to the following:** (CFR: 41.7) K2.02 Explosive valves **A2. Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** (CFR: 41.5 / 45.6) A2.05 Loss of SBLC tank heaters	3.1 3.1	32 33

212000 RPS	X			X								**K1. Knowledge of the physical connections and/or causeeffect relationships between REACTOR PROTECTION SYSTEM and the following:** (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.06 Control rod drive hydraulic system **K4. Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following:** (CFR: 41.7) K4.12 Bypassing of selected SCRAM signals (manually and automatically): Plant-Specific	3.5 3.9	34 35
215003 IRM										X		**A4. Ability to manually operate and/or monitor in the control room:** (CFR: 41.7 / 45.5 to 45.8) A4.06 Detector drives	3.0	36
215004 Source Range Monitor										X		**A4. Ability to manually operate and/or monitor in the control room:** (CFR: 41.7 / 45.5 to 45.8) A4.01 SRM count rate and period	3.9	37
215005 APRM / LPRM							X				X	**A1. Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including:** (CFR: 41.5 / 45.5) A1.03 Control rod block status **215005 Average Power Range Monitor/Local Power Range Monitor System** **2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. |** **(CFR: 41.5 / 43.5 / 45.12) |** ~~2.2.1 **Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.**~~ ~~| (CFR: 41.5 / 41.10 / 43.5 / 43.6 / 45.1)~~	3.6 4.2	38 39
217000 RCIC								X				**A2. Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** (CFR: 41.5 / 45.6) A2.16 Low condensate storage tank level	3.5	40

218000 ADS									X			**A3. Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including:** (CFR: 41.7 / 45.7) A3.07 Lights and alarms	3.7	41
223002 PCIS/Nuclear Steam Supply Shutoff						X						**K6. Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF :** (CFR: 41.7 / 45.7) K6.06 Various process instrumentation	2.8	42
239002 SRVs			X									**K3. Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on following:** (CFR: 41.7 / 45.4) K3.03 Ability to rapidly depressurize the reactor	4.3	43
259002 Reactor Water Level Control					X							**K5. Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM :** (CFR: 41.5 / 45.3) K5.03 Water level measurement	3.1	44
261000 SGTS							X					**A1. Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including:** (CFR: 41.5 / 45.5) A1.02 Primary containment pressure	3.1	45
262001 AC Electrical Distribution	X										X	K1. **Knowledge of the physical connections and/or cause effect relationships between A.C. ELECTRICAL DISTRIBUTION and the following:** (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.04 Uninterruptible power supply **226001 A.C. Electrical Distribution** 2.1.27 **Knowledge of system purpose and/or function.** | (CFR: 41.7)	3.1 3.9	46 47
262002 UPS (AC/DC)				X								**K4. Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following:** (CFR: 41.7) K4.01 Transfer from preferred power to alternate power supplies	3.1	48
263000 DC Electrical Distribution		X										**K2. Knowledge of electrical power supplies to the following:** (CFR: 41.7) K2.01 Major D.C. loads	3.1	49

264000 EDGs						X					X	**K6. Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET) :** (CFR: 41.7 / 45.7) K6.02 Fuel oil pumps **264000 Emergency Generators (Diesel/Jet)** 2.1.1 **Knowledge of conduct of operations requirements.** (CFR: 41.10 / 45.13)	3.6 3.8	50 51
300000 Instrument Air	X											**K1 Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following:** (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.05 Main Steam Isolation Valve air	3.1	52
400000 Component Cooling Water						X						**K6 Knowledge of the effect that a loss or malfunction of the following will have on the CCWS:** (CFR: 41.7 / 45.7) K6.06 Heat exchangers and condensers	2.9	53
K/A Category Point Totals:												Group Point Total:		26

ES-401-1 BWR Examination Outline Form ES-401-1

Plant Systems - Tier 2/Group 2 (RO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	Q#
201001 CRD Hydraulic	X											**K1. Knowledge of the physical connections and/or causeeffect relationships between CONTROL ROD DRIVE HYDRAULIC SYSTEM and the following:** (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.09 Plant air systems	3.1	54
201002 RMCS														
201003 Control Rod and Drive Mechanism							X					**A1. Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including:** (CFR: 41.5 / 45.5) A1.03 CRD drive water flow	2.9	55
201004 RSCS														
~~201005 RCIS~~			~~X~~									~~**K3. Knowledge of the effect that a loss or malfunction of the ROD SEQUENCE CONTROL SYSTEM (PLANT SPECIFIC) will have on following:**~~ ~~(CFR: 41.7 / 45.4)~~ ~~K3.01 Reactor manual control: BWR-4,5~~	~~3.3~~	~~56~~
201006 RWM														
202001 Recirculation														
202002 Recirculation Flow Control														
204000 RWCU											X	**204000 Reactor Water Cleanup System** 2.1.41 **Knowledge of the refueling process.** | (CFR: 41.2 / 41.10 / 43.6 / 45.13)	2.8	57
214000 RPIS														
215001 Traversing In-core Probe														
215002 RBM														
216000 Nuclear Boiler Inst.								X				**A2. Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** (CFR: 41.5 / 45.6) A2.04 Detector diaphragm failure or leakage	2.9	58
219000 RHR/LPCI: Torus/Pool Cooling Mode														

