Three Mile Island Unit 1
Route 441 South, P.O. Box 480
Middletown, PA 17057

Telephone 717-948-8000

Exelon Nuclear

January 08, 2010
TMI-10-002

U.S. NRC Region I Administrator
475 Allendale Road
King of Prussia, PA 19406

Three Mile Island Unit 1
Facility Operating License DPR –50
NRC Docket No. 50-289

Subject: Submittal of Initial Operator Licensing Examination Outline

Enclosed are the examination outlines, supporting the Reactor Operator initial license written examination scheduled for the week of April 12, 2010, at Three Mile Island Unit 1.

This submittal includes all appropriate examination standard forms and outlines in accordance with NUREG 1021, "Operator Licensing Examination Standards," Revision 9 Supplement 1.

In accordance with NUREG 1021, Revision 9 Supplement 1, Section ES-201, "Initial Operator Licensing Examination Process," please ensure that these materials are withheld from public disclosure until after the examinations are complete.

Should you have any questions concerning this letter, please contact Adam Miller of Regulatory Assurance at (717) 948-8128.

For questions concerning examination materials, please contact Greg Hoek at (717) 948-2027.

Respectfully

For
W. Noll

William Noll
Vice President TMI, Unit I

WN/awm

Enclosures: (Fed Ex to Joe D'Antonio, Chief Examiner, NRC Region I)
Examination Security Agreements (Form ES-201-3)
PWR Examination Outline (Form ES-401-2)
Generic Knowledge and Abilities Outline (Tier 3) (Form ES-401-3)
Statement detailing method of Written Exam Outline generation
Record of Rejected K/As (Form ES-401-4)
Completed Checklists:
Examination Outline Quality Checklist (Form ES-201-2)

cc: (without attachments)
Chief, NRC Operator Licensing Branch
NRC Senior Resident Inspector – TMI Unit 1

Facility: TMI						Date of Exam: 04/12/10									
Tier	Group	RO K/A Category Points											SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Plant Evaluations	1	3	3	3				3	3			3	18			6
	2	1	2	1				1	2			2	9			4
	Tier Totals	4	5	4				4	5			5	27			10
2. Plant Systems	1	3	3	1	3	3	3	2	2	3	2	3	28			5
	2	1	1	1	0	1	1	1	1	1	1	1	10			3
	Tier Totals	4	4	2	3	4	4	3	3	4	3	4	38			8

3. Generic Knowledge & Abilities	1	2	3	4	Total	1	2	3	4	Total
	2	2	3	3	10					7

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 that are not included on the outline should be added. Refer to section D.1.b of ES-401, for guidance regarding 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 KAs 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/A's

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IR) 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 10CFR55.43

TMI

Written Examination Outline

Emergency and Abnormal Plant Evolutions - Tier 1 Group 1

EAPE#/Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
015 / 17 / Reactor Coolant Pump Malfunctions / 4	X						AK1.03 - Knowledge of the operational implications of the following concepts as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): The basis for operating at a reduced power level when one RCP is out of service	3.0	39
007 / Reactor Trip / 1	X						EK1.04 - Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Decrease in reactor power following reactor trip (prompt drop and subsequent decay)	3.6	40
022 / Loss of Reactor Coolant Makeup / 2	X						AK1.03 - Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Pump Makeup: Relationship between charging flow and PZR level	3.0	41
009 / Small Break LOCA / 3		X					EK2.03 - Knowledge of the interrelations between the small break LOCA and the following: S/Gs	3.0	42
E05 / Steam Line Rupture - Excessive Heat Transfer / 4		X					EK2.1 - Knowledge of the interrelations between the (Excessive Heat Transfer) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.8	43
008 / Pressurizer Vapor Space Accident / 3		X					AK2.01 - Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Valves	2.7	44
054 / Loss of Main Feedwater / 4			X				AK3.05 - Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): HPI/PORV cycling upon total feedwater loss	4.6	45
056 / Loss of Off-site 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	46
011 / Large Break LOCA / 3			X				EK3.08 - Knowledge of the reasons for the following responses as they apply to the Large Break LOCA: Flowpath for sump recirculation	3.9	47
038 / Steam Generator Tube Rupture / 3				X			EA1.24 - Ability to operate and monitor the following as they apply to a SGTR: Safety injection pump ammeter and indicators	3.6	48

TMI
Written Examination Outline
Emergency and Abnormal Plant Evolutions - Tier 1 Group 1

EAPE#/Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
065 / Loss of Instrument Air / 8				X			AA1.02 - Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Components served by instrument air to minimize drain on system	2.6	49
027 / Pressurizer Pressure Control System Malfunction / 3				X			AA1.02 - Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: SCR-controlled heaters in manual mode	3.1	50
029 / Anticipated Transient Without Scram (ATWS) / 1					X		EA2.09 - Ability to determine or interpret the following as they apply to a ATWS: Occurrence of a main turbine/reactor trip	4.4	51
025 / Loss of Residual Heat Removal 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	52
055 / Station Blackout / 6					X		EA2.01 - Ability to determine or interpret the following as they apply to a Station Blackout: Existing valve positioning on a loss of instrument air system	3.4	53
057 / Loss of Vital AC Electrical Instrument Bus / 6						X	2.4.45 - Emergency Procedures / Plan: Ability to prioritize and interpret the significance of each annunciator or alarm.	4.1	54
E04 / Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4						X	2.1.19 - Conduct of Operations: Ability to use plant computers to evaluate system or component status.	3.9	55
026 / Loss of Component Cooling Water / 8						X	2.1.31 - Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	4.6	56
K/A CategoryTotals	3	3	3	3	3	3	Group Point Total:		18

TMI

Written Examination Outline

Emergency and Abnormal Plant Evolutions - Tier 1 Group 2

EAPE#/Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
024 / Emergency Boration / 1	X						AK1.01 - Knowledge of the operational implications of the following concepts as they apply to Emergency Boration: Relationship between boron addition and change in T-ave	3.4	57
A03 / Loss NNI-X/Y / 7		X					AK2.2 - Knowledge of the interrelations between the (Loss of NNI-Y) 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.	3.3	58
E14 / EOP Rules and Enclosures / 5			X				EK3.4 - Knowledge of the reasons for the following responses as they apply to the (EOP Enclosures) RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.	3.5	59
005 / Inoperable/Stuck Control Rod / 1				X			AA1.04 - Ability to operate and / or monitor the following as they apply to the Inoperable / Stuck Control Rod: Reactor and turbine power	3.9	60
E08 / LOCA Cooldown - Depress. / 4					X		EA2.1 - Ability to determine and interpret the following as they apply to the (LOCA Cooldown) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	2.8	61
076 / High Reactor Coolant Activity / 9						X	2.1.31 - Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	4.6	62
A07 / Flooding / 8						X	2.1.28 - Conduct of Operations: Knowledge of the purpose and function of major system components and controls.	4.1	63
028 / Pressurizer Level Control Malfunction / 2		X					AK2.03 - Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: Controllers and positioners	2.6	64

TMI

Written Examination Outline

Emergency and Abnormal Plant Evolutions - Tier 1 Group 2

EAPE#/Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
060 / Accidental Gaseous RadWaste Release / 9					X		AA2.01 - Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: A radiation-level alarm, as to whether the cause was due to a gradual (in time) signal increase or due to a sudden increase (a "spike"), including the use of strip-chart recorders, meter and alarm observations	3.1	65
K/A Category Totals	1	2	1	1	2	2	Group Point Total:		9

TMI

Written Examination Outline

Plant Systems - Tier 2 Group 1

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
006 Emergency Core Cooling	X											K1.10 - Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: Safety injection tank heating system	2.6	1
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	2
062 AC Electrical Distribution		X										K2.01 - Knowledge of bus power supplies to the following: Major system loads	3.3	3
005 Residual Heat Removal		X										K2.01 - Knowledge of bus power supplies to the following: RHR pumps	3.0	4
022 Containment Cooling			X									K3.01 - Knowledge of the effect that a loss or malfunction of the CCS will have on the following: Containment equipment subject to damage by high or low temperature, humidity, and pressure	2.9	5
078 Instrument Air				X								K4.02 - Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Cross-over to other air systems	3.2	6
063 DC Electrical Distribution				X								K4.01 - Knowledge of dc electrical system design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control	2.7	7
003 Reactor Coolant Pump				X								K4.07 - Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Minimizing RCS leakage (mechanical seals)	3.2	8
004 Chemical and Volume Control					X							K5.37 - Knowledge of the operational implications of the following concepts as they apply to the CVCS: Effects of boron saturation on ion exchanger behavior	2.6	9
007 Pressurizer Relief/Quench Tank					X							K5.02 - Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR	3.1	10

TMI

Written Examination Outline

Plant Systems - Tier 2 Group 1

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
010 Pressurizer Pressure Control						X						K6.03 - Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PZR sprays and heaters	3.2	11
061 Auxillary/Emergency Feedwater						X						K6.02 - Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps	2.6	12
073 Process Radiation Monitoring							X					A1.01 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels	3.2	13
076 Service Water							X					A1.02 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures.	2.6	14
059 Main Feedwater								X				A2.05 - Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture in MFW suction or discharge line	3.1	15
013 Engineered Safety Features Actuation								X				A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; Rapid depressurization	4.4	16
064 Emergency Diesel Generator									X			A3.08 - Ability to monitor automatic operation of the ED/G system, including: Consequences of automatic transfer to automatic position after the ED/G is stopped	3.7	17
012 Reactor Protection									X			A3.05 - Ability to monitor automatic operation of the RPS, including: Single and multiple channel trip indicators	3.6	18

TMI

Written Examination Outline

Plant Systems - Tier 2 Group 1

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
008 Component Cooling Water										X		A4.04 - Ability to manually operate and/or monitor in the control room: Startup of a CCW pump when the system is shut down.	2.6	19
103 Containment										X		A4.06 - Ability to manually operate and/or monitor in the control room: Operation of the containment personnel airlock door	2.7	20
039 Main and Reheat Steam											X	2.4.4 - Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.5	21
012 Reactor Protection											X	2.2.12 - Equipment Control: Knowledge of surveillance procedures.	3.7	22
064 Emergency Diesel Generator											X	2.1.19 - Conduct of Operations: Ability to use plant computers to evaluate system or component status.	3.9	23
005 Residual Heat Removal	X											K1.04 - Knowledge of the physical connections and/or cause-effect relationships between the RHRS and the following systems: CVCS	2.9	24
003 Reactor Coolant Pump									X			A3.04 - Ability to monitor automatic operation of the RCPS, including: RCS flow	3.6	25
013 Engineered Safety Features Actuation		X										K2.01 - Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control	3.6	26
004 Chemical and Volume Control						X						K6.05 - Knowledge of the effect of a loss or malfunction on the following CVCS components: Sensors and detectors	2.5	27
061 Auxiliary/Emergency Feedwater					X							K5.02 - Knowledge of the operational implications of the following concepts as the apply to the AFW: Decay heat sources and magnitude	3.2	28
K/A Category Totals	3	3	1	3	3	3	2	2	3	2	3	Group Point Total:		28

TMI
Written Examination Outline
Plant Systems - Tier 2 Group 2

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
015 Nuclear Instrumentation		X										K2.01 - Knowledge of bus power supplies to the following: NIS channels, components, and interconnections	3.3	29
045 Main Turbine Generator	X											K1.06 - Knowledge of the physical connections and/or cause-effect relationships between the MT/G system and the following systems: RCS, during steam valve test	2.6	30
068 Liquid Radwaste					X							K5.04 - Knowledge of the operational implication of the following concepts as they apply to the Liquid Radwaste System: Biological hazards of radiation and the resulting goal of ALARA	3.2	31
041 Steam Dump/Turbine Bypass Control							X					A1.02 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure	3.1	32
016 Non-Nuclear Instrumentation			X									K3.02 - Knowledge of the effect that a loss or malfunction of the NNIS will have on the following: PZR LCS	3.4	33
001 Control Rod Drive									X			A3.02 - Ability to monitor automatic operation of the CRDS, including: Rod height	3.7	34
029 Containment Purge										X		A4.04 - Ability to manually operate and/or monitor in the control room: Containment evacuation signal	3.5	35
034 Fuel Handling Equipment								X				A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Mispositioned fuel element	3.3	36
072 Area Radiation Monitoring											X	2.4.50 - Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	4.2	37
035 Steam Generator						X						K6.03 - Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: S/G level detector	2.6	38

TMI

Written Examination Outline

Plant Systems - Tier 2 Group 2

System #/Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
K/A Category Totals	1	1	1	0	1	1	1	1	1	1	1	Group Point Total:		10

Facility: TMI Date:

Category	K/A #	Topic	RO IR	RO Q#	SRO-Only IR	SRO-Only Q#
1. Conduct of Operations						
	2.1.28	Knowledge of the purpose and function of major system components and controls.	4.1	66		
	2.1.13	Knowledge of facility requirements for controlling vital / controlled access.	2.5	67		
	Subtotal			2		
2. Equipment Control						
	2.2.39	Knowledge of less than or equal to one hour technical specification action statements for systems.	3.9	68		
	2.2.21	Knowledge of pre- and post-maintenance operability requirements.	2.9	69		
	Subtotal			2		
3. Radiation Control						
	2.3.11	Ability to control radiation releases.	3.8	70		
	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.4	71		
	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9	75		
	Subtotal			3		
4. Emergency Procedures / Plan						
	2.4.29	Knowledge of the emergency plan.	3.1	72		
	2.4.25	Knowledge of fire protection procedures.	3.3	73		
	2.4.28	Knowledge of procedures relating to a security event (non-safeguards information).	3.2	74		
	Subtotal			3		
Tier 3 Point Total				10		7

Three Mile Island
2010 NRC Initial License Written Examination
Written Examination Outline Methodology

The written examination outline was developed using a proprietary electronic random outline generator developed by Western Technical Services, Inc.

The software was designed to provide a written examination outline in accordance with the criteria contained in NUREG 1021, Revision 9, Supplement 1.

The application was developed using Visual Basic code, relying on a true random function based on the PC system clock. The random generator selects topics in a Microsoft Access Database containing Revision 2, Supplement 1 of the PWR K&A catalogue. The selected data is then written to a separate data table. The process for selection of topics is similar to the guidance in ES-401, Attachment 1.

The attached outline report and plant specific suppression profile report are written directly from the data tables created by the software. Electronic copies of the data tables are on file.

The process used to develop the outlines is as follows:

- For Tier 1 and Tier 2 generic items, only the items required to be included in accordance with ES-401, Section D.1.b are included in the generation process.

- The TMI plant suppression profile lists all suppressed topics, either at the Topic level (System/EPE) or at the statement level. These items were suppressed prior to the electronic generation process. *(Note: Topics were not pre-suppressed for TMI)*

- Outline is generated for all topics with KA importance ≥2.5.

- Typically, 25 SRO topics are randomly selected from Tier 1 AA2 and required generic items, Tier 2 A2 and required generic items, and Tier 3 generic items (All with ties to 10CFR55.43). 75 RO topics are randomly selected to complete the outline, 100 topics total. *(Note: This exam contains only 75 RO items)*

- The exam report generated lists the topic (Question) number in the far right column. RO topics are numbered 1-75, and SRO topics are numbered 76–100. The SRO topics are written in red ink for ease of identification. *(Note: This exam contains only 75 RO items)*

- Items that are rejected after the initial generation process are automatically placed on the rejected items page. The software tracks whether items are added manually or by random generation, and a report of outline modification may be generated.

- Disposition of any item randomly selected but not included in the outline is documented and included.

Tier / Group	Randomly Selected KA	Reason for Rejection
1 / 2	A03 /AA2.1 replaced by E08 / EA2.1	EAPE A03 was inadvertently selected twice in T1 G2.
2 / 1	012 / 2.2.4 replaced by 012 / 2.2.12	The subject K/A isn't relevant at the subject facility. (Multi-Unit Generic topic)
2 / 1	078 /K3.03 replaced by 078 /K4.02	The subject KA isn't relevant because there is only one applicable unit.
2 / 2	028 /K3.01 replaced by 016 /K3.02	The subject KA is from a system that exists but is no longer used at the facility or required by facility EOPs

ADAMS MASTER EXAM FILE PACKAGE

ADAMS DOCUMENT COVER SHEET

DOCUMENT TITLE: TMI - RETAKE - Draft Written Exam

ESTIMATED PAGE COUNT:	______
AVAILABILITY:	**Public**
DOCUMENT SENSITIVITY:	**Non-Sensitive**
KEYWORD:	**NRR-079, SUNSI Review Complete**
ADAMS Package Accession #	**ML092470034**

SECURITY:	**Access Level:**
R1-DRS-BS	**Owner**
NRC Users	**Viewer**
General Users	**Viewer**
DPC	**Owner**
CJB1	**Owner**

Document Accession No: ML___________

SITE:	TMI - RETAKE
Exam DATES:	04/12/10
Chief Examiner:	J. D'Antonio
TAC NO:	U01820

ADAMS Profiled & entered into DPC on _________ by _____

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K1.10	
	Importance Rating	2.6	

Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: Safety injection tank heating system

Proposed Question: RO Question # 1

BWST heaters automatically operate to maintain BWST temperature above the Technical Specification minimum temperature of ____(1)____ and are automatically tripped at a level setpoint of ____(2)____.

A. (1) 45°F
(2) 52 feet

B. (1) 45°F
(2) 6.3 feet

C. (1) 40°F
(2) 52 feet

D. (1) 40°F
(2) 6.3 feet

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. Plausible because the temperature is the normal lower temperature for the operation of the BWST heaters. (Operate to ensure minimum 45-55°F) Second part is plausible because below 52 feet, heaters must be operated manually because the temperature sensor will be uncovered. Therefore, both values will be familiar to the applicant.

B. **Incorrect.** 1st part wrong, 2nd part correct. First part plausible same reason as option A. Second part is correct. At 6.3 feet, heaters trip on low-low-level to prevent damage due to uncover.

C. **Incorrect.** 1st part correct, 2nd part wrong. TS 3.3.1.1.a requires minimum temperature of 40°F for BWST. Second part is plausible for same reason as option A.

D. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-212-C001 (p13-14; Rev 7) electric heaters and heat trace automatically control temperature of tank contents. According to TQ-TM-104-212-C001 (p13-14, 38, 41-42 and 50; Rev 7), the BWST immersion heaters are normally operated automatically and cycle between 45-55°F. However, Technical Specifications requires a minimum temperature of 40°F. According to TQ-TM-104-212-C001, the heaters operate automatically when level is above 52 feet, required to be operated manually when level is below 52 feet, and trip on lo-lo level when level reaches 6.3 feet.

Technical Reference(s): TQ-TM-104-212-C001 (p13-14, 38, 41-42, 50 and 93; Rev 7) (Attach if not previously provided)
Technical Specification 3.3.1.1.a
OP-TM-MAP-E0104 rev 2 step 3.0

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-GLO-2, 6, 9 and 10 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of the physical connections (i.e. Height of the BWST Temperature Detector and the mode of operation under varying levels between the ECCS and the Safety injection tank heating system.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	026	K1.02	
	Importance Rating	4.1		

Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: Cooling water

Proposed Question: RO Question # 2

Plant conditions:

- Reactor trip due to LOCA.
- RCS pressure is 1400 psig and lowering.
- Reactor Building pressure is 22 psig and rising.
- All equipment is operating as designed.

Which ONE (1) of the following describes the status of (1) Reactor Building Spray Pumps, and (2) Decay Heat CCW and River Water Pumps (DC/DR-P-1A & B)?

A. (1) Running
(2) Started on ES Block 1

B. (1) Running
(2) Started on ES Block 3

C. (1) NOT Running
(2) Started on ES Block 1

D. (1) NOT Running
(2) Started on ES Block 3

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** Plausible because a LOCA is in progress and all ES actuations have occurred with exception of Building Spray initiation. BS Pumps would not have started at this point. However, all other ES equipment is running. Additionally, DC/DR Pumps have received an AUTO start signal, but start on ES Block 3, not ES Block 1.

B. **Incorrect.** Plausible because DC/DR Pumps start on ES Block 3. Also plausible because ES Actuations have occurred, EXCEPT for Building Spray Pump actuation

C. **Incorrect.** Plausible because Building Spray status is correct, and also because the applicant may not know which ES Block the DC/DR Pumps started on. Wrong for same reason as option A, because ES Block 1 is incorrect.

D. **Correct.** DC/DR Pumps will have started on either the 4 # RB ES Actuation, or the 1600# RCS pressure ES actuation. They start on ES Block 3. Reactor Building Spray Pumps are not started as they have not received the actuation signal.

Technical Reference(s):	TQ-TM-104-214-C001 (p25; Rev 6) TQ-TM-104-533-C001 (p28; Rev 4) TQ-TM-104-642-C001 (p29; Rev 4)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective:	214-GLO3 533-GLO1, 5, and 8 642-GLO5	(As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. what cause cooling water systems to start, how are the signals generated) of the cause-effect relationships between the CSS and its cooling water systems (i.e. DC/DR)

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	062	K2.01	
	Importance Rating	3.3		

Knowledge of bus power supplies to the following: Major system loads

Proposed Question: RO Question # 3

The Reactor is at 100%.

A plant transient occurs that causes the following plant response:

- MAP-AA-1-3, 4KV BOP FDR BKR TRIP
- MAP-AA-1-4, 4KV BOP MOTOR TRIP
- Various other Control Room alarms are in alarm
- Loss of CW-P-1B and CW-P-1E

Based on the above conditions and transient, which ONE (1) of the following components' normal power supply was lost?

A. Main Vacuum Pump, VA-P-1C

B. Instrument Air Compressor, IA-P-4

C. Primary Control Rod Drive Power Supply

D. Secondary Services Closed Cooling Water Pump, SC-P-1C

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because according to TQ-TM-104-331-C001 (p14; Rev 3) the Main Vacuum Pump, VA-P-1C, is powered from BOP 480 VAC Bus 1C. However, according to TQ-TM-104-731-C001 (p110-112; Rev 2), 480 VAC Bus 1C is normally supplied from BOP 4160 VAC Bus 1A, and can be backed up by 480 VAC Bus 1N.

B. **Incorrect.** This is plausible because according to TQ-TM-104-850-C001 (p25; Rev 1) the Instrument Air Compressor, IA-P-4, is powered from BOP 480 VAC Bus 1J. However, according to TQ-TM-104-731-C001 (p110-112; Rev 2), 480 VAC Bus 1C is normally supplied from BOP 4160 VAC Bus 1C, and can be backed up by 480 VAC Bus 1N.

C. **Correct.** According to MAP AA-1-3 (p1-2; Rev 6), a 4KV Breaker Tripping was required to generate the 4kV BOP FDR BKR TRIP, and according to MAP AA-1-4 (p1-2; Rev 5), the CW-P-1B and 1E Breakers tripping would have been sufficient to cause the 4kV BOP MOTOR TRIP Alarm. These alarms indicate that 4160 Volt Bus 1B has experienced an electrical fault of some kind. TQ-TM-104-731-C001 (p111; Rev 2), The 4160 Turb. Plant Bus 1B supplies electrical power to 480 Volt switchgear 1F, 1G, 1H, and 1U. According to TQ-TM-104-622-C001 (p28; Rev 4), The CRD System is powered from two BOP 480V buses (G & L).

D. **Incorrect.** This is plausible because according to TQ-TM-104-544-C001 (p20; Rev 3) the Secondary Services Closed Cooling Water Pump, SC-P-1C, is powered from 480 VAC Bus 1N.

Technical Reference(s): OP-TM-AOP-011 Rev 1 pg 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 622-GLO4 331-GLO4 850-GLO4 544-GLO4 (As available)

Question Source: Bank #

Modified Bank # IR-AOP-011-PCO-5-Q01 (Note changes or attach parent)

New

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of bus power supplies to Major system loads (i.e. Control Rod Drive, Vacuum Pump, IAS Compressor, and SSCCW Pump).

The question is at the Comprehension/Analysis cognitive level because the operator must assemble several pieces of information (i.e. that (1) an electrical fault has occurred associated with Bus 1B, that (2) that Bus 1B powers bus 1G, that (3) Bus 1G provides at least one half of the power to the Control Rod Drive System, and (4) the normal power supplies to the C Main Vacuum Pump, the Main IAS Air Compressor, and the C Secondary Side CCW Pump); and the draw a conclusion (i.e. that Control Rod Drive Power has been affected).

The Reactor is at 100% power with the following lineup:

- CO-P-1A and CO-P-1C are Operating
- CO-P-2A and CO-P-2C are Operating
- CO-P-1B is Out of Service for maintenance and expected to be returned to service in 6 hours.
- ULD is in Hand with ICS in Auto

A plant transient occurs that causes the following plant response:

- MAP-AA-1-3, 4KV BOP FDR BKR TRIP
- MAP-AA-1-4, 4KV BOP MOTOR TRIP, and various other control room alarms
- Loss of CW-P-1B and CW-P-1E

Based on the above conditions and transient, which one of the following components' normal power supply was lost?

A. Main Vacuum Pump, VA-P-1C

B. Instrument Air Compressor, IA-P-4

C. Primary Control Rod Drive Power Supply

D. Secondary Services Closed Cooling Water Pump, SC-P-1C

Answer: C

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	005	K2.01	
	Importance Rating	3.0		

Knowledge of bus power supplies to the following: RHR pumps

Proposed Question: RO Question # 4

During surveillance testing, if DH-P-1A should trip, where would an AO be dispatched to verify an adequate source of power for this pump?

A. 1P 480v ES Bus.

B. 1S 480v ES Bus.

C. 1D 4160v ES Bus.

D. 1E 4160v ES Bus.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may incorrectly believe that the DH Pumps are 480 VAC, rather than 4160 VAC. If so, according to TQ-TM-104-701-C001 (p6-7; Rev 3), the 1P 480v ES Bus is powered from the A Train of ES Equipment.

B. **Incorrect.** This is plausible because the operator may incorrectly believe that the DH Pumps are 480 VAC, rather than 4160 VAC; AND that the A Train of equipment is powered from the 1E 4160 ES Bus train. If so, according to TQ-TM-104-701-C001 (p6-7; Rev 3), the 1S 480v ES Bus is powered from the 1E 4160 ES Bus Train of ES Equipment.

C. **Correct.** According to TQ-TM-104-212-C001 (p28; Rev 7), the Decay Heat

Removal Pumps (DH-P-1A/1B) are powered by the 1D 4160 VAC ES Bus and 1E 4160 VAC Bus respectively.

D. **Incorrect.** This is plausible because according to TQ-TM-104-701-C001 (p6-7; Rev 3) there are two trains of 4160 ES equipment, and the operator may correctly believe the pumps to be powered from 4160 VAC, rather than 480 VAC, but incorrectly believe that they are powered from the 1E, rather than the 1D, 4160 ES Bus Train of ES Equipment.

Technical Reference(s): TQ-TM-104-212-C001 (p28; Rev 7) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-GLO4 (As available)

Question Source: Bank #

Modified Bank # OA-534-GLO-4-Q02 (Note changes or attach parent)

New

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of bus power supplies to the DHR (RHR) pumps (i.e. 4160 VAC vs. 480 VAC; 1D ES Train vs. 1E ES Train).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. which Electrical Bus powers A Train equipment, what electrical voltage is needed to run the DH Pump motors).

The modification of this question included changing the operational conditions (surveillance vs. emergency) and the Pumps.

Exam Bank Searches:
Section 34 quest for 34005K201: None
(175) IR-424-GLO-4-Q02 (Modify)
(176) IR-424-GLO-4-Q03 (Modify)
(206) OA-534-GLO-4-Q02 (Modify) (Selected)

During RB Emergency Cooling operations, if RR-P-1A should trip, where would an AO be dispatched to verify an adequate source of power for this pump?

A. 1E 4160v ES Bus.

B. 1T 480v Screen House ES Bus.

C. 1D 4160v ES Bus.

D. 1R 480v Screen House ES Bus.

Answer: C

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	022	K3.01	
	Importance Rating	2.9		

Knowledge of the effect that a loss or malfunction of the CCS will have on the following: Containment equipment subject to damage by high or low temperature, humidity, and pressure

Proposed Question: RO Question # 5

Plant conditions:

- Reactor is at 100% power.
- Both Reactor Compartment Cooling Fans (AH-E-2A/B) have failed.
- H&V Panel HVA-1-9, Reactor Compartment Air Temp Hi, is actuated.
- Highest Reactor Compartment air temperature is 300°F and steady.

Which ONE (1) of the following describes the concern associated with the elevated Reactor Compartment air temperature?

A. Reactor Compartment concrete structures are subject to immediate failure.

B. Reactor Compartment concrete structures are subject to degradation over a period of time.

C. Excore Nuclear Instrumentation detectors are subject to immediate failure.

D. Excore Nuclear Instrumentation detectors are subject to degradation over a period of time.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** According to TQ-TM-104-824-C001 (p47; Rev 3), AH-E-2A & 2B and associated cooling units are designed to limit local concrete temperature to under 150°F during normal operating conditions. Concrete temperatures above 300 °F can cause structural failure. According to OP-TM-HVA-109 (p3 of 3; Rev 0) sustained temperatures of > 300°F for > 1 week may have permanent degrading effects on Reactor Compartment concrete structures. There is no immediate concern for concrete degradation.

B. **Correct.** According to TQ-TM-104-824-C001 (p47; Rev 3), AH-E-2A & 2B and associated cooling units are designed to limit local concrete temperature to under 150°F during normal operating conditions. Concrete temperatures above 300 °F can cause structural failure. According to OP-TM-HVA-109 (p3 of 3; Rev 0) sustained temperatures of > 300°F for > 1 week may have permanent degrading effects on Reactor Compartment concrete structures.

C. **Incorrect.** This is plausible because according to TQ-TM-104-623-C001 (p5-6, 8; Rev 1) there are three different types of Nuclear Detectors in use at TMI; Compensated and Uncompensated Ion Chambers as well as Fission Chambers; and each are located outside the Reactor Vessel. The operator may incorrectly believe that the Excore Nuclear Detectors is affected by the elevated temperature. While according to TQ-TM-104-623-C001 (p44; Rev 1), while their associated electronic systems can be affected by electromagnetic radiation from portable communications equipment, and according to TQ-TM-104-623-C001 (p31-33; Rev 1) factors that change the flux leakage conditions; such as core reloads, power level changes, normal steady state operation, vessel downcomer water densities, and the expected shift in power distribution over core life, can affect Power Range calibration, there is no indication that elevated Reactor Compartment temperatures will have any effect on the nuclear instruments.

D. **Incorrect.** This is plausible because according to TQ-TM-104-623-C001 (p5-6, 8; Rev 1) there are three different types of Nuclear Detectors in use at TMI; Compensated and Uncompensated Ion Chambers as well as Fission Chambers; and each are located outside the Reactor Vessel. The operator may incorrectly believe that the Excore Nuclear Detectors is affected by the elevated temperature. While according to TQ-TM-104-623-C001 (p44; Rev 1), while their associated electronic systems can be affected by electromagnetic radiation from portable communications equipment, and according to TQ-TM-104-623-C001 (p31-33; Rev 1) factors that change the flux leakage conditions; such as core reloads, power level changes, normal steady state operation, vessel downcomer water densities, and the expected shift in power distribution over core life, can affect Power Range calibration, there is no indication that elevated Reactor Compartment temperatures will have any effect on the nuclear instruments.

Technical Reference(s):	TQ-TM-104-824-C001 (p47; Rev 3) OP-TM-HVA-109 (p3of3; Rev 0)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 824-GLO7 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. which component will be operationally limiting when both Reactor Compartment fans have failed) of the effect that a loss or malfunction of the CCS will have on Containment equipment subject to damage by high or low temperature, humidity, and pressure.

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. which component is operationally limiting in high temperature situation in RC, how components such as NIS Detectors are affected by RC high ambient temperature).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	078	K4.02	
	Importance Rating	3.2		

Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Cross-over to other air systems

Proposed Question: RO Question # 6

Plant conditions:

- 100% power.
- IA-P-4 Instrument Air Compressor trips for unknown reasons.
- IA-P-4 is removed from service and expected to be out of service for three days.

Which ONE (1) of the following identifies the NORMAL position of the Instrument Air to Service Air Cross Connect Valves, IA-V-2712/SA-V-232, AND their PRESENT position under existing plant conditions?

A. OPEN;
OPEN.

B. OPEN;
CLOSED.

C. CLOSED;
OPEN.

D. CLOSED;
CLOSED.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the valves must be open to provide a backup for the Instrument Air System. In actuality, the Service Air System is still providing a backup for the Instrument Air System through IA-V-1.

B. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-850 C001 (p19; Rev 1), the Instrument Air System maintains the Service Air System Receivers at pressure via IA-V-2712, IA-V-232 and a 3/16"hole in the disc of the outlet check valve on each receiver. The Service Air Compressors, SA-P-1A/B are normally in standby. According to TQ-TM-104-850 C001 (p20-21; Rev 1), The Service Air System can be used to backup the Instrument Air System automatically via IA-V-1 at 80 psig, however, the Service Air is NOT clean, dry, or oil free air, and therefore, is only used as a backup to Instrument Air when directed by the Control Room. According to TQ-TM-104-850 C001 (p61-62; Rev 1) and 1104-25 (p8; Rev 139), the cross connect valves may be used during all plant operating modes, and in fact, the preferred alignment is to have IA-P-4 supply both Instrument Air and Service Air while SA-P-1A/B remains in standby. This is done so that the operator can reduce run time and maintenance costs on SA-P-1A/B as well as make clean, dry, oil free air available in the Service Air System for breathing purposes if needed. However, whenever IA-P-4 is removed from service for extended periods (>1 shift), the cross- connect valves IA-V-2712/SA-V-232 should be closed in order to eliminate a potential path for wet and/or oil laden air from entering Instrument Air System.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because operator may incorrectly believe that since the SA system is not dry clean air, that the two systems are NOT normally cross tied (See D); AND because the operator may incorrectly believe that the valves must be open to provide a backup for the Instrument Air System (See A).

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because operator may incorrectly believe that since the SA system is not dry clean air, that the two systems are NOT normally cross tied. However, they are. There are other design features to ensure that wet unfiltered air is NOT placed directly into the IA System from the SA System.

Technical Reference(s): TQ-TM-104-850 C001 (p 61-62; Rev 1) (Attach if not previously provided)

1104-25 (p8; Rev 141)

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO3 and 9 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	4
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge of the effect that a malfunction of the IAS (i.e. loss of IA-P-4) will have on existing cross-tied units (i.e. IAS and SAS are normally cross-tied, but will be realigned due to the failure of IA-P-4).

The question is at the Comprehension/Analysis cognitive level because the operator must possess information that is non-intuitive such as (1) that the IAS and the SAS are normally cross-tied, and that (2) procedures direct that the cross tie be closed when IA-P-4 is OOS; and understand the implications of it, such as that (1) simply because the SA and the IA are cross tied, does NOT mean that wet unfiltered SA will contaminate the IAS, and that (2) the SA System even when not cross-tied through the stated valves provides backup for the IAS through the Standby IAS filter/dryers, and not directly into the IAS.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	063	K4.01	
	Importance Rating	2.7		

Knowledge of dc electrical system design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control

Proposed Question: RO Question # 7

Which ONE (1) of the following describes the status of the power transfer scheme for 1M DC Distribution Panel following an RCS 1600 psig ES actuation?

A. AUTOMATIC transfer is blocked but the MANUAL transfer is operable.

B. MANUAL transfer is blocked but the AUTOMATIC transfer is operable.

C. AUTOMATIC AND MANUAL transfer are blocked.

D. AUTOMATIC AND MANUAL transfer are operable.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may incorrectly believe that the manual transfer is still operable, especially since according to OP-TM-211-000 (p15; rev 19) the power source to the 1M panel must be manually selected to ensure HPI operability.

B. **Incorrect.** This is plausible because the operator may incorrectly believe that the automatic transfer is still operable, especially since According to TQ-TM-104-642-C001 (p4; Rev 4), the ESAS is a system that *automatically* actuates core cooling for the RCS, isolates the RB from the environment, actuates RB emergency cooling, actuates RB Spray and responds to a loss of electrical power to 4160 VAC ES buses to ensure that the plant response to transient conditions are consistent with the Accident Analyses contained in the TMI-1 UFSAR.

C. **Correct.** According to OPM A-03 (p4; Rev 12) the 1M DC Bus has an auto transfer switch to auto swap power supplies on a loss of power to the selected bus; DC Panel 1A or 1B. According to TQ-TM-104-642-C001 (p39-41, 57; Rev 4) the ESAS prevents auto/manual transfer of DC Bus 1M during times when the ESAS is followed by Under Voltage, and Under Voltage is followed by an ESAS, or when the two occur simultaneously.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that manual transfer capability is always available, and that the interlocks associated with the 1M Bus during an ESAS enables the auto transfer so that all equipment remains available. (NOTE: This is NOT a use of "None of the Above" which is prohibited by NUREG-1021 for two reasons: (1) it is obviously not a direct use since the words "None of the Above" are NOT used, (2) this is not an implied use, because the distractor has plausibility).

Technical Reference(s): TQ-TM-104-642-C001 (p40; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO5 (As available)

Question Source: Bank # IR-642-GLO-4-Q01

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: TMI Feb 2000

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. how the interlock functions with an ESAS) of a dc electrical system interlock which provides for the following the manual and automatic transfer of control of DC Bus 1M.

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. Manual Transfer cannot be made with an ESAS active, automatic transfer cannot be made with an ESAS active).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	003	K4.07	
	Importance Rating	3.2		

Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Minimizing RCS leakage (mechanical seals)

Proposed Question: RO Question # 8

The plant is operating at 100% power.

Which ONE (1) of the following describes the normal leak-off flow rate for the RCP #1 Seal, AND the normal leak-off flow path (i.e. where does the flow go from the #1 seal)?

A. 3 gpm; AND
To the #2 Seal ONLY.

B. 3 gpm; AND
To the Makeup Tank and #2 Seal.

C. 5 gpm; AND
To the #2 Seal ONLY.

D. 5 gpm; AND
To the Makeup Tank and #2 Seal.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the leakoff flow passes to the #2 Seal, through it, and then on to the Makeup Tank. This would be the case if the operator believed that the #2 seal was a film riding seal like the #1 seal, and NOT a rubbing face seal (i.e. mechanical seal).

B. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-226-C001 (p9-10; Rev 4), the normal seal leakoff flow for the #1 seal is 3 gpm. According to TQ-TM-104-226-C001 (p11; Rev 4), the leakage flow path from the #1 seal flows upward along the shaft to the #2 seal area at which point most of the water exits through #1 seal leakoff to be returned to Makeup Tank.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that 5 gpm is normal flow (See D). This is plausible because the operator may incorrectly believe that the leakoff flow passes to the #2 Seal, through it, and then on to the Makeup Tank (See A).

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that 5 gpm is normal flow. After all, it is within the range of the instrumentation and observable to the operator. However, according to TQ-TM-104-226-C001 (p33; Rev 4), the MCB Annunciator for RCP #1 Leak Off High Flow alarms at 5 gpm. Therefore, this leakoff flow would be considered abnormally high flow.

Technical Reference(s): TQ-TM-104-226-C001 (p11; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO (As available)

Question Source: Bank #

Modified Bank # TMI: OA-226-GLO-2-Q04 WTS: 60881 (Note changes or attach parent)

New

Question History: Last NRC Exam: WTS Point Beach 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. design of the seal package, normal #1 seal leakoff Flowrates and flow paths) of RCPS design feature(s) which provide for minimizing RCS leakage (mechanical seals).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. normal #1 seal leakoff Flowrates and flow paths, the #1 Seal is a film riding seal and the #2 seal is a rubbing face seal).

The modification of this question included taking the bank Question and changing it from the #2 seal to the #1 seal; and then incorporating the WTS question into it.

Exam Bank Searches:
Section 32 quest for 32002K407: None
Section 34 quest for 34003K407: None
(8) OA-226-GLO-2-Q04 (Modify) Selected

Which of the following is the normal leak off flow rate for RCP Seal #2?

A. 0 GPM

B. 3 GPM

C. 3 GPH

D. 5 GPM

Answer: C

WTS 60881 Point Beach 2007

Which of the following describes the RCP #1 seal leak-off flow path at 100% Reactor Power?

A: #2 Seal Only

B: VCT and #2 Seal

C: VCT and Standpipe

D: #2 Seal and Standpipe

Proposed Answer: B

The modification included a combination of both the Bank Question (modified for seal #1), and the WTS Question.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	004	K5.37	
	Importance Rating	2.6		

Knowledge of the operational implications of the following concepts as they apply to the CVCS: Effects of boron saturation on ion exchanger behavior

Proposed Question: RO Question # 9

Boron concentration in the reactor coolant system has been lowering steadily at approximately 10 ppm per hour while using the deborating demineralizer.

After several hours, the rate of boron reduction lowered to 2 ppm per hour.

Which ONE (1) of the following statements best explains a possible cause for the change in deboration rate?

A. Temperature of the reactor coolant passing through the demineralizer has lowered.

B. pH of the reactor coolant system has risen significantly.

C. Flow through the deborating resins has risen sharply.

D. The deborating resins are becoming boron saturated.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OPM N-9 (p193-5; Rev 4) higher temperatures result in higher resin film diffusion rates, which is directly proportional to the rate of ion exchange. Therefore, a decrease in demineralizer temperature will result in a slower rate of exchange and the resin holding more boron. However, the deborating process is a constant temperature process, and lowering temperature is NOT expected over several hours. Therefore, lowering temperature is NOT the best explanation for the change in boration rate.

B. **Incorrect.** This is plausible because according to OPM N-9 (p96-97; Rev 4) an increased pH means an increase in the number of Hydroxyl (OH^-) in the coolant, and according to TQ-TM-104-561-C001 (p63-64; Rev 2) it is the OH^- ions that react with the Boric Acid ions in the deborating Ion Exchanger. Therefore, the operator may incorrectly believe that an increased pH will result in a lower rate of ion exchange. However, because of the simultaneous use of mixed bed demineralizers, the deborating process is a constant pH process, and an increase in pH is NOT expected over several hours. Therefore, an increase in pH is NOT the best explanation for the change in boration rate.

C. **Incorrect.** This is plausible because according to OPM N-9 (p193-5; Rev 4) increased flow rates result in decreased contact time between the resin and the reactor coolant, which result in lower resin film diffusion rates, which is directly proportional to the rate of ion exchange. Therefore, an increase in demineralizer flow will result in a slower rate of exchange and the resin holding more boron. However, the deborating process is a constant flow process, and an increase in flow is NOT expected over several hours. Therefore, increased flow is NOT the best explanation for the change in boration rate.

D. **Correct.** According to TQ-TM-104-211-C001 (p22; Rev 3) anion resin is used to remove boron ions from the RCS. TQ-TM-104-561-C001 (p63-64; Rev 2) the anion resin is originally in the hydroxide form. When exposed to borated reactor coolant the resin is transformed to the borate form by the removal of boric acid from the reactor coolant. The boric acid will chemically bond to the resin as well as being absorbed by the resin beads. These reactions continue until the resin becomes saturated and an equilibrium is established between the resin and the reactor coolant water. Until the resin reaches equilibrium, the demineralizer effluent boron concentration will be lower than the reactor coolant system (causing a decrease in RCS boron concentration). As the process continues it would be expected that the rate of boron concentration reduction is lowered.

Technical Reference(s):	TQ-TM-104-211-C001 (p23; Rev 3) TQ-TM-104-561-C001 (p64; Rev 2) OPM N-9 (p96-97, 193-5; Rev 4)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO10 (As available)

Question Source:	Bank #	IR-211-GLO-3-Q01	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Old Question ID SR4A09-21-Q01 Migrated from TMI OPS 8/05	Last NRC Exam: NA	

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. what is the normal or expected mode of operation for the deborating Ion Exchanger) of the operational implications of the effects of boron saturation on ion exchanger behavior as they apply to the CVCS.

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. The Deborating Ion Exchanger will remove boron at constantly decreasing rates over time as it operates).

Exam Bank Searches:
Question Preview 31 for 31004K537:
(44) IR-211-GLO-3-Q01 (Selected)
Section 32 quest for 32004K537: None

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 K5.02	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR

Proposed Question: RO Question # 10

Plant conditions:

- 1103-11, RCS Water Level Control, is controlling procedure preparing for RCS heatup.
- RCS final fill operation has been completed, with flow into the RCS terminated.
- Pressurizer level is 390 inches.
- Pressurizer temperature is 190 degrees F.
- Reactor Coolant Drain Tank (RCDT) pressure is 2.0 psig.
- Operator energizes Pressurizer heaters to form a steam bubble in the Pressurizer.

Event, 60 minutes later:

- RCS Pressure indication reaches 22 psig.
- RCDT level begins to rise.

Based on these conditions, which ONE (1) of the following describes the source of the water flowing into the RCDT?

A. Hot leg vent(s).

B. Pressurizer vent.

C. RCP Seal Standpipe(s).

D. Center Control Rod Drive mechanism vent.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** RCDT will begin to rise when pressure is high enough to overfill one or both hot legs (flow will initiate out the hot leg vent(s). According to Steps 3.3.2.11-15 of 1103-11 (p17-21, 49 Rev 67), when Pzr level is 390 ±2 inches RCS fill is terminated, and the manual Pzr vent valves are closed. When Pzr temperature has been > 230F for > 30 minutes, the Pzr Vent valves to the RCDT Sparger (RC-V-28 and 44) are closed. With a steam bubble in the Pzr, maintaining RCS pressure at 22 psig water will be forced out of the Pzr and into the RCS causing water to issue from the Hot Leg vents (RC-V15A/B, RC-V46A/B and RC-V-14A/B) and into the RDCT. At this point a Steam Bubble exists in the Pzr.

B. **Incorrect.** This is plausible because the Pzr vents to the RCDT, and the Pzr manual vent valves are open during the RCS filling process, even at the point that the Pzr heaters are energized (Steps 10-11 of 1103-11).

C. **Incorrect.** This is plausible because the RCP Seal standpipes overflow and drain to the RCDT. However, flow from the RCP standpipes is not possible at these conditions, and the standpipe bypass valves, RC-V-33A-D, are closed in accordance with OP-TM-220-000 (p33; Rev 13).

D. **Incorrect.** This is plausible because the CRD venting system vents to the RCDT. However, In accordance with 1103-11, Enclosure 3A, (p5of 6; Rev 67), the CRDM are vented and closed when the RCS is being filled, and Pzr level is between 280-360 inches. This would have already occurred at this point in the procedure.

Technical Reference(s):	1103-11 (p17-21, 49 Rev 67) 1103-11, Enclosure 3A, (p5of 6; Rev 67) OP-TM-220-000 (p33; Rev 13)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-12-PCO4 (As available)

Question Source:	Bank #	IR-GOP-012-PCO-4-Q01	
	Modified Bank #		(Note changes or attach parent)

New

Question History: NA   Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. indications of a steam bubble in the Pzr such as vent valves closed, 20-25 psig in the RCS and water issuing from the high point vents) of the operational implications of the method of forming a steam bubble in the PZR as it applies to PRTS.

The question is at the Comprehension/Analysis cognitive level because the operator must understand the process of drawing a bubble in the Pzr (i.e. with water level above the heaters but below the top of the Pzr, the heaters are energized, the water is brought to saturation and boiling occurs, and then the Pzr Vents are closed), and then relate it to its consequence (i.e. once the vents are close, water will back up into the system and ultimately issue from the high point, which must be known as well).

Exam Bank Searches:
Question Preview 35 for 35007K502:
(6) IR-GOP-012-PCO-4-Q01 (Selected)

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	010	K6.03	
	Importance Rating	3.2		

Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PZR sprays and heaters

Proposed Question: RO Question # 11

Plant conditions:

- Plant is at 100% power.
- The Pressurizer Spray Valve (RC-V-1) is CLOSED and will NOT OPEN.
- All Pressurizer Heaters are de-energized, attempts to re-energize have failed.

The crew enters OP-TM-AOP-043, Loss of Pressurizer (Solid Ops Cooldown), due to loss of Pressurizer Heaters.

Present plant conditions:

- RCS Subcooling is 38°F.
- Pressurizer Level is 218".

Which ONE (1) of the following identifies actions that should be taken within this procedure under the present conditions to control RCS pressure?

A. Close the Spray Valve Block Valve (RC-V-3); AND
Place MU-V-17 to hand to maintain SCM > 40°F.

B. Place MU-V-17 to hand to maintain SCM > 40°F; AND
Initiate a plant shutdown and cooldown.

C. Isolate Letdown; AND
Initiate a plant shutdown and cooldown.

D. Isolate Letdown; AND
Close the Spray Valve Block Valve (RC-V-3).

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1^{st} part correct, 2^{nd} part correct. According to OP-TM-AOP-0431 (p3; Rev 2), this procedure attempts to maintain Pzr pressure control with a steam bubble as long as possible, while attempting to correct the problem. While this includes raising Pzr level to squeeze the bubble to maintain RCS pressure, this strategy is limited. According to OP-TM-AOP-0431 (p3; Rev 2), Step 3.5, the operator is directed to close RC-V-3 so that the spray bypass flow stops maximizing the amount of time that the steam bubble is maintained. Also, according to Step 3.7, the operator is directed to place MU-V-17 in hand and raise makeup flow to maintain > 40°F of SCM. As makeup flow is increased, Pzr level will increase, and squeeze the steam bubble, raising Pzr pressure.

B. **Incorrect.** 1^{st} part correct, 2^{nd} part wrong. This is plausible because the operator may incorrectly believe that a plant shutdown and cooldown should be initiated under the present conditions. According to OP-TM-AOP-0431 (6; Rev 2), Step 3.3, when Pzr level gets to 315" a plant shutdown will be initiated. According to OP-TM-AOP-0431 (8; Rev 2), Step 3.11, the steam bubble in the Pzr will remain until the fluid in the Pzr cools to the hot leg temperature. While shutdown and cooldown are strategies incorporated into the total mitigation, these activities come well after taking steps to prolong steam bubble operation.

C. **Incorrect.** 1^{st} part wrong, 2^{nd} part wrong. This is plausible because the operator may incorrectly believe that a plant shutdown and cooldown should be initiated under the present conditions (See B), and because the operator may incorrectly believe that isolating Letdown will assist in retaining a steam bubble (See D).

D. **Incorrect.** 1^{st} part wrong, 2^{nd} part correct. This is plausible because the operator may incorrectly believe that isolating Letdown will assist in retaining a steam bubble. If Letdown is isolated, while RCP Seal flow is allowed to continue, overall RCS inventory will increase and tend to compress the steam bubble. However, the strategy for compressing the steam bubble must be based on SCM, and not directly upon Pzr Level.

Technical Reference(s): OP-TM-AOP-0431 (p7; Rev 2)
OP-TM-AOP-043 (p3; Rev 2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	220-GLO10		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	NA	Last NRC Exam: NA	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. steam bubble will diminish over time, and the actions that the operator can take to lengthen the diminishing time) of the effect of a loss of the PZR sprays and heaters will have on the PZR PCS.

The question is at the Comprehension/Analysis cognitive level because the operator must assess each action, and its effectiveness given the set of stated conditions, and then determine which actions are effective from three that are clearly effective throughout the entire strategy (i.e. control makeup flow to maintain SCM, close the Spray valve Block Valve, and initiate a plant shutdown and cooldown; and one that while ineffective overall, has a variation (i.e. controlling letdown flow) that is effective in the ultimate strategy of controlling plant pressure while solid).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061	K6.02	
	Importance Rating	2.6		

Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps

Proposed Question: RO Question # 12

Plant conditions:

- Emergency Feedwater Pump EF-P-1 OOS.
- OTSG B Tube Rupture occurs while at 100% power.
- Reactor tripped.
- Subcooling Margin was lost and OP-TM-EOP-010 Rule 1, "Loss of Subcooling Margin (SCM)" was initiated.
- Loss of offsite power (LOOP) with a failure of the EG-Y-1B Emergency Diesel to start.
- Emergency Feedwater Pump EF-P-2A is feeding both OTSGs as follows:
  - OTSG A - 312 gpm
  - OTSG B - 210 gpm

In accordance with OP-TM-EOP-010 Rule 4, "FWC Feedwater Control", which ONE (1) of the following actions is required?

A. Stop feeding OTSG "B" and throttle Emergency Feedwater Control Valve EF-V-30A to feed "A" OTSG at between 430 gpm and 515 gpm

B. Stop feeding OTSG "B" and fully open Emergency Feedwater Control Valve EF-V-30A to feed OTSG "A" at the maximum available rate

C. Throttle Emergency Feedwater Control Valves EF-V-30A and EF-V-30B to feed each OTSG at >215 gpm each

D. Throttle Emergency Feedwater Control Valves EF-V-30A and EF-V-30B such that total emergency feedwater flow to both OTSG's is greater than 430 gpm

Proposed Answer: A

Explanation (Optional):

A. **Correct**. According to TQ-TM-104-424-C001 (p6; Rev 6), there are three EFW Pumps; the steam driven pump (EF-P-1), and two motor driven pumps (EF-P-2A and EF-P-2B). In the stated condition only the A Train Motor Driven EFW Pump is operating (EF-P-2A). According to OP-TM-EOP-010 (p8; Rev 11) the operator must first consider whether or not EFW is actuated, and if so, as is the case in the stated conditions, verify that two or more EFW Pumps are operating. Upon determining that only one EFW Pump is operating the operator will implement the Step 1 RNO. This step requires that if only one EFW Motor Driven Pump is operating, flow to the OTSGs must be maintained < 515 gpm. According to OP-TM-EOP-0101 (p20; Rev 3) Step 1 of Rule 4 is intended to prevent runout in the event only one EF-P-2 is operating. According to Step 2 of OP-TM-EOP-010 (p8; Rev 11) the operator must then check whether or not SCM is > 25°F or the OTSG level is between 75-85% Operating Range Level. The stated conditions indicate that SCM is < 25°F, and the B OTSG has a Tube Rupture, so level criteria will NOT be met requiring the operator to implement the Step 2 RNO. The applicable "IF" statement in the Step 2 RNO states that if only one OTSG is available or OTSG tube leakage exists, then feed with EFW > 430 gpm to the good OTSG and end the procedure.

B. **Incorrect**. This is plausible because the operator may be unaware of the runout prevention criteria in Step 1. If there were more than one EFW Pump running, this would be a correct answer.

C. **Incorrect**. This is plausible because the operator may confuse the strategies when SCM is < 25°F. If there were no tube leakage (i.e. < 1 gpm) identified, this would be the correct strategy.

D. **Incorrect**. This is plausible because the operator may be unaware of the strategy to stop feeding an OTSG that is ruptured. According to OP-TM-EOP-0101 (p21; Rev 3) the minimum total EFW flow necessary to maintain a heat sink is 430 gpm. The operator may incorrectly believe that the safe strategy is to maintain a total of 430 gpm regardless of the OTSG Tube status.

Technical Reference(s): TQ-TM-104-424-C001 (p6; Rev 6) (Attach if not previously provided)

OP-TM-EOP-010 (p8; Rev 11)

TM-EOP-0101 (p20-21; Rev 3)

Proposed References to be provided to applicants during examination: None

Learning Objective:	424-GLO11		(As available)
Question Source:	Bank #	IR-EOPR4-PCO-4-Q04	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:		Last NRC Exam: TMI 2007 Q19	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	8	
	55.43		

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. when only one pump is available flow must be limited to prevent pump runout) of the effect of a loss or malfunction of the EFW Pumps will have on the AFW components.

The question is at the Comprehension/Analysis cognitive level because the operator must assemble at least five pieces of information (i.e. that (1) that there is only one EFW Pump operating, that (2) one OTSG has a Tue Rupture, that (3) SCM is < 25°F, that (4) there are EFW pump runout prevention strategies that come in to play with only one EFW Pump Running, and that (5) the operator must not feed a OTSG with > 1 gpm Tube Leakage); and draw a conclusion about how adjust EFW flow in a post-trip situation (i.e. limit flow to the good OTSG and maintain flow > 430 gpm but < 515 gpm).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	073	A1.01	
	Importance Rating	3.2		

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels

Proposed Question: RO Question # 13

Plant conditions:

- Reactor Building Purge in progress.
- MAP C-1-1 RADIATION LEVEL HI alarm is received.
- The operator observes that RM-A-9 (Reactor Building Purge) is in HI alarm.

In addition to automatically isolating the RB Purge, which ONE (1) of the following identifies a component, or set of components, that will change state?

A. WDL-V-534 and WDL-V-535 (RB sump drain isolation) will CLOSE, ONLY.

B. The RB Evacuation Alarm will sound, ONLY.

C. The MAP-5 Iodine Sampler will start; AND
WDL-V-534 and WDL-V-535 (RB sump drain isolation) will CLOSE.

D. The MAP-5 Iodine Sampler will start; AND
The RB Evacuation Alarm will sound.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the RB sump drain isolation valves will close, and the operator may be unaware that this monitor starts the MAP-5 Iodine Sampler (There are only four atmospheric monitors that do; A-5, A-8 and A-15).

B. **Incorrect.** This is plausible because according to OP-TM-MAP-C0101 (p18; Rev 0), the operator must evacuate all personnel from Containment. Additionally, according to TQ-TM-104-623-C001 (p28; Rev 1) the Source Range (NI-11 and NI-12) instrumentation will automatically actuate the RB Evacuation alarm, therefore, the operator may incorrectly believe that the RB Evacuation Alarm is automatically actuated by RM-A-9 as well.

C. **Correct.** According to TQ-TM-104-661-C001 (p54; Rev 3), when RM-A-9 enters the alarm conditions three automatic functions will occur; (1) the RB sump drain valves, WDL-V-534 and 535 will CLOSE, (2) the RB Purge Valves, AH-V-1A/1B/1C and 1D will CLOSE, and (3) the MAP 5 Iodine Sampler will start.

D. **Incorrect.** This is plausible because the MAP-5 Iodine Sampler will start, and the operator may incorrectly believe that the RB Evacuation Alarm is automatically actuated by RM-A-9 (See B).

Technical Reference(s):	TQ-TM-104-661-C001 (p54, 55; Rev 4) OP-TM-MAP-C0101 (p17, 18; Rev 1)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO2, 5 (As available)

Question Source: Bank #

Modified Bank # WTS 60125 (Note changes or attach parent)

New

Question History: Last NRC Exam: Oconee 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to monitor changes in parameters such as Radiation levels (to prevent exceeding design limits) (i.e. Monitor enters the alarm condition) associated with operating the PRM system controls (i.e. ensuring automatic functions take place).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. which functions occur when a radiation monitor enters the alarm condition, and identify them correctly).

The modification of this WTS Bank question included modification from the Oconee plant to TMI.

Exam Bank Searches:
Question Preview 37 for 37073A101 – None
(90) **QR-661-GLO-8-Q02 (Modify this question to appear as the WTS Bank Question)**
WTS: 60125 Oconee 2007 can be modified for TMI and used as a Bank Question.
Modified Question

Plant Conditions:

- Plant Shutdown in progress to start a Refueling Outage.
- Currently Reactor power is 20% and lowering per the Shutdown Procedure.
- Reactor Building Purge has been started.

Event Occurrence:

- A small RCS leak has occurred.
- The Reactor Building Purge is automatically terminated.
- RCS pressure is 2100 psig and relatively steady.
- Reactor Building pressure is 1.35 psig and rising slowly.

Which of the following has caused the termination of the purge?

A. The low RCS pressure has actuated an ES signal that has isolated the RB purge.

B. The Reactor Building Exhaust Radiation Monitor, RM-A-9 has actuated the isolation.

C. The high Reactor Building pressure has initiated a Reactor Trip which isolated the RB purge.

D. The rise in Reactor Building temperature due to the leak has caused the RB purge isolation.

Answer: B

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	076	A1.02	
	Importance Rating	2.6		

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures.

Proposed Question: RO Question # 14

Plant conditions:

- The plant is operating at 85% power.
- Secondary Component Cooling Water (SC) temperatures have been slowly rising.
- SC Cooler Outlet Temperature (A0322) indicates 97°F and slowly rising.
- SC Surge Tank level is 3.5 feet and stable.
- SC Pump discharge pressure is 85 psig.
- Secondary River Water Pumps are tripped.
- The crew has entered OP-TM-AOP-033, Loss of Secondary Component Cooling.
- All Nuclear River Water Pumps are available.

In accordance with OP-TM-AOP-033, which ONE (1) of the following describes actions required to mitigate this event?

A. Reduce turbine generator load to restore SC Cooler outlet temperature to less than 95°F and CLOSE all Secondary River Pump discharge valves.

B. Cross tie Nuclear River system to Secondary River system by opening NR-V-2, Nuclear River to Secondary River isolation and NR-V-7, Nuclear River to Secondary River cross tie.

C. Trip the main turbine, place all Secondary Closed Pumps in PTL, and initiate shutdown of SC-cooled components.

D. Reduce Main Generator reactive load to restore SC Cooler outlet temperature to less than 95°F and OPEN all Secondary River Pump discharge valves.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** OP-TM-AOP-033 requires load reduction to restore temperature below 95°F. Additionally Secondary River Pump Discharge Valves will be closed.

B. **Incorrect.** Plausible because this would provide cooling to Secondary components, wrong because OP-TM-531-902 requires the reactor to be shut down as a prerequisite.

C. **Incorrect.** Plausible because SC cools turbine components, and tripping turbine is the logical choice. However, power is too high, and SC Surge Tank level and pump discharge pressure are normal.

D. **Incorrect.** Plausible because actions in this option are correct, but discharge valves must be closed. If applicant does not know cross-tie arrangement, this option could be selected.

Technical Reference(s): OP-TM-AOP-033, (p3, 5); Rev 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 544-GLO5, 8
626-GLO8 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate the ability monitor changes in parameters (to prevent exceeding design limits) including reactor and turbine building closed cooling water temperatures associated with operating the SWS controls (i.e. identification of actions required to reduce temperature below a limit).

The question is at the Comprehension/Application cognitive level because the operator must incorporate plant conditions into a thought process that determines appropriate actions

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	059	A2.05	
	Importance Rating	3.1		

Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture in MFW suction or discharge line

Proposed Question: RO Question # 15

Plant conditions:

- The plant is at 100% power.
- A large rupture (i.e. double ended shear) occurs in the common Main Feedwater Pump suction header.

Which ONE (1) of the following describes the operation of the Main Feedwater (MFW) System for this condition, AND the procedure(s) that the operating crew will use following transition from, or directed by, OP-TM-EOP-001, Reactor Trip?

A. Both MFW Pumps will trip on Low Suction Pressure;
OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer.

B. MFW Pumps will remain running;
OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer.

C. MFW Pumps will remain running;
OP-TM-EOP-010, Emergency Procedures Rules, Guides and Graphs.

D. Both MFW Pumps will trip on Low Suction Pressure;
OP-TM-EOP-010, Emergency Procedures Rules, Guides and Graphs.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that the MFW Pumps are tripped on low suction pressure to protect the pumps (See D); and because the operator may incorrectly believe that this event has resulted in Excessive Primary-to-Secondary Heat Transfer (XHT) as it could if break was on other side of containment check valve.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that this event has resulted in Excessive Primary-to-Secondary Heat Transfer (XHT). If the operator believed this, according to OP-TM-EOP-001 (p5; Rev 10), Step 3.1, the crew would transition to OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer. The operator may believe that XHT is occurring because According to TMI 1 FSAR (p14.1-29; REV. 19, APRIL 2008), a steam line break rather than a feed line break would result in a need to implement EOP-3.

C. **Correct.** 1st part correct, 2nd part correct. According to TMI 1 FSAR (p14.2-46; Rev 19, APRIL 2008), a loss of feedwater may result from abnormal closure of the feedwater isolation valves, control valve failure, or pump failure. The loss of feedwater flow results in a loss of heat sink, primary system heatup, increased pressurizer level and pressure, and reactor trip on high RCS pressure. In this case, both pumps are lost immediately with a break in the suction line. According to TMI 1 FSAR (p14.2-48/49; REV. 18, APRIL 2006), TABLE 14.2-22 (Sheet 3 of 3), the reactor trip will occur in about 17-18 seconds on the analyzed LOFW accident, and according to TMI 1 FSAR (p14.2-48/49; Rev 19, APRIL 2008) be more abrupt if the LOFW was due to a Feed Line Break. According to TQ-TM-104-401-C001 (p42-43; Rev 2), the main feedwater flow control block valves are interlocked to close following reactor trip. According to OP-TM-EOP-001 (p1; Rev 10) the entry condition for EOP-1 is any unplanned condition requiring an automatic or manual reactor trip. Additionally, this procedure will eventually address EOP-10 Rules and Guides that the crew may or must implement. For instance, at Step 3.6 (p5; Rev 10) the crew will be directed to verify OTSG levels greater than setpoint. Since the initiating event, a LOFW, will have substantially reduced the OTSG inventory, this may not be the case, and the crew will be directed to initiate Rule 4, Feedwater Control." On the other hand, if the OTSG level criteria is satisfied, the crew will proceed through EOP-1 to Steps 3.9 through 11 (p7; Rev 10) in which the operating crew will be directed to initiate Guides, 9, 6 and 8 during the course of EOP-1 implementation.

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that the MFW Pumps are tripped on low suction pressure to protect the pumps. According to TQ-TM-104-401-C001 (p42-43; Rev 2), the MFW Pumps are protected against low suction pressure by an automatic trip which occurs when the pump suction valves (CO-V-9A/B) are closed. In the case of the Suction Piping Rupture, the pumps are not protected by a low suction pressure trip. (They may, however, trip on overspeed, or if the steam/water mixture results in tripping of Condensate or Booster Pumps.