223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode				X								**K4. Knowledge of RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE design feature(s) and/or interlocks which provide for the following:** (CFR: 41.7) K4.12 Prevention of inadvertent containment spray activation	2.9	59
230000 RHR/LPCI: Torus/Pool Spray Mode														
233000 Fuel Pool Cooling/Cleanup		X										**K2. Knowledge of electrical power supplies to the following:** (CFR: 41.7) K2.02 RHR pumps	2.8	60
234000 Fuel Handling Equipment														
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator						X						**K6. Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM :** (CFR: 41.7 / 45.7) K6.18 Low pressure stop and control valves: Plant-Specific.	2.6	61
245000 Main Turbine Gen. / Aux.														
256000 Reactor Condensate									X			**A3. Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including:** (CFR: 41.7 / 45.7) A3.07 Feedwater heater level	2.9	62
259001 Reactor Feedwater										X		**A4. Ability to manually operate and/or monitor in the control room:** (CFR: 41.7 / 45.5 to 45.8) A4.06 Feedwater inlet temperature	3.4	63
268000 Radwaste														
271000 Offgas														
272000 Radiation Monitoring		X										**K2. Knowledge of electrical power supplies to the following:** (CFR: 41.7) K2.03 Stack gas radiation monitoring system	2.5	64
286000 Fire Protection														
288000 Plant Ventilation														
290001 Secondary CTMT					X							**K5. Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT :** (CFR: 41.5 / 45.3) K5.01 Vacuum breaker operation: BWR-4	3.3	65
290003 Control Room HVAC														

290002 Reactor Vessel Internals			X									**K3. Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following:** (CFR:41.7 / 45.4) K3.03 Reactor power	3.3	56
K/A Category Point Totals:	1	2	1	1	1	1	1	1	1	1	1	Group Point Total:		12

ES-401 BWR Examination Outline Form ES-401-1
Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (SRO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	Q#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4									
295003 Partial or Complete Loss of AC / 6					X		**AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER :** (CFR: 41.10 / 43.5 / 45.13) ~~AA2.03 Battery status: Plant-Specific~~ AA2.04 System lineups.	3.2	76
295004 Partial or Total Loss of DC Pwr / 6						X	**295004 Partial or Complete Loss of D.C. Power** 2.2.19 **Knowledge of maintenance work order requirements.** | (CFR: 41.10 / 43.5 / 45.13)	3.4	77
295005 Main Turbine Generator Trip / 3						X	**295005 Main Turbine Generator Trip** ~~2.1.42~~ ~~**Knowledge of new and spent fuel movement procedures.**~~ ~~(CFR: 41.10 / 43.7 / 45.13)~~ ~~2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.~~ 2.4.6 Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)	~~4.3~~ 4.7	78
295006 SCRAM / 1									
295016 Control Room Abandonment / 7									
295018 Partial or Total Loss of CCW / 8					X		AA2. **Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:** (CFR: 41.10 / 43,5 / 45.13) AA2.03 Cause for partial or complete loss	3.5	79
295019 Partial or Total Loss of Inst. Air / 8									
~~295021 Loss of Shutdown Cooling / 4~~					~~X~~		~~**AA2. Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING :**~~ ~~(CFR: 41.10 / 43.5 / 45.13)~~ ~~AA2.01 Reactor water heatup/cooldown rate~~	~~3.6~~	~~79~~
295023 Refueling Acc / 8									
295024 High Drywell Pressure / 5									
295025 High Reactor Pressure / 3						X	**295025 High Reactor Pressure** 2.3.14 **Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.** (CFR: 41.12 / 43.4 / 45.10)	3.8	80

295026 Suppression Pool High Water Temp. / 5									
295027 High Containment Temperature / 5									
295028 High Drywell Temperature / 5									
295030 Low Suppression Pool Wtr Lvl / 5									
295031 Reactor Low Water Level / 2						X	**295031 Reactor Low Water Level** 2.1.39 **Knowledge of conservative decision making practices.** | (CFR: 41.10 / 43.5 / 45.12)	4.3	81
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1									
295038 High Off-site Release Rate / 9					X		**EA2. Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE :** (CFR: 41.10 / 43.5 / 45.13) EA2.03 †Radiation levels	4.3	82
600000 Plant Fire On Site / 8									
700000 Generator Voltage and Electric Grid Disturbances / 6									
K/A Category Totals:							Group Point Total:		7

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (SRO)		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	Q#
295002 Loss of Main Condenser Vac / 3									
295007 High Reactor Pressure / 3					X		**AA2. Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE** : (CFR: 41.10 / 43.5 / 45.13) AA2.01 Reactor pressure	4.1	83
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5									
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1						X	**295014 Inadvertent Reactivity Addition** 2.4.23 **Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.** | (CFR: 41.10 / 43.5 / 45.13)	4.4	84
295015 Incomplete SCRAM / 1									
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1									
295029 High Suppression Pool Wtr Lvl / 5									
295032 High Secondary Containment Area Temperature / 5									
295033 High Secondary Containment Area Radiation Levels / 9									
295034 Secondary Containment Ventilation High Radiation / 9									
295035 Secondary Containment High Differential Pressure / 5					X		**EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE:** (CFR: 41.8 to 41.10) EA2.01 Secondary containment pressure: Plant-Specific	3.8	85
295036 Secondary Containment High Sump/Area Water Level / 5									
500000 High CTMT Hydrogen Conc. / 5									

K/A Category Point Totals:							Group Point Total:		3

ES-401												BWR Examination Outline Plant Systems - Tier 2/Group 1 (SRO)		Form ES-401-1
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	Q#
203000 RHR/LPCI: Injection Mode														
205000 Shutdown Cooling											X	**205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)** 2.1.36 **Knowledge of procedures and limitations involved in core alterations.** | (CFR: 41.10 / 43.6 / 45.7)	4.1	86
206000 HPCI														
207000 Isolation (Emergency) Condenser														
209001 LPCS														
209002 HPCS														
211000 SLC								X				**A2. Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** (CFR: 41.5 / 45.6) A2.06 Valve openings	3.3	87
212000 RPS														
215003 IRM														
215004 Source Range Monitor														
215005 APRM / LPRM											X	**215005 Average Power Range Monitor/Local Power Range Monitor System** 2.2.22 **Knowledge of limiting conditions for operations and safety limits.** | (CFR: 41.5 / 43.2 / 45.2)	4.7	88
217000 RCIC														
218000 ADS														
223002 PCIS/Nuclear Steam Supply Shutoff														
239002 SRVs														
259002 Reactor Water Level Control														
261000 SGTS														

262001 AC Electrical Distribution								X				**A2. Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** (CFR: 41.5 / 45.6) A2.07 Energizing a dead bus	3.2	89
262002 UPS (AC/DC)														
263000 DC Electrical Distribution														
264000 EDGs											X	**264000 Emergency Generators (Diesel/Jet)** 2.4.28 **Knowledge of procedures relating to a security event (non-safeguards information).** | (CFR: 41.10 / 43.5 / 45.13)	4.1	90
300000 Instrument Air														
400000 Component Cooling Water														
K/A Category Point Totals:												Group Point Total:		5

ES-401 401-1	BWR Examination Outline Plant Systems - Tier 2/Group 2 (SRO)												Form ES-		
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	Q#
201001 CRD Hydraulic														
201002 RMCS														
201003 Control Rod and Drive Mechanism								X				**A2. Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** (CFR: 41.5 / 45.6) A2.10 †Excessive SCRAM time for a given drive mechanism	3.4	91
201004 RSCS														
201005 RCIS														
201006 RWM														
202001 Recirculation														
202002 Recirculation Flow Control														
204000 RWCU														
214000 RPIS														
215001 Traversing In-core Probe														
215002 RBM														
216000 Nuclear Boiler Inst.														
219000 RHR/LPCI: Torus/Pool Cooling Mode														
223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode								X				**A2. Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** (CFR: 41.5 / 45.6) A2.07 Emergency generator failure	3.8	92
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment														
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator														

245000 Main Turbine Gen. / Aux.														
256000 Reactor Condensate														
259001 Reactor Feedwater														
268000 Radwaste											X	**268000 Radwaste** 2.2.42 **Ability to recognize system parameters that are entry-level conditions for Technical Specifications.** (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)	4.6	93
271000 Offgas														
272000 Radiation Monitoring														
286000 Fire Protection														
288000 Plant Ventilation														
290001 Secondary CTMT														
290003 Control Room HVAC														
290002 Reactor Vessel Internals														
K/A Category Point Totals:												Group Point Total:		3

Facility: Cooper Nuclear Station Cooper Nuclear Station			Date of Exam: July 28, 2014			
Category	K/A #	Topic	RO		SRO-Only	
			IR	Q#	IR	Q#
1. Conduct of Operations	2.1.18	2.1.18 **Ability to make accurate, clear, and concise logs, records, status boards, and reports.** | (CFR: 41.10 / 45.12 / 45.13)	3.6	66		
	2.1.31	2.1.31 **Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.** | (CFR: 41.10 / 45.12)	4.6	67		
	2.1.7	2.1.7 **Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. |** (CFR: 41.5 / 43.5 / 45.12 / 45.13)			4.7	94
	2.1.35	2.1.35 **Knowledge of the fuel-handling responsibilities of SROs.** | (CFR: 41.10 / 43.7)			3.9	95
	2.1.					
	2.1.					
	Subtotal		2		2	
2. Equipment Control	2.2.2	2.2.2 **Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.** | (CFR: 41.6 / 41.7 / 45.2)	4.6	68		
	2.2.12	2.2.12 **Knowledge of surveillance procedures.** (CFR: 41.10 / 45.13)	3.7	69		
	2.2.41	2.2.41 **Ability to obtain and interpret station electrical and mechanical drawings.** (CFR: 41.10 / 45.12 / 45.13)	3.5	70		
	2.2.5	2.2.5 **Knowledge of the process for making design or operating changes to the facility.** | (CFR: 41.10 / 43.3 / 45.13)			3.2	96
	2.2.17	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.** (CFR: 41.10 / 43.5 / 45.13)			3.8	97
	2.2.					
	Subtotal		3		2	
3.	2.3.4	2.3.4 **Knowledge of radiation exposure limits under normal or emergency conditions.** (CFR: 41.12 / 43.4 / 45.10)	3.2	71		

Radiation Control	2.3.13	2.3.13 **Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.** | (CFR: 41.12 / 43.4 / 45.9 / 45.10)	3.4	72		
	2.3.7	2.3.7 **Ability to comply with radiation work permit requirements during normal or abnormal conditions.** (CFR: 41.12 / 45.10)	3.5	73		
	2.3.6	2.3.6 **Ability to approve release permits.** | (CFR: 41.13 / 43.4 / 45.10)			3.8	98
	2.3.12	2.3.12 **Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc**. (CFR: 41.12 / 45.9 / 45.10)			3.7	99
	2.3.					
	Subtotal		3		2	
4. Emergency Procedures / Plan	2.4.18	2.4.18 **Knowledge of the specific bases for EOPs**. | (CFR: 41.10 / 43.1 / 45.13)	3.3	74		
	2.4.26	2.4.26 **Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.** (CFR: 41.10 / 43.5 / 45.12)	3.1	75		
	2.4.47	2.4.47 **Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.** | (CFR: 41.10 / 43.5 / 45.12)			4.2	100
	2.4.					
	2.4.					
	2.4.					
	Subtotal		2		1	
Tier 3 Point Total			10	10	7	7