Technical Reference(s):	TMI 1 FSAR (p14.1-29, 14.2-46, 48-49; Rev 19, APRIL 2008) TQ-TM-104-401-C001 (p42-43; Rev 2) OP-TM-EOP-001 (p1, 5, 7; Rev 10)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 401-GLO5, 11 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to (a) predict the impacts of a rupture in MFW suction or discharge line on the MFW (Feed Control Block Valves auto close, while MFW may auto trip, but not on low suction pressure); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations (i.e. identify post-reactor trip procedure flow path).

The question is at the Comprehension/Analysis cognitive level because the operator must possess information about the event that is occurring (i.e. MFW Suction pipe rupture) and how it affects the MFW System operation (i.e. the Feed Control Block Valves auto close, while MFW may auto trip, but not on low suction pressure), and relate that to the selection of appropriate procedures (i.e. Rx trip to XHT, or Rx trip and EOP Rules/Guides).

Facility:	TMI		
Vendor:	B&W		
Exam Date:	4/2010		
Exam Type:	R		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013	A2.03
	Importance Rating	4.4	

Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; Rapid depressurization

Proposed Question: RO Question # 16

Plant conditions:

- The plant is being shutdown to repair a steam leak in the RB.
- A plant cooldown is in progress in accordance with 1102-11, Plant Cooldown.
- RCS Pressure is 1300 psig.
- RCS Temperature is 400°F.
- RB pressure is 0.8 psig and rising at 0.1 psig every five minutes.

Subsequently:

- RCS pressure starts to rapidly lower at 150 psig per minute.
- RB pressure is 0.9 psig and rising slowly.
- RM-A-5, Condenser Off Gas Monitor, goes into HIGH ALARM.

Which ONE (1) of the following predicts the amount of time before an automatic ESAS occurs, AND describes the action that the crew should take?

A. Approximately 2-3 minutes; AND
Manually actuate the 1600 PSIG ESAS immediately, and enter OP-TM-EOP-005, OTSG Tube Leakage.

B. Approximately 2-3 minutes; AND
Enter OP-TM-EOP-005, OTSG Tube Leakage, and do NOT actuate any ESAS prior to reaching the automatic setpoint being reached, unless directed.

C. 5-10 minutes; AND
Manually actuate the 1600 PSIG ESAS immediately, and enter OP-TM-EOP-005, OTSG Tube Leakage.

D. 5-10 minutes; AND
Enter OP-TM-EOP-005, OTSG Tube Leakage, and do NOT actuate any ESAS prior to reaching the automatic setpoint being reached, unless directed.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. The operator may incorrectly believe that the ESAS actuation occurs at 900 psig, rather than 500 psig (See B); and because the operator may be unaware of the OS-24 exception for proactively actuating ESAS, or misdiagnose the event and incorrectly believe that they are applying to OS-24 rule correctly (See C).

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that the ESAS signal occurs at 900 psig, the setpoint at which the 500 psig signal can be blocked, rather than 500 psig. If so, the time for the low RCS pressure ESAS would be approximately 2-3 minutes.

C. **Correct.** 1st part correct, 2nd part correct. Under these circumstances the operator would anticipate the automatic signal and proactively take the action, according to OS-24.

D. **Incorrect.** 1st part correct, 2nd part wrong. According to TQ-TM-104-642-C001 (p5, 9; Rev 4), there are two ESAS signals based on RCS Pressure; one at 1600 psig, and a second at 500 psig. Both have the ability to be bypassed for cooldown purposes. According to TQ-TM-104-642-C001 (p13-14; Rev 4) the 500 psig signal provides Safeguards protection against a LOCA during cooldown after the 1600 psig bistables have been bypassed. Additionally, there is a 900 psig bistable that allows operator to bypass the 500 psig signal during cooldown or during an accident condition to gain component control. According to 1102-11, Enclosure 2 (p4 of 13; Rev 138), for the stated conditions, the 1600 psig ESAS would have been placed in BYPASS between 1750 and 1650 psig during the cooldown. Also, according to 1102-11, Enclosure 2 (p6 of 13; Rev 138), the 500 psig ESAS will be placed in BYPASS between 900 and 550 psig during the cooldown, and therefore at the present time is still fully enabled. Furthermore, according to 1102-11, Enclosure 2 (p7 of 13; Rev 138), the Core Flood Tank Isolation Valves (CF-V-1A and B) are NOT closed until 700-650 psig, and therefore, these valves will be open at the start of the event, and tend to retard the pressure decrease. Therefore, with a depressurization rate of 150 psig per minute, the 500 psig ESAS will occur in approximately 5 minutes, or longer, perhaps as much as ten, as the RCS pressure depressurization rate is slowed by the injection of the CFTs. According to OS-24 (p19; Rev 17), Step 4.6.2, if a parameter or trend indicates that a safety system will actuate, then proactive action should be taken to place the system in the required position. The reactor should be tripped & ES manually actuated prior to reaching an ES automatic setpoint. According to OP-TM-EOP-0051 (p4; Rev 1), Step 3.4, in a post-trip condition a valid unexpected alarm resulting from offgas monitors must be assumed to indicate primary to secondary leakage, and entry into EOP-005 is appropriate. According to OP-TM-EOP-0051 (p5-6; Rev 1), steps are provided to indicate that Guide 9 is to be used to maintain Pzr and MU Tank level to achieve a more controlled power reduction and plant shutdown.

Technical Reference(s): TQ-TM-104-642-C001 (p5, 9, 13-14, 42; Rev 4) (Attach if not previously provided)
1102-11, Enclosure 2 (p4&6 of 13; Rev 138)
OS-24 (p19; Rev 17)

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO3, 5 and 10 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

The KA is matched because the operator must demonstrate the ability to (a) predict the impacts of a rapid depressurization on the ESFAS (i.e. how long to arrive at an auto ESAS from given condition); and (b) based on those predictions, use procedures (i.e. identify the correct actions to take with respect to manually actuating ESAS) to correct, control, or mitigate the consequences.

The question is at the Comprehension/Analysis cognitive level because the operator must possess information about the event that is occurring (i.e. what ESAS signals are bypassed, what signals are active) and how the event affects the ESAS System operation (i.e. based on stated conditions when will an ESAS occur), diagnose that a SGTL is occurring, and then relate this to appropriate actions identified within OS-24.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	064	A3.08	
	Importance Rating	3.7		

Ability to monitor automatic operation of the ED/G system, including: Consequences of automatic transfer to automatic position after the ED/G is stopped

Proposed Question: RO Question # 17

Plant conditions:

- The plant is at 50% power.
- The crew is completing OP-TM-AOP-046, Inadvertent ESAS Actuation, due to an inadvertent actuation of the Train B 1600 psig ESAS.
- The ARO has stopped EG-Y-1B by pressing the STOP pushbutton in the Control Room.
- Two minutes after stopping the Engine, a Loss of Offsite Power (LOOP) occurs.

Assuming that there is no further operator action, which ONE (1) of the following identifies how EG-Y-1B will respond?

A. EG-Y-1B will automatically start immediately.

B. EG-Y-1B will automatically start after a time delay.

C. EG-Y-1B will NOT automatically start, but may be manually started immediately.

D. EG-Y-1B will NOT automatically start, but may be manually started ONLY after a time delay has completed.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to OP-TM-AOP-0461 (p20; Rev 0) after an inadvertent actuation of a Train B ESAS, EG-Y-1B will be running. According to OP-TM-AOP-046 (p59; Rev 0) Step 6.11, if the EG-Y-1B is running the operator will be directed to initiate OP-TM-861-902 to secure the Diesel and place it in ES-Standby. Upon entry into OP-TM-861-902 (p2; Rev 10), since EG-Y-1B is running unloaded, Step 4.1 will direct the operator to go to Section 5.4 to shutdown EG-Y-1B and place it in ES-Standby. According to OP-TM-861-902 (p6; Rev 10), Step 5.4.8, the operator will depress the EG-Y-1B STOP PB. According to TQ-TM-104-861-C001 (p74-75; Rev 6), depressing any stop pushbutton energizes the stopping relay (5 Relay), which is a 60 second Time Delay Drop Out (TDDO) Relay. This means that when the 5 Relay energizes the contacts switch position, and when it de-energizes a time delay is initiated. After the time delay, the contacts switch back to their original position. Therefore, when any stop pushbutton is pressed the 5 Relay will energize which energizes the governor shutdown solenoid which moves the fuel racks to the NO FUEL position, effectively preventing another manual start for at least 60 seconds. However, the circuit to the governor shutdown solenoid has two normally closed contacts (ES1 and ES2) that will open to ensure that if an automatic start is called for, the auto start occurs as required, except in the case that the governor shutdown solenoid is maintain energized by the generator 86 lockout relay which picks up for differential over-current. Since there is no indication of an overcurrent lockout the governor shutdown solenoid is free to respond to any automatic signal. According to TQ-TM-104-861-C001 (p72-73; Rev 6), an ESAS or UV will energize the Emergency Start Relays (ES1 and ES2), the cranking time delay relays (T2A and T2B), and the start relays (4A and 4B). This relay status energizes the start relays which de-energizes the Air Start Solenoid Valves (AS1 and AS2) and the Air Vent Valve (AV) allowing starting air to be emitted into the cylinders, and the engine will start to crank.

B. **Incorrect.** This is plausible because the EG-Y-1B has an interlock to prevent any manual start after a normal (i.e. manual) shutdown or automatic shutdown for a minimum of 60 seconds after either one of the STOP pushbuttons are depressed, or an adverse condition has initiated an automatic shutdown via the Shutdown (SDR) Relay. According to OP-TM-861-902 (p2; Rev 10), Step 4.2.3 addresses this 60 second Time Delay providing actions in the case that the engine will not start automatically and a manual start is needed. Because of this, the operator may incorrectly believe that the EG-Y-1B will start after a time delay.

C. **Incorrect.** This is plausible because the operator may believe that the unit must be restored to standby status before the engine will even attempt to start, even on a LOOP. According to OP-TM-861-902 (p7; Rev 10) Steps 5.4.9-17 must be completed to fully restore the Diesel to ES-Standby status and the operator may incorrectly believe that since these steps have NOT been performed, the unit will not start on a UV/ESAS, but may be manually started as required.

D. **Incorrect.** This is plausible because this response if characteristic of an Engine that has undergone a Start Failure, and since there are additional steps to complete to restore the engine to ES-Standby (See B) the operator may incorrectly believe that the unit will manually start, but recognizing there is a time delay in the start circuitry, may believe it must time out.

Technical Reference(s):	TQ-TM-104-861-C001 (p72-75; Rev 6) OP-TM-861-902 (p2, 6, 7; Rev 10) OP-TM-AOP-0461 (p20; Rev 0) OP-TM-AOP-046 (p59; Rev 0)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO5, 10 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

The KA is matched because the operator must demonstrate the ability to monitor automatic operation of the ED/G system, including the consequences of automatic transfer to automatic position after the ED/G is stopped (i.e. identify what will happen to the diesel on UV/ESAS, 30 seconds after STOP pushbutton is pressed).

The question is at the Comprehension/Analysis cognitive level because the operator must understand how the Emergency Diesel Generator starting circuit UV/ESAS start functions in the presence of the normal shutdown cycle after auto start. This requires knowledge of both circuits, and integrating the two, together to predict an outcome.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	012	A3.05	
	Importance Rating	3.6		

Ability to monitor automatic operation of the RPS, including: Single and multiple channel trip indicators

Proposed Question: RO Question # 18

Plant conditions:

- The plant is at 100% power.
- Main Control Board Annunciator G-1-2 RPS CHANNEL TRIP, alarms.
- The plant is stabilized.
- The crew diagnoses that the Narrow Range RCS Pressure Transmitter (RC3A-PT1) has lost power.

Which ONE (1) of the following describes how this will be indicated on the Reactor Trip Modules (RTM)?

A. The A RTM Subsystem Trip lamp turns BRIGHT ONLY;
The other RTMs have one Protective Subsystem lamp that turns BRIGHT.

B. The A RTM Subsystem Trip lamp turns BRIGHT ONLY;
The other RTMs remain unchanged.

C. The A RTM Subsystem Trip AND Protective Subsystem No. 1 lamps turn BRIGHT;
The other RTMs have one Protective Subsystem lamp that turns BRIGHT.

D. The A RTM Subsystem Trip AND Protective Subsystem No. 1 lamps turn BRIGHT;
The other RTMs remain unchanged.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that when a channel is tripped there is only one lamp that needs to be BRIGHT to indicate this.

B. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that when a channel is tripped there is only one lamp that needs to be BRIGHT to indicate this; and because the operator may incorrectly believe that there are no lamps from other RTMs associated with a given RTM.

C. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-624-C001 (p29; Rev 2), RCS Narrow Range Pressure Detector RC3A-PT1 feeds the "A" RPS cabinet. According to TQ-TM-104-641-C001 (p27; Rev 1) an individual input component may lose power and fail in a tripped state. As an example, If RCS pressure transmitter, such as RC3A-PT1, loses power and fails low, it causes a Low RCS pressure trip and trips the channel (i.e. Channel A). When this occurs, it will produce changes in the Reactor Trip Modules (RTM) that the operator may observe. According to TQ-TM-104-641-C001 (p32-33; Rev 1), all lamps on each RTM are normally DIM. However, the SUBSYSTEM TRIP lamp turns BRIGHT on the associated RTM when the channel relay, in this case KA, an "Output Memory" relay, is tripped. Additionally, this relay operates an "Output State" relay in each of the four RTMs (KA1, KA2, KA3 and KA4). These relays, when operated turn the respective PROTECTIVE SUBSYSTEM lamp on each of the RTMs BRIGHT. According to OP-TM-MAP-G0102 (p2of2; Rev 2), the operator is directed to check the RPS bistables to determine the cause of the trip, and a Note is provided just prior to this direction to ensure that the operator understands that the tripped bistables should have two BRIGHT lamps, an Output State lamp (associated with KA1-4 Relay) and an Output Memory lamp (associated with KA Relay).

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that there are no lamps from other RTMs associated with a given RTM.

Technical Reference(s):	TQ-TM-104-624-C001 (p29; Rev 2) TQ-TM-104-641-C001 (p27, 32-33; Rev 1) OP-TM-MAP-G0102 (p2of2; Rev 3)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	641-GLO5		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	NA	Last NRC Exam: NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments:

The KA is matched because the operator must demonstrate the ability to monitor automatic operation of the RPS, including single (i.e. how does the A RTM change due to a changed input) and multiple channel trip indicators (i.e. how do the remaining three RTMs change due to a failed input on another channel).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. that the RTM Module has a lamp turning BRIGHT when the Output Memory Relay de-energizes; and that each RTM has a lamp turning BRIGHT when its own Output State Relay de-energizes as well as an associated Output State Relay with each of the other Channels).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	008	A4.04	
	Importance Rating	2.6		

Ability to manually operate and/or monitor in the control room: Startup of a CCW pump when the system is shut down.

Proposed Question: RO Question # 19

Plant conditions:

- The plant is in cold shutdown at the end of a Refueling Outage.
- Containment Isolation Valves IC-V-2,3,4,6 are closed.
- OP-TM-541-101, Shifting Primary Component Cooling to Operating Mode, is in progress.
- Nuclear Service Closed Cooling Water Pump NS-P-1A is running.
- Both Intermediate Closed Cooling Water Pumps IC-P-1A and 1B are in Pull To Lock (PTL).
- The operating crew is getting ready to start IC-P-1A.

Which ONE (1) of the following describes the condition that will cause IC-P-1A to START, AND identifies the expected condition of the IC SYSTEM FLOW LO alarm after the pump is started?

A. IC-P-1A will start when the operator takes the control switch out of PULL TO LOCK (PTL);
When the pump starts the IC SYSTEM FLOW LO is expected to clear.

B. IC-P-1A will start when the operator takes the control switch out of PULL TO LOCK (PTL);
When the pump starts the IC SYSTEM FLOW LO is expected to remain in.

C. IC-P-1A will start when the operator takes the control switch to NORMAL AFTER START (NAS);
When the pump starts the IC SYSTEM FLOW LO is expected to clear.

D. IC-P-1A will start when the operator takes the control switch to NORMAL AFTER START (NAS);
When the pump starts the IC SYSTEM FLOW LO is expected to remain in.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible if the operator believes incorrectly that the system valves are opened upon pump start, and that when the pump is started the alarm will clear. However, it is intuitive that when a pump is started and associated system low flow alarm should clear.

B. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-531-000 (p2; Rev 10), in the plant shutdown mode both IC Pumps are in PTL during the Shutdown Mode, and in the Operating Mode IC-P-1A is Running, while IC-P-1B is in Normal After Stop. Additionally, IC-V-2, 3 and 4 are closed isolating flow to system loads. According to TQ-TM-104-531-C001 (p34, 39; Rev 5), the IC Pumps have a Pistol Grip Control Switch which includes an NORMAL AFTER STOP, STOP, NORMAL AFTER START and START position, with a PULL TO LOCK (PTL) feature from the NORMAL AFTER STOP position. An interlock exists such that when ICCW system flow is less than 550 gpm, the standby pump will start. The standby pump control switch must be in the AFTER STOP position to enable the automatic starts. When the operator takes the IC-P-1A Control Switch from the PTL position to the NORML AFTER STOP position, the control circuitry will see this pump as in the Standby Mode, and since there will be NO flow in the System, the pump will start. In fact, according to OP-TM-531-101 (p4; Rev 2), a NOTE is provided just prior to the step directing the operator to place the pump control switch in the NORMAL AFTER START position that notifies the operator that the IC Flow Interlock will cause the IC-P-1A to start when the control switch is taken from the PTL position. Therefore, the Pump will start when the operator takes the control switch out of PTL. Immediately following the pump start the operator is directed to check system flow > 40 gpm. At this time total system flow will be low since IC-V-2, 3, 4 and 6 are closed, therefore when the pump is started the system will have relatively little flow, according to TQ-TM-104-531-C001 (p40; Rev 5), only through IC-V-74 which will automatically open when IC-V-2, 3, 4 and 6 are closed. According to OP-TM-MAP-C0202 (p1of1; Rev 2), the IC SYSTEM FLOW LO alarm will alarm if the flow through the system is < 550 gpm. Immediately after the pump start, and before the opening of IC-V-2, 3, 4 and 6, the system flow will be lower than 550 gpm, and therefore the alarm will NOT clear until the system valves are subsequently opened. In fact, in accordance with OP-TM-531-101 (p4; Rev 2), subsequent steps are provided to open these system valves and then test the interlock.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible if the operator incorrectly believes that the first pump started will not function as a pump in standby (See D); and if the operator believes incorrectly that the system valves are opened upon pump start, and that when the pump is started the alarm will clear (See A).

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible if the operator incorrectly believes that the first pump started will not function as a pump in standby. This is supported by the fact that the standby feature is normally associated with the non-running pump. However, in this case, the operator must make the connection that the first pump started is under the control of the no flow interlock.

Technical Reference(s): OP-TM-541-000 (p2; Rev 11)
TQ-TM-104-531-C001 (p34, 39-40; Rev 5)
OP-TM-541-101 (p4; Rev 2)
MAP-C0202 (p1of1; Rev 2)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 541-GLO2, 5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to manually operate (i.e. associate expected pump starts with control switch positions) and monitor (i.e. predict system flow after pump start) the startup of a CCW pump when the system is shut down in the control room.

The question is at the Comprehension/Analysis cognitive level because the operator must assemble at least three pieces of information (i.e. that (1) the pumps are controlled by a five position control switch with a PTL feature (and know the meaning of this), that (2) the pumps are subject to a system low flow interlock, and that (3) the System valves are closed during startup of the system); and draw a conclusion (i.e. that the pumps will start when moved from PTL, and there will be little flow through the system at the time of pump start).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	103	A4.06	
	Importance Rating	2.7		

Ability to manually operate and/or monitor in the control room: Operation of the containment personnel airlock door

Proposed Question: RO Question # 20

Containment entry is in progress.

Which ONE (1) of the following describes the Control Room indications available to the operator to determine the status of the Containment Personnel Hatch Doors?

A. Both doors have Open/Closed indication.

B. Both doors have Open indication ONLY.

C. Only the INNER door has Open/Closed indication.

D. Only the OUTER door has Open/Closed indication.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to TQ-TM-104-240-C001 (p12-13, 37; Rev 2) Panel PL has backlit status lights for both the RB Personnel Lock Outer and Inner Doors. When the light is Green the door is closed. When the door is Red the door is Open.

B. **Incorrect.** This is plausible because the operator may incorrectly believe that the only indication in the Control Room associated with the RB Personnel Hatch doors is the abnormal indication of OPEN.

C. **Incorrect.** This is plausible because the operator may incorrectly believe that the only indication in the Control Room associated with the RB Personnel Hatch doors is the inner door which must be closed to maintain Containment Integrity.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that the only indication in the Control Room associated with the RB Personnel Hatch doors is the outer door which must be opened first on a Containment Entry.

Technical Reference(s): TQ-TM-104-240-C001 (p12-13, 37; Rev 2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 240-GLO2, 5 (As available)

Question Source: Bank # IR-240-GLO-2-Q02

Modified Bank # (Note changes or attach parent)

New

Question History: ILT 03-1 NRC Audit #2 Q28 Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 9

55.43

Comments:

The KA is matched because the operator must demonstrate the ability monitor (i.e. identify the status lights associated with the hatch doors) the operation of the containment personnel airlock door in the control room.

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. Which doors associated with the RB personnel hatch have status indication lights, what do these lights specifically indicate).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	039	2.4.4	
	Importance Rating	4.5		

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: RO Question # 21

Which ONE (1) of the following would require entry into OP-TM-AOP-051 (Secondary Side High Energy Leak)?

A. A small Steam Leak is discovered in the Turbine Building and can be isolated by closing MS-V-5A, Main Steam Isolation valve to FW-P-1A.

B. Reactor Building Pressure is at 1.5 psig and slowly rising, with indication of rising counts on RM-A-2, RB Rad Monitor.

C. A Main Steam Line Break in the Reactor Building resulting in a rapid De-pressurization of an OTSG.

D. MWe is slowly lowering concurrent with a Fire Alarm in the Intermediate Building.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible because the symptoms stated are indicative of a steam leak, and the operator may incorrectly believe that entry into AOP-051 is needed. However, according to OP-TM-AOP-0511 (p3; Rev 0) AOP-051 is entered for steam leaks that affect large portions of the plant and therefore it is

NOT obvious to the operator what needs to be done to isolate the leak. If, as stated in the conditions, the operator knows what action to take to isolate the leak, there is no need to enter AOP-051.

B. **Incorrect.** This is plausible because the operator may misdiagnose the presence of radioactivity, and focus only on the increasing RB temperature and Pressure. According to OP-TM-AOP-0511 (5; Rev 0), AOP-051 may also be entered if RB temperature and pressure are approaching the TS limits because this is symptomatic of a high energy line break in the RB. However the presence of radioactivity indicates that the leak is not from a secondary system High Energy Line, but rather a primary system. According to OP-TM-AOP-0511 (p2; Rev 0), a steam leak will result in a secondary side fluid being released to the environment which will NOT pose a significant radiation release. In this case, According to OP-TM-AOP-0501 (P2; Rev 1), radiation monitors are used to identify and isolate the leak within AOP-050, and NOT AOP-051.

C. **Incorrect.** This is plausible because these symptoms are characteristic of a Steam Rupture, and the Operator may incorrectly believe that entry into AOP-051 is warranted. However, according to OS-24 (p3; Rev 17) Definition 3.5 XHT is undesired heat removal by one or both OTSGs. The stated symptoms are characteristic of XHT. According to OP-TM-AOP-0511 (p4; Rev 0), the EOP network is designed to handle steam leaks that are large enough to cause XHT or FW leaks that are large enough to cause a LOHT.

D. **Correct.** According to OP-TM-AOP-0511 (p1; Rev 0), a secondary side steam leak will bypass steam away from the Turbine, and if large enough will cause electrical generation to lower. According to OP-TM-AOP-0511 (p2; Rev 0), a steam leak may cause fire alarms to actuate, and in fact, this could be the first indication of a steam leak to the control room. According to OP-TM-AOP-0511 (p4; Rev 0), Entry Conditions into AOP-051 are ALL of the following: (1) indications of a Secondary side Steam leak (As previously stated loss in MWe and fire alarms), (2) the leak does NOT cause XHT or LOHT, (3) the OTSGs are being used for RCS heat removal. Therefore, an indication of slowly lowering MWe concurrent with a Fire Alarm in the Intermediate Building would be grounds for entering AOP-051.

Technical Reference(s): OP-TM-AOP-0511 (p1-5; Rev 0) (Attach if not previously provided)
OS-24 (p3; Rev 18)
OP-TM-AOP-0501 (P2; Rev 1)

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP051-GLO1 (As available)

Question Source: Bank # IR-AOP051-PCO-2-

Q01

Modified Bank # (Note changes or attach parent)

New

Question History:	NA	Last NRC Exam:	NA

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:

The KA is matched because the operator must demonstrate the ability to recognize abnormal indications for system operating parameters (i.e. decreasing MWe coupled with fire alarms) which are entry-level conditions for emergency and abnormal operating procedures (specifically, AOP-051, HELB).

The question is at the Comprehensive/Analysis cognitive level because the operator must evaluate four sets of conditions, and sort them from each other (i.e. known leak location and action, radioactivity involved, rupture vs. leak, and unknown specific leak location), and then compare them to the entry conditions for a procedure.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	012	2.2.12	
	Importance Rating	3.7		

Equipment Control: Knowledge of surveillance procedures.

Proposed Question: RO Question # 22

The Reactor is at 100% power.

The last channel checks for Reactor Protection System (RPS) channels were performed at 0900 on April 12, 2010.

In accordance with Technical Specifications, which ONE (1) of the following is the LATEST time that RPS channel checks may be performed, and remain within the allowable surveillance period, including extensions, if applicable?

A. 2100, April 12, 2010

B. 2400, April 12, 2010

C. 0900, April 13, 2010

D. 1500, April 13, 2010

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** RPS Channel Checks are required every 12 hours (S for once per shift). Therefore, this would be correct if there were no allowable extensions. It is plausible to account for no allowable extension because surveillances designated (F) have no allowable extension.

B. **Correct.** For a 12 hour required surveillance, the allowable extension is 1.25 times the surveillance interval, or 3 additional hours. Therefore, the surveillance must be performed within 15 total hours of the previous surveillance.

C. **Incorrect.** This allows for 24 hours, and is plausible because this is the closest surveillance interval (D) to the actual requirement. Also logical to not consider extension, since (F) surveillances do not allow extensions.

D. **Incorrect.** This allows for 24 hours, and is plausible because this is the closest surveillance interval (D) to the actual requirement. Additionally, this time allows for the 1.25 time extension on the Daily surveillance. If the applicant believes the interval is daily, he/she would choose this option if they also apply the extension.

Technical Reference(s): Technical Specification Definition 1.25, Amendment 199 (Attach if not previously provided)
Technical Specification Table 4.1-1 (Amendment 262)

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO10 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

The KA is matched because the operator must demonstrate the knowledge of surveillance procedures (i.e. in practice how frequently this is performed, and the allowable extension).

The question is at the Analysis cognitive level because the operator must recall how often a channel check is performed, and calculate the latest the surveillance can be performed.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	064	2.1.19	
	Importance Rating	3.9		

Conduct of Operations: Ability to use plant computers to evaluate system or component status.

Proposed Question: RO Question # 23

Plant conditions:

- Plant is at 100% power.
- The A Emergency Diesel Generator (EG-Y-1A) has been manually started for a test, and load is being raised in accordance with 1107-3, Diesel Generator.

Which ONE (1) of the following identifies the HIGHEST allowable Diesel Generator Megawatt load, and the indication available that would warn the crew that the load is being exceeded?

A. 2.5 MW; Computer alarm 'EG-Y-1A(1B) Electrical Load'.

B. 2.5 MW; Main Annunciator Panel A, 'Diesel Gen 1A(1B) Overload' alarm.

C. 3.0 MW; Computer alarm 'EG-Y-1A(1B) Electrical Load'.

D. 3.0 MW; Main Annunciator Panel A, 'Diesel Gen 1A(1B) Overload' alarm.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** Plausible because the alarm point is correct. 1105-10A, Page E2-

152 shows alarm at 3.0 MW. Option is plausible because reactive load limit is 2.5 MVAR, and is also indicated by a computer alarm.

B. **Incorrect.** Incorrect but plausible because the MVAR rating is 2.5 MVARs. Additionally, the MAP alarm is incorrect because the limit for that alarm (MAP A-3-1), is current flow (amperes). Plausible because high current flow may potentially relate to high MW output, but wrong because the alarm does not indicate MW output.

C. **Correct.** In accordance with 1107-3, maximum DG output should not exceed 3.0 MW. A computer alarm warns the crew that 3.0 MW has been exceeded.

D. **Incorrect.** Plausible for same reasons as option B, as an overload conditions could be related to high MW output. However, the MAP alarm is a direct indication of high current flow, not high MW output.

Technical Reference(s): 1107-3 (sec 2.1, p10; Rev 126C) (Attach if not previously provided)

1105-10A, pE2-152

MAP A-3-1

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO5, 10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to use plant computers to evaluate component status (i.e. Emergency Diesel Generator MVAR Loading and Generator load).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. that the procedure requiring evaluation of operation of the Emergency Diesel Generators requires the knowledge that the plant computer alarm warns of a high MW and MVAR output).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005 K1.04	
	Importance Rating	2.9	

Knowledge of the physical connections and/or cause-effect relationships between the RHRS and the following systems: CVCS

Proposed Question: RO Question # 24

Plant conditions:

- While operating at power a LOCA occurred.
- The plant tripped and the 1600 psig/ RB 4 psig ESAS has actuated.
- BWST Level has lowered to less than 15 ft.
- The crew has just transferred to RB Sump Recirc and placed HPI in "piggyback" mode IAW OP-TM-211-901, Emergency Injection HPI/LPI.

Which ONE (1) of the following identifies the position of MU-V-14A and 14B (Makeup Pump Suction from the BWST), AND the reason for this position?

A. OPEN;
Allows for recirculation back to the BWST if the MU Pump trips.

B. OPEN;
Protects the Make-up Pump from loss of suction if the DH Pump trips.

C. CLOSED;
To prevent release of high radioactivity water to the environment.

D. CLOSED;
To ensure that RB Sump Recirc flow requirements are met for Decay Heat Removal.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may incorrectly believe that the valve is open to allow recirculation in the event of the MU Pump trip, thinking that in the piggyback mode the RCS pressure is too high to permit flow into the RCS from the DH Pump alone. However, in this case the DH Pump would simply recirc back to its suction until the RCS pressure dropped to < 250 psig (i.e. the head of the DH Pumps).

B. **Correct.** According to TQ-TM-104-211-C001 (p26; Rev 3), both MU-V-14A and 14B (Makeup Pump Suction from the BWST) automatically open on an ESAS of HPI. Section 4.3 of OP-TM-211-901 (p13; Rev 5) places HPI in the piggyback mode, and this procedure does not re-position MU-V-14A and 14B. According to OP-TM-EOP-0101 (p64; Rev 2) in the event of a LOCA which results in RB Sump Recirculation, MU-V-14A and B are left open as HPI is placed in the piggyback mode in order to provide an alternate suction path for the MU Pump if the DH Pump were to trip. This procedure provides clarification saying that if DH-V-7A or 7B was open and MU-V-14A or 14B was not providing a leak tight seal, then 14A and 14B must be closed to prevent the continued release of potential high activity water to the environment, as well as the loss of RCS inventory. However, since the conditions stated have just placed the system in the piggyback mode, this BWST leakage check, will come subsequent to this operation (in Guide 22). Therefore, the MU-V-14 valves are open to ensure an alternate suction path for the MU Pumps.

C. **Incorrect.** This is plausible because the operator may incorrectly believe that the valves are closed to prevent back-leakage from the discharge of the DH Pump into the BWST (Which is vented to atmosphere) via MU-V-14A and B. However, the DH Pump discharge is sufficient to seat the BWST borated water source check valve, and effectively block any reverse flow into the BWST. In fact, in Guide 22, In accordance with OP-TM-EOP-010 (p30; Rev 10), Guide 22, (p1of4), the operator is directed to check for leakage into the BWST when DH-V-7A and 7B are open, and if leakage occurs, MU-V-14A and B must be closed.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that having MU-V-14A and B open will divert flow from the recirculation operation. This could be true if the Makeup Pump BWST suction check valve was not seated, but the major cause for concern if that were to happen would be contamination of BWST contents.