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	2.3.7	Question 11. Could not write a psychometrically correct question dealing with High Drywell Pressure and the ability to comply with radiation work permit requirements during normal or abnormal conditions. Randomly selected another Generic K/A.
SRO 1/1	2.1.42	Question 78. Could not write a psychometrically correct question dealing with a Main Turbine Generator Trip and how that relates to the knowledge of new and spent fuel movement procedures. Randomly selected another Generic K/A.
SRO 1/1	295021.AA2.01	Question 79. This question was too similar to RO Question 9 and needed to be replaced. Could not write a psychometrically correct question dealing with a Loss of Shutdown Cooling that would be different than Question 9. Reselected from the non-used Abnormal Plant Evolution K/As and randomly selected 295018 Partial or Total Loss of CCW. Then randomly selected one of the three A2 K/A because the last two importance values were too low.
Tier 2/Group 1 (RO)	217000.A2.17	Question 40. The high suppression pool suction swap has been removed and there is no other operationally significant impacts associated with high suppression pool level. Using a random number generator 217000.A2.16 was selected from the remaining A2 topics to replace 217000.A2.17. GPJ
Tier 2/Group 1 (RO)	218000.A3.03	Question 41. Acoustic tail pipe monitors are not installed at Cooper Nuclear Station. Using a random number generator, 218000.A3.07 was selected from the remaining topics to replace 218000.A3.03. GPJ
Tier 2/Group 1 (RO)	259002.K5.01	Question 44. Cooper Nuclear Station no longer uses GEMAC/Foxboro/Bailey controller for reactor water level control. Using a random number generator, 259002.K5.03 was selected from the remaining topics to replace 250002.K5.01. GPJ
Tier 2/Group 2 (RO)	216000.A2.13	Question 58. Unable to develop an operationally and psychometrically valid question to this KA. Using a random number generator 216000.A2.04 was selected to replace 216000.A2.13. GPJ
Tier 2/Group1 (RO)	212000.K4.04	Question 35. Cooper Nuclear Station does not have an interlock that prevents supplying both RPS buses simultaneously from their alternate power source. Using a random number generator 212000.K4.12 was selected to replace 212000.K4.04
Tier 2/Group 2 (RO)	201004.K3.01	Question 56. The original randomly selected KA is on the line for RCIS (201005) which is a BWR6 system and is non-applicable to Cooper Nuclear Station (BWR4); Additionally if the intent is to sample 201004 (RSCS) this system is no longer used at Cooper Nuclear Station. Using a random number generator a new system was chosen from Tier 2/Group 2 that was not previously sampled. 290002 Reactor Vessel Internals was selected. Since a K3 topic was being replaced from the K3 topics a random selection yielded K3.03 as the replacement. 290002.K3.03 replaces 201004.K3.01.
Tier 1/Group 1 (RO)	295005 AA1.03	Question 4. The number of steps the candidate must mentally asses to connect the main turbine trip and reactor manual control system/control rod information system is excessive and difficult to write a psychometrically valid question. Using a random number generator 295005 AA1.01 was selected to replace 295005 AA1.03.
Tier 1/Group 1 (RO)	295018 AK3.05	Question 7. Cannot write a psychometrically valid question for this KA. Using a random number generator 295018 AK3.02 was selected to replace 295018 AK3.02.

Tier 2/group 2 (RO)	241000.K6.18	Question 61. Unable to develop an operationally and psychometrically valid test item for this KA. Using a random number generator 241000.K6.05 was selected to replace 241000.K6.18.
Tier 2/Group 2	215005.G2.2.1	Question 39. No applicability of generic KA 2.2.1 as it relates to 215005 could be found as the bases for a question. Using a random number generator 2.2.44 was selected to replace 2.2.1.
Tier 1/Group 1	295003 AA2.03	Question 76. Unable to develop an operationally valid test item for this KA. Using a random number generator 295003 AA2.04 was selected to replace AA2.03
Tier 1/Group 1	295005 G2.4.6	Question 78: Unable to develop an operationally valid test item at the SRO level for this K/A. Selected 2.4.6.

Facility: Cooper Nuclear Station	Date of Examination: 2014
Examination Level: RO X SRO	Operating Test Number: 1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations SKL034-21-131	R,D	Suppression Pool Temp Calculation K/A: 2.1.12 (2.9/4.0)
Conduct of Operations SKL034-50-XX	R,N	Determine RHR Pump NPSH Flow Limit K/A: 2.1.25 (3.9/4.2)
Equipment Control SKL034-50-78	R,D	Perform Diesel Generator Fuel Oil Availability K/A 2.2.37 (3.6/4.6)
Radiation Control SKL034-30-63	R,D	Radiation Protection Table Top K/A 2.3.13 (3.4/3.8)
Emergency Procedures/Plan		

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; ≤ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank (≥ 1)
(P)revious 2 exams (≤ 1; randomly selected)

Facility: Cooper Nuclear Station	Date of Examination: 2014
Examination Level: RO SRO X	Operating Test Number:

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations SKL034-50-80	R,N	Determine required Actions for Plant Chemistry Out of Specification K/A 2.1.34 (SRO 3.5)
Conduct of Operations SKL034-20-114	R,D	Determine Shift Staffing Requirements for Mode Change K/A 2.1.4 (SRO 3.4)
Equipment Control SKL034-50-18	R,D	Determine Post-Maintenance Testing Requirements K/A 2.2.7 (2.0/3.2)
Radiation Control SKL034-30-63	R,D	Radiation Protection Table Top K/A 2.3.13 (3.4/3.8)
Emergency Procedures/Plan SKL034-30-61	S,D	Protective Action Recommendation (PAR) Table Top X K/A 2.4.44 (2.1 / 4.0)

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; ≤ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank (≥ 1)
(P)revious 2 exams (≤ 1; randomly selected)

Facility: Cooper Nuclear Station		Date of Examination: 2014	
Exam Level: RO X SRO-I SRO-U		Operating Test No.: 1	

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
a. Placing SDG In Service From Control Room K/A 262001.K4.06 (3.6 / 3.9) and 264000.A4.04 (3.7/3.7)	S, L, D, P	6
b. Shifting from Single Element to Auto (3 element) K/A 259002 A4.06 (3.1/3.2)	A, S, D	2
c. Recover from Manual Scoop Tube Operations K/A A.2.05 (RO 3.1 / SRO 3.1) A 2.09 (RO3.1 / SRO 3.0)	S, D	1
d. Transfer RPSPP1B from RPS MG Set B to CDP-1A K/A 212000 A1.11 (3.4 / 3.3) and 212000.A2.02 (3.7/3.9)	S, N	7
e. RPV Depressurization with RWCU K/A 204000 A4.03 (3.2 / 3.6), 204000.A1.04 (2.8/2.8)	A, S, E, L, N	3
f. Start HPCI From The ASD Room K/A 206000 A1.02 (4.2/4.2)	S, EN, D	4
g. Startup the Suppression Pool Cooling Mode of RHR (Alternate Path) K/A 219000 A4.02 (3.7 / 3.5)	A, S, D	5
h. Separation of REC Critical Loops K/A 295018 A1A.03 (3.3 / 3.4) and AK3.07 (3.1/3.2)	S, E, M	8
In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. Operate the Diesel Fire Pump Manually K/A 286000; A4.06(3.4/3.4)	A, E, D	8
j. Place 24 VDC Batteries and Associated Chargers in service K/A: 263000 K4.01(3.1/3.4) and 2.2.2 (4.6/4.1)	D	6
k. Conduct Alternate Rod Insertion (Vent Scram Air Header) K/A: 295037 EA1.05 (3.9/4.0)	E, R, D	1

@ 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	

Facility: Cooper Nuclear Station	Date of Examination: 2014
Exam Level: RO SRO-I X SRO-U	Operating Test No.: 1

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
a. Placing SDG In Service From Control Room K/A 262001.K4.06 (3.6 / 3.9) and 264000.A4.04 (3.7/3.7)	S, L, D, P	6
b. Shifting from Single Element to Auto (3 element) K/A 259002 A4.06 (3.1/3.2)	A, S, D	2
c.		
d. Transfer RPSPP1B from RPS MG Set B to CDP-1A K/A 212000 A1.11 (3.4 / 3.3) and 212000.A2.02 (3.7/3.9)	S, N	7
e. RPV Depressurization with RWCU K/A 204000 A4.03 (3.2 / 3.6), 204000.A1.04 (2.8/2.8)	A, S, E, L, N	3
f. Start HPCI From The ASD Room K/A 206000 A1.02 (4.2/4.2)	S, D, EN	4
g Startup the Suppression Pool Cooling Mode of RHR (Alternate Path) K/A 219000 A4.02 (3.7 / 3.5)	A, S, D	5
h. Separation of REC Critical Loops K/A 295018 A1A.03 (3.3 / 3.4) and AK3.07 (3.1/3.2)	S, E, M	8

In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

i. Operate the Diesel Fire Pump Manually K/A 286000; A4.06(3.4/3.4)	A, E, D	8
j. Place 24 VDC Batteries and Associated Chargers in service K/A: 263000 K4.01(3.1/3.4) and 2.2.2 (4.6/4.1)	D	6
l. k. Conduct Alternate Rod Insertion (Vent Scram Air Header) K/A: 295037 EA1.05 (3.9/4.0)	E, R, D	1

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

Facility: Cooper Nuclear Station	Date of Examination: 2014
Exam Level: RO SRO-I SRO-U X	Operating Test No.: 1

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
a.		
b.		
c.		
d. Transfer RPSPP1B from RPS MG Set B to CDP-1A K/A 212000 A1.11 (3.4 / 3.3) and 212000.A2.02 (3.7/3.9)	S, N	7
e. RPV Depressurization with RWCU K/A 204000 A4.03 (3.2 / 3.6), 204000.A1.04 (2.8/2.8)	A, S, E, L, N	3
f. Start HPCI From The ASD Room K/A 206000 A1.02 (4.2/4.2)	S, D, EN	4
g.		
h.		

In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

System / JPM Title	Type Code*	Safety Function
i. Operate the Diesel Fire Pump Manually K/A 286000; A4.06(3.4/3.4)	A, E, D	8
j.		
k. Conduct Alternate Rod Insertion (Vent Scram Air Header) K/A: 295037 EA1.05 (3.9/4.0)	E, R, D	1

@ 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	

Facility: Cooper Nuclear Station Scenario No.: 1 Op-Test No.: 1

Examiners: ______________ Operators: ______________

Initial Conditions: Plant operating at 85% power (BCL.

Turnover: Continue startup to 100% power.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N (BOP	Place Circulating Water Pump D in service (2.2.3)
2	N/A	R (ATC)	Raise Reactor Power using reactor recirculation (2.1.10)
3	RR17a	I (ATC) TS (SRO	RR Pump A runs back in speed on the power rise and must be locked out to halt to speed drop. (Abnormal Procedure 2.4RR). Enters TS LCO 3.4.1 because pump speeds vary by more than 5%.
4	RC08	I (BOP)	PCIS instrument causes RCIC-MO-16 to isolate. TS LCO 3.5.3, Condition A.
5	ED19c	C (BOP)	125V DC power panel AA-3 loses power. (Emergency Procedure 5.3DC125)
6	EG03	C (BOP) R (ATC)	H2 Gas leak in Generator (Abnormal Procedure 2.4 Gen) Reduce power using Recirculation Pump Flow (2.1.10)
7	N/A RP15	M (Crew) I (BOP)	Unit SCRAM due to Generator hydrogen leak (EOP 1A, 6A,7A) PCIS Group 6 failure. Required manual group isolation using radiation monitor switches out of operate.
8	RD15	C , (©) (ATC)	ATWS (6 rods) Manually insert control rods with RMCS (EOP 5.8.3)
9	FW18b MS02b	C , (©) (ATC)	FW Line B Break inside DW (transition to ECCS for RPV Level Control) 2.4MC-RF and 2.4PC.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (T)echnical Specifications, (©) CT

Initial Conditions: 85% power (BCL), “D” CWP out of service,

Turnover:

The “D” CWP is returned to service, 2.2.3 completed to step 5.5, and backwash completed. Raise power to 90% using reactor recirculation system.