Technical Reference(s):	TQ-TM-104-211-C001 (p26; Rev 3) Section 4.3 of OP-TM-211-901 (p13; Rev 5) OP-TM-EOP-0101 (p66; Rev 3)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO3, 8
212-GLO3, 8 (As available)

Question Source: Bank #

Modified Bank # IR-212-GLO-2-Q01 (Note changes or attach parent)

New

Question History: Old Question ID SR4A11-03-Q01 Migrated from TMI OPS 8/05 Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate the knowledge (i.e. normal position of MU-V-14A/B during piggyback operations, the reason for leaving the valves open during piggyback operations) of the cause and effect relationships between the RHRS and the CVCS. Physical connection demonstrated by knowledge that DH-R (RHR) pumps are supplying NPSH to MU-P (CVCS) pumps and cause effect that if a DH-P trips MU-V-14 valves (CVCS) must be open to replace DH-R (RHR) as suction source.

The question is at the Comprehension/Analysis cognitive level because the operator must identify that there is an advantage for leaving the MU-V-14A/B valves open during piggyback operations, in light of the fact that there are situations that may arise where it is advantageous to have them closed (i.e. BWST leakage), and then determine what situation exists in the present condition (i.e. that they should be OPEN).

The modification included the addition of identification of the position of MU-V-14A/B to avoid teaching in the Stem, and to lend plausibility.

Exam Bank Searches:
Section 34 quest for 34005K104 – None
(50) **IR-212-GLO-2-Q01 (Can probably connect KA with modifications – avoid teaching in the stem)**

WTS: One question (62764) on previous TMI Exams related to this KA. This question was Q31 on the 2008 NRC Exam. I cannot use this Question
Modified Question

MU-V-14A/B are left open during piggyback operation.

Which of the following statements describes the reason for this action?

A. ESAS prevents closing these valves.

B. To allow use of the entire inventory of the BWST.

C. Allows for recirc back to the BWST if the make-up pump trips.

D. Protects the make-up pump from loss of suction if the decay heat removal pump trips.

Answer: D

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	003	A3.04	
	Importance Rating	3.6		

Ability to monitor automatic operation of the RCPS, including: RCS flow

Proposed Question: RO Question # 25

Plant conditions:

- Plant is at 100% power.
- RCS Loop A flow is 72 x 10^6 lbm/hr.
- RCS Loop B flow is 72 x 10^6 lbm/hr.

Plant event:

- The RC-P-1B trips.
- The plant runs back and stabilizes at a lower power level.

Which ONE (1) of the following describes the status of the RC-P-1B AC Oil Lift Pump, AND the RCS loop flow indication?

A. The AC Oil Lift Pump is RUNNING;
RCS Loop B flow indication has remained the same.

B. The AC Oil Lift Pump is RUNNING;
RCS Loop B flow indication has increased slightly.

C. The AC Oil Lift Pump is OFF;
RCS Loop B flow indication has remained the same.

D. The AC Oil Lift Pump is OFF;
RCS Loop B flow indication has increased slightly.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that flow in the Loop opposite the Loop where the Pump has tripped will remain unaffected by the Pump trip, and this is the normal flow when two pumps are running.

B. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-226-C001 (p15; Rev 4) If an RCP trips, the AC oil lift pump starts automatically on breaker open signal. The DC oil lift pump starts on "breaker open" and low pressure of 1000 psig. The applicant may believe that the DC pump starts, with the AC Pump providing the backup. According to TQ-TM-104-226-C001 (p52-53; Rev 4), the total flow for 4 RCPs running is 144 E6 lbm/hr, including 72 E6 lbm/hr/Loop. This is described in the initial conditions. However, the total flow for 3 RCPs running is 108 E6 lbm/hr, with 33 E6 lbm/hr/"A" Loop, and 75 E6 lbm/hr/"B" Loop. In other words, the B Loop flow has slightly increased. This is also reflected in OPM B-02 (p43; Rev 8), which indicates that when one RCP trips the percentage of total core flow from of all running pumps increases. While it increases slightly for the two pumps in the non-affected loop (i.e. from 25 to 26% of total core flow), it increases substantially in the affected loop's running pump (i.e. 25 to 32% of total core flow). However, the total affect in the loop with the stopped pump is a reduction by more than half since much of the flow is back flowing through the stopped pump.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that the AC Lift Oil Pump requires a 1000 psig signal to start automatically (See D); and because the operator may incorrectly believe that flow in the Loop opposite the Loop where the Pump has tripped will remain unaffected by the Pump trip (See A).

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that the AC Lift Oil Pump requires a 1000 psig signal to start automatically. Both AC and DC pumps do receive automatic start signals under specific conditions. (DC will start if AC does not produce a pressure of 1000 psig) The applicant may believe that the DC pump receives a start signal, and that the AC pump provides backup.

Technical Reference(s): TQ-TM-104-226-C001 (p15, 52-Rev 4)
OPM B-02 (p43; Rev 8) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	226-GLO2, 10	(As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to monitor automatic operation of the RCPS (i.e. Oil Lift System auto operation on pump trip), including RCS flow (i.e. how the non-affected loop flow changes on RCP trip).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013	K2.01
	Importance Rating	3.6	

Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

Proposed Question: RO Question # 26

Plant conditions:

- A loss of 1A 250 Volt DC has occurred.
- The crew is attempting to determine the cause of the failure.
- Subsequently, a loss of off-site power occurs, and the reactor trips.

Which ONE (1) of the following describes the operation of 1A Emergency Diesel Generator for this event?

A. EDG 1A started when 1A 250 volt DC was lost; output breaker closes and equipment loads as required for plant conditions when off-site power was lost.

B. EDG 1A started when 1A 250 volt DC was lost; DC excitation is lost; output breaker remains open, and equipment powered by the associated 4160 volt bus remains de-energized upon loss of off-site power.

C. EDG 1A remains shut down throughout the event because air start system is failed; may be manually started by manually opening air start valves; output breaker must be manually closed to supply the associated 4160 volt bus.

D. EDG 1A remains shut down throughout the event because air start system is failed; EDG is unavailable to supply the 4160 volt bus because DC excitation is unavailable.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** Plausible because the EDG and safeguards control system has multiple electrical inputs, from VBA and VBC, both of which receive power from 250 volt DC 1A. Wrong because excitation and DC control power is lost with loss of the DC bus, so output breaker closure will NOT occur

B. **Correct.** According to TQ-TM-104-861-C001 (p38-39; Rev 6) are primarily from 480 VAC 1A ES MCC and 125/250 VDC Distribution Panel 1P. According to TQ-TM-104-642-C001 (p74-75; Rev 4), a 250 Volt battery failure simultaneous with a loss of offsite power results in the loss of the two associated inverters. Loss of the Train A battery will result in loss of inverter A and inverter C. This will result in the ESAS output relays in Actuation Cabinet compartments 4A and 4C deenergizing resulting in actuation of Train A, however, since the Train A battery has failed, no equipment actuation will occur. The diesel will start but its output breaker will not close. Additionally, according to OP-TM-AOP-0231 (p7; Rev 1), EG-Y-1A will start on a loss of A DC because the air start valves will fail open, and not close as they normally would when the Diesel starts. While the engine will start, according to OP-TM-AOP-0231 (p12; Rev 1), on a loss of DC A there is a loss of excitation power, and control power to its output breaker. Therefore, while the engine starts it will not repower its ES Bus.

C. **Incorrect.** Plausible because it is logical that the system will not start without control power. In this case, failure of air start system causes valves to fail open, starting the EDG. But without control power available, the EDG will not start or supply the bus

D. **Incorrect.** Plausible because it is logical that the system will not start without control power. In this case, failure of air start system causes valves to fail open, starting the EDG. The remainder of this option is correct

Technical Reference(s): OP-TM-AOP-023 (p33; Rev 1) (Attach if not previously provided)
OP-TM-AOP-0231 (p7, 12; Rev 1)

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Modified Bank #		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of bus power supplies to ESFAS/safeguards equipment control (i.e. 1A 250 Volt DC Distribution System powers starts valves for Diesel, excitation power and provides control power to output breaker)

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. function of Diesel is to start without AC Power, and provide it in an emergency, in an LOOP and a train related loss of DC, all electrical power is lost to train).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004	K6.05
	Importance Rating	2.5	

Knowledge of the effect of a loss or malfunction on the following CVCS components: Sensors and detectors

Proposed Question: RO Question # 27

Plant conditions:

- Plant has just tripped from 100% power.
- The operating crew is implementing the appropriate post-trip procedures.
- RCS Pressure is 1750 psig and stable.
- RCS Temperature is 540°F and stable.
- Makeup Tank Level is 82 inches.
- Makeup Tank Pressure is 28 psig.
- Makeup flow is 180 gpm.
- Letdown flow is 45 gpm.

Subsequently:

- The operator observes Makeup Tank Level (MU14) to be 85 inches and rising.
- The operator observes Makeup Tank Pressure (MU17) to be 25 psig and lowering.

Assuming plant conditions do not change, which ONE (1) of the following describes the observed conditions, AND, if any, the operational concerns?

A. These indications are expected for the plant conditions; AND
There are no operational concerns.

B. MU Tank Pressure (MU17) indicator is failing low: AND
The operator has approximately 5 minutes before the MU Tank relief valve lifts.

C. MU Tank Level (MU14) indicator is failing high; AND
The operator has approximately 5 minutes before the MU Pump loses suction.

D. MU Tank Level (MU14) indicator is failing high; AND
The operator has approximately 15 minutes before the MU Pump loses suction.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible because as stated in TQ-TM-104-211-C001 (p64-65; Rev 3) and OP-TM-MAP-D0302 (p1of3; Rev 4) and D0303 (p1of2; Rev 3), MU tank level and pressure have an interrelated relationship; when level changes, pressure will as well. Since both are changing this is plausible, and if this was expected, there would be no operational concerns. However, the changes of the two parameters are not consistent.

B. **Incorrect.** This is plausible because both the MU Tank Level and Pressure indications are changing, and they are changing in an inconsistent manner; therefore, one of them must be in error. If the operator believed that the MU Tank Pressure Transmitter were failing low, and in fact the MU Tank level were increasing (by the same magnitude that the level is actually decreasing), the Tank would be full in approximately five minutes, and the operator would be expected to take some action to prevent component damage.

C. **Incorrect.** This is plausible because the MU Tank level instrument is failing high (See D); and because according to TQ-TM-104-211-C001 (p68; Rev 3), the MU Tank is sized so that the useful tank water volume will prevent the tank from emptying during power level changes and provide five minutes of makeup at maximum flow rate. The operator may incorrectly believe that flow is at maximum.

D. **Correct.** According to TQ-TM-104-211-C001 (p64-65; Rev 3), there is a relationship between MU Tank level and pressure. During steady state operation the Makeup Tank level and pressure should be maintained between 72 to 92 inches and 24 to 34 psig. According to OP-TM-MAP-D0302 (p1of3; Rev 4) and D0303 (p1of2; Rev 3), high level is NOT a function of MU Tank pressure, but low Pressure is a function of lowering MU Tank level. Since in the stated conditions, MU Tank level is increasing while MU Tank pressure is decreasing, a failed MU Tank Level transmitter might be suspected. This is strengthened by the fact that the inventory balance into and out of the MU Tank requires that the level be lowering. For instance, the MU Pump is taking 200 gpm out of the tank, 180 gpm through MU-V-17, and 20 gpm through the RCP Seals, while at the same time only 45 gpm of Letdown Flow is flowing into the tank. Based on these conditions, MU Tank level should be lowering. Since level is increasing, the operator will diagnose a failed level instrument. According to TQ-TM-104-211-C001 (p68; Rev 3), the makeup tank was sized such that the useful tank water volume will prevent the tank from emptying during power level changes and provide five minutes of makeup at maximum flow rate. The nominal tank volume is 600 cu ft. The calculated tank volume is 575 cu ft. This was required to provide the control room operator time to switch to an alternate suction source for the makeup pump or initiate makeup to the makeup tank before the makeup tank could empty. The tank actually provides more than 10 minutes of maximum makeup flow. Between a tank level of 73 inches (theoretical normal level) and 0 inches indicated, there is about 2250 gallons. Assuming a Reactor Coolant System pressure of 1600 psig following a reactor trip, the makeup line flow is about 180 gpm with makeup control valve MU-V17 full open. With 180 gpm makeup plus 20 gpm seal inleakage minus 45 gpm letdown, the net makeup to the Reactor Coolant System or net outflow from the makeup tank is about 155 gpm which would take nearly 15 minutes to remove the 2250 gallons from the makeup tank. With the initial level in the MU Tank of 82 ", and the MU Tank Pressure trending down, this is indicative of a lowering level, and would place the actual level within the approximate level stated by the example in the Training Lesson Plan. Therefore, the operator will have approximately 15 minutes to allow shifting suction of the MU Pump to a viable source such as the BWST before NPSH and gas entrainment concerns start to be of concern.

Technical Reference(s): TQ-TM-104-211-C001 (p64-65, 68; Rev 3) (Attach if not previously provided)
OP-TM-MAP-D0302 (p1of3; Rev 4)
OP-TM-MAP- D0303 (p1of2; Rev 3)

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO7 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. How do MU Tank Level and Pressure track during plant changes, how much time the operator has to complete the transfer of MU Pump suction) of the effect of a malfunction of the MU Tank Level such that it fails high, on the CVCS components (i.e. loss of level indication will require shifting suction of the MU Pump to another source).

The question is at the Comprehension/Analysis cognitive level because the operator must compare the relationship between MU tank level and pressure, and the present changing conditions, and then draw a conclusion about their validity, and then based on their conclusion, estimate the time available for actions to prevent component damage.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061	K5.02	
	Importance Rating	3.2		

Knowledge of the operational implications of the following concepts as the apply to the AFW: Decay heat sources and magnitude

Proposed Question: RO Question # 28

Which ONE (1) of the following situations would require the HIGHEST EFW flow rate to maintain post-trip RCS temperature stable following an automatic reactor trip?

(Assume the same post-trip temperature is maintained in each case and MFW is NOT available.)

A. A trip with off-site power available;
Following a ten day run at 100% power.

B. A trip with a coincident loss of off-site power;
Following a ten day run at 100% power.

C. A trip with off-site power available;
After reaching 100% power following a four day mini-outage.

D. A trip with a coincident loss of off-site power;
After reaching 100% power following a four day mini-outage.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to OPM O-6 (p34; Rev 5), available decay heat is a function of two things; (1) past power history, and (2) time since reactor trip. Therefore, the highest EFW flowrate will be experienced with the ten day run at 100%, rather than the trip upon arrival at 100%, after a short outage. According to TQ-TM-104-424-C001 (p28; Rev 6) and Technical Specification 3.04 (p3-26b; Amendment 242), the most demanding design basis event requiring EFW is a loss of normal feedwater (LOFW) with off-site power available requiring EFW flow to overcome the additional heat input from the RCPs.

B. **Incorrect.** 1st part wrong, 2nd part correct. On the LOOP the RCPs will trip. Therefore this event will be a lower heat load that the EFW System will need to overcome. This is plausible because the operator may incorrectly believe that additional EFW is needed to establish Natural Circulation in the RCS.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may confuse the concepts of Decay Heat Load with xenon production or transients. If the operator does this the operator may believe that Decay Heat load which was peaking during the shutdown, and then produced as the power level increased, peaked again following the trip, from a higher level than would have occurred following a similar trip from a ten day (equilibrium) at power run.

D. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may confuse the concepts of Decay Heat Load with xenon production or transients (See C); and because the operator may incorrectly believe that additional EFW is needed to establish Natural Circulation in the RCS (See B).

Technical Reference(s): OPM O-6 (p34; Rev 5) (Attach if not previously provided)
TM-TM-104-424-C001 (p28; Rev 6)
Technical Specification 3.04 (p3-26b; Amendment 242)

Proposed References to be provided to applicants during examination: None

Learning Objective: 424-GLO7 (As available)

Question Source: Bank # IR-220-GLO-8-Q01
Modified Bank # (Note changes or attach parent)
New

Question History: Moved to ILT 05-01 Comp Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content:	55.41	5
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge the operational implications of the concept of decay heat sources (i.e. heat load is a function of power operation and RCP status) and magnitude (i.e. direct result of power history) as it applies to the EFW.

The question is at the Memory cognitive level because the operator must recall bits of information to answer the question (i.e. the magnitude of decay heat load is dependent upon power history, and that higher power history produces a higher decay heat rate at the point of trip, a LOOP will result in a trip of the RCPs and limit the post trip heat level that must be removed from the RCS).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	015	K2.01	
	Importance Rating	3.3		

Knowledge of bus power supplies to the following: NIS channels, components, and interconnections

Proposed Question: RO Question # 29

The plant is shutdown.

Maintenance of 120V Vital AC Distribution Panel VBB is scheduled for an outage in which the bus is expected to be de-energized for one shift.

Which ONE (1) of the following identifies the nuclear instrumentation that will be unavailable during this maintenance?

A. Intermediate Range NI-4; AND
Power Range NI-5.

B. Intermediate Range NI-4; AND
Power Range NI-6.

C. Source and Wide Range NI-12/12A; AND
Power Range NI-6.

D. Source and Wide Range NI-12/12A; AND
Power Range NI-5.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because every Nuclear Instrument channel is provided power from a 120V Vital AC Distribution Panel. However, according to TQ-TM-104-623-C001 (p25; Rev 1), 120V Vital AC Distribution Panel VBD provides power to Intermediate Range NI-4, and 120V Vital AC Distribution Panel VBA provides power to Power Range NI-5.

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because every Nuclear Instrument channel is provided power from a 120V Vital AC Distribution Panel. However, according to TQ-TM-104-623-C001 (p25; Rev 1), 120V Vital AC Distribution Panel VBD provides power to Intermediate Range NI-4.

C. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-623-C001 (p25; Rev 1), 120V Vital AC Distribution Panel VBB provides power to Source and Wide Range NI-12/12A and Power Range NI-6.

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because every Nuclear Instrument channel is provided power from a 120V Vital AC Distribution Panel. However, according to TQ-TM-104-623-C001 (p25; Rev 1), 120V Vital AC Distribution Panel VBA provides power to Power Range NI-5.

Technical Reference(s): TQ-TM-104-623-C001 (p25; Rev 1) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of bus power supplies to the NIS channels, components, and interconnections (i.e. which instruments are powered from which Vital Buses).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. what nuclear instruments are powered by VBB).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	045	K1.06	
	Importance Rating	2.6		

Knowledge of the physical connections and/or cause-effect relationships between the MT/G system and the following systems: RCS, during steam valve test

Proposed Question: RO Question # 30

Plant conditions:

- The plant is at 100% power.
- ICS is in AUTO.
- Turbine Valve testing is in progress in accordance with OP-TM-301-301, Turbine Valve Partial Stroke Test.

Plant event:

Main Turbine Stop Valve #1 fails completely CLOSED.

Which ONE (1) of the following describes how the magnitude of the Loop ΔT Error (i.e. SG A/B Load Ratio) will INITIALLY trend, AND the power level to which the plant must be lowered?

A. Increase;
70%.

B. Increase;
90%.

C. Remains constant;
70%.

D. Remains constant;
90%.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. The operator may incorrectly believe that plant power must be reduced to 70%. According to OP-TM-301-000 (p7; Rev 13), there are two additional power restrictions on an inadvertent valve closure during testing. One restriction applies to a Turbine Control Valve closure, and the other if two Combined Intermediate Valves were to inadvertently close. If the Turbine Control Valve goes closed during testing, power must be reduced to limit the remaining Control Valves to < 85% open, which means that power must be reduced to about 85-90%. However, this is NOT based on flow imbalance, but on the HP Rotor vibration concerns that it produces. Secondly, if two CIVs are closed during testing, meaning that a follower valve closed with its leader when it was being tested, plant power must be reduced to 70% based on MSR operational concerns.

B. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-621-000 (p15; Rev 1) Attachment 7.2, the SG A/B Load Ratio is the ΔTc Error signal, and obtained by subtracting ΔTca from ΔTcb. According to TQ-TM-104-301-C001 (p10-11, 31; Rev 2), there are four Main Steam lines, two each from each OTSG. Each Steam line has a Turbine Stop Valve which combines into a common header downstream of these valves, but upstream of the four Turbine Control Valves. This is reflected in Drawing 302-011, Main Steam. When the Turbine Stop Valve closes, the steam flow in the other three Main Steam lines, one from the A OTSG and two from the B OTSG will increase. When steam flow increases from the B OTSG, Loop B Tcold will lower. On the other hand, in the A OTSG, one steam line has increased steam flow, but the other is completely shut off resulting in an overall reduction of steam flow from the A OTSG, and an increase in Loop A Tcold. Since the Loop Tcolds are changing in opposite directions, the magnitude of the ΔT Error will increase. According to OP-TM-301-000 (p6; Rev 13) if a Turbine Stop Valve goes closed, plant power must be reduced to < 90% because the excessive steam flows from the opposite OTSG can contribute to tube cracking.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may confuse the Turbine Stop Valves and the Turbine Control Valves (See D); and operator may incorrectly believe that plant power must be reduced to 70% (See A).

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may confuse the Turbine Stop Valves and the Turbine Control Valves. If so, the operator may incorrectly assume that steam flow to all four steam lines is equalized and simply flowing into the Turbine through three valves instead of four. If this were so, there would be no difference in the steam flows coming from each OTSG, the heat transfer rates would be the same, and Loop Tcolds would be equalized, resulting in the ΔT Error remaining constant.

Technical Reference(s): OP-TM-621-000 (p15; Rev 1) (Attach if not previously provided)
OP-TM-301-000 (p6-7; Rev 13)
Drawing 302-011, Main Steam

Proposed References to be provided to applicants during examination: None

Learning Objective: 301-GLO3, 9 and 10 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4
55.43

Comments:
The KA is matched because the operator must demonstrate knowledge of the physical connections between the MT/G system and the RCS, during steam valve test (i.e. A Turbine Stop Valve failing closed will result in a steam imbalance from the OTSGs)

The question is at the Comprehension/Analysis cognitive level because the operator must assemble three pieces of information (i.e. that (1) how does reduced steam rates from an OTSG affect the RCS, the (2) physical design of the Main Steam/Turbine System including where the steam chest cross connect is located, and (3) how does the system respond when an inadvertent valve closure occurs during testing); and draw a conclusion (i.e. that a steam flow imbalance exists); and then know to what power level, power must be reduced.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	068	K5.04	
	Importance Rating	3.2		

Knowledge of the operational implication of the following concepts as they apply to the Liquid Radwaste System: Biological hazards of radiation and the resulting goal of ALARA

Proposed Question: RO Question # 31

Which ONE (1) of the following actions or precautions associated with the Liquid Radwaste System is performed to ensure lower radiation levels to site personnel and the ability to achieve ALARA goals?

A. Ensuring liquid stored in a RCBT has a boric acid concentration of less than 6% by weight.

B. Maintaining temperature within specified limits for radwaste flow through cation demineralizers.

C. Ensuring timely processing of Chem Cleaning Building Liquid Radwaste Holding Tank CC-T-1 contents using the Liquid Waste Evaporators.

D. Ensuring Evaporator Feed Tank level is maintained above 12 inches during a continuous evaporator run.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because this is a precaution and the operator may incorrectly believe that the storing of high BA concentrated solution in the RBCT is based on preventing increased radiation levels. However, according to OP-TM-232-000 (p2; Rev 5) the RCBTs shall not be used to store solution containing more than 6% by weight of boric acid in any circumstance to prevent boron crystallization. The operator may believe that the crystallized Boric Acid will somehow result in increased radiation levels.

B. **Incorrect.** This is plausible because the operator may incorrectly believe that exceeding temperature limits through the demineralizer will increase radiation levels. The reason for this precaution is to prevent heat damage to the demineralizers, which will decrease efficiency and create flow restrictions.

C. **Correct.** According to TQ-TM-104-232-C001 (p114; Rev 3), Surveillance Procedure 1301-11.1, Operability of the Liquid Radwaste Treatment System, is used to help ensure availability of the WDL System for operation as required by ODCM, Section 2.2.1.3, Liquid Radwaste Treatment System, CONTROL. The Acceptance Criteria is interpreted to include a minimum level of operability of the system is to have one of the waste streams available for use, which includes one WDL Evaporator. According to TQ-TM-104-232-C001 (p107-108; Rev 3), in 2002, the untimely processing of radioactive liquid waste resulted in slightly elevated dose rates and personnel exposures. Individuals who routinely worked near the pre-evaporation holding tank were determined to have increased exposure to radiation. A survey of the office space located near the tank found slightly elevated dose rates in the office space that were determined to result from water with higher than normal radioactivity levels stored in the tank in the liquid radwaste processing building. The water was transferred to the tank following the return of the reactor to service after the fall 2001 outage. The cause of the higher radiation levels was Maintenance and Operations issues with the operability of the liquid radwaste evaporator preventing the processing of water in the holding tank in a timely manner.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that a low feed tank level may result in higher radiation levels (Less shielding). The reason for this precaution is to prevent uncovering heaters.

Technical Reference(s): TQ-TM-104-232-C001 (p73, 115,116; Rev 3) (Attach if not previously provided)

OP-TM-232-000 (p2; Rev 5)

Proposed References to be provided to applicants during examination: None

Learning Objective: 232-GLO7, 13, 14 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge of the operational implication of the concept of the biological hazards of radiation and the resulting goal of ALARA as they apply to the Liquid Radwaste System (i.e. operational methods that are based in reducing radiation exposure).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. the basis for various operational restrictions).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	041	A1.02	
	Importance Rating	3.1		

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure

Proposed Question: RO Question # 32

Plant conditions:

- Plant is at 100% power.
- All Systems are in AUTO.

Which ONE (1) of the following correctly completes the statement below?

On a reactor trip the Turbine Bypass Valves open to control Turbine Header Pressure at ____1____. The Atmospheric Dump Valves will start to open at ____2____ and be fully open at ____3____.

A. (1) 960 psig
(2) 1010 psig
(3) 1026 psig

B. (1) 960 psig
(2) 1026 psig
(3) 1052 psig

C. (1) 1010 psig
(2) 1026 psig
(3) 1052 psig

D. (1) 1010 psig
(2) 1052 psig

(3) 1060 psig

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the Turbine Bypass valves will open at 960 psig while at 100% power, and the other two setpoints are associated with system operation. According to TQ-TM-104-411-C001 (p28-31; Rev 3), During normal operation the setting is 885 psig, corresponding to 47.5% on the dial. Setpoint biases applied by the ICS are 10, 75 and 125, resulting in normal automatic adjustable control setpoints of 895, 960 and 1010 psig. The 960 psig setpoint is used when ICS ULD > 15% to prevent inadvertent Turbine Bypass Valve operation during normal plant transients. The 1010 psig setpoint is used to prevent excessive Pressurizer Level decrease on reactor trip.

B. **Incorrect.** This is plausible because the Turbine Bypass valves will open at 960 psig while at 100% power, and the other two setpoints are correct (See A).

C. **Correct.** According to TQ-TM-104-621-C001 (p137-140; Rev 2) and OP-TM-411-000 (p8; Rev 10), immediately following the reactor trip, the turbine header pressure setpoint will shift to 125 psig bias. Header pressure will be controlled by Turbine Bypass Valves at 1010 psig after the initial surge to approximately 1100 psig, where the Turbine Bypass Atmospheric and Main Steam Relief Valves will respond. The Turbine Bypass Atmospheric Valves begin opening at 1026 psig and will be full open at 1052 psig. The nine (9) main steam relief valves per OTSG will relieve at the following setpoints: MSV-17 A, B, C, D at 1050 psig, MSV-18 A, B, C, D at 1060 psig, MSV-19 A, B, C, D at 1080 psig, MSV-20 A, B, C, D at 1092.5 psig and MSV-21 A, B at 1040 psig.

D. **Incorrect.** This is plausible because the two setpoints offered as ADV start to open and full open setpoints are setpoints associated with system operation. For instance, according to TQ-TM-104-621-C001 (p137-140; Rev 2), the ADVs will be full open at 1052 psig and the 2nd lowest set MSSV.

Technical Reference(s):	TQ-TM-104-621-C001 (p137-140; Rev 2) TQ-TM-104-411-C001 (p29, 31; Rev 3) OP-TM-411-000 (p9; Rev 11)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	411-GLO-3 621-GLO-8	(As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:
The KA is matched because the operator must demonstrate ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including steam pressure (i.e. pressures at which SDS valves open).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. what are the setpoints that the valves control at during a reactor trip).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	016	K3.02	
	Importance Rating	3.4		

Knowledge of the effect that a loss or malfunction of the NNIS will have on the following: PLCS

Proposed Question: RO Question # 33

Plant conditions:

- The plant is at 80% power.
- Pressurizer Level Control is selected to RC1-LT1 on CC.
- A leak is occurring on the fixed reference leg of RC1-LT1.
- Indicated Pressurizer Level is changing at a rate of 200" per minute.

Which ONE (1) of the following describes the effect on Makeup Tank level and Makeup flow?

A. Makeup Tank Level Drops as Makeup flow rises.

B. Makeup Tank Level Rises as Makeup Flow lowers.

C. No effect on Makeup Tank Level or Makeup flow; SASS will automatically select the redundant transmitter.

D. No effect on Makeup Tank Level or Makeup flow; actual RCS inventory and pressurizer volume is unchanged.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. A loss of the pressurizer level reference leg will cause indicated pressurizer level to read high (increased DP between variable and reference legs) If pressurizer level reads high, Makeup flow will begin to be reduced. With Letdown to the MU Tank constant and reduced Makeup flow, Makeup Tank level will rise. This option is plausible because the applicant may confuse reference and variable legs, and come to this conclusion. This option describes a variable leg failure.

B. Correct. A loss of the pressurizer level reference leg will cause indicated pressurizer level to read high (increased DP between variable and reference legs) If pressurizer level reads high, Makeup flow will begin to be reduced. With Letdown to the MU Tank constant and reduced Makeup flow, Makeup Tank level will rise.

C. Incorrect. Plausible because a rapid failure of most non-nuclear instrumentation will actuate SASS. Pressurizer level control is not equipped with SASS so this option is wrong.

D. Incorrect. Plausible because it is true that pressurizer volume and mass is unchanged (true statement) but Makeup Flow is controlled by indicated pressurizer level, not actual.

Technical Reference(s): TQ-TM-104-624-C001, (p35) Rev 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-11 (As available)

Question Source: Bank # QR-624-GLO-11-Q10

Modified Bank # (Note changes or attach parent)

New

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

KA matched because the effect of a loss of non-nuclear instrumentation (PZR level) is evaluated as to effect on the pressurizer level control system. (Makeup flow)

Comprehension level because the applicant must determine the effect on an operating system by deducing the effect of a failure on part of the system

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	001	A3.02	
	Importance Rating	3.7		

Ability to monitor automatic operation of the CRDS, including: Rod height

Proposed Question: RO Question # 34

Plant conditions:

- The plant is at 100% power.
- All systems are in AUTO.

Event:

- The Tavg input to the Reactor Demand H/A Station (RC 12 TAS) fails HIGH.
- During the rod movement of Group 7, an AMBER lamp below Rod 1 is LIT.
- There are <u>NO</u> amber fault lights on Diamond Control Panel.
- Without any operator action the plant stabilizes at 90% power.

Which ONE (1) of the following identifies the status of the Group 7 Rods?

A. They are LOWER than they were before the event; AND
Rod 1 is between 7 and 9 inches from the Group average height.

B. They are LOWER than they were before the event; AND
Rod 1 is more than 9 inches from the Group average height.

C. They are HIGHER than they were before the event; AND
Rod 1 is between 7 and 9 inches from the Group average height.

D. They are HIGHER than they were before the event; AND
Rod 1 is more than 9 inches from the Group average height.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-621-C001 (p163; Rev 2), if T_{avg} fails high although actual T_{avg} has not changed, T_{avg} modification to reactor demand causes control rods to drive into the core and feedwater demand to rise, lowering T_{avg} and reactor power. The end result will find power at 90% with indicated T_{avg} at 579°F but actually below 579°F. Therefore, Control Rods will drive inward and be LOWER than they were at the start of the event. According to OP-TM-622-000 (p7; Rev 2), the asymmetric rod alarm is indicated by amber FAULT lamp on PI Panel and by main annunciator, CRD PATTERN ASYMETRICAL. The alarm is caused by a rod being misaligned 7 inches or more from the group average position. According to OP-TM-MAP-G0201 (p1of1; Rev 1), the setpoint of the CRD PATTERN ASYMETRIC annunciator is any rod > 7" (5%) from its absolute average position.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to OP-TM-622-000 (p7; Rev 2) the asymmetric rod fault is indicated by ASYM RODS lamp (Amber) on CRD Control Panel, and results from a rod being misaligned from the group average by 9 inches. According to OPM F-01 (p48; Rev 7) a rod that is 9 inches is 6.5% from its group average. The operator may incorrectly believe that the amber light indicates that the rod is 9 inches or 6.5% from the group average.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the failure of the Tavg input to the Reactor Demand H/A Station will cause an upset to the controller which is in AUTO, requiring rods to move inward or outward. The operator may incorrectly diagnose that the control rods move outward in this event.

D. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible the operator may incorrectly diagnose that the control rods move outward in this event (See C); and because the operator may incorrectly believe that the amber light indicates that the rod is 9 inches or 6.5% from the group average (See B).

Technical Reference(s): TQ-TM-104-621-C001 (p165; Rev 2) (Attach if not previously provided)
OP-TM-622-000 (p7; Rev 3)
OP-TM-MAP-G0201 (p1of1; Rev 1)
OPM F-01 (p48; Rev 7)

Proposed References to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

The KA is matched because the operator must demonstrate the ability to monitor automatic operation of the CRDS (i.e. which way the rods move in a control system failure), including rod height (i.e. identify the cause of an asymmetric rod height failure by correctly interpreting system indications).

The question is at the Comprehension/Analysis cognitive level because the operator must assemble several pieces of information (i.e. (1) how the Reactor Demand Control works, (2) how Reactor demand control is affected by an signal failure, (3) the meaning of an AMBER light on the Control Rod PIP); and draw a conclusion (i.e. that the rods have lowered, and that one rod is between 5-6.5% of the Group Average Height).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	029	A4.04	
	Importance Rating	3.5		

Ability to manually operate and/or monitor in the control room: Containment evacuation signal

Proposed Question: RO Question # 35

The plant is in COLD SHUTDOWN.
Source Range indicates 20 CPS.

Which ONE (1) of the following describes the MINIMUM rise in Source Range Count Rate that will automatically actuate the Reactor Building Evacuation Alarm, AND the location(s) that it can be MANUALLY initiated from the Control Room?

A. 10 CPS; AND
MANUALLY initiated ONLY from Panel PL.

B. 10 CPS; AND
MANUALLY initiated from BOTH Panels PL and PRF.

C. 20 CPS; AND
MANUALLY initiated from BOTH Panels PL and PRF.

D. 20 CPS; AND
MANUALLY initiated ONLY from Panel PL.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the reset of the alarm is 30 CPS (20 CPS + 10 CPS).

B. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the reset of the alarm is 30 CPS and because the operator may incorrectly believe that the alarm can manually be initiated from PRF (Only Automatically by SR) (See C).

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to TQ-TM-104-623-C001 (p28; Rev 1), Source Range (NI-11 and NI-12) will automatically actuate the RB Evacuation Alarm if enabled on PRF. However, the Source Range instruments will only actuate the alarm automatically when the instrument is reading greater than 40 CPS (20 CPS + 20 CPS) . RB Evacuation Alarm can only be manually initiated from Control Room Panel PL. The operator may incorrectly believe that the alarm can manually be initiated from PRF.

D. **Correct.** 1st part correct, 2nd part correct. The automatic setpoint is 40 CPS on PRF and the alarm is manually initiated by pressing and releasing the pushbutton on Control Room Panel PL. At that point, the signal will sound throughout the plant for about 30 seconds and then automatically stop.

Technical Reference(s): TQ-TM-104-623-C001 (p28; Rev 1) (Attach if not previously provided)

1105-12 (p13-14; Rev 25)

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO5 824-GLO5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to manually operate and monitor the containment evacuation signal in the control room (i.e. be able to distinguish the tone from other site emergency alarms, and identify the location for manual operation within the Control Room).

The question is at the Comprehension cognitive level because the operator must calculate what rise in Source Rate Counts would actuate the alarm automatically. As well as recall where can the RB Evacuation Alarm be operated from.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	033	A2.03	
	Importance Rating	3.1		

Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal spent fuel pool water level or loss of water level

Proposed Question: RO Question # 36

Plant conditions:

- Reactor Coolant System is in Refueling Shutdown condition.
- The Fuel Transfer Canal has been filled.
- Fuel Transfer Canal isolation Valves FH-V-1A and FH-V-1B are Closed.

Event and subsequent conditions.

- Loss of Offsite Power (LOOP).
- Spent Fuel Pool Temperature is 135°F and rising slowly.
- Alarm PLB-2-9, "Spent Fuel Pool A Level Lo", is illuminated.
- Alarm PLB-2-10, "Spent Fuel Pool B Level Lo", is illuminated.
- Spent Fuel Pool level is 342' and lowering rapidly.

Which ONE (1) of the following actions is required in accordance with OP-TM-AOP-035, "Loss of Spent Fuel Cooling"?

A. Place both trains of Spent Fuel Pool Cooling in service to increase cooling.

B. Open FH-V-1A and FH-V-1B to equalize level and temperature with the refueling cavity.

C. Commence makeup to the Spent Fuel Pool from the BWST to restore boron concentration.

D. Initiate Attachment 1 for makeup to the Spent Fuel Pool from Fire Service Water to restore level.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** Plausible since both trains would be started if temperature is >190 °F and pool level was not <343' 6". Wrong because this action is currently not required by procedure.

B. **Incorrect.** Plausible since opening these valves would transfer some water to the Spent Fuel Pool; however the correct action is to close them to isolate the leak to the Spent Fuel Pool and maintain cavity level.

C. **Incorrect.** Plausible since this is the procedure for transferring water to the Spent Fuel Pool; however it utilizes the RCBTs, not the BWST transfer pumps, which are not available in LOOP.

D. **Correct.** Procedure states "if at any time level <343' 6 - - - "

Technical Reference(s):	OP-TM-AOP-035, Loss of Spent Fuel Pool Cooling (p1) Rev 2	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: TQ-AA-210-3203 Obj 4 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: NA Last NRC Exam: TMI 2007 RO

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	4
	55.43	

Comments:

The KA is matched because the operator must demonstrate the ability to take action for low spent fuel pool level in accordance with plant procedures

The question is at the Comprehensive/Analysis cognitive level because the operator must demonstrate an understanding of the action required and determine corrective action based upon a combination of temperatures and levels

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	072	2.4.50	
	Importance Rating	4.2		

Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question: RO Question # 37

Plant Conditions:

- RM-G-20, RCDT Discharge Gamma Monitor, has been removed from service in accordance with 1105-8, Radiation Monitoring System, for minor maintenance on the digital display.
- The maintenance has been completed and now the operator is getting ready to restore RM-G-20 to service using the same procedure.

Which ONE (1) of the following correctly completes the statements below?

While restoring the instrument to service, the RM-G-20 Interlock Defeat Switch is returned to NORMAL ____1____. The MOST ACCURATE method of verifying RM-G-20 Alarm setpoints is by ____2____.

A. (1) just prior to re-energizing the instrument
(2) verifying that I&C has completed the appropriate calibration procedure

B. (1) just prior to re-energizing the instrument
(2) by pressing the ALERT and ALARM pushbuttons and reading the instrument

C. (1) after the instrument has been re-energized and warmed up
(2) verifying that I&C has completed the appropriate calibration procedure

D. (1) after the instrument has been re-energized and warmed up
(2) by pressing the ALERT and ALARM pushbuttons and reading the instrument

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to MAP C-2-1 (p1of3; Rev 35), the RM-G-20 is equipped with an interlock defeat switch, and the RAD MON INTERLOCK BYPASS annunciator will alarm when an instrument's Interlock Defeat Switch is placed in the Defeat position. The operator may incorrectly believe that the interlock defeat switch must be in the NORMAL position whenever the instrument is energized to ensure that the associated automatic functions of the instrument are in effect.

B. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that the interlock defeat switch must be in the NORMAL position whenever the instrument is energized (See A); and the operator may incorrectly believe that they can verify the setpoints by depressing the ALERT and ALARM pushbuttons (See D).

C. **Correct.** 1st part correct, 2nd part correct. According to 1105-8 (p43-44; Rev 83), Section 3.5, to remove the instrument from service the operator must place interlock mode switch in DEFEAT, place the monitor in OFF, and then, when appropriate, return monitor to service IAW Section 3.1. According to 1105-8 (p9-10; Rev 84), Section 3.1.4, to return the instrument to service will ensure the bypass switch on the control panel "PRF" is in the DEFEAT position to preclude automatic interlock actuation while placing the monitor in service. Then, the operator will Press then release the ON/OFF pushbutton, and Allow approximately 5 minutes for the electronics to warm up for guaranteed accuracy. Next, the operator will perform a source check, and verify the ALERT and ALARM setpoints by either directing I&C to perform the appropriate setpoints adjustment procedure, or by verifying that the appropriate I&C procedure is on file. The operator will then restore the bypass switches to "normal" position to reinstate the interlock function. Therefore, the Interlock Defeat Switch is restored to NORMAL after the instrument has been re-energized and warmed for 5 minutes, and the setpoints verification is an administrative verification with I&C.

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to 1105-8 (p7-8; Rev 84), these instruments have the capability of the operator pressing the ALERT and ALARM pushbuttons to reveal their respective setpoints. Therefore, the operator may incorrectly believe that they can verify the setpoints by depressing these pushbuttons. However, in the power up procedure for these instruments a caution is provided which states "Reading Alarm setpoints from the meter face by pressing the alarm pushbutton does not give an accurate indication of actual setpoint. This feature is not calibrated and should not be relied upon for accurate setpoint information." Therefore, this method should never be relied upon to verify the setpoints.

Technical Reference(s): 1105-8 (p7-10, 43-44; Rev 84) (Attach if not previously provided)
MAP-C (p1of3; Rev 35)

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO3, 10 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43

Comments:
The KA is matched because the operator must demonstrate the ability to verify ARM system alarm setpoints (i.e. method used) and operate controls (i.e. when to operate Interlock Defeat Switch when energizing instrument) identified in the alarm response manual (i.e. MAP C-1-1).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. The operation of the Interlock Defeat Switch during power up, the verification of setpoints by depressing the ALERT/ALARM pushbuttons is procedurally prohibited).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	035	K6.03	
	Importance Rating	2.6		

Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: S/G level detector

Proposed Question: RO Question # 38

Plant conditions:

- 100% power.
- All major controls in AUTO.
- The DP Cell diaphragm ruptures on the A OTSG Operating Range Level transmitter causing an instantaneous change in the level signal to the channel selected as the controlling input.

Assuming no operator action, which ONE (1) of the following correctly describes OTSG level response?

A. Level transmitter output fails HIGH;
Actual will level remain the same due to SASS transfer.

B. Level transmitter output fails HIGH;
Actual level would decrease until the low level limit is reached.

C. Level transmitter output fails LOW;
Actual level will remain the same due to SASS transfer.

D. Level transmitter output fails LOW;
Actual level would increase until the high level limit is reached.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to OPM B-4 (p16-17; Rev 9), the Operating Range OTSG Level transmitters are Foxboro force balance type with diaphragm sensors. The transmitters measure the ΔP between the OTSG 96 inch and 388 inch level taps, with a corresponding level indication of 0-100%, using a wet reference leg system. When the detector is detecting a low level, the measured ΔP is at its highest. Likewise, when the detector is detecting a high level, the measured ΔP is at its lowest. Therefore, when the diaphragm ruptures, and the ΔP is equalized, the instrument fails HIGH. According to OP-TM-104-624-C001 (p6-7; Rev 2), the OTSG "A" Operate Level instruments (LT-1044/1045, LT-1040/1041) are monitored by SASS, the Smart Automatic Signal Selector. According to OP-TM-104-624-C001 (p24; Rev 2), If SASS senses one of the parallel instruments more than 3% of full scale away from the other, it will announce a MISMATCH (MAP H-3-2). An AUTOMATIC transfer will not occur if a SASS monitored channel is in MISMATCH. If SASS senses one of the parallel instruments changing more than 8%/sec. (SASS ACTUATION), it will automatically select the other instrument and provide a computer alarm. Therefore, in this instance the SASS will sense the deviation, auto select another controlling instrument, and actual level will remain the same.

B. **Incorrect.** 1st part correct, 2nd part wrong. According to OP-TM-104-624-C001 (p7-10; Rev 2), there are at least 15 instruments or sets of instruments in the NNI System that are NOT monitored by the SASS. The operator may incorrectly believe that the Operating Range OTSG Level Instrument is one of them.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may easily confuse the failure mechanism on the Differential Pressure wet reference leg OTSG Level transmitters. The operator may incorrectly believe that a 0 ΔP will indicate a LOW level indication.

D. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that a 0 ΔP will indicate a LOW level indication (See C); and because the operator may incorrectly believe that the Operating Range OTSG Level Instrument is NOT monitored by SASS (See B).

Technical Reference(s): OPM B-4 (p16-17; Rev 9) (Attach if not previously provided)
OP-TM-104-624-C001 (p6-7, 24; Rev 2)

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO5
624-GLO2, 3 (As available)

Question Source:	Bank #	WTS 58507	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: NA Last NRC Exam: Davis-Besse 2005

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:
The KA is matched because the operator must demonstrate knowledge of the effect of a malfunction on a S/G level detector will have on the OTSG (i.e. what happens when controlling OTSG Level detector fails).

The question is at the Comprehension/Analysis cognitive level because the operator must assemble three pieces of information (i.e. (1) how does the level detector work, (2) what happens when the level detector diaphragm fails, and (3) whether or not SASS monitors Operating Range of OTSG); and draw a conclusion (i.e. that the level will fail high and that the channel will be deselected for control by SASS).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015	AK1.03
	Importance Rating	3.0	

Knowledge of the operational implications of the following concepts as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): The basis for operating at a reduced power level when one RCP is out of service

Proposed Question: RO Question # 39

Plant conditions:

- The plant is at 100% Power.
- RC-P-1D #1 Seal Leakoff is below the maximum limit but has been slowly rising for several shifts.
- OP-TM-AOP-040, RCP #1 Seal Failure, has been implemented.
- A conservative decision has been made to stop RC-P-1D.

Which ONE (1) of the following correctly completes the following statement?

Prior to stopping RC-P-1D, maximum reactor power must be less than ____1____% to prevent an automatic reactor trip that protects against exceeding ____2____ assumption limits.

A. (1) 75
(2) linear heat rate

B. (1) 75
(2) departure from nucleate boiling ratio

C. (1) 49
(2) linear heat rate

D. (1) 49
(2) departure from nucleate boiling ratio

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because Technical Specification 2.3 (p2-6&7; Amendment 262) Linear Heat Rate (i.e. power peaking in KW/ft) is discussed in TS Basis as a limit associated with other trips. The operator may incorrectly believe that the power level reduction is made to ensure LHGR limits rather than DNB limits are maintained.

B. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-AOP-040 (p3; Rev 0), the Rx power limit is <75% for three RCP operation. According to Technical Specification 2.3 (p2-6; Amendment 262), the power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power to flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1 are given in the COLR (p36; Rev 5). Therefore, the power level is reduced to protect against exceeding DNB limits.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that 49% is power level for the stated conditions (See D); and because the operator may incorrectly believe that the power level reduction is made to ensure LHGR limits rather than DNB limits are maintained (See A).

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to OP-TM-AOP-040 (p3; Rev 0), 49% is the limit for one RCP operating in each loop. The operator may incorrectly believe that 49% is power level for the stated conditions.

Technical Reference(s): Technical Specification 2.3 (p2-6; Amendment 262) (Attach if not previously provided)
OP-TM-AOP-040 (p3; Rev 0)

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-10 (As available)

Question Source: Bank #

	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	NA	Last NRC Exam: NA	

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	5
	55.43	

Comments:
The KA is matched because the operator must demonstrate knowledge (i.e. the power limit and the basis for that limit when an RCP must be stopped at power) of the operational implications of the concept of the basis for operating at a reduced power level when one RCP is out of service as they apply to the Reactor Coolant Pump Malfunctions Loss of RC Flow.

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. maximum power levels for stopping a pump, why the power level must be reduced).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	007	EK1.04	
	Importance Rating	3.6		

Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Decrease in reactor power following reactor trip (prompt drop and subsequent decay)

Proposed Question: RO Question # 40

Plant Conditions:

- Reactor trip from 100% power.
- Group 6 Rod 1 stuck full out.

Which ONE (1) of the following correctly describes the Unit Reactor Operator's observations (1) power 30 seconds after trip, (2) power decay FIVE minutes after the trip?

A. (1) Power reading <5%
(2) Normal negative 1/3 DPM SUR

B. (1) Power reading <5%
(2) Less negative than 1/3 DPM SUR

C. (1) Power reading >5%
(2) Normal negative 1/3 DPM SUR

D. (1) Power reading >5%
(2) Less negative than 1/3 DPM SUR

Proposed Answer: A

Explanation (Optional):

A. **Correct.** One rod stuck will not prevent S/D, and decay is only affected by long lived daughters after 5 minutes.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the SUR it may be believed that the stuck rod would raise the reactivity and hold power up.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because 2nd party is correct but it may be believed the stuck rod would hold power above normal S/D for initial drop.

D. **Incorrect.** 1st part wrong, 2nd part wrong, plausible for reasons stated above.

Technical Reference(s): OP-TM-EOP-0011 rev 1 Step 2.2 second paragraph (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 823-GLO5, 10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. when is the reactor on negative one-third dpm SUR, what are the effects of a stuck rod) of the operational implications of the concept of the decrease in reactor power following reactor trip (prompt drop and subsequent decay) as they apply to the reactor trip.

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate understanding of the reactor and the nuclear instrumentation system works on a reactor trip, with stuck rod.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022	AK1.03
	Importance Rating	3.0	

Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Pump Makeup: Relationship between charging flow and PZR level

Proposed Question: RO Question # 41

Plant conditions:

- The plant is at 100% power with the ICS in full automatic.
- The operating Makeup Pump trips on motor overload.
- MAP G-2-5 PZR LEVEL HI-LO annunciator has alarmed.
- Pressurizer level is 196 inches and slowly lowering.

Which ONE (1) of the following identifies actions that the operator would be expected to take while restoring a Makeup Pump to operation?

A. De-energize all Pzr Heaters; AND
CLOSE MU-V-33A-D, Seal Leakoff Valves.

B. CLOSE MU-V-3, Letdown Isolation Valve; AND
CLOSE MU-V-33A-D, Seal Leakoff Valves.

C. De-energize all Pzr Heaters; AND
Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves.

D. CLOSE MU-V-3, Letdown Isolation Valve; AND
Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that that the heaters will need to be deenergized due to low pressurizer level (See C); and because the operator may incorrectly believe that procedures direct the closure of MU-V-33A-D to save RCS inventory from being lost (See B).

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that procedures direct the closure of MU-V-33A-D to save RCS inventory from being lost. While their closure may save inventory, the closure of these valves would also eliminate their cooling water, and the seals will overheat.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that that the heaters will need to be deenergized due to low pressurizer level. According to OP-TM-MAP-G0305 (p1of2; Rev 2), at 80 inches in the Pzr the Pzr Heaters will automatically trip. However, this level has not been reached, and the need to de-energize the Pzr heaters is not identified in either EOP-10 Guide 9, or AOP-041.

D. **Correct.** 1st part correct, 2nd part correct. In this situation there is no makeup flow and therefore the RCS will start to lose inventory, and Pzr level will decrease. The source of inventory loss is both normal Letdown (45 gpm), and the RCP #1 Seal Leakoff which is normally about 3 gpm. According to OP-TM-MAP-G0205 (p1of2; Rev 3), if Pzr level can NOT be restored with automatic or manual control of MU-V-17, then initiate OP-TM-EOP-010, Guide 9. According to OP-TM-EOP-010 (p21; Rev 11), if the Pzr level is low the operator will be directed to check to see if a MU Pump is running. If there is no MU Pump running the operator will be directed to isolate Letdown and initiate OP-TM-AOP-041, Loss of Seal Injection. In accordance with OP-TM-AOP-041 (p1; Rev 4) the operator must ensure that MU-V-32 is in HAND and CLOSED. According to OP-TM-AOP-0411 (p3; Rev 3), this is done to ensure that seal injection is restored in a controlled manner. In accordance with OP-TM-AOP-041 (p3; Rev 4) if at least one MU Pump is NOT running, the operator must ensure that MU-V-17 is CLOSED. Since MU-V-17 would have been in AUTO control previously, and the Pzr Level being low would have driven the valve OPEN, the operator must take the controls to HAND and close the valve. According to OP-TM-AOP-0411 (p3; Rev 3), this is done to prevent uncontrolled flow when the pump is started, and ensure that Pzr level is restored in a controlled manner, rather than a rapid recovery which would otherwise occur. Therefore, the operator will be required to isolate letdown and take both MU-V-17 and 32 to CLOSE during the recovery.

Technical Reference(s): OP-TM-MAP-G0205 (p1of2; Rev 3) OP-TM-MAP-G0305 (p1of2; (Attach if not previously provided)

Rev 2)

OP-TM-AOP-041 (p1, 3; Rev 4)
OP-TM-EOP-010 (p21; Rev 11)
OP-TM-AOP-0411 (p3; Rev 3)

Proposed References to be provided to applicants during examination: None

Learning Objective: A41-PCO-1, 2, and 6
E10-PCO- 1, 2, and 6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. what actions must be taken) of the operational implications (i.e. RCS Inventory is being lost, and action is needed in preparation for MU Pump restart) of the concept of the relationship between charging flow and Pzr level as it applies to the Loss of Reactor Coolant Pump Makeup.

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate understanding of recovery process (i.e. actions must be taken to limit inventory loss, but not at the expense of damaging equipment, actions are needed in preparation for MU Pump restart.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009	EK2.03
	Importance Rating	3.0	

Knowledge of the interrelations between the small break LOCA and the following: S/Gs

Proposed Question: RO Question # 42

A small break LOCA has occurred, and the loops are in the drain down phase.

Which ONE (1) of the following identifies an action that the operator will take to help promote Pool Boiler-Condenser Mode (BCM) Cooling?

A. Manually cycle the PORV.

B. Raise OTSG Level to 75-85%.

C. Verify that Adequate HPI exists.

D. Stop all Reactor Coolant Pumps.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because it is an action that the operator is directed to take within OP-TM-EOP-002, and the operator may incorrectly believe that it is directed to specifically promote Pool BCM Cooling. However, according to OP-TM-EOP-0021 (p8; Rev 1), the PORVs are manually cycled when the RCS heats up and re-pressurizes during the period that BCM is being established. The direction to manually cycle minimizes the number of cycles on the PORV that would otherwise occur if the PORV were allowed to cycle at the automatic lift and blowdown setpoints.

B. **Correct.** According to TMI FSAR (p14.2-25; Rev 19, APRIL 2008), during the loop draining phase, the steam voids that develop in the U-bends can become large enough that the primary liquid level is displaced into the steam generator tube region below the EFW nozzles. The improved primary-to-secondary heat transfer can then be restored, through condensation on the tubes wetted by the EFW. This heat transfer process within a once-through steam generator (OTSG) is referred to as boiler-condenser mode, or BCM, cooling. Later in the loop draining phase, Pool BCM cooling can occur if the RCS tube liquid level decreases below the secondary liquid level. This cooling process will continue if (1) RCS condensation and ECCS injection do not cause the RCS liquid level to increase above the secondary level, (2) the secondary fluid temperature is maintained below the temperature of the steam on the primary side of the OTSG tubes, and (3) the secondary liquid level is high enough that the secondary OTSG thermal center remains several feet above the RCP spillover elevation. According to OP-TM-EOP-0021 (p1; Rev 2), the EOP actions includes directions to ensure that the secondary fluid temperature is maintained below the temperature of the steam on the primary side of the OTSG tubes, and the secondary liquid level is high enough that the secondary OTSG thermal center remains several feet above the RCP spillover elevation. For instance, according to OP-TM-EOP-0021 (p8; Rev 2), Step 4.4. directs the operator to raise OTSG Operating Range Level to 75-85% to prepare the OTSG water levels for BCM heat transfer. Additionally, EFW rather than MFW is the preferred source of feedwater since it is colder water and that will tend to keep the ΔT between the primary and secondary fluids longer. These actions will promote Pool BCM.

C. **Incorrect.** This is plausible because it is an action that the operator is directed to take within OP-TM-EOP-002, and the operator may incorrectly believe that it is directed to specifically promote Pool BCM Cooling. However, according to OP-TM-EOP-0021 (p3; Rev 2), HPI/LPI are actuated to ensure core cooling exists, and according to OP-TM-EOP-0021 (p5; Rev 2), the specific step of verifying that Adequate HPI exists at Step 3.9 is to determine the mitigation strategy. For instance, if SCM is lost and Adequate HPI does not exist, the operator will initiate a Rapid RCS Cooldown which will direct specific steps be taken to promote Pool BCM Cooling.

D. **Incorrect.** This is plausible because it is an action that the operator is directed to take within OP-TM-EOP-002, and the operator may incorrectly believe that it is directed to specifically promote Pool BCM Cooling. However, according to OP-TM-EOP-0021 (p3; Rev 1), the RCPS are stopped to ensure that ECCS can be successful for any size RCS break.

Technical Reference(s):	TMI FSAR (p14.2-25; Rev 19, APRIL 2008) OP-TM-EOP-0021 (p1, 3, 5, 8; Rev 2)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: E02-PCO-1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of the interrelations between the small break LOCA and the OTSGs (i.e. what is Pool BCM and what actions are taken to promote this).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	E05	EK2.1	
	Importance Rating	3.8		

Knowledge of the interrelations between the (Excessive Heat Transfer) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO Question # 43

Plant conditions:

- The CRS directed a MANUAL 1600# ESAS actuation due to a steam break inside the Reactor Building on the A OTSG.
- The operating crew is performing EOP-003, Excessive Heat Transfer.
- Phase 1 and Phase 2 Isolation have been performed on the A OTSG.
- The RCS continues to cool down while the A OTSG inventory is consumed.

Which ONE (1) of the following predicts the direction of the tube to shell differential temperature (TSDT) on the B OTSG, AND the resultant stress on the tubes?

A. Negative; AND
Tensile.

B. Negative; AND
Compressive.

C. Positive; AND
Tensile.

D. Positive; AND
Compressive.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	008	AK2.01	
	Importance Rating	2.7		

Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Valves

Proposed Question: RO Question # 44

Plant Conditions:

- Power was reduced rapidly from 100% to 60% power on Load Dispatcher orders due to a major problem on the grid.
- Pressurizer Spray failed to actuate automatically and RCS Pressure reached 2275 PSIG during the transient.
- Main Annunciator alarm G-1-6, PZR SAFETY OR PORV OPEN (DP) actuated and is still illuminated.
- Main Annunciator alarm G-1-7, PORV OPEN (ACOUSTIC) alarmed and has cleared.
- PPC alarm A0518, RC-RV1A TAILPIPE DELTA TEMPERATURE is in alarm.
- RCS pressure is 2220 PSIG, lowering slowly.
- RCDT pressure is rising slowly.

Which ONE (1) of the following identifies the event that is occurring, and the action required?

A. Relief Valve RC-RV-2 is leaking and a reactor trip is required.

B. Safety Valve RC-RV-1A is leaking and a reactor trip is required.

C. Safety Valve RC-RV-1A is leaking and a reactor trip is <u>not</u> required.

D. Relief Valve RC-RV-2 is leaking and a reactor trip is <u>not</u> required.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may not consider all parameters. Alarm G-1-7, which is specific to the PORV, actuated during the transient. However, the MAP indicates that the acoustic monitor may also actuate if a PZR Safety lifts. The operator may incorrectly believe, based on the actuation of this alarm, that it was the PORV that opened below setpoint. According to OP-TM-PPC-A0517 (p2of3; Rev 2) if the PORV were leaking the course of action would be permissible.

B. **Incorrect.** This is plausible because Tailpipe Delta T indicates the safety valve is open. If the operator does not know the safety valve TS then the next logical choice would be tripping based on safety boundary leakage(incorrect application).

C. **Correct.** According to OP-TM-MAP-G0106 (p1of 2; Rev 3), RC-RV-2 will open if RCS Pressure rises to > 2450 psig, or RC-RV-1A or 1B will open if RCS Pressure rises to > 2500 psig. Since pressure rose only to 2275 psig, it must be concluded that whichever valve opened, it opened below its setpoint. According to OP-TM-MAP-G0106 (p1of 2; Rev 3), RC-RV-1A, RC-RV-1B, or PORV RC-RV-2 opening will cause this alarm. The operator will then be directed to observe the actual associated ΔP indication to determine which valve is OPEN. The procedure indicates that the G-1-7 PORV OPEN (Acoustic) and the tailpipe differential temperatures may also be used to determine which valve is open. According to OP-TM-MAP-G0107 (p1of2; Rev 2), the PORV is typically responsible for causing this alarm, however, the primary safeties may cause this alarm to occur if any actually lift. Since the PPC alarm A0518, RC-RV1A TAILPIPE DELTA TEMPERATURE is still in alarm, it must be concluded that RC-RV-1A is the valve that opened. A reactor trip is not required because pressure is above the reactor trip setpoint.

D. **Incorrect.** This is plausible because the operator may not consider all parameters. Alarm G-1-7, which is specific to the PORV, actuated during the transient. However, the MAP indicates that the acoustic monitor may also actuate if a PZR Safety lifts. According to OP-TM-MAP-G0106 (p1of 2; Rev 3) and OP-TM-MAP-G0107 (p1of 2; Rev 3)

Technical Reference(s):	OP-TM-MAP-G0106 (p1/2of 2; Rev 3) OP-TM-MAP-G0107 (p1of1; Rev 2) OP-TM-PPC-A0518 (p1; Rev 2)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO3, 5, 14 (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: NA Last NRC Exam: 2008-1 Q1 (Modified)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of the interrelations between the Pressurizer Vapor Space Accident and the system valves (i.e. diagnosing a Pzr PORV/Safety that has lifted below setpoint, actions to take because of this).

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate understanding of the Pzr PORV/Safety valve instrumentation, and diagnose a failure given a set of circumstances.

Exam Bank Searches:
Questions 41 for 41008AK201 – None

Modified 2008 NRC Exam Question #1 by changing conditions from a leaking PZR PORV to a Pressurizer Safety Valve that had lifted clearly below setpoint and modified all distracters to align with conditions and permissible actions.
Modified Question

008 AK2.01 (**Direct Match**)

Same System/EAPE previous RO exams

Previous Exam	Question #	KA
08-1 NRC	**1**	**008 AK2.01**
08-1 Audit	39	008 AA2.04
07-1 NRC	40	008 AA2.17
05-1 NRC	40	008 AA2.04

Same System/EAPE previous SRO exams: None

08-1 NRC Q1

Plant Conditions:

- The plant is at 60% power following the loss of the A Feedwater Pump
- Highest RCS Pressure during the transient was 2250 psig
- Main Annunciator alarm G-1-7, PZR SAFETY OR PORV OPEN (DP) alarmed and then cleared following the transient
- Main Annunciator alarm G-1-8, PORV OPEN (ACOUSTIC) alarmed and then cleared following the transient
- PPC alarm A0517, RC-RV2 TAILPIPE DELTA TEMPERATURE is in alarm
- RCS pressure continues to lower slowly
- RCDT pressure continues to rise slowly

Which ONE of the following actions will be required?

A. Remove power from the PORV to ensure it is closed.

B. Commence a plant shutdown and cooldown due to the RCS leak.

C. Reduce RCS pressure to 1970-2000 psig for two hours to stop the leakage.

D. Close the PORV Block Valve and quickly cycle the PORV to try and reseat it.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	054	AK3.05	
	Importance Rating	4.6		

Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): HPI/PORV cycling upon total feedwater loss

Proposed Question: RO Question # 45

Plant conditions:

- A total loss of feedwater capability has occurred.
- The operating crew is implementing OP-TM-EOP-009, HPI Cooling.

Which ONE (1) of the following correctly completes the statement below?

HPI flow will be initiated by manually actuating the ____1____ ESAS and the PZR PORV ____2____.

A. (1) 4 PSIG
(2) is opened in MANUAL to minimize the chance of valve failure.

B. (1) 4 PSIG
(2) cycles at the AUTO setpoint to minimize the loss of coolant inventory.

C. (1) 1600 PSIG
(2) is opened in MANUAL to minimize the chance of valve failure.

D. (1) 1600 PSIG
(2) cycles at the AUTO setpoint to minimize the loss of coolant inventory.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-EOP-009 (p3; Rev 6), Step 3.3, the operator is directed to ensure that the 4 psig ESAS has been initiated. According to OP-TM-EOP-0091 (p2; Rev 1), using the 4 psig ESAS accomplishes two important functions: (1) initiation of HPI, and (2) initiation of RB isolation and cooling. Initiation of HPI is essential for HPI Cooling, and actuation of HPI by the 4 psig ESAS aligns makeup and HPI components with 2 pumps injecting through four valves to the RCS. Additionally, alignment of Containment Isolation valves and RB cooling for LOCA conditions is appropriate before opening the PORV. High energy content reactor coolant will be released into the RDCT and eventually rupture the pressure disc releasing the coolant into the RB. The 4 psig ESAS will also isolate letdown, maintaining reactor coolant in the RB as long as possible for long term cooling. According to OP-TM-EOP-009 (p3; Rev 6), Step 3.6, the operator is directed to open the PORV. According to OP-TM-EOP-0091 (p3; Rev 1), the PORV is manually opened to preclude automatic cyclic operation and the chance for PORV failure is minimized.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that RCS inventory is of the utmost concern. However, the PORV is opened to provide a cooling path, not operated to minimize the loss of inventory. According to OP-TM-EOP-009 (p3; Rev 6), Step 3.4, the operator is directed to verify Adequate HPI flow. If Adequate HPI flow does not exist the operator will be directed around the step to open the PORV. According to OP-TM-EOP-0091 (p3; Rev 1), if adequate HPI flow is NOT available, then the operator must take actions to ensure that the existing coolant remains in the RCS, and inventory loss becomes the predominant concern. The operator may confuse the two concerns for the given conditions and incorrectly believe that the action of allowing the PORV to cycle automatically is done to conserve inventory.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because it is another means of ESAS actuation and the operator may incorrectly believe that this actuation is preferable to the 4 psig ESAS.

D. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that this actuation is preferable to the 4 psig ESAS (See C); and because the operator may incorrectly believe that RCS inventory is of the utmost concern (See B).

Technical Reference(s): OP-TM-EOP-009 (p3; Rev 6) (Attach if not previously provided)
OP-TM-EOP-0091 (p2-3; Rev 1)

Proposed References to be provided to applicants during examination: None

Learning Objective:	E09-PCO-6	(As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge of the reasons for the HPI/PORV cycling upon total feedwater loss as it applies to the Loss of Main Feedwater (i.e. minimize chance for PORV failure).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. 4 psig ESAS is manually actuated, PORV is manually actuated to minimize the potential for failure).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	056	AK3.02	
	Importance Rating	4.4		

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

Proposed Question: RO Question # 46

Plant conditions:

- The plant is at 100% power.
- Normal equipment lineups.

Event:

- Loss of offsite power (LOOP).
- EG-Y-1B fails to start.
- IC-P-1A fails to restart when 1A ES MCC is reenergized.
- RCP Seal #1 inlet temperatures are 239°F.
- RCP Seal Water Temperatures at the radial bearings are 221°F.

Which ONE (1) of the following actions should be taken to cool the RCP seals?

A. Initiate a plant cooldown.

B. Start MU-P-1A and reestablish seal injection flow.

C. Cross connect the 1P and 1S 480V busses and start IC-P-1B.

D. Isolate one letdown cooler to reduce Intermediate Closed Cooling Water temperature.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to OP-TM-211-000 (p1; Rev 53), in the normal ES Standby Mode, MU-P-1B is in operation, and MU-P-1A and MU-P-1C are ES Selected. According to TQ-TM-104-740-C001 (p21-22; Rev 3), when an undervoltage condition occurs on an ES 4160V bus, the associated EDG will auto-start. As soon as the diesel generator is up to speed and voltage, the generator breakers will automatically close, providing that both bus supply breakers and the tie breaker are open. Then, the previously running block 1 loads re-energize. Since only the MU-P-1B was previously running, no MU Pump will be running upon re-energizing of the 1D 4160V bus. Since no IC Pump is running either, there will be no cooling to the RCP Thermal Barriers, and a total loss of seal cooling will occur. According to OP-TM-AOP-020 (p3; Rev 13), Step 3.4, the operator will be directed to verify that at least one of the IC Pumps are operating. Since none are, the RNO, since RCP Seal temperature is >235°F, will direct the operator to initiate OP-TM-226-901, Loss of all RCP Seal Cooling. According to OP-TM-226-901 (p1; Rev 3) the procedure provides actions required when all RCP seal cooling has been lost and RCP#1 seal temperature has exceeded 235°F. Within this procedure precautions are provided to ensure that thermal barrier cooling is not restored to an RCP once this procedure is entered. According to OP-TM-226-901 (p3; Rev 3), when at least one DH train is available, the operator is directed to start a plant cooldown. The A Train of DH is available, and therefore, this step is applicable.

B. **Incorrect.** This is plausible because according to OP-TM-AOP-020 (p3; Rev 13), Step 3.5, the operator will be directed to verify seal injection flow > 22gpm, and then initiate OP-TM-AOP-041, Loss of Seal Injection, when it is determined that SI is < 22 gpm. According to OP-TM-AOP-041 (p3; Rev 4), Step 3.4, the operator will address steps to start a MU Pump. However, a previous step in OP-TM-AOP-020 will have provided guidance to initiate OP-TM-226-901, Loss of all RCP Seal Cooling, and this procedure will provide a direction (Precaution 3.1.1) to NOT start a MU Pump.

C. **Incorrect.** This is plausible because According to OP-TM-AOP-020 (p9; Rev 13), if the 1E 4160V Bus cannot be energized, the operator is directed to initiate OP-TM-732-902, Energize 1S 480V Bus using ES Bus Cross Tie. However, according to OP-TM-732-902 (p1; Rev 2), the purpose of this action is to energize the B train battery chargers and maintain power to VBB and VBD. But, since according to TQ-TM-104-531-C001 (p26; Rev 5) IC-P-1B is powered from 1B ES 480V MCC which is powered from the 1S 480V Bus, the operator may incorrectly believe that this is done to restart IC-P-1B and cool the seals.

D. **Incorrect.** This is plausible because this action may have been an action previously taken in the leakage AOP to minimize ICCW temperature and improve seal cooling; however a plant cooldown must be commenced to cool down the seals since temperature is >235 degrees.

Technical Reference(s):	OP-TM-211-000 (p1; Rev 53) OP-TM-226-901 (p1, 3; Rev 3) OP-TM-732-902 (p1; Rev 2) TQ-TM-104-740-C001 (p21-22; Rev 3) TQ-TM-104-531-C001 (p26; Rev 5) OP-TM-AOP-020 (p3, 9; Rev 13) OP-TM-AOP-041 (p3; Rev 4)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: A20-PCO-4 (As available)

Question Source:	Bank #	IR-AOP020-PCO-4-Q03	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. if the seals are > 235°F, cooling cannot be restored and a plant cooldown is needed) of the reasons for the actions contained in EOP for loss of offsite power as they apply to the Loss of Offsite Power. The Bank Question documentation indicated that the question requires that the student understand the strategy to protect the both stations batteries when one 4kv ES bus is unavailable following a loss of offsite power.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	011	EK3.08
	Importance Rating	3.9	

Knowledge of the reasons for the following responses as they apply to the Large Break LOCA: Flowpath for sump recirculation

Proposed Question: RO Question # 47

Plant conditions:

- A large break LOCA has occurred.
- LPI flow is 2500 gpm in each train.
- SCM is 2°F, rising slowly.
- Containment Pressure is 8 psig, lowering with spray actuated.
- The crew is performing EOP-006, LOCA Cooldown, Step 3.9 – Initiate GUIDE 20, Prior to Transfer to RB Sump.

Which ONE (1) of the following correctly completes the statements below?

GUIDE 20 requires that the first step actions be completed prior to BWST level reaching less than 15 feet because ____1____. Considering the current conditions, HPI will be ____2____ when BWST level lowers to less than 15 feet.

A. (1) establishing LPI flow from the RB sump cannot be successfully completed without completion of GUIDE 20
(2) terminated

B. (1) establishing LPI flow from the RB sump cannot be successfully completed without completion of GUIDE 20
(2) placed in the piggyback mode

C. (1) rising radiation levels may make areas of the Auxiliary Building inaccessible for critical operations that must be performed later in the event

(2) terminated

D. (1) rising radiation levels may make areas of the Auxiliary Building inaccessible for critical operations that must be performed later in the event
(2) placed in the piggyback mode

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may not know the general content of GUIDE 20, if candidate believes GUIDES 22 and 23 can not be performed without competition of GUIDE 20. The steps in GUIDE 20 are NOT directly related to the implementation of LPI RB Sump Recirculation.

B. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may not know the general content of GUIDE 20 (See A); and the operator may incorrectly believe that the conditions for HPI termination are NOT met (See D).

C. **Correct.** 1st part correct, 2nd part correct. According to OP-TM- EOP-0061 (p7; Rev 2), Step 3.9 indicates that future tasks in the Auxiliary Building may not be able to be performed due to radiation problems so critical operations are performed in advance of recirculation. Additionally, according to OP-TM-EOP-010 (p28; Rev 11), Guide 20, (p1of1) when BWST level is < 15 feet, the operator is directed to verify HPI is shutdown or placed in piggyback mode. According to OP-TM-211-901 (p13; Rev 5), Step 4.3.2, if at any time BWST level is < 15 feet, the operator is directed to evaluate RCS Subcooling and LPI. If Subcooling is < 25°F AND LPI flow is > 1250 gpm, then the operator is directed to terminate HPI in accordance with Attachment 7.3, Throttling HPI. Since both of these conditions are met, HPI will be terminated and NOT placed in the piggyback mode.

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because placing the HPI in piggyback operation is the operational choice if conditions for HPI termination are NOT met. The operator may incorrectly believe that the conditions for HPI termination are NOT met.

Technical Reference(s): OP-TM- EOP-0061 (p7; Rev 2) (Attach if not previously provided)
OP-TM-EOP-010 (p28; Rev 11)
OP-TM-211-901 (p13; Rev 5)

Proposed References to be provided to applicants during examination: None

Learning Objective: E06-PCO-1, 2 and 5 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge of the reasons (i.e. basis for performing/completing GUIDE 20 at the proper time, the status of HPI during recirculation) for the flowpath for sump recirculation as they apply to the Large Break LOCA.

The question is at the Comprehension/Analysis cognitive level because the operator must evaluate parameters and decide on a course of action which demonstrates understanding. If the conditions were changed, a different answer, which is available, is appropriate.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	038	EA1.24	
	Importance Rating	3.6		

Ability to operate and monitor the following as they apply to a SGTR: Safety injection pump ammeter and indicators

Proposed Question: RO Question # 48

Plant conditions:

- A loss of off-site power (LOOP) resulted in a reactor trip.
- An OTSG "A" tube rupture occurred coincident with the trip.
- HPI has been initiated.
- The crew is performing OP-TM-EOP-005, OTSG Tube Leakage.
- Crew is throttling HPI to minimize SCM.
- The crew has closed MU-V-16A and MU-V-16D HPI injection valves.

Which ONE (1) of the following correctly completes the statement below?

When the TWO HPI valves above are closed, flow will be indicated on ____1____ HPI Flow Indicators, and the loading on "A" Emergency Diesel Generator will be ____2____ than before the throttling began.

A. (1) two
(2) higher

B. (1) two
(2) lower

C. (1) four
(2) higher

D. (1) four
(2) lower

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may not properly think through a typical pump curve.

B. **Correct.** 1st part correct, 2nd part correct. In accordance with OP-TM-211-901(p21; Rev 5), Attachment 7.3, Step 7, the operator is directed to close MU-V-16 valves to establish flow through one valve on each train and retain HPI flow through four RCS nozzles. Therefore, one valve is closed in each train but the selection is such that flow continues to all four nozzles. According to Drawing 301-661 (Rev 59), MU-P-1A will discharge to a common header with MU-V-16A and 16B, while MU-P-1C will discharge to a common header with MU-V-16C and 16D. Each of the MU-V-16 valves discharge to a Loop Cold Leg downstream of the RCP. For instance, MU-V-16A discharges top Loop A, downstream of RCP A, and so on. However, downstream of each MU-V-16 is a cross-connect pipe with the opposite loop. For instance, downstream of MU-V-16A is a cross-connect pipe with the downstream of MU-V-16C; and a similar arrangement exists downstream of MU-V-16 B and D. Therefore, when the operator is directed to close MU-V-16 valves to establish flow through one valve on each train and retain HPI flow through four RCS nozzles, the operator must close either MU-V-16A and D, or MU-V-16B and C. In this way, two valves will remain open, however, flow will be directed to all four loops. However, since flow transmitters FT-1126, 1127, 1128 and 1129 are located upstream of the respective MU-V-16 valves, while flow is retained to all four Tcold nozzles, flow will only be indicated in one train A and one Train B line. For instance, if the operator closes MU-V-16A and D, flow through MU-V-16B will be directed to Loop A Tc RCP B discharge as it normally would, and flow will be indicated on FT-1127; and at the same time flow would be directed through the cross connect to Loop B Tc RCP D discharge. However, the Loop flow associated with MU-V-16D, since this valves is closed will indicate zero. At the same time, flow through MU-V-16C will be directed to Loop B Tc RCP C discharge as it normally would, and flow will be indicated on FT-1128; and at the same time flow would be directed through the cross connect to Loop A Tc RCP A discharge. However, the Loop flow associated with MU-V-16A, since this valves is closed will indicate zero. Therefore, while flow is directed to four entry points of the RCS, flow will only be indicated on two HPI flow nozzles. The second part is correct because flow is being reduced from a multi-stage centrifugal pump. According to OPM N-3 (p88; Rev 5), the MU Pump Brake Horsepower Curve represents the power requirements of the motor as system flow increases. According to OPM N-3 (p114; Rev 5), the MU Pump Curve reveals that as the system flow increases, show do the pump horsepower requirements. The opposite is also true. As the system flow decreases, the pump power requirements will be less. Since, the Emergency Diesels are powering the 4160V busses, the loading on the Diesels after HPI throttling will be lower.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that the flow transmitters to all four loops are indicating flow (See D); and because the operator may not properly think through a typical pump curve.

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may not be aware that the MU piping system is cross tied in such a manner that closing two valves in the proceduralized combination will result in a continuation of flow through all four piping paths, BUT an indication of flow through only two flow transmitters. Therefore, two of the four HPI Valves are closed, and flow is indicated in two of the four, even though flow continues to all four nozzles. The operator may incorrectly believe that the flow transmitters to all four loops are indicating flow.

Technical Reference(s): OP-TM-211-901 (p13, 21; Rev 5) (Attach if not previously provided)

OPM N-3 (p88, 114; Rev 5)

Drawing 301-661 (Rev 59)

Proposed References to be provided to applicants during examination: None

Learning Objective: E10-PCO-1 and 6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10, 14

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to operate and monitor the Safety injection pump ammeter and indicators as they apply to a SGTR as closely as possible without available amperage indication. Developer could not locate evidence of amperage indication in the lesson plan covering the Makeup Pumps. If there is amperage indication, then the second part could be changed easily. Comparing EDG loading before and after is based on connection to Pump Curve. But, since there is only HIGHER or LOWER as a choice, a logical choice is available by applying fundamental knowledge.

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate an understanding of the consequences of procedurally driven actions upon indications.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	065	AA1.02	
	Importance Rating	2.6		

Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Components served by instrument air to minimize drain on system

Proposed Question: RO Question # 49

Plant conditions:

- An Instrument Air (IA) leak has reduced IA header pressure to 50 psig.
- The IA leak has not been isolated.
- OP-TM-AOP-028, "Loss of Instrument Air" has been entered.
- The reactor has been tripped.
- The "A" 2 Hour Emergency Air Header Pressure is now reading 1500 psig.
- The "B" 2 Hour Emergency Air Header Pressure is now reading 1725 psig.

Which ONE (1) of the following valves is capable of being manually shifted from an "A" 2 Hour Header Supply to a "B" 2 Hour Header Supply?

A. EF-V-30A, EFW Control Valve to OTSG "A".

B. MU-V-20, RCP Seal Injection RB Isolation Valve.

C. MS-V-6, EF-P-1 Main Steam Pressure Control Valve.

D. RR-V-6, Emergency Cooling Coil Backpressure Regulator Valve.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-AOP-028 (p21; Rev 4), Attachment 6.2, this valve is supplied emergency operating air by the Train A 2-hour Air System. However, the backup is that EF-V-30D (also for OTSG "A") is supplied from the "B" 2 Hr Header. The operator may incorrectly believe that EF-V-30A may be supplied by both the Train A and B 2-hour Air System.

B. **Incorrect.** This is plausible because according to OP-TM-AOP-028 (p3; Rev 4), Step 3.2, directs the operator to dispatch an operator to block open MU-V-20. According to OP-TM-AOP-0281 (p7; Rev 3), this is done to ensure that RCP Cooling is NOT lost. The operator may incorrectly believe that this valve is important enough to have a 2-hour back up of IA.

C. **Correct.** According to OP-TM-AOP-028 (p15; Rev 4), Step 4.13 the operator is directed to open IA-V-1632, the MS-V-6 Backup 2 Hour Air B Distribution Header Valve, if (IAAT) the A 2-hour emergency air system pressure is reducing to > 200 psig below the B 2-hour emergency air system pressure. According to OP-TM-AOP-0281 (p12; Rev 3) this is to provide more time before local control of MS-V-6 is required if the A 2-hour emergency system is being rapidly depleted.

D. **Incorrect.** This is plausible because according to OP-TM-AOP-028 (p21; Rev 4), Attachment 6.2, this valve is supplied emergency operating air by the Train A 2-hour Air System. However, there is no alternate supply. RR Pressure is maintained by closing RR-V-6 and controlling RR-V-5. The operator may incorrectly believe that RR-V-6 may be supplied by both the Train A and B 2-hour Air System.

Technical Reference(s): OP-TM-AOP-028 (p15, 21; Rev 4) (Attach if not previously provided)
OP-TM-AOP-0281 (p7, 12; Rev 3)

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO11 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to operate and/or monitor components served by instrument air to minimize drain on system as they apply to the Loss of Instrument Air. The actions in the AOP focus on preserving critical functions and restoring pressure while the leak is located and isolated. Attempted to meet K/A by requiring knowledge of action that can be implemented when a critical function is threatened by lowering pressure in the 2 Hr Header (the final source for EFW control from the control room).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. Which valves can be receive air from both the A and B Trains of the 2-hour Emergency Air System).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	027	AA1.02	
	Importance Rating	3.1		

Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: SCR-controlled heaters in manual mode

Proposed Question: RO Question # 50

Plant conditions:

- The plant is at 100% Power.
- Troubleshooting is in progress after failure of the automatic operation of the Pressurizer Pressure Controller.
- The Pressurizer Pressure Controller has been placed in HAND.
- All Pressurizer Heaters are available.
- RCS pressure indicates 2135 psig.
- The URO raises the demand setting on the controller.

Which ONE (1) of the following describes the operation of pressurizer heaters for this condition?

A. RCS pressure rises; As pressure rises, Banks 1 and 2 will de-energize at a <u>lower</u> setpoint than Bank 3.

B. RCS pressure rises; As pressure rises, Banks 1 and 2 will de-energize at a <u>higher</u> setpoint than Bank 3.

C. RCS pressure lowers; As pressure lowers, Banks 1 and 2 will re-energize at the <u>same</u> setpoint as Bank 3.

D. RCS pressure lowers; As pressure lowers, Banks 4 and 5 will re-energize at the <u>same</u> setpoint as Banks 1, 2, and 3.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** Plausible because raising demand on the controller calls for increased pressure. Banks 1 and 2 de-energize at a different pressure than Bank 3. Incorrect because de-energization pressure is higher than Bank 3.

B. **Correct.** Raising demand calls for increased pressure. The SCR controlled heaters (Banks 1, 2, 3) are used to cycle normal pressure between 2135 psig and 2147 psig (Bank 3) and 2135-2155 psig (Banks 1 and 2).

C. **Incorrect.** Incorrect but plausible because if controller output was raised instead of controller demand, then RCS pressure would lower. Controller operation in different modes is easily confused. Also plausible because if pressure did lower, Banks 1, 2, and 3 would energize at the same pressure.

D. **Incorrect.** Incorrect but plausible because if controller output was raised instead of controller demand, then RCS pressure would lower. Controller operation in different modes is easily confused. Also plausible because if pressure did lower, Banks 1, 2, and 3 would energize at the same time, but Banks 4 and 5 are operated on bistables, and will energize at a lower pressure than Banks 1, 2, and 3.

Technical Reference(s): OP-TM-220-000 (p19; Rev 13) (Attach if not previously provided)
TQ-TM-104-220-C001 (p8,9 Rev 4)

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO3, 5 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to operate and/or monitor the SCR-controlled heaters in manual mode as they apply to the Pressurizer Pressure Control Malfunctions (i.e. requires knowledge of the proper setting for the Pressurizer Pressure Controller (in HAND) to energize the SCR-controlled heaters and the status of an important heater protective interlock with the controller in an off-normal setting).

The question is at the Comprehension cognitive level because the operator must understand controller status and the resulting effect of changing one of the controller processes

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	029	EA2.09	
	Importance Rating	4.4		

Ability to determine or interpret the following as they apply to a ATWS: Occurrence of a main turbine/reactor trip

Proposed Question: RO Question # 51

Plant conditions:

- A plant shutdown is in progress.
- Power Range Instruments read as follows:
  - NI-5 44%
  - NI-6 46%
  - NI-7 44%
  - NI-8 47%

Event:

- Alarm K-1-1, TURBINE TRIP, actuates.
- EHC Hydraulic Pressure is 375 psig.
- Two Turbine Stop Valves indicate closed.
- RCS pressure peaks at 2350 psig.
- The reactor has NOT tripped.

Which ONE (1) of the following describes the operational status of the Reactor Protection System (RPS) and the Diverse Scram System (DSS)?

A. BOTH the RPS and the DSS are operating normally.

B. ONLY the RPS has failed to operate as designed.

C. ONLY the DSS has failed to operate as designed.

D. BOTH the RPS and the DSS have failed to operate as designed.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible if the operator incorrectly believes that since 2 of 4 NI PR Channels are less than 45% power that the RPS has bypassed the Turbine Trip input into the RPS.

B. **Correct.** According to TQ-TM-104-623-C001 (p11-12; Rev 1), each Power Channel inputs to the Turbine Trip Anticipatory Trip Bypass. According to TQ-TM-104-641-C001 (p48-49; Rev 1), there are four pressure switches to monitor the main turbine hydraulic fluid header pressure downstream of the master trip solenoid valve in the EHC. These switches are set to open at 400 psig and decreasing and reset to close at 500 psig and increasing hydraulic fluid header pressure. The main turbine trip bypass is automatically placed in effect when the reactor power is less than 45%FP and is automatically removed when reactor power is raised above 45%FP. Since reactor power is >45% on two NI and EHC pressure is 375 psig, and the reactor is NOT tripped, the RPS has failed and an ATWS has occurred.

C. **Incorrect.** This is plausible if the operator incorrectly believes that since 2 of 4 NI PR Channels are less than 45% power that the RPS has bypassed the Turbine Trip input into the RPS; and if the operator has a gross conceptual error of the DSS and incorrectly believes that the DSS operates in a parallel path with the RPS, rather than as a backup, and carries the error forward believing that the DSS operates automatically at 2350 psig rather than 2500 psig.

D. **Incorrect.** This is plausible because the operator may not know what causes the DSS to automatically operate, and believed correctly that since the RPS should have operated and did not, that the DSS should have operated as well. According to OPM F-02 (p123; Rev 9), the DSS will automatically operate to open breakers 1G-2A and 1L-2A automatically if RCS pressure is > 2500 psig, or manually. Therefore, there is no reason to believe that the system has failed.

Technical Reference(s): TQ-TM-104-623-C001 (p11-12; Rev 1) (Attach if not previously provided)
TQ-TM-104-641-C001 (p48-49; Rev 1)
OPM F-02 (p123; Rev 9)

Proposed References to be provided to applicants during examination: None

Learning Objective:	641-GLO-3 and 5 621-GLO-5		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	NA	Last NRC Exam: NA	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

The KA is matched because the operator must demonstrate the ability to determine or interpret the occurrence of a main turbine/reactor trip as it applies to an ATWS (i.e. requiring application of knowledge that indications associated with an incomplete turbine trip did still provide sufficient information to the RPS that should have generated an automatic reactor trip signal).

The question is at the Comprehension/Analysis cognitive level because the operator must assemble several pieces of information (i.e. that (1) the present turbine conditions require a reactor trip, that (2) the automatic trip has failed, and that (3) the ATWS under the stated conditions would not have actuated the DSS); and draw a conclusion (i.e. what is the operational status of the RPS/DSS).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	025	AA2.02	
	Importance Rating	3.4		

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

Proposed Question: RO Question # 52

Plant Conditions:

- Rx Vessel Head being de-tensioned.
- "A" Decay Heat Removal train in service.
- "B" Decay Heat Removal train in standby.
- "A" Decay Heat Closed Cooling Water Surge Tank level is rising at a rate equivalent to 5 gpm.
- RB Sump level has a slight upward trend over the last eight hours.
- RM-A-2, RB Atmosphere, count rate is steady.
- RM-A-9, RB Purge Exhaust, count rate is steady.
- RM-L-2, "A" DHCCW, count rate has risen.

Which ONE (1) of the following describes the required action?

A. Isolate "A" DHCCW tank vent.

B. Initiate Containment Isolation.

C. Transfer heat removal to the OTSGs and secure DHR.

D. Swap cooling to "B" DHR and close DH-V-12A and DH-V-38A.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect** This is plausible because according to OP-TM-AOP-060 (p5; Rev 4), Step 3.8, which will be addressed before the transition to section 4.0, the operator will be directed to isolate the leak if it can be done so without affecting DHR operation. However, this action will not isolate the leak, but only prevent its overflow.

B. **Incorrect.** This is plausible because according to OP-TM-AOP-060 (p13; Rev 4), Step 3.18, directs the operator to initiate Containment Isolation. If the operator incorrectly believed that the leak was in Containment this action would be appropriate. On the other hand that operator may have merely remembered the step, associating it with AOP-60 and not have understood its purpose.

C. **Incorrect.** This is plausible because according to OP-TM-AOP-060 (p9; Rev 4), Step 3.13, the operator is directed to initiate EOP-030, which would initiate these actions. However, the RCS is not available and this section is skipped because the leak is NOT in containment. If the operator incorrectly believed that the leak was in Containment this action would be appropriate. On the other hand that operator may have merely remembered the step, associating it with AOP-60 and not have understood its purpose.

D. **Correct.** The operator will diagnose a leak into the DHCCW System rather than a leak into the RB Atmosphere. The RB Air monitors are steady, and a slight upward trend on the RB Sump over several hours may not be unexpected. According to OP-TM-AOP-060 (p1; Rev 4), since DHR is providing core cooling and leakage from the system is observed the entry conditions for the AOP are met. According to OP-TM-AOP-060 (p7; Rev 4), the operator will check to see if the leak is in Containment, and if NOT, go to section 4, Leak in Auxiliary Building or Fuel Handling Building. Since the leak is into the DHCCW Surge Tank, Section 4.0 will be addressed. According to OP-TM-AOP-060 (p19; Rev 4), Step 4.5, If the A DHR Train is in service, and the A DC Surge Tank Level is increasing, the operator is directed to place the B DHR Train in service, then CLOSE DH-V-12A and 38A. This will isolate the likely source of the leak.

Technical Reference(s): OP-TM-AOP-060 (p1, 5, 7, 9, 13, 19; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: A60-PCO-1, 2 and 4 (As available)

Question Source: Bank # QR-A60-PCO4-Q01

Modified Bank # (Note changes or attach parent)

New

Question History: Listed as "new" as of 1/30/08. No history of use. Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to determine and interpret leakage of reactor coolant from RHR into closed cooling water system or into reactor building atmosphere as they apply to the Loss of Residual Heat Removal System (i.e. the operator must interpret standard parameters determine where the leak is, and decide on an action).

The question is at the Comprehension/Analysis cognitive level because the operator must consider a set of parameters and decide where a leak is within the DHR System, and then apply procedure strategies, which demonstrates understanding of the event.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	055	EA2.01	
	Importance Rating	3.4		

Ability to determine or interpret the following as they apply to a Station Blackout: Existing valve positioning on a loss of instrument air system

Proposed Question: RO Question # 53

Plant conditions:

- A loss of off-site power (LOOP) resulted in a reactor trip.
- Neither 4160 V ES Bus has energized.
- The operating crew is performing OP-TM-AOP-020, Loss of Station Power – Section 4.0, Station Blackout.
- Instrument Air (IA) Pressure is lowering slowly.

Which ONE (1) of the following correctly completes the statement below?

If Instrument Air Pressure continues to lower, CO-V-7, HOTWELL LEVEL NORMAL MAKEUP VALVE, will fail ____1____ and CO-V-8, HOTWELL LEVEL EMERGENCY MAKEUP VALVE, will fail ____2____.

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-421-C001 (p10; Rev 5) and OP-TM-AOP-028 (p22; Rev 4), both CO-V-7 and 8 fail OPEN on a loss of air. It is important to know the fail position of both valves because the position of CO-V-7 must be changed if it becomes necessary to shift the suction of EFW to the hotwell.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to TQ-TM-104-421-C001 (p10; Rev 5) this would be the normal operational sequencing of these valves: CO-V-7 opens before CO-V-8. The operator may incorrectly believe that CO-V-8 fails closed on a loss of air.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to TQ-TM-104-421-C001 (p10; Rev 5) this is the alignment for the emergency suction source for EFW to allow suction to EF Pumps with vacuum broken. However, it is not the LOIA fail position. The operator may incorrectly believe that CO-V-7 fails closed on a loss of air.

D. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that CO-V-8 fails closed on a loss of air (See B); and because the operator may incorrectly believe that CO-V-7 fails closed on a loss of air (See C).

Technical Reference(s): TQ-TM-104-421-C001 (p10; Rev 5) (Attach if not previously provided)

OP-TM-AOP-028 (p22; Rev 4)

Proposed References to be provided to applicants during examination: None

Learning Objective: 421-GLO-3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:	55.41	4
	55.43	

Comments:

Meets K/A by predicting the fail position of two valves that may be required to be properly positioned later in the event.

The KA is matched because the operator must demonstrate the ability to determine or interpret existing valve positioning on a loss of instrument air system as they apply to a Station Blackout (i.e. by predicting the fail position of two valves that may be required to be properly positioned later in the event).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. how does CO-V-7 fail on a loss of air, how does CO-V-8 fail on a loss of air).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	057	2.4.45	
	Importance Rating	4.1		

Emergency Procedures / Plan: Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: RO Question # 54

Plant Conditions:

- Reactor is at 75% power, ICS is in full automatic.
- Three (3) Reactor Coolant Pumps (RCP's) are operating.
- RC-P-1A is tripped for breaker maintenance.
- MAP F-3-1, RC LOOP A FLOW LO, alarm is actuated.

Event:

- Numerous alarms including A-1-6, INVERTER FAILED, actuate.
- Loss of 120 Volt AC Vital Bus B is confirmed.
- Reactor power is stable at 75%.

Which ONE (1) of the following describes the required action?

A. Enter TS 3.0.1 due to an inoperable RPS Channel.

B. Place the ICS and Diamond Control Stations in MANUAL.

C. Trip the Reactor and enter OP-TM-EOP-001, Reactor Trip.

D. Refer to 1107-2B, 120 Volt Vital Electrical System, to energize VBB.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible if the operator incorrectly believes the loss of an entire Channel ("B") makes the RPS inoperable. The action (if all RCP's were running) would be to verify that all Channel B bistables are in the tripped condition.

B. **Incorrect.** This is plausible because according to OP-TM-AOP-016 (p3; Rev 1), Step 3.6, the operator is directed to place the ICS and Diamond Control Stations in MANUAL in accordance with OP-TM-621-471, ICS Manual Control, to prevent rod movement on one power supply. However, with one flow channel already tripped, the RPS should have tripped the reactor on indication of no RCPs operating in the A Loop.

C. **Correct.** According to TQ-TM-104-641-C001 (p45-46, Rev 1), the RCP power is monitored to determine that the pumps are running. Loss of a single pump initiates eight independent contact closure signals (four in each independent RC pump power monitor train). The contact closure signals are monitored by the RPS channel RC pump contact monitor which counts the number of RC pumps in operation, identifies the loop in which the pumps are operating, and outputs a signal representing the allowed power level for the present pump status. The reactor is tripped when the number of operating pumps in each loop does not correspond to the number of reactor coolant pumps required to be in operation for the existing reactor power. Each of the four RCP power monitor channels is powered from respective vital bus; i.e., RC-P-1A power monitor is powered from VBA, RC-P-1B power monitor channels from VBB, etc. Therefore, when VBB is de-energized the RPS detects this as the RCP is NOT running. Since the A RCP is also not running, the RPS will detect that both RCPs in Loop A are NOT operating, and generate a reactor trip signal regardless of the power level. Since the reactor is still operating at power, an ATWS exists. According to OS-24 (p13; Rev 18), the operator may take action independent of procedures to initiate a manual trip when it is apparent that an automatic action has failed. Since a reactor trip did not occur, the operator should ensure that it does occur.

D. **Incorrect.** This is plausible because according to OP-TM-MAP A0106 (p3of3; Rev 1) manual action 4.10, the operator is directed to re-energize the affected Vital Bus from TRA, TRB or alternate inverter as directed by SM/CRS, IAW 1107-2B. However, this is premature since the reason for the loss of the bus has not been determined. Re-energizing without knowing the root cause for the bus failure is an unsafe action.