Description:

BOP Operator places the 4th Circulating Water Pump in service per procedure 2.2.3 Circulating Water System Operation.

The ATC raises reactor power using Reactor Recirculation flow at least 5% power using procedure 2.1.10, Station Power Changes. During the power rise, RR Pump A runs back in speed requiring the ATC to stop the speed reduction by locking out the pump's scoop tube. Abnormal Procedure 2.4RR is entered. The CRS enters TS LCO 3.4.1, Condition B when RR pump speeds differ by more than 5%.

RCIC PCIS instrument failure causes RCIC-MO-16 steam supply valve to isolate. The instrument is repaired and the valve is reopened. RCIC is declared inoperable per TS 3.5.3.

The 125V DC Panel AA-3 loses power. Reactor Recirculation Pump A trips and the crew enters Procedures 2.4RR and 5.3DC125. Single loop operations are entered and effected loss of DC loads are shifted as needed.

The BOP operator responds to a H_2 Gas leak on the main generator. Abnormal procedure 2.4GEN-H2 will be entered requiring the operator to remove the generator from service due to excessive leakage.

The ATC is directed to scram the reactor as hydrogen pressure lowers to 30 psig. When the reactor is SCRAMMED 6 control rods fail to fully insert (ATWS). The SRO enters EOPs 6A and 7A. The ATC manually inserts control rods per EOP 5.8.3 using the reactor manual control system. Post scram the PCIS Group 6 isolation fails to actuate requiring the BOP to manually insert the isolation signal per operating procedures.

A small Feedwater line break occurs inside the drywell requiring the BOP operator to utilize ECCS systems to maintain reactor water level, and both the ATC and BOP operators take actions to restore RPV water level and containment parameters. The drywell requires containment sprays to maintain pressure within acceptable limits.

The scenario ends when Drywell pressure is being controlled and Reactor Water Level is restored to the normal band and all control rods are inserted.

Critical Task List

Critical Task	**During failure to scram conditions, insert control rods using one or more methods contained within Procedure 5.8.3 before addressing other tasks that detract from inserting control rods in an expeditious manner.**

Basis:
Achieving reactor shutdown is one of the primary goals of EOP 6A.

Critical Task	**Event 11- (BOP) Initiate Drywell spray when torus pressure exceeds 10 psig and before Pressure Suppression Pressure (PSP) Graph is exceeded.**

Basis:

Regarding drywell temperature, if operation of all available drywell cooling is unable to terminate increasing drywell temperature before the structural design temperature limit of 280°F is reached, drywell sprays are initiated to affect the required drywell temperature reduction status of the DSIL and adequate core cooling permitting. Spray operation effects a drywell pressure and temperature reduction through the combined effects of evaporative cooling and convective cooling.

Regarding drywell pressure, operation of drywell sprays reduces primary containment pressure by condensing any steam that may be present and by absorbing heat from the containment atmosphere through the combined effects of evaporative and convective cooling. Drywell sprays are initiated when torus pressure exceeds the Torus Spray Initiation Pressure (10# torus pressure) to preclude chugging the cyclic condensation of steam at the downcomer openings of the drywell vents. When a steam bubble collapses at the exit of the downcomers, the rush of water drawn into the downcomers to fill the void induces stresses at the junction of the downcomers and the vent header in Mark I containments and at the junction of the downcomers. Repeated application of such stresses could cause fatigue failure of these joints; thereby, creating a direct path between the drywell and torus. When drywell sprays are initiated, the resulting pressure reduction opens the vacuum breakers, drawing non-condensable from the torus back into the drywell.

This condition defines the Torus Spray Initiation Pressure. As the drywell atmosphere is purged to the torus and replaced by steam, torus pressure increases. The SCSIP is the lowest torus pressure which can occur when 95% of the non-condensable in the drywell have been transferred to the torus. Since the failure mode is based on fatigue failure, a precise time limit or pressure cannot be provided. Therefore, prompt initiation of drywell sprays is required based on existing EOP priorities.

Procedures used
2.2.3 Circulating Water System Operation (Normal)
Annunciator C-1/E-3, Diesel Gen 1 Trouble (ARP)
2.1.10, Station Power Changes (Normal)
Annunciator 9-4-3/C-2 for scoop tube lockup (ARP)

2.4RR Abnormal for scoop tube lockup (AOP)
Annunciator 9-5-2/E-3 for loss of DC panel (ARP)
5.3DC125 for DC 125 issues (EOP)
Annunciator B-2/B-4 for H2 leakage (ARP)
2.4GEN-H2 Abnormal for H2 leakage (AOP)
2.1.5 Scram (EOP)
EOP-1A
EOP 6A
EOP 7A
5.8.3 Alternate Rod Insertion (EOP)
2.4MC-RF, Condensate or Feedwater Abnormal (AOP)
2.4 PC (AOP)
EOP-3A
Manually insert PCIS per procedure 2.1.22.

Facility: Cooper Nuclear Station Scenario No.: 2 Op-Test No.: 1

Examiners: ______ Operators: ______

Initial Conditions: 100% power, EOL, no equipment out of service,

Turnover: Lower power to 95% for HPCI Test. Following HPCI Test return to 100% power. 6.HPCI.103 completed to Step 4.13.