Technical Reference(s): OS-24 (p13; Rev 18) (Attach if not previously provided)
TQ-TM-104-641-C001 (p43-44, Rev 1)
OP-TM-MAP A0106 (p3of3; Rev 1)
OP-TM-AOP-016 (p3; Rev 1)

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO11, 12 (As available)

Question Source: Bank #

Modified Bank # QR-120240-PCO-4-Q03 (Note changes or attach parent)

New

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:
The KA is matched because the operator must demonstrate the ability to prioritize and interpret the significance of each annunciator or alarm (i.e. requiring interpretation of the A-1-6 alarm within the context of the existing conditions and selecting the highest priority action. In this case recognizing the failure of the RPS to initiate an automatic trip).

The question is at the Comprehension/Analysis cognitive level because the operator must assemble three pieces of information (i.e. that (1) the present RCP/RCP indications conditions require a reactor trip, that (2) the automatic trip has failed, and that (3) the operator has the authority/responsibility to trip the reactor manually if the automatic trip fails); and draw a conclusion (i.e. that manual reactor trip is required). This demonstrates understanding of the RPS operation.

Exam Bank Searches:
Section 42-3, Q57.
(57) **QR-120240-PCO-4-Q03**
Revised 42-3 Q57 by changing two distracters: (B) because the original could lead the applicant to the correct answer; (C) because the ICS and Diamond Controls are “higher level” activities as compared to the original distracter. Re-ordered all choices IAW length. The correct answer remains the same. The bank question appears to be intended for open-reference LORT but the developer has seen many similar questions on NRC closed reference examinations.
Modified Question

The Plant is operating at 75% power, ICS in full automatic with the following conditions:

- Three (3) Reactor Coolant Pumps (RCP's) are operating (RC-P-1A is tripped).
- MAP F-1-3, RC LOOP A FLOW LO, alarm is actuated.

When the following event occurs:

- MAP A-1-6, INVERTER FAILED, is received due to a loss of 120 Volt AC Vital Bus B.

The following is observed:

- Reactor power is constant at 75%.

What actions need to be taken immediately?

A. Trip the Reactor and enter OP-TM-EOP-001, Reactor Trip.

B. Enter 1102-4, Power Operations and reduce Reactor power to less than 55%.

C. Verify DC-V-2B and DC-V-65B Console Controllers have failed to their ES positions.

D. Refer to 1107-2B, 120 Volt Vital Electrical System, and take action to re-energize VBB.

Answer: A

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	E04	2.1.19	
	Importance Rating	3.9		

Conduct of Operations: Ability to use plant computers to evaluate system or component status.

Proposed Question: RO Question # 55

Plant conditions:

- Loss of off-site power (LOOP) at 100% power results in a reactor trip.
- No source of feedwater has been established to the OTSG's.
- Natural Circulation has NOT been confirmed.
- The crew is performing OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer.
- SCM is 35°F, and lowering steadily.

Which ONE (1) of the following identifies the SCM instrument to be monitored, AND the next procedure to be implemented if SCM continues to lower?

A. Saturation Margin Meter (PCL) TI-977 or TI-978; AND
OP-TM-EOP-002, Loss of 25°F Subcooling Margin.

B. Saturation Margin Meter (PCL) TI-977 or TI-978; AND
OP-TM-EOP-009, HPI Cooling.

C. Computer Point C4008 or C4132; AND
OP-TM-EOP-002, Loss of 25°F Subcooling Margin.

D. Computer Point C4008 or C4132; AND
OP-TM-EOP-009, HPI Cooling.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may be unaware of the idiosyncrasies of the Wide Range Thot instruments (See B); and the operator may incorrectly believe that transition to OP-TM-EOP-002 is appropriate (See C).

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because these are the normal SCM instruments used when RCP's are running or natural circulation is verified. The operator may be unaware of the idiosyncrasies of these instruments.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because transition to OP-TM-EOP-002 would be the path for loss of SCM under most EOP conditions but not from EOP-004. The operator may incorrectly believe that transition to OP-TM-EOP-002 is appropriate.

D. **Correct.** 1st part correct, 2nd part correct. According to OS-24 (p21; Rev 18), Section 4.7.2, if all RCPs are shutdown and natural circulation has not been verified in both loops, the operator is directed to use incore subcooling margin C4008 of C4132 to determine SCM. According to TQ-TM-104-624-C001 (p52; Rev 2), a Backup Incore Thermocouple Readout (BIRO), is a diverse readout system, (redundant to the computer) for monitoring core exit thermocouple temperatures. According to OP-TM-EOP-004 (p3; Rev 7), Step 3.6, the operator is directed to go to OP-TM-EOP-009, IAAT RCS is approaching 25°F SCM. According to OP-TM-EOP-0041 (p2; Rev 2), Step 2.2 there are two major mitigation strategies within EOP-004, based on whether or not Feedwater is available. If feedwater is available, then OTSG pressure is lowered and level is raised to promote primary-to-secondary heat transfer. On the other hand, if feedwater is NOT available, as is the case here, EOP-009, HPI Cooling will be initiated if SCM approaches < 25°F.

Technical Reference(s): OS-24 (p21; Rev 18), Section 4.7.2 (Attach if not previously provided)
OP-TM-EOP-004 (p3; Rev 6)
OP-TM-EOP-0041 (p2; Rev 1), Step 2.2

Proposed References to be provided to applicants during examination: None

Learning Objective: E04-PCO-1, 2, 3 and 4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

	New	X	
Question History:	NA	Last NRC Exam: NA	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments:

Considered higher order because the correct answer would be different if RCS circulation conditions were changed.

The KA is matched because the operator must demonstrate the ability to use plant computers to evaluate system or component status associated with Inadequate Heat Transfer (i.e. by requiring knowledge of when to use computer generated SCM using the in-core thermocouples rather than normal SCM using Thot.).

The question is at the Comprehension/Analysis cognitive level because the operator must evaluate the stated information and consider follow-up procedures. If the stated conditions were different, the distracters would be correct. This demonstrates that the operator must possess understanding of the event that is occurring.

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	026	2.1.31	
	Importance Rating	4.6		

Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question: RO Question # 56

Plant Conditions:

- The plant was at 35% power when NS Surge Tank Level started lowering.
- The crew entered OP-TM-AOP-031, Loss of Nuclear Services Component Cooling (NS).
- The reactor has been tripped.
- The crew has completed the required AOP actions and is awaiting leak isolation.

In addition to the NS Pumps, which ONE (1) of the following identifies pump control switch(es) that has/have been placed in PULL-TO-LOCK (PTL) during this procedure?

A. Makeup Pump MU-P-1B ONLY.

B. Makeup Pump MU-P-1B AND
RCDT Pump ONLY.

C. Makeup Pump MU-P-1B AND
Spent Fuel Cooling Pumps (SF) AND
Waste Gas Compressors ONLY.

D. Makeup Pump MU-P-1B AND
RCDT Pump AND
Spent Fuel Cooling Pumps (SF) AND
Waste Gas Compressors

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-AOP-031 (p5; Rev 4) Steps 3.6 and 3.7, either MU-P-1A of MU-P-1C will need to go into service. According to OP-TM-AOP-0311 (p7; Rev 1) this is done to prevent damage to MU-P-1B which receives its cooling from the NS System. However, this answer is incomplete. The operator may be unaware that the SF Pumps are placed in PTL as well (See C).

B. **Incorrect.** This is plausible because MU-P-1B is placed in PTL (See A/C). However, the RCDT Pump is NOT placed in PTL because normal cooling has not been lost. According to OP-TM-AOP-032 (p5; Rev 2), when the Intermediate Closed Cooling System is affected in a similar fashion to the NS System, the RDCT Pump will be placed in PTL. The operator may incorrectly believe that this event has resulted in a loss of the RCDT Pump. Plausibility is enhanced by the fact that the NS System provides cooling water to components within the RB.

C. **Correct.** According to OP-TM-AOP-031 (p5; Rev 4) Steps 3.6 and 3.7, either MU-P-1A of MU-P-1C will need to go into service. According to OP-TM-AOP-0311 (p7; Rev 1) this is done to prevent damage to MU-P-1B which receives its cooling from the NS System. MU-P-1A and MU-P-1C can also receive cooling from the respective DH CCW System. According to OP-TM-AOP-031 (p9; Rev 4) Step 3.16, both SF Pumps will need to be removed from service. According to OP-TM-AOP-0311 (p8; Rev 1) this is done to because the NS System provides cooling water to the SF Coolers, and running the pumps will simply add heat to the SFP. It is recognized that SFP heat up is expected, and the AOP for loss of SF Cooling may need to be implemented as appropriate.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that this event has resulted in a loss of the RCDT Pump.

Technical Reference(s): OP-TM-AOP-031 (p5, 9; Rev 4) Steps 3.6, 3.7,3.16, and 3.20 (Attach if not previously provided)

OP-TM-AOP-031 (p7-8; Rev 1)

OP-TM-AOP-032 (p5; Rev 2)

Proposed References to be provided to applicants during examination: None

Learning Objective: A31-C001 Ob 1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup associated with a loss of Component Cooling (Meets K/A by requiring knowledge of the required position for a pump control switch with a loss of NSCCW in effect).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. Pumps that are placed in PTL within AOP-031).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	024	AK1.01	
	Importance Rating	3.4		

Knowledge of the operational implications of the following concepts as they apply to Emergency Boration: Relationship between boron addition and change in T-ave

Proposed Question: RO Question # 57

Plant conditions:

- ICS in Hand for CRD Breaker testing
- An ATWS is in progress.
- Reactor power is 45%.
- The crew is performing actions of Rule 5, Emergency Boration.
- RCS boration flow has been 55 GPM for 5 minutes.

Which ONE (1) of the following describes the effect of the Emergency Boration on Tave, and (2) the normal flowpath for Emergency Boration in accordance with Rule 5?

A. (1) Tave will be constant as reactor power is lowering
(2) MU-V-51 or MU-V-10

B. (1) Tave will be constant as reactor power is lowering
(2) MU-V-14A or MU-V-14B

C. (1) Tave will be lowering as reactor power is lowering
(2) MU-V-51 or MU-V-10

D. (1) Tave will be lowering as reactor power is lowering
(2) MU-V-14A or MU-V-14B

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. Plausible because the operator may misunderstand the affect of boration on Tave during ATWS conditions. If power was lower, Tave would not change, as temperature would be controlled by Decay heat and RCP heat, as well as TBVs. Additionally, MU-V-51 and MU-V-10 are identified as backup boration flowpaths in accordance with Guide 1.

B. **Incorrect.** 1st part wrong, 2nd part correct. Plausible because the operator may misunderstand the affect of boration on Tave during ATWS conditions. If power was lower, Tave would not change, as temperature would be controlled by Decay heat and RCP heat, as well as TBVs. The second part is correct in accordance with Rule 5.

C. **Incorrect.** 1st part correct, 2nd part incorrect. 1st part correct because as boron is added, neutrons are absorbed, reactor power is lowered, and RCS temperature is lowered. Second half is plausible because both of the valves identified are part of emergency boration flowpaths, but they are identified as backup paths in accordance with Guide 1.

D. **Correct.** Normal boration flowpath is correct, and temperature effect is correct as show on the boron worth curve boron adds negative reactivity. As power is reduced constant level in the OTSGs will cause a reduction in Tave.

Technical Reference(s): OP-TM-EOP-010, Rev 11(Rule 5) (Attach if not previously provided)

OPM N-07 (p215; Rev 5)

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO7 (As available)

623-GLO5, 7

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	5
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge of the operational implications of the relationship between boron addition and change in T-ave as they apply to Emergency Boration).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	A03	AK2.2	
	Importance Rating	3.3		

Knowledge of the interrelations between the (Loss of NNI-Y) 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.

Proposed Question: RO Question # 58

Plant conditions:

- The unit is in Cold Shutdown condition.
- RCS temperature is being lowered by 5°F.
- Decay Heat Removal Trains A and B are operating.
- DC-V-2A and B (DH Removal Cooler Inlet Valve) are throttled 15% OPEN.
- DC-V-65A and B (DH Removal Cooler Bypass Valve) are throttled 40% OPEN.

Event:

- Vital Bus "A" de-energizes.

Which ONE (1) of the following correctly completes the statement below?

The cooldown rate ____1____ and the flow rate through the A Train Decay Heat Removal Cooler Bypass line ____2____.

A. (1) lowers
(2) goes to ZERO

B. (1) lowers
(2) rises

C. (1) rises
(2) rises

D. (1) rises
(2) goes to ZERO

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that that DC-V-2A fails CLOSED. Since the B Train will still be operating the cooldown rate will be lower, and with DC-V-65A failing CLOSED, flow rate through the A Train Decay Heat Removal Cooler Bypass line will go to ZERO.

B. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that DC-V-65A fails OPEN and that DC-V-2A fails CLOSED. Since the B Train will still be operating the cooldown rate will be lower with the A Train in full Cooler Bypass.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that DC-V-65A fails OPEN, but since DC-V-2A fails open as well, the overriding effect is that of overcooling.

D. **Correct.** 1st part correct, 2nd part correct. With maximum flow through the cooler the cooldown rate will rise, and, with DH-V-65 CLOSED, bypass line flow will go to ZERO. According to OP-TM-1107-2B (p5; Rev 24A), control power is lost to the Foxboro Spec 200 controllers used in control loops for DC-V-2A/B and DC-V-65A/B, DC-V-2A/B fail open while DC-V-65A/B will fail closed. This means that full Decay Heat Closed Cooling Water System (DC) flow will be sent to the Train A Decay Heat Removal (DH) cooler, causing the cooldown rate to rise. This is reflected in OP-TM-AOP-0151 (p12; Rev 0) which indicates that when VBA is lost with DHR operating, DC-V-2A will fail open and DC-V-65A will fail closed. If only the A DHR Train is operating it is likely that the RCS temperature is low enough that failing the valves to full cooling positions will have little effect on the RCS Temperature. However, since both trains are operating, the operator can adjust the B Train to stabilize RCS temperature.

Technical Reference(s): OP-TM-1107-2B (p5; Rev 24A) (Attach if not previously provided)
OP-TM-AOP-0151 (p12; Rev 0)

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP015-C001 Ob 1 (As available)

734-GLO11

Question Source: Bank #

Modified Bank # QR-543-GLO-6-Q02 (Note changes or attach parent)

New

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of the interrelations between the Loss of NNI-Y (VBA) and the DHR System (i.e. how is two train cooldown affected by a loss of VBA), and relations between the proper operation of these systems to the operation of the facility.

The question is at the Comprehension/Analysis cognitive level because the operator must understand how the DC System is affected by a loss of VBA while in the DHR Mode, and consider how this affects a cooldown using two trains of DHR.

The modification included changing the bank question to more closely match the K/A by adding a cooldown to the conditions and requiring knowledge of what will happen to the cooldown and flow rate through the bypass line rather than reciting the fail position of the valves. Two trains were placed in operation to lend plausibility to Answer (See OP-TM-AOP-015 (p12; Rev 0).

Exam Bank Searches:
Question Preview 43a for 43A03AK22:
(18) QR-543-GLO-6-Q02
See COMMENTS for changes.
Modified Question

Initial plant conditions:

- Reactor is in Cold Shutdown condition.
- Decay Heat Removal Train A is operating.
- DC-V-2A (DH Removal Cooler inlet valve) is throttled 50% OPEN.
- DC-V-65A (DH Removal Cooler bypass valve) is throttled 50% OPEN.

A loss of Vital Bus "A" occurs.

Based on these conditions, identify the ONE set of statements below that describes DC-V-2A and DC-V-65A automatic response.

A. 1) DC-V-2A fails fully open,
(2) DC-V-65A fails fully closed.

B. 1) DC-V-2A fails fully open,
(2) DC-V-65A fails fully open.

C. 1) DC-V-2A fails fully closed,
(2) DC-V-65A fails fully open.

D. 1) DC-V-2A fails fully closed,
(2) DC-V-65A fails fully closed.

Answer: A

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	E14	EK3.4	
	Importance Rating	3.5		

Knowledge of the reasons for the following responses as they apply to the (EOP Enclosures) RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Proposed Question: RO Question # 59

Sequence of events:

- Reactor trip due to low RCS pressure.
- Automatic 1600 psig ESAS actuation.
- Loss of RCS Subcooled Margin.
- All RCPs were tripped.

Current plant conditions:

- RCS pressure is 780 psig.
- Core exit thermocouple temperature is 485°F.
- The plant computer is UNAVAILABLE.
- An RCS cooldown is planned.

Which ONE (1) of the following describes the maximum RCS cooldown rate limit for the conditions, AND the required time interval (per 1102-11, Plant Cooldown) for recording data during the cooldown?

A. 50 °F/hour; AND
Every 5 minutes.

B. 50 °F/hour; AND
Every 30 minutes.

C. 100 °F/hour; AND
Every 5 minutes.

D. 100 °F/hour; AND
Every 30 minutes.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-EOP-010 (p23; Rev 10) if RCS Temperature is >255°F, and no RCPs are operating (i.e. a natural circulation cooldown) the RCS Cooldown Rate is limited to 50 °F/hr. According to OP-TM-EOP-0101 (46; Rev 71), the intent of limiting the cooldown rate during a natural circulation cooldown is to prevent drawing a steam bubble in the RV Head. According to 1102-11, Enclosure 4 (5of7; Rev 138), every 30 minutes, the operator is directed to plot a point and record the time on the P/T curves (Encl 4 Fig 1 or 1A). If the plant computer calculated cooldown rates are unavailable, the operator is instead directed to complete Enclosure 4 Data Sheet every 5 minutes. A note prior to this step identifies that these requirements are for compliance with Tech. Spec. 3.1.2.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the 30 minute interval is for logging data when the computer is available and operating.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because this is the maximum cooldown rate permitted by TS but only with RCP's operating.

D. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because this is the maximum cooldown rate permitted by TS but only with RCP's operating. The 30 minute interval is for logging data when the computer is available.

Technical Reference(s): OP-TM-EOP-010 (p23; Rev 11) (Attach if not previously provided)
OP-TM-EOP-0101 (47; Rev 3)
1102-11, Enclosure 4 (5of7; Rev 140)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP010-C001 Ob 2, 6 (As available)

Question Source: Bank #

Modified Bank # IR-EOPG11-PCO-4-Q01 (Note changes or attach parent)

New

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of the reasons for the RO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated as they apply to the EOP Enclosures (i.e. by requiring application of knowledge of cooldown limits specified in an EOP-010 GUIDE. The second part requires knowledge of monitoring by a control room position during the cooldown. The reason part of the K/A is implied by demonstrating knowledge of the logging requirement (indicated by the 1102-11 NOTE preceding the logging requirement step).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. cooldown rate during NC Cooldown, logging requirements during cooldown with computer available).

The modification involved changes to two limit choices and added cooldown monitoring requirements of 1102-11, PLANT COOLDOWN, Enclosure 4.

Exam Bank Searches:
Question Preview 43-a for 43E14EK34 – None
(103) IR-EOPG11-PCO-4-Q01 (Perhaps can modify)
Section 43 Q103 modified to meet this K/A. Changed to two limit choices and added cooldown monitoring requirements of 1102-11, PLANT COOLDOWN, Enclosure 4.
Modified Question

Sequence of events:

? Reactor trip due to low RCS pressure.
? Automatic 1600 psig ESAS actuation.
? Loss of RCS Subcooled Margin.
? All RCPs were tripped.

Current plant conditions:

? RCS pressure is 780 psig.
? Core exit thermocouple temperature is 485°F.

Identify the ONE (1) statement below that describes the MAXIMUM ALLOWABLE RCS cooldown rate limit for these conditions.

A. 40°F per hour in accordance with OP-TM-EOP-006, LOCA Cooldown.

B. 50°F per hour, since RCPs are not running in accordance with OP-TM-EOP-010 Guide 11, Cooldown Rate (CDR) Limits.

C. 100°F per hour in accordance with TMI Technical Specifications.

D. NO maximum cooldown rate limit applies, since the RCS had been saturated, in accordance with OP-TM-EOP-010 Guide 11, Cooldown Rate (CDR) Limits.

Answer: B

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	005	AA1.04	
	Importance Rating	3.9		

Ability to operate and / or monitor the following as they apply to the Inoperable / Stuck Control Rod: Reactor and turbine power

Proposed Question: RO Question # 60

Plant Conditions:

- Reactor power is 100%.
- Due to Feedwater Control System instability, the crew determines that a power reduction to approximately 75% power is necessary.
- During the power reduction, one Group 7 rod failed to move and is 10 inches misaligned from the group.
- Power is currently 85%.
- All major controllers are in AUTO.
- Efforts to move the misaligned rod to the group have been unsuccessful and a decision has been made to trim the group to within the limits of the misaligned rod.

Which ONE (1) of the following identifies the maximum power level permitted while the steps are performed to align the group to the stuck rod?

A. 70%

B. 60%

C. 50%

D. 5%

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-AOP-062 (p11; Rev 2), Step 3.17.a, 70% is the reset value for the overpower trip setpoint if realignment cannot be accomplished.

B. **Correct.** Power permitted by the conditions (RCS Flow) is 100%. 60% of the maximum permissible power is 60% power.

C. **Incorrect.** According to OP-TM-AOP-062 (p7; Rev 2), Step 3.10, the operator is directed to checking power less than 60% of the allowable power for the operating RCPs. This option is incorrect because it is lower than the maximum permissible power, but plausible because it represents 60% of the current power level (60% of 85 is 51)

D. **Incorrect.** This is plausible because according to OP-TM-AOP-062 (p3; Rev 2), Step 3.6, this is a benchmark power in AOP-062 to continue realignment efforts or to go to hot shutdown.

Technical Reference(s): OP-TM-AOP-062 (p 7; Rev 2) (Attach if not previously provided)
Technical Specification 3.5.2.2.e rev 246, 4.7.1.2 Rev 211

Proposed References to be provided to applicants during examination: None

Learning Objective: A62-C001 Ob 1,2 and 5 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10
55.43

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to operate and monitor the reactor and turbine power as they apply to the Inoperable/Stuck Control Rod (i.e. by requiring knowledge of power limit for off-normal conditions with a stuck/misaligned rod).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. Rod trim may occur as long as power level is < 60% of maximum permissible power).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	E08 EA2.1	
	Importance Rating	2.8	

Ability to determine and interpret the following as they apply to the (LOCA Cooldown) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: RO Question # 61

Event:

- The operating crew initiated a manual reactor trip from 100% power when RCS pressure began to lower.
- The crew is performing OP-TM-EOP-001, Reactor Trip.

Following are the current plant conditions:

- RCS Pressure 1670 psig, lowering slowly.
- MU-V-217 is full open.
- Letdown is isolated per Guide 9.
- PZR level is 85", lowering slowly.
- HPI is initiated per Guide 9.
- Reactor Building Pressure is 1.2 psig, rising slowly.
- Tavg is 540˚F and steady.
- Turbine Bypass Valves are closed.
- OTSG pressures are 1010 psig.

Which ONE (1) of the following procedures will be implemented next?

A. OP-TM-EOP-002, Loss of 25 ˚F Subcooled Margin.

B. OP-TM-EOP-003, Excessive Primary-to-Secondary Heat Transfer.

C. OP-TM-EOP-006, LOCA Cooldown.

D. OP-TM-EOP-009, HPI Cooling.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because <25˚F SCM is considered throughout EOP-001 but, in this case, SCM can be calculated to be ≈72˚F, and a transition to EOP-002 is unwarranted.

B. **Incorrect.** This is plausible because XHT is considered throughout EOP-001 but, in this case, all OS-24 criteria are NOT met. According to OS-24 (p3; Rev 17) XHT is defined as undesired heat removal by one or both OTSGs, and can be confirmed if all the following conditions exist: (1) RCS average temperature is below 540˚F, (2) uncontrolled lowering of RCS temperature, and (3) Tsat for OTSG pressure is < Tcold on affected OTSGs. Since RCS Temperature is 540˚F and steady, XHT is NOT occurring, and the transition to EOP-003 is unwarranted.

C. **Correct.** According to OP-TM-EOP-001 (p9; Rev 10), Step 18, IAAT Pzr Level cannot be maintained without HPI, then go to EOP-006. According to OP-TM-EOP-0011 (p13; Rev 1), the purpose of this step is to direct entry into EOP-006 for events where SCM was not lost but an RCS Leak exists which requires continued use of HPI to maintain Pzr level. In the stated conditions, SCM can be calculated to be ≈72˚F, and Pzr level continues to lower slowly without HPI. Therefore the transition is warranted.

D. **Incorrect.** This is plausible because PZR level is lowering but the primary purpose of EOP-009 is cooling rather than maintaining PZR level. There is no direct transition from EOP-001 to EOP-009.

Technical Reference(s): OP-TM-EOP-001 (p9; Rev 10) (Attach if not previously provided)
OP-TM-EOP-0011 (p13; Rev 1)
OS-24 (p5; Rev 18)

Proposed References to be provided to applicants during examination: None

Learning Objective: E06-PCO-1 (As available)

Question Source: Bank # IS-EOP006-PCO-2-Q01

Modified Bank # (Note changes or attach parent)

New

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to determine and interpret the facility conditions and selection of appropriate procedures during abnormal and emergency operations as they apply to the LOCA Cooldown (i.e. by determining that EOP-006 entry is required based on a set of conditions).

The question is at the Comprehension/Analysis cognitive level because the operator must determine the condition of the plant by calculating SCM, and determining if XHT exists, and then, knowing the purpose of each procedure listed, apply the appropriate procedure.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	076	2.1.31
	Importance Rating	4.6	

Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question: RO Question # 62

Plant conditions:

- The unit is operating with some known fuel leakage.
- Power was rapidly reduced from 100% to 25% due to a disturbance on the grid.

Which ONE (1) of the following identifies the time interval after the power reduction before a change in letdown activity will be detected on the Radiation Monitoring Channel, AND the instrument that will isolate letdown by closing MU-V-2A and MU-V-2B?

A. 1-3 minutes; AND
RM-L-1 HI.

B. 1-3 minutes; AND
RM-L-1 LO.

C. 30-60 minutes; AND
RM-L-1 HI.

D. 30-60 minutes; AND
RM-L-1 LO.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the change in letdown activity will occur rapidly but the flow rate through the detector causes a significant delay. According to TQ-TM-104-661-C001 (p38-39; Rev 3), RM-L-1 has a 3 minute delay coil to allow for decay of N-16. The operator may confuse the concept of the use of a delay coil, with the concept of instrument delay due to a limited system flowrate.

B. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may confuse the concept of the use of a delay coil, with the concept of instrument delay due to a limited system flowrate (See A); and because the operator may incorrectly believe that the auto Letdown isolation occurs on the Lo Range instrument.

C. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-661-C001 (p13; Rev 3), there are two RCS Letdown monitors, RM-L-1 Lo and RM-L-1 Hi. According to TQ-TM-104-661-C001 (p37-39; Rev 3), for reliability, RM-L-1 provides redundant detectors to measure the range equivalent to 0.1% to 1% failed fuel. According to OP-TM-MAP-C0101 (p43; Rev 1), MU-V-2A and 2B automatically close when RM-L-1 (High Range) reaches the High Alarm setpoint, while there are no automatic actions associated with the RM-L-1 LO instrument. Additionally, there is a NOTE prior to the Manual Action Required Steps that state that there is a delay time between changes in RCS activity and RM-L-1 response of 30 to 60 minutes depending on the RM-L-1 flowrate.

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the auto Letdown isolation occurs on the Lo Range instrument.

Technical Reference(s):	TQ-TM-104-661-C001 (p13, 37-39; Rev 3) OP-TM-MAP-C0101 (p43; Rev 1)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	661-GLO-5 and 10	(As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: NA   Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	11
	55.43	

Comments:

The KA is matched because the operator must demonstrate the ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup associated with High Reactor Coolant Activity (i.e. by requiring knowledge of indication response to changes in specific activity and the channel that would require monitoring the position of the letdown isolation valves.).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. how does the RCS Letdown monitor respond to changes in RCS activity, which instrument auto isolates letdown).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	A07	2.1.28	
	Importance Rating	4.1		

Conduct of Operations: Knowledge of the purpose and function of major system components and controls.

Proposed Question: RO Question # 63

Plant conditions:

- A large break LOCA has occurred.
- Cooling has been in the RB Sump Recirculation Mode for 2 hours.
- RB FLOOD LEVEL was 56 inches but has started to slowly rise.
- OP-TM-EOP-10, Emergency Procedure Rules, Guides and Graphs – GUIDE 22, RB Sump Recirculation, requires investigation of in-leakage at 67 inches.

Which ONE (1) of the following correctly completes the statement below?

The primary reason for maintaining RB FLOOD LEVEL below a maximum value is to ________.

A. maintain sump pH within limits for iodine solubility

B. maintain boron concentration for shutdown margin assumptions

C. prevent the introduction of non-primary grade water for core cooling

D. prevent adverse effects on critical instruments in the reactor building

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible because Sump pH is an accident analysis consideration but samples are performed to monitor this parameter.

B. **Incorrect.** This is plausible because SDM is an accident analysis consideration but samples are performed to monitor this parameter.

C. **Incorrect.** This is plausible because water quality is a typical consideration for RCS coolant but "prevent" is not the issue because significant in-leakage has occurred when 67" is reached.

D. **Correct.** According to OP-TM-EOP-0101 (p62; Rev 3), Guide 21, the purpose of Guide 21 is to transfer suction of the DH and the BS Pumps from the BWST to the RB Sump. This transition is needed if BWST Level is <15 feet, or if RB Flood Level is > 54 inches. In these cases continued injection from the BWST would flood some RB instruments. According to OP-TM-EOP-0101 (p65; Rev 3), Guide 22, provides the operator with direction on how to operate during RB Sump Recirculation. According to OP-TM-EOP-010 (p31; Rev 11), Guide 22, Step 9, directs the operator to verify that RB Flood Level is stable between 37 and 67 inches. According to OP-TM-EOP-0101 (p67; Rev 3), Guide 22, this level must remain below 67 inches to ensure that RB instruments are NOT adversely affected.

Technical Reference(s): OP-TM-EOP-0101 (p62, 65, 67; Rev 3) (Attach if not previously provided)
OP-TM-EOP-010 (p31; Rev 11)

Proposed References to be provided to applicants during examination: None

Learning Objective: E10-PCO-1, 2 and 3 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. the need to limit the level of flooding in the RB post-LOCA due to the adverse impact of some RB instruments) of the purpose and function of major system components and controls as it applies to flooding.

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. why the RB Level is limited in a post-LOCAS environment).

Facility:	TMI
Vendor:	B&W
Exam Date:	4/2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	028	AK2.03
	Importance Rating	2.6	

Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: Controllers and positioners

Proposed Question: RO Question # 64

The Plant is operating normally at 100% power with ICS in full automatic and all other major controllers in AUTO.

Subsequently, the Pressurizer Temperature compensation signal fails to the lowest value of the range.

Assuming no operator action, which ONE (1) of the following correctly completes the statements below?

Makeup Flow Control Valve, MU-V-17 will initially ____1____. Actual Pressurizer Level will ____2____.

A. (1) open
(2) continuously rise until a reactor trip occurs on HIGH RCS PRESSURE

B. (1) close
(2) continuously lower until a reactor trip occurs on LOW RCS PRESSURE

C. (1) open
(2) rise and then stabilize at a value ABOVE the HIGH LEVEL ALARM setpoint

D. (1) close
(2) lower and then stabilize at a value BELOW the LOW LEVEL ALARM setpoint

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the MU-V-17 will open raising Pzr Level which, in turn will raise Pzr Pressure. However, even if pressure rises the PZR Spray Valve is unaffected by this failure and will maintain RCS Pressure below the trip setpoint.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because operator may not understand how the temperature compensation signal works, and believe that the indicated Pzr level is higher than actual. If this were the case the operator would incorrectly believe that MU-V-17 would close to reduce level. If Pzr level was being lowered, the reactor trip setpoint on low RCS pressure is being approached.

C. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-624-C001 (p71; Rev 2), loss of the temperature compensating signal causes indicated PZR level to be lower than actual PZR level because the density of the reference leg is higher than it would be with the compensation, causing the instrument's measured ΔP to be higher than actual. Since 0% level is indicated when there is maximum ΔP, indicated level would approach minimum level; or the instrument will read lower than normal. This is reflected in OP-TM-1105-6, Figure 2, (p28; Rev 36) where uncompensated inches of Pzr Level is compared to compensated. For a given uncompensated level, and Pzr temperature, compensated level is always higher than uncompensated level. Therefore, if compensation is lost, indicated level will always be lower than actual (i.e. compensated) because the instruments are calibrated with temperature compensation included. According to TQ-TM-104-624-C001 (p35; Rev 2), compensated Pzr level is compared to a level setpoint and used to position MU-V-17 for auto level control. If the level drops below setpoint, MU-V-17 is automatically opened, and if level rises above setpoint, MU-V-17 is automatically closed. Since indicated level is now lower than setpoint, MU-V-17 would OPEN to raise level until the variable leg column is high enough to compensate for the loss of the temperature compensation signal, and then arrive at a new equilibrium position.