Event No.	Malf. No.	Event Type*	Event Description
1	SW04	C (BOP) TS (SRO)	RHRSW Pump trips when started. (2.2.70) The SRO addresses Tech Spec 3.7.1 Condition A for one pump INOP
2	N/A	N (BOP)	Place RHR Loop B in Suppression Pool Cooling (2.2.69.3)
3	N/A	R (ATC)	Lower Power with Reactor Recirculation (2.1.10)
4	HP12	N (BOP) TS (SRO)	Perform HPCI Full Flow Test (6.HPCI.103) Enters Tech Spec 3.5.1C and declares HPCI INOP
5	RD04a	I (ATC)	In Service CRD FCV fails closed.
6	RR26c RR26d	I (BOP) TS (SRO)	LIS-72C and LIS-72D fail downscale. RCIC initiates. (Abnormal Procedure 2.4CSCS) LCO 3.5.3 Condition A.
7	IA05	C (BOP)	Instrument Air dryer plugs. (Abnormal Procedure 5.2AIR)
8	RR50A RR10A, RR11A	C (ATC)	Reactor Recirculation Pump high Vibration (ARP) Recirculation Pump Seal Failure (ARP) (Abnormal Procedures 2.4RR and 2.4PC)
9	RR31A RD01	M © (All) C © (ATC)	LOCA – RR Suction Line Break (EOP 1A, 3A) Scram Discharge Volume Drain Vlv. Fails to close

10	O/R	C © (BOP)	RHR Loop A Drywell Spray Valves de-energize.
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (T)echnical Specifications, (©) CT			

Initial Conditions: 100% power, EOL, HPCI INOP for testing.

Turnover:

Lower power to 95% for HPCI Test. Following HPCI Test return to 100% power. 6.HPCI.103 completed to step 4.13

Description:

RHR Suppression Pool cooling is placed into service in preparation for the HPCI run. RHR Service Water Booster system is started and then RHR is placed into service.

The B RHRSW Pump trips when B loop of RHRSW is placed in service per Procedure 2.2.70, RHR Service Water Booster Pump System, to support Suppression Pool Cooling. The other pump in the loop is started per procedure. The CRS addresses Tech Spec LCO 3.7.1 and declares the tripped pump inoperable.

The BOP Operator places B Loop of RHR in Suppression Pool Cooling operation to support the HPCI Full Flow Test per 2.2.69.3, RHR Suppression Pool Cooing and Containment Spray.

The ATC reduces reactor power using reactor recirculation flow to 95% reactor power in preparation for the HPCI Full Flow Test per 2.1.10 Station Power Changes.

The BOP Operator performs the HPCI Full Flow Test per 6.HPCI.103. At step 4.16 the test fails and the SRO addresses Tech Spec 3.5.1C and declares HPCI INOP.

The in-service CRD FCV fails closed requiring the ATC to enter Abnormal Procedure 2.4CRD which requires shifting FCVs per procedure 2.2.8.

The BOP responds to alarms 9-3-2/A-5, 9-3-2/B-5 and 9-4-1/A-1. RCIC starts and must be secured per Abnormal Procedure 2.4CSCS. The SRO enters TS LCO 3.5.3, Condition A for RCIC being inoperable.

The BOP responds to alarm A-4/A-5, CONTROL AIR LOW PRESSURE, due to instrument air dryers plugging. The BOP enters Abnormal Procedure 5.2AIR and opens a motor operated valve that bypasses the air dryer. Once the other air dryer train is placed in service the bypass valve can be closed.

The ATC responds to high vibration on the “A” Reactor Recirculation Pump per alarm 9-4-3/C-3 and begins to reduce recirculation pump speed to reduce vibrations. The vibration continues and the Recirculation Pump seals fail resulting in a LOCA inside the Drywell. The pump is secured and isolated and the primary containment is vented with Standby Gas Treatment. After a time delay the RR suction pipe breaks. The SRO enters EOP 1A and directs the ATC to SCRAM the reactor then enters EOP 3A to address containment pressure issues.

During the SCRAM recovery, one of the SDV drain valves fails to close resulting in a primary system leaking into secondary containment. The ATC recognizes this and closes the valve using the control switch on Panel 9-5 and reports the action to the SRO.

The SRO directs the BOP operator to utilize Drywell Sprays and control drywell pressure between 2 psig and 10 psig per EOP 3A. If the operator fails to control drywell pressure the valves will NOT close resulting in a negative pressure in primary containment.

The scenario ends when the primary containment pressure is being controlled in band (+2 to +10 psig) as directed by the SRO and Reactor water level is in the normal range.

Critical Task List

Critical Task	**Event 9- (ATC) Isolate primary system discharging into secondary containment before any Max Safe Operating Limit is reached.** **NOTE to Examiners: The SDIV drain valve (AO-33) on the south volume fails to isolate when required post scram.**

Critical Task	**Event 9 – (BOP) Initiate Drywell Spray prior to Drywell pressure exceeding the Pressure Suppression Pressure (PSP) graph.** **Note to Examiners: Either Loop of RHR may be used for Torus Spray and Drywell Spray. Only A Loop of RHR may be used for Torus cooling.**

Critical Task	**Event 10 – (BOP) Secure RHR Loop A drywell sprays or lower Loop B sprays prior to Drywell Pressure reaching negative pressure.** **NOTE to Examiners: During performance of Drywell Sprays using procedure 2.2.69.3 during execution of EOP 3A if the operator fails to recognize DW Spray valves have lost power the valves will fail to close resulting in negative DW pressure and challenging PC Integrity.**

Facility: Cooper Nuclear Station Scenario No.: 3 Op-Test No.: 1

Examiners: ______________________ Operators: ______________________

Initial Conditions: 4%, BCL, power, plant startup in progress.