D. **Incorrect.** 1st part wrong, 2nd part correct (for error carried forward). This is plausible because the operator may not understand how the temperature compensation signal works, and believe that the indicated Pzr level is higher than actual. If this were the case the operator would incorrectly believe that MU-V-17 would close to reduce level, in which case level would approach the low level alarm setpoint.

Technical Reference(s): TQ-TM-104-624-C001 (p35, 71; Rev 2) (Attach if not previously provided)
OP-TM-1105-6, Figure 2, (p28; Rev 36)

Proposed References to be provided to applicants during examination:			None
Learning Objective:	624-GLO-5 and 10		(As available)
Question Source:	Bank #		
	Modified Bank #	QR-120229-PCO-4-Q01	(Note changes or attach parent)
	New		
Question History:	NA	Last NRC Exam: NA	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments:
The KA is matched because the operator must demonstrate knowledge of the interrelations between the Pressurizer Level Control Malfunctions and controllers and positioners (i.e. by requiring demonstration of knowledge of Pressurizer Level controller response to loss of a compensating signal.).

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate understanding of how temperature compensation affects the Pzr level indicating system, and how MU-V-17 would respond to a loss of Pzr level instrumentation temperature compensation.

The modification changed choices to meet K/A for positioner/controller response.

Exam Bank Searches:
Question Preview 42-3 for 42028AK203 – None
(12) **QR-120229-PCO-4-Q01 (Modify)**
Modified Question

The Plant is operating normally at 100% power with ICS in full automatic with the following Plant conditions:

- Four (4) Reactor Coolant Pumps (RCP's) are operating.
- Pressurizer level is 220 inches and steady.
- Pressurizer Spray Valve, RC-V-1, is in automatic.
- Pressure Operated Relief Valve, RC-RV-2, is in automatic.
- Pressurizer Heaters are being controlled in automatic.

The following event occurs:

- Temperature compensation is lost.

What is the indicated Pressurizer level following this event and what actions are required?

"Current level is indicating...

A. 250 inches; re-energize Pressurizer heaters as needed for pressure control and isolate Letdown flow.

B. 150 inches; transfer MU-V-17 to manual control and adjust to control Pressurizer level.

C. 150 inches; re-energize Pressurizer heaters as needed for pressure control and isolate Letdown flow.

D. 250 inches; transfer MU-V-17 to manual control and adjust to maintain Makeup Tank level constant.

Answer: B

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	060	AA2.01
	Importance Rating	3.1	

Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: A radiation-level alarm, as to whether the cause was due to a gradual (in time) signal increase or due to a sudden increase (a "spike"), including the use of strip-chart recorders, meter and alarm observations

Proposed Question: RO Question # 65

Plant conditions:

- The plant is at 100% power.
- Alarm MAP-C-1-1, RADIATION LEVEL HI just actuated.
- RM-A-5, Condenser Vacuum Pump Exhaust, is in ALERT.
- No other instruments are presently alarming.

Which ONE (1) of the following describes how the operator can determine whether the alarm was due to a sudden or gradual increase in radioactivity levels, AND a Condenser Vacuum Pump Exhaust instrument that could be used to confirm that the RM-A-5 detector has not failed?

A. Check the trend history on the PPC, ONLY; AND RM-A-15.

B. Check the trend history on the PPC, ONLY; AND RM-G-25 (RM-A-5 HI-HI).

C. Check the trend history on the PPC or directly on the PRF Panel; AND RM-A-15.

D. Check the trend history on the PPC or directly on the PRF Panel; AND RM-G-25 (RM-A-5 HI-HI).

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-661-C001 (p57-59; Rev 3) normal monitoring the integrity of the primary system with respect to the secondary steam system is accomplished through the use of RM-A-5, and its backup RM-A-15. The PPC provides for recording of these functions. Therefore, for the operator to determine the characterization of the alarming condition, the operator must check the trend history on the PPC. According to TQ-TM-104-661-C001 (p29-30; Rev 3), the condenser vacuum exhaust is monitored by atmospheric monitor RM-A-5 and its back-up RM-A-15. RM-A-5 actually has two ranges, including a HI range. RM-A-5 and 15 are digital meters which utilize titanium end window beta scintillation detectors, however their readouts are digital. Range $10\text{-}10^7$ CPM, while RM-A-5 GAS HI uses GM Tube and has a range of $10\text{-}10^6$ CPM (to 10^3 mc/cc). Additionally, according to TQ-TM-104-661-C001 (p22-23; Rev 3) RM-G-25 (also referred to as RM-A-5 Hi-Hi) is an ion chamber used to extend the range to of RM-A-5 to 100,000 microcuries/cc of Xe133. According to OP-TM-MAP-C-0101 (p9; Rev 1), IAAT RM-G-25 (RM-A-5 HI-HI) comes on scale, the operator is directed to de-energize RM-A-5 HI. This is because this instrument, which indicates at a higher range than RM-A-5, is at the high end of the scale, or off-scale high when RM-G-25 comes on scale. Since this is the case, RM-G-25 will NOT be able to be used to confirm that RM-A-5 has risen to the ALERT LEVEL. Since RM-A-15 is the back instrument to RM-A-5, only this instrument, among the choices offered can be used to confirm the level rise of RM-A-5.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because RM-G-25 is a radiation monitor that monitors the Condenser Vacuum Pump Exhaust. However, it extends the range of the RM-A-5 and will not register a reading in the lower levels of this instrument, such as at the ALERT Level. The operator may incorrectly believe that RM-G-25, which is associated with RM-A-5, can be used to confirm readings at this level.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to TQ-TM-104-661-C001 (p36; Rev 3), all controls necessary for operation of the atmospheric monitors, are mounted on the individual monitor cabinet. In addition, duplicate readout meters are located in the Control Room on Panel PRF. However, the trend history of the RM instrumentation can only be read on the PPC. The operator may incorrectly believe that a trend history can be determined by looking at the meter face.

D. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that a trend history can be determined by looking at the meter face (See C); and because the operator may incorrectly believe that RM-G-25, which is associated with RM-A-5, can be used to confirm readings at this level (See B).

Technical Reference(s):	TQ-TM-104-661-C001 (p29-30, 57-60; Rev 3) OP-TM-MAP-C0101 (p8-11; Rev 1)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-5 and 6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 11

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to determine and interpret a radiation-level alarm, as to whether the cause was due to a gradual (in time) signal increase or due to a sudden increase (a "spike"), including the use of strip-chart recorders, meter and alarm observations, as they apply to the Accidental Gaseous Radwaste (i.e. by requiring knowledge of the channel to be used to verify another channel is responding properly and to identify where the trend history can be viewed).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. where can the operator determine RM instrument trend history, which instrument serves as a backup to RM-A-5).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.28	
	Importance Rating	4.1		

Conduct of Operations: Knowledge of the purpose and function of major system components and controls.

Proposed Question: RO Question # 66

Plant conditions:

- The unit has been operating at 100% power for 13 days.
- A Main Feedwater Pump trips. All equipment operates as designed.
  - o Demanded Feedwater Flow is currently the equivalent of 90%.
  - o Actual Feedwater Flow is 60%.
  - o Demanded flux is currently the equivalent of 95%.
  - o Actual flux is 90%.

At this moment in time, which ONE (1) of the following describes how cross limits will affect the ICS?

A. Feedwater to Reactor cross limits are in effect. Demanded flux will be modified with a -10% decrease signal.

B. Feedwater to Reactor cross limits are in effect. Demanded flux will be modified with a -25% decrease signal.

C. Reactor to Feedwater cross limits are in effect. Demanded flux will be modified with a -25% decrease signal.

D. Reactor to Feedwater cross limits are in effect. Demanded flux will be modified with a -10% decrease signal.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** Plausible because Feedwater to Reactor cross limits are in effect due to unavailability of feedwater to provide adequate flow. Incorrect because the modification signal would be greater than 10% because after 5% error, the signal is modified to within 5% of the difference. 10% is a plausible option because the applicant may easily confuse neutron error versus feedwater demand with the actual cross limit in effect.

B. **Correct.** Demanded flux will be modified by a 25% decrease signal, because after the initial 5% error, a 25% modification will bring demanded flux to within 5% of feed flow.

C. **Incorrect.** Plausible because cross-limits are in effect, but incorrect because the wrong cross-limit is identified. If reactor to feedwater cross-limit was in effect, feedwater demand would be run up to limit neutron error. In this case, the value is correct, lending additional plausibility to the option, but the cross-limit is incorrect.

D. **Incorrect.** Plausible because cross-limits are in effect, but incorrect because the wrong cross-limit is identified. If reactor to feedwater cross-limit was in effect, feedwater demand would be run up to limit neutron error. 10% is a plausible option because the applicant may easily confuse neutron error versus feedwater demand with the actual cross limit in effect.

Technical Reference(s): TQ-TM-104-621-C001 (p56-57; Rev 2) (Attach if not previously provided)

OP-TM-MAP-H0105 REV1

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-GLO8 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: NA Last NRC Exam: Davis Besse 2005

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge of the purpose and function of major system components and controls, in this case ICS Cross-Limits.

The question is at the Comprehension/Analysis cognitive level because the operator must understand the purpose and function of cross-limits, and apply them to an event that will result in cross-limits modifying a neutron flux demand signal

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.13	
	Importance Rating	2.5		

Conduct of Operations: Knowledge of facility requirements for controlling vital/controlled access.

Proposed Question: RO Question # 67

The plant is shutdown for a Refueling/Maintenance Outage.

Which ONE (1) of the following describes an activity that will require notification of the Security Supervisor?

A. Powering up the RB Fuel Transfer System.

B. Opening FH-V-1A and 1B to fill the Fuel Transfer Canal.

C. Drain down of the Circulating Water System and disassembly of CW-P-1F.

D. Drain down of the Condensate System/Hotwell and disassembly of CO-P-1A.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** According to OP-TM-1507-7 (p8; Rev 28A), Step 7.1.4, prior to Refueling Operations the Fuel Transfer equipment is powered up by closing breakers at RB Distribution Panel D-8, 1A Reactor Plant MCC, 1B Reactor Plant MCC, and the 120V Panel AB-4. There are no requirements identified to notify the Security Supervisor. Since these actions allow plant personnel to relocate new and spent fuel assemblies, the operator may incorrectly believe that these actions require notification to the Security Supervisor.

B. **Incorrect.** This is plausible because the Transfer Tube Isolation Valves (FH-V-1A and 1B) are normally Locked Closed. According to OP-TM-220-558 (p45; Rev 7), Step 5.3.2, when draining the FTC, the FH-V-1 valves are closed and locked; and left in this configuration. However, according to OP-TM-220-558 (p17; Rev 7), Step 4.4.32, when these valves are opened, there is no direction to contact Security. Because these valves will allow direct access to spent and new nuclear fuel, the operator may incorrectly believe that Security must be notified before these valves are open.

C. **Correct.** According to OP-TM-511-000 (p4; Rev 11), Step 2.2.4, to prevent a breach of the PA Security Boundary whenever the CW System is drained the Security Supervisor must be notified prior to disassembly of a CW Pump, Expansion Joint, or flange on CW-V-13A-D, or 4A and 4B. Therefore, since the valve internals will need to be disassembled, the flange around these valves will need to be opened. This is reflected in the System Operating Procedure. According to OP-TM-511-162 (p4; Rev 4), the operator is directed to notify the Security Supervisor when the CW System is being drained with CW-13C and CW-13D opened.

D. **Incorrect.** This is plausible because in a similar fashion to the CWS, it involves draining and opening a system. The operator may confuse the requirements associated with the CWS for the CO System. According to OP-TM-421-151 (p3; Rev 4), the CO system can be drained to CO-T-1A and CO-T-1B, and then the hotwell drained to the PWST in accordance with OP-TM-421-535. According to OP-TM-421-535 (p1; Rev 0), Section 3.3, this procedure requires a Spectacle Flange to be in the OPEN position, which the operator may associate with Security Related requirements, and the notification of both the RadPro and Chemistry Managers. The operator may incorrectly believe that the Security Department must be notified as well.

Technical Reference(s): OP-TM-511-000 (p4; Rev 11)
OP-TM-511-162 (p4; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 511-GLO-9 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge of facility requirements for controlling vital / controlled access (i.e. situations under which plant activities require a notification to Security).

The question is at the Comprehensive/Analysis cognitive level because the operator must demonstrate an understanding of why the Security Supervisor must be called (i.e. breaking the CWS Piping system, which has piping diameters large enough to support human passage, allows personnel from outside plant Vital Areas to enter into a plant Vital Area).

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	G2	2.2.39	
	Importance Rating	3.9		

Equipment Control: Knowledge of less than one hour technical specification action statements for systems.

Proposed Question: RO Question # 68

Given the following:

- Reactor Startup is in progress.
- Reactor Power is 7%.
- BOTH Intermediate Range NI channels are exhibiting erratic behavior and are failing low.
- Troubleshooting and repair will be completed in approximately 12 hours.

In accordance with Technical Specifications which ONE (1) of the following actions, if any, is required?

A. NO action is required. Intermediate Range channels are NOT required above 5% power.

B. Reactor power must be held stable below 10% until at least ONE Intermediate Range channel is returned to service.

C. Within ONE hour, initiate action to place the unit in HOT SHUTDOWN.

D. Within ONE hour, initiate action to verify the operability of at least TWO Power Range detectors, <u>or</u> be in Hot Shutdown within ONE hour.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** Plausible because 5% is the value that the reactor enters POWER OPERATION. Wrong because the actual value IAW Technical Specifications is 10%

B. **Incorrect.** Plausible because 10% will also be a familiar value to the applicants. However, with power below 10%, that is when IR is required. If power were greater than 10%, they would not be required.

C. **Correct.** In accordance with Technical Specification 3.5.1, at least ONE IR is required below 10% power. With minimum channels not met, action is required IAW Technical Specification 3.0.1.

D. **Incorrect.** Plausible because during power operation, two Power Range channels are required, The applicant may confuse requirements for IR/PR channels and choose this option knowing that shutdown is required if minimum NI channels are NOT met

Technical Reference(s): Technical Specification 3.5.1, Table 3.5-1 (Attach if not previously provided)

Technical Specification 3.0.1

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO 14 (As available)

Question Source: Bank # IS-623-14-Q01

Modified Bank # (Note changes or attach parent)

New

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments: Question on Cert exam 05-1 (IS level item but entry to TS 3.0.1 and operability requirement for Nis, including modes of operation, considered RO)

The KA is matched because the operator must demonstrate knowledge of less than one hour technical specification action statements for systems (i.e. RPS Instrumentation).

The question is at the Application level cognitive level because the operator must assemble information (i.e. When and number of IR instruments required) and determine that conditions for entry into T.S. 3.0.1 apply.

This item was significantly EDITORIALLY modified with distractor option modifications, but is considered a bank question

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	G2	2.2.37	
	Importance Rating	3.6		

Equipment Control: Ability to determine operability and/or availability of safety related equipment.

Proposed Question: RO Question # 69

Plant conditions:

- The plant is at 100% power.
- The following conditions exist for the Core Flood Tanks:

	CF-T-1A	CF-T-1B
Level	11.4 feet	12.0 feet
Pressure	590 psig	630 psig
Boron Conc.	2270 ppm	2370 ppm

- CF-T-1A has 1 level instrument and 1 pressure instrument out of service.
- CF-T-1B has 1 level instrument and 1 pressure instrument out of service.

Which ONE (1) of the following describes the status of Technical Specification LCO 3.3.1.2 for the Core Flooding System?

A. CF-T-1B is INOPERABLE because boron concentration is out of specification.

B. CF-T-1A is INOPERABLE because level is out of specification.

C. CF-T-1B is INOPERABLE because pressure is out of specification.

D. The Core Flooding System is INOPERABLE because the minimum requirement for instrumentation is NOT met.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** Plausible because boron concentration is on the high end of the specification, although within the appropriate range.

B. **Incorrect.** Plausible because level is lower than CF-T-1B, but level may be as low as 11 feet and still meet the criteria for volume (930 cubic ft +/-30), so option is incorrect. Applicant must understand which level is assigned to minimum volume.

C. **Correct.** Pressure is required to be 600 psig +/-25 psig. CF-T-1B is above the specification for pressure, and the action statement will have to be entered due to inoperability of CF-T-1B

D. **Incorrect.** Although 1 channel of 2 parameters on each train are out of service, the minimum instrumentation requirement is still met. If more than 1 channel on any parameter of a train was out of service, the associated Core Flood Tank would be considered inoperable

Technical Reference(s): TS 3.3.1.2 TS section 3.3 page 3-21, 3-22 (Attach if not previously provided)
OP-TM-213-000 (p5; Rev 8)

Proposed References to be provided to applicants during examination: None

Learning Objective: 213-GLO-14 (As available)

Question Source: Bank #
Modified Bank # IR-213-GLO-14-Q01 (Note changes or attach parent)
New

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must determine operability of Core Flooding System based upon given parameters and indications

Original question for this item: (Parameters changed to provide different correct answer.)

Given the following data for the Core Flood Tanks:

	CF-T-1A	CF-T-1B
LEVEL	11.0'	12.0'
PRESSURE	580 psig	620 psig
BORON	2400 ppm	2300 ppm
Instrument Status	1 LEVEL & 1 PRESS OOS.	2 Level Channels OOS.

Which one of the following statements identifies one reason why the Core Flood system is inoperable?

A CF-T-1A is inoperable due to level channel and pressure channel out of service.

B CF-T-1B is inoperable due to 2 level channels out of service.

C CF-T-1A is inoperable due to low Core Flood Tank Pressure.

D CF-T-1B is inoperable due to high Core Flood Tank Pressure.

Answer: B

IR-213-GLO-14-Q01

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.11	
	Importance Rating	3.8		

Radiation Control: Ability to control radiation releases.

Proposed Question: RO Question # 70

Plant conditions:

- Waste Gas Decay Tank WDG-T-1A radioactive gas release is in progress.
- WDG-T-1A pressure is at 60 psig, lowering in response to adjustments to release control valve WDG-V-47.
- RM-A-6 Auxiliary Building exhaust monitor activity is as expected.
- RM-A-7 activity is as expected.
- RM-A-8 Auxiliary and Fuel Handling Building combined exhaust monitor activity is HIGHER than expected during the release, but lower than the high alarm setpoint.
- WDG-T-1B (isolated) pressure is 75 psig, lowering continuously at 1 psig every 10 minutes.
- WDG-T-1C (in service) pressure is steady at 10 psig.

Results when operator closes WDG-V-47, Waste Gas Release Stop and Control Valve:

- FT-123 gas release flow rate lowers to zero scfm.
- WDG-T-1B pressure continues to lower.

Based on these conditions, identify the ONE selection below that describes actions that would terminate the unplanned Waste Gas System leak.

A. Ensure WDG-T-1B gas re-use valve WDG-V-27 is closed.

B. Ensure WDG-T-1B release outlet valve WDG-V-31 is closed.

C. Terminate WDG-T-1B Beckman Gas Analyzer sampling.

D. Initiate request for maintenance to gag closed WDG-T-1B relief valve WDG-V-37.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** WDG-T-1B relief valve is leaking. Open re-use valve would not affect activity seen by RM-A-8, Aux and FH Bldg combined exhaust ventilation radiation monitor.
Distracter is plausible since WDG-T-1B tank pressure would reduce gradually with this valve open.

B. **Incorrect.** WDG-T-1B relief valve is leaking. If WDG-V-31 was open, WDG-T-1B tank pressure would reduce, but RM-A-7 and RM-A-8 activity would BOTH be higher than expected.
Distracter is plausible since WDG-T-1B gas pressure would reduce with this valve open.

C. **Incorrect.** WDG-T-1B relief valve is leaking. Beckman analyzer flow does not reduce tank pressure as described in the stem. Beckman analyzer discharge flow returns back to the waste gas compressor suction header, to be pumped into the in-service Waste Gas Decay Tank, rather than released to the environment.
Distracter is plausible since differential pressure is needed to develop analyzer sample flow.

D. **Correct.** Waste Gas Decay Tank relief valves discharge to exhaust duct ventilation, and RM-A-7 detector would not see this activity (but RM-A-8 would). FT-123 would not detect this flow rate. Relief valve is not isolable.

Technical Reference(s): 302-694 Rev 45. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 231-GLO-11 (As available)

Question Source: Bank # IR-231-GLO-11-Q01

Modified Bank # (Note changes or attach parent)

New

Question History: NA Last NRC Exam: NA

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

The KA is matched because the operator must demonstrate the ability to control radiation releases by requiring knowledge of where the release is occurring and the method of isolation.

The question is at the Comprehension cognitive level because the operator must determine by multiple RMS indications where the release is occurring, and using this knowledge assess how it can be isolated.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.14	
	Importance Rating	3.4		

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: RO Question # 71

Plant conditions:

- A Tube Rupture has occurred on OTSG "B".
- The crew is performing actions of EOP-5, OTSG Tube Leakage.
- Reactor power is 10%.
- The crew is preparing to trip the reactor.

Which ONE of the following describes the operation of TBVs, and the reason?

A. Controllers are placed in HAND and operated as required to prevent MSSVs from lifting and prevent radioactive release.

B. Controllers are placed in HAND and operated as required to ensure maximum RCS subcooling is maintained as the RCS depressurizes.

C. Controllers are placed in AUTO with setpoints below MSSV lift pressure to prevent MSSVs from lifting and prevent radioactive release.

D. Controllers are placed in AUTO with setpoints at normal values to ensure maximum RCS subcooling is maintained as the RCS depressurizes.

Proposed Answer: A

Explanation (Optional):

A. Correct. EOP 5, step 3.22 places TBVs in hand and step 3.23 ensures MSSVs do not lift. Basis document explains that radioactive release to environment is to be avoided.

B. Incorrect. Plausible because this is how the TBVs will be operated. Incorrect because during OTSG tube rupture, maximum RCS subcooling is undesirable, as it will increase primary to secondary leakage. Minimum RCS subcooling is desirable.

C. Incorrect. Plausible because TBVs are normally operated in AUTO post-trip, and having a setpoint that prevents MSSVs from lifting is a logical way to operate. Wrong because they are placed in HAND control.

D. Incorrect. Plausible because TBVs are normally operated in AUTO post-trip. Wrong because RCS subcooling is not maintained at maximum during an OTSG tube rupture

Technical Reference(s): OP-TM-EOP-5, (p13; Rev 7) (Attach if not previously provided)
OP-TM-EOP-0051, (p11; Rev 2)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-005-PCO-1 (As available)

Question Source: Bank # WTS 64973
Modified Bank # (Note changes or attach parent)
New

Question History: NA Last NRC Exam: Wolf Creek 2009

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12
55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities (i.e. reason for operation of TBVs following OTSG tube rupture).

The question is at the Memory cognitive level because the operator must recall information, such as the operation of equipment during EOPs

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.29	
	Importance Rating	3.1		

Emergency Procedures / Plan: Knowledge of the emergency plan.

Proposed Question: RO Question # 72

Plant conditions:

- An ALERT has been declared.

Which ONE (1) of the following identifies the communications system used to notify the state and local agencies, AND the Nuclear Regulatory Commission (NRC)?

A. BOTH are notified using the Emergency Notification System (ENS).

B. The state and local agencies are notified with the Nuclear Accident Reporting System (NARS); AND
The NRC is notified with the Emergency Notification System (ENS).

C. The state and local agencies are notified with the Emergency Notification System (ENS); AND
The NRC is notified with the Nuclear Accident Reporting System (NARS).

D. BOTH are notified using the Nuclear Accident Reporting System (NARS).

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because according to EP-AA-1009 (p5-11 through13; Rev 13), in addition to the two systems offered, there are several communications systems that support the Emergency Plan including a BRP Line, an Emergency Plan Private Branch Exchange (PBX), the Site PBX, the Coatesville PBX, and the radio systems. Various systems are used at various times for various purposes. Therefore, the operator may incorrectly believe that both the State and local agencies as well as the NRC are notified using the Emergency Notification System, and that the NARS is used later in the accident by the TSC.

B. **Correct.** According to EP-AA-114 (p1; Rev 9), the Nuclear Accident Reporting System (NARS) is a telecommunications network and form used to transmit information to the appropriate State and local agencies. Additionally, the Emergency Notification System (ENS) is a telecommunications network and worksheet used to transmit information to the Nuclear Regulatory Commission.

C. **Incorrect.** The operator may reverse the two system purposes and incorrectly believe that the state and local agencies are notified with the Emergency Notification System (ENS) and that the NRC is notified with the Nuclear Accident Reporting System (NARS).

D. **Incorrect.** This is plausible because according to EP-AA-1009 (p5-11 through13; Rev 13), in addition to the two systems offered, there are several communications systems that support the Emergency Plan including a BRP Line, an Emergency Plan Private Branch Exchange (PBX), the Site PBX, the Coatesville PBX, and the radio systems. Various systems are used at various times for various purposes. Therefore, the operator may incorrectly believe that both the State and local agencies as well as the NRC are notified using the NARS, and that the ENS is used later in the accident by the TSC.

Technical Reference(s): EP-AA-114 (p1; Rev 9) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EPAA101007 (As available)

Question Source: Bank #

Modified Bank # IR-EP101007-Q01 (Note changes or attach parent)

New

Question History: Used on SRO 03-1 Comp 3 exam question # 74 Last NRC Exam: Davis Besse May 2004

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of the emergency plan (i.e. which communication system is used to notify which agencies).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. which communication system is used to notify which agencies).

The modification changed distractors to focus on the two major communications systems.

Exam Bank Searches:
Section 2 quest for 2429:
(271) IR-EP101007-Q02
(272) IR-EP101007-Q01 (Randomly Selected)
Modified Question

The plant was at 100% power.
An Alert has been declared.
The state and local agencies are notified using the ___________ and the NRC is notified using the __________.

A. Nuclear Accident Reporting System, Emergency Notification System

B. Bureau of Radiation Protection Line, Emergency Notification System

C. Nuclear Accident Reporting System, Emergency Plan Private Branch Exchange

D. Pennsylvania Emergency Management Agency Radio System, Emergency Plan Private Branch Exchange

Answer: A

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.25	
	Importance Rating	3.3		

Emergency Procedures / Plan: Knowledge of fire protection procedures.

Proposed Question: RO Question # 73

Plant conditions:

- The plant is at 100% power.
- A small electrical fire has been detected inside the Main Control Board.
- After initial attempts to extinguish the fire, several relays inside the Main Control Board continue to burn.
- The crew is performing the actions of OP-TM-AOP-001, FIRE.

Based upon current plant conditions, which ONE (1) of the following describes actions that are required in accordance with OP-TM-AOP-001?

A. Evacuate the Control Room <u>and</u> ensure RC-V-2 is CLOSED.

B. Evacuate the Control Room <u>and</u> ensure MU-V-14A and MU-V-14B are OPEN.

C. Trip the reactor <u>and</u> ensure MS-V-8A and MS-V-8B are OPEN.

D. Trip the reactor <u>and</u> ensure MU-V-14A and MU-V-14B are OPEN.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-AOP-001 (p3; Rev 6), Step 3.9, the operating crew will determine if ability to shut down reactor is jeopardized, and use EOP-20 if it is. Currently there are no indications of plant upset or inability to control the unit from the control room. RC-V-2 must be closed, providing additional plausibility to the option.

B. **Incorrect.** This is plausible because according to OP-TM-AOP-001 (p3; Rev 6), Step 3.9, the operating crew will determine if ability to shut down reactor is jeopardized, and use EOP-20 if it is. Currently there are no indications of plant upset or inability to control the unit from the control room. The additional action to ensure MU-V-14A and MU-V-14B are open is correct, lending additional plausibility to the item

C. **Incorrect.** This is plausible because a reactor trip is required in accordance with OP-TM-AOP-001, step 3.10. Additionally, operation of MS-V-8A and MS-V-8B is required. However, the valves are required to be closed, not open. Therefore, this option will be incorrect.

D. **Correct.** OP-TM-AOP-001, step 3.10 states that if a fire in the control room cannot be promptly extinguished, a reactor trip must be performed. Additionally, MS-V-8A and MS-V-8B must be closed, RC-V-2 must be closed, and MU-V-14A and MU-V-14B must be open.

Technical Reference(s): OP-TM-AOP-001 (p1, 3, 5, 25, 31; Rev 6) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: A01-PCO-1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

The KA is matched because the operator must demonstrate knowledge of fire protection procedures (i.e. What actions are required for fire in CR that cannot be immediately extinguished).

The question is at the Comprehension cognitive level because the operator must recognize that a small fire that continues to burn after initial attempts at extinguishing requires additional action, and additionally because the applicant must make a determination of action based upon plant conditions that are not directly defined by immediate operator actions.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.28	
	Importance Rating	3.2		

Emergency Procedures / Plan: Knowledge of procedures relating to a security event.

Proposed Question: RO Question # 74

Plant conditions:

- The plant is operating at 100% power.
- A Security Event is in progress.

Which ONE (1) of the following correctly completes the TWO separate statements below?

OP-TM-AOP-008, Security Threat/Intrusion, will be entered if Security Codes ____1____ are declared.

If remote actions are necessary in accordance with Attachment 5, Remote Operator Actions, ____2____ is dispatched to the 1E 4160V Bus.

A. (1) BLUE or YELLOW, ONLY
(2) an Auxiliary Operator

B. (1) BLUE or YELLOW, ONLY
(2) the Assistant Reactor Operator (ARO)

C. (1) WHITE, YELLOW or BLUE
(2) an Auxiliary Operator

D. (1) WHITE, YELLOW or BLUE
(2) the Assistant Reactor Operator (ARO)

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that this event will NOT be covered in the AOP (See B); and because the operator may incorrectly believe that this is an AO task (See C).

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because this is the lowest threat level Security Code. According to OP-TM-AOP-0081 (p1; Rev 5), the Code WHITE is the "surface probable" threat and the "Aircraft informational" threat. None of these actions are a significant deviation from operations normally performed. On the other hand both Code YELLOW and BLUE involve penetration of the Protected Area, and the operator may incorrectly believe that this event will NOT be covered in the AOP.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because most actions performed outside the Control Room are Auxiliary Operator tasks. The operator may incorrectly believe that this is an AO task also.

D. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-AOP-008 (p1; Rev 6) entry into the procedure is required on a declaration of a Code WHITE, YELLOW or BLUE. According to OP-TM-AOP-008 (p3; Rev 6), Step 3.4, if a Code BLUE or YELLOW is declared the ARO is dispatched to the 1E 4160V Bus. According to OP-TM-AOP-008 (p5; Rev 6), Step 3.7, if a Code WHITE is declared the ARO is dispatched to the 1E 4160V Bus. According to OP-TM-AOP-0081 (p3-4; Rev 5), the ARO is sent because if all communications are lost with the Control Room including page and radio systems and EP-P-2B is operating, indicating that the Control Room is performing Attachment 1, the ARO has indication, without communication, that the protected area has been breached and should continue with the attachment.

Technical Reference(s): OP-TM-AOP-008 (p1, 3, 5; Rev 6) (Attach if not previously provided)
OP-TM-AOP-0081 (p3-4; Rev 5)

Proposed References to be provided to applicants during examination: None

Learning Objective: A08-PCO-1, 5 and 6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

	New	X	
Question History:	NA	Last NRC Exam: NA	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments:

The KA is matched because the operator must demonstrate knowledge of procedures relating to a security event (i.e. what are the entry conditions for AOP-008, who performs remote actions in the procedure).

The question is at the Comprehension/Analysis cognitive level because the operator must understand that the ARO is sent rather than the AO to perform the remote actions. Even though the AO is normally sent to perform the remote actions in the plant, this procedure specifically requires the ARO be sent because there is a possibility that all communications with the Control Room will be lost, and the ARO will know that they are to take actions independent of direction from the SM/CRS based on observed indications.

Facility: TMI

Vendor: B&W

Exam Date: 4/2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.15	
	Importance Rating	2.9		

Radiation Control: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: RO Question # 75

A 4# ESAS has occurred.

The CRS orders a determination of whether it is a steam leak or RCS leak.

RM-A-2 Reactor Building Atmospheric Monitor ___(1)___ be used to determine whether the pressure increase is from RCS based on ___(2)___.

A. (1) CAN
(2) sensitivity to fission product gasses

B. (1) CAN NOT
(2) it is isolated from Containment

C. (1) CAN
(2) sensitivity to Iodine

D. (1) CAN NOT
(2) wetting of charcoal and paper filters blocking flow

Proposed Answer: B

Explanation (Optional):