Turnover: Procedure 2.1.1, Steps 4.22 and 5.36 are completed, 2.2.77 Attachment 1, Step 1.14 is complete, 2.2.28.1 Step 5.14 is complete. Fuel is being de-channeled on the refueling floor, No Tech Specs Limitations in effect. Reactor Coolant samples (1xE-3 Micro-Curies / CC) indicate higher than normal gross activity level for this point in a startup. Site Management and Reactor Engineering have indicated the startup can continue. PRA risk is green. Continue plant startup.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R (ATC)	Continue startup using control rods
2	NM05A	I (ATC) TS (SRO)	IRM power supply failure (ARP 9-5-1/D-8) Tech Specs LCO 3.3.1.1, Reactor Protection System (RPS) Instrumentation
3	SW01d N/A	C (BOP) TS (SRO)	Service Water Pump D Trip (ARP B-3/B-7) Tech Spec LCO 3.7.2 Service Water (SW) System and Ultimate Heat Sink (UHS) for number of Operable SW Pumps.
4	RD03d N/A	C (ATC) TS (SRO)	Control Rod Drifts IN (2.4 CRD) Tech Spec LCO 3.1.3 for Inoperable Control Rod.
5	HP05,CR 01, CR03	I (BOP)	HPCI Inadvertent Initiation / Fuel Failure (2.4CSCS)
6	HP06, HP09, HP15	M (All) © R (ATC)	LOCA - Steam line break into Secondary Containment (EOP-5A, 2.4 OG, 5.1RAD) SCRAM (2.1.5)
7	N/A	M © (BOP)	Emergency Depressurization (>2 above Max Safe) EOP-2A
8	FW28B	I (ATC)	Reactor Feedpump fails to trip on high level
9	PC18a/b	I (BOP)	SGT Fans A & B fail to auto start.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (T)echnical Specifications, (©) CT

Initial Conditions: 4%, BCL, power, plant startup in progress.

Turnover:

Procedure 2.1.1, Steps 4.22 and 5.36 are completed, 2.2.77 Attachment 1, Step 1.14 is complete, 2.2.28.1 Step 5.14 is complete, LCOs and Logs are complete for change to MODE 1. Fuel is being de-channeled on the refueling floor, No Tech Specs Limitations in effect. Reactor Coolant samples (1xE-3 Micro-Curies / CC) indicate higher than normal gross activity level for this point in a startup. Site Management and Reactor Engineering have indicated the startup can continue normally. PRA risk is green. Continue plant startup.

Description:

A startup is in progress following a refueling outage. Reactor Power is approximately 4% and the intention is to raise power. Reactor Coolant activity is above normal. The ATC continues the plant startup using control rods using procedure 2.1.1, Startup and procedure 10.13, Control Rod Movement Sequence.

During control rod withdrawal, one IRM power supply fails requiring the ATC to bypass the IRM per ARP 9-5-1/D-8. The SRO addresses Tech Spec 3.3.1.1, Reactor Protection System (RPS) Instrumentation and declares IRM B inoperable. Potential LCO 3.3.1 is logged. The half reactor scram is reset per Procedure 2.1.5.

The BOP operator responds to a Service Water Pump Trip per ARP B-3/B-7 and responds by starting another pump.

The SRO addresses Tech Spec LCO 3.7.2 Service Water (SW) System and Ultimate Heat Sink (UHS) for number of Operable SW Pumps and determines 30 day LCO entry is required per 3.7.2 and 7 day LCO entry is required per 3.8.1 with DG-1 declared inoperable.

One Control Rod drifts in due to a leaking outlet scram valve and the ATC responds per 2.4CRD to fully insert the Control Rod using the emergency in switch. The control rod is fully inserted and the SRO enters Tech Spec 3.1.6 because BPWS is not met. The control rod is declared INOP and de-activated. The SRO addresses Tech Spec LCO 3.1.3 for Inoperable Control Rod.

An Inadvertent HPCI Actuation and Injection occurs resulting in a power excursion that results in some Fuel Failure. The BOP Operator responds per 2.4CSCS and secures HPCI. During the process of tripping HPCI the HPCI Steam line breaks in Secondary Containment. The SRO enters EOP 5A due to high Reactor Building temperature and radiation. EOP 5A provides direction to "isolate systems discharging into the area"

As area temperatures and radiation continue to rise the SRO directs the ATC to SCRAM the reactor per 2.1.5, prior to commencing emergency depressurization, due to the inability to isolate the HPCI steam line break and follows up by entering 5.1 RAD and 2.4 OG.

The crew will be able to anticipate emergency depressurization but will have to close the MSIVs due to MSL HI-HI RAD. Once two or more areas are above Max Safe the SRO will direct the BOP operator to Emergency Depressurize the Reactor per EOP 2A prior to exceeding 312^{o}F anywhere in the reactor building.

The SGT fans fail to auto start and the BOP operator must manually start them to prevent the secondary containment pressure from becoming positive and a potential radioactivity release.

During recovery of reactor water level using Feedwater the remaining Reactor Feedpump fails to trip on high level requiring the BOP operator to control RPV level using ECCS Systems.

The scenario ends when the RPV level has been restored to the normal range following emergency depressurization and the SGT system has restored negative pressure in the secondary containment.

Critical task List

Critical Task	**Event 6- (ATC) With a primary system discharging into secondary containment through an un-isolable break, SCRAM the reactor per EOP 5A and EOP 1A prior to commencing Emergency Depressurization.**

	Critical Task	**Event 7- (BOP) With a primary system discharging into secondary containment through an un-isolable break, execute Emergency Depressurization when Maximum Safe Operating Values are exceeded in two or more areas for the same parameter per EOP2A prior to exceeding 312^{0}F anywhere in the reactor building.